

CONF-860203--

DE86 900357

**CONF-860203**

**(DE86900357)**

**December 1985**

Distribution Category UC-41

# **HEALTH PHYSICS CONSIDERATIONS IN DECONTAMINATION AND DECOMMISSIONING**

**Proceedings  
of the  
Nineteenth Midyear Topical Symposium  
HEALTH PHYSICS SOCIETY**

**February 2-6, 1986  
Knoxville, Tennessee**

**Hosted by the  
East Tennessee Chapter**

## **HEALTH PHYSICS SOCIETY**

### **Officers for 1985-1986**

JAMES E. WATSON, JR., *President*  
JOHN W. POSTON, *President-Elect*  
DUANE C. HALL, *Treasurer*  
CARLYLE J. ROBERTS, *Secretary*  
RICHARD J. BURK, JR., *Executive Secretary*

### **Board of Directors**

James E. Watson, Jr., *Chairperson of the Board, '87*

Robert E. Alexander, '86  
William J. Bair, '86  
Ernest A. Belvin, Jr., '87  
Bruce B. Boecker, '88  
John P. Corley, '86  
Thomas R. Crites, '88  
Duane C. Hall, '88  
Roscoe M. Hall, Jr., '87  
Constantine J. Maletskos, '87  
Francis X. Masseo, '86  
Kenneth L. Miller, '88  
Mary E. Moore, '86  
John W. Poston, '88  
Carlyle J. Roberts, '87

## NINETEENTH MIDYEAR TOPICAL SYMPOSIUM

Tom W. Oakes, *General Chairperson*  
Terry J. Oakes, *Administrative Assistant*  
Alice C. Wittmer, *Secretary*

### Executive Committee

Tom W. Oakes, *Chairperson*  
William A. Alexander  
James S. Bogard  
Emily D. Copenhaver  
Howard W. Dickson  
Joe D. Eddlemon  
Ray D. Foley  
John R. Frazier  
William A. Goldsmith  
Jerry B. Hunt  
Donald G. Jacobs  
Dennis C. Parzyck  
Paul S. Rohwer  
David R. Simpson  
Richard D. Smith  
Rick V. Stachowiak  
Harvel A. Wright

### Publicity Committee

Emily D. Copenhaver, *Chairperson*  
Helen M. Braunstein  
Dana K. Jennings  
Victor L. Turner

### Finance Committee

Howard W. Dickson, *Chairperson*  
Ray D. Foley  
Jerry B. Hunt  
Tom W. Oakes  
Phil T. Perdue  
Marvin H. Shanks

### Audio-Visual and Communications Committee

William A. Alexander, *Chairperson*  
B. Annette Campbell  
Billy M. Eisenhower  
Edward B. Wagner

### Registration Committee

William A. Goldsmith, *Chairperson*  
William L. Beck  
Bill D. Carden  
Phyllis R. Cotten  
R. Dan Glenn  
J. Michael Mahathy  
R. Scott Moore  
Alice C. Wittmer

### Steering and Policy Committee

Richard D. Smith, *Chairperson*  
John A. Auxier  
Carol D. Berger  
Hal M. Butler  
Roger J. Cloutier  
Howard W. Dickson  
Donald G. Jacobs  
Dennis C. Parzyck  
James E. Turner  
Tom W. Oakes – Ex Officio

### Technical Program Committee

David R. Simpson, *Chairperson*  
James D. Berger  
John H. Coobs  
Fred F. Haywood  
Troyce D. Jones  
Tim E. Myrick

### Public Relations Committee

Ray D. Foley, *Chairperson*  
Emily D. Copenhaver  
Brian A. Kelly

### Local Arrangements Committee

Donald G. Jacobs, *Chairperson*  
John F. Alexander  
Belgin D. Barkenbus  
Hal M. Butler  
Howard W. Dickson  
Fred F. Haywood  
Paul S. Rohwer

### Exhibitors Committee

Harvel A. Wright, *Chairperson*  
Carol D. Berger  
Joe D. Eddlemon  
Robert N. Hamm  
Fred D. Haywood  
Wilma S. Nichols  
Tony A. Rhea  
Paula A. Trinoskey

**Special Events Committee**

Jerry B. Hunt, *Chairperson*  
John E. Buddenbaum  
Howard M. Hubbard  
Marcia L. Thomas

**Technical Tours and Transportation**

James S. Bogard, *Chairperson*  
Alex J. Boerner  
John E. Buddenbaum  
Gerry R. Eisele  
John R. Frazier

**Spouses Committee**

Rick V. Stochowiak, *Chairperson*  
Rita Dickson  
Patti P. Frazier  
Terry J. Oakes  
Walt F. Ohnesorge, Jr.  
Penny Parzyck  
Sandy Rohwer

**East Tennessee Chapter Liaison Committee**

Dennis C. Parzyck, *Chairperson*  
Carol D. Berger  
James R. Muir  
Paul S. Rohwer

**Publications Committee**

John R. Frazier, *Chairperson*  
William L. Beck  
James S. Bogard

Fred F. Haywood  
Cathy R. Hickey  
Edward N. Lazo  
Michael G. Stabin

**Vendors Committee**

Joe D. Eddlemon, *Chairperson*  
Karen B. Abbott  
Ken J. Babeaux  
William A. Cross, III  
David Katzman  
Harold H. Koppell  
Joe C. Lochamy  
Marvin G. Nushan  
David Silverstein  
John P. Smalling  
H. Dwane Stevens  
Al Zirkes

**Special Arrangements Committee**

Paul S. Rohwer, *Chairperson*  
Bill D. Carden  
Shilling F. Huang  
James R. Muir  
R. Keith Owenby  
M. Guvan Yalcintas

**Food and Refreshments**

M. Guvan Yalcintas

**Continuing Education**

Paul W. Frame

## Technical Program Foreword

The East Tennessee Chapter of the Health Physics Society (ETC-HPS) is pleased to host this symposium to discuss "Health Physics Considerations in Decontamination and Decommissioning." We strongly believe this timely topic will be of great benefit to all those attending. Interest in decontaminating and decommissioning (D&D) outmoded nuclear facilities is rapidly increasing. Many government facilities built during the 1940s and early 1950s have been retired for a number of years and are being examined for final disposition. In the private sector, D&D is an increasing concern as early nuclear power reactors approach the end of their operational lifetime. There is great interest in the progress of the decommissioning effort at the Shippingport Station and the expertise used in recovering Unit II at Three Mile Island (TMI). The TMI work has major applications to future decontamination techniques and methods for dealing with highly contaminated equipment and components. Given the current life expectancies of nuclear power reactors, 20 additional reactors may need decommissioning within the next 15 years.

Decontaminating and decommissioning experience to date has primarily involved relatively low levels of radiation, so the health physics problems have not been significant. However, as alternatives for D&D are examined, both the immediate and long-term radiological hazards need to be explored in more detail. Therefore, a symposium dedicated to the health physics considerations is especially appropriate at this time.

There are many reasons for having this symposium. Here are a few.

- The "art" of D&D is currently evolving. The number of health physicists with experience in the various aspects of D&D is presently quite small, but the demand is growing rapidly as more facilities reach the end of their operating life.
- The D&D activities of retired facilities can involve the full spectrum of radiation protection needs such as high exposure rates, large collective doses, airborne hazards, etc.
- The D&D objectives differ greatly from routine maintenance or repair. Routine maintenance entails some decontamination, but D&D involves decontamination to levels acceptable for landfill or recycle.
- Major waste management problems for health physicists may be created if D&D activities are not planned responsibly.
- Health physics activities must be carefully considered to avoid "surprises" such as the "niobium problem" discussed in the January 22, 1982 issue of *Science*. Health physicists must convince the public that we are conducting D&D activities in a responsible, professional manner.
- The D&D involves applications of advanced remote technology at sites which were not initially designed for such methods. Size and space limitations will require precise engineering, and health physics aspects of these operations must be considered.

Many goals are possible for this symposium, but special emphasis has been placed on:

- Sharing methods for reducing radiation exposure and contamination.
- Comparing rationales for selecting various D&D options (i.e., immediate dismantlement, protective storage, etc.) both for the near-term risk to on-site personnel and the long-term risk to the public.
- Establishing personal communications between health physicists and other professionals with common problems, and on providing means for the continued exchange of information.
- Providing an opportunity to show the public that the industry is aware of the coming problems and is developing the expertise and methodology to safely decommission the nuclear facilities constructed over the past 40 years.

In conclusion, we hope that all those attending this symposium will be rewarded with a greater appreciation and understanding of the total role of health physics in the decontamination and decommissioning of nuclear facilities.

David R. Simpson, Ph.D., CHP  
Technical Program Chairman

# CONTENTS

## CRITERIA/LEGAL CONSIDERATIONS I

	<b>Page</b>
<b>DEVELOPMENT OF RESIDUAL RADIOACTIVITY CRITERIA</b> Stanley Lichtman and Melinda Ronca-Battista	<b>1</b>
<b>NATIONAL AND INTERNATIONAL CONSIDERATIONS OF A <i>DE MINIMIS</i> DOSE</b> David Kocher	<b>9</b>
<b>ALLOWABLE RESIDUAL CONTAMINATION LEVELS FOR DECOMMISSIONING: REVISIONS TO THE METHOD</b> W. E. Kennedy, Jr. and B. A. Napier	<b>19</b>
<b>DOSE GUIDELINES FOR DECONTAMINATION AND DECOMMISSIONING PROJECTS</b> Paul C. Voilleque	<b>27</b>
<b>A MANUAL FOR IMPLEMENTING RESIDUAL RADIOACTIVITY GUIDELINES</b> Thomas L. Gilbert, K. F. Eckerman, W. R. Hansen, J. W. Healy, W. E. Kennedy, B. A. Napier, and J. K. Solday	<b>37</b>
<b>RESIDUAL SURFACE CONTAMINATION LIMITS: PROBLEMS IN INTERPRETATION AND IMPLEMENTATION</b> R. A. Wynveen, H. J. Moe, M. J. Robinet, and V. R. Veluri	<b>39</b>
<b>DECISION MAKING DURING RADIOLOGICAL DESIGNATION SURVEYS</b> Jeanette Eng	<b>51</b>
<b>UMTRA CONSENT FORM ACQUISITION: A SURVEY OF NONRESPONDENTS</b> Laura L. Gonsalves, Dena Carpenter, Thomas B. Borak, and Philip Kearney	<b>53</b>

## APPLIED HEALTH PHYSICS I

<b>HEALTH PHYSICS PROGRAM FOR THE EDMONTON URANIUM MILL DECOMMISSIONING PROJECT</b> J. L. Polehn, R. G. Wallace, R. P. Reed, and G. T. Wilson	<b>63</b>
<b>ESTABLISHING A COMPREHENSIVE COST-EFFECTIVE RADIOLOGICAL PROTECTION PROGRAM FOR DECOMMISSIONING PROJECT</b> M. A. Maan, R. P. Nair, and S. Schafer	<b>69</b>
<b>INDUSTRIAL SAFETY AND HEALTH PHYSICS INTERACTIONS DURING SLURRY TRANSFER OF 1000 CURIES OF RADIUM RESIDUES AT THE NIAGARA FALLS STORAGE SITE (NFSS)</b> Peter A. Williams	<b>79</b>
<b>HEALTH PHYSICS EXPERIENCE DURING A URANIUM AND THORIUM POND CLOSURE</b> Leslie W. Cole and Stephen V. Prewett	<b>81</b>

## ASSESSMENT

<b>AN OVERVIEW OF DECOMMISSIONING AND DECONTAMINATION TECHNICAL INFORMATION SUPPORT ACTIVITIES FUNDED BY THE U. S. DEPARTMENT OF ENERGY'S REMEDIAL ACTION PROGRAMS</b> Park T. Owen	<b>89</b>
--	-----------

<b>DECOMMISSIONING PLANNING AND THE ASSESSMENT OF ALTERNATIVES FOR THE HANFORD PRODUCTION REACTORS</b>	<b>95</b>
C. E. Miller, R. F. Potter, and J. C. Chattin	
<b>ENVIRONMENTAL ASSESSMENT RELATING TO THE DECOMMISSIONING OF STRONTIUM SEMIWORKS FACILITY</b>	<b>105</b>
V. Q. Hale and D. R. Speer	
<b>EVALUATION OF DOSES AND RISKS FROM DIFFERENT DECONTAMINATION AND DECOMMISSIONING STRATEGIES USING THE PRESTO-II METHODOLOGY</b>	<b>115</b>
D. E. Fields	
<b>ESTIMATING DOSES AND RISKS ASSOCIATED WITH DECONTAMINATION AND DECOMMISSIONING ACTIVITIES USING THE CRRIS</b>	<b>125</b>
C. W. Miller, A. L. Sjoreen, and S. J. Cotter	
<b>A COMPOSITE EMPIRICAL AND ANALYTICAL METHOD TO PREDICT THE MAGNITUDE OF SURFACE CONTAMINATION IN SEMI-ACCESSIBLE AREAS</b>	<b>135</b>
C. H. Distenfeld and R. J. Vallem	
<b>CRITERIA/LEGAL CONSIDERATIONS II</b>	
<b>IMPLEMENTATION OF DECOMMISSIONING CRITERIA IN THE CONCEPTUAL DESIGN OF THE MRS FACILITY</b>	<b>143</b>
D. L. Gross, A. D. Wilcox, and S. Huang	
<b>A SIMPLIFIED PATHWAY ANALYSIS APPROACH FOR ESTABLISHING LIMITS FOR SOIL CONTAMINATION</b>	<b>151</b>
John E. Till, Mark S. Whittaker, and Robert E. Moore	
<b>LEGAL AND ETHICAL ISSUES RAISED IN CONSIDERING RESIDENTIAL DECONTAMINATION OPTIONS FOR TECHNOLOGICALLY-ENHANCED RADIOACTIVE CONTAMINATION</b>	<b>161</b>
Larry Jensen and Joyce Feldman	
<b>PUBLIC INFORMATION EXPERIENCE IN THE URANIUM MILL TAILINGS REMEDIAL ACTION (UMTRA) PROJECT</b>	<b>171</b>
H. R. Meyer and D. J. Skinner	
<b>FINANCIAL ASSURANCE FOR DECONTAMINATION AND DECOMMISSIONING: A TEXAS PERSPECTIVE</b>	<b>181</b>
C. D. Rao, Stephen D. Etter, and Timothy W. Dziuk	
<b>ENVIRONMENTAL</b>	
<b>THE USE OF SITE GROUNDWATER INVESTIGATIONS IN THE DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR FACILITIES</b>	<b>191</b>
Stephen A. Molello	
<b>COMPARISONS OF PASSIVE ENVIRONMENTAL RADON MONITORS (PERMs) AND TRACK ETCH <sup>(R)</sup> RADON DETECTORS DURING REMEDIAL ACTIONS AT THE NIAGARA FALLS STORAGE SITE</b>	<b>197</b>
William C. Borden and Mark E. Kaye	
<b>ENVIRONMENTAL MONITORING REQUIREMENTS DURING REMEDIAL ACTION AND STABILIZATION OF URANIUM MILL TAILINGS (UMTRA PROJECT)</b>	<b>199</b>
D. J. Skinner, H. R. Meyer, and L. G. Hoffman	
<b>RADIOLOGICAL CHARACTERIZATION OF THE WELDON SPRING QUARRY WELDON SPRING, MISSOURI</b>	<b>211</b>
C. R. Hickey, R. D. Glenn, and K. L. Christian	
<b>ENVIRONMENTAL IMPACTS OF THE RELEASE OF A TRANSURANIC ACTINIDE, AMERICIUM-241, FROM A CONTAMINATED FACILITY</b>	<b>217</b>
J. Wang and P. Merry-Libby	

## DECOMMISSIONING EQUIPMENT AND METHODS

<b>A REVIEW OF CURRENT RESEARCH INTO SOIL DECONTAMINATION AND VOLUME REDUCTION TECHNIQUES FOR PURPOSES OF TRANSPORTATION AND DISPOSAL</b>	<b>227</b>
Fred F. Haywood	
<b>TMI-2 POST-ACCIDENT DOSE REDUCTION THROUGH COATING REMOVAL</b>	<b>229</b>
S. Scott Hinds, M. D. Pavlek, II, and Kenneth R. Hoyt	
<b>THE USE OF URETHANE FOAM IN THE DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR FACILITIES</b>	<b>235</b>
Rudy E. Hermetz	
<b>REMOTE METHODS FOR DECONTAMINATION AND DECOMMISSIONING OPERATIONS</b>	<b>241</b>
J. R. DeVore	
<b>JOHNSTON ATOLL SITE CLEANUP - SOME PRACTICAL CONCERNS</b>	<b>251</b>
Bruce E. Vesper, T. H. Mohaupt, M. A. Dooley, and E. Bramlett	
<b>POST-ACCIDENT RECOVERY OPERATIONS AT TMI-2</b>	<b>253</b>
Dale J. Merchant and James E. Tarpinian	
<b>THE IMPLICATION OF REMOTE TECHNOLOGY FOR DECONTAMINATION ACTIVITIES</b>	<b>261</b>
B. G. Kniazewycz, J. T. Irving, D. C. Long, and H. Wagoner	
<b>THE DECONTAMINATION AND DECOMMISSIONING OF EXTRACTION CELL 3 AT THE WEST VALLEY DEMONSTRATION PROJECT</b>	<b>271</b>
E. D. Jones	
<b>DECONNING DURING THE 1984 H. B. ROBINSON STEAM GENERATOR REPLACEMENT PROJECT</b>	<b>293</b>
Bruce A. Meyer	

## POSTER SESSION: INSTRUMENTATION

<b>DETERMINATION OF U-238, Ra-226, AND Th-232 BY GAMMA-RAY SPECTROMETRY IN UNPROCESSED SOIL SAMPLES</b>	<b>295</b>
Katherine H. Dendahl and Jeffrey A. Brown	
<b>AN IN SITU MONITORING TECHNIQUE FOR DETERMINING POST REMEDIAL ACTION CONCENTRATIONS OF RADIUM IN SOIL</b>	<b>297</b>
Nels R. Johnson and R. Dan Glenn	
<b>IN SITU DETERMINATION OF U-238 CONCENTRATIONS IN SURFACE SOIL</b>	<b>299</b>
Jeffrey A. Brown and Richard C. Robertson	
<b>RADIOLOGICAL FIELD SURVEY PROBLEMS &amp; SOLUTIONS (?)</b>	<b>301</b>
E. J. Deming and A. J. Boerner	
<b>MEASUREMENT OF RADON AND RADON PROGENY IN RADIUM CONTAMINATED RESIDENTIAL PROPERTIES IN NEW JERSEY</b>	<b>303</b>
J. Eng, A. D. White, and R. Hopper	
<b>A PORTABLE WIDE RANGE INSTRUMENT FOR BETA RADIATION MONITORING WITH MINIMUM ANGULAR DEPENDENCE</b>	<b>311</b>
J. Milakovic, G. W. Frank, and M. J. Slobodien	
<b>JOHNSTON ATOLL CLEANUP - ACCURATE MONITORING FOR ACTINIDES</b>	<b>321</b>
Bruce E. Vesper, T. H. Mohaupt, M. A. Dooley, and E. Bramlett	
<b>EVALUATION OF USE OF THE "NNC GAMMA-12" PORTAL MONITOR FOR INTERNAL DOSIMETRY SCREENING DURING DECOMMISSIONING</b>	<b>323</b>
G. Wilhelm and J. Youd	
<b>RADIOACTIVE WASTE ASSAY AT THE WEST VALLEY DEMONSTRATION PROJECT</b>	<b>331</b>
S. Marchetti, C. W. McVay, and J. R. Stimmel	

<b>A COMPARISON OF THE RESPONSE OF A NAI SCINTILLATION CRYSTAL WITH A PRESSURIZED IONIZATION CHAMBER AS A FUNCTION OF ALTITUDE, RADIATION LEVEL, AND RA-226 CONCENTRATION</b>	<b>339</b>
Richard Provencher, Gregory Smith, T. B. Borak, and P. Kearney	
<b>USING A HPGe DETECTOR AND MICRO R METER FOR QUICK AND SIMPLE ESTIMATE OF RADIONUCLIDE ACTIVITY IN LSA SHIPPING CONTAINERS</b>	<b>347</b>
M. J. Robinet, A. L. Justus, and R. A. Wynveen	
<b>RAPID ESTIMATION OF <sup>226</sup>RA IN SOIL FOR THE GRAND JUNCTION RASA/UMTRA PROJECT</b>	<b>359</b>
J. B. Kark, T. B. Borak, P. D. Kearney, and A. S. Rood	
<b>A FAST SORTING MEASUREMENT TECHNIQUE TO DETERMINE DECONTAMINATION PRIORITY</b>	<b>369</b>
C. H. Distenfeld, B. Brosey, and H. Igarashi	
<b>APPLIED HEALTH PHYSICS II</b>	
<b>HEALTH PHYSICS EXPERIENCE IN THE DECONTAMINATION AND DECOMMISSIONING OF THE 203-S, 204-S, and 205-S URANIUM NITRATE HEXAHYDRATE PURIFICATION AND WASTE TANK CAR UNLOADING FACILITIES</b>	<b>379</b>
O. R. Rasmussen	
<b>PLACEMENT OF THE RADIOCHEMICAL PROCESSING PLANT AT OAK RIDGE NATIONAL LABORATORY INTO A SAFE STANDBY CONDITION</b>	<b>389</b>
D. W. Holladay, C. D. Bopp, A. J. Farmer, J. K. Johnson, C. H. Miller, B. A. Powers, and E. D. Collins	
<b>MEASUREMENTS AND STATUS OF A DECOMMISSIONED EXPERIMENTAL REACTOR - SEFOR</b>	<b>401</b>
L. West, C. Ragland, and T. Welty	
<b>HEALTH PHYSICS OR INDUSTRIAL HYGIENE</b>	<b>403</b>
Jere K. Allen	
<b>TRAINING REQUIREMENTS FOR HEALTH PHYSICISTS IN THE DECONTAMINATION/DECOMMISSIONING FIELD</b>	<b>407</b>
E. D. Copenhaver and D. C. Parzyck	
<b>WASTE CLASSIFICATION AND DISPOSAL</b>	
<b>SITE CLEAN-UP REQUIREMENTS: WHERE DOES ONE DISPOSE OF "DIRT"?</b>	<b>417</b>
Joyce Feldman	
<b>DECONTAMINATION/DECOMMISSIONING OF MIXED HAZARDOUS AND RADIOACTIVE WASTE DISPOSAL SITES</b>	<b>423</b>
R. R. Ferguson, D. G. Jacobs, and C. W. Kimbrough	
<b>THE LABORATORY PROTOCOL FOR CO-CONTAMINATED WASTE</b>	<b>431</b>
D. P. Kharkar, D. Rodgers, L. Penfold, G. Dunstan, and L. Leventhal	
<b>DISPOSAL OF SLIGHTLY CONTAMINATED RADIOACTIVE WASTES FROM NUCLEAR POWER PLANTS</b>	<b>437</b>
Edward F. Brannagan, Jr. and Frank J. Congel	
<b>FEASIBILITY OF CENTRALIZING DISPOSAL AND PROCESSING FACILITIES FOR LOW LEVEL BIOMEDICAL WASTE</b>	<b>443</b>
T. W. Osborne, M. S. Terpilak, and E. A. Tupin	
<b>D &amp; D PROJECT EXPERIENCE</b>	
<b>AN OVERVIEW OF THE SURPLUS FACILITIES MANAGEMENT PROGRAM</b>	<b>459</b>
C. E. Miller, Jr. and B. L. Baumann	

<b>INTACT DECOMMISSIONING OPTION FOR NUCLEAR POWER PLANTS</b> S. L. Ostrow, O. I. Oztunali, and T. J. Grant	<b>461</b>
<b>DECONTAMINATION OF THE CHEMICAL CRANE ROOM AND DECONTAMINATION AND DECOMMISSIONING OF THE EXTRACTION CHEMICAL ROOM AT THE WEST VALLEY DEMONSTRATION PROJECT</b> E. C. Phillips and M. P. Golden	<b>471</b>
<b>DECONTAMINATION AND DECOMMISSIONING OF CONTAMINATED SURPLUS FACILITIES IN THE 200 AREAS OF THE HANFORD SITE</b> D. R. Speer and W. F. Heine	<b>487</b>
<b>REMEDIAL ACTION OF RADIUM CONTAMINATED RESIDENTIAL PROPERTIES</b> Duncan White and Jeanette Eng	<b>497</b>
<b>EXPERIENCE GAINED DURING REMEDIAL ACTION AT RESIDENTIAL PROPERTIES CONTAMINATED WITH Th-232 AND Ra-226</b> William L. Beck, Richard Lynt, Edward N. Lazo, and Michael R. McDougall	<b>505</b>
<b>DECONTAMINATION AND DECOMMISSIONING OF A LUMINOUS DIAL PAINTING FACILITY: RADIOLOGICAL CHARACTERIZATION, SEGREGATION AND DISPOSAL OF BUILDING MATERIALS</b> D. Ed, L. Chu, P. Chepulis, and M. Hamel	<b>513</b>
<b>POSTER SESSION: GENERAL</b>	
<b>CORRELATION BETWEEN GAMMA RADIATION LEVELS AND SOIL RADIUM CONCENTRATIONS AT THE EDMONTON URANIUM MILL SITE</b> R. G. Wallace, R. P. Reed, J. L. Polehn, and G. T. Wilson	<b>523</b>
<b>ANALYSIS OF URANIUM AND THORIUM CONTAMINATED SOILS DURING A CLEANUP AND INTERMENT OPERATION</b> Leslie W. Cole and Michael Bollenbacher	<b>533</b>
<b>IDAHO NATIONAL ENGINEERING LABORATORY RELEASE CRITERIA FOR DECONTAMINATION AND DECOMMISSIONING</b> Max R. Dolenc and Marilyn J. Case	<b>535</b>
<b>DECONTAMINATION OF AN AMERICIUM-241 SITE</b> G. L. Murphy, E. J. Deming, R. J. Caniano, T. R. Emswiler, and G. E. Kirsch	<b>537</b>
<b>INVENTORY OF SITES USED TO DEVELOP RESIDUAL RADIOACTIVITY CRITERIA</b> Melinda Ronca-Battista and James C. Hardeman, Jr.	<b>539</b>
<b>DATABASE CONSTRUCTION FOR ASSESSMENT OF CHRONIC RADIONUCLIDE RELEASES</b> L. M. Hively, A. L. Sjoreen, and J. L. Bledsoe	<b>543</b>
<b>THE USE OF COMPUTERIZED, 3-DIMENSIONAL SOLIDS MODELING AND DATA BASE MANAGEMENT TO SUPPORT RADIATION MAPPING AND ALARA PLANNING</b> Richard D. Schauss and Michael S. Slobodien	<b>553</b>
<b>HEALTH PHYSICS AND OPERATIONAL EXPERIENCE GAINED FROM SLURRY TRANSFER OF WASTES CONTAINING 780 CURIES OF RADIUM-226</b> William C. Borden and Roger R. Land	<b>563</b>
<b>ASSURANCE PROGRAM FOR REMEDIAL ACTION (APRA) MICROCOMPUTER-OPERATED BIBLIOGRAPHY MANAGEMENT SYSTEM</b> R. D. Stenner, D. K. Washburn, and D. H. Denham	<b>565</b>
<b>DECOMMISSIONING THE WORLD'S FIRST COMMERCIAL NUCLEAR FUEL REPROCESSING FACILITY</b> Stephen Marchetti and Dean Hoffman	<b>575</b>

**THE FINAL RESTING PLACE OF A CESIUM-137 GAUGE**

**579**

**James L. McNees, Kirksey E. Whatley, and Cornelius Maryland**

**AUTHOR INDEX**

**581**

## DEVELOPMENT OF RESIDUAL RADIOACTIVITY CRITERIA

Stanley Lichtman  
Melinda Ronca-Battista

U.S. Environmental Protection Agency  
Office of Radiation Programs  
Washington, D.C. 20460

### ABSTRACT

The U.S. Environmental Protection Agency (EPA) is developing public health radiation protection criteria for residual radioactivity following cleanup of contaminated lands and facilities. Satisfaction of the criteria will allow the site to be safely released from all controls based on radioactivity. The criteria may be issued as broad Federal guidance, specific environmental standards, or both. The criteria will apply to sites owned or used by licensees of the Nuclear Regulatory Commission or its Agreement States, the Department of Energy, the Department of Defense, the former Atomic Energy Commission and Manhattan Engineering District, and where naturally-occurring or accelerator-produced materials have been used. The EPA is supporting the development of this project with the following activities. First, an inventory of never licensed or otherwise poorly documented sites is being compiled in conjunction with the National Conference of Radiation Control Program Directors, to ensure that the criteria are appropriate for all site types and satisfy existing and future decommissioning needs. Second, the relationship between residual radioactivity and public exposures under a variety of site and future use conditions is being studied. This information will enable EPA to evaluate the health impacts of alternative levels of residual radioactivity. Finally, non-technical issues to be resolved include how the criteria and the supporting analyses could apply to recycled materials, how these criteria will relate to "below regulatory concern" wastes, and whether the criteria should include limits on planned reliance on institutional controls.

### PURPOSE OF THE CRITERIA

The Environmental Protection Agency (EPA) is developing residual radioactivity criteria for cleanup of land and facilities contaminated with radioactive materials. The purpose of these criteria is to assure protection of public health and the environment after such facilities are shut down, decommissioned, and released from restrictions based on radioactivity. The main objective is to limit health risks to current and future generations and to assure that cleanup costs are reasonable in comparison to the long and short term benefits of cleanup. Other Federal and State agencies would retain their authority to determine which sites under their jurisdiction should be cleaned up for such unrestricted use.

## CIRCUMSTANCES FOR WHICH RESIDUAL RADIOACTIVITY CRITERIA ARE NEEDED

Radioactive materials have been used at more than 20,000 sites in the United States. These may be grouped into three categories: (1) licensees of the Nuclear Regulatory Commission (NRC) or its Agreement States, (2) government sites, such as those of the Department of Energy (DOE) and Department of Defense (DOD), and (3) nongovernment sites where naturally-occurring or accelerator-produced radioactive materials have been used.

The NRC and its Agreement States have issued over 22,000 licenses for use of radioactive materials covered under the Atomic Energy Act (AEA). (The AEA does not cover accelerator-produced and most naturally radioactive materials.) More than 95 percent of these licensees use sealed radioactive sources or small quantities in research or medicine that result in little or no contamination of land or facilities. Of more concern are approximately 300 major licensees, such as nuclear power plants, other components of the nuclear fuel cycle, and research reactors.

The DOE controls about 25 large complexes, including national laboratories and nuclear weapons research and testing sites. There are approximately 500 facilities in the DOE "Surplus Facilities Management Program" scheduled for decommissioning over the next twenty years. They include reactors, solid waste burial grounds, fuel reprocessing facilities, radiochemical laboratories, and waste treatment plants. The DOE is also responsible for sites operated by the former Atomic Energy Commission and the Manhattan Engineering District, 25 of which are now in the "Formerly Utilized Sites Remedial Action Program" (FUSRAP). Many FUSRAP sites were used for research, processing, and storage of uranium and thorium ores, and are contaminated primarily with naturally-occurring radionuclides.

DOD facilities vary widely in function and size. They include hospitals, laboratories, proving grounds, bombing and gunnery practice ranges, missile launch sites, weapons manufacturing and storage facilities, and reactors. These sites range from those that contain only small enclosed radiation sources, such as radium and tritium instruments, to dispersed sources, such as laboratory waste areas and test ranges contaminated with radioactive products.

The third category consists of all remaining sites not under license by NRC or Agreement States or directly controlled by the Federal government. Included are about a thousand particle accelerator sites that usually contain only small amounts of short-lived radioactivity after their shutdown. Other sites contain naturally-occurring radionuclides in forms ranging from small radiation sources, such as radium needles, and residues from hospital and research activities, to large volumes of mostly low-level dispersed contamination, including tailings from rare earth ore processing and depleted uranium from armor piercing shell tests.

In addition to an administrative categorization, as described above, we may consider two classes of entities to which residual radioactivity criteria could apply: (1) decommissioning fixed entities (i.e., land and permanent buildings on a site); (2) decontaminating removable, reusable entities (i.e., equipment and materials). We plan initially to emphasize development of criteria for fixed entities, to lay a foundation for dealing with the less tractable category of removable entities.

This program will not address sites already covered by Public Law 95-604, the Uranium Mill Tailings Radiation Control Act of 1978, and Public Law 92-314, which established a remedial action program for contaminated buildings in Grand Junction, Colorado. Radioactive waste disposal sites that are not intended for unrestricted public access will also be excluded. Note, however, that EPA has disposal standards for radioactive wastes in various rulemaking stages: final standards for uranium mill tailings (48 F.R. 590, January 5, 1983 and 48 F.R. 45926, October 7, 1983) and for spent fuel and high-level and transuranic radioactive waste (50 F.R. 38066, September 19, 1985); and an advance notice of proposed rulemaking for low-level radioactive waste (48 F.R. 39563, August 31, 1983). Management or disposal of high volume radioactive mining wastes, such as phosphate and gypsum piles, and uranium mining wastes, which are being studied under EPA's Resource Conservation and Recovery Act program, will also not be covered by these criteria.

#### APPLICABLE EPA AUTHORITIES

EPA could develop radiation protection criteria to cover all NRC licensees and most Federal sites under two types of authorities. The first is the authority of the Administrator to develop general recommendations to serve as radiation protection guidance to all Federal agencies. Agencies use such Federal guidance, which may consist of general principles, specific policies, and/or numerical criteria, as the basis for developing and implementing their own regulatory standards and procedures. Federal guidance may address any radiation protection issue, but would not directly cover materials regulated exclusively by the States. States traditionally have voluntarily followed Federal radiation protection guidance, however. Federal guidance can be difficult to develop because a consensus of Federal agencies is desirable, in practice, and the President must approve the guidance.

The second type of authority provides for the direct establishment of regulatory standards by EPA. The most important is the Atomic Energy Act of 1954 (AEA), as amended, under which EPA may establish generally applicable environmental radiation standards that apply to any radioactive material covered by the AEA. AEA standards must be expressed as limits on radiation exposure or levels or the quantities or concentrations of radioactive materials. Non-AEA materials, such as naturally-occurring or accelerator-produced radioactive material (NARM), would not be covered by such standards. EPA may use other authorities for such radionuclides, such as the Resource Conservation and Recovery Act (RCRA) and the Toxic Substances Control Act (TSCA).

We believe EPA's Federal guidance and AEA authorities are those most suitable for addressing the broad radiation protection issues involved in establishing residual radioactivity criteria. Congress intended RCRA and TSCA for specific types of radioactive materials or hazardous circumstances, none of which closely match the predominant materials and circumstances for which these residual radioactivity criteria are to be developed: RCRA deals primarily with controlling waste; TSCA deals with use of materials; with respect to radioactive hazards, both deal only with NARM.

## ISSUES

In the course of this work, we intend to obtain and evaluate information in the following areas:

1. Basis for the criteria: What relative importance should be assigned to various factors in determining criteria for unrestricted public access, now and in the future, to sites that contain residual radioactivity materials? Among the factors we plan to consider are magnitudes of current and future individual risks, cumulative effects on populations now and in the future in relation to the half-life and environmental mobility of the residual contamination, and the technical and economic practicality of implementation of cleanup.

2. Form of the criteria: (a) Contamination of sites where radioactive materials have been used may exhibit widely varying characteristics. The materials may be well contained or widely dispersed, manmade or natural, low or high in radioactivity, short- or long-lived, and in a variety of chemical forms. Different criteria could be devised for different types of contamination, or some general criteria might be developed that can be applied differently to different circumstances. EPA intends to determine whether different forms of radiation protection criteria are necessary and appropriate for different types of contamination. (b) Cleanup costs for previously closed or abandoned sites may be much higher than for operating sites. Similarly, costs may be lower for future sites because of better planning for efficient decommissioning. We intend to consider whether such cost differences might justify less stringent criteria for previously closed than for presently operating or future sites.

3. Guidance versus standards: Guidance may be numerical, narrative, or both; it may address quantitative radiation protection requirements as well as procedural considerations. Standards are generally more specific than guidance. Implementing agencies have greater discretion in applying guidance than standards. Perhaps broad guidance should be formulated initially as a basis for more specific subsequent standards. We intend to determine whether guidance, standards, or a combination, is the most appropriate form for criteria.

4. Assessment needs: EPA needs to evaluate the effects of alternative criteria. Therefore, we intend to assess the health and environmental benefits, costs, and technical feasibility of achieving various levels of residual radioactivity at sites that are to be made available for unrestricted public use. Because DOE and NRC licensed sites are relatively well documented, such information is particularly needed for old sites or those that have not been regulated for radioactivity.

5. Institutional controls: For some sites it may be possible to reduce occupational radiation exposure and cleanup costs by deferring cleanup until radioactivity levels decline by radioactive decay. In the interim, the use of sites must be restricted by institutional controls. In other recent actions for radioactive wastes, EPA has expressed its preference to not rely primarily on institutional controls for long-term protection from radiation hazards. We intend to assess whether EPA needs to consider a time limit on such use of institutional controls in decommissioning to assure adequate protection of public health and the environment.

## CRITERIA DEVELOPMENT

We are pursuing two complementary approaches to developing residual radioactivity criteria. The first is to develop Federal guidance for residual radioactivity, which would update existing generic guidance\* and provide a framework for developing optimal ("ALARA") criteria for specific categories of residual radioactivity sources. Such Federal guidance would express broad radiation risk limitation principles applicable to residual radioactivity that the general public, health physics community, and Federal agencies accept as fair, sensible, and practical.

This process of developing a consensus on broad guidance principles cannot be scheduled. It will initially require extensive consultation with and review by other Federal agencies. We plan to interact with the States, health physics community, and general public at appropriate stages, including through workshops.

Second, in order to justify generic criteria we must assess what is generally achievable in the real world. This is needed so that (1) we have a general assessment of the health and cost implications of various cleanup criteria, (2) we will understand the conditions under which criteria for unrestricted use can and cannot be reasonably satisfied, and (3) we do not require less public health and environmental protection than can be reasonably achieved as a general rule. We intend to assess different levels of residual radioactivity for broad classes of sites by analyzing the health effects, costs, and technical feasibility of implementation. For example, existing work suggests the general technical and economic feasibility of achieving maximum residual radiation exposure levels of about 10 mrem/y at major nuclear fuel cycle facilities, such as reactors. Lower residual exposure levels are certainly achievable for other kinds of facilities where only short-lived or sealed radiation sources have been present, such as at many hospitals and laboratories. We also believe, based on our work on uranium milling and rare earth processing wastes, that where large volumes of radon-emitting materials have been handled only higher residual dose levels may be feasible. These preliminary conclusions are based on existing data and analyses reflect engineering judgments about practical cleanup procedures for specific levels and situations. There have been very few systematic analyses of the costs and benefits of achieving other cleanup levels for a broad range of sites.

To generate such systematic analyses, we plan to group facilities into categories according to similarities of size, operational characteristics, radioactive materials, cleanup methods, and usefulness after decommissioning. Scenarios will be developed whereby people having unrestricted access would be exposed to residual radioactivity. A range of residual radioactivity levels

---

\*In the U.S., the only official generic guidance applicable to residual radioactivity consists of recommendations made by the Federal Radiation Council and approved by the President in 1960. The guidance recommends, for example, that whole body dose rates be limited to 0.5 rem/y for individuals in the general population. The maximum individual lifetime risk associated with this dose rate, approximately  $10^{-2}$ , deserves review in light of more recent reviews, e.g., by the International Commission on Radiological Protection.

will be analyzed by appropriately varying the cleanup operations analyzed in existing literature. Costs and benefits of each cleanup level will be determined. Both the maximum individual risk and the long-term cumulative population exposure will be considered. The analysis should identify reasonably achievable risk levels for each major site and facility category, and indicate whether the same criteria may be applied in common to all or only to a number of such categories.

Because many cleanup methods have not been applied in a full-scale manner, uncertainties associated with pilot studies will have to be considered. Uncertainties may be a key factor in deciding whether guidance or standards is the most appropriate and effective regulatory approach.

## EARLY PRODUCTS

Much of the analysis discussed above will take several years to complete. We are planning two products, however, that should provide useful information earlier.

### 1. Source-to-Dose Relationship for Residual Radioactivity

The first such product is a pathway analysis relating residual radioactivity levels to doses to average members of a critical group, i.e., on-site individuals. This work will cover the radionuclides and depositional media that are important in decommissioning -- specifically, natural and man-made radionuclides with half-lives greater than about 1 year, in the fixed media of soil and building surfaces. We need this relationship between residual radioactivity and dose for several reasons. First, the relationship will be used for evaluating the effect of the criteria on people exposed to residual radioactivity. Second, the relationship is needed for evaluating the benefits and costs of achieving different levels of cleanup. Finally, if the criteria are formulated in terms of dose or risk limits, implementers will need to know the relationship of the criteria to the residual radioactivity levels.

We have carefully reviewed the technical literature and believe that most of this source-to-dose assessment method can be assembled from existing models and from recent developmental work by the Department of Energy for residual radioactivity in soil. The weakest link in this regard is the evaluation of water pathways.

In October 1985, we received a preliminary report that we had commissioned on the analytical deficiencies of current source-to-dose assessment methods. We are sponsoring further work to develop, review, simplify, and package a method that will be adequate for the purposes described above. We intend to consult closely with other interested agencies and peer reviewers in developing the final product.

## 2. Inventory of Sites

In developing residual radioactivity criteria, EPA needs to ensure that an appropriate range of site and facility characteristics are considered. Especially needed is information on sites and facilities that are poorly documented because they are not Federally-owned or regulated by Federal or State governments for radioactivity. These include old and non-licensed sites that are not now covered under the Uranium Mill Tailings Radiation Control Act, the Comprehensive Environmental Response Compensation and Liability Act ("Superfund"), or DOE's Formerly Utilized Sites/Remedial Action Program.

Accordingly, at our request, the Conference of State Radiation Control Program Directors has collected available information from all the States on such sites. In addition to the exclusions described above, the solicitation does not cover sites consisting entirely of high volume mining wastes, such as phosphate and gypsum piles that EPA is studying separately under a RCRA program. Similarly, we have excluded sites that contain high volumes of marginally radioactive wastes, such as coal ash, and other high volume residues with specific activities less than 15 pCi/g.

The information obtained will help to establish the range of circumstances for which criteria are needed, and the suitability of candidate criteria for dealing with actual situations. The information will also aid in identifying adequate model sites and facilities for analyzing the technical and economic feasibility of candidate criteria and in assessing overall costs and benefits. The results of this study are presented elsewhere at this Symposium (Melinda Ronca-Battista and James C. Hardeman, "Inventory of Sites Used to Develop Residual Radioactivity Criteria").

# NATIONAL AND INTERNATIONAL CONSIDERATIONS OF A DE MINIMIS DOSE\*

David C. Kocher  
Health and Safety Research Division  
Oak Ridge National Laboratory  
P.O. Box X  
Oak Ridge, TN 37831

## ABSTRACT

This paper reviews current efforts by national and international agencies to establish a generally applicable de minimis radiation dose. A de minimis dose is derived from a level of risk that is widely regarded as negligible by the general public, and defines a level below which control of radiation exposures would be deliberately and specifically curtailed. A de minimis dose can be used to derive quantities of radionuclides in various materials that would be exempt or below regulatory concern for such purposes as resale, recycling, or disposal. The specification of exempt levels of radioactivity could lead to significant reductions in the required capacity of radioactive waste storage and disposal facilities and in the costs associated with management of slightly contaminated materials. A de minimis dose must be set well below the limit on acceptable dose for public exposures. A de minimis dose in the range 0.1-30 mrem (0.001-0.3 mSv) per year committed effective dose equivalent has been considered by various agencies, with most recommendations at 1 or 5 mrem per year. A value of 30 mrem per year may be too high for a de minimis dose, because a dose limit of 25 mrem per year is widely used in the U.S. for regulating specific practices (e.g., low-level waste disposal). A value as low as 0.1 mrem per year could lead to great difficulties in measuring associated levels of radioactivity. We propose a de minimis dose of 1 mrem (0.01 mSv) per year committed effective dose equivalent averaged over a lifetime and 5 mrem (0.05 mSv) in any year. These values are 1% of the limits on acceptable dose from all sources of exposure that recently have been proposed by the National Council on Radiation Protection and Measurements. The proposed de minimis dose corresponds to a lifetime risk from continuous exposure of about  $10^{-5}$ .

## INTRODUCTION

Radiation-protection authorities generally recognize that there are levels of radioactivity so low as to be of no concern (1); i.e., potential risks from radiation exposures that might result from unrestricted use of materials containing such low levels of radioactivity would be of no concern to members of the general public. These so-called "exempt" materials thus could be handled in all respects as if they were nonradioactive. The establishment of exempt levels of radioactivity in slightly contaminated materials would have two primary benefits: [1] the materials would be suitable for unrestricted recycle or sale to the general public, or [2] the materials could be placed in disposal facilities for nonradioactive wastes, thus providing significant savings in disposal costs and in the required capacity of radioactive waste disposal facilities.

\* Research sponsored by the U.S. Department of Energy under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

By acceptance of this article, the publisher or recipient acknowledges the U.S. Government's right to retain a nonexclusive, royalty-free license in and to any copyright covering the article.

The primary purpose of this paper is to review current efforts by national and international agencies to develop generally applicable criteria that could be used to establish exempt quantities of radioactivity. These criteria generally are in the form of limits on radiation dose to individuals in the general public that would be below regulatory concern. Such a limit is referred to as a de minimis dose. Establishment of a de minimis dose by regulatory authorities would provide the most defensible basis for deriving exempt quantities of radioactivity in all radiation practices.

The next section of this paper discusses the concept of a de minimis dose and its relation to limits on acceptable dose and the ALARA (As Low As Reasonably Achievable) principle. The third section then discusses the general principles, based on the concept of negligible risk, that are used to derive a generally applicable de minimis dose, and presents the recommendations of national and international agencies for a de minimis dose. The last section presents a set of recommendations for a de minimis dose that we have developed on the basis of this review.

## CONCEPT OF DE MINIMIS DOSE

The concept of a de minimis dose is best understood in relation to other fundamental concepts in radiation protection of the general public including [1] a limit on acceptable dose from all sources of exposure, [2] limits on acceptable dose from specific practices, and [3] application of the ALARA principle to specific sites and specific practices. These concepts and their interrelationships are shown in Fig. 1, which is adapted from Meinhold (1).

### Limit on Acceptable Dose from All Sources

The top line in Fig. 1 represents the dose limit for any member of the general public from all sources of exposure, exclusive of natural background radiation and deliberate medical practices. This dose limit represents a risk that is regarded as "acceptable" by most individuals, and shall not be exceeded, except in unusual circumstances, regardless of the cost of doing so.

In the U.S., the current limit on annual dose\* from all sources, as established by the Nuclear Regulatory Commission (NRC), is 500 mrem to whole body or gonads, with higher limits for other organs and tissues (2). Similar limits on annual dose to whole body or the critical organ have been established by the U.S. Department of Energy (DOE) for all DOE and DOE-contractor operations (3). The National Council on Radiation Protection and Measurements (NCRP) recently has recommended a limit on annual committed effective dose equivalent of 100 mrem (1 mSv) for continuous or repeated exposures and a limit of 500 mrem (5 mSv) for occasional exposures (4). The committed effective dose equivalent is a weighted sum of committed dose equivalents received by several body organs and tissues, and is a quantity that is intended to be proportional to stochastic risk for either uniform or nonuniform irradiation of the body (5).

### Limits on Acceptable Dose from Specific Practices

Regulatory authorities in the U.S. have established generally applicable limits on acceptable dose for specific practices that often are well below the

---

\* This paper emphasizes dose equivalents in units of mrem to facilitate comparison with radiation standards in the U.S. The relation between mrem and the SI units of mSv is 1 mrem = 0.01 mSv.

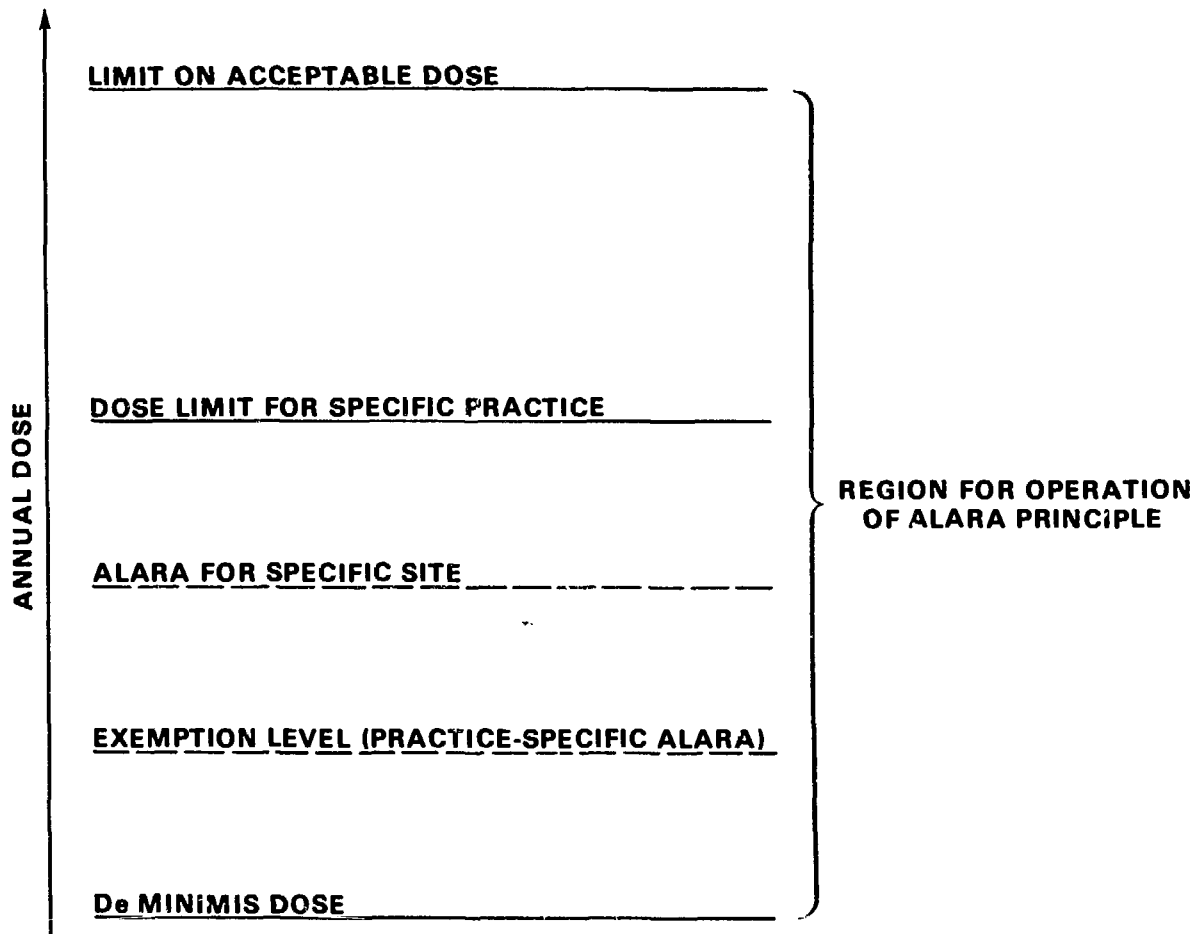


Fig. 1. Relationships between limit on acceptable dose, dose limits for specific practices, doses resulting from application of the ALARA principle to specific sites and specific practices, and de minimis dose. The vertical scale for dose is arbitrary.

limit on acceptable dose from all sources. For example, the Environmental Protection Agency (EPA) has established a limit on annual dose to whole body of 25 mrem for several practices (6-8), and the NRC has established the same limit for low-level waste disposal (9).

The dose limits for specific practices represent a judgment by the regulatory authorities that the limits are "reasonably achievable" at any site. However, the dose limit that is judged to be reasonably achievable may vary from one practice to another. For example, the EPA's remedial action standards for inactive uranium processing sites (10) correspond to annual doses considerably in excess of 25 mrem (11). However, the dose limit for a specific practice must not exceed the dose limit from all sources of exposure.

The generally applicable dose limit for a specific practice is represented by the second line in Fig. 1. This limit would apply to that practice at any site.

#### Application of the ALARA Principle

Once regulatory authorities have established generally applicable dose limits for specific practices, application of the ALARA principle may reduce maximum doses to individuals still further, provided it is cost-effective to do so. The ALARA principle may be applied on a site-specific or a practice-specific basis.

Application of the ALARA principle to specific sites is an essential aspect of radiation-protection standards in the U.S. [e.g., see refs. (2-4)], and involves a balancing of reductions in population dose with the increased costs required to achieve such reductions (5). The annual dose that results from site-specific application of the ALARA principle is represented by the upper dashed line in Fig. 1. This dose will not exceed the limit for the particular practice, but application of the ALARA principle to different sites generally will result in different individual doses.

General application of the ALARA principle to a specific practice usually leads to the determination of levels of radioactivity that are exempt or below regulatory concern, irrespective of site. The determination of exempt levels of radioactivity for a specific practice essentially represents a judgment by the regulatory authorities that the doses associated with those levels are ALARA for any site. For example, several NRC rulemakings specify quantities of radionuclides that are generally exempt from certain licensing requirements (2,12-14). The annual dose resulting from practice-specific application of the ALARA principle is represented by the lower dashed line in Fig. 1. This dose will not exceed the dose obtained by applying the ALARA principle to the given practice at a specific site (i.e., the upper dashed line in Fig. 1), but may vary from one practice to another.

#### De Minimis Dose

The concept of a de minimis dose arises from the consideration that there must be a limit beyond which no further reduction in dose should be attempted using the ALARA principle, for either specific sites or specific practices. Thus, a de minimis dose is of no concern to regulatory authorities, as are doses associated with exempt quantities of radioactivity for specific practices, but such a dose also defines a level below which control of radiation exposures would be deliberately and specifically curtailed (1).

A de minimis dose, which is represented by the lowest line in Fig. 1, corresponds to a "negligible" level of risk, as distinct from an "acceptable" level of risk. Thus, the de minimis dose must be set well below the limit on acceptable dose from all sources of exposure. The NCRP recommends, for example, that the de minimis dose should be two orders of magnitude below the limit on acceptable dose from all sources (4).

While the de minimis dose constitutes a lower limit to ALARA, this dose is not the goal of ALARA, since what is "as low as reasonably achievable" for a given practice or a given site may result in doses that are above the de minimis level (1). The ALARA principle requires only that reasonable efforts be made to reduce exposures as far below the dose limit as practicable, taking into account a variety of technological, social, and economic factors, not that doses must be reduced to de minimis levels.

## Summary of Concepts

The radiation-protection concepts described above and their interrelationships may be summarized as follows. A dose limit from all sources of exposure, exclusive of natural background and deliberate medical practices, corresponds to a level of risk that is generally "acceptable" to the general public. The limit on acceptable dose thus constitutes a ceiling for application of the ALARA principle; i.e., the dose limit must be met regardless of the cost of doing so. Regulatory authorities then may establish lower dose limits for specific practices, based on a judgment that such doses are "reasonably achievable" at any site, and these limits may vary from one practice to another. Application of the ALARA principle leads to further reductions in individual doses on a site-specific or a practice-specific basis. The latter application usually leads to the determination of quantities of radionuclides that are "exempt" or "below regulatory concern," based on a judgment by regulatory authorities that the doses associated with exempt quantities of radioactivity are indeed ALARA at any site. The de minimis dose corresponds to a level of risk that is widely regarded as "negligible" by the general public. The de minimis dose must be well below the limit on acceptable dose from all sources, and constitutes a floor for application of the ALARA principle; i.e., efforts to reduce doses below the de minimis level would be deliberately and specifically curtailed. Establishment of a de minimis dose by regulatory authorities thus would provide the most defensible basis for determining exempt quantities of radioactivity for specific practices.

## RECOMMENDATIONS OF NATIONAL AND INTERNATIONAL AGENCIES FOR A DE MINIMIS DOSE

This section reviews current efforts by national and international agencies to establish a generally applicable de minimis dose. We first discuss the general approach, based on the concept of negligible risk, that is used in defining a de minimis dose. We then review the specific recommendations for a de minimis dose.

### General Approach to Defining a De Minimis Dose

The general approach that has been adopted in defining a de minimis dose is based on the concept of negligible risk [e.g., see refs. (1) and (4)]. This approach is outlined below.

- [1] On the basis of risks that are widely accepted by the general public, a negligible lifetime risk from radiation exposure is defined. The negligible risk must be well below the level of acceptable risk on which radiation protection standards are based.
- [2] From the negligible lifetime risk from radiation exposure so defined, a lifetime de minimis dose is derived using an accepted factor for the risk per unit dose. This risk factor is based on a linear no-threshold, dose-response hypothesis, and usually is assumed to be in the range  $1-2 \times 10^{-4}$  per rem ( $1-2 \times 10^{-2}$  per Sv) (5).
- [3] From the lifetime de minimis dose so derived, an annual de minimis dose is obtained by assuming exposure over the 70-year lifetime of an average individual; i.e., the lifetime de minimis dose is divided into equal annualized increments.

Thus, in this approach, a negligible lifetime risk is expressed in terms of a de minimis dose for each year of exposure. Alternatively, an annual negligible risk can be defined that forms the basis for deriving the annual de minimis dose.

The general approach to defining a de minimis dose does not involve consideration of specific practices or facility locations. However, using a de minimis dose to derive exempt quantities of radioactivity will involve consideration of specific practices (e.g., airborne releases vs land disposal) and, perhaps, specific locations (e.g., disposal in humid vs arid environments).

#### Recommendations of National and International Agencies

The values of an annual de minimis dose that have been considered by regulatory authorities in the U.S. and elsewhere are summarized in Table 1; values are listed in order of decreasing dose. These recommendations are discussed in the following paragraphs, beginning with those from U.S. agencies.

The NRC has not yet established a generally applicable de minimis dose. However, the NRC apparently is considering a de minimis dose as part of an extensive revision of radiation protection standards in 10 CFR Part 20 (15). The suggested value is based on an assumed negligible lifetime risk of  $10^{-6}$ . The NRC also is considering a "regulatory cutoff level" of 1 mrem per year for members of the general public in population dose calculations (15); i.e., individual doses below 1 mrem per year would be excluded in applying the ALARA principle to specific sites.

The EPA is considering levels of radioactivity that would be below regulatory concern for airborne emissions (8) and low-level waste disposal (16). The EPA has performed calculations for low-level waste disposal which suggest that many low-activity waste streams would yield individual doses less than 1 mrem per year, although doses as large as 70 mrem per year also were obtained (16). However, the EPA has not yet indicated a dose that might be de minimis for waste disposal or any other purpose.

The two recommendations attributed to the DOE represent proposals by DOE contractors (17-19), neither of which has been endorsed by the agency. An important feature of the proposal from EG&G Idaho (17,18) is that the de minimis dose should decrease as the size of the exposed population increases. The value of 30 mrem per year, for example, likely would apply to the case of inadvertent intrusion onto a disposal facility for exempt radioactive wastes.

Table 1. Recommendations of National and International Agencies  
on a De Minimis Dose

Agency	Annual Dose (mrem) <sup>a</sup>
U.S. Department of Energy <sup>b</sup>	30 <sup>c</sup>
Atomic Energy Control Board (Canada)	5
National Radiological Protection Board (U.K.)	5 <sup>d</sup>
U.S. Department of Energy <sup>e</sup>	1
National Council on Radiation Protection and Measurements	1
International Atomic Energy Agency	1 <sup>f</sup>
U.S. Nuclear Regulatory Commission <sup>g</sup>	0.1 (?)
U.S. Environmental Protection Agency	?

<sup>a</sup>Values are annual committed effective dose equivalents, and may be converted to units of mSv by multiplying by 0.01.

<sup>b</sup>Proposal of EGYG Idaho (17,18) for DOE's National Low-Level Radioactive Waste Management Program.

<sup>c</sup>Value applies to populations less than 1,000. recommended value is 3 mrem for populations of  $10^3$ - $10^6$  and 0.3 mrem for populations greater than  $10^6$ .

<sup>d</sup>Recommended value is 0.5 mrem when an individual could be exposed to several exempt sources.

<sup>e</sup>Proposal from Paducah Gaseous Diffusion Plant (19).

<sup>f</sup>Value is intended for application only to man-made radionuclides.

<sup>g</sup>Unpublished proposal for revision of 10 CFR Part 20.

The recent recommendation from the NCRP is based on an assumed negligible risk of  $10^{-7}$  per year, and is 1% of the proposed limit on acceptable dose from all sources for continuous exposures of 100 mrem (1 mSv) per year (4). The NCRP also has recommended that population dose assessments exclude those individuals who receive doses less than 1 mrem (0.01 mSv) per year.

The recent proposal from the Atomic Energy Control Board of Canada is based on an assumed negligible risk of  $10^{-6}$  per year (20). A further recommendation is that the potential for exposure of large populations to the de minimis dose will be small; e.g., the radiological impacts from disposal of exempt materials will be localized.

The recommendation from the U.K.'s National Radiological Protection Board is 1% of the dose limit for individuals in the general public (21). However, the recommended de minimis dose from any practice is reduced by a factor of 10 when an individual could be exposed to several exempt sources.

The proposal attributed to the International Atomic Energy Agency represents the recommendation of two advisory groups (22,23), and is based on an assumed negligible risk of  $10^{-7}$  per year. One advisory group also recommended that a practice may be left unregulated if the population dose commitment were less than 100 person-rem (1 person-Sv) per year of operation (23). The advisory groups also point out that the proposed de minimis dose essentially applies only to man-made radionuclides, because it is illogical to apply a limit of 1 mrem (0.01 mSv) per year for exempting naturally occurring radionuclides which, in their undisturbed state, lead to doses greater than the de minimis value. This is a particularly important consideration in determining exempt concentrations of naturally occurring radionuclides for purposes of disposal.

In summary, the recommendations of the various national and international agencies on a generally applicable de minimis dose are in the range 0.1-30 mrem (0.001-0.3 mSv) per year. Most recommendations focus on 1 or 5 mrem per year.

#### RECOMMENDATIONS FOR A DE MINIMIS DOSE

We would offer the following comments on the various proposals for a de minimis dose.

- [1] A de minimis dose should be expressed in terms of the committed effective dose equivalent, as recommended by the ICRP (5) and the NCRP (4). The effective dose equivalent is more closely related to the goal of risk limitation than is the dose equivalent to whole body or the critical organ which has been used by U.S. Government agencies [e.g., see refs. (2), (3), and (6-9)].
- [2] A de minimis dose need not be expressed in terms of a limit for each year of exposure, as is customary in all of the proposals reviewed, primarily because the negligible risks on which the de minimis dose are based generally are not constant over a lifetime. The alternative of expressing an annual de minimis dose as an average over a lifetime is more closely related to the basic goal of limiting lifetime risk. This approach also encourages proper consideration of the age-dependence of dose and risk in deriving exempt quantities of radionuclides from the de minimis dose. One can specify both a limit on annual dose averaged over a lifetime and a higher dose limit in any year.
- [3] An annual dose of 30 mrem (17,18) is probably too high to be acceptable as de minimis in the U.S., because of the widespread use of 25 mrem as a dose limit for specific practices (6-9). A de minimis dose, by definition, must be below any dose that is of regulatory concern.
- [4] An annual dose of 0.1 mrem (15) may be too low to be acceptable as de minimis, because of the great difficulties that likely would be encountered in measuring associated quantities of radioactivity. Furthermore, this dose is only about 0.1% of the dose from natural background radiation. Such a low de minimis dose for each source perhaps could be justified if exposures of individuals to many exempt sources were likely.
- [5] An annual de minimis dose in the range 1-5 mrem (4,20-23) seems most reasonable. Such de minimis doses correspond to lifetime risks from continuous exposure of about  $10^{-4}$ - $10^{-5}$ .

From the considerations outlined above, we offer the following proposal for a de minimis dose:

- [1] a limit on annual committed effective dose equivalent averaged over a lifetime of 1 mrem (0.01 mSv); and
- [2] a limit on committed effective dose equivalent in any year of 5 mrem (0.05 mSv).

This proposal is based largely on the recommendation of the NCRP that a limit on de minimis dose be set at 1% of the limit on acceptable dose from all sources of exposure, and that the limit on acceptable dose be set at 100 mrem (1 mSv) per year for continuous exposures and 500 mrem (5 mSv) per year for occasional exposures (4). The proposed de minimis dose corresponds to a lifetime risk from continuous exposure of about  $10^{-5}$ .

#### REFERENCES

1. C. B. Meinhold, "Considerations of de minimis and Exempt Quantities," paper presented at National Council on Radiation Protection and Measurements 1985 Annual Meeting on Waste Disposal, Washington, D.C., April 3-4, 1985.
2. U.S. Nuclear Regulatory Commission, "Part 20 - Standards for Protection Against Radiation," in Code of Federal Regulations, Title 10, Parts 0 to 199, pp. 234-273, U.S. Government Printing Office, Washington, D.C., 1985.
3. U.S. Department of Energy, "Environmental Protection, Safety, and Health Protection Program for DOE Operations," Order 5480.1, May 5, 1980.
4. National Council on Radiation Protection and Measurements, Recommendations on Radiation Exposure Limits, draft report, July 25, 1985.
5. International Commission on Radiological Protection, "Recommendations of the International Commission on Radiological Protection," ICRP Publication 26, Ann. ICRP 1, No. 3, pp. 1-53, 1977.
6. U.S. Environmental Protection Agency, "Part 190 - Environmental Radiation Protection Standards for Nuclear Power Operations," in Code of Federal Regulations, Title 40, Parts 190 to 399, pp. 6-7, U.S. Government Printing Office, Washington, D.C., 1984.
7. U.S. Environmental Protection Agency, "Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," 40 CFR Part 191, Final Rule, August 15, 1985; see also Fed. Registr. 47, pp. 58196-58206, 1982.
8. U.S. Environmental Protection Agency, "National Emission Standards for Hazardous Air Pollutants; Standards for Radionuclides," 40 CFR Part 61, Fed. Registr. 50, pp. 5190-5200, 1985.
9. U.S. Nuclear Regulatory Commission, "Licensing Requirements for Land Disposal of Radioactive Wastes," 10 CFR Part 61, Fed. Registr. 47, pp. 57446-57482, 1982.

10. U.S. Environmental Protection Agency, "Part 192 - Health and Environmental Protection Standards for Uranium and Thorium Mill Tailings," in Code of Federal Regulations, Title 40, Parts 190 to 399, pp. 7-14, U.S. Government Printing Office, Washington, D.C., 1984.
11. Final Environmental Impact Statement for Remedial Action Standards for Inactive Uranium Processing Sites (40 CFR 192), Vol. I, EPA 520/4-82-013-1, U.S. Environmental Protection Agency, 1982.
12. U.S. Nuclear Regulatory Commission, "Part 30 - Rules of General Applicability to Domestic Licensing of Byproduct Material," in Code of Federal Regulations, Title 10, Parts 0 to 199, pp. 284-305, U.S. Government Printing Office, Washington, D.C., 1985.
13. U.S. Nuclear Regulatory Commission, "Part 40 - Domestic Licensing of Source Material," in Code of Federal Regulations, Title 10, Parts 0 to 199, pp. 366-391, U.S. Government Printing Office, Washington, D.C., 1985.
14. U.S. Nuclear Regulatory Commission, "Part 71 - Packaging and Transportation of Radioactive Material," in Code of Federal Regulations, Title 10, Parts 0 to 199, pp. 654-690, U.S. Government Printing Office, Washington, D.C., 1985.
15. U.S. Nuclear Regulatory Commission, "Standards for Protection Against Radiation," unpublished Proposed Rulemaking, August 1982.
16. U.S. Environmental Protection Agency, Proposed Low-Level Radioactive Waste Standards (40 CFR 193), draft Background Information Document, March 13, 1985.
17. Threshold Limit Guidance, draft report, DOE/LLW-41T, EG&G Idaho, January 1985.
18. Development of Threshold Guidance, draft report, DOE/LLW-40T, EG&G Idaho, January 1985.
19. T. A. Jokerst, Proposed De Minimis Radioactivity in Waste, KY/B-243, Paducah Gaseous Diffusion Plant, 1984.
20. Atomic Energy Control Board (Canada), "The Basis for Exempting the Disposal of Certain Radioactive Materials from Licensing," draft Regulatory Policy Statement, May 6, 1985.
21. "NRC Staff Proposes Part 20 Revisions but Key Controversies Remain," p. 10 in Inside NRC, May 27, 1985.
22. W. E. Kennedy, Jr., "Foreign Travel Report," Pacific Northwest Laboratory, January 16, 1985.
23. "Statement by the Group of Senior Experts on General Principles for Exemptions from the Basic Safety Standards," International Atomic Energy Agency, Vienna, April 24, 1985.

ALLOWABLE RESIDUAL CONTAMINATION LEVELS  
FOR DECOMMISSIONING: REVISIONS TO THE METHOD<sup>(a)</sup>

W. E. Kennedy, Jr. and B. A. Napier  
Pacific Northwest Laboratory  
P.O. Box 999  
Richland, Washington 99352

ABSTRACT

In 1983, Pacific Northwest Laboratory published a series of reports that describe the Allowable Residual Contamination Level (ARCL) method for decommissioning U.S. Department of Energy (DOE) nuclear facilities and sites. The purpose of the method was to provide engineers with a useful tool for conducting safety and cost comparisons for decommissioning alternatives. The ARCL results are based on a scenario/exposure-pathway analysis and compliance with an annual radiation dose limit. Recently, revisions were made to the method that are designed to make it more useful for site- or facility-specific model applications. These revisions include analysis of additional exposure scenarios and pathways, the addition of area/volume correction factors to account for small areas or volumes of residual contamination, and an evaluation of barrier and waste-form credit factors to be applied in the analysis. This paper contains a description of the area/volume correction factors that are intended to permit the model to account for small areas or volumes of residual contamination.

INTRODUCTION

The Allowable Residual Contamination Level (ARCL) method was developed during 1983 to provide data useful to UNC Nuclear Industries engineers in conducting safety and cost comparisons for evaluating Hanford Site decommissioning alternatives.<sup>(1,2)</sup> The ARCL method and example results presented were based on a radiation scenario/exposure-pathway analysis and compliance with an annual radiation dose limit. To permit full consideration of decommissioning alternatives, three specific modes of future use of the land and facilities were included in the scope of the method. The three modes of use were restricted, controlled, and unrestricted. Specific radiation exposure scenarios for individuals who may use the land or facilities after decommissioning were defined and modeled for each mode of future use and for various modes of soil contamination.

Since 1983, the ARCL method has been adopted as the official analysis method for decommissioning property at the Hanford Site. In addition, UNC Nuclear Industries staff (with concurrence from the Richland Operations Office of the U.S. Department of Energy) have identified additional assumptions to be used in planning exercises. As a result, questions have been raised about the application of the method to site-specific conditions. To answer these questions, and to make the method easier to apply to field situations, modifications to the original method have become necessary.

---

(a) Work supported by the U.S. Department of Energy's Surplus Facilities Management Program Office through UNC Nuclear Industries' Decommissioning Programs Department.

This paper describes the area/volume modifications to the scenario-specific dose conversion factors developed by Pacific Northwest Laboratory (PNL) staff for use in the ARCL method. The modifications are in three major areas: 1) area correction factors for unconfined soil contamination, 2) volume correction factors for confined soil contamination (buried at depths between 1-5 m), and 3) root penetration and area correction factors for deeply confined soil contamination (buried at depths of 5 m or greater).

## SURFACE SOIL CONTAMINATION

The Residential/Home-Garden exposure scenario used in the original ARCL analysis (1,2) is designed to represent the unrestricted use radiation exposure conditions for an individual who resides on a site that has been decommissioned. The maximally exposed individual is assumed to spend 12 h/d outdoors on the site. During this time he is exposed to direct penetrating radiation from the contaminated soil. The individual is also assumed to inhale resuspended contamination in the surface soil for 12 h/d, with an assumed air concentration calculated using a time-dependent resuspension factor to account for environmental "aging" of radionuclides (2; pp 47-48). The individual is also assumed to raise and ingest 50% of his fruit and vegetable diet in a backyard home garden located in the contaminated surface soil. This exposure scenario is designed to approximate the conditions described for the intruder agricultura scenario by the U.S. Nuclear Regulatory Commission (NRC) in the Draft Environmental Impact Statement (DEIS) in support of 10 CFR Part 61.(3)

The Residential/Home-Garden scenario was designed to model sites with rather large areas of contaminated surface or subsurface soil. No attempt was made in the original ARCL modeling effort to account for sites of limited size (i.e., a contaminated surface area of less than 1 ha). However, in a related modeling application, PNL developed area correction factors for near surface (0 to 1 m of cover) soil contamination.(4,5) The results were documented as part of the ONSITE/MAXI1 computer program for the U.S. NRC.

Area correction factors for the Residential/Home-Garden scenario were developed as an extension of the modeling effort begun in the ONSITE/MAXI1 code documentation. These correction factors were designed to modify the ARCL dose conversion factors, by radiation exposure pathway, to account for contaminated sites smaller than 1 ha. A scenario modification factor worksheet, compatible with the previous ARCL worksheets (1,2) for unconfined surface soil, was also developed.

## External Exposure Area Correction Factors

For external exposure, a sensitivity study was conducted for various beta-gamma emitters to determine the penetrating (total-body) exposure rate versus source area using the ISOSHLID (6) computer program. The results of the calculations are shown in Figure 1. Because the curves in Figure 1 are parallel, there appears to be a uniform correction for site area over a large range of penetrating (total-body) source energies. Thus, we have determined the ratio of the exposure rates (for small to large area sources) and plotted the results versus the fractional hectare of source area, as shown in Figure 2. This figure defines the area correction factors for the direct exposure pathway for various fractions of a hectare. The correction factors default to a value of 1.0 (no correction) at about 0.125 ha, or about 1,250 m<sup>2</sup>. This area corresponds to a site size of about 35 x 35 m.

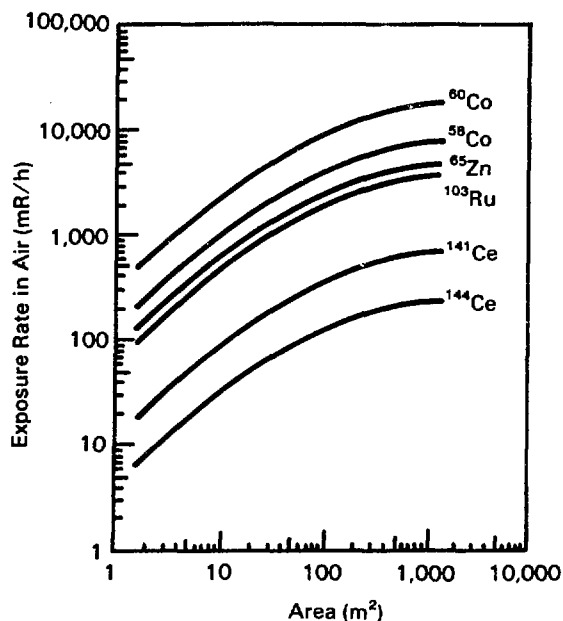


Figure 1 Direct Exposure Rate  
Versus Source Area.

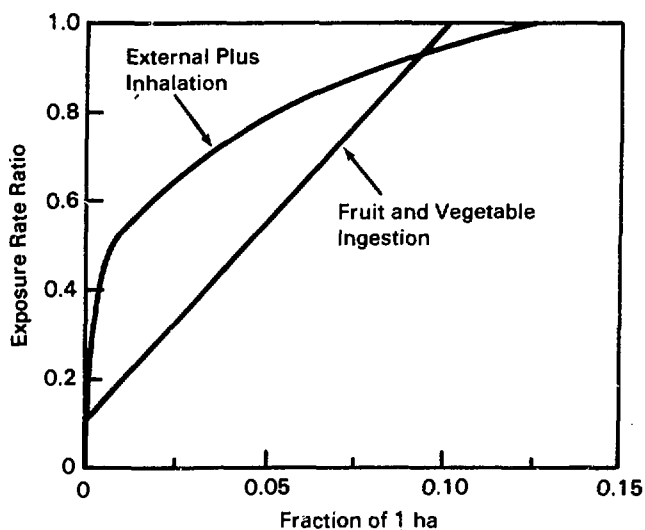


Figure 2 Allowable Residual Contamination  
Level Scenario Modification Factors -  
Unconfined Soil (0-1 m Cover).

#### Inhalation Area Correction Factors

After carefully considering the potential magnitude of the area correction for the inhalation exposure pathway, it was determined that the same relationship defined for external exposure would be applied. This is because many of

the experiments relating air and soil concentrations were performed on sites of limited size (i.e., sites with areas of 1 ha or less). (7,8,9) Thus, it is perhaps not unreasonable to assume that the area correction factor would approach 1.0 for a relatively small site (0.1 to 0.2 ha). The combined external plus inhalation area correction factors to be used in the ARCL method are shown in Figure 2.

#### Ingestion Area Correction Factors

The Residential/Home-Garden scenario used in the ARCL method is based on the assumption that the maximally exposed individual grows a backyard garden from which he obtains half of his annual fruit and vegetable diet. This assumption is made because residential family gardens commonly provide a significant fraction of the family diet. To produce half of the fruit and vegetable diet, the garden would have to occupy an area of at least 0.1 ha. This value is based on the following estimates for the general area required to produce a year's supply of garden food products: 1) about 50 m<sup>2</sup> would be required for leafy and other above-ground vegetables, 2) about 200 m<sup>2</sup> would be required for root vegetables and grains, and 3) somewhat over 200 m<sup>2</sup> would be required for fruit trees. (5) Of course, these estimates would vary depending upon the agricultural methods used and the climatic conditions encountered. It is suggested by these general estimates that 0.1 ha would be sufficient for producing half of the annual fruit and vegetable diet for a family of 4. Thus, the fraction of the plant food diet that is assumed to be grown in contaminated soil is assumed to be proportional to the contaminated area for sites smaller than 0.1 ha, as shown in Figure 2.

#### CONFINED SOIL CONTAMINATION (1-5 m) OF COVER

As an extension to the Residential/Home-Garden scenario used in the original ARCL analysis (1,2), confined soil contamination located between 1-4 m from the surface was considered. However, for the confined soil contamination scenario, it was assumed that the individual digs the basement for a house mixing buried contaminated material with clean soil. The individual was assumed to distribute this material uniformly around the surface of the site. The final concentration of the radioactive material on the surface of the site was estimated in Kennedy and Napier (2) by assuming that 200 m<sup>3</sup> of radioactive material was mixed with 800 m<sup>3</sup> of clean overburden. Thus, the resulting soil contamination level in this scenario is a direct function of the volume of radioactive material encountered during the excavation. Again, the maximally exposed individual is assumed to spend 12 h/d outdoors on the site during which he is exposed to penetrating radiation from contaminated soil. The individual also is assumed to inhale resuspended soil contamination for 12 h/d and reise and ingest 50% of his annual fruit and vegetable diet in the contaminated surface soil. This scenario is designed to approximate the conditions described for the intruder agriculture scenario by the U.S. NRC in the Draft Environmental Impact Statement in support of 10 CFR Part 61.(3)

Volume correction factors were developed next for the Residential/Home-Garden scenario as an extension of the ARCL method. (1,2) These volume correction factors are designed to modify the radionuclide concentration in soil that is determined for the scenario analysis to account for various volumes of radioactive materials and various depths of overburden. They were developed to be applied only to the Residential/Home-Garden scenario for residual contamination located between 1-5 m from the surface after decommissioning.

## Description of Volume Correction Factors

In the original ARCL Residential/Home-Garden scenario, the basement digging activities of the maximally exposed individual were defined.(1,2) An individual was assumed to dig a basement for a house into a subsurface radioactive soil (or debris) zone. The radioactive soil was assumed to be 2.5 m from the surface. Typical surface areas for a house were defined as being about 20 m by 10 m, for an area of 200 m<sup>2</sup>. This dimension was used as the dimension of the foundation hole. The foundation hole was assumed to be 3.5 m deep, with surface dimensions of 26 m by 16 m. The total volume of the excavation was assumed to be about 1,000 m<sup>3</sup>. Again, these assumptions were designed to approximate the assumptions used by the U.S. NRC in the DEIS supporting 10 CFR 61.(3)

This scenario and these defined conditions were intended to be used by UNC Nuclear Industries engineers for planning purposes, fully realizing that different cover depths or volumes of radioactive materials would change the estimated results. However, it was felt that this single set of scenario conditions was adequate for an initial comparison of decommissioning alternatives. Now that a full evaluation of site-specific field conditions is required, modifications must be made to the original assumptions used in this scenario to permit the development of volume correction factors.

To fully consider wastes potentially buried to a depth of 5 m, the scenario description is modified as follows. The base area of the basement and total volume of the excavation are assumed to remain the same (i.e., an area of 20 m by 10 m and a volume of 1,000 m<sup>3</sup>); however, the depth of the basement is changed from 3.5 m to 5 m. The ARCL scenario modification factor is described by the ratio of the volume of contaminated soil or radioactive material encountered in the excavation to the volume of the excavation (1,000 m<sup>3</sup>).

This ARCL modification is intended to account for a wide variety of potential conditions for residual radioactivity located at depths between 1-5 m after decommissioning operations. These conditions include the limited volumes of radioactive materials associated with historical spills and uniform thin "sheets" of subsurface radioactivity.

The following points should be noted about the ARCL volume correction factor:

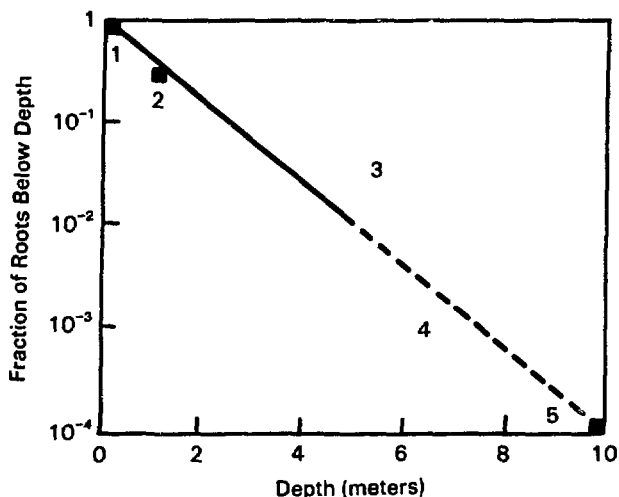
- The default value for material in the top meter of soil is assumed to be 1.0 (i.e., the area correction factor previously discussed should be used instead of the volume correction factor).
- The default value for material deeper than 5 m is assumed to be 0.01, (i.e., the dose-conversion factors for confined soil with 5 m of cover should be used instead of a volume correction factor).
- If 200 m<sup>3</sup> of radioactive material is encountered, the volume correction factor equals the factor identified in the original method (1,2) and the estimated results are identical.

### CONFINED SOIL CONTAMINATION (GREATER THAN 5 M OF COVER)

A second confined soil contamination condition was considered in the original method (1,2) that consisted of the burial of radioactive material under a cover greater than 5 m deep. For this soil contamination condition, a root penetration scenario was defined using an assumed root penetration factor of 1%. The root penetration factor used was based on previous work by Napier (10) in modeling the fraction of plant roots that penetrate overburden to various

depths. Figure 3 shows crop root penetration as a function of depth. For the analysis, the crop uptake of radionuclides was assumed to be directly proportional to the fraction of active roots that penetrate the overburden. The root penetration value of 1% (corresponding to an approximate overburden depth of about 5 m) was used to provide a modeling approximation for deeply buried material. Other root penetration values, corresponding to alternative overburden depths, may be used as depth correction factors.

Root penetration and contaminated area modifications, as defined for the surface soil contamination condition, can be made to the root penetration scenario in the original ARCL method. Each of these modifications are intended to provide additional flexibility in modeling the site-specific conditions that may be encountered during decommissioning. These modifications are especially needed to help account for the diversity of facilities and conditions existing at the Hanford Site.



#### Legend

1. 100% of plant roots are assumed to be at or below the soil surface.
2. 70% of alfalfa roots are in the top 1 meter. (11)
3. Solid line adapted from results for cumulative percent frequency of rooting depth for 1,012 vascular plants. (12)
4. Extrapolation of point 3 (dashed line).
5. Arbitrary 10-meter cutoff.

Figure 3 Crop Root Penetration Depths (10).

#### SUMMARY

The ARCL method was originally developed in 1983 to assist UNC Nuclear Industries engineers in evaluating Hanford Site decommissioning alternatives. Since its development, the ARCL method has been adopted as the official analysis method for decommissioning property at the Hanford Site. The ARCL method uses a radiation scenario/exposure pathway analysis and to evaluate compliance with an annual dose limit. For ease in application, worksheets were provided that rely on scenario-specific dose-conversion factors in evaluating the potential doses

from a specified inventory of radioactive materials. These dose conversion factors were developed for assumed exposure and contaminated material conditions. To account for conditions that differ from the assumed conditions, and to provide flexibility in the method, modifications to the original dose conversion factors and the overall method application were required. These modifications include: 1) area correction factors for unconfined (surface) soil contamination, 2) volume correction factors for confined soil contamination (buried at depths between 1-5 m), and 3) root penetration and area correction factors for deeply confined soil contamination (buried at depths of 5 m or greater).

## REFERENCES

1. W. E. Kennedy, Jr. and B. A. Napier, Allowable Residual Contamination Levels for Decommissioning the 115-F and 117-F Facilities at the Hanford Site, PNL-4704, Pacific Northwest Laboratory, Richland, Washington, 1983.
2. W. E. Kennedy, Jr. and B. A. Napier, Allowable Residual Contamination Levels for Decommissioning Facilities in the 1DD Areas of the Hanford Site, PNL-4722, Pacific Northwest Laboratory, Richland, Washington, 1983.
3. U.S. Nuclear Regulatory Commission, Draft Environmental Impact Statement on 10 CFR Part 61, Licensing Requirements for Land Disposal of Radioactive Waste, NUREG-0782, Washington, D.C., 1981.
4. B. A. Napier, R. A. Peloquin, W. E. Kennedy, Jr., and S. M. Neuder, Intruder Dose Pathway Analysis for the Onsite Disposal of Radioactive Wastes: The ONSITE/MAXI1 Computer Program, NUREG/CR-3620, U.S. Nuclear Regulatory Commission, Washington, D.C., 1984.
5. W. E. Kennedy, Jr., R. A. Peloquin, B. A. Napier, and S. M. Neuder, Intruder Dose Pathway Analysis for the Onsite Disposal of Radioactive Wastes: Supplements to the ONSITE/MAXI1 Computer Program, NUREG/CR-3620, Suppl. 1, U.S. Nuclear Regulatory Commission, Washington, D.C., 1985.
6. G. L. Simmons, J. J. Regimbal, J. Greenborg, E. L. Kelly, Jr., H. H. Van Tuyl. 1967. ISOSHLD II Code Revision to Include Calculation of Dose Rate from Shielded Bremsstrahlung Sources, BNWL-236, Suppl. 1, Pacific Northwest Laboratory, Richland, Washington, 1967.
7. R. C. Milham, J. F. Schubert, J. R. Watts, A. L. Boni, and J. C. Corey, "Measured Plutonium Resuspension and Resulting Dose from Agricultural Operations on an Old Field at the Savannah River Plant in the Southeastern United States," in Proceedings of an IAEA/USERDA Symposium on Transuranium Nuclides in the Environment, E. I. du Pont de Nemours & Co., Aiken, South Carolina, 1974.
8. G. A. Sehmel, "Particle Resuspension from an Asphalt Road," in Atmosphere-Surface Exchange of Particulate and Gaseous Pollutants, CONF-740921, p. 862-880, National Technical Information Service, Springfield, Virginia, 1974.
9. G. A. Sehmel and F. D. Lloyd, "Resuspension Rates from a Circular Field Source." In Pacific Northwest Annual Report for 1975 to the DOE Assistant Secretary for the Environment, Part 3, Atmospheric Sciences, pp. 96-98, BNWL-2000, Pacific Northwest Laboratory, Richland, Washington, 1976.

10. B. A. Napier, A Method for Determining "Allowable Residual Contamination Levels" of Radionuclide Mixtures in Soil, PNL-3852, Pacific Northwest Laboratory, Richland, Washington, 1982.
11. Washington State University, Alfalfa Irrigation, E.M 3468, Cooperative Extension Service, Pullman, Washington, 1970.
12. T. S. Foxx, G. D. Tierney, and T. M. Williams. Rooting Depths of Plants Relative to Biological and Environmental Factors, LA-10254-MS, Los Alamos National Laboratory, Los Alamos, New Mexico.

DOSE GUIDELINES FOR  
DECONTAMINATION AND DECOMMISSIONING PROJECTS\*

Paul G. Voillequé  
Science Applications International Corporation  
P.O. Box 696  
Idaho Falls, Idaho 83402

ABSTRACT

Dose guidelines are needed for evaluations of the potential exposure of members of the public to residual radioactivity following decontamination and decommissioning (D&D). These are then translated into guidelines for radionuclide concentration in media of interest, such as soil, using scenarios that define potential exposure pathways. The scenarios consider individuals whose location and behavior cause them to receive doses higher than the average person. These persons form the critical groups for radiation exposure following D&D activities. Two dose guidelines were developed. The higher annual dose guideline (D, mrem) defines an action level. If the projected annual dose (H, mrem) exceeds D, then action to reduce the potential dose is required. The lower annual dose guideline (DD, mrem) defines a no-action level. Projected doses below this level are "trifling" and of no regulatory concern. Selection of the two dose guidelines was based upon a review of existing national and international guidance. The effective dose equivalent was judged to be the appropriate dosimetric quantity. The selected value of D is 100 mrem, corresponding to a risk of about  $10^{-5}$  per year. The selected value of DD is 1 mrem, carrying a risk of about  $10^{-7}$  per year. It is emphasized that DD is not a goal or target for D&D activities, but a no-action level. For projected doses in the range  $DD < H < D$ , the ALARA principle applies. The cost and feasibility of clean-up as well as the benefits must be evaluated to determine the ALARA endpoint for a particular action.

INTRODUCTION

Dose guidelines are needed to bound the potential exposure of the general public following decontamination and decommissioning (D&D) operations. The guidelines refer to doses that due to exposure to residual radioactivity and are in addition to doses from natural sources. Members of the public can be exposed to residual radionuclides operations through use of surplus equipment and facilities, intrusion into facilities, and farming on sections of land released to the public domain. Exposure scenarios consider members of the public whose location and behavior cause

---

\*This paper is based on work performed under contract for EG&G Idaho, Inc. The views presented are those of the author and do not represent the official position of EG&G Idaho, Inc. or the Department of Energy.

them to receive doses higher than those received by the rest of the population. These individuals thus form critical groups for the various exposure situations that may occur following D&D.

Two guidelines are required. The higher annual dose guideline (D, mrem) defines an action level. If the projected annual dose (H, mrem) exceeds D, then action to reduce contamination level or dose rate is required. The dose guideline D must not exceed the annual dose limit applicable for members of the public. The term "guideline" is used rather than "limit" because, as indicated, the two may be different.

When action is undertaken, the principle of reducing the dose to a level that is as low as is reasonably achievable (ALARA) applies and is emphasized. Cleanup actions would cease when costs of achieving an incremental dose reduction were not balanced by the incremental benefits expected. The ALARA endpoint will differ for different situations. In each case, approval of the ALARA endpoint must be documented. A vigorous ALARA program is an essential component of the D&D program.

The lower annual dose guideline (DD, mrem) defines a no-action level. The dose DD is sufficiently small that the risk of any health effects is below regulatory concern. This level of risk has also been called the de minimis level (from the Latin "de minimis non curat lex", roughly: "the Law does not concern itself with trifles"). For projected doses below this level ( $H < DD$ ) the risks are so "trifling" that no action is necessary to reduce them further.

It is emphasized that the de minimis risk level is not a goal for D&D projects. The goals are (a) to assure that projected doses do not exceed the action level (D) and (b) to demonstrate (and document) that the D&D process has followed the ALARA principle. The risk level DD defines the lower limit of the range of projected doses in which the ALARA procedure applies.

In the next section, the existing and anticipated radiation protection guidance of advisory groups and regulatory bodies is reviewed. That guidance limits the range of possible values for D and DD. The recommended dose guidelines are presented in the last section.

## REVIEW OF EXISTING GUIDANCE

Established and proposed radiation protection guidance of advisory and regulatory bodies has defined the range of possible values of guidelines D and DD. The value of D cannot exceed the corresponding annual dose limit. Previous regulatory actions also provide an upper bound for DD, a dose level that is below regulatory concern.

There are several organizations that develop and publish guidance in the field of radiation protection. The International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP) are advisory bodies that provide basic guidance at the international and national levels, respectively. Their recommendations are very important, but are not legally binding. The Federal Radiation Council (FRC) whose functions are now the responsibility of the Environmental Protection Agency (EPA), provides radiation protection guidance for all

Federal agencies. Among those agencies are the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE). The current and anticipated guidance of each organization cited is discussed in the subsections that follow.

#### International Commission on Radiological Protection

The most recent fundamental guidance of the ICRP was presented in ICRP Publication 26 (1). The ICRP recommended that the acceptability of occupational and public exposure to radiation be judged by comparing the risks of such exposure with other human activities. An average mortality risk of  $10^{-4}$  year<sup>-1</sup> was selected as a guide value for a "safe" occupation (2). The risk of death associated with radiation-induced cancer is estimated by the ICRP to be about  $10^{-4}$  per rem. The total risk of genetic damage in all future generations is also estimated to be about  $10^{-4}$  per rem. The sum of these two risks, when rounded up, is  $2 \times 10^{-4}$  per rem.

After reviewing other risks of modern life to which members of the public are exposed, the ICRP concluded that acceptable risks of death for the general public, due to activities over which it has little control, were about an order of magnitude lower than the risks of a safe occupation. The ICRP suggested that a risk in the range of  $10^{-6}$  to  $10^{-5}$  year<sup>-1</sup> would probably be acceptable to individuals. The risk level of  $10^{-5}$  year<sup>-1</sup> corresponds approximately to an annual whole body dose equivalent of 0.1 rem. The ICRP chose to maintain the annual limit of 0.5 rem for the general public, noting that the limit is applied to exposure of critical groups within the population and that the average dose actually received is generally considerably lower than that estimated for the critical group.

The ICRP expected that radiation exposure of members of the public for an extended period to annual doses approaching 0.5 rem would occur infrequently. However, if such exposure were anticipated, the ICRP deemed it prudent to abide by the risk level rather than the dose limit. This risk-based guide for continuous public exposure is conceptually most similar to the guideline for action for members of the public described above. The ICRP policy to limit the risk committed during each year is reflected in the use of the committed dose equivalent (3).

The risk to the individual from mixed internal and external exposure can be estimated using these concepts. The sum of the risk-weighted committed dose equivalents received by all tissues is termed the effective dose equivalent (4,5). The effective dose equivalent is the quantity that is limited, as described above, in the current ICRP radiation protection recommendations.

#### National Council on Radiation Protection and Measurements

The current recommendations of the NCRP were published in NCRP Report No. 39 (6). An evaluation of the general validity of radiation protection philosophy was later made by the NCRP and the current recommendations were reaffirmed at the time (7). The NCRP annual dose limit for whole body exposure of individual members of the public is 0.5 rem. The NCRP limit for continuous exposure of the U.S. population is 0.17 rem/year. The same limit was derived by the NCRP from consideration of (a) somatic (principally cancer induction) risks and (b) the risk of genetic effects in future generations.

The NCRP is currently developing a philosophical background for a dose limitation system that is totally risk-based. Although a formal NCRP report is not expected for some time, the outline of a proposed system was discussed at the 1984 NCRP Annual Meeting. The system under consideration contains limitations on risk that are similar to those of the ICRP.

In NCRP Report No. 77 (8), three recommendations regarding exposure to individual members of the public are quoted from a document in preparation. The quote is given below:

- "1. a maximum limit of 500 mrem in any one year (other than medical and natural background) to an individual member of the public is still recommended, but not for continuous or repeated exposures;
2. it is recommended that continuous exposures (other than medical and natural background) resulting in a dose equivalent of 100 mrem/y or more to individual members of the public be avoided;
3. in situations where continuous exposures resulting in dose equivalents of 100 mrem/y cannot be avoided, such as from elevated or enhanced natural sources, NCRP recommends a remedial action level of 500 mrem/y from external radiation from all sources except medical."

#### Environmental Protection Agency

The Environmental Protection Agency received the authority of the former Federal Radiation Council to "advise the President with respect to radiation matters directly or indirectly affecting health, including guidance for all Federal agencies in the formulation of radiation standards". The Federal Radiation Council published radiation protection guidance for Federal agencies in 1960 and 1961 (9,10). Those recommendations are still in force.

The EPA is in the process of adopting a system similar to that in Reference 1. An initial draft proposal for radiation workers, which introduced a scheme of risk weighting factors that differed from those in Reference 1, was roundly criticized. A revised EPA proposal, presumably more in concert with the approach of the ICRP, is in preparation.

In separate actions, the EPA has proposed or established a number of criteria for specific activities. Examples are (a) the guidelines for the nuclear fuel cycle (11), (b) environmental standards for the management and disposal of spent nuclear fuel, high-level and transuranic radioactive wastes (12), (c) the proposed Clean Air Act standards for DOE facilities, NRC-licensed facilities and non-DOE Federal facilities, underground uranium mines, and elemental phosphorous plants (13), and (d) standards for remedial action of inactive uranium processing sites (14). The annual dose criteria for whole body exposure from these activities range from 10 to 170 mrem. The EPA criteria were established after at least qualitative consideration of the potential benefits and costs of alternatives. That is, these criteria represent approximate ALARA endpoints for specific source categories. A partial exception to this characterization is the set of remedial action criteria. The remedial action criteria are similar in concept to the D&D action guideline D, although the costs of remediation were clearly taken into account when the EPA standards were developed.

## Nuclear Regulatory Commission

The basic radiation protection standards of the Nuclear Regulatory Commission (15) are closely related to the Radiation Protection Guides issued by the Federal Radiation Council. A draft proposed revision of 10 CFR 20 has been circulated informally (1983) and comments have been received by the NRC.

The 1983 draft revision of 10 CFR 20 included a specific proposal for a de minimis level: an annual effective dose equivalent of 1 mrem to an individual. As noted, the guideline DD corresponds to a de minimis dose or a level below regulatory concern. The NRC view of the de minimis concept and the rationale used in the selection of the proposed value are discussed by Cunningham (16).

Currently, the NRC is reevaluating residual radioactive contamination limits to provide more specific requirements related to decommissioning of NRC-licensed nuclear facilities (17). It is expected that the Commission will require residual radioactive contamination levels in unrestricted areas be unlikely to cause an individual to receive an annual effective dose equivalent in excess of a level set exclusively for decommissioning and NRC license terminations.

Existing NRC requirements for license terminations are contained in Regulatory Guide 1.86 (RG 1.86) (18). Licensees are required to make a reasonable effort to eliminate residual contamination from the facility premises. A comprehensive radiation survey should be performed to establish that the contamination is below the levels given in Table 1 of RG 1.86. Levels are specified only for surface contamination by alpha and beta-gamma radionuclides. Guidance for radioactive contamination in soil is not included in RG 1.86. The levels specified in RG 1.86 reflect an analysis of relative hazard categories rather than strict dosimetric calculations for each radionuclide. The levels in RG 1.86 are the same as those originally proposed in a 1974 draft standard submitted to the American National Standards Institute (ANSI). The draft standard is now referred to as ANSI N13.12. Surface contamination guidance in the current draft standard (19) is similar, but not identical, to that in RG 1.86.

## Department of Energy

The Department of Energy, as a Federal agency, is guided by the recommendations of the Federal Radiation Council. Thus, the FRC limits on dose equivalent form the current basis for DOE policies and future DOE policy will be consistent with the basic Federal guidance promulgated by the EPA. A general policy of the Department of Energy is that its guidance will be consistent with that applied by the Nuclear Regulatory Commission. This policy is expressed in Chapter V of DOE Order 5480.1A as follows:

"Nuclear facilities are sited, designed, constructed, modified, operated, maintained, and decommissioned in accordance with generally uniform standards, guides and codes which are consistent with those applied to comparable licensed nuclear facilities." (20)

Chapter V of DOE Order 5820.2 contains policy statements for Decontamination and Decommissioning of Surplus Facilities (21). According to DOE policy, the decontamination and decommissioning of surplus facilities for which DOE is responsible shall be managed in a safe, cost-effective manner to assure that exposure to radiation and hazardous chemicals complies with DOE standards. Radiological criteria (release criteria for residual radioactivity) shall be developed, as needed, based on accepted radiation protection standards and consideration of natural background radiation levels.

Guidelines are being developed for residual radioactive materials at sites identified by the Formerly Utilized Sites Remedial Action Program (FUSRAP) and at remote sites identified by the Surplus Facilities Management Program (SFMP). The guidance employs an annual effective dose equivalent guide of 100 mrem for exposure periods exceeding 5 years. Surface contamination guides that are currently employed by the NRC are applied to equipment and structures that would be released (not to those that would be demolished and buried) (22).

#### Other Groups

The Atomic Industrial Forum (AIF) sponsored a study of de minimis concentrations of radionuclides in radioactive wastes. The concentration values were derived from a de minimis annual dose level, selected to be 1 mrem (23). The AIF has also funded a study of the benefits, risks, and costs of establishing regulatory cutoff (de minimis) levels of radioactivity in certain waste streams from nuclear power plants. The study is considering three annual dose cutoff levels: 0.1, 1.0, and 10 mrem. Publication of results of the current study is expected in 1985.

Two International Atomic Energy Agency (IAEA) Advisory Groups have addressed the question of de minimis quantities of certain solid radioactive wastes released to the environment. The approach used was to relate the potential doses from such disposal to an acceptable level of risk to the public. It was concluded that a risk level of  $10^{-7}$  year<sup>-1</sup> was an appropriate level for involuntary risks to radiation from such wastes. On this basis, a de minimis level for annual dose to members of the public of about 1 mrem was chosen (24, 25, 26).

#### RECOMMENDED DOSE GUIDELINES

The dose guidelines recommended for use in D&D projects are presented in this section. The recommendations are consistent with current thinking about radiation dose and risk criteria of the international (ICRP) and national (NCRP) advisory bodies and with expected actions by other Federal agencies (EPA and NRC) and by the Department of Energy. Recommendations made in this section anticipate changes in Federal guidance and corresponding changes in DOE standards. Because most of the D&D work will be conducted in future years, it is prudent to anticipate the context and magnitude of the dose limits that will apply at that time.

The first recommendation is that the dosimetric quantity to which the guidelines apply should be the effective dose equivalent. It is expected that future guidance for Federal agencies will also be expressed in those terms.

The recommended annual dose guideline for exposure of a critical group who are part of the general public is 100 mrem. This recommendation follows the approach of the ICRP and the NCRP for bounding the risk of long term exposure (1,8). This dose quantity and guide value are the same as that used in the draft DOE FUSRAP/SFMP guidelines (22). This recommended action level limits the risk to members of the public to  $10^{-5}$  year<sup>-1</sup>. As stated previously, and reemphasized here, the ALARA principle (27) is an integral part of dose guideline framework for D&D activities. It applies for projected doses to members of the public (H) in the range  $DD < H < D$ . The dose and risk to the public will be reduced to an appropriate and approved ALARA endpoint for each D&D project.

The recommended de minimis dose guideline (DD) is 1 mrem. As indicated in the previous section, previous studies of a de minimis dose have most frequently selected this value. This dose corresponds to annual risk of death of  $10^{-7}$  year<sup>-1</sup>. It is less than one percent of the annual background radiation dose in the United States is about ten percent of the annual variation in the background dose. The health consequences of such a dose are minimal. A quotation from ICRP Publication 22 illustrates this point:

"At low levels of individual dose, e.g., those small by comparison with variations in local natural background, the risk to the individual is so small that his health and welfare will not be significantly changed by the presence or absence of the radiation dose." (27)

Perhaps the most dangerous aspect of the dose guideline DD is that it will be misinterpreted and its achievement considered a requirement or at least a target for all D&D activities. As stated, the de minimis dose is not a goal for any D&D project. Rather, it defines a lower bound for the application of the ALARA principle in the program. The cost and engineering feasibility of cleanup actions as well as their benefits must be evaluated (28) to determine the ALARA endpoint for each project. The "ALARA endpoint" is not a synonym of "de minimis dose". If the true goals of the D&D process and the meanings of the guidelines are repeated frequently enough, logic will prevail.

## APPLICATION

Concentration guide values that correspond to the dose guidelines can be derived by considering potential exposure scenarios and analyzing the radionuclide transport pathways that would result in radiation doses to the critical group. Calculations of concentration guides should be as realistic as possible, consistent with the amount of site specific data available.

The list of radionuclides of interest in a D&D project frequently contains many that are classified as difficult-to-measure (DTM). These radionuclides either emit no gamma rays or emit gamma rays with low energy or abundance. Alpha particles are the principle emissions of many of the DTM radionuclides. In situ monitoring of these radionuclides at low concentration levels is not feasible. Samples must be collected and could be taken to the laboratory for radiochemical analysis. However, Healy has pointed out that taking a crude laboratory to the field location provides more timely results. In the cleanup of residual Pu at TA-1 in Los Alamos,

a simple but efficient detector in the field was used for direct counting of samples of soil.

On the other hand, some radionuclides of interest emit energetic and abundant gamma rays as well as beta particles. These radionuclides, generally fission and activation products, are much easier to detect and can be measured in the field.

#### ACKNOWLEDGEMENT

The work on which this paper is based benefitted from the suggestions of many reviewers. Particularly helpful in clarifying the concepts and relationships presented were the comments of a review group consisting of Jack Corley, Keith Eckerman, Tom Gilbert, Jack Healy, Lamar Johnson, Charles King, and Bruce Napier.

#### REFERENCES

1. International Commission on Radiological Protection, Recommendations of the International Commission on Radiological Protection, ICRP Publication 26, Pergamon Press, Oxford, 1977.
2. International Commission on Radiological Protection, Problems in Developing an Index of Harm, ICRP Publication 27, Pergamon Press, Oxford, 1977.
3. International Commission on Radiological Protection, Statement from the 1984 Stockholm Meeting of the ICRP, Annals of the ICRP, 14, 1, 1984.
4. International Commission on Radiological Protection, Statement from the 1978 Stockholm Meeting of the ICRP, Annals of the ICRP, 2, unnumbered page, 1978.
5. International Commission on Radiological Protection, A Compilation of the Major Concepts and Quantities in Use by ICRP, ICRP Publication 42, Pergamon Press, Oxford, 1974.
6. National Council on Radiation Protection and Measurements, Basic Radiation Protection Criteria, NCRP Report No. 39, National Council on Radiation Protection and Measurements, Washington, D.C., 1971.
7. National Council on Radiation Protection and Measurements, Review of the Current State of Radiation Protection Philosophy, NCRP Report No. 43, National Council on Radiation Protection Measurements, Washington, D.C., 1975.
8. National Council on Radiation Protection and Measurements, Exposures From the Uranium Series With Emphasis on Radon and Its Daughters, NCRP Report No. 77, National Council on Radiation Protection and Measurements, Washington, D.C., 1984.

9. Federal Radiation Council, Background Material for the Development of Radiation Protection Standards, FRC Report No. 1, U.S. Department of Health, Education, and Welfare, Washington, D.C., 1960.
10. Federal Radiation Council, Background Material for the Development of Radiation Protection Standards, FRC Report No. 2, U.S. Government Printing Office, Washington, D.C., 1961.
11. U.S. Environmental Protection Agency, Environmental Radiation Protection Standards for Nuclear Power Operations, Federal Register, 42, 2858, 1977.
12. U.S. Environmental Protection Agency, Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes, Federal Register, 47, 58196, 1982.
13. U.S. Environmental Protection Agency, National Emission Standards for Hazardous Air Pollutants; Standards for Radionuclides, Federal Register, 48, 14076, 1983.
14. U.S. Environmental Protection Agency, Standards for Remedial Actions at Inactive Uranium Processing Sites, Federal Register, 48, 590, 1983.
15. U.S. Nuclear Regulatory Commission, Standards for Protection Against Radiation, Title 10, Code of Federal Regulations. Part 20 (10 CFR 20), current issue, 1985.
16. G. H. Cunningham, "The de minimis Concept From a Regulatory Standpoint," in Conference on Radiation Issues for the Nuclear Industry, E. David Harward (Ed), Atomic Industrial Forum, Bethesda, MD, 1983.
17. U.S. Nuclear Regulatory Commission, Plan for Reevaluation of NRC Policy on Decommissioning of Nuclear Facilities, NRC Report NUREG-0436, 1978.
18. U.S. Nuclear Regulatory Commission, Termination of Operating Licenses for Nuclear Reactors, NRC Regulatory Guide 1.86, 1974.
19. Health Physics Society Standards Committee, Surface Radioactivity Guides for Materials, Equipment, and Facilities to be Released for Uncontrolled Use, Draft Standard, Under revision, 1984.
20. U.S. Department of Energy, Environmental Protection, Safety and Health Protection Program for DOE Operations, DOE Order 5480.1A, 1981.
21. U.S. Department of Energy, Radioactive Waste Management, DOE Order 5820.2, 1984.
22. T. L. Gilbert (Coordinator), U.S. Department of Energy Guidelines for Residual Radioactive Materials at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites, Draft guide, 1984.
23. W. A. Rodger, S. S. Stanton, R. L. Frendberg, and H. W. Morton, de minimus (sic) Concentrations of Radionuclides in Solid Wastes, Atomic Industrial Forum Report AIF/NESP-016, Washington, D.C., 1978.

24. International Atomic Energy Agency, Considerations Concerning "de minimis" Quantities of Radioactive Waste Suitable for Dumping at Sea Under a General Permit, Report of Advisory Group, IAEA-TECDOC-244, 1981.
25. International Atomic Energy Agency, de minimis Concepts in Radioactive Waste Disposal, Report of Advisory Group, IAEA-TECDOC-282, 1981.
26. International Atomic Energy Agency, Definition of de minimis Quantities for Release of Low-Level Solid Radioactive Waste Into the Terrestrial Environment from Incineration Plant and Landfill Facilities, Methodological Implications, Report of Advisory Group, Unnumbered, 1982.
27. International Commission on Radiological Protection, Implications of Commission Recommendations that Doses be Kept as Low as Readily Achievable, ICRP Publication 22, Pergamon Press, Oxford, 1973.
28. International Commission on Radiological Protection, Cost-Benefit Analysis in the Optimization of Radiation Protection, ICRP Publication 37, Pergamon Press, Oxford, 1983.

## A MANUAL FOR IMPLEMENTING RESIDUAL RADIOACTIVITY GUIDELINES

Thomas L. Gilbert  
K. F. Eckerman  
W. R. Hansen  
J. W. Healy  
W. E. Kennedy  
B. A. Napier  
J. K. Solday  
Argonne National Laboratory  
Building 203  
Argonne, IL 60439

### ABSTRACT

The U.S. Department of Energy (DOE) has recently issued guidelines for residual radioactivity at Formerly Utilized Sites Remedial Action Program (FUSRAP) and remote Surplus Facilities Management Program (SFMP) sites. A manual for implementing these guidelines has been prepared jointly by four DOE laboratories (ANL, LANL, ORNL, and PNL) and is being issued as a supplement to the guidelines. The manual presents procedures and tables for deriving site-specific guidelines for levels of residual radionuclide concentrations in soil that must not be exceeded if a site is to be released for unrestricted use. Guidance for implementing DOE "ALARA" policy for remedial actions is also included.

The concentration factor method is used in the pathway analysis for deriving soil guidelines. The analysis has been structured in a manner that explicitly identifies all of the factors involved. Tables are provided for dose-conversion factors and pathway factors from which environmental transport factors for each radionuclide and pathway may be calculated. The scenarios used for deriving the environmental transport factors and dose conversion factors, and the manner in which the information provided in the manual is used to derive site-specific soil guidelines will be presented.

RESIDUAL SURFACE CONTAMINATION LIMITS:  
PROBLEMS IN INTERPRETATION AND IMPLEMENTATION

R. A. Wynveen, H. J. Moe, M. J. Robinet and V. R. Veluri  
Argonne National Laboratory  
Argonne, Illinois 60439

ABSTRACT

During the decontamination and decommissioning of facilities requiring remedial action, one is often faced with making decisions for the disposition of materials and equipment having various levels of residual surface radioactive contamination. The existing limits for the release of materials and equipment with such surface contamination have evolved historically from the proposed ANSI Standard N328, and U.S.N.R.C. Regulatory Guide 1.86.

Experience gained by the Health Physics Group at Argonne National Laboratory in decommissioning a number of varied facilities has revealed several difficulties in the interpretation and implementation of these limits. Included in these difficulties are: i) Interpretation of these limits with respect to certain radionuclides and radionuclide chains, ii) Apparent inconsistencies in the assignment of certain radionuclides to a given hazard group, iii) Limited guidance in the treatment of contaminated material and equipment containing contributions from several radionuclides, and iv) Implementation problems in the interpretation of instrument readings, particularly when the contamination level is at or near the instrument detection limit.

These and other problems are addressed from the standpoint of practical field applications. In addition, certain suggested modifications, designed to remove inconsistencies in the limits, are presented.

INTRODUCTION

Since the early days of the nuclear industry, surface radioactive contamination was recognized as a potential for internal radiation exposure of people. Control of surface contamination has, therefore, been an integral part of applied health physics and is especially important where materials and facilities are released for unrestricted use by the public.

There has been considerable effort in the United States and other countries to establish a firm basis for setting surface contamination limits. A number of papers have been published on the subject (references 1-8), one by Dunster as early as 1955. However, there has been a consensus that because there is no unique relationship between surface contamination levels and the resulting radiation dose to humans, the limits must be some-

what arbitrary.

The primary focus of this paper is not an evaluation of the numerical values of the limits, but rather to point out some, seemingly, serious ambiguities and inconsistencies relevant to the practical application of current surface contamination limits and then to propose some possible solutions.

#### BASES FOR SURFACE CONTAMINATION LIMITS

Though the numerical values of the limits are not the issue here, the indicated problems can, perhaps, be better appreciated if we have some insight about the bases which have been used for deriving and "selecting" surface contamination limits. Since the primary reason for controlling surface contamination is control of internal exposures, most of the early efforts concentrated on the risk associated with contamination entering the body by way of ingestion and inhalation.

##### Ingestion

The basic rationale for relating surface contamination to direct ingestion was that a person should be allowed to ingest the same amount of activity per day from a surface as one is allowed to ingest per day from drinking water which had a concentration of one MPC (now obsolete Maximum Permissible Concentration). For example, if the MPC for  $^{226}\text{Ra}$  in water was about 1 dis/minute per  $\text{cm}^3$ , then a person who drank 100  $\text{cm}^3$  of water per hour would ingest 100 dis/min each hour. If one then assumed that a person might ingest all of that activity from an area of 2  $\text{cm}^2$  per hour, the surface contamination limit for  $^{226}\text{Ra}$  would be 100 dis/min-hour divided by 2  $\text{cm}^2/\text{h} = 50$  dis/minute per  $\text{cm}^2$  (5,000 dis/min per 100  $\text{cm}^2$ ). Dunster (reference 1) assumed that a person might ingest all of the activity from 10  $\text{cm}^2$  of contaminated skin per day while Healey (reference 2) assumed an area of 1  $\text{cm}^2$  per hour.

There was concern that using this approach for radionuclides with much higher MPC values could lead to unnecessary radiation exposure via direct radiation, percutaneous absorption, etc. (see reference 2), and it would be prudent to arbitrarily select an upper limit value.

##### Inhalation

One approach used to relate surface contamination to inhalation was based on the fact that a portion of surface contamination can be resuspended and become available for inhalation. The ratio of resuspended activity concentration (i.e.,  $\mu\text{Ci}/\text{cm}^3$ ) to surface activity (i.e.,  $\mu\text{Ci}/\text{cm}^2$ ) is defined as the resuspension factor  $k$  ( $\text{cm}^{-1}$ ). Dunster (reference 3) found that  $k$  was in the range of  $10^{-7}/\text{cm}$  to  $2 \times 10^{-8}/\text{cm}$ . Using the MPC for air

(now replaced by the DAC) as an acceptable resuspended activity concentration, the corresponding surface contamination was simply the MPC divided by the resuspension factor. For example, an MPC of about  $1 \times 10^{-7}$  dis/minute per  $\text{cm}^3$  for  $^{239}\text{Pu}$  and a resuspension factor of  $2 \times 10^{-8}/\text{cm}$  would mean that the surface contamination limit for  $^{239}\text{Pu}$  would be 5 dis/min per  $\text{cm}^2$  (500 dis/minute per 100  $\text{cm}^2$ ). The use of resuspension factors was not widely accepted and other approaches were used to relate surface contamination to airborne activity concentration.

Most of the derived values for surface contamination limits were considered unnecessarily high because it was felt that the industry could easily afford lower values.

## EVOLUTION OF SURFACE CONTAMINATION LIMITS

The first United States surface contamination limits for materials and facilities to be released for unrestricted use were developed under the procedures of American National Standards Institute by Working Group 7 of the Health Physics Society Standards Committee for American National Standards Main Committee N13 (ANSI N13) on Radiation Protection. In about 1974, the Committee completed a proposed draft of ANSI N328 (1974), but it was never formally sent to ANSI. However, N328 was adopted by the NRC in the form of Regulatory Guide 1.86. Subsequent Health Physics Standards Committees adapted N328 into drafts and proposals of ANSI N13.12. Proposed ANSI N13.12 has never been made an ANSI Standard. In 1983, DOE published OR0831 which was an adaptation of the proposed N328. DOE later dropped the N328 version in favor of the Regulatory Guide 1.86 version and issued a final adoption of 1.86 in July of 1985.

The chronological chain of guideline proposals, adoptions and adaptations is shown in Figure 1. There is not sufficient space here to list all of the standards or to discuss each of them in detail. However, the following is a brief synopsis of the essential parts in each version of the limits.

### Proposed ANSI N328 (1974)

The limits selected by the N328 Committee were indirectly tied to MPC values but the Committee made it clear that these limits were by a consensus based on what appeared to be safe and practical with the existing technology. The limit for  $^{90}\text{Sr}$  was arbitrarily taken as 1000 dis/min per 100  $\text{cm}^2$  because it was approximately the value of background  $^{90}\text{Sr}$  contamination produced by fallout from past nuclear weapon tests. In addition, the Committee decided to use 5000 dis/min per 100  $\text{cm}^2$  as an upper limit for all radionuclides, since higher values could lead to unnecessarily high direct radiation exposure. Nuclides with maximum beta energies less than 150 keV were not considered. An abbreviation of the table of limits from N328 is shown in Table 1.

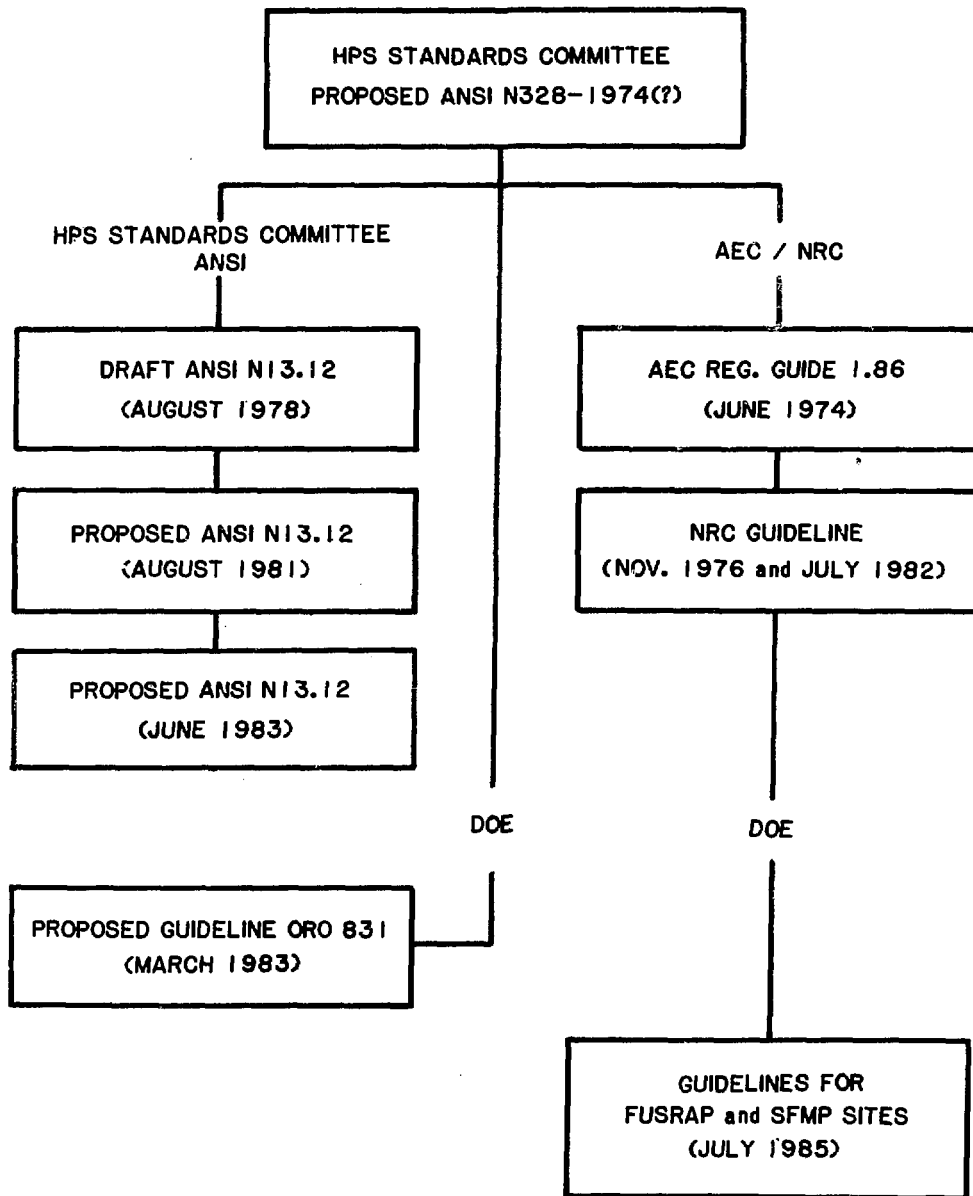


Figure 1. Interrelation of Surface Contamination Guidelines.

Table 1  
Surface Contamination Limits:  
Abbreviated Table From Proposed ANSI N328 (1974)

Nuclide	dpm/100 cm <sup>2</sup>	
	Total	Removable
Group 1: Nuclides for which the MPC for air is $2 \times 10^{-13}$ Ci/m <sup>3</sup> or less or for which the MPC for water is $2 \times 10^{-7}$ Ci/m <sup>3</sup> or less; includes Pu-239, Ra-226, Pb-210, I-125, I-129, etc.	100	20
Group 2: Those nuclides not in Group 1 for which the MPC for air is $1 \times 10^{-12}$ Ci/m <sup>3</sup> or less or for which the MPC for water is $1 \times 10^{-6}$ Ci/m <sup>3</sup> or less; includes Po-210; Sr-90; Th-232; U-232, etc.	1,000	200
Group 3; those nuclides not in Group 1 or Group 2.	5,000	1,000

#### REGULATORY GUIDE 1.86 (1974) AND NRC (1982)

An abbreviated version of the table of limit values from Regulatory Guide 1.86 (1974) and NRC (1982) is shown in Table 2. The only significant difference between the 1.86 (1974) table and the NRC (1982) table is that the NRC (1982) table has a footnote f which states: "The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h at 1 cm and 1.0 mrad/h at 1 cm respectively, measured through 7 milligrams per square centimeter of total absorber."

Table 2  
Surface Contamination Limits:  
Abbreviated Table From Regulatory Guide 1.86 (1974)

Nuclide	dpm/100 cm <sup>2</sup>		
	Average Total	Maximum Total	Removable
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay products	5,000	15,000 $\alpha$	1,000 $\alpha$
Beta-gamma emitters except Sr-90 and others noted above	5,000 $\beta$ - $\gamma$	15,000 $\beta$ - $\gamma$	1,000 $\beta$ - $\gamma$

## DOE GUIDELINE (JULY 1985)

The DOE (1985) table of limits is identical to the Regulatory Guide 1.86 (1974) table. It includes the NRC (1982) footnote on average and maximum dose rates of 0.2 and 1.0 mrad/h at 1 cm.

## DRAFT ANSI N13.12 (1978)

An abbreviation of the table of limits in ANSI N13.12 (1978) is shown in Table 3. ANSI N13.12 (1978) is different from N328 (1974) in that "nondetectable" was used for Group 1, and "nondetectable  $\beta\gamma$ " and 2000 $\alpha$  were used for Group 2.

Table 3  
Surface Contamination Limits:  
Abbreviated Table From Draft ANSI N13.12 (1978)

Nuclide	dpm/100 cm <sup>2</sup>	
	Total	Removable
Group 1: nuclides for which the MPC for air is $2 \times 10^{-13}$ Ci/m <sup>3</sup> or less or for which the nonoccupational MPC for water is $2 \times 10^{-7}$ Ci/m <sup>3</sup> or less; includes Pu-239, Ra-226, Pb-210, I-125, I-129, etc.	Nondetectable	20
Group 2: those nuclides not in Group 1 for which the MPC for air is $1 \times 10^{-12}$ Ci/m <sup>3</sup> or less or for which the MPC for water is $1 \times 10^{-6}$ Ci/m <sup>3</sup> or less; includes Po-210; Sr-90; Th-232; U-232, etc.	2,000 $\alpha$ Nondetectable $\beta\gamma$	200
Group 3: those nuclides not in Group 1 or Group 2.	5,000	1,000

## PROPOSED ANSI N13.12 (1981)

An abbreviation of the table of limits in proposed ANSI N13.12 (1981) is shown in Table 4. The limits in proposed ANSI N13.12 (1981) essentially returned to those of N328 (1974) except that the total value for Group 2 was changed from 1000 dpm/100 cm<sup>2</sup> to 5000 dpm/100 cm<sup>2</sup> and the iodines were moved to Group 2.

## PROPOSED ANSI N13.12 (1983)

The 1983 revision of proposed N13.12 was identical to proposed N13.12 (1981) except that it excluded depleted uranium, <sup>238</sup>U and <sup>232</sup>Th from Group 1. Table 5 is an abbreviation of the table of limits in proposed ANSI N13.12 (1983).

Table 4  
Surface Contamination Limits:  
Abbreviated Table From Proposed ANSI N13.12 (1981)

Group	Description	(dpm/100 cm <sup>2</sup> )	
		Total	Removable
1	Alpha emitters except Nat-U and Nat-Th	300	20
2	Sr-90, I-125, I-129, I-131, Ra-228	5,000	200
3	Beta emitters not in Group 2 with E <sub>max</sub> >150 keV; Nat-U and Nat-Th	5,000	1,000

Table 5  
Surface Contamination Limits:  
Abbreviation of Table From Proposed ANSI N13.12 (1983)

Group	Description	(dpm/100 cm <sup>2</sup> )	
		Total	Removable
1	Alpha emitters except Nat-U and Nat-Th, depleted uranium, U-238 and Th-232	300	20
2	Sr-90, I-125, I-129, I-131, Ra-228	5,000	200
3	All radionuclides not in groups 1 or 2 except beta emitters with E <sub>max</sub> < 150 keV	5,000	1,000

DOE ORO 831 (1983)

The ORO 831 (1983) guidelines were essentially identical to N328 (1974) except that a footnote was added which stated: "In the event of the occurrence of mixtures of radionuclides, the fraction contributed by each constituent to its own limit shall be determined, and the sum of the fraction shall be less than 1."

## INTERPRETATION AND IMPLEMENTATION PROBLEMS

### What is Surface Contamination?

The Regulatory Guide 1.86 version of the limits all list a total average value of 5000 dpm  $\beta\gamma/100 \text{ cm}^2$  for beta gamma emitters. How does one determine what fraction of high energy photons is "fixed" on the surface and what is in the material? The N328 (1974) Committee stated specifically that the proposed limits did not deal with radioactive contamination dispersed through material so the  $\beta\gamma$  terminology was not used (see Table 1). The footnote f in NRC (1982) further clouds the issue by stating that the dose rate from surface contamination must not exceed an average of 0.2 mrad/h at 1 cm and a maximum of 1 mrad/h at 1 cm. In addition to the problems of measuring true dose rate, why is it necessary to know the dose rate from a contaminated wall or floor at 1 cm? Perhaps, surface contamination limits should deal only with short range radiation.

### $^{226}\text{Ra}$ or $^{226}\text{Ra}$ and Daughters?

All versions of the limits list a value of 100 dpm/100  $\text{cm}^2$  for  $^{226}\text{Ra}$ . Does this really mean 100 dpm/100  $\text{cm}^2$  of only  $^{226}\text{Ra}$ , or does it mean 100 dpm  $\alpha$  from  $^{226}\text{Ra}$  and its daughters? If it means only  $^{226}\text{Ra}$ , one could use a gamma ray spectrometer and measure the 186.2 keV photons from  $^{226}\text{Ra}$  but there would be ambiguity about what was on the surface versus what was in the material. On the other hand, one could measure the alpha activity. If the daughters are in equilibrium, about 5 alpha particles are emitted for each transformation of  $^{226}\text{Ra}$ . Then, if 100 dpm  $\alpha/100 \text{ cm}^2$  was measured it would mean that the  $^{226}\text{Ra}$  activity was about 20 dpm/100  $\text{cm}^2$ . If 500 dpm  $\alpha/100 \text{ cm}^2$  was measured it would mean that the  $^{226}\text{Ra}$  activity was about 100 dpm/100  $\text{cm}^2$ . This interpretation decision alone could make a significant difference in the cost of decontamination and decommissioning.

### Natural-U, Natural-Th, and their Daughters

All versions of Regulatory Guide 1.86 and N13.12 (1981) list values for Nat-U and associated decay products, and Nat-Th and associated decay products which are greater than the values for some of their individual decay products. In Table 2 for example, the limit for natural-U is 5000 dpm/100  $\text{cm}^2$  while the limit for one of its daughters,  $^{226}\text{Ra}$ , is only 100 dpm/100  $\text{cm}^2$ . Unprocessed uranium ore involves over 35 radionuclides which collectively emit more than 10 alpha particles for each transformation of the parents. How can 35 radionuclides be less hazardous than a select few of them?

Besides the issue of the limits for a chain of nuclides versus that for individual members of the same chain, what, for example, does 5000 dpm/100 cm<sup>2</sup> for natural-U mean? As with the <sup>226</sup>Ra example, is it 5000 dpm  $\alpha$ /100 cm<sup>2</sup>, or what?

#### MIXTURES OF RADIONUCLIDES

Only ORO 831 (1983) has addressed the possibility of having mixtures of radionuclides which have different limit values. For example, if the limit for <sup>239</sup>Pu is 100 dpm $\alpha$ /100 cm<sup>2</sup> and that for <sup>90</sup>Sr is 1000 dpm $\beta$ /100 cm<sup>2</sup>, the acceptable levels of each (in dpm/100 cm<sup>2</sup>) could be:

<sup>239</sup> Pu	<sup>90</sup> Sr
100	0
80	200
50	500
0	1000 etc.

The mixture rule implies that a unique combination of reduced limit values are based on each field measurement. If there is 80 dpm $\alpha$ /200 cm<sup>2</sup> from <sup>239</sup>Pu, there could be 200 dpm $\beta$ /100 cm<sup>2</sup> from <sup>90</sup>Sr, etc. In application, it is impractical to make this kind of continuous adjustment of the limits in the field. Rather, for each mixture, a set of reduced limit values must be selected and applied uniformly.

There is also a measurement problem associated with mixtures which have alpha limits of 100 dpm/100 cm<sup>2</sup>. Measuring 100 dpm  $\alpha$ /100 cm<sup>2</sup> with a current state-of-the-art field instrument having an effective detection efficiency of between 0.1 and 0.5 cpm/dpm produces only 10-50 cpm.

In addition to the large uncertainty in this reading itself at this low level, if Sr-90 contamination were also present, the allowable contamination limit for Pu-239 would be even lower than 100 dpm/100 cm<sup>2</sup>. The allowable contamination limit would then be below the practical detection limit of the instrument.

A further difficulty in the application of the mixture rule is that the identity of the contaminants may not be well known in practical situations. This is particularly true for situations in which the initial contamination levels exceed the limits, but not by a large factor. This makes identification, particularly for radionuclides like Pu-239 very difficult, since one may be unable to obtain large enough samples for adequate analysis.

#### A PROPOSED PARTIAL SOLUTION

As we have attempted to demonstrate, there are different versions

of surface contamination limits. These embody rather minor differences in the numerical values which have been more or less "selected" by arbitrary consensus. However, the validity of the limit values in regard to actual hazard is not an acceptable excuse for ambiguity or inconsistency. In Table 6 an alternate version of the surface contamination limits is offered. The numerical value of the numbers is not the main issue and no attempt is being made to evaluate their validity in the context of risk potential. This is, seemingly, a separate issue. Its major purpose is to attempt to minimize ambiguity and inconsistency. Note that each limit value given in Table 6, states specifically what quantity is to be measured.

Table 6  
Alternative Surface Contamination Limits  
Argonne National Laboratory (1985)

Radionuclides <sup>1,2</sup>	(dpm/100 cm <sup>2</sup> )		
	Average Total	Maximum Total <sup>3</sup>	Removable
Transuranics, <sup>226</sup> Ra+D, <sup>228</sup> Ra+D <sup>230</sup> Th+D, <sup>232</sup> Th+D, <sup>231</sup> Pa+D <sup>238</sup> U+D, <sup>235</sup> U+D	100α	300α	20α
<sup>228</sup> Th, <sup>230</sup> Th, <sup>232</sup> Th <sup>238</sup> U, <sup>234</sup> U, <sup>235</sup> U (see footnote 4)	1000α	3000α	200α
<sup>90</sup> Sr- <sup>90</sup> Y, <sup>126</sup> I, <sup>131</sup> I, <sup>133</sup> I <sup>125</sup> I, <sup>129</sup> I	1000β	3000β	200β
Beta Emitters Other Than Those Listed Above <sup>5</sup>	5000β	15,000β	1,000β

<sup>1</sup>A nuclide +D indicates that the guideline value is for the sum of the activity from the nuclide and its daughters. Daughter nuclides are assumed to be in equilibrium with the parent nuclide. However, it is not necessary to make corrections for disequilibrium.

<sup>2</sup>Where there may be a mixture of both alpha and beta emitting nuclides, an attempt should be made to reduce the average total alpha contamination to zero and to apply the full value of the beta limit. Where there is a mixture of only alpha emitting nuclides, the most restrictive alpha limit should be applied. Similarly, where there is a mixture of only beta emitting nuclides, the most restrictive beta limit should be applied.

<sup>3</sup>The maximum contamination level applies to an area of not more than 100 cm<sup>2</sup>.

<sup>4</sup>Applies only to separated radionuclides where "near-term" potential hazards are given priority. Long-term hazard considerations would require application of the limits for the decay chains.

<sup>5</sup>In some cases, it may be impractical to measure in situ surface contamination of nuclides, such as tritium, which have very low energy beta particles. In such cases, the effective surface contamination may be calculated, for an assumed contamination depth, from measured activity per gram in surface samples. The assumed depth of contamination should not be less than the maximum range of the beta particle.

## CONCLUSION

While the problems highlighted in this paper focused on application of surface contamination limits as related to decontamination and decommissioning, these problems have relevance to all areas of health physics where surface contamination limits are applied. In real life, most health physicists are probably forced to use their professional judgement in interpreting surface contamination limits and they do not worry about what is or is not actually written in the guidelines. As a result, the same limits often have different interpretations. It seems that this is a condition which the profession cannot continue to support. Use of limits which have problems with inconsistency and interpretation could raise serious questions about the credibility of some D&D efforts.

We have attempted to whet your appetites to look at the limits from the practical field measurement standpoint. This is especially important since the measurements are most often made not by professional health physicists, but by technicians. In addition, we hope that we may have stimulated interest to attempt to provide sound justification for the merical values.

In 1964, the Health Physics Society cosponsored the First International Symposium on Surface Contamination. That symposium was held in Gatlinburg, Tennessee. Twenty-one years later, it seems appropriate that at this meeting, also being held in Tennessee, that the time has come to refocus on the surface contamination limits and give them the attention they deserve.

## REFERENCES

1. H. J. Dunster, "Surface Contamination Measurements as an Index of Control of Radioactive Material," Health Physicis . 8, pp. 353-356, 1962.
2. J. W. Healey, "Surface Contamination: Decision Levels," Los Alamos Scientific Laboratory, Report LA-4558-MS, National Technical Information Service, Springfield, VA. 1971.
3. H. J. Dunster, "The Concept of Derived Working Limits for Surface Contamination," in Surface Contamination edited by B.R. Fish, pp. 139-147, Pergamon Press, New York 1967.

4. H. J. Dunster, "Contamination of Surfaces by Radioactive Materials: The Derivation of Maximum Permissible Levels," Atomics . 6, pp. 233-239, 1955.
5. N. B. Schultz and A. F. Becker, "Correlation of Uranium Alpha Surface Contamination, Air-Borne Concentration, and Urinary Excretion Rate," Health Physicis . 9, pp. 901-909, 1963.
6. A. F. Becher, "The Development of Surface Alpha Contamination Limits" Symposium on Occupational Health Experience and Practices in the Uranium Industry, Health and Safety Laboratory Report HASL-58, pp. 151-156, 1958.
7. B. R. Fish, Editor, Surface Contamination, Proceedings of a Symposium held at Gatlinburg, Tennessee, June 1964, Pergamon Press, New York, 1967.
8. J. Shapiro, "Test for the Evaluation of Airborne Hazards from Radioactive Surface Contamination," Harvard School of Public Health, Boston, Massachusetts, (paper presented at June 1969 Annual Meeting of the Health Physics Society).

## DECISION MAKING DURING RADIOLOGICAL DESIGNATION SURVEYS

Jeanette Eng  
NJ Dept. of Environmental Protection  
Bureau of Radiation Protection  
380 Scotch Road  
Trenton, New Jersey 08628

### ABSTRACT

Investigations to identify contaminated sites are often performed under intense public scrutiny. Local officials, property owners and the general public seek assessments in time frames much shorter than investigators may consider sufficient to fully understand a problem. The designation survey is a critical factor in the entire remedial action program. Overly conservative designation processes can lead to many more properties requiring detailed characterization and possible economic and legal consequences. On the other hand, a cursory designation process may lead to too few properties and a premature judgement concerning the existence and extent of contamination with possible economic and legal consequences. Remedial action and investigative agencies need to evaluate their designation survey procedures. As an illustration, several New Jersey sites will be reviewed in terms of the approach various federal and state agencies have taken in the initial identification and designation processes.

**UMTRA CONSENT FORM ACQUISITION:  
A SURVEY OF NONRESPONDENTS**

Laura L. Gonsalves, Dena Carpenter\*, Thomas B. Borak, Philip Kearney†  
Department of Radiology and Radiation Biology  
Department of Physics†  
Colorado State University  
Fort Collins, Colorado 80523

Oak Ridge National Laboratory\*  
P.O. Box 2567  
Grand Junction, Colorado 81502

**ABSTRACT**

The Radiological Survey Activities group of the Health and Safety Research Division at Oak Ridge National Laboratory is the Inclusion Survey Contractor (ISC) for the Uranium Mill Tailings Remedial Action (UMTRA) project in Grand Junction, Colorado. The ISC is responsible for performing any required radiological surveys and data analyses for the recommendation of inclusion or exclusion of designated properties in the UMTRA project. One of the responsibilities of the ISC is to obtain consent from the property owners to conduct radiological surveys. In Grand Junction, Colorado 30-40% of the owners of designated properties have not responded to the consent-for-access requests sent by certified mail. A questionnaire was designed to identify and study this nonresponse through personal interviews with 100 randomly selected nonrespondents. A profile of the population of nonrespondents, reasons for nonresponse, as well as suggestions to encourage response were identified and analyzed.

**INTRODUCTION**

On November 8, 1978 Congress passed Public Law 95-604, the Uranium Mill Tailings Radiation Control Act (UMTRCA) (1). This act required the Federal government to perform remedial action on 24 former uranium mill sites that were under contract with the Manhattan Engineering District (MED) and the U. S. Atomic Energy Commission (AEC) from the early 1940s through 1970, and on each site's vicinity properties. A vicinity property was identified as any property in the vicinity of the mill site which had been contaminated with uranium mill tailings originating from that site. Contamination of vicinity properties generally occurred due to windblown losses from uranium mill tailings piles as well as the removal and use of the tailings as a sand substitute in construction projects.

The responsibility for conducting the remedial actions at the 24 sites was delegated to the Department of Energy (DOE), Uranium Mill Tailings Remedial Action (UMTRA) Project Office. Assisting the DOE in this project is the Inclusion Survey Contractor, the Radiological Survey Activities (RSA) group of the Health and Safety Research Division (HASRD) at Oak Ridge National Laboratory (ORNL). The duties of the ISC are 1) the acquisition of consent from property owners to conduct radiological surveys; 2) the performance of an inclusion survey on the property; and 3) the reporting of results of the inclusion survey to DOE.

The acquisition of consent-for-access from property owners is often the most difficult and challenging responsibility of the ISC. Inclusion of a vicinity property in the UMTRA program is recommended by the ISC based on the results of a radiological survey. Without an agreement for access from the property owner, no radiological survey can be performed. The success or failure of consent acquisition depends on the local perception of risk from mill tailings and of the subsequent merit of the remedial action program.

This paper focuses on nonresponse to the consent-for-access requests. In Grand Junction 30-40% of the owners of designated properties (i.e. properties formally identified by DOE as potential candidates for remedial action) have not responded to the consent-for-access requests sent by certified mail. A profile of the population of nonrespondents, reasons for nonresponse, as well as suggestions to encourage response are identified and analyzed in order to address this problem of nonresponse.

## **BACKGROUND**

Grand Junction, CO is the location of one of the 24 uranium mill sites to be cleaned up under the UMTRA project. The Climax uranium mill in Grand Junction processed uranium ore for the Federal government from 1951 through 1970. Until 1966 when the Colorado Department of Health (CDH) stopped the mill from releasing tailings from the site, mill tailings were used extensively in construction and backfill on public and private property throughout Grand Junction (Mesa County). Approximately 7000 vicinity properties designated by the DOE for possible remedial action are located in Grand Junction.

Grand Junction has a long history of attention to its vicinity properties. As early as 1965, survey crews from several organizations - CDH, AEC and the Federal Public Health Service (now the Environmental Protection Agency (EPA)) - performed radiological surveys in order to determine the number of contaminated properties in the area. In 1972 the State of Colorado required radiological surveys and necessary remedial action to be performed before construction on a property would be approved. Also in 1972 Congress passed Public Law 92-314 which authorized the AEC to support a State of Colorado program, the Grand Junction Remedial Action Program (GJRAP), for the clean-up of vicinity properties in Mesa County. GJRAP has been limited to remedial action on tailings that affect radiation levels inside habitable structures, usually under or within 10 ft of the structure.

Using the information obtained from door-to-door surveys performed over this period, as well as mobile and aerial surveys, the DOE compiled a list of designated properties in Grand Junction. Unlike GJRAP, the UMTRA program is responsible for cleaning up all eligible vicinity properties, including open land. Under the UMTRA program, the ISC is required to survey these designated properties so that inclusion in the program can be determined. Data from previous GJRAP surveys cannot be preemptively used for this determination because the criteria for inclusion in UMTRAP are based on EPA standards not used in GJRAP, and radiological measurements of property greater than 10 ft from the habitable structure were not documented in the GJRAP surveys. Inclusion surveys of these designated properties by the ISC, therefore, are now in progress.

## METHODS

A questionnaire was designed to identify and study the problem of nonresponse to the consent-for-access requests sent to designated property owners by the ISC. The questionnaire consisted of two open-ended questions:

- 1) If you did receive the consent-for-access request, why did you choose not to respond?
- 2) What would have encouraged you to respond?

The questionnaire included a section requesting demographic information of age, length of residence, level of formal education and gross annual income.

The survey was conducted through unannounced person-to-person interviews with a randomly selected sample of nonrespondents. The 100 nonrespondents to be surveyed were selected in the following manner. Owners of 506 out of a total of 1553 designated properties did not respond to the consent-for-access requests sent by certified mail in ISC mailings from November 21, 1984 through June 10, 1985. From this list of nonrespondents, properties owned by businesses, multiple listings of properties owned by the same person and properties whose owners had not received the consent-for-access request due to change of address or owner were eliminated. This resulted in a list of 344 eligible nonrespondents from which 100 were randomly selected to be surveyed.

The data acquired from the survey were analyzed and the frequency distributions of age, length of residence, income, education and responses to each question were described. Responses which occurred more than 10 times were analyzed by chi square tests to determine any statistically significant association to age, length of residence, income and education.

## RESULTS

The nonrespondents were characterized according to age, length of residence, income and level of formal education from the information obtained. Figure 1 shows the distribution of nonrespondents according to age. It depicts an older population with 36% of those sampled in the 65+ years category. This is not surprising given that the majority of the designated properties are located in the older section of the city and many retired people live in this area. This is also reflected in the distribution of length of residence. Figure 2 shows that 86 out of 99 have lived in Grand Junction for more than 10 years. Figure 3 depicts the distribution of income of the property owners surveyed. Though balanced in the income brackets greater than \$10,000/yr, it also manifests the marked presence of a retired population living on a fixed income of social security and retirement benefits of less than \$10,000/yr. Figure 4 shows that the highest level of formal education completed by those who were surveyed is almost symmetrically distributed about the majority who completed high school and/or had some college education. This indicates that the nonrespondents are well-educated.

Those surveyed were asked why they had chosen not to respond to the consent-for-access request. The reasons given were noted and analyzed. The reasons are ordered according to their frequency and percentage of the total number of reasons in Table 1. Generally, more than one reason was offered by each interviewee. Hence, before any association of age, length of residence, income or educational level to the reason for nonresponse could be tested, it was necessary to determine whether the number of reasons given in each category was well-correlated to the number of interviewees in that same category. If

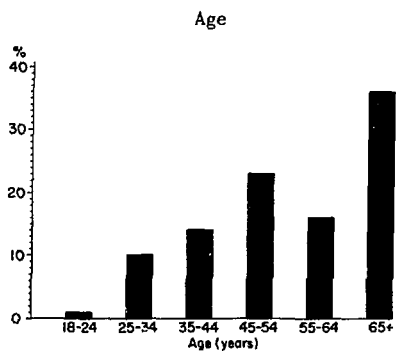


Figure 1: Age Distribution of 100 Persons Surveyed (100 responses).

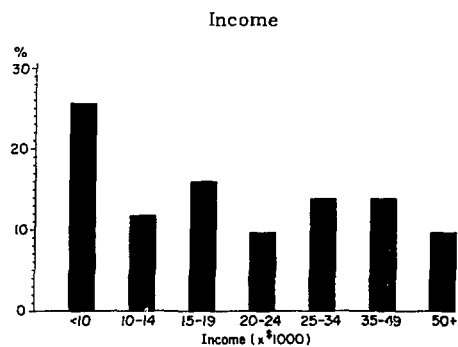


Figure 3: Income Distribution of 100 Persons Surveyed (94 responses).

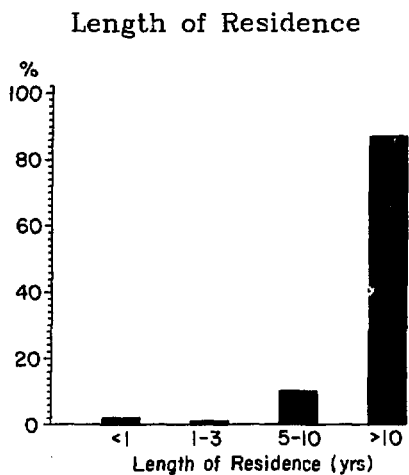


Figure 2: Distribution of Length of Residence of 100 Persons Surveyed (99 responses).

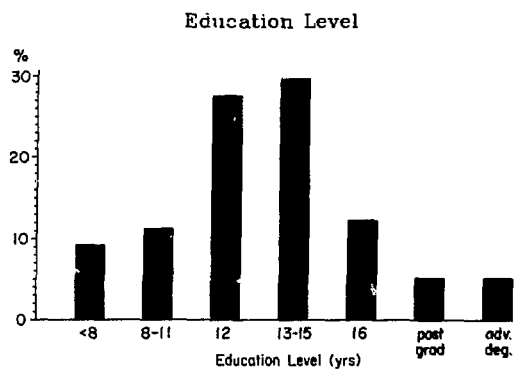


Figure 4: Distribution of Education Level of 100 Persons Surveyed (98 responses).

this were not the case, a bias would be introduced due to a disproportionate number of reasons per category analyzed. For example, the number of responses per age group should be correlated to the number of interviewees in each age group. Otherwise, interviewees of any one age group would have given too many or too few responses, thereby overweighting or underweighting their influence over the kinds of responses.

Table 1: Reasons for Nonresponse

Reasons	Frequency	Percent
Already surveyed under GJRAP or other program	60	32.8
Believe UMTRAP is a waste of taxpayer's money	25	13.7
Believe no harmful effects from tailings	24	13.1
Procrastination	16	8.7
Resent intrusion and disruption of my life	9	4.9
Apathy	8	4.4
Disruption in rental and/or sale of property	7	3.8
Assumed no tailings present due to time of construction	7	3.8
Forgot about it	7	3.8
Observed inadequacy in remedial action under GJRAP	5	2.7
Assumed already responded	4	2.2
Removed tailings from property personally	4	2.2
Still considering	3	1.6
Intimidated by officiality of consent form	2	1.1
Feared personal cost	1	0.5
Assumed not applicable-wrong address on form	1	0.5
<b>TOTAL</b>	<b>183</b>	<b>100</b>

Correlation coefficients for the number of reasons per category and the number of interviewees per category were derived for the variables of age ( $r=.996$ ), length of residence ( $r=1.0$ ), income ( $r=.988$ ) and education ( $r=.977$ ). These values show an excellent correlation between each set of data and therefore one can assume no bias due to multiple responses.

The most common responses to the question - "Why did you choose not to respond?" - were

- 1) Already surveyed under GJRAP or other program;
- 2) Believe UMTRAP is a waste of taxpayers' money;
- 3) Believe no harmful effects from tailings; and
- 4) Procrastination.

These reasons were then analyzed according to age, length of residence, income and education. Chi square tests were performed to determine any statistically significant association between reason given and the above variables. Since a chi square test requires an expected frequency greater or equal to one, categories within each variable were combined. Table 2 gives the probability

values for each chi square test. The chi square tests showed no statistically significant association at  $\alpha = 0.05$  between reason given and age, length of residence, income or education.

Table 2. The Most Common Reasons for Nonresponse are Categorized by Age, Length of Residence, Income and Education. The Actual and Expected Frequencies are Presented for Each Cell.

CATEGORY		Previous Survey	No Risk	Waste of Money	Procrastination	TOTAL	
AGE (years)	18-44	11 10.6	2 4.3	2 4.3	7 2.8	22	<div>CHI-SQUARE</div> <div>df Value Prob</div> <div>9 11.269 0.258</div>
	45-54	15 13.5	5 5.4	5 5.4	3 3.8	28	
	55-64	10 10.6	5 4.3	5 4.3	2 2.6	22	
	65+	24 25.2	12 10.1	12 10.1	4 8.7	52	
	TOTAL	60	24	24	16	124	
Length of Residence (years)	≤ 10	7 6.3	1 2.5	2 2.5	3 1.7	13	<div>CHI-SQUARE</div> <div>df Value Prob</div> <div>3 2.229 0.526</div>
	> 10	52 52.7	22 20.5	21 20.5	13 14.3	106	
	TOTAL	59	23	23	16	121	
Income (x \$1000)	< 10	11 13.1	6 4.4	7 4.6	2 3.7	26	<div>CHI-SQUARE</div> <div>df Value Prob</div> <div>9 4.626 0.866</div>
	10-19	19 16.7	6 6.2	6 6.9	6 5.2	37	
	20-34	12 12.6	4 4.2	4 4.6	5 3.5	25	
	35+	15 12.6	3 4.2	4 4.6	3 3.5	25	
	TOTAL	57	19	21	16	113	
Education (years)	0-11	6 10.1	6 4.1	5 4.1	2 2.7	21	<div>CHI-SQUARE</div> <div>df Value Prob</div> <div>9 6.361 0.703</div>
	12	16 17.3	7 7.0	9 7.0	4 4.7	36	
	13-15	21 17.7	5 7.2	4 7.2	7 4.8	37	
	≥ 16	14 13.9	6 5.7	6 5.7	3 3.8	29	
	TOTAL	59	24	24	16	123	

The nonrespondents surveyed were asked what would have encouraged them to respond to the consent-for-access form. The suggestions given are ordered

according to their frequency and percentage of the total number of suggestions in Table 3. More than one suggestion was generally offered by each interviewee. Again, it was necessary to determine if the number of suggestions per category was well-correlated to the number of interviewees per category in order to ensure no bias. Correlation coefficients for the number of suggestions per category and the number of interviewees per category were derived for the variables of age ( $r=.995$ ), length of residence ( $r=1.0$ ), income ( $r=.991$ ) and educational level ( $r=.999$ ). These values also show an excellent correlation between each set of data and therefore one can assume no bias due to multiple responses.

Table 3: What Would Have Encouraged Response

Suggestions	Frequency	Percent
If form had been delivered in person	60	37.3
If understood the difference between UMTRAP and GJRAP better	58	36.0
Nothing	14	8.7
Any follow-up	12	7.5
If resale of house helped by compliance	4	2.5
If agreement for access had been for shorter period	3	1.9
If required by law	3	1.9
If understood consent-for-access form and cover letter better	3	1.9
If rental property surveyed when unoccupied	2	1.2
If contacted by phone	1	0.6
If it were made clear that a response was necessary	1	0.6
TOTAL	161	100

The most common response to the question - "What would have encouraged you to respond?" - were

- 1) If the consent-for-access form had been delivered in person;
- 2) If understood the difference between UMTRAP and GJRAP better;
- 3) Nothing; and
- 4) Any follow-up.

Chi square tests were performed in order to determine any statistically significant association between the suggestion offered and age, length of residence, income or education. At  $\alpha = 0.05$ , no statistically significant association was found between the suggestions to encourage response and any of the above variables (Table 4).

## DISCUSSION

The results of this study may aid in the acquisition of consent-for-access forms in Grand Junction and in all 24 UMTRA sites. The UMTRA sites are by

definition communities which coexisted with uranium mills for several decades. Many members of these communities have worked in the uranium mills and are not convinced that the tailings are a potential health risk. It is perhaps understandable that 30-40% of the property owners have not responded positively to the UMTRA program in one such community, Grand Junction.

Although the chi square tests between the reason for nonresponse and age, as well as suggestions to encourage response and age indicated no overall

Table 4. The Most Common Suggestions to Encourage Response are Categorized by Age, Length of Residence, Income and Education. The Actual and Expected Frequencies are Presented for Each Cell.

	Better Understood	Personal Contact	Any Follow-Up	Nothing	TOTAL	
Age (years)	18-44	18 15.0	12 15.3	5 3.1	2 3.6	37
	45-54	14 14.2	13 14.4	5 2.9	3 3.4	35
	55-64	8 8.9	10 9.1	1 1.8	3 2.2	22
	65+	18 19.9	24 20.2	1 4.1	6 4.8	49
	TOTAL	58	59	12	14	143
Length of Residence (years)	≤ 10	9 7.3	6 7.4	2 1.5	1 1.8	18
	> 10	48 49.7	52 50.6	10 10.5	13 12.2	123
	TOTAL	57	58	12	14	141
Income (x \$1000)	< 10	12 14.7	16 14.4	1 3.1	6 2.8	35
	10-19	19 17.2	16 16.9	5 3.6	1 3.3	41
	20-34	14 12.2	8 11.9	5 2.6	2 2.3	29
	35+	12 13.0	18 12.8	1 2.7	2 2.5	31
	TOTAL	57	56	12	11	136
Education (years)	0-11	10 11.8	14 12.0	2 2.5	3 2.7	29
	12	18 17.6	18 17.9	3 3.6	4 3.9	43
	13-15	18 16.3	16 16.6	3 3.4	3 3.7	40
	≥ 16	12 12.3	11 12.5	4 2.5	3 2.7	30
	TOTAL	58	59	12	13	142

CHI-SQUARE		
df	Value	Prob
9	9.278	0.412

CHI-SQUARE		
df	Value	Prob
3	1.335	0.721

CHI-SQUARE		
df	Value	Prob
9	14.085	0.120

CHI-SQUARE		
df	Value	Prob
9	2.256	0.987

association, there are trends in the data that are of interest. Seventy-one percent of the property owners who responded with "Believe no harmful effects from tailings" and "Believe UMTRAP is a waste of taxpayers' money" were 55 years of age or older. When asked "What would have encouraged you to respond?", this older population generally either answered "If the form had been delivered in person" or "Nothing". Many of the property owners of this age group have been associated with the uranium industry. The property owners in the age group of 18-54, tended to give "Procrastination" as a reason for nonresponse. Eighty-three percent of those who answered that any form of follow-up would have encouraged them to respond were of this same age group. These are trends which could be characteristic in all of the 24 UMTRA communities, and should be considered in the design of any follow-up program or media campaign for consent acquisition.

The uniqueness of Grand Junction from other UMTRA sites should not be overlooked. This is apparent in that the most common reason for nonresponse was "Already surveyed under GJRAP or other program." Grand Junction has a long history of vicinity property surveys, particularly under GJRAP. Many of those interviewed had had several radiological surveys of their properties over the years. Most of these property owners received CDH official survey reports stating that "no indication of uranium mill tailings was found under or against the structure of your property" and that any tailings found elsewhere on the property did "not present any problem of radiation exposure to you." Their lack of enthusiasm to another survey is not surprising. Those interviewed were generally unaware of any difference between GJRAP and UMTRAP. Consequently, UMTRAP was viewed as another "waste of taxpayers' money."

This problem was anticipated by the ISC. A pamphlet explaining the differences between GJRAP and UMTRAP is included in the consent-for-access packet sent to the property owner. The included cover letter states "Even though your property may have been surveyed in the past under the Grand Junction Remedial Action program, another survey may be required to ensure that the radiation levels do not exceed the standards established by the EPA for UMTRA." In these cases either this information is not understood or is not read by the property owner.

This problem is best addressed by person-to-person contact with the property owner. Property owners commonly requested information regarding the difference between the two programs. These personal discussions were quite effective. As a result 80% of the property owners interviewed signed the consent-for-access form. The success of personal contact in explaining the difference between GJRAP and UMTRAP is indicated in the survey responses to "What would have encouraged you to respond?". Most of those who answered "If the form had been delivered in person", also answered "If understood the difference between GJRAP and UMTRAP better." These property owners were more receptive to arguments of the value of an UMTRA survey when approached through personal discussion.

## **CONCLUSION**

The survey results suggest several options in the design of a follow-up procedure to be implemented by the ISC.

- 1) A follow-up letter is recommended. Property owners who did not respond due to procrastination would most likely respond to a second mailing.

2) A person-to-person interview with the property owner, though more costly, is probably the most effective method for consent acquisition from the older nonrespondents. Personal communication with the property owner can often change the owner's negative impression of UMTRAP. Also, the confusion between GJRAP and UMTRAP is most effectively resolved by personal discussion.

3) Local media coverage of the UMTRA program would quite likely increase the response to the initial consent-for-access mailings. Conversations with nonrespondents indicate that any media campaign should focus on the benefits of UMTRAP to the local community. The majority of nonrespondents are longtime residents of Grand Junction and are concerned about the future of the city.

To gain the cooperation of the nonrespondents it is not sufficient to stress the possible health risks from uranium mill tailings. Emphasis should also be placed on the UMTRA program's ability to help eliminate Grand Junction's negative public image and bolster its faltering economy.

#### **ACKNOWLEDGMENT**

The research performed by Laura L. Gonsalves was under appointment to the Nuclear Engineering, Health Physics, and Radioactive Waste Management Fellowships program administered by Oak Ridge Associated Universities for the U. S. Department of Energy.

#### **REFERENCE**

1. U. S. Department of Energy. Vicinity Properties Management and Implementation Manual. Uranium Mill Tailings Remedial Action Project Office, Albuquerque, NM, 1984.

## HEALTH PHYSICS PROGRAM FOR THE EDMONT URANIUM MILL DECOMMISSIONING PROJECT

J. L. Polehn, R. G. Wallace, R. P. Reed  
Tennessee Valley Authority  
401 UBB-C  
Chattanooga, TN 37401  
And  
G. T. Wilson  
Silver King Mines, Inc.  
P.O. Box 49  
Edgemont, SD 57735

### ABSTRACT

The Tennessee Valley Authority (TVA) is actively involved in decommissioning a uranium mill located near the town of Edgemont, South Dakota. The Edgemont Mill Decommissioning Project, which is unique in many respects, will involve dismantlement of the old inactive mill building and excavation and transportation of several million tons of uranium mill tailings to a permanent disposal site. To ensure that workers are adequately protected from radiation exposure during decommissioning operations, a health physics program appropriate for the decommissioning situation was developed. The Edgemont Mill Decommissioning Project Health Physics Manual (HPM) gives the programmatic requirements for worker radiation protection. The requirements of the HPM are implemented by means of detailed onsite operating procedures. The Edgemont project health physics program was developed using currently available regulations and guidance for an operating uranium mill with appropriate modifications for decommissioning. This paper discusses the development, implementation, and documentation of that program.

### INTRODUCTION

The decommissioning of the Edgemont Uranium Mill Site is being conducted by Silver King Mines (SKM), under contract to TVA. In developing the health physics program for this project, TVA and SKM found there were few guides or regulations pertaining to the uranium mill decommissioning situation. Most of the guides and regulations available were for operating uranium mills or nuclear power plants. Therefore, it was necessary to adapt these guides and regulations to our purpose considering the potential hazards of the residual radioactive materials at the site. This was not always easy because, on occasion, it was necessary to readjust our perspective (i.e., we would slip into thought patterns appropriate for operating mills or nuclear power plants) as well as overcome a lack of information on the uranium mill decommissioning subject.

### DEVELOPMENT

Our initial approach to the Edgemont project was to develop generic health physics criteria for the site which served as a foundation for serial development of the health physics program. These criteria came to be known as the Radiological Safety Plan (RSP).

Topics determined to be of major significance for inclusion in the RSP are listed in Table 1 with the major source regulation or guide used in developing the requirements in the topics.

Table 1: RSP Topics (1)

<u>Topic</u>	<u>Source Guide</u>
1. Health Physics Organization/ . . . . . Qualifications/Inspections	U.S. NRC Regulatory Guide 8.31
2. Training . . . . .	10 CFR Parts 19 and 21; U.S. NRC Regulatory Guides 8.13 and 8.31
3. ALARA . . . . .	10 CFR Part 20
4. Direct Radiation Exposure . . . . .	10 CFR Part 20
5. Airborne Radioactivity . . . . .	10 CFR Part 20; U.S. NRC Regulatory Guides 8.15 and 8.30; NUREG-0041
6. Bioassay . . . . .	10 CFR Part 20; ICRP's 2, 10, and 30; U.S. NRC Regulatory Guides 8.11 and 8.22
7. Contamination Control and . . . . . Limits	U.S. NRC Regulatory Guide 8.30; NRC position paper, Guidelines for Decontamination of Facilities & Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-product, Source, or Special Nuclear Material, 11/76
8. Quality Control/Quality . . . . . Assurance	10 CFR Part 20; U.S. NRC Regulatory Guides 4.15, 8.15, and 8.25
9. Accidents	

Since the requirements of the RSP originated from regulations and guides for operating uranium mills and nuclear power plants, it was necessary to make adjustments to these regulations and guides for the special problems associated with uranium mill decommissioning. Special problems which required a significant amount of attention included: setting appropriate contamination controls limits, examining the potential need for respiratory protection, developing appropriate criteria for setting Maximum Permissible Concentrations (MPC's), developing a health physics training program appropriate for the decommissioning situation, and obtaining qualified health physics personnel at the remote Edgemont, South Dakota site. Basic health physics commonsense and an understanding of the physical processes involved in decommissioning activities, in concert with available regulations and guides, were used to address these special problems.

Since the RSP was a programmatic document, the detail in the document was insufficient to provide direction for day-to-day activities (i.e., procedures were needed). Thus, the RSP topics were expanded to form another more detailed document known as the Edgemont Mill Decommissioning HPM. The purpose of this document was to specify basic radiation protection standards and special TVA requirements applicable to the project as well as delineate responsibilities and onsite health physics implementing procedures necessary to carry out the program. The topics addressed in the HPM are listed in Table 2.

Table 2: HPM Topics (2)

1. Administrative Controls
2. Access Control
3. Radiation Exposure Control
4. Health Physics Instrumentation
5. Emergencies
6. Quality Assurance

This first version of the HPM basically was designed to replace the RSP (i.e., include the programmatic requirements of the RSP as well as the specifics of procedures). The plan of incorporating the two documents did not work because the integrated document became cumbersome and disorganized and at the same time did not contain the detail necessary to adequately perform certain tasks. Thus, the HPM was revised. It basically became a more detailed version of the RSP. The onsite implementing procedures were separated from the HPM into individual documents so the necessary detail could be incorporated. These documents now make up the contents of the onsite procedures manual.

In addition to the above-mentioned changes, responsibility for various parts of the program changed. SKM originally was given total responsibility for the HPM, subject to TVA review and approval. However, the review process proved to be cumbersome and time-consuming. TVA, therefore, assumed responsibility for the HPM (i.e., the programmatic portion of the program) with SKM retaining responsibility for development of the onsite procedures. To expedite development of the procedures, TVA provided draft procedures in TVA accepted format to SKM. SKM adapted these procedures to the Edgemont operations. TVA performed final review of the procedures as part of its oversight role.

#### Health Physics Program Content

The health physics program basically consists of: a training program, an access control program, and a radiation exposure control program. The training program topics for all employees are covered in Table 3.

Table 3: Training Program Topics (3)

1. Methods of Reducing Radiation Exposure
2. Risk versus Exposure
3. Prenatal Radiation Exposure
4. Work Limitations
5. Exposure Limits
6. Employee Courses of Action

In addition to this training, supervisors receive instruction on the health and radiation aspects of the specific jobs they will supervise. Also, an annual abbreviated retraining course is provided to all employees. The retraining course addresses: relevant information which has become available during the previous year, a review of safety problems that have occurred during the previous year, changes in regulations and license conditions, exposure trends, and other current topics. An exam is given after training and retraining. Test results are discussed with the employee.

The access control portion of the program basically consists of a fence around the site with controlled entrance and exit locations equipped with friskers. This control provides site security, ensures compliance with regulatory requirements, prevents the spread of contamination, and prevents unnecessary personnel exposure.

The radiation control portion of the program consists of posting, monitoring, and decontamination programs based on external exposure, MPC, and contamination limits developed for the project. Development of the external exposure limits was straight forward. These limits were taken from 10 CFR Part 20 requirements. In contrast, the MPC limits for the site required significant amounts of attention since there were no MPC limits specifically for uranium depleted materials and since inhalation of airborne materials is believed to be the critical pathway for the site. Basically, the methodology from 10 CFR Part 20 Appendix B Notes 1 and 4 were used to develop these limits. Contamination control limits also required significant amounts of attention because of the potential for ingestion of radioactive materials. The contamination control limits are based on U.S. NRC Regulatory Guide 8.30 and the NRC Position Paper cited earlier. These limits are set points for surface contamination on equipment and personnel leaving the site. These limits are also used in eating areas, change rooms, laboratories, and offices. In addition to the "equipment release" contamination limits, the site has contamination control limits to designate controlled areas within the fence (i.e., areas which require special precautions).

## IMPLEMENTATION

The Edgemont health physics program is implemented through the HPM and onsite operating procedures by the SKM resident manager. The SKM Radiation Safety Officer (RSO) has direct responsibility for the program (i.e., day-to-day supervision of the Health Physics Staff and overall implementation of the radiation protection program). Site employees and visitors are responsible for complying with all rules, notices, and operating procedures as well as promptly reporting to management any safety problems they identify.

TVA plays an oversight role in the implementation of the health physics program. TVA's oversight role includes periodic unannounced inspections and an annual evaluation of the radiological protection program. In addition, TVA provides technical assistance to the site as requested.

The Edgemont onsite operating procedures were developed to ensure safe day-to-day decommissioning operations. The procedures, for the most part, have been designed so the detail is sufficient for knowledgeable personnel to perform the tasks without supervision. The procedures address the topics listed in Table 4:

Table 4: Procedure Topics (3)

1. HP Procedures and Document Control
2. HP Records
3. Instrument Calibration & Daily Performance Checks
4. Laboratory Counting Procedures
5. Access Control
6. Occupational Radiation Exposure Control
7. Radiation Protection Training
8. Dust Control
9. Respiratory Protection
10. Environmental Radiation Monitoring Sample Collection
11. Total Suspended Particulate Sampling
12. Nonconformance
13. Radiological Airborne Sampling and Calibration

These procedures can be supplemented as needed. Allowance for emergency changes has also been included.

#### DOCUMENTATION

The Edgemont health physics program is documented in the form of the HPM and procedures. Issuance of the HPM is controlled by TVA while the onsite procedures are controlled by SKM (i.e., with TVA oversight).

Forms for surveys, dose calculations, calibration checks, etc., are included in the procedures. In addition, inspections and evaluations are documented and reported to management (i.e., SKM, and/or TVA as appropriate).

#### CONCLUSIONS

Development of the health physics program for the Edgemont site required a cautious attitude as well as numerous hours of research. An important consideration in the development of the program was to balance health effects savings with the cost of these savings. Basic commonsense in combination with good health physics practices and an understanding of decommissioning processes have led to what we believe to be a good program.

Development of the Edgemont health physics program has never quite ended (i.e., it is an ongoing project). This occurs as a result of new information being brought to light or as apparent deficiencies are noted. As this occurs, the program area affected is reevaluated and changed if it is determined that the new information or deficiency is significant enough for incorporation or correction, respectively.

Lastly, development of the health physics program for the Edgemont site required a strong commitment from various levels of management as well as project employees. This commitment to the safe decommissioning of the site is clearly reflected in the following statement in the HPM: "Edgemont mill decommissioning project management and employees will take all reasonable precautions in the performance of the decommissioning project to protect the health and ensure the safety of employees and the public." (2)

REFERENCES:

1. Edgemont Mill Decommissioning Radiological Safety Plan, Tennessee Valley Authority, September 1983
2. Edgemont Mill Decommissioning Health Physics Manual, Revision 3, Tennessee Valley Authority, July 10, 1985
3. Edgemont Mill Decommissioning on Site Operating Procedures, Silver King Mines, Inc., May 3, 1985

# **ESTABLISHING A COMPREHENSIVE COST-EFFECTIVE RADIOLOGICAL PROTECTION PROGRAM FOR DECOMMISSIONING PROJECT**

M.A. Maan, R.P. Nair and S. Schafer  
Atomic Energy of Canada Limited  
CANDU Operations

## **1. INTRODUCTION**

Atomic Energy of Canada Limited (AECL), as virtually every other organization involved in the nuclear field, is committed to the implementation of radiological safety measures necessary to protect health and safety of its employees and the general public. AECL has established itself as the Canadian centre for pure and applied research in the nuclear and related fields, has evolved and developed commercially the CANDU nuclear power system, use of isotopes, and irradiation devices, and has provided the stimuli for development of a large secondary industry capable of producing equipment or services to meet the nuclear market needs.

With over 30 years of experience in designing and operating nuclear reactors, AECL has undertaken a new challenge, that of decommissioning a prototype nuclear power plant. The conversion of Gentilly-1 from an operational plant to a waste storage facility involved a transfer of irradiated fuel from wet to dry (canister) storage, and an extensive decontamination effort intended to ensure safe passive storage conditions, requiring only minimal surveillance. The program has been severely constrained by the imposition of a two-year schedule and a ceiling on expenditures for completion of the task.

The constraints so defined made the requirement for cost-effectiveness mandatory. Yet, because of the commitment to excellence, the radiological protection program was to run its course without any compromises on safety standards, and with keeping the latest ICRP recommendations, on ALARA\* risk control, in mind.

The logic in establishing the program, radiation protection organization, responsibilities and staffing is described in Section 2.

In Section 3, the personnel selection, training and qualification program is described.

The dose control and ALARA program is given in Section 4, while in Section 5, the health physics facilities and equipment are described.

Section 6 covers the experience with field execution of the program, a description of the development of procedures, quality assurance, and health physics interaction with the project.

Section 7 deals with the organization of the project and the subject of management oversight analytic trees.

## **2. RADIOLOGICAL PROTECTION PROGRAM ORGANIZATION**

### **2.1 General**

The Radiological Protection Program defines provisions required in the planning, execution, and the follow-up stages of decommissioning, to ensure compliance with the ALARA dose limitation principle. The adopted radiation protection decommissioning objectives are deemed to be consistent with the recommendations of the International Commission on Radiological Protection, Publication No. 26, 30, and 37; and with the Basic Safety Standards for Radiation Protection, prepared under the joint auspices of the IAEA, WHO, ILO, and NEA (see References 1, 2, 3, and 4).

### **2.2 Program**

At the outset, it was recognized that the Radiation Protection Program must have full support from management and in keeping with this precept, a role and a budget to meet the needs of the program were

\*ALARA . . . As Low As Reasonably Achievable, social and economic constraints being taken into account.

defined and approved by management. This step was followed by the definition of short-term and long-term goals for the Radiation Protection Program, consistent with other related goals such as the total budget and schedule for the project, regulatory and company policies, and resources available for the task.

The following goals were defined, stated in measurable terms.

#### **2.2.1 Short-Term**

- Appointment of RP/HP Manager and Staff.
- Resources supplied to develop the team.
- Collaboration with external consultants to streamline the operation and establish general validity of approach.
- Define the long-term radiation protection objectives and the method of applying ALARA.
- Define Health Physics Principles and Radiation Protection Standards for the project, and present to regulatory authorities for approval.
- Carry out in situ the characterization radiation survey.
- Develop methodology for radiation exposure estimates and reviews.
- Define training and qualification program.
- Develop radiation protection and health physics procedures, including an emergency plan.
- Identify, specify, and procure instrumentation and protective equipment (PE).
- Place the instruments and PE in service.
- Train and qualify necessary manpower.
- Modify facilities and establish a radiological zoning plan.
- Establish dosimetric system acceptable to the regulatory authorities.
- Establish RP and HP record keeping system, develop forms, routines, contractual arrangements, and data processing system (including software).
- Establish checks and balances to satisfy QA requirements.

#### **2.2.2 Long-Term**

- Excellence, to be measured in terms of safety quality, corporate "citizenship" and public regard.
- Cost Effectiveness, to do more for less, avoid unnecessary waste of resources, streamline the decommissioning process by avoiding trivial problems.
- Competitiveness, to ensure that the know-how acquired by "doing it" is on a level which is marketable on its own merit of contribution to prospective clients' projects.
- Profitability, to set a stage for developing the expertise in decommissioning health physics into a self supporting business activity.

With respect to the long-term goals, it should be kept in mind that AECL possesses an excellent access to expertise in all aspects of radiation management such as safety engineering, risk assessment, reliability studies, pathways analyses, etc., available within the Corporation.

### **2.3 ORGANIZATION**

#### **2.3.1 Reporting and Responsibilities**

The responsibility for radiological protection, at the execution level, is shared by the Health and Safety Group (HSG) and the Project. The unimpeded function of the HSG is assured by an independent reporting route direct to senior management. (See Figure 1.)

The Health and Safety Group is responsible for establishing the regulations and policies, while the Project ensures the regulations are implemented, and decommissioning effort proceeds in compliance with these.

The responsibilities were clearly defined in a document which formed a part of the license and was approved by the regulatory board (reference 5).

#### **2.3.2 Staffing**

To handle a peaking manpower of about 130 persons at site, the personnel related to the HSG comprised 3 professionals, one technician, and one administrative assistant.

The Radiation Protection personnel (attached to project) included 1 supervisor, 2 experts in radiation protection, and 9 field technicians trained in the RP and surveying techniques (see also Section 3).

The position of Head, Health and Safety Group at site, encompasses also the function of a Senior Site Health Physicist, and was approved by the Canadian Regulatory Authority within conditions of the license (the incumbent possesses many years of experience and has a doctoral degree in the field).

The qualifications are briefly discussed in Section 3.

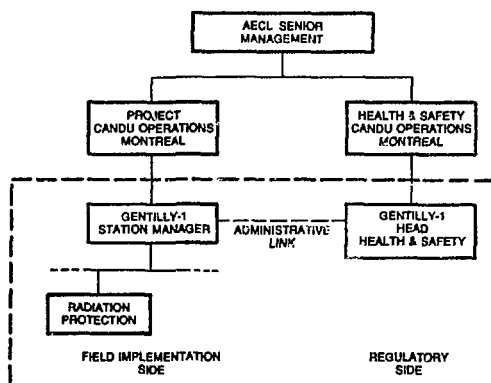


FIGURE 1 GENTILLY-1 SITE RADIOLOGICAL PROTECTION ORGANIZATION AND ADMINISTRATION

### 3. PERSONNEL QUALIFICATIONS AND TRAINING

The project implementation schedule called for a skeleton crew of the operators needed to wind down the operation and execute an orderly transition from operational to static conditions. At the same time, there was a steady influx to site of the contract personnel, needed to carry out the deconstruction activities.

Normally, in the context of Canadian nuclear industries, the policy of licensees is to rely on the so-called "autoprotection", practice, i.e. to train all the personnel to the level necessary for them to be allowed on their own responsibility into the controlled areas of the plant. In the light of the manpower requirements (steady growth in a relatively short span of time), and the time needed to obtain sufficient confidence in allowing personnel to "protect themselves", the autoprotection scheme had to be modified for Gentilly-1.

The most effective solution was found in modifying the training requirements to meet actual needs: For example, the tradesmen most in demand (operators, mechanics, electricians) were subjected to a formal training of a 2-weeks duration, followed by the examinations and on-the-job training for an additional period of time.

The experience has shown that majority of the personnel sent through the training process could be qualified at least to the level allowing them to look after themselves ("yellow" badge). A small number of persons were disqualified, but allowed to remain in the workforce on tasks involving essentially non-radioactive work ("orange" badge).

Among the workers allowed to plan their own work (with assistance from RP/HP specialists), it became soon apparent that there was a core group of responsible individuals with quick grasp of issues. These were sought, identified and qualified one step higher (i.e. "green" badge) to allow creation of a task force of persons who could be trusted to protect other, less experienced personnel. There was no formal training beyond the level "yellow"; however, the selection and appointment into "green" status represented a rigorous selection process, involving supervisors, peers and an approval of the appointment by the site senior health physicist, on a personal assessment basis. In all cases, these "green" badges had substantial RP experience from other nuclear sites.

The classroom and field training program was formulated to cater to specific site conditions, and was supplemented by the latest teaching methods (videotapes, actual instrumentation and "live" assignments in the field). An important part of the record keeping process was a clause in the qualification chain which involved each person's signature attesting to the fact that he/she actually received the training and understood the resultant responsibility.

The training instructors qualifications and unbiased assessment was ensured by subcontracting part of the training to a third party (utility — Hydro-Quebec; university — Ecole Polytechnique in Montreal).

The course content encompassed the theory and the practices of radiation protection, site specific hazards and instrumentation, site specific procedures (rad-safety and industrial), reporting chain, qualifica-

tions, emergency plans and rescue, etc. The contents and examinations were formally approved by the Canadian Regulatory authorities.

With the creation of a "green-badge" reserve, the increased number of less experienced persons could be utilized on tasks involving decommissioning of active systems. The "green" badge holders have acted as sponsors to the others, and in this way the training program was not as extensive (and costly), and the selection of personnel was not as rigorous as would have been necessary if full "autoprotection" policy had been adopted.

During the period of peaking manpower demand, there were 62 qualified orange badge holders, 43 yellow badges and 22 qualified green badge holders.

## **4. DOSE CONTROL AND ALARA IMPLEMENTATION**

### **4.1 Dose Limitation System**

Provisions are required to comply with the objectives that:

- a) Unnecessary exposure shall be avoided.
- b) The doses from any necessary exposures shall not exceed the prescribed limits.
- c) The necessary, planned exposures must be kept as low as is reasonably achievable taking into account the economic and social factors (see Section 4.5 for interpretation).

By the time the task is to be executed in the field, the engineering preparation of the decommissioning task already included ALARA engineering effort. The decommissioning and verification measures were defined under the overall quality assurance requirements of the project.

Based on experience with operational nuclear facilities, it was found advantageous to establish an individual whole body operational dose equivalent limit; in the G-1 case it was\* 40 mSv/person in any one year period. (Canadian utilities generally establish an administrative operational dose limit at 30 mSv/a-person; this was considered unnecessary in case of decommissioning where the effort does not last over the lifetime of an individual). In any case, the ALARA effort was expected to control the individual doses well below this limit. In fact, the operational dose control put into effect at site was devised to ensure that no individual atomic radiation worker would receive doses significantly higher than the average.

\*The regulatory limit equals 50 mSv/a, for an atomic radiation worker (ARW).

The limits for non-qualified staff, female workers, etc. were similarly set below the respective regulatory limits.

No limits were assumed for collective radiation exposure. Instead, tasks with a risk of significant dose exposure were identified from radiation exposure estimates. The detailed dose estimate obtained for the task of irradiated fuel transfer from wet to dry storage, for example became a sort of collective dose target against which the dose trend monitoring program could be applied. During the task execution, the health physicist's office monitored the situation closely. As confidently expected, the "actuals" were lower than predicted.

### **4.2 Monitored Performance Parameters**

In monitoring the R/P performance, HSG identified many parameters on which to judge the result of RP program. These included: collective radiation exposure; maximum dose to an individual; average individual doses (by work group); accumulated station collective dose; dose trending by the period; general radiation fields in monitored areas; gaseous, liquid and solid effluent releases; etc.

The performance was periodically assessed, and the results published in the form of quarterly reports. Preparation of these reports was greatly facilitated by data processing done routinely on IBM-PC's and software created for that purpose.

### **4.3 Dosimetry Program**

#### **4.3.1 External Exposures**

The monitoring of whole-body gamma and skin beta-gamma doses was accomplished by using the system of thermoluminescent dosimeters (T.L.D.). Two-week dosimetry period was chosen to coincide with the reading periods of a subcontractor (Chalk River Nuclear Laboratories and Hydro-Quebec). The raw printouts received from the subcontractors, listing the persons vital identification data and the dose read in the period, were compared with other data received from the field. In many cases, the self reading of the "direct reading dosimeters" (DRD) resulted in recording minute doses higher than that reported by the TLD's. In keeping with the principle of conservative assessment, the station senior site health physicist has authorized a procedure by which the dose assessment, and final dose accounting for each individual in-

cluded data from the DRD records, suitably recalculated for the type of exposure ("rad" to "rem" conversion). In addition to this, an individual's dosimetric file contains an authenticated copy of all work permits where the individual's name is entered. In this way, the conditions of work performed, the ambient fields, the dose-rates in the work location, air contamination data and time spent in the fields could all be reconstructed.

One benefit from this is immediate: each worker has to participate in the dose assessment process, and the dose accumulation could be checked against the official records posted on visible plant locations. The workers can check the data against their own "memory", and there is a direct rapport between what they feel should be in their dossiers, and what they actually see.

#### **4.3.2 Internal Exposures**

An extraordinary effort is made to ensure that internal exposures are kept at zero. During decommissioning, the obnoxious short-lived radionuclides have usually decayed, and by keeping with the procedures in force, it is possible, indeed imperative that internal dose uptakes are eliminated by engineering methods.

To ensure that there is no uncontrolled uptake, all employees entering the site, are subjected to a whole body count. The same applies for all terminations; one final WB count is taken before an employee is discharged. Furthermore, the work groups are periodically scheduled to undergo a quick screening, using the sensitive portal monitor (Gamma-I2, NNC) calibrated to detect less than 1% of ALI on site specific radionuclides.

The additional biosurveillance includes broad search counts of urine samples (periodic for critical work groups), a program of air monitoring (continuous alarming function) in work locations, and keeping all areas free of any loose contamination. The critical areas (e.g. lunchrooms) are surveyed on a daily basis (including some areas checked even more frequently). The surveys are performed by a group of specially trained and qualified surveyors, and current survey results are circulated to the HSG on a daily basis. The surveys are clearly defined, to include swipe checks, alpha, radon monitoring, etc. to conform with the up-to-date standards. The system is in place to record, review and assess any incidents. The records of all WBC readings, the urine analyses and all other related records are permanently stored to allow possible exposure investigations.

#### **4.4 Exposure Limitations - Work Techniques**

With a target of zero internal exposure, which translates into zero uptake, the emphasis is on providing suitable engineered solutions to activities which have a potential for airborne contamination. The use of local barriers (plastic tents, exhausts, etc.) is mandatory for work that may create airborne contamination and is considered as part of the ALARA procedures. In addition, use of the personal protective clothing and respiratory equipment is usually prescribed whenever there is uncertainty as to possible airborne hazard. These techniques are well known and there is no need to describe them here in any more detail.

#### **4.5 ALARA Implementation**

AECL adopted an ALARA optimization program which is based on the so called "aggregative" method, in which the available option is assessed, by taking into account the dose saving estimate, and the cost of option. The assessment is done against a single criterion, in this case a ceiling on the cost of the dose reduction measure.

Management controls the key variable, by pre-approving the input parameters (for example the cost of manpower replacement). Thus, the decision making can be done on a working level (by the engineer), and the method could be used project-wide without sacrificing the consistency of decisions.

The optimization is applied during the preparation of detailed decommissioning plans (engineering stage).

##### **4.5.1 Preamble**

A preliminary man-rem audit is carried out, to determine the expected collective exposure (a sum of collective exposures from individual decommissioning tasks). This usually allows an estimate of an individual dose in the critical group, prior to optimization. The magnitude of individual dose determines, in turn, the variable cost ceiling per unit of exposure saved, as will be shown below; see also references 6 and 7.

In case of Gentilly-1, with its low radiation fields, the projected individual doses were so low that automatically, cost allowance has fallen into the very low range (\$200 per rem saved).

#### 4.5.2 Determining the Available Options

The user asks a question: "Is there any alternative available; a modification; a change in equipment; anything?"

If the answer is "No", the system is ALARA, and no further action is needed.

If the answer is "Yes", the optimization effort continued to the next step.

#### 4.5.3 Cost-Effectiveness Test

Should the option be such that both, the dose and the cost is reduced, the change should be implemented. In case there is more than one alternative available, the one offering the best yield is selected.

In case the decision has to be made whether the increased cost is justified, the user must determine:

- the total cost of the option (\$)
- the total collective dose saved (man-rem)
- the individual average exposure in the critical group (rem/person-period)
- the maximum allowable cost of exposure avoidance (\$/1 man-rem avoided)

It is the latter which has to be determined from the management pre-approved input parameters. In practice, this is done by consulting a suitably prepared nomogram showing the ceiling variable cost as a function of individual dose. The closer the dose to the individual dose limit, the higher the cost allowed, and vice versa. The decision logic is given in Figure 2.

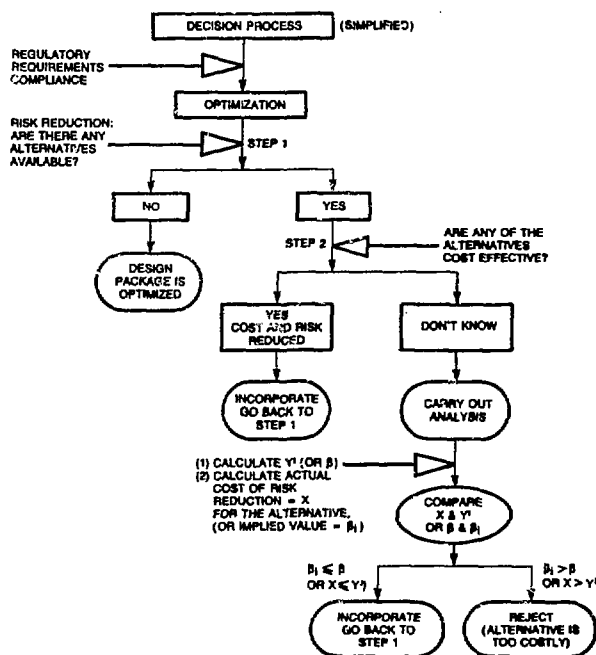


FIGURE 2 DECISION PROCESS

The cost of the option is compared with the "ceiling", calculated as the product of (\$/man-rem avoided x collective dose saved). If the cost is much higher than the ceiling, the idea is obviously not good, and probably it is a waste of money.

#### 4.5.4 Experience with ALARA at Gentilly-1

To a cursory reader, the aforementioned method may not be very attractive because it is reminiscent of weary "academic" attempts to control the process at the decision level where detailed analyses are simply not practical.

However, it deserves, in our opinion, another look. The advantage of this costing method has been immediately realized in Gentilly-1 simply because the risks, and the attendant doses were so low. The resulting restriction on cost of further dose reduction was so severe that after several attempts to quantify

the expenditures for dose reduction, further effort was abandoned! The reason was: by consistent application of the costing formula, the allowed expenditures were in the order of magnitude of "tens" to "hundreds" of dollars total. The theoretical assessment involved professional's time costing essentially more money than the money justifiably spent. Thus, it was clear to everyone involved, and the decision was properly documented, that unless the situation in the field deviated significantly upwards from projected, there was absolutely no justification for further ALARA effort.

## **5. HEALTH PHYSICS FACILITIES AND EQUIPMENT**

The provision of adequate instrumentation was considered essential for the performance of RP plan. During peak demand, the instruments at G-1 site included: 4 low volume air samplers; 4 high volume air samplers; 2 continuous (beta) air monitors with alarming function, 8 Beta/Gamma portable rad-monitors (0-5 R/h range); 2 portable RM (range 0-5000 microR/h); 2 Teletectors, i.e. extendable underwater probes (0-100 R/h); 8 portable contamination meters with supply of probes (0-500,000 CPM); 14 semi-portable CM with probes (0-500,000 CPM) used as interzonal friskers; 12 ratemeters (six ranges up to 100,000 CPM); 2 mini-scalars; 4 gamma monitors with alarming function (1-500 mR/h); 70 Direct Reading Dosimeters; 1 Portable Alpha Monitor with probes; 2 portal monitors; 8 hand monitors; 1 Waste Curie Monitor; 1 Multichannel Analyzer; and a number of auxiliary equipment, such as pancake probes, floor monitor probes; radiation alert monitors (0-50 mR/h) for personal use; and Digital Alarming Dosimeters (SuperDAD).

An adequate supply of sources to check and calibrate the instruments had also been secured. The calibration routines and evidence of verification was procedurally established and kept under close surveillance. The periodic recalculation of source decay was also performed, and results posted with the sources. An independent recalibration by a contractor using recognized standards was implemented on a bi-annual basis. Specific procedures were developed and implemented with respect to defective instruments or probes.

An emergency kit equipment was defined and located in the assembly areas. The periodic checking of all emergency equipment was also procedurally established.

Besides instrumentation, the original purchases covered the consumables (gloves, disposable clothing, rubbers, plastic, tapes, marking signs, ropes, garbage bags, etc.), which were controlled on a "minimum" level in the stores.

The experience at G-1 indicates that it is cheaper to marginally over-supply site with instruments and protective equipment, with resulting continuous progress in the work area, than to allow contractors or workers to stay idle on account of shortage of suitable instruments. The lesson has been learned early enough when the instruments originally delivered to site were found defective. The temporary shortage, before the replacements were obtained, proved to be a blessing in disguise.

The question may be posed on how "cost-effective" it is to spend a sizeable investment on a purchase of instruments, especially when the decommissioning activity was curtailed to two years only. The answer is, even during such a short time, the savings on manpower time alone justified extra instruments. Furthermore, the existing equipment was given a very good care and should continue to hold its useful value beyond the two year project lifecycle.

## **6. FIELD EXPERIENCE WITH EXECUTION OF THE PROGRAM**

The work at Gentilly-1 could be typically divided into three main activities:

- transfer of the fuel into dry storage
- decontamination and dismantling of active components
- demolition and removal of clean (or cleaned) material

Organization of the RP coverage, under HSG surveillance followed closely the three main activities.

### **6.1 Fuel Transfer RP Coverage**

The crew carrying out this part of the decommissioning program was dedicated to the task. All the workers were fully trained and qualified to self protection status "Yellow". The supervisor was qualified "Green", and an experienced radiation protection technician was assigned a full time function of continuous support to the group. This included planning and execution of daily surveys, instrument checks, monitoring of instruments "entrusted" with alarming function, and monitoring personnel dose uptakes. The data was daily reviewed and recorded by the HP group. There was no significant deviation from planned exposures and the work proceeded smoothly.

It should be recalled that the engineering package for design and manufacturing of remote tools, flasks, TV cameras, shielding stations, transfer vehicles and pathways, as well as the repository design

were all done with full HP participation, and assessed to be ALARA. The radiation exposure review estimates (collective dose) were adopted as a field "target", and the dose trend monitored against the "target" so established (see also 4.1 and 4.2).

The work has been done on the basis of a required Radiation Work Permit, which was periodically renewed to maintain its validity (the contents rarely changed, except for the names of workers). The copies of the permits are kept in the permanent records of the HS Group.

## **6.2 Decontamination and Dismantling**

The task of decontamination was handled under the specialist's direction, and carried out essentially by the dedicated group of decontamination technicians. According to needs, this group was occasionally supplemented by contract workers, in accordance with the overall progress schedule.

As a rule, all the workers in the core group were fully trained in radiation protection, and again, qualified at least "Yellow", with a number of "Green" badge holders among the workers.

The decontamination group cooperated closely with the group of "operators", whenever their work involved isolation of active (ie. functioning) systems which were in an orderly manner being taken out of service.

The cleanup was performed either *in situ*, taking all necessary precautions specified in the work permits (RWP), or, the items were relocated to a decontamination centre where the cleaning could be accomplished with better results due to specialized equipment available there.

The waste generated was categorized and inventorized, using the Waste Curie Monitor or in accordance with authorized procedure. Active waste was destined to remain on site, while clear trash or equipment was under close HSG scrutiny released for disposal elsewhere.

The planning of progress & liaison between groups was scrutinized, and the exceptions (eg. a contaminated spot left behind) were well documented to ensure "hold-up" points in the contractor's progress of work. The surveys before the work begun, during the work and after clean-up, were performed by a dedicated crew of specially trained RP surveyors, using approved techniques, instruments and documentation flow. All of the pertinent records shall be kept in permanent storage.

## **6.3 Demolition and Removal of Clean Material**

This represented a major activity, and with a few exceptions, it was handled by an outside contractor.

Contractor's staff was given a very basic training up to "Orange" status, and records are kept with respect to whole body counts prior to employment and after the termination. The staff is also being issued TLD badges but it was not necessary to qualify them as ARWs simply because obtaining significant dose would be very unlikely.

## **6.4 Integration of Field Work with RP Program**

It is unfortunate that the pressures being put on nuclear industry from outside have caused many RP managers to hear the word "ALARA" and cringe. This is a pity because it was precisely the ALARA argument which has allowed some flexibility in decision making, and probably saved a lot of money to an otherwise strained budget.

A case in point may be the decision to place the change rooms (makeshift facilities) directly in the field, thus allowing only workers in the most active zones to wear radiation clothing (colour coded white). The majority of the workers could be allowed to wear the work clothes (dark), and use protective lab coats (dark) when necessary, all day, including wearing them in the office area and cafeteria. There was no need to change the clothes when taking coffee breaks or lunch. This was allowed on the basis of rigorous survey (twice daily) and monitoring facilities set up prior to entering the unzoned cafeteria. Only a very small number of dedicated task force workers had to undergo periodic clothes change but that was integrated into their optimized (12 hours) shift routines.

Another example was the decision to allow lunching on the roof of the Reactor Building. The workers could "protect" themselves because of the training received, there were surveys available, etc. However, the main reason was simply realization that the "risk" of exposure (nil resulted) could be traded against risk of injury resulting from frequent climbing up or down from the top of the R/B.

Yet another (subject to paper presented as well, reference 8) example is the decision to use a portal monitor for "go-no-go" screening of personnel who may conceivably receive an internal contamination. Instead of time consuming body counts, suitably spread-out schedule calls the workers for fast check, usually not more than 5 minutes in duration. This allowed much higher frequency of examination, and afforded the HSG a real control over the potential of inadvertent internal contamination uptakes.

There could be many other examples: use of Waste Curie Counter to speed up sorting of the bagged waste and increase confidence in release of clean trash to public domain; procurement of many inexpensive and dependable instruments ("Monitor 4") at a fraction of the cost of similar instruments (with full measuring capabilities) for field use; modifying calibration schedules to actual use of instruments; etc. The bottom line was - since we have ALARA, and our commitment to ALARA is sincere, let us use it in the proper sense of the word "reasonable". If necessary, use ALARA argument in the reverse direction, ie. justify why it is not necessary to spend money.

The involvement of HSG in the preparation of detailed decommissioning work plans has been procedurally secured, and has resulted in uncounted savings because of the health physics non-interference in the project's undertakings (on account of its being not justified, ie. ALARA).

## **7. MORT, PROCEDURES, AUDITS & QUALITY ASSURANCE**

The Health and Safety function in a two-tier structure was independent of the Station/Project manager, thus allowing the desirable level of objectivity in assessing the compliance with health and safety guidelines.

The RP group, on the other hand, works in conjunction with the project in devising procedures to be followed, appropriate to the specific tasks. RP generates and promulgates permits, surveys and establishes work methods in compliance with the accepted standards. The procedures are only operative when countersigned and issued as approved by the Head, Health and Safety Group (Health Physicist).

Before any activity can be initiated, a "radiological work permit" and/or "dangerous work permit" is required, as appropriate. Each permit lists the activities to be performed and the personnel involved, protective measures to be enforced, and other pertinent data. Most often reference is made to the survey report prepared by the RP immediately prior to job start up. In case specified criteria were exceeded, the work permit had to be cleared up by the HSG. The log lists the identities of the individuals covered by the permit, and records of previous and cumulative exposure levels for each involved worker.

The station was divided into zones, the highest being Zone 3 (also "rubber areas") requiring special work methods and access. All work areas involving active equipment were classed as Zone 3 as soon as any decontamination activity commenced.

The work, the supporting routines, access to and from active areas, etc. all are covered by a set of approved "Radiation Protection Procedures". Activities involving qualifications, training, dosimetry, etc. are covered by a separate set of procedures named "Health Physics Procedures".

In defining the interactions, it was found advantageous to process systematically through all elements of a program by use of analytical trees. An example is given in Figure 3: Management Oversight Risk Tree (MORT). The analytic trees start with a single desirable activity (eg. optimization, No. 5) and systematically proceed through lower levels until all important factors are identified and their position specifically understood, in the overall system.

Each branch has some degree of interaction with the others. Where such interface could be clearly identified, transfer functions are inserted (triangles with arrows and letters).

The use of analytical trees helps in detecting and correcting an oversight or omission. Also, carefully structured trees are certain to avoid duplication of effort in the interface areas.

The end result is that areas of responsibilities could be clearly defined, and assessment of performance, especially from the QA point of view, can provide greater assurance that a given area is, in fact, adequate or inadequate.

An informed evaluation of management's oversight is considered critical in reviewing the adequacy of any of the elements of the radiation protection program (reference 9).

The success of a program can be often attributed to "management being in control" provided it is recognized that the "control" is only one of possible functions which, if performed inadequately, can result in deficiencies.

The other element, essential to success, is to have a fully functional, "non-defective" program, free of causes for problems. A well prepared logic behind the MORT analytical tool has gone a long way in developing a hitch-free RP Program for the decommissioning of Gentilly-1.

The internal evaluation of the performance at G-1 site was performed in conjunction with the overall Q.A. program. While the QA program concentrated on matters involving compliance with the procedures, and documentation for RP, Zoning, Work Permits, Personnel Qualifications and Training, Decontamination and Inspection Procedures, etc., the QA audits were complemented by regular audits in the field, carried out by the HSG. The basic difference was that of a content: the QA audit concentrates on

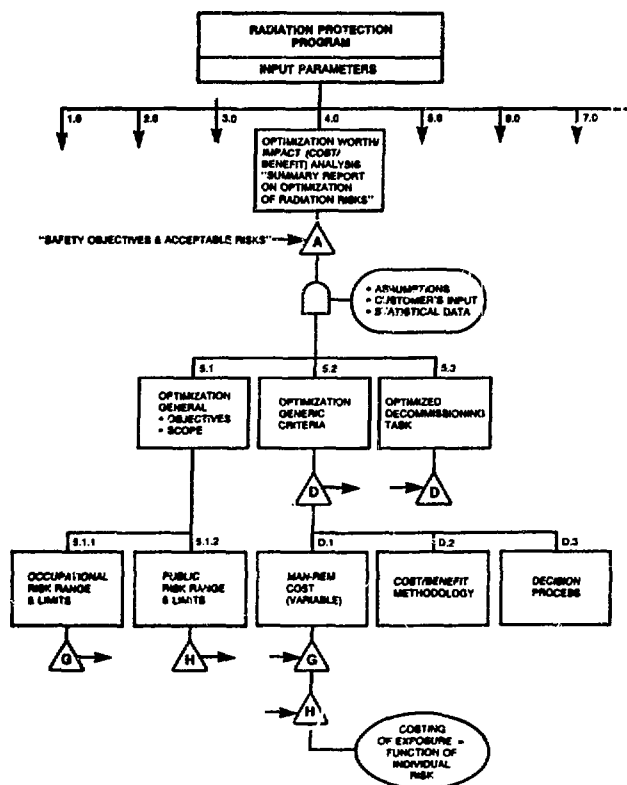


FIGURE 3 EXCERPT FROM MANAGEMENT OVERSIGHT RISK TREE LOGIC DIAGRAM (MORT)

assessing whether the prescribed QA requirements are met, while the HSG audit actually ensured a technical check that, for e.g., survey results reported by the RP are indeed correct and supported by the independent survey results; or, that the trash reviewed and declared clear is, indeed, clean, etc..

In this way, the HSG provided the project with valuable service in sponsoring the validity of records, documentation, reports, calibration records, surveys and dosimetric data. The professional level of RP program and its ultimate success has been thus truly assured.

## 8. ACKNOWLEDGEMENT

The authors wish to acknowledge a contribution to the success of the RP program to all those, whose knowledge, commitment to excellence beyond the call of duty, and attitude made the difference: Messrs. G. Powaschuk, J. Michalko, G. D'Aoust, G. Wilhelm, J. Youd, P. Denault, P.H. Dufour, A. Desrochers, J. Castonguay, R. Pressé and many others.

## Reference

- 1 International Commission on Radiological Protection, Recommendations of the ICRP, Publication 26, Annals of the ICRP, Volume 1, No. 3, Pergamon Press, Oxford (1977) 53 p.
- 2 International Commission on Radiological Protection, Recommendations of the ICRP, Publication 30 (1978-83) "Limits for Intakes of Radionuclides by Workers".
- 3 International Commission of Radiological Protection, "Cost-Benefit Analysis in the Optimization of Radiation Protection", Publication 37, Annals of the ICRP, Volume 10, No. (2/3) Pergamon Press, Oxford (1983), 75 p.
- 4 IAEA: Basic Safety Standard for Radiological Protection (Vienna 1982)
- 5 Gentilly-1 Health Guidelines and Radiation Protection Standards, Technical Document, Atomic Energy of Canada Limited, CANDU Operations, Rev. 3 (1985).
- 6 Gentilly-1 Guidelines on ALARA Implementation, Technical Document, Atomic Energy of Canada Limited, CANDU Operations, Rev. 0 (1984).
- 7 Maan, M.A., "Cost Effective Radiation Risk Management in a Generic Nuclear Plant Design", Proceeding, X-th International (Regional) Congress of IRPA, October (1982), Avignon, France, pp. 391-400 (also issued as AECL-7536).
- 8 Wilhelm, G., and Youd, J., "Evaluation of use of the "NNC-Gamma 12" Portal monitor for Internal Dosimetry Screening during Decommissioning" HPS 19th Midyear Topical Symposium, Knoxville, TN, February 2-6, 1986.
- 9 NUREG-0855: "Health Physics Appraisal Program (1982), Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C.

INDUSTRIAL SAFETY AND HEALTH PHYSICS INTERACTIONS DURING  
SLURRY TRANSFER OF 1000 CURIES OF RADIUM RESIDUES  
AT THE NIAGARA FALLS STORAGE SITE (NFSS)

Peter A. Williams  
Bechtel National, Inc., P. O. Box 350, Oak Ridge, TN 37831

ABSTRACT

Bechtel National, Inc. performed radiological decontamination operations on the Niagara Falls Storage Site in Lewiston, New York. A primary activity was the hydraulic transfer of 3,000 cubic yards of highly contaminated radium residues from the 160' tall K-65 Tower, approximately 5,000' across the site to an engineered storage facility.

The industrial safety hazards associated with the residue transfer were unique and involved the constant coordination of radiological controls and personal safety constraints for personnel working 160' above ground level. Safety was of paramount importance for the life-threatening confined space entries, 160' fall potential, approximately 5,000 man-hours trips, 2.5 million cubic feet of breathing air supplied, heat stress concerns, and implementation of emergency rescue provisions with subcontractor personnel and local fire-rescue agencies.

The methods and techniques of the planning, operational monitoring, and enforcement of the safety controls during this radiological cleanup operation will be presented. Emphasis will be directed to problem areas and solutions that can be applied to similar radiological operations.

# HEALTH PHYSICS EXPERIENCE DURING A URANIUM AND THORIUM POND CLOSURE

by

Leslie W. Cole  
Stephen V. Prewett  
Aerojet Heavy Metals Company  
Jonesborough, Tennessee

## ABSTRACT

Aerojet Heavy Metals Company (AHMC) recently completed a pond de-commissioning project which involved the movement and interment of approximately 600,000 cubic feet of uranium- and thorium-contaminated soil. This work was performed by a construction company using standard earthmoving equipment and their general labor force. In addition to the construction company employees, geotechnical quality assurance was performed by a soils test engineer and overall construction management was performed by a resident project engineer. Certification of cleanup and inventory sampling were done by contract personnel also.

This total action involved a significant number of people and large amounts of equipment working in a contaminated area. All of the work was performed with health physics and contamination control. A summary of the internal and external exposure results, ambient air quality measurements, personal air sampling measurements and excretion data will be presented for the project employees. All measures of exposure were uniformly low, as had been anticipated in project planning. The dosimetry effort was primarily aimed at verifying the anticipated low exposures.

Several large pieces of equipment such as track excavators, bulldozers, and scrapers were decontaminated. The concentrations of uranium and thorium present in the soil indicated minimal effort would be required for these operations. In practice, equipment typically could be decontaminated in less than two hours using a fire hose and a scraping device to remove imbedded contaminated soil. This greatly facilitated effective use of equipment and reduced the cost of the project significantly. Equipment breakdowns can pose a major problem when working with contaminated material. Experience indicates that all equipment should be in a good state of repair prior to its introduction into the controlled area.

In general, a large complicated project involving several different parties, a large amount of equipment and a large work area was completed with minimal exposure and no spread of contamination to uncontrolled areas. Detailed pre-planning with the project management staff and careful coordination with the actual construction contractor is required. Predictive methods of contamination and air activities can be valuable in assessing the program requirements.

## INTRODUCTION

The Aerojet Heavy Metals Company (AHMC) fabricates products made of depleted uranium. The AHMC facility, located in Jonesborough, Tennessee, was acquired by the Aerojet Ordnance Company in late 1976. Most of the ongoing operations were continued by Aerojet. Early in the operation of this facility, there was a uranium hexafluoride to uranium tetrafluoride conversion facility and a small thorium processing facility. Liquid wastes were treated and released to a 2.5 million gallon capacity holding pond. By 1981, it was apparent that the holding pond would not be adequate for planned expansion. The company decided to construct a modern water treatment facility to treat all process liquids. In addition, it was evident that the holding pond would have to be decommissioned. This decommissioning became the major element in a complete environmental action plan. A complete radiological survey of the site conducted by Radiation Management Corporation in 1982 identified some contaminated soil around the original buildings and along the bank of the holding pond nearest the building. Building expansion during 1982-83 required moving approximately 150,000 ft<sup>3</sup> of soil to a temporary storage location on the site.

Decommissioning the pond required several different operations. It was first necessary to remove the water. Then the sludge which had settled on the pond bottom had to be removed and disposed of properly. Finally, the pond bottom was excavated, liner placed in the bottom, the contaminated soil placed in and the top lined.

Before the final excavation of the pond bottom and construction of the interment site could proceed, it was necessary to construct a retaining wall on the downslope side of the pond. The construction of the retaining wall was not performed under radiological control procedures and will not be discussed further.

## OBJECTIVE

AHMC desired to remove all the contaminated water from the existing holding pond, remove the high concentration/mobility sludge from the bottom of the pond, and permanently inter all contaminated soils on the original pond site. The company was prepared to commit significant resources to undertake these operations and wished them complete in as timely a fashion as possible. The entire process would have to be approved by the appropriate regulatory agencies and monitored for compliance with objectives and costs by senior plant management. To meet these objectives the plant used a matrix organization headed by the Director of Environmental Health and Safety to manage the project. Supporting disciplines in health physics, waste management, plant engineering and purchasing were used. In addition to the plant staff, a significant contractor workforce was also employed for various phases of the operation. These contractors were controlled by the project manager and under contract to the company.

## DEWATERING OPERATION

At the start of the project, the pond contained approximately 2.2 million gallons of water. This was a result of past operations and stopped with the installation and permitting of a new liquid waste treatment facility. The plant was designed with the excess capacity to treat the remaining pond water in a timely fashion. The treatment process is fairly straightforward, with a physical separation made by precipitation and subsequent sludge water partitioning. Water quality for compliance with the permit limits was monitored prior to discharge of the water. Due to continued rainfall during the pond closure operations, the actual water treated was 4.5 million gallons in approximate 5,000-gallon batches. Initial treatability work was done on a bench scale with the operators used for routine plant waste water and after the process was proven out, an additional shift with operators was added to the waste water treatment plant to expedite water removal.

Four employees were primarily associated with the removal of the water from the pond. This removal activity included repositioning the intake for the pump to transfer the water from the pond to the pump house, and removal of the sludge generated as a result of the treatment from the sludge press. Routine operations could be carried out by either the technician or the utility operator, as required. Routine plant dosimetry in the form of whole body TLD's, routine bioassay, and in-vivo lung counts were performed. Additionally, one or two fixed position air samplers were operated in the actual waste water treatment plant. These results were compiled and showed no significant deviation from routine plant personnel working in uncontrolled areas such as the warehouse.

The universally low results were in agreement with our assessment of the radiological risk associated with this operation. Primary exposure concern was deemed to be internal and the most likely pathway to be ingestion of the sludge, either from the pond or from the filter presses as a result of plant operations. The material was always handled in either a fluid or damp state and airborne activity was not considered to be a likely exposure mechanism. The concentrations and low specific activity of the sludge also indicated unlikely direct exposure problems. The uniformly good results on internal measurements such as bioassay and lung burdens indicated care on the part of the waste water treatment plant operators in handling the concentrated sludge and the less frequent movement of the intake for the waste water treatment plant collection. We believe that any material with these low concentrations can be handled with reasonable care in the liquid state. Contamination due to leaks is always a potential problem and should be addressed in the facility's design and operation but does not pose other than ordinary contamination problem to the employees. Filter press sludge is a much higher contamination risk and should be dealt with accordingly. Standard waterproof clothing and gloves with safety glasses were found to be adequate with minimal operator training. Routine health physics coverage consisted of daily air sample collection and random checks for compliance with protective clothing and dosimetry requirements.

## SLUDGE REMOVAL

After dewatering was complete, AHMC contracted with Westinghouse Hittman to remove the sludge remaining at the bottom of the pond. This included packaging in a form suitable for burial and shipping to the burial site. The sludge was the most concentrated of the radioactive contaminants in the area and also exhibited a higher solubility. The sludge was removed by being shoveled into 55-gallon drums and lifted from the pond site via crane. To facilitate the decontamination of the exterior barrel surfaces, outer liners were used which were removed at the time of transport from the pond site to a process holding area. Varying numbers of people were used during the actual operation but typically in the range of six to ten actual workers. Further processing required an additional two to three employees in addition to supervision and health physics coverage for the operation.

The employees removing the sludge from the pond site were dressed in standard cotton coveralls, rubber boots for walking on the pond surface, and plastic raingear for outer covering. Hoods were provided to protect the head and neck areas and rubber gloves for the hands. A small change line was provided at the edge of the pond boundary where employees could remove their outer rubber or plastic garments, gloves and hoods prior to entering a small trailer to change out of the coveralls. Selected employees wore a personal breathing zone air sampler during the work day. All employees were on a routine bioassay and whole body dosimetry program. Additional airborne contaminant monitoring was done on a series of exposition RAS-1 air samplers around the perimeter of the pond site and also at the processing location for removed sludge.

During the entire operating period the site was in a wet condition due to ambient rainfall. The sludge did not become airborne. As a result, the mechanical handling operation and all air samplers (personal and fixed) were uniformly low. The routine bioassay sampling program detected no uranium excretion above background for the employees in question. During the sludge removal operation, a health physics technician was employed full time to monitor all the activities. The work force in general was local labor that had been through a several-hour health physics orientation tailored to their special work assignment. Compliance with the requirements was good and the detailed supervision available prevented any mishaps. It was not anticipated that whole body exposures would be an area of concern, given the short duration of the project and the relatively low results on the TLD's around the pond site used for area monitors. The further processing required to render the sludge suitable for burial took place in a tent and involved drying it to meet the no water criteria. Each drum of sludge was sampled and filled to capacity for economical burial and shipment. This involved handling material which was much drier and had a much greater potential for becoming an airborne hazard. The fixed position and personal air samplers for this area also indicated extremely low exposure levels and have been included as part of the general results.

The equipment used was generally metal, plastic and wood. Contaminants were introduced to surfaces in a wet form and could easily be

washed off with plain water and moderate scrubbing action. Some of the material, when dried, was somewhat baked onto the surface and required the use of a small steam ginny to remove the contaminants. Some more porous materials such as wooden surfaces and the interior cloth linings of boots, etc. became more imbedded with contaminants. These were generally cleanable but because of low value and low volume may not have been economically attractive in all cases. These decisions were made on a case-by-case basis by the site contractor.

## SITE CONSTRUCTION

The final phase of our remedial action project consisted of interring contaminated soils in a clay lined site. This was a major earthmoving project, taking approximately ten weeks to complete using standard equipment. The project involved certification of clean and inventory sampling, excavation of all contaminated soils, and construction of the clay envelope of the final repository. In general, all earthmoving was accomplished with large conventional equipment and all soil sampling was accomplished with a core or auger-type sampler.

All employees received a health physics orientation prior to the start of the project. In addition, all prospective contractors received information and evaluation of potential plans during their bid preparation phase. The bid package itself also contained sections on the restrictions necessary for health physics and contamination control and allowed the contractor to assign a cost to these items. Preparatory site work included the location of a mobile change trailer with water and electricity for use by the work force. Additionally, fire hoses and pumps were collected for use on equipment decontamination. During the bid preparation period and the bid package, it was stressed that all equipment should be in good working order prior to its entrance into the controlled area.

Actual work consisted basically of the following sequence of steps. First, the north end of the pond contaminated soil was removed down to an area that could be certified clean and stockpiled on top of the contaminated soil on the south end of the pond. As the final removal operation took place, a large-track excavator was used to reach out and scrape the soil off without cross-contamination. Sections were surveyed according to our pre-approved sampling plan and certified to be clean and appropriately marked to exclude traffic and further excavation. This proceeded until a sufficiently large area to begin construction of the bottom of the clay liner had been done. At this point some of the equipment used to excavate the contaminated soil was cleaned for use to haul clean clay fill in. This cleaning consisted of washing with a fire hose driven by a relatively low pressure, high volume pump. Difficult to remove contaminated soil was scraped or brushed by hand. Typically, a large piece of earthmoving equipment or a truck could be cleaned and certified within a two-hour period. Equipment breakdowns did occur and necessitated either cleaning the equipment or orientation of the repair force and work on the equipment in the contaminated state. This underscored the early emphasis of proper equipment maintenance before introduction to the controlled area. After the liner was complete, the contaminated soil stockpiled was placed on the clean liner, creating a new controlled

area. Additional excavation on the south end of the pond proceeded until the clean surface had been achieved again. At this point, another section of bottom liner was installed and a stockpile of contaminated soil was moved to the new liner. Some additional excavation of contaminated soil took place to collect all material at the original pond site. When this was complete, the finish grading of the contaminated soil was done with proper drainage and stability. At this point all the earthmoving equipment was decontaminated for the last time and either removed from the site or used for the clean clay cap.

During the contaminated soil removal operations air quality was monitored by a series of fixed position RAS-1 air samplers and by breathing zone samplers mounted on representative pieces of equipment. In general, the construction season was wet and dust control measures were not used. Both hoses and a water truck had been provided for dust control to minimize airborne contamination as required. In order to enhance compliance of breathing zone air samplers, the sampler was typically placed in the equipment rather than worn by the operator. Samplers were taken out at the start of each work day and collected at the end. Results were available in approximately 12 hours and reviewed with the project manager on a regular basis. All employees were also placed on a routine bioassay program with sample collection prior to start of work on Monday morning. TLD's were also issued for regular employees and processed with the normal plant TLD's. These results are uniformly low and indicate background levels of exposure for the work force. Good compliance with all practices was noted by the contractor personnel. At various times throughout the operation, people were transferred in for particular skills or for added labor and were removed for other requirements. In general, the stable work force of long-term employees made for a harmonious working relationship. New employees received their health physics orientation and detailed instructions prior to the commencement of work.

#### INSTRUMENTATION AND ANALYTICAL PROCEDURES

Most of the survey was done with a Ludlum Model 3 Survey Meter with a thin window Geiger-Meuller detector. This instrument has an efficiency of approximately six percent for surface contamination. It is estimated that if contamination were homogeneously distributed throughout the soil, it would be possible to detect approximately 100 picocuries per gram of uranium. Careful survey may allow somewhat lower levels, but not as low as 35 picocuries per gram. The Ludlum instrument is ideal for surveying equipment, clothing and machinery. Its size and the configuration of the detector allow easy access to small openings and uneven surfaces. Surface contaminations of 2000 dpm per 100 cm<sup>2</sup> are easily detectable.

Smear surveys were performed on all equipment leaving the construction area. All smears were measured on a Tennelec Model 5100 Series II. This instrument will detect Alpha contamination below 20 dpm per smear in a one-minute count. Smears were typically counted for one-half minute since the standard was 220 dpm alpha/100 cm<sup>2</sup> for release.

Air samples were analyzed using the Tennelec Model 5100 Series II. Fixed samples which were collected at 40 liters per minute for a 24-hour period had a minimum detection level of less than  $5 \times 10^{-14}$  microcuries per milliliter, which is one percent of the unrestricted maximum permissible concentration. The breathing zone samples were collected at two liters per minute for a full work day. The minimum detectable level for this sample size is about  $1.5 \times 10^{-12}$  microcuries per milliliter, or about 30% of the unrestricted maximum permissible concentration. The measurements on the breathing zone samples were made as though they were occupational samples.

All contractor personnel were monitored for external and internal exposure. The external monitoring was performed by Eberline Laboratories TLD Service. TLD's were collected on a monthly basis. The internal monitoring was through urinalysis. The analysis was done by Controls for Environmental Pollution, Inc. The urine sample frequency was at two-week intervals.

All personnel working on the ground in the area wore coveralls and shoe covers. Each monitored himself after changing out when leaving the controlled area. No contamination was found on any of the workers.

Analysis of the soil was made using a gamma ray spectroscopy system. This analysis is the subject of another paper.

## RESULTS AND CONCLUSIONS

The highest external whole body exposure was 22 mRem for one of the contractor personnel involved in the sludge removal. He was involved for about 70 days. The average whole body exposure was less than 0.25 mRem per day.

Air sampling did not indicate concentrations above the unrestricted limit permitted concentration level for any day of the operation. The average concentration was less than five percent of the unrestricted concentration limit for the fixed samples around the site and the breathing zone samples were at or below the minimum detectable level of 30% of the concentration limit for an unrestricted area.

The analysis of urine indicated only 14% of the samples above the minimum detectable level of 5 ug/l. The highest sample was 18 ug/l and this was one of the workers involved with the sludge removal. The highest among the workers involved with the earthmoving phase was 6 ug/l.

Equipment leaving the site was cleaned in the construction area with little difficulty. High-pressure water and brushes were sufficient for all the large equipment. Survey time was considerable but no difficulties were encountered. At the end of the project six large pieces of equipment were released in a single day by two technicians who did all the surveying. Equipment operators cleaned the equipment.

The entire project, while spread over approximately 20 months, was completed successfully from a health physics standpoint. No equipment or personnel were found to have external contamination after following proper protective and cleaning procedures. A project such as this can be monitored with a small number of health physics trained personnel supplemented by a minimally-trained contractor work force. Detailed pre-planning of all facets of such a project is an essential element.

**AN OVERVIEW OF DECOMMISSIONING AND DECONTAMINATION  
TECHNICAL INFORMATION SUPPORT ACTIVITIES FUNDED BY THE  
U.S. DEPARTMENT OF ENERGY'S REMEDIAL ACTION PROGRAMS**

**Park T. Owen  
Hazardous Materials Information Group  
Information Research and Analysis  
Information Resources Organization  
Oak Ridge National Laboratory  
Oak Ridge, Tennessee 37831**

**ABSTRACT**

In 1979 the U.S. Department of Energy (DOE) established the Remedial Action Program Information Center (RAPIC) at the Oak Ridge National Laboratory to provide technical information support to the Surplus Facilities Management Program, DOE's national decommissioning and decontamination (D&D) program which is managed by DOE's Richland Operations Office and UNC Nuclear Industries, Inc., and to the other DOE remedial action programs. Specific information activities that RAPIC performs to support the DOE's programs and the D&D community include: maintaining a computerized bibliographic database containing about 4500 annotated citations (about 2000 on D&D) and a database of 1800 contacts involved with remedial action work at radioactively contaminated sites; publishing an annual bibliography, Nuclear Facility Decommissioning and Site Remedial Actions, A Selected Bibliography, GENL/EIS-154 (six volumes published); maintaining a document repository and providing copies of requested publications; and performing manual and computerized searches of the technical literature. The most significant RAPIC function is serving as a focal point for D&D information. With the extensive resources at its command, RAPIC is in a unique position to provide a comprehensive information base to the D&D community. DOE makes these services available to foster good public relations and promote cooperation and information exchange.

---

\* Operated by Martin Marietta Energy Systems, Inc., for the U.S. Department of Energy under Contract No. DE-AC05-84OR21400.

## INTRODUCTION

In 1979 the U.S. Department of Energy (DOE) established the Remedial Action Program Information Center (RAPIC) at the Oak Ridge National Laboratory (ORNL) to provide technical information support to the Surplus Facilities Management Program (SFMP), DOE's national decommissioning and decontamination (DgD) program, and to other remedial action programs. RAPIC is funded by the two divisions of the U.S. Department of Energy that compose the DOE Remedial Action Program: Division of Facility and Site Decommissioning Projects and Division of Uranium Mill Tailings Projects. RAPIC operates under the auspices of the Office of Surplus Facilities Management, UNC Nuclear Industries, Inc., Richland, Washington, and the Surplus Facilities Management Office, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

RAPIC was established at ORNL because of ORNL's expertise in technical information management and analysis. The primary function of RAPIC is to ensure effective transfer of technical information pertinent to remedial action program activities to DOE staff, contractors, and governmental agencies. The programs and organizations that RAPIC was specifically created to serve include:

- Surplus Facilities Management Program  
Lead Field Office - DOE Richland Operations Office  
Lead Technical Contractor - UNC Nuclear Industries, Inc., Office of Surplus Facilities Management (OSFM)
- Formerly Utilized Sites Remedial Action Program  
Lead Field Office - DOE Oak Ridge Operations Office  
Lead Technical Contractor - Bechtel National, Inc.
- Uranium Mill Tailings Remedial Action Program  
Lead Field Office - DOE Albuquerque Operations Office  
Lead Technical Contractor - Jacobs Engineering Group, Inc.
- Grand Junction Remedial Action Program  
Lead Field Office - DOE Grand Junction Area Office  
Lead Technical Contractor - Colorado Department of Health
- Technical Measurements Center  
Lead Office - DOE Grand Junction Area Office  
Lead Technical Contractor - Bendix Field Engineering Corporation

Specific technical information tasks that RAPIC performs to support the DOE's programs and the DgD community include:

1. maintaining a computerized bibliographic database of the technical literature pertinent to remedial actions at radioactively contaminated sites/facilities,
2. publishing an annual bibliography of all documents identified as being pertinent to the remedial action programs,
3. collecting hardcopies of documents referenced in the database and providing copies of requested documents,
4. performing manual and computerized searches of the technical literature upon request,
5. maintaining a database of contacts involved with remedial action work,
6. providing technical information support at program-sponsored meetings and conferences.

## NUCLEAR FACILITY DECOMMISSIONING AND SITE REMEDIAL ACTIONS DATABASE

RAPIC maintains an online computerized database, Nuclear Facility Decommissioning and Site Remedial Actions, which is a comprehensive source of technical information related to D&D and other aspects of remedial actions at radioactively contaminated sites. Reports, journal articles, conference papers, books, theses, and patents from worldwide sources are scanned for inclusion in the database. The database, which was designed specifically for the Remedial Action Program, currently contains approximately 4500 annotated citations, of which approximately 2000 are pertinent to D&D activities. RAPIC developed 22 data fields to facilitate efficient online retrieval of needed citations, as well as to accommodate specific reporting requirements for the DOE programs. DOE program structure, as well as other parameters of interest to the program, is included in the multilevel indexing system.

The first level of indexing places documents into one of nine major subject categories, which include general areas of remedial actions and specific DOE remedial action programs. The primary categories are:

- ◆ Surplus Facilities Management Program
- ◆ Nuclear Facility Decommissioning
- ◆ Formerly Utilized Sites Remedial Action Program
- ◆ Facilities Contaminated with Natural Radioactivity
- ◆ Uranium Mill Tailings Remedial Action Program
- ◆ Grand Junction Remedial Action Program
- ◆ Uranium Mill Tailings Management
- ◆ Technical Measurements Center
- ◆ General Remedial Action Program Studies

The Surplus Facilities Management Program (SFMP) category contains references pertaining to the SFMP, program sites (located primarily on federal reservations), and specific D&D technology developed by the program.

The Nuclear Facilities Decommissioning category contains foreign D&D information as well as any D&D technology and domestic site-specific information that is not a part of the SFMP.

The Formerly Utilized Sites Remedial Action Program (FUSRAP) category contains references pertinent to FUSRAP management or to FUSRAP sites. These sites were used by the Manhattan Engineer District or by the Atomic Energy Commission from the 1940s through the 1960s for the processing, handling, storage, or shipment of radioactive materials.

The category for Facilities Contaminated with Natural Radioactivity contains information detailing remedial action work at industrial or research facilities (other than at FUSRAP sites) that are contaminated with naturally occurring radioactive materials.

The Uranium Mill Tailings Remedial Action Program (UMTRAP) category contains information pertinent to UMTRAP management or to approximately 25 UMTRAP sites, located primarily in the western United States. These sites are inactive uranium milling sites that were operated under government contract.

The Grand Junction Remedial Action Program category contains information pertinent to the remedial actions that are under way in Grand Junction, Colorado. This program is concerned with local structures that have in, under, or adjacent to their walls or foundations uranium mill tailings that originated from the Grand Junction uranium tailings pile.

The category for Uranium Mill Tailings Management contains foreign site-specific information, as well as any basic or applied research and domestic site-specific mill tailings information that is not part of UMTRA.

The Technical Measurements Center category contains reports published by the Technical Measurements Center (TMC), Grand Junction, Colorado, on the subject of detection and measurement of radioactive/hazardous contaminants, instrument calibration, and field calibration facilities.

The category for General Remedial Action Program studies contains references pertinent to DOE's Division of Facility and Site Decommissioning Projects, Division of Uranium Mill Tailings Projects, or activities associated with two or more of the DOE remedial action programs.

The secondary level of indexing identifies the major technical activity discussed in the document. The subcategories are:

- ◆ Design, Planning, and Regulations
- ◆ Environmental Studies and Site Surveys
- ◆ Health, Safety, and Biomedical Studies
- ◆ Decontamination Studies
- ◆ Dismantlement and Demolition
- ◆ Waste Disposal
- ◆ Remedial Action Experience
- ◆ General Studies

In addition to the bibliographic citation (title, author, author's affiliation, and publication description) and the various indexing fields to facilitate rapid retrieval, each record contains an informative abstract of the document.

Six volumes of a selected bibliography entitled Nuclear Facility Decommissioning and Site Remedial Actions, A Selected Bibliography (ORNL/EIS-154/V1 through V6) have been published. Chapters and subchapters of these bibliographies correspond to the major categories and subcategories of the database. Each volume contains approximately 700 citations to documents that were added to the database during the previous fiscal year. The bibliographies have a worldwide distribution of more than 1200.

It is preferred that researchers use these published bibliographies as their "first-line" reference tool; however, the database can be accessed through RAPIC for more current literature listings or for a comprehensive subject search of the entire database.

## OTHER ACTIVITIES

In addition to the document collection developed for the remedial action program, RAPIC and its associated information centers house a central document collection in excess of 10,000 titles pertaining to various aspects of the nuclear fuel cycle, primarily nuclear waste management.

Another RAPIC activity involves providing technical information support at program-sponsored meetings and conferences by setting up information exhibits and conducting poster sessions.

RAPIC also developed and continually updates the Remedial Action Contacts database. This database contains the names, addresses, and telephone numbers of approximately 1800 individuals involved with remedial action work at radioactively contaminated sites. The database also details technical areas of interest, as well as remedial action program involvement. The database is a useful reference for intra- and inter-group communication, for announcement of meetings and conferences, and for document distribution.

Communication established with D&D designated contacts and and researchers from Nuclear Energy Agency countries has resulted in an efficient information exchange process and has enhanced DOE's international technology transfer program. For example, RAPIC assisted UNC Nuclear Industries, Inc., in the preparation of the Compendium on Decommissioning Activities in NEA Member Countries (published January 1985), which was supported by DOE.

RAPIC's most important function is to serve as a focal point for D&D information. With the resources at its command, RAPIC is in a unique position to provide to requesters a comprehensive information base on various aspects of D&D activities, including, but not limited to:

- ◆ Planning and design
- ◆ Legislation, regulations, criteria, standards, and guidelines
- ◆ Characterization surveys
- ◆ Radiation detection instrumentation
- ◆ Environmental effects
- ◆ Operating procedures and methodologies
- ◆ Health physics considerations in operations
- ◆ Specific applications of technology and equipment
- ◆ Post remedial action surveys
- ◆ Surveillance programs
- ◆ Certification for reuse of sites

## CONCLUSION

The assistance that has been given in response to these and many other varied requests could be provided only by a group with the experience and personal contacts that the RAPIC staff has accrued in the area of nuclear facility decommissioning and remedial actions.

In summary, RAPIC is uniquely capable of serving as a comprehensive information broker for all remedial action activities. RAPIC has made itself highly visible in the remedial action field, and, through RAPIC, researchers and managers have access to a variety of technical information capabilities tailored to their needs.

All RAPIC services are provided at no cost to the requester. DOE makes these services available to all interested in remedial action activities in order to foster good public relations and promote cooperation and information exchange among the D&D community.

Any questions or information requests should be addressed to:

Park T. Owen  
Remedial Action Program Information Center  
Oak Ridge National Laboratory  
P.O. Box X, Building 2001  
Oak Ridge, Tennessee 37831  
Telephone: 615/576-0568 or FTS 626-0568

## DECOMMISSIONING PLANNING AND THE ASSESSMENT OF ALTERNATIVES FOR THE HANFORD PRODUCTION REACTORS

C. E. Miller  
U.S. Department of Energy  
P. O. Box 550  
Richland, Washington 99352

R. F. Potter and J. C. Chattin  
UNC Nuclear Industries  
P. O. Box 490  
Richland, Washington 99352

### ABSTRACT

The U.S. Department of Energy Hanford Site, near Richland, Washington, is the location of eight shut-down plutonium production reactors that are now in safe storage. These reactors each operated between 14 and 24 years, until shutdown of the last reactor in 1971.

Comprehensive decommissioning planning has been conducted for the reactors and their support facilities; actual decommissioning work has begun on some support facilities. Decommissioning criteria used at Hanford are discussed in this paper, as well as major planning assumptions. Decommissioning alternatives have been identified and assessed. The final decommissioning alternative will be selected through the National Environmental Policy Act (NEPA) process. The NEPA process for the Hanford reactor decommissioning has been initiated and will include a complete Environmental Impact Statement.

### INTRODUCTION

Several years ago, the U.S. Department of Energy began assessing alternatives and planning the decommissioning of eight shut-down plutonium production reactors located on the DOE Hanford Site in Washington State (Figure 1). These graphite-moderated, water-cooled reactors were built and started up between 1944 and 1955. In the 1960's, production needs declined and the reactors were one-by-one permanently shut down, the last of them in 1971.

### FACILITY DESCRIPTION

The eight shut-down Hanford reactors are similar to one another but very different in design and operation from commercial power-producing reactors. They were not used to generate electricity, and have no steam generating systems, pressure vessels, recirculating coolant components, or containment structures normally associated with commercial reactor plants. These reactors were cooled with pretreated Columbia River water, which made a single pass through the reactor, then was sent to a holdup basin before being returned to the river.

Each reactor consists principally of a graphite-core block penetrated by up to 3000 horizontal process tubes. Unirradiated fuel elements (cylinders nominally 1-1/2 in. in diameter and 8 in. long) were loaded end-to-end into the reactor process tubes via nozzles on the reactor front face. Once fueled, the reactor was brought critical and allowed to run until the fuel elements had produced a calculated quantity of plutonium. Then new fuel elements were charged into the process tubes, pushing the irradiated fuel elements out of the rear face ends of the process tubes and into a water-filled fuel storage basin. Irradiated fuel elements were periodically collected from the fuel storage basin and shipped to the plutonium extraction plant in the 200 Area of the Hanford Site (Figure 1). Many thousands of fuel elements were thus irradiated in each of these reactors.

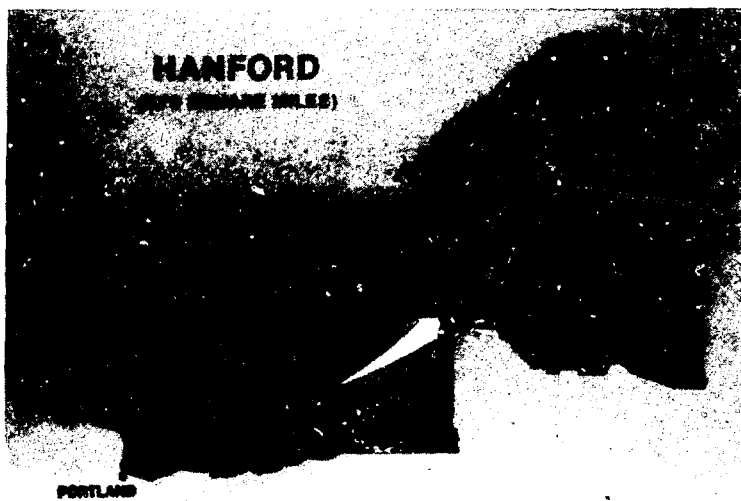


Figure 1. Hanford Site Map. Inset shows the 100 Area production reactor sites along the Columbia River.

### Physical Features

Each reactor block is housed within a reinforced concrete and concrete block reactor building. A typical reactor building is some 250 feet long, 250 feet wide and 100 feet high. It has massive 3 to 5 foot thick reinforced concrete walls around the reactor block at the lower levels, with lighter construction above.

The reactor block (Figure 2) consists of a stack of graphite blocks encased in a cast iron thermal shield, and around that a biological shield consisting of either concrete or alternating layers of masonite and steel, and finally an outer steel skin. A typical reactor block weighs approximately 9,000 tons and measures 46 feet high, 46 feet wide and 40 feet deep. It is within these graphite blocks and their surrounding shielding that the bulk of the radioactivity associated with the Hanford reactors is contained, primarily carbon-14 and cobalt-60.

105-F  
REACTOR BLOCK CONSTRUCTION DETAILS

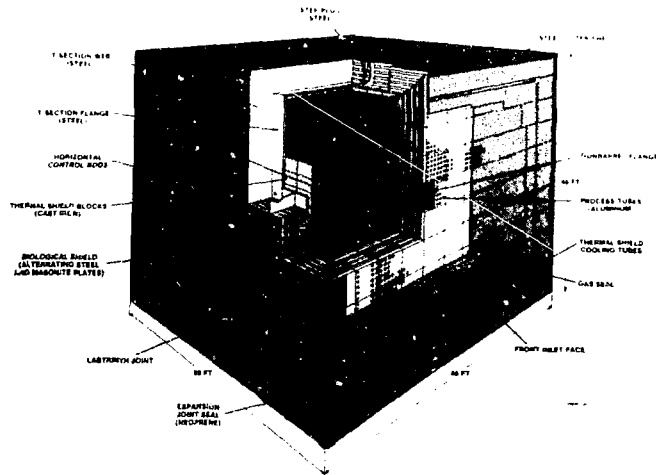


Figure 2. Typical Reactor Block. The graphite core is encased within a massive structure of cast iron, masonite, and steel shielding materials.

### Status Today

Starting in the 1960's, the eight reactors were shut down and defueled. They have been in safe storage for the past 14 to 20 years. This safe storage consists of maintenance and surveillance activities conducted to ensure industrial, radiological and environmental safety. These are only interim measures until a more permanent method of disposition is chosen.

### Radiological Condition

Each reactor block graphite core and its surrounding shielding contains up to 8,000 curies. Table 1 summarizes the estimated radionuclide inventory in a typical shut-down Hanford production reactor block.

### DECOMMISSIONING PLANNING EXPERIENCE AT HANFORD

Major decommissioning project work at the Hanford 100 Area began in 1983, and has included reactor exhaust stacks, gas recirculation and exhaust filter buildings, laboratory facilities, fuel examination facilities, contaminated effluent system piping, and fuel storage basins. The decommissioning work conducted so far at Hanford has provided valuable experience in identifying and assessing decommissioning alternatives and planning for maximum project cost-effectiveness and safety. For each facility decommissioned, various alternatives were proposed and examined, and final selections were based on such factors as expected occupational exposure, cost, project duration, manpower requirements, and expected radioactive waste volumes.

TABLE 1  
ESTIMATED INVENTORY OF RADIONUCLIDES IN A TYPICAL 100 AREA  
SHUT-DOWN REACTOR

Note: Typical inventory for one of eight production reactors.

<u>Radioactive Material</u>	<u>Half-Life (yr)</u>	<u>Total Inventory (Ci)</u>
<sup>3</sup> H	12.33	740
<sup>14</sup> C	5,730	3,600
<sup>60</sup> Co	5.27	3,100
<sup>63</sup> Ni	100	730
<sup>90</sup> Sr	29	0.002
<sup>93</sup> Mo	3,500	2
<sup>94</sup> Nb	20,000	2
<sup>137</sup> Cs	30.17	0.002
<sup>152</sup> Eu	13.4	8
<sup>154</sup> Eu	8.2	0.3
<sup>238</sup> Pu	87.74	4
<sup>239/240</sup> Pu	24,110	3
Estimated Total Ci:		8,000

#### Decommissioning Criteria Used at Hanford

Historically at Hanford, equipment and materials are released for unrestricted use when it has been determined that they are "free from contamination." Though this criterion is essential when releasing materials and equipment for offsite use, it can result in unreasonably high decommissioning costs when applied to large facilities. Consequently, a method was developed to define the amount of radioactive material that may safely remain after a facility has been decommissioned. Known as the Allowable Residual Contamination Level, or ARCL method,(1) it defines realistic exposure scenarios based on an analysis of potential pathways to humans for radiation exposure.

The predicted potential radiation doses are then calculated and compared to an established dose limit to define the ARCL for the specific mixture of radionuclides present at a specific facility or location. If the predicted potential dose exceeds the limit, then additional remedial action must be taken.

The ARCL methodology was used, for example, in the demolition and in-place burial of four reactor exhaust stacks. Even though stack demolition is a much smaller project than reactor decommissioning, using ARCL for this work saved \$1/2 million in decontamination and radioactive waste handling and burial costs. All Hanford decommissioning work must meet the DOE requirements for radioactive waste management.(2)

#### Planning for Reactor Decommissioning

The planning process for decommissioning the Hanford production reactors began when the reactors were originally shut down. The first step was the radiological and physical characterization of the reactors and the reactor buildings. This provided DOE with the basis for its present maintenance and surveillance program, which has effectively stabilized the facilities for the present.

The following five assumptions were used in identifying and assessing possible methods for decommissioning the Hanford production reactors.

1. The Allowable Residual Contamination Limits (ARCL) methodology(1) will be utilized to calculate the allowable quantities of residual radioactivity that can remain in the site for the in situ decommissioning alternative.
2. The radioactive waste materials in the Hanford reactors meet the DOE definition(2) of low-level wastes.
3. The reactor facilities and the land they occupy can, if necessary, be institutionally controlled for a period of up to 100 years. Institutional control means regulating the use of the decommissioned site by local, state, or federal authority.
4. Radioactive facilities decommissioned in situ will be isolated in a manner that provides a degree of protection to the public and environment equivalent to that afforded by 10CFR61.(3)
5. Intrusion barriers can, if necessary, be designed to last at least 500 years. Such barriers may be either engineered (concrete, riprap, etc.) or a stable earth cover up to 5 meters thick. Intrusion barrier requirements can be based on ARCL calculations.

#### NEPA PROCESS FOR REACTOR DECOMMISSIONING

Decommissioning the reactors themselves will involve a comprehensive NEPA process. Because of the proximity of the reactors to the Columbia River, and the large quantities of carbon-14 contained within them, the NEPA process for the project will include the development and submittal of a complete environmental impact statement (EIS).

The major milestones in the EIS process are the issuance of the Notice of Intent, the draft EIS, the final EIS, and the Record of Decision. On May 16, 1985, DOE published in the Federal Register a Notice of Intent to prepare an EIS pertaining to the decommissioning of these eight production reactors. The DOE Richland Operations Office is responsible for preparing the EIS document.

The EIS will address several decommissioning alternatives, identifying for each alternative its potential short-term and long-term environmental impacts, plus important engineering and cost considerations. The alternatives will include, as a minimum: 1) safe storage followed by deferred dismantlement, 2) immediate dismantlement, 3) in situ (or in place) disposal, and 4) taking no new action. Each of these alternatives is described below.

#### Development and Assessment of Decommissioning Alternatives

As shown in Figure 3, the reactors can be either decommissioned where they are in the 100 Areas, or moved to the Hanford 200 Area low-level waste disposal site for permanent disposition. And further, if moved to the 200 Area, they could be moved immediately or after a safe storage period of 75 years, which would allow the shorter half-life radionuclides like cobalt-60 time to decay. Within these three main scenarios (leave in 100 Area, move to 200 Area immediately, move to 200 Area later), seven decommissioning alternatives have been evaluated. They are discussed in the following paragraphs.

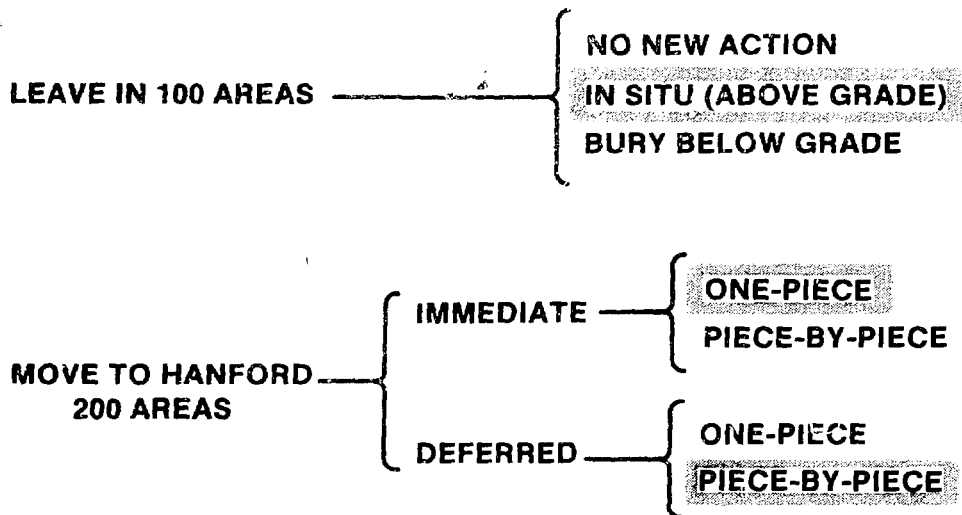


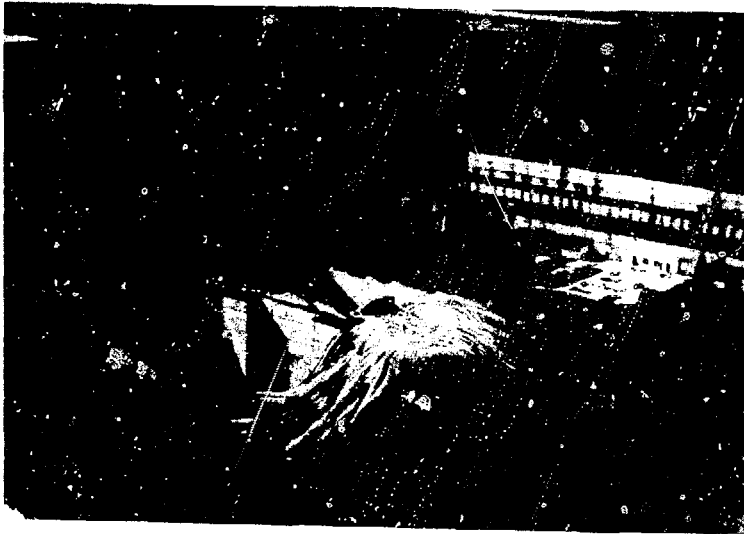
Figure 3. Hanford Reactor Decommissioning Alternatives. Highlighted alternatives have been recommended for consideration in the NEPA process.

#### Leave in 100 Areas

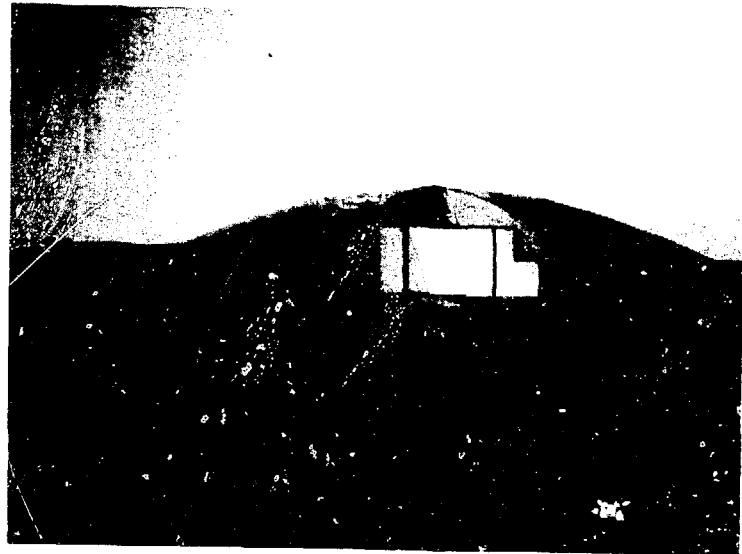
1. No New Action. In this alternative, the reactors would be left where they are and the present maintenance and surveillance program would continue. Although not technically recommended for these facilities, this alternative is addressed in the EIS process since it provides baseline environmental impact, cost, and other data for comparison purposes.
2. In Situ. In this alternative (Figure 4), the perimeter portions of the reactor building would be demolished, leaving the heavy shielding walls in place and the reactor block intact on its foundation. Major voids in and around the reactor block would be filled and any openings would be sealed. Then the block would be covered over to a minimum depth of 5 meters using building rubble and natural gravel as a long-term protective barrier. This alternative was recommended for further consideration in the NEPA process.
3. Bury Below Grade. This alternative was not recommended for further consideration because it would not yield any environmental, safety, or cost advantages over other alternatives. The cost and engineering challenges involved would be essentially the same as in moving the reactor block to the 200 Area.

#### Move to Hanford 200 Area

1. Immediate One-Piece Dismantlement. In this alternative, the reactor building would be demolished and the reactor block would be removed in one piece by excavating under the block, moving a heavy, tracked crawler under it, and positioning the block on the crawler. The crawler would transport the block to the Hanford 200 Area. This alternative was recommended for further consideration in the NEPA process.



Earth and building rubble are mounded over the partially demolished reactor building.



The finished mound, seeded with natural plant growth, would blend with the surrounding terrain.

Figure 4. In Situ Decommissioning of a Hanford Production Reactor (artist's concept).

2. Immediate Piece-by-Piece Dismantlement. This alternative would involve the complete dismantlement of the reactor building, and the segmentation and removal of the reactor block itself. Although this alternative is technically feasible, it would involve high occupational exposure (some 2000 man-rem), would require the design, fabrication, and use of special equipment, would result in a very high total decommissioning cost, and produce some 4 million ft<sup>3</sup> of solid radioactive waste. This alternative was not recommended for further consideration.
3. Deferred One-Piece Dismantlement. This alternative would involve storing the reactors until the level of radioactivity within the reactor block had decayed. The optimum safe storage period for this purpose has been determined to be 75 years. Following the 75-year safe storage period, the reactor would be dismantled and removed in one piece. This alternative is not recommended for further consideration because there would be no significant occupational exposure or technical advantage in deferring one-piece removal for 75 years.
4. Deferred Piece-by-Piece Dismantlement. This alternative would involve a 75-year safe storage period, allowing the level of radioactivity to decay to the point where hands-on dismantlement work can efficiently take place. At the end of the 75-year storage period the reactor buildings would be demolished, the reactors dismantled, and the waste taken to low-level waste burial areas on the Hanford Site. This alternative was recommended for further consideration.

#### Alternatives Recommended for NEPA Process

In May 1984, UNC Nuclear Industries (UNC) published UNI-2619, an Assessment of Decommissioning Alternatives for the Shut-Down Hanford 100 Area Reactors.(4) This document presents estimated costs, occupational radiation exposures, manpower requirements, waste volumes, and project completion times for several feasible decommissioning alternatives. The three alternatives assessed in UNI-2619 were:

- o Safe storage (75 years) followed by complete piece-by-piece dismantlement,
- o Immediate one-piece dismantlement, and
- o In situ disposal (burial in place)

Summary data for these three alternatives are presented in Table 2.

#### Final Selection of the Decommissioning Alternative

The NEPA process will consider the recommended alternatives described above as well as any additional viable alternatives raised during the preparation and review of the EIS. Then DOE will issue a Record of Decision identifying the chosen alternative. Once this is done, the actual decommissioning of the reactors can begin, and it can begin with confidence that an environmentally acceptable and cost-effective decommissioning alternative has been chosen.

TABLE 2

SUMMARY DATA FOR THREE DECOMMISSIONING ALTERNATIVES  
(Estimated totals for all eight Hanford reactors)

Assessment Factor	DECOMMISSIONING ALTERNATIVE		
	In-Situ	Safe Storage/ Deferred Dismantlement	Immediate Dismantlement
Cost*	\$40 million	\$420 million	\$400 million
Occupational Exposure	40 man-rem	900 man-rem	500 man-rem
Manpower	230 man-yr	2900 man-yr	1800 man-yr
Completion Time	10 yr	100 yr	20 yr
Waste Volume**	6000 ft <sup>3</sup>	1,800,000 ft <sup>3</sup>	3,400,000 ft <sup>3</sup>

\*Costs are escalated and reflect latest conceptual engineering and design estimates.

\*\*That volume of waste removed and taken to an approved waste burial site in the 200 Area of the Hanford Site.

## REFERENCES

1. Kennedy, W.D., Jr. and Napier, B.A., Allowable Residual Contamination Levels for Decommissioning Facilities in the 100 Areas of the Hanford Site, PNL-4722/UNI-2522, Pacific Northwest Laboratories, Richland, WA 1983.
2. DOE Orders, "Radioactive Waste Management," Order 5820.2, U.S. Department of Energy, Washington, D.C., February 1984.
3. Code of Federal Regulations, 10 CFR 61, "Licensing Requirements for Local Disposal of Radioactive Waste," U.S. Nuclear Regulatory Commission, Washington, D.C., 1981.
4. Adams, J.A., et al., Assessment of Decommissioning Alternatives for the Shut-Down Hanford 100 Area Reactors, UNI-2619, UNC Nuclear Industries, Richland, WA, 1984.

## **ENVIRONMENTAL ASSESSMENT RELATING TO THE DECOMMISSIONING OF STRONTIUM SEMIWORKS FACILITY**

V. Q. Hale and D. R. Speer  
Rockwell Hanford Operations  
P. O. Box 800  
Richland, Washington 99352

This Work Was Performed for the U.S. Department of Energy  
Under Contract DE-AC06-77RL01030

### **ABSTRACT**

The Strontium Semiworks complex on the Hanford Site was a pilot plant for radioisotope separation processes. The Strontium Semiworks facility was operated from 1949 to 1967 at which time it was retired and has been maintained in safe storage since then. The conditions of the building, the cost of safe storage, and the potential for radioisotope release lead to a decision in 1983 to decommission the facilities. The decommissioning alternatives considered were 1) earthen entombment without demolition, 2) partial dismantlement/entombment, 3) razing of all above ground structures, 4) total dismantlement, and 5) no action. An Environmental Assessment for the project was prepared in 1984 and a Finding of No Significant Impact issued on May 15, 1985. The approved method for decommissioning was Alternative 2 based on stability of the end product, cost of the project and projected impacts.

### **INTRODUCTION**

The Strontium Semiworks complex was used as a pilot plant for various radioisotope separation processes on the Hanford Site from 1949 to 1967. The complex consists of eleven contaminated facilities that are surplus to Department of Energy (DOE) needs. Since 1967 the facilities have been maintained in a safe storage condition requiring continued routine surveillance and maintenance. Because of the continuing costs for surveillance and maintenance and the desire to end perpetual active institutional control, decommissioning of the Strontium Semiworks complex was initiated in 1983. The initiation of decommissioning required the preparation of environmental impact documentation as required by the National Environmental Policy Act. Therefore an Environmental Assessment (EA) was prepared which resulted in the issuance of a Finding of No Significant Impact on May 15, 1985. This manuscript describes the salient points of that EA.

### **PROJECT DESCRIPTION**

The Strontium Semiworks complex consists of a process building; chemical and gas preparation buildings; a fan house with a 61-m (200-ft) tall stack; a solvent handling building; a storage and change house; and other support structures, some of which are attached to the process building (figure 1).

The principle structure is the process building consisting of three process cells, two other cells for storage and loadout, hot shop for cell equipment maintenance and an air treatment room. The cells are constructed of reinforced concrete walls on a 1.5-m (5-ft) thick concrete base pad which is 7.6 m (25 ft) below grade. The main process cell has 1.5 m (5 ft) thick walls and extends to 9.1 m (30 ft) above grade. The other cells have thinner walls and extend only up to grade. All of the cells contain a variety of tanks, columns and piping. Across the bottom of the three process cells and connected to the cells

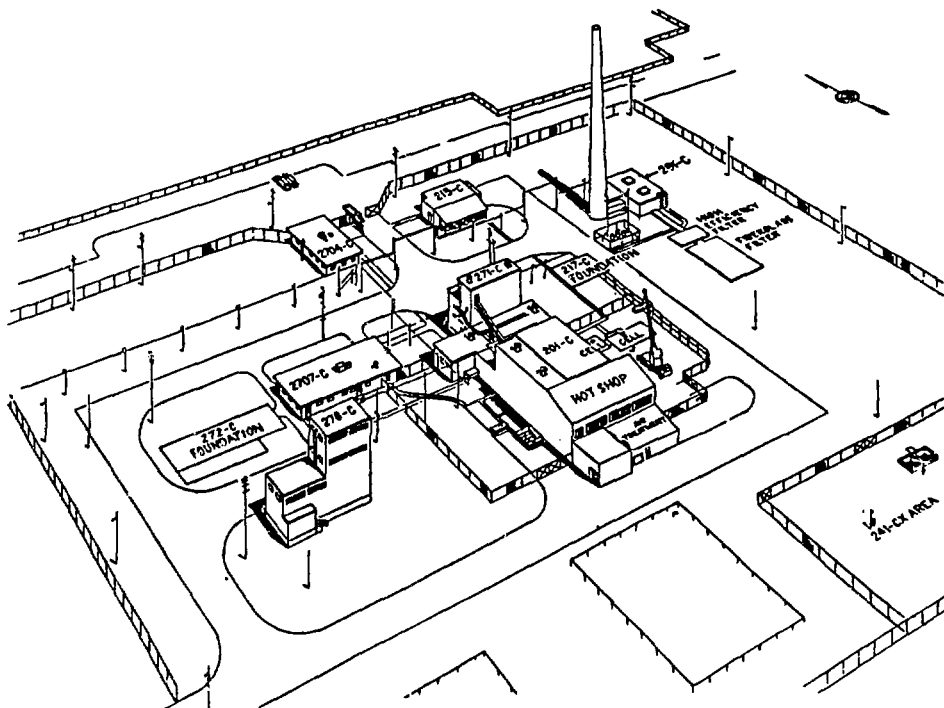


Figure 1. Overview of Strontium Semiworks Complex.

by louvered openings, is a concrete air tunnel that leads through deep bed fiberglass filters (DBFF) to the main stack. The rest of the buildings and the ancillary structures that surround the process cells are of steel and transite construction.

The most highly contaminated areas are the process cells and their related equipment. There is also some contamination in the galleries, hot shop and air treatment room. Following process pilot studies there were extensive decontamination campaigns and the resulting residual contamination consists of fission products with small amounts of transuranics (TRU). The quantities of TRU are sufficiently low to qualify the facility and any waste material removed as non-TRU low level waste (LLW).

The air exhaust system contains an estimated 22 TBq (600 Ci) of beta-gamma and 0.17 TBq (4.6 Ci) of alpha emitting isotopes. This system was not decontaminated between processes and has been in continuous use since the facility was constructed. In the early 1950s, high efficiency particulate air

(HEPA) filters were installed underground in series with the DBFF filters. However, most of the contamination in this system is in the tunnel and fiberglass filters.

### DECOMMISSIONING ALTERNATIVES

Various alternatives for decommissioning the Strontium Semiworks were evaluated. Whereas the site consists of eleven facilities and the disposition of each is described, only the 201-C Process Building was analyzed for the decommissioning alternatives because of its configuration and contamination levels. The No Action alternative would apply to all of the facilities. The alternatives for 201-C were entombment; partial dismantlement/entombment; razing; total dismantlement; and no action.

These alternatives are described as follows:

Alternative 1: Earthen Entombment. In this alternative, the above grade portion of B-Cell and the extracellular facilities would be left standing. The cells would be filled by pressure grouting, as would the A- and C-Cell sample trenches. Debris from other buildings would be placed in the cells, interlayered with grout. The interior and exterior transite walls of the galleries would be removed and all of the rooms filled with grout. Bottom ash would be used to fill the hot shop, backfill around the building, and fill the above-grade part of B-Cell. An earthen barrier consisting of rip-rap and soil would be placed over the whole site.

Alternative 2: Partial Dismantlement/Entombment. This alternative is to demolish 201-C to 3 m (10 ft) above grade and fill the remaining cells and galleries with concrete. Radioactive debris would be placed in the cells prior to their being filled or would be removed to a burial ground. Nonradioactive debris would be excessed or buried at a land fill. The site could be covered with an engineered barrier consisting of bottom ash, rip-rap, and soil.

Alternative 3: Razing. This alternative involves the razing of the extracellular facilities, demolition of the above grade portion of B-Cell, backfilling the cells with debris and concrete and placing an earthen barrier over the site. The below grade cell contents would remain in place and be filled with grout.

Alternative 4: Total Dismantlement. In this alternative all of the facilities would be dismantled and removed for disposal in a land burial site. The site would then be graded and released for unrestricted use.

Alternative 5: Continued Surveillance and Maintenance. This is the no action alternative in which no decommissioning activities would take place. Surveillance and maintenance would continue indefinitely in order to maintain a safe storage mode.

The other facilities within the Strontium Semiworks complex are being decommissioned using conventional decontamination and dismantlement methods which will vary between alternatives only in the extent to which they are dismantled. The following is a description of the decommissioning process for each of these ancillary facilities.

The Aqueous Makeup and Control Building, 271-C. Contaminated components in the control room have been removed and buried. Noncontaminated piping, tanks and control panels have been taken from the building and stored outside. The 296-C-2 Stack has been removed and buried. The building has been dismantled.

The Fan House, 291-C and Stack 291-C-1. This air exhaust system will be maintained until the below grade cells are filled. Exhaust will then be routed through a temporary above ground duct. The stack will be decontaminated internally and felled with explosives into a trench. The fan house will be dismantled. The associated below-grade ducts and filter cells will be filled with grout.

The Waste Storage Tanks, 241-CX-70, 71, and 72 will be emptied as far as practicable, the liquid taken to a tank farm and the tanks stabilized in place by filling with grout.

The other buildings that make up the Semiworks complex are the Solvent Handling Building (276-C), the Gas Preparation Building (215-C) and the Storage and Change House (2707-C). These buildings have been decontaminated as necessary and put to other uses.

Where facilities are demolished, the rubble is being retained and used as backfill material where possible. All pipelines into the area will be cut at the Semiworks site boundary, plugged with grout, and capped.

#### PROPOSED ACTION

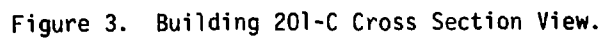
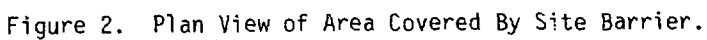
The approved decommissioning method for the complex is Alternative 2 based upon stability of the end product, cost and projected impacts of performing the work. The approved action is to demolish 201-C to ten feet above grade by first dismantling the second and third level galleries, the hot shop, air treatment room, and the 271-C aqueous makeup facilities. The first level galleries and the A, C, D, and E-Cells would be filled with concrete, and B-Cell would receive a partial concrete fill. The top 6.7 m (22 ft) of B-Cell would then be demolished.

The U.S. DOE Order 5820.2 provides guidance on acceptable practices for disposal of LLW via shallow land burial. The proposed in situ entombment provides a greater degree of waste isolation than would dismantlement and burial as LLW per DOE Order 5820.2. Therefore, the approved action is considered to provide adequate isolation and protection while keeping costs and exposures low. Permanent markers will be placed on the concrete monolith as warning to anyone who might dig into the mound after a projected minimum 100 year period of active governmental control. These markers provide an additional margin of safety even though the government intends to maintain active institutional control of the site indefinitely. This action will leave the site as a LLW site (figure 2).

All of the alternatives evaluated were considered practicable. Total dismantlement is the only alternative that would not dispose of the facility in situ. This alternative was not chosen because of its high cost and high occupational exposure (see table 1). Although Alternative 1 (earthen entombment) has the lowest exposure and cost, it was not selected because it would require the demolition of the 2707-C Building and the construction of a 12 m (40 ft) high earthen barrier. Construction of an earthen barrier of only 6 m (20 ft) in height over the site in the proposed alternative results in significantly less long term maintenance costs when compared to a 12 m (40 ft) high barrier. The completed site will appear as shown in figure 3.

#### AFFECTED ENVIRONMENT

The 200 East Area is located near the center of the 570-sq mi semiarid Hanford Site in southeastern Washington (figure 4). The nuclear waste processing operations are located on the 200 Area Plateau which is about seven



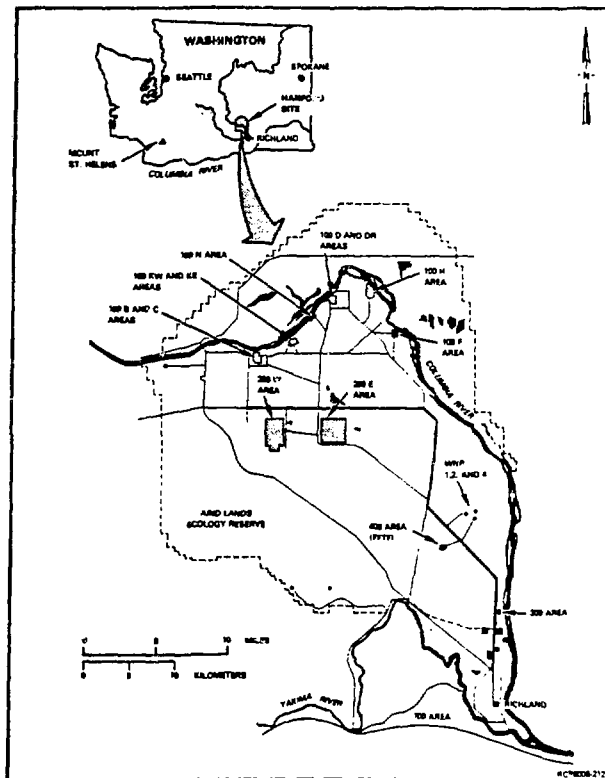


Figure 4. The Hanford Site.

miles from the Columbia River and above the projected 100 and 500 year floods. Approximately 300 feet of unconsolidated soil and geologic materials lie between the soil surface and the water table. The nearest population center is Richland, about 34 km (21 mi) southeast of the 200 East Area.

The Hanford Site is classified as a shrub-steppe grassland with the dominant plant community at the proposed site being sagebrush/cheatgrass. Common plants in the area are Russian thistle, rabbitbrush, and mustards.

A total of 27 mammal species have been identified on the Hanford Site. These include mule deer, coyotes, badgers, jackrabbits, Townsend's ground squirrels, deer mice and pocket mice.

Several bird species appear in the area. Raptors include Swainson's hawks, red-tailed hawks, American kestrels, long-eared owls, and burrowing owls. Song and perching birds that nest in the area include horned larks, sage sparrows, common ravens, loggerhead shrikes, western meadowlarks, and magpies. Hanford Site ponds are stopover points for migrating waterfowl and shorebirds and home to such species as mallards, American coots, buffleheads, long-billed curlews.

Snakes common to the area include gopher snakes, northern Pacific rattlesnakes and western yellow-bellied racers. The most common reptiles are side-blotched lizards. Insects, which are abundant on the Hanford Site, include several genera with grasshoppers and darkling beetles probably the most common.

The bald eagle is a winter resident but no species of plant or animal registered as rare, threatened, or endangered are known to depend on the habitats unique to the 200 Area Plateau.

A survey of the 200 Area Plateau by professional archeologists revealed no archeological or historical sites at or near the proposed project.

#### POTENTIAL ENVIRONMENTAL IMPACTS

Table 1 is a comparison of the estimated routine exposures and costs for the alternatives.

Table 1. Exposures and Costs for the Alternatives

	Occupational Exposure (man-Sv)	Costs (Millions 1984 dollars)
Partial Dismantlement/ Entombment	.027 - .041	3.5
Approved Action	.11 - .21	4.9
Razing	.25 - .58	8.0
Total Dismantlement	1.35 - 4.00	18.0
No Action (1st 100 years)	.05	10

A worst case hypothetical accident was determined to result from dropping a coverblock on the DBFF. This is assumed to release 0.25% of the total radionuclide inventory to the atmosphere in the respirable particle size range. The estimated radiation doses to the work force and general public are shown in table 2.

Table 2. Doses Resulting from a Worst Case Accident

	Work Force (man-Sv)		Public (man-Sv)	
	1 year dose	50 year dose	1 year dose	50 year dose
Total body	.12	1.70	.025	2.00
Bone	1.90	31.00	.25	10.00
Lung	.53	.53	.059	.059
Liver	.26	7.80	.029	.87

A long-term preliminary hazards analysis was prepared to assess causes, effects, severity, frequency, and mitigative measures for the hazard potential resulting from the proposed action. This analysis considered the two facilities with the greatest quantities of residual radioactivity, i.e., the process cells and the DBFF. The Analysis assumed that active institutional control can be relied on for 100 years and that passive controls may remain effective for hundreds of years. However, to provide a consequence analysis, an intrusion of the site is assumed. The radionuclide inventory of the subsystems is as follows.

Table 3. Inventory of Residual Radionuclides.

Isotopes	Subsystem Inventory, GBq (Ci)	
	Process Cells	DBFF
Pu-238	137 (3.7)	16 (0.43)
Pu-239	181 (4.9)	21 (0.57)
Pu-240	44 (1.2)	5 (0.14)
Cm-244	0	63 (1.7)
Am-241	7.4 (0.2)	81 (2.2)
Total Beta	3.3E+05 (9000)	2.2E+04 (600)

The most likely causes of airborne and surface contamination are intrusion events. Water intrusion that would result in groundwater and river contamination is highly improbable due to the nature and quantity of the isotopes and the sorptive capacity of the Hanford soils. Estimates of the vadose zone travel time for Pu-239 range from 1,000 years to infinity as evidenced by an evaluation that showed this isotope did not migrate even under wet soil conditions (Smith, 1980).

Active intrusion by a salvage digger and a homesteader were considered. Each was assumed to dig into the subsystem sites and the homesteader was assumed to ingest food grown on the excavated material. The resulting doses are calculated to be as follows.

Table 4. Calculated Doses to Intruders.

	Dose (Sv)			
	Salvage Digger		Homesteader	
	1 year	50 year	1 year	50 year
Total Body	1.3E-04	5.5E-03	0	3.6E-06
Bone	2.5E-03	.075	3.0E-07	8.5E-05
Lung	7.0	.073	7.1E-07	8.7E-07

Nonradiological impacts from this project will result from disposal of solid waste, dust and emissions from equipment, and noise. The largest noise will result from dropping the stack. The dust, emissions and noise will dissipate rapidly and will not have any lasting effects. A large volume of soil will be required to construct the barrier and this may require reclamation for stabilization of the borrow area.

#### CONCLUSIONS

An evaluation of cost, impacts, and stability of the end product showed that the proposed action was the most cost effective for the alternatives. The Environmental Assessment has been approved and a Finding of No significant Impact (FONSI) issued.

#### REFERENCES

- Smith, R. M., 1980, "216-B-5 Reverse Well Characterization Study,"  
RHO-ST-37, Rockwell Hanford Operations, Richland, Washington
- U.S. Department of Energy Order 5820.2, "Radioactive Waste Management"
- U.S. Department of Energy, 1985, "Environmental Assessment Relating  
to the Decommissioning of Strontium Semiworks Facility,"  
DOE/EA-0259, Department of Energy, Richland, Washington.

EVALUATION OF DOSES AND RISKS FROM DIFFERENT DECONTAMINATION AND  
DECOMMISSIONING STRATEGIES USING THE PRESTO-II METHODOLOGY

D. E. Fields  
Oak Ridge National Laboratory  
P. O. Box X  
Oak Ridge, TN 37831

ABSTRACT

The PRESTO-II methodology may be applied to evaluate doses and health risks from a variety of decontamination and decommissioning activities. This methodology has been implemented in the form of a computer code that has been applied to several sites, and that has been extensively documented. Radionuclide inventories are specified as separate contamination sources either present on the ground surface, covered by non-radioactive soils but lying above the water table, suspended in the atmosphere, or dissolved in surface waters. Hydrologic transport mechanisms considered in the PRESTO-II methodology include chemical exchange, ponding and overflow, surface water transport, groundwater transport, and pumping contaminated groundwater from wells. Varied scenarios of water usage are treated. Atmospheric inputs are based on both resuspension factor and resuspension rate approaches, with inhalation and immersion doses based on a Gaussian plume transport calculation. Site activities that are considered include land clearing, farming, and residing on the site. Exposure and dose calculations are derived from the U.S. Nuclear Regulatory Commission Reg. Guide 1.109 approach, while risk calculations use a life-table approach developed for the U.S. Environmental Protection Agency (EPA). Internal dose conversion factors are taken from ICRP 26 and 30, while risk conversion factors are values suggested by EPA.

INTRODUCTION

Sites under consideration for decontamination and decommissioning (D&D) often pose problems similar to those posed at sites under consideration as low-level waste (LLW) disposal facilities, or sites that have been used for past disposal of low-level wastes. For both types of sites, radionuclides have either been, or may be, dispersed both on the soil surface and in the near-surface zones. There may have been radionuclide spillage into local water sources, and injections into the atmosphere. Furthermore, consideration of future activities at both D&D and LLW sites leads to the possibility of concomitant human exposure. The choice of options to be exercised at a given site, whether for waste disposal or for D&D, defines important factors that determine future human exposure to such radionuclides. Because of these similarities, mathematical models that have been devised primarily for assessing human exposure from operations at LLW sites are likely to be useful for D&D sites as well. One model that has been recently documented, and that has been applied to several LLW disposal sites, is the PRESTO-II model.

---

\* Research sponsored by the U.S. Department of Energy under contract DE-AC05-84OR21400 with the Martin Marietta Energy Systems, Inc.

The PRESTO-II model is an implementation of the PRESTO methodology that was developed in 1983 for the Environmental Protection Agency (EPA). User documentation for the PRESTO-II model is in press (1). The model has been applied to several LLW disposal sites, and has been used to examine whether it might be possible to set de minimis radionuclide concentrations for certain waste streams (2). We will describe the PRESTO-II model and discuss its applicability to D&D activities.

#### IMPLEMENTATIONS OF THE PRESTO METHODOLOGY

The U.S. Environmental Protection Agency is responsible for developing a generally applicable environmental standard for the disposal of low-level radioactive waste (LLW). A methodology was developed to assist in assessing the potential human health impacts from such operations. This methodology was called PRESTO (Prediction of Radiation Effects from Shallow Trench Operations) (3). To assist in application of the PRESTO methodology, at least two different computer codes have been developed to estimate radionuclide transport and health risk. Documentation for Version 2.3 of the PRESTO-EPA model was delivered to EPA in April, 1983, and the EPA is continuing to develop and document versions of the PRESTO-EPA code (4).

In this paper we discuss the second code, PRESTO-II (1). PRESTO-II has been applied to simulate radionuclide transport at several DOE low-level radionuclide disposal sites, and for the NRC in support of a de minimis classification for waste (2). PRESTO-II differs from the EPA code in the following respects:

- PRESTO-II contains what may be a more defensible approach for computing infiltration through the trench cap. The input data-intensive calculation of trench-cap infiltration used in subroutine INFIL of PRESTO-EPA has been replaced by a simpler approach that computes this important variable from experimentally determined permeability and hourly precipitation values. Still other approaches to infiltration have been added as options: (1) use of yearly precipitation values; (2) user-specification of infiltration; and (3) estimation of trench cap infiltration as a fraction of calculated watershed infiltration. All of these methods provide values for infiltration through an intact trench cap - infiltration through failed portions of the cap is computed in a separate calculation.
- Watershed infiltration, an important variable for determining radionuclide weathering from the surface soils, is determined in PRESTO-II using a parametric evapotranspiration equation devised by Morton (5,6). This model requires input values for the following site variables: annual precipitation, mean sea level pressure, latitude, monthly dew point, monthly ambient air temperatures, and the observed fraction of maximum monthly sunshine.

According to Morton, this evapotranspiration model has been verified over a wide range of climates and reasonable estimates of water balance have been obtained for 122 river basins on three continents. Wallace (7) reports the Morton model to be superior to both the Thornwaite-Mather (8) and Penman (9) approaches to modeling evapotranspiration in certain environments.

- The computation of trench water balance (including water level, water overflow, and water percolation through the trench bottom), is made using a finite element approach in the PRESTO-II code. In comparison, a finite difference approach with a one-year time step is used in the PRESTO-EPA code.

The advantages of the finite element approach are that unphysical predictions of trench overflow are avoided in wet years, and that the time-averaged vertical extent of the saturated zone within the trench is available for use in other calculations (such as those determining radionuclide leaching and contaminated water release from the trench).

- A streamlined algorithm for describing radionuclide transport through subsurface pathways is used in the PRESTO-II code. The calculation of a correction factor for the combined effects of dispersion and radioactive decay used in the PRESTO-EPA code was not used in PRESTO-II. This correction factor differed from unity only in some scenarios in which the magnitude of human exposure was small.
- Additional user options for routing of contaminated water have been implemented in PRESTO-II. For example, contaminated aquifer water that is not pumped from the aquifer at the well may be routed to surface streams.
- The PRESTO-II code has profited from the discovery of several errors in early versions of the PRESTO-EPA model. These errors have been found and corrected in many cases through joint efforts by the authors of this report and by EPA researchers.

#### CODE APPLICABILITY

PRESTO-II is a computer code developed to estimate possible health effects from operations at LLW disposal sites. As previously mentioned, the code is also applicable to D&D sites. As an initial condition, up to 40 radionuclides may be specified according to the mass of each radionuclide present on the soil surface, in the air, in a local stream, or in a trench. As presently structured, the code can be used to make estimates for a period of up to 1000 years. The code is intended to be as generic as possible. Most of the transport methodology is derived from previously published work; e.g., (10). Different transport mechanisms may be considered, but there has been no intent to tailor the code to a particular site. Nevertheless, the assumption in the PRESTO methodology of saturated hydrologic transport results in the prediction of more rapid radionuclide transport than would actually occur at some arid sites. In most cases, the model results are conservative for such sites. The DARTAB model (11) is used in modified form as an important module of PRESTO-II to generate human health-risk estimates from radionuclide concentration and intake values. Cancer risks are calculated for individuals or populations over the assessment period using a life-table approach developed by EPA (12).

The PRESTO-II code construction is modular to allow alternate submodels or subroutines to be substituted if necessary. Many of the submodels included in PRESTO-II were developed for other types of assessments and have been adapted for use in estimating health effects from shallow land disposal of LLW.

The PRESTO-II model is most appropriately used as a screening model, and site specific codes should be considered in cases where PRESTO-II numerical results fall close to reference values used for decision making. The model may be used to estimate doses to populations as an aid to the ALARA process. It may also be used to estimate doses to maximally exposed individual members of the public by specification in the input data that the location of the individual and of the well, used for obtaining water for drinking, crop irrigation, and livestock watering, is at the site boundary. The code contains algorithms to track both radionuclide amounts and radionuclide concentrations in well water. Radionuclides are first withdrawn from the well for human ingestion in an amount appropriate for the number of exposed persons specified, and remaining radionuclides may be withdrawn to be used for irrigating crops and watering livestock. A second reason to account for both amount and concentration of

radionuclides at the well is to insure that the amount withdrawn at the well (based on the calculated concentration at the well) does not exceed the amount available. As a result of this calculation, the dose and risk calculated for a single individual may exceed the dose and risk calculated for individual members of a large population taking water from a well.

Use of PRESTO-II to calculate population doses for considerations of ALARA, or to calculate maximally exposed onsite or offsite individual doses for consideration of compliance with maximum exposure regulations, is determined by the user's choice of population size, well location, and other input option values and parameters.

PRESTO-II is designed to consider the following near-surface storage or disposal situation. Surface contamination may be present. The sub-surface contamination zone is treated as three regions (13) -- a trench, a vertical transport zone, and a horizontal transport zone, or aquifer. If an actual trench is present, the following considerations apply. Waste materials may be buried in many types of containers ranging from plastic bags or cardboard boxes to steel drums. The trench may be backfilled to eliminate voids and decrease subsidence, thus minimizing possible trench cap cracking. When completely filled, the trench may be assumed covered with a cap of soil or clay mounded above grade to facilitate runoff and decrease infiltration of precipitation.

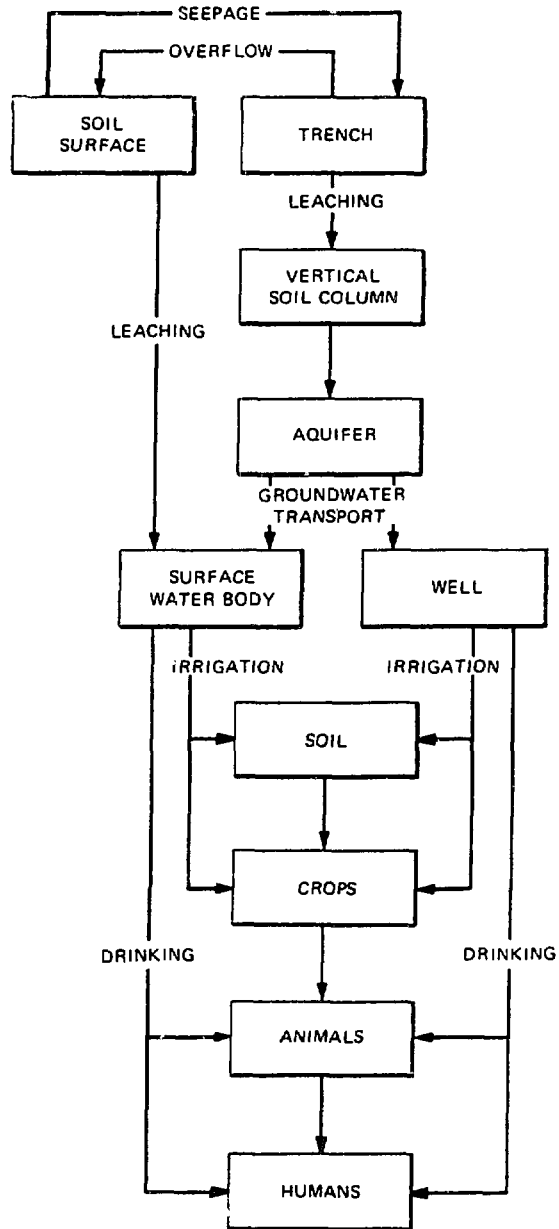
#### MECHANISMS CONSIDERED

Water, principally from precipitation, is often a primary transport medium for radioactivity from LLW stored in shallow trenches or in surface soils. Environmental pathways driven by hydrologic mechanisms are shown schematically in Figure 1. Precipitation falling on the trench may either infiltrate the soil or trench cap, run off the surface, or evaporate, and radionuclides may be transported from a LLW disposal trench by groundwater or by precipitation runoff. Runoff also causes erosion of the soil surface and the trench cap. Water percolating downward through the trench to the subtrench soil zone may ultimately enter an aquifer. Radionuclides that finally reach the aquifer will generally be transported more slowly than the characteristic flow velocity of water in the aquifer because the radionuclides interact with the solid materials in the aquifer. Some of the radionuclides which enter the aquifer may eventually reach irrigation or drinking wells or surface streams.

Contaminated water from the trench may overflow onto the surface soil. Once overflow has occurred, radionuclides may be transported by runoff water to nearby streams and become available for human consumption via irrigation or drinking.

Humans may also be exposed to radionuclides transported from D&D sites by atmospheric processes. Atmospheric transport processes are summarized in Figure 2. Radionuclides left on the soil surface by trench overflow, by spillage during disposal operations, or by complete removal of the trench cap may be suspended in the atmosphere and transported downwind where they may be inhaled or deposited on vegetation and soil. In the case of deposition, the radioactivity may produce external exposure to humans or may enter into the food chain and result in radioactivity in crops, meat, and milk.

The code allows the user to select human exposure scenarios including: migration of radioactivity from the trench in hydrologic and atmospheric environmental pathways to food and drinking water, the presence of a resident intruder on the site, and farming of the site. Processes considered in calculating individual or population exposure include: groundwater transport, precipitation runoff, trench water overflow and seepage, chemical exchange, trench cap erosion, stream dilution, and resuspension and atmospheric dispersion of contaminated soil followed by inhalation or deposition on crops and land.



**Figure 1. Hydrologic environmental transport pathways.**

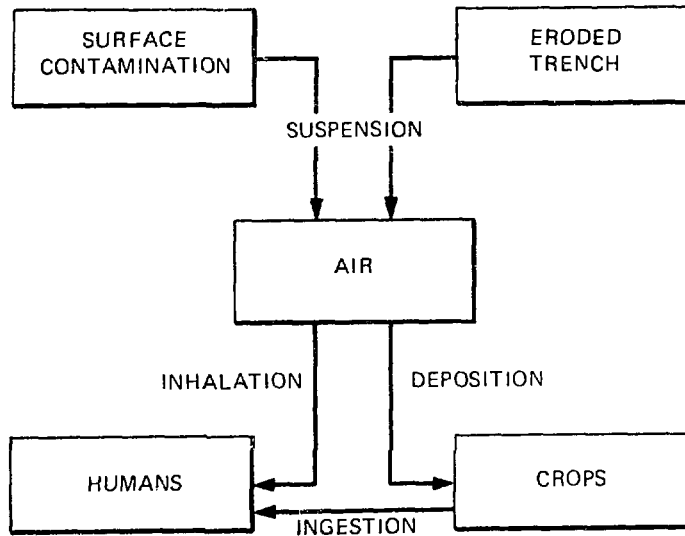


Figure 2. Atmospheric environmental transport pathways.

The termination of LLW disposal operations (as associated with final trench closure), or the completion of D&D operations at a site, is the starting time for the total assessment period considered by the code. The user specifies the total length of the assessment period, from 1 to 1000 years. The health effects for a user-specified length of time within this assessment period are calculated. The averaging time for health effects can be as short as any single year of the assessment period or as long as the entire assessment period. The maximum concentration of each radionuclide and the year of this maximum for each of the four major exposure pathways for each nuclide is calculated. The initiation time and period for mechanical suspension of the surface soil by farming is specified by the user. The user can also specify the time at which the trench cap begins to fail and the temporal profile of this failure, which results in exposure of the trench contents. The code time step is fixed at one year.

The resident intruder scenario assumes that an intruder digs into the disposal trench while building a residence. The individual will receive external exposure to the buried radionuclides due to time spent in the basement of the residence, which is assumed to have walls made of trench material. The user-specified length of residency, time in the assessment period when the residence is first occupied, and the composition of the initial trench inventory will all contribute to the dose from external exposure.

Farming the site after loss of institutional control is treated as a separate intrusion scenario. By farming the site a farmer will be affected by most of the hydrologic and atmospheric transport processes described above. In contrast to an off-site population, however, a large portion of the farmer's food may consist of crops grown in the contaminated zone near the trench. The water used by the farmer for irrigation, drinking, and livestock may be taken from directly beneath the trench or from a nearby stream and thus contain a much higher concentration of radioactivity than water used by the off-site population. The farming process may also mechanically suspend contaminated soil in the atmosphere. Such mechanical suspension may impact both the farmer and a downwind population.

Annual-average concentrations of each radionuclide in environmental media (such as well water or the atmosphere) over the assessment period are used to calculate radionuclide concentrations in foodstuffs. Foodstuff information and human ingestion and breathing rates are utilized to calculate the annual average radionuclide intake per individual in the population by ingestion and inhalation. These intake data are used by the exposure and risk submodels to estimate dose rate and health risk.

The health risk estimation methodology assumes that each member of the population is a member of a cohort that is exposed to constant, annual-averaged radionuclide concentration levels. For atmospheric transport calculations, the total population is assumed to reside within the same 22.5-degree sector. User-specified parameters give the fraction of the year that the wind blows into that sector. A user option allows the results of the atmospheric dispersion calculation in the code to be replaced by an externally calculated dispersion coefficient which considers several population centers. Each member of the population is assumed to eat the same quantities of food (vegetables, beef, and milk). These foods are assumed produced on the same fields and spray irrigated with contaminated water. Contaminated water is assumed drunk by beef and dairy cattle. The code user may independently specify the fractions of water taken from wells or streams and used for human consumption, for irrigation, or for livestock consumption.

PRESTO-II considers only one scenario per computer run. The scenario to be simulated may be structured by the user by changing values of user-specified parameters such as population size and location, distance to well, percent cap failure, resuspension rate, etc.

## APPLICATIONS

The PRESTO-II code has been applied to simulate transport from three low-level waste disposal sites (1). Input data were compiled for each of three example locations: Barnwell, South Carolina; Beatty, Nevada; and West Valley, New York. Due to the lack of availability of some data, these data must be regarded as example values designed to verify operation of the code, rather than as unanimously agreed upon numerical specifications of these sites. Results from each of the three sites suggest that there may be low health risk to nearby populations from shallow land disposal of LLW. A summary of results for one scenario at the Barnwell site is presented in Table I.

A second set of applications of PRESTO-II has been used to support consideration by the US Nuclear Regulatory Commission of a de minimis disposal standard for certain waste streams (2, 14).

PRESTO-II has been used at Oak Ridge National Laboratory to examine the consequences of storage or disposal of low-level radioactive waste at various site on the Oak Ridge Reservation. It is also being used to explore the consequences of various remedial activities on the Reservation.

Finally, the code is being applied to evaluate the adoption of various LLW disposal strategies by underdeveloped countries (15).

## SUPPORTING CODES AND DOCUMENTATION

To assist the PRESTO-II user in compiling radionuclide data, a radionuclide Daughter Inventory Generator (DIG) computer code has been written (16). This code facilitates the construction of the site inventory radionuclide data set for the PRESTO-II code. The DIG code accepts a table of radionuclide names and amounts initially present in a waste stream and produces a tabulation of radionuclide names and amounts present after a user-specified elapsed time. This resultant radionuclide inventory characterizes wastes that have undergone radioactive decay and daughter ingrowth before leaching and transport, thus

**Table I. Results of example problem based on Barnwell, SC data.  
Estimated health risks to surrounding population  
for 1000 years following trench closure (1)**

---

Site	Barnwell, S.C.
Population	7033
Deaths/year	5.5E-4
Annual risk per person	7.2E-8

**Contributions to net annual death rate by pathway or mode**

INTERNAL	5.5E-4
INGESTION	5.5E-4
INHALATION	1.8E-14
EXTERNAL	9.1E-13
AIR IMMERSION	2.3E-17
GROUND SURFACE	9.1E-13

**Contributions to net annual death rate by radionuclide  
(20 selected radionuclides):**

H-3	2.6E-14
C-14	2.5E-5
Mn-54	2.1E-16
Fe-55	3.6E-16
Co-60	3.6E-13
Zn-65	3.8E-16
Sr-90	7.9E-16
Tc-99	5.3E-4
Cs-134	3.2E-14
Cs-137	2.3E-13
Ba-137	5.8E-13
Ce-144	2.2E-20
Pb-210	4.1E-17
Ra-226	4.8E-19
Th-232	7.4E-18
U-234	1.1E-17
U-235	1.6E-17
U-238	1.1E-14
Pu-238	1.6E-18
Pu-239	2.9E-18

---

identifying daughter radionuclides that should be considered for inclusion in the pollutant source term. The output of the DIG code also includes radionuclide decay constants, which are required in the PRESTO-II code input data set.

A second code has also been written to assist the user in preparing PRESTO-II data sets. This second code is PRESTO-PREP (17). The PRESTO-PREP code accesses radionuclide data bases to prepare a radionuclide data set in the proper format for reading by PRESTO-II code.

#### CONCLUDING REMARKS

PRESTO-II, while designed for simulating radionuclide transport and human exposure associated with LLW sites, has characteristics that make it useful in evaluating the relative merits of various D&D alternatives. The code is documented and has been applied to various sites. In addition to the code itself, several supporting codes and methodologies, including PRESTO-PREP (17), DIG (14), and DARTAB (11), have also been documented.

PRESTO-II, like any complex computer code, may be misapplied. Misapplication may consist of trying to apply the code to examine a site where one or more modeling assumptions are invalid, or of choosing values of input parameters that do not accurately reflect variables such as radionuclide inventory, site meteorology, surface and subsurface hydrology and geology, and future population demographics. One serious misapplication of the PRESTO-II methodology would be to use carelessly chosen  $k_d$  values to determine hydrologic transport. Element-specific values of  $k_d$  vary widely as sites, chemical forms, and other waste components are changed.

Some release and transport scenarios, such as those necessary to consider major meteorological changes over the simulation period, injection of subsurface radionuclides into the atmosphere, in-situ combustion of wastes or contaminated materials, and radionuclide transport by burrowing animals or plant roots, are not considered in the PRESTO-II model and code. The user must make significant changes to the code and the input data if those mechanisms are to be considered.

#### REFERENCES

1. Fields, D. E., R. O. Chester, C. A. Little, C. J. Emerson, and G. Hiromoto, PRESTO-II: A Low-Level Radioactive Waste Environmental Transport and Risk Assessment Code, Oak Ridge National Laboratory Report ORNL-5970 (in administrative review).
2. Fields, D. E. and C. J. Emerson, Unrestricted Disposal of Minimal Activity Levels of Radioactive Wastes: Exposure and Risk Calculations. Oak Ridge National Laboratory Report ORNL-6001 (August 1984).
3. Little, C. A., D. E. Fields, C. J. Emerson, and G. Hiromoto. 1981. Environmental Assessment Model for Shallow Land Disposal of Low-Level Radioactive Wastes: Interim Report, ORNL/TM-7943, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
4. Bandrowski, M. S., and Cheng-Yeng Hung. 1985. "Environmental Transport Pathways of the EPA Model (PRESTO-EPA) Used to Determine Health Impact from Low-Level Radioactive Waste Disposal, Proceedings of the Eighteenth Midyear Topical Symposium of the Health Physics Society, January 6-10, 1985, Colorado Springs, Colorado.

5. Morton, F. I. 1976. "Climatological Estimates of Evapotranspiration," Proc. ASCE, J. of Hydraulics Div. Paper 11974, 102(HY3):257-290.
6. Morton, F. I. 1978. "Estimating Evapotranspiration from Potential Evaporation: Practicality of an Iconoclastic Approach," J. Hydrology 38:1-32.
7. Wallace, R. W. 1978. A Comparison of Evapotranspiration Estimates Using DOE Hanford Climatological Data, PNL-2698, Pacific Northwest Laboratory, Richland, Washington.
8. Thornthwaite, C. W., and J. R. Mather. 1955. "The Water Balance," Climatol. 8(1), Laboratory of Climatology, Drexel Institute, 104 p.
9. Penman, H. L. 1948. "Gas and Vapor Movement in the Soil: I. The Diffusion of Vapor through Porous Solids," Journal, Agr. Sci. 30:437-461.
10. USNRC. 1977. Regulatory Guide 1.109: Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, U.S. Nuclear Regulatory Commission, Washington, D.C.
11. Begovich, C. L., K. F. Eckerman, E. C. Schlatter, and S. Y. Ohr. 1981. DARTAB: A Program to Combine Airborne Radionuclide Environmental Exposure Data with Dosimetric and Health Effects Data to Generate Tabulations of Predicted Impacts, ORNL-5692, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
12. Dunning, D. E., Jr., R. W. Leggett, and M. G. Yalcintas. 1980. A Combined Methodology for Estimating Dose Rates and Health Effects from Exposures to Radioactive Pollutants, ORNL/TM-7105, Oak Ridge
13. Fields, D. E. 1982. "Computation of Radionuclide Particulate Finite Area Fugitive Source Strengths," Health Phys. 44(6):653-655.
14. Fields, D. E., "Exposure and Risk Calculations for Disposal of Wastes Having Minimal Radioactivity", Proceedings of Waste Management '84, Waste Isolation in the U.S., Technical Programs and Public Education, March 11-15, 1984, Tucson, Arizona, pp 661-664.
15. Yalcintas, M. G., and D. E. Fields, "Consideration of Alternatives Regarding Shallow Land Disposal of Low-Level Radioactive Waste for a Developing Country," accepted for presentation at Waste Management '86, Tuscon, Arizona, March 2-6, 1986.
16. Fields, D. E., and R. D. Sharp, Radionuclide Daughter Inventory Generator Code (DIG), ORNL-6110, Oak Ridge National Laboratory (September, 1985).
17. Bell, M. A., C. J. Emerson, and D. E. Fields, PRESTO-PREP: A Data Preprocessor for the PRESTO-II Code. Oak Ridge National Laboratory Report ORNL-5981 (July 1984).

# ESTIMATING DOSES AND RISKS ASSOCIATED WITH DECONTAMINATION AND DECOMMISSIONING ACTIVITIES USING THE CRRIS

C. W. Miller,<sup>\*</sup> A. L. Sjoreen,<sup>#</sup> and S. J. Cotter<sup>\*</sup>  
Oak Ridge National Laboratory  
P. O. Box X  
Oak Ridge, TN 37831

## ABSTRACT

The Computerized Radiological Risk Investigation System (CRRIS) is applicable to determining doses and risks from a variety of decontamination and decommissioning activities. For example, concentrations in air from resuspended radionuclides initially deposited on the ground surface and the concentrations of deposited radionuclides in various soil layers can be obtained. The CRRIS will estimate exposure to radon and its progeny in terms of working-level months, and will compute the resulting health risks. The CRRIS consists of seven integrated computer codes that stand alone or are run as a system to calculate environmental transport, doses, and risks. PRIMUS output provides other CRRIS codes the capability to handle radionuclide decay chains. ANEMOS and RETADD-II calculate atmospheric dispersion and deposition for local and regional distances, respectively. Multiple ANEMOS runs for sources within a small area are combined on a master grid by SUMIT. MLSOIL is used to estimate effective ground surface concentrations for dose computations. TERRA calculates foodchain transport, and ANDROS calculates individual or population exposures, doses, and risks. Applications of the CRRIS to decontamination problems are discussed.

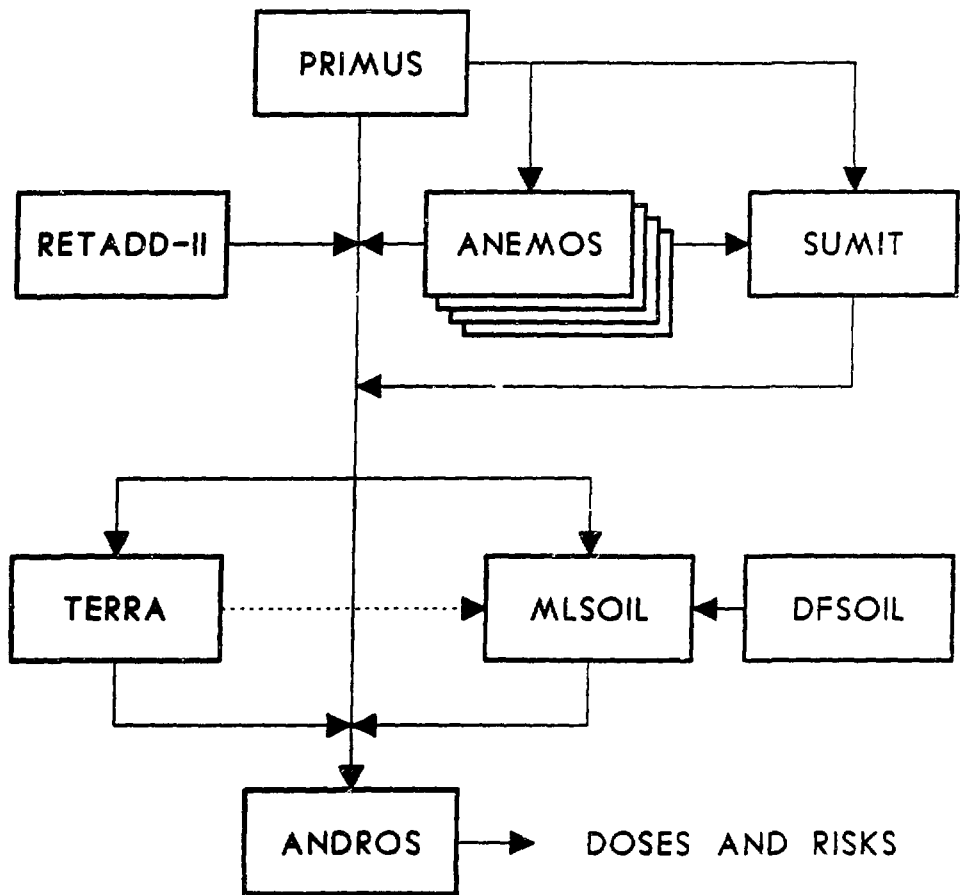
## INTRODUCTION

Work is being completed on an integrated set of computer codes and data bases to serve as a state-of-the-art radiological risk assessment methodology for the U. S. Environmental Protection Agency (EPA). While designed primarily to assist EPA in regulating radioactive effluents covered by the Clean Air Act amendments of 1977, these codes can also be used in a variety of decontamination and decommissioning activities. Shown in Fig. 1 are the codes that comprise this Computerized Radiological Risk Investigation System, or CRRIS, and their interrelationships. The remainder of this article will briefly examine each of the CRRIS codes and discuss their use in performing nuclear risk assessments, especially those for decommissioning and decontamination activities.

---

<sup>\*</sup>Health and Safety Research Division

<sup>#</sup>Computing and Telecommunications Division



**Figure 1. The computer codes which comprise the CRRIS system and the various pathways of interaction among them.**

## PRIMUS

The PRIMUS code (1) is generally the first code run each time a new set of released radionuclides (source term) is evaluated using CRRIS. This is a program for the Preparation of Radionuclide Ingrowth Matrices from User-specified Sources. Based on the radionuclides in the source term supplied as input by the user, the PRIMUS code sets up matrices of decay constants for the calculation of daughter ingrowth by accessing a documented radionuclide decay data base (2). The parent decay and daughter buildup calculations are not handled directly in PRIMUS. Instead, the other CRRIS codes use the PRIMUS data files to make these calculations via appropriate subroutines.

## ANEMOS

The purpose of the ANEMOS code (3) is to estimate concentrations in air and ground deposition rates for Atmospheric Nuclides Emitted from Multiple Operating Sources. ANEMOS calculates the average concentrations around a source for releases that occur over an extended period of time, such as a year.

The calculations made in ANEMOS are based on the use of a modified point-source, straight-line, 22.5-degree sector-averaged, Gaussian-plume atmospheric dispersion model (4). The emission source for an ANEMOS calculation may also be a finite area with the computational grid centered on the centroid of the assumed circular area. Basically, air concentrations and ground deposition rates for an area source are calculated by interchanging the source and the receptor points and approximating the effect of a point source (receptor) on a sector (area source).

For a point or area source, the radiological source term is specified by the name of each parent radionuclide, its particle sizes, its lung clearance classes, and its release rate (activity/s). For a windblown source, the activity density (activity/g) and the fraction of the time that the windblown source remains dry are entered instead of a release rate for each radionuclide to be considered. A model developed by Mills, Dahlman, and Olson (5) is used to calculate the radionuclide release rates.

ANEMOS requires as basic meteorological data a joint frequency distribution of wind direction, wind speed class, and atmospheric stability class over the time period of interest in each run. The basic set of dispersion parameters used in ANEMOS is that developed by Smith (6). These parameters may be modified by the user to account for building wake effects (7). Also, radionuclides may be removed from the plume during transport by dry and wet deposition processes.

The output of ANEMOS is presented for 16 sectors of a circular grid with nuclide-specific air concentrations and ground deposition rates. ANEMOS can calculate both the sector-averaged concentrations and deposition rates at a given set of downwind distances in each sector, and the average of these quantities over an area within each sector bounded by two successive downwind distances. These results, along with a user-supplied description of the source and pertinent information on the parameters used to make the run, are printed for inspection by the user and written to a file for use by other codes in the CRRIS.

## SUMIT

ANEMOS makes calculations for only one release point in a given run and does not account for contributions from other nearby sources. The purpose of SUMIT (8) is to provide for the Systematic Unification of Multiple Interpolated Tables that have been generated by individual runs of ANEMOS. SUMIT combines the results of these individual runs onto a master grid.

The grid system used in ANEMOS is generally set up for computational convenience. This may not, however, be the desired grid from an assessment point of view. As a result, provision has been made in SUMIT to interpolate an ANEMOS output to other locations. These other locations may be [1] a rectangular grid of specified dimensions, [2] a different circular grid of specified dimensions, or [3] specific locations within the original assessment area as designated by the user, e.g., site boundaries.

## RETADD-II

For downwind distances on a regional or continental scale, a Regional Trajectory And Diffusion-Deposition model, RETADD-II (9), has been incorporated into CRRIS. RETADD-II is intended for the estimation of average dispersion over a time period not shorter than a month. RETADD-II operates by calculating tropospheric wind trajectories which advect or carry material away from the source of emission. Superimposed on these trajectories are the vertical and horizontal diffusion rates of the advected material which are a function of travel time. Loss of material by deposition on the ground is also accounted for in RETADD-II, as is radioactive decay and ingrowth of daughter species. Also, both wet and dry plume depletion effects may be considered. The source of emission may be either at ground level or elevated. This option allows for realistic treatment of plume depletion from elevated sources during the initial part of the advection and diffusion process. RETADD-II could also be used to examine individual trajectories.

The calculation of wind trajectories uses a data base of historical upper-air wind data. The premise behind this approach is that a climatology which is typical for the assessment run is chosen from the historical record and used in the simulation. For instance, to predict dispersion patterns during a given time period with RETADD-II, one would use upper-air wind data for that month, season, or year of interest.

## TERRA

The computer code TERRA (10) is used to calculate the Transport of Environmentally Relaxed Radionuclides through Agriculture. TERRA calculates the buildup of radionuclides on the ground surface and in the root zone of irrigated and nonirrigated agricultural soils. From root-zone soil concentrations, TERRA calculates root uptake of radionuclides into food and feed crops. Deposition of radionuclides on exposed surfaces of food and feed crops is calculated by using the ground deposition rate and vegetation-specific interception fractions. Finally, the transport of nuclides to beef and cow's milk from ingested feeds is calculated from feed concentrations. During transport, consideration is made for leaching of nuclides from root-zone soil,

variation in plant physiology and structure, site-specific cattle management practices, and resuspension of ground surface-deposited material.

Ground deposition rates may be input directly into TERRA or they may be obtained from ANEMOS, SUMIT, or RETADD-II. Also, TERRA calculations may be made for either a circular or rectangular grid or a point location, depending on the grid defined by either ANEMOS, SUMIT, RETADD-II, or the user. TERRA writes a file including the data read from the atmospheric model used and the environmental concentrations computed.

## SITE

The environmental transport codes used in CRRIS rely heavily on a variety of transfer factors for estimating the movement of radionuclides between various environmental compartments. It is preferable to have site-specific data for these parameters, but this is seldom possible. As a result, a major effort has been made to acquire a data base of geographically-dependent default parameters describing agricultural production and productivity, climate, and other considerations across the conterminous United States. These parameters are available on a  $1/2 \times 1/2$ -degree longitude-latitude basis and are part of a data base of Specific Information on the Terrestrial Environment, called SITE (11). For a given location, as specified by a longitude-latitude coordinate, TERRA simulates terrestrial transport by incorporating 20 of the SITE parameters into its calculations. The remaining 14 parameters not used by the TERRA code are either used by or are available for use by the other codes of CRRIS.

The agricultural parameters in SITE were derived from the United States county-averaged values given in the report by Shor, Baes, and Sharp (12) which analyzes the 1974 Census of Agriculture. Climatological parameters were interpolated from long-term averages recorded by U.S. weather stations. Demographic parameters describing the fraction of the population in various urbanization categories were available from the 1970 U.S. Census and the population estimate was taken from the 1980 U.S. Census.

## MLSOIL AND DFSOIL

The MLSOIL (Multiple Layer Soil model) (13) code is used to calculate an equivalent ground-surface concentration to be used in computation of external doses. This effective ground-surface concentration is equal to (the computed dose in air from the concentration in the soil layers) divided by (the dose-rate factor for computing dose in air from a plane concentration). MLSOIL uses a five-compartment linear-transfer model to calculate the time-dependent concentrations of radionuclides in the soil resulting from deposition on the ground surface. The model considers leaching through the soil as well as radioactive decay and build-up. The element-specific transfer coefficients used in this model are a function of the soil-water distribution coefficient and environmental parameters. The DFSOIL (Dose conversion Factors for radionuclides in Soil) code calculates dose-rate factors to determine the dose in air per unit concentration at 1 m above the ground for each of the five soil layers used in MLSOIL and the dose per unit concentration from a surface plane source. MLSOIL and DFSOIL used together produce a soil concentration that can be used in ANDROS as an alternative to the soil concentration computed in TERRA.

## ANDROS

The ANDROS computer code (14) is used for the Assessment of Nuclide Doses and Risks with Option Selection. ANDROS couples radionuclide concentrations generated by the other codes in CRRIS with human intake factors and dose and risk factors to arrive at individual or collective doses and health risks.

Doses and risks are calculated for the released radionuclides and their daughter products; results are tabulated for doses and risks due to released radionuclides (including doses and risks from all daughters) or for individual ("exposure") radionuclides. The specific effects available are dose, dose equivalent, risk, risk equivalent, and life lost per premature death. Also, risk, risk equivalent, and life lost due to exposure to radon progeny may be computed in terms of working levels. Intake of food groupings may be calculated for either individuals or populations. Doses and risks for external pathways may be corrected for the time the exposed persons spend indoors, and concentrations in the food pathway can be further decayed due to storage time.

The dosimetric and health-effects data base used in ANDROS was developed using the RADRISK computer code (15,16). The basic life-table methodology used to derive the health effects data in RADRISK was developed by EPA's Office of Radiation Programs. ANDROS reads dosimetric data files that consider internal exposure due to inhalation and ingestion of radionuclides as well as external exposure from photons emitted by radionuclides in the air and on the ground. The dosimetric files used by ANDROS contain organ-specific dose rates (mrad/year per pCi/year) for each pathway. Internal dose commitment values are then tabulated using the fact that the dose rate after either 50 years or 70 years of uniform chronic intake is numerically equivalent to the 50-year or 70-year dose commitment, respectively, for a unit intake. The dose commitment is the total dose over a future period associated with an acute intake. ANDROS uses these values to estimate the risk to the population for somatic effects. Genetic effects in offspring of exposed parents are estimated on the basis of the average total absorbed dose to the gonads assuming a 30-year exposure period. In addition to calculating doses to individual organs, ANDROS also calculates a weighted whole-body dose that combines doses from various organs on the basis of the relative risk for somatic effects as a result of a dose to a specific organ.

ANDROS also requires as input the rates of food consumption. In calculating the ingestion exposure, the user selects the fraction of foods consumed that are grown in home gardens and the fractions that are grown elsewhere inside the assessment area depending on whether the area is classified as rural-farm, rural-nonfarm, or urban. The SITE data base is then accessed to determine production, population, and fractions of the rural/urban classification. The amount ingested from the assessment area is then the maximum of the amount produced or the fraction consumed with the remainder being imported from outside the area. The user can select radionuclide concentrations in imported food ingested. A second option exists to assume that the food within the area is consumed by an unspecified population-at-large regardless of the local population characteristics.

## APPLICATIONS

The CRRIS is a flexible tool for performing radiological assessments. As indicated by Fig. 1, various combinations of the CRRIS codes may be used to tailor an assessment to user needs. For example, if only inhalation and air

immersion doses and risks are to be calculated, then the user may eliminate TERRA and MLSOIL from the assessment. On the other hand, if ground deposition rates and air concentrations are known for a given location, then the user may perform the assessment using PRIMUS, TERRA, MLSOIL, and ANDROS.

Because of the built-in flexibility of the CRRIS, this system of computer codes is ideally suited for determining doses and risks from a variety of decontamination and decommissioning activities. For example, concentrations in air from resuspended radionuclides initially deposited on the ground surface over a large area can be calculated, making the CRRIS suitable for assessing the impact of uranium mining and milling tailings piles. Decay of the parent radionuclides and build up of radiologically-important daughter products is an integral part of the CRRIS. The extensive location-dependent data bases allows the necessary population, agricultural, and other data necessary for the assessment to be obtained relatively easily.

In recent years many sites involved in the handling of radioactive materials have been found to be a source of environmental contamination. Extensive radiological monitoring and remedial action activities have been carried out at some of these locations. The CRRIS is suited to assist in the characterization of potentially-contaminated sites. Both the aerial extent of the potential contamination and its potential health impacts can be estimated if sufficient information is available to characterize the nature of the environmental radionuclide releases. Results of CRRIS computations can thus be used to prioritize monitoring activities.

Exposures to radon emissions from both natural and industrially-enhanced sources are becoming of increasing concern in the U. S. The CRRIS will compute exposures from radon and its progeny in terms of working-level months, and will compute the resulting health risks. By using the RETADD-II code to calculate air concentrations and ground deposition rates, the continental impact of radon releases can be estimated.

## DISCUSSION

The CRRIS codes should be used for assessments of long-term releases to the atmosphere and chronic exposure conditions. None of the transport codes are intended to address accidental or acute releases. The default parameters have been selected with these points in mind. Furthermore, almost all parameters are not associated with a single value, but a distribution of values encountered in the environment. Often these parameter distributions span an order of magnitude or more. The default parameters chosen for CRRIS are mostly median estimates of their distributions and reflect an effort to choose "reasonable" values.

In using CRRIS computer codes, it is the responsibility of the user to determine if the input parameters, options, default parameter values, and codes selected are representative of the problem being addressed. To this end, the documentation of the individual codes addresses questions concerning theory, parameter values, and applicability of the codes to certain types of problems.

## CONCLUSIONS

Because of its modular construction the CRRIS codes provide an alternative to assessment codes which incorporate all calculations into a single program. Each code of CRRIS contains well-documented default parameters for ease of operation, but all defaults may be readily overridden. Additionally, numerous options for transport, dose, and risk calculations and for output display in all codes make CRRIS a very versatile system for performing nuclear assessments, including those related to decontamination and decommissioning activities.

## ACKNOWLEDGEMENT

This research was sponsored by the Office of Radiation Programs, U.S. Environmental Protection Agency (EPA), under Interagency Agreement DW89930177-01-1 under Martin Marietta Energy Systems, Inc., contract DE-AC05-84OR21400 with the U.S. Department of Energy. Although the research described in this article has been funded by the EPA, it has not been subjected to the Agency's required peer and policy review and, therefore, does not necessarily reflect the views of the Agency and no official endorsement should be inferred.

## REFERENCES

1. O. W. Hermann, C. F. Baes III, C. W. Miller, C. L. Begovich, and A. L. Sjoreen, PRIMUS: A Computer Code for the Preparation of Radionuclide Ingrowth Matrices from User-specified Sources, ORNL-5912, Oak Ridge National Laboratory, 1984.
2. D. C. Kocher, Radioactive Decay Data Tables, DOE/TIC-11026, U.S. Department of Energy, 1981.
3. C. W. Miller, C. L. Begovich, O. W. Hermann and A. L. Sjoreen, ANEMOS: A Computer Code to Estimate Air Concentrations and Ground Deposition Rates for Atmospheric Nuclides Emitted from Multiple Operating Sources, ORNL-5913, Oak Ridge National Laboratory (in press).
4. D. Slade (ed.), Meteorology and Atomic Energy - 1968, TID-24190, U. S. Atomic Energy Commission, 1968.
5. M. T. Mills, R. G. Dahlman, and J. S. Olson, Ground Level Air Concentrations from a Tailings Area During a Windstorm, ORNL/TM-4375, Oak Ridge National Laboratory, 1975.
6. F. B. Smith, "A scheme for estimating the vertical dispersion of a plume from a source near ground level," Chapt. XVII, In Proceedings: Third Meeting of the Expert Panel on Air Pollution Modeling, Paris, France, 2-3 October 1972, NATO-CCHS Report 14, North Atlantic Treaty Organization, Brussels, Belgium, 1972.
7. A. H. Huber and W. H. Snyder, "Building wake effects on short stack effluents," In Preprint Volume Third Symposium on Atmospheric Diffusion and Air Quality, American Meteorological Society, Boston, 1976.

8. C. L. Begovich, N. J. DeBliek, J. T. Holdeman, Jr., A. L. Sjoreen, and C. W. Miller, SUMIT: A Computer Code to Interpolate and Sum Single Release Atmospheric Model Results onto a Master Grid, ORNL-5914, Oak Ridge National Laboratory, 1984.
9. B. D. Murphy, S. Y. Ohr, and C. L. Begovich, RETADD-II: A Long Range Atmospheric Trajectory Model with Consistent Treatment of Deposition Loss and Species Growth and Decay, ORNL/CSD-99, Oak Ridge National Laboratory, 1984.
10. C. F. Baes III, R. D. Sharp, A. L. Sjoreen, and O. W. Hermann, TERRA: A Computer Code for Calculation of the Transport of Environmentally Released Radionuclides Through Agriculture, ORNL-5785, Oak Ridge National Laboratory, 1984.
11. C. F. Baes III, R. D. Sharp, A. L. Sjoreen, and R. W. Shor, A Review and Analysis of Parameters for Assessing Transport of Environmentally Released Radionuclides Through Agriculture, ORNL-5786, Oak Ridge National Laboratory, 1984.
12. R. W. Shor, C. F. Baes III, and R. D. Sharp, Agricultural Production in the United States by County: A Compilation of Information from the 1974 Census of Agriculture for Use in Terrestrial Food Chain Transport and Assessment Models, ORNL-5768, Oak Ridge National Laboratory, 1982.
13. A. L. Sjoreen, D. C. Kocher, G. G. Killough, and C. W. Miller, MLSOIL and DFSOIL - Computer Codes to Estimate Effective Ground Surface Concentrations for Dose Computations, ORNL-5974, Oak Ridge National Laboratory, 1984.
14. C. L. Begovich, A. L. Sjoreen, S. Y. Ohr, and R. O. Chester, ANDROS: A Code for Assessment of Nuclide Doses and Risks with Option Selection, ORNL-5889, Oak Ridge National Laboratory (in press).
15. D. E. Dunning, Jr., R. W. Leggett, and M. G. Yalcintas, A Combined Methodology for Estimating Dose Rates and Health Effects from Radioactive Pollutants, ORNL/TM-7105, Oak Ridge National Laboratory, 1980.
16. R. E. Sullivan, N. S. Nelson, W. H. Ellett, D. E. Dunning, Jr., R. W. Leggett, M. G. Yalcintas, and K. F. Eckerman, Estimates of Health Risk from Exposure to Radioactive Pollutants, ORNL/TM-7745, Oak Ridge National Laboratory, 1981.

A COMPOSITE EMPIRICAL AND ANALYTICAL METHOD TO PREDICT THE  
MAGNITUDE OF SURFACE CONTAMINATION IN SEMI-ACCESSIBLE AREAS

C. H. Distenfeld\* and R. J. Vallem\*  
GPU Nuclear Corporation  
P.O. Box 480  
Middletown, Penna. 17057

\*Bechtel National Inc.

ABSTRACT

The purpose of this study was to develop an exposure rate model for the radiation fields within the lower portion of the "A" D-ring in the Three Mile Island Unit 2 (TMI-2) reactor building (RB) containment. The model was based on vertical strings of thermoluminescent dosimeter (TLD) measurements made adjacent to the steam generator. The measurements were made with standard Panasonic personnel dosimeters assembled on plastic strips to produce "TLD strings". The dosimeters were vertically spaced about 30 cm apart. The TLD strings were lowered about 7.8 meters from the elevation 308' platform to the elevation 282.5' basement floor. The strings were placed midway between the vertical axis of the steam generator and cold legs. About 3 meters above the basement floor the string passed a metal platform that had been submerged in primary coolant released during the accident.

Six discrete surface sources were mathematically modeled to closely correspond to the surfaces that exposed the TLD strings. The models used were a disc geometry for the concrete floor; vertical surface cylinders for the cold legs and steam generator; a horizontal surface cylinder for the "J" connection between the cold legs and steam generator; a rectangular surface geometry for the platforms; and a constant term to account for unmodeled sources. The six models were used to calculate responses corresponding to the TLD locations. This resulted in a matrix of 15 equations with six unknowns. An iterative method requiring "best" positive solutions was used to solve the overspecified set simultaneously. The results indicate that the surface activity varies from 5E9 to 6E11 Bq/m<sup>2</sup> (14 to 1670  $\mu$ Ci/cm<sup>2</sup>).

INTRODUCTION

The Three Mile Island Unit 2 (TMI-2) reactor building (RB) has been sufficiently recovered to allow near routine access to the entry and operating levels. Present access was gained by a cyclic process that involved "hands on" characterization and decontamination that further improved general accessibility. The RB basement is excepted from this favorable scenario. General area exposure rates in the RB basement vary from 5 to 25 R/h; too high for direct characterization measurements and exposure reduction means. Semi and totally remote radiation measurement methods had to be developed to gather an understanding of the source terms so that basement exposure reduction engineering could be started.

This paper describes one method that employs direct operator controlled instruments or dosimeters on long strings; and the comprehensive analysis necessary to interpret the data for surface concentrations of radioactive materials. The instrumentation part of the method is well known and will not be considered here.

This study characterizes the basement region of the "A" D-ring between the floor at elevation 282.5', and the platform at elevation 308'. The components of exposure rate due to the otherwise inaccessible basement floor in the "A" D-ring and the cold legs were determined. The cold legs represented the lowest elevation in the reactor coolant system (RCS) where degraded fuel or core debris could accumulate. The material presented will attempt to illuminate the method; it is general and can be used in other semi-accessible locations.

## METHOD

Radiation fields are composed of the sum of all direct or scattered rays from coupled sources. For a given location, the field is a simple sum of all contributions of all component sources. For a second location reasonably close to the first, the radiation field may be different from the first but still due to the same sources. The difference is due to changes in geometry and/or radiation absorption for some or all of the sources between the two locations as follows:

$$\dot{X} = \sum_{i=1}^n g_i S_i$$

where:

$g_i$  is the geometry factor for source  $i$  to location  $k$

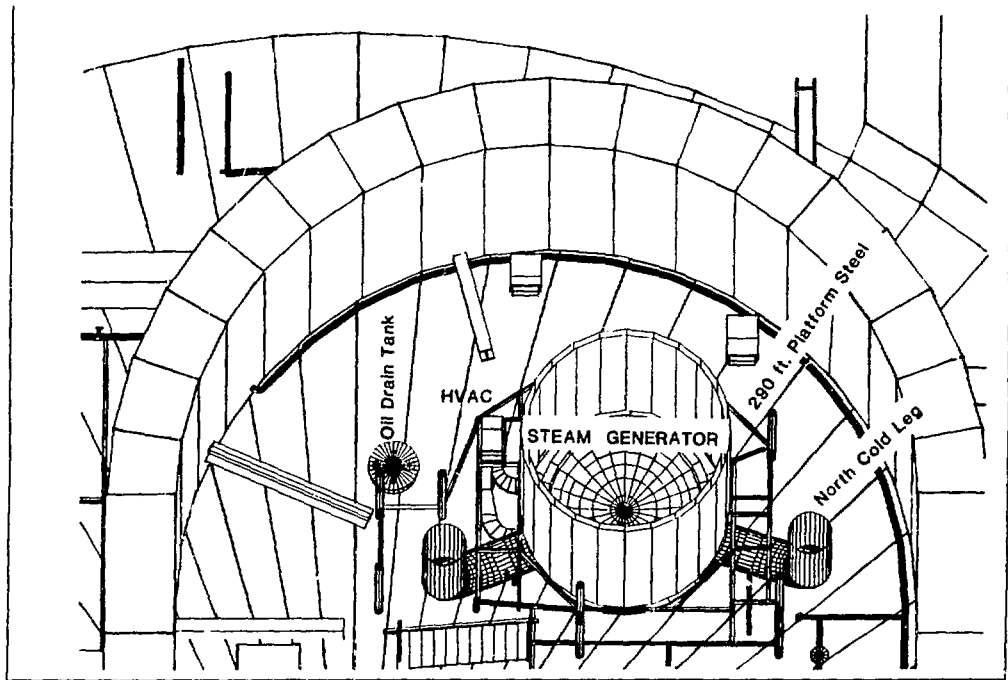
$S_i$  is the source strength for the  $i$ (th) source

$n$  is the number of discrete sources

Radiation measurements,  $\dot{X}_k$ , are selected to bridge the region containing the source components of interest. Geometry factors are calculated for each source to each measurement location. The results are a series of simple linear equations, one for each measurement location corresponding to the number of the unknown sources. Results can be improved by solving an overspecified set of equations where more measurements and corresponding equations exist than unknown sources. Finally, the solution method must reject negative terms, and calculate the "best" solution set for the unknown positive sources.

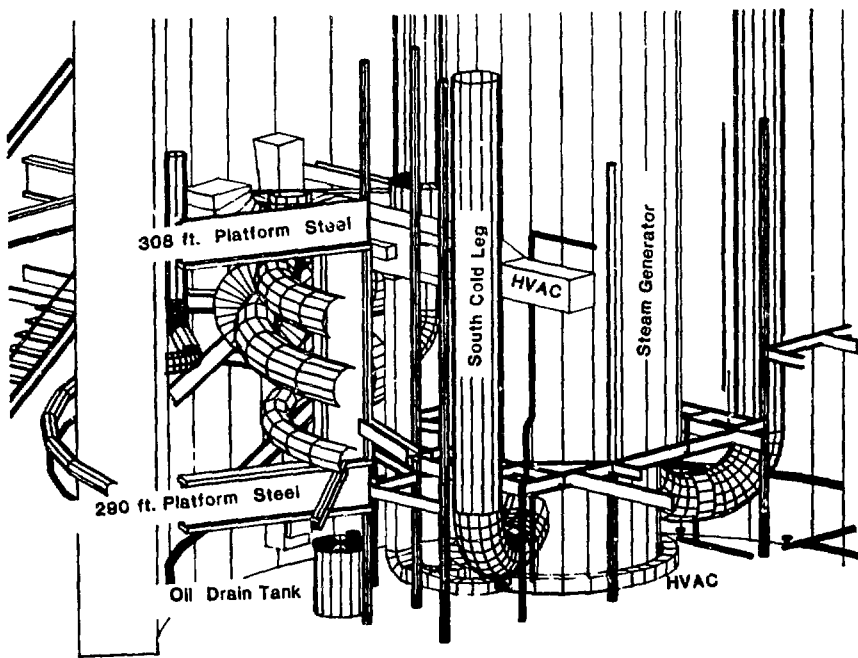
This straightforward approach depends on two less clear ingredients:

- geometry calculations
- unfolding program



Three Meter Vertical Slice Above Floor in "A" D-Ring

Fig. 1



"A" D-ring Elevation

Fig. 2

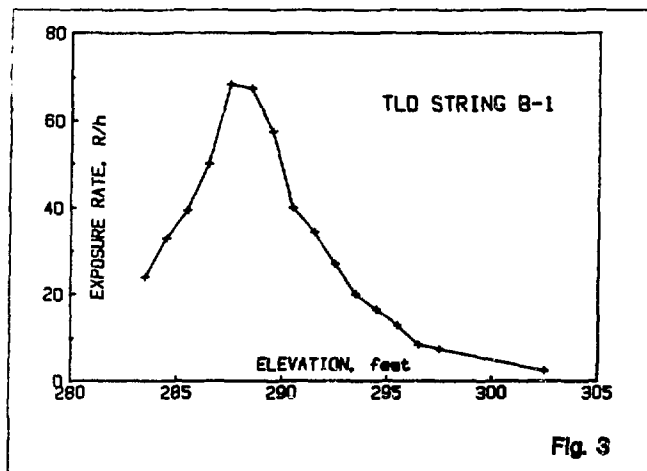
## SOURCE ANALYSIS

The starting point for this work is an analysis of probable sources. This includes all specifiers except intensity such as:

- a. surface shape
- b. surface or volume source
- c. activity distribution
- d. location of source

Figure 1 graphically depicts the TMI-2 RB basement. The horizontal slice covers the space 2.6m (8.5') above the floor; the area that was submerged in water released as a consequence of the accident. Since the cold legs and basement floor were the locations of interest, not all basement structures needed to be considered. Figure 2 is more specific for the sources under consideration. The smaller "I" beams shown near the bottom of Figure 2 support the 290' elevation steel platform that was submerged for nearly 30 months. Submergence should enhance incorporation of fission products into corrosion films, concrete voids, paint films, and may have trapped water in unsealed but closed spaces such as within multifoil mirror insulation.

Figure 3 depicts the exposure rate in the basement near the periphery of the building as measured with a chained string of personnel dosimeters through a floor penetration. The amplitude is thought to be due to a "bathtub ring" phenomenon where an insoluble film may have plated out on surfaces near the air-water interface.



The problem of concern was the source strength of the north and south cold legs and the basement floor in the "A" D-ring. Figure 4 is a presentation of TLD (Reference 1) data expressed in terms of the exposure rate (Reference 2). The TLD strings were suspended from a platform 7.8 meters above the floor tangent to the respective cold leg. The sources assumed to be significant contributors to the measured exposure fields are shown on Table 1. Care was used to include all close sources. A background term was used to account for all unspecified sources

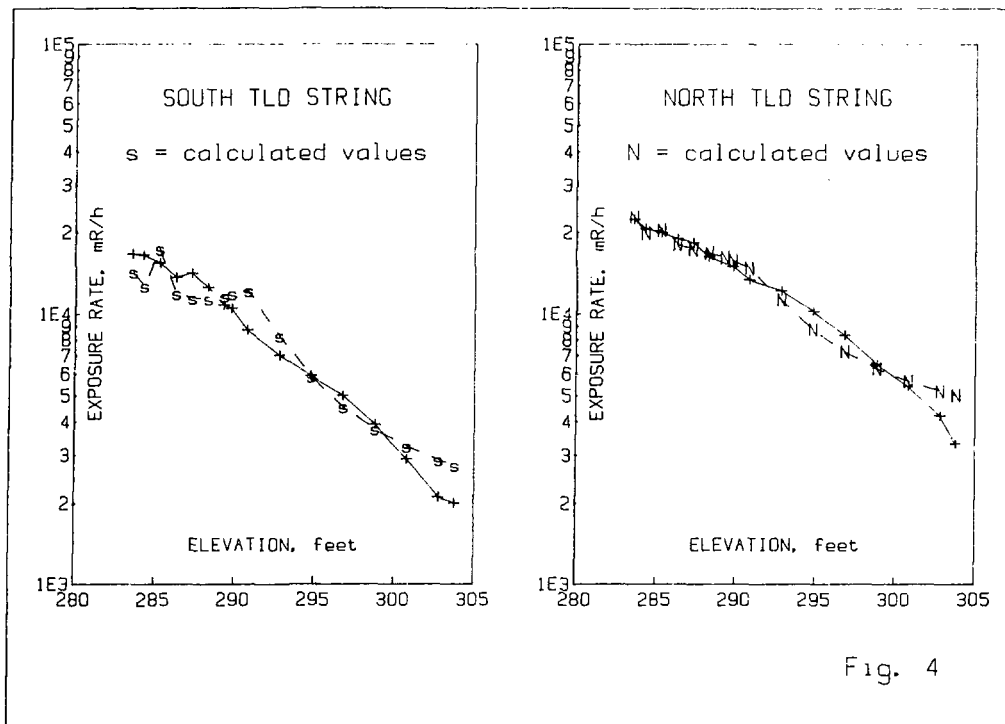


Table 1  
Geometry of Sources

Source	Geometry	Dimensions m	Minimum Offset
			Distance m
Steam Generator	Vertical Cylinder	4.12 Diam. by 2.6	0.91
Cold Leg	Vertical Cylinder	1.12 Diam. by 2.26	0.91
Cold Leg	Horizontal Cylinder	1.12 Diam. by 1.12	0.02
Bathtub Ring	Vertical Cylinder	4.12 Diam. by 0.15	0.91
	Vertical Cylinder	1.12 Diam. by 0.15	0.91
Steel Platforms	Rectangular Area	1.03 by 4.0	0.10
		1.03 by 6.1	
Concrete Floor	Disc	12.2 Diam.	
General	Constant		

## SOURCE GEOMETRY

For simplification, the source geometries were taken to be surface films containing only Cs-137. Abundant measurements indicate Cs-137 accounts for 90-95% (Reference 3 and 4) of the exposure rate. The field of view was between tangents to the surfaces. Geometry factors were calculated for a uniform surface source of  $3.7E8 \text{ Bq/m}^2$  ( $1.0 \text{ } \mu\text{Ci/cm}^2$ ) for disc, cylinder, or rectangular surfaces in view. The surface geometry factor was the integral of line sources 0.01 m wide and the length indicated in Table 1. The expression used to calculate the surface geometry factor was:

$$g_i = \sum_{1-L}^L \Gamma (\theta_2 + \theta_1) / 4\pi * a$$

where

- $\Gamma$  Cs-137 exposure rate constant  $2.48E-9 \text{ mR}^*\text{m}^2/\text{Bq}$   
( $3300 \text{ mR}^*\text{cm}^2/\text{h}^*\mu\text{Ci}$ )
- $a$  is the perpendicular distance between location and line
- $\theta_2, \theta_1$  angle in radians from location to line ends
- $L$  is the peripheral surface, m

## UNFOLDING PROGRAM

The program used to unfold the sources was a composite of well known routines with a modification that allowed only positive solutions. Accuracy of results can be improved by expanding the number of equations beyond the number of unknowns. This produces a rectangular matrix of equation constants with the form:

$$\begin{matrix} k & k & k & k & k \\ g & S & + & g & S & + & g & S & + & g & S & + & S & = & \dot{X} \\ f & f & J & J & c & c & BR & BR & p & p & B & B & k \end{matrix}$$

For the k(th) location, the geometry factors represent:

- $g_f$  ; floor (includes active water, sludge, & concrete)
- $g_J$  ; cold legs (horizontal section, also termed "J" legs)
- $g_c$  ; vertical steam generator and cold leg surfaces
- $g_{BR}$  ; bathtub ring
- $g_p$  ; platform at 290' elevation
- $g_{BK}$  ; background defined as unity

and the corresponding source terms are the required result. Dimensions were selected so that the product of source and geometry terms were increments of the exposure rate.

Solutions for large systems of equations use well established techniques that converge to the result. The Gauss-Jordan elimination method is efficient for large numbers of equations, but suffers roundoff errors due to the large numbers of arithmetic calculations performed with the precision of a mini (IBM-clone) computer. The Gauss-Seidel iterative method avoids this problem. An initial approximate solution is repeatedly refined until the result is acceptable. Roundoff errors do not accumulate since each iteration is a unique solution. Appropriate positive solutions can be extracted by running the program in steps for increased numbers of iterations and tighter tolerances. Appropriate solutions will converge to near constant values for the unknowns if the matrix reasonably corresponds to the physical system of source assumptions, geometry calculations, and radiation measurement precision and location.

Overall accuracy can be improved by solving a larger system of equations than required. The rectangular matrix described earlier can be squared by the usual matrix multiplication of the original and the transpose, and the constant vector by the original matrix. This method is described by Miller (Reference 5) and many others.

An approximation to the BASIC coding used can be found in Reference 5. Miller's "Best Fit" program can be altered by substituting the Gauss-Seidel iterative method for the Gauss-Jordan elimination routines. A more extensive input structure and coding for positive solutions can provide an unfolding system.

## RESULTS

The north string calculation results are included in Table 2. The results are expressed in units of  $\mu\text{Ci}/\text{cm}^2$  except for the background result which is in units of exposure rate ( $\text{mR}/\text{h}$ ). Column 1 provides the number of iterations required to automatically terminate the calculation under given precision constraints; column 2, provides the corresponding solutions. It is clear that the individual surface source solutions smoothly approach final values that are reasonably approximated by the output for the 241 iterations.

Table 2  
North String Results,  $\mu\text{Ci}/\text{cm}^2$ \*

<u>No. of</u> <u>Iterations</u>	<u>Precision</u>	<u>Cold</u> <u>Legs</u>	<u>Floor</u>	<u>Platform</u>	<u>Bathtub</u> <u>Ring</u>	<u>Steam Generator</u> <u>Contamination</u>	<u>Bkg.</u>
31	0.8	5.4	243	13	2417	223	2852
41	0.4	6.0	231	22	1935	241	3013
71	0.2	6.4	221	23.9	1780	254	3157
146	0.1	6.5	216	22	1714	261	3240
241	0.06	6.5	215	21.1	1689	263	3274

\*Background in Units of  $\text{mR}/\text{h}$

The solution sets for both strings were used to calculate the exposure rate for the measurement locations. The graphic results are included on Figure 4. The north string match between calculated and measured values is superior to the south string match.

Model adjustment was not attempted since the result satisfied the need for a preliminary evaluation of surface contamination in an inaccessible area prior to exposure reduction engineering.

#### REFERENCES

1. Dosimeter model 802 AS, Panasonic Company,  
One Panasonic Way, Secaucus, NJ 07094
2. Algorithm by P. Plato, University of Michigan, June 1982 under  
contract with GPU Nuclear Corporation
3. Editors: C. McIsaac, D. Keefer, "TMI-2 Reactor Building Source Term  
Measurements: Surfaces and Basement Water and Sediment,  
"GEND 042, October 1984
4. Editor: C. McIsaac, "Surface Activity and Radiation Field  
Measurements of the TMI-2 Reactor Building Gross Decontamination  
Experiment, "GEND 037, Oct. 1983
5. A. Miller, Basic Programs for Scientists and Engineers  
Sybex, Berkeley, 1981

## IMPLEMENTATION OF DECOMMISSIONING CRITERIA IN THE CONCEPTUAL DESIGN OF THE MRS FACILITY

D. L. Gross, A. D. Wilcox, and S. Huang  
The Ralph M. Parsons Company  
Pasadena, California 91124

### ABSTRACT

The United States Department of Energy (DOE) selected the Ralph M. Parsons Company (RMP) to prepare the conceptual design of the Monitored Retrievable Storage (MRS) Facility. The purpose of this facility is to consolidate and temporarily store spent fuel from civilian nuclear power plants. In addition, it will overpack, handle, and store high-level radioactive waste from non-defense related sources. The Functional Design Criteria (FDC) prepared by Pacific Northwest Laboratories, as well as 10 CFR 72, requires the facility to be designed for decommissioning, with provisions to facilitate decontamination of structures and equipment to minimize the volume of radioactive wastes and contaminated equipment at the time of decommissioning.

Many problems associated with decommissioning a nuclear facility have been identified in recent years and the design for the MRS Facility presents a unique opportunity for RMP to implement decommissioning criteria into the conceptual design of a major nuclear facility. The provisions made in the design to facilitate decommissioning include good housekeeping during operations, controlled personnel access, access for equipment removal, equipment design, installed radiation monitors, adequate work space, installed decontamination systems and areas, control of all effluents, and operational documentation. These topics will be the major points of discussion for this paper.

### INTRODUCTION

A conceptual design for a Monitored Retrievable Storage (MRS) Facility has been developed to provide several integral functions for the Federal nuclear waste management program. The most important functions of the MRS Facility are to reduce the number of miles fuel assemblies must be transported, reduce the complexity of equipment at the repository site, coordinate the waste shipments to the repository, and provide the Federal government the capability to receive and store nuclear waste while the repository is under construction. This paper will describe the plans to decontaminate and decommission the MRS Facility that have been developed during the conceptual design.

The MRS Facility was designed to receive, handle, and temporarily store spent nuclear fuel and high-level waste from civilian nuclear power facilities. The facility has the capability to process 3,600 metric tons of

uranium (MTU) per year with a working storage of 1,000 MTU inside the Receiving and Handling (R&H) Building, and 15,000 MTU in concrete casks or drywells at an onsite storage field. The design has assumed that the annual throughput will consist of 60% by weight PWR fuel assemblies and 40% by weight BWR fuel assemblies. Therefore, the MRS will be able to process 12,400 fuel assemblies per year and store 51,700 consolidated and unconsolidated assemblies in the storage field. The DOE intends to license the MRS Facility to possess a total of 15,000 MTU of spent fuel.

## FACILITY DESCRIPTION

The MRS has been designed to receive all of the currently known and licensed train and truck shipping casks. After the train or trucks come onsite, they are inspected, then directed to the R&H Building, where the shipping cask is removed and mated to a port through which the fuel assemblies are unloaded into the process cells. The R&H Building has two process cells for PWR assemblies and two process cells for BWR assemblies, all of which are interchangeable. These cells will become the most contaminated areas of the facility and will require the most effort when it is time to decommission the facility.

After the assemblies are removed from the shipping casks, they will be placed in lag storage pits until the consolidation equipment is ready to process them. The three PWR or seven BWR assemblies are removed from lag storage and clamped into position for a laser cutting head to remove the end fittings of the assemblies. The assemblies are rotated into a horizontal position where the fuel rods are simultaneously pulled from the assembly skeleton and consolidated into a 12-inch diameter circular bundle. This bundle is then pushed through a port in the wall into a clean stainless steel canister. The assembly skeleton is placed into a low-speed shredder for volume reduction and the ensuing scraps are packaged in 55-gallon steel drums. The filled storage canister is placed in a chamber where it is evacuated, back-filled with an inert gas and has a lid welded on, after which it is checked for leaks and decontaminated by a liquid freon system. These canisters are stored in the canyon cell lag storage vault until they are ready to be shipped offsite to a repository or until they can be loaded into the concrete storage casks or drywells. The sealed concrete storage casks can hold 12 canisters, or up to forty five 55-gallon drums of site-generated radwaste, such as skeleton scraps or other high-level waste and the drywells hold one canister each, or five 55-gallon drums.

The canyon cell contains the large lag storage vaults, the canister welding and testing stations and equipment to handle high-level waste (HLW) packages. When casks containing the HLW packages are received onsite, they will be brought to the R&H Building, where the cask will be unloaded and mated to a port in the canyon cell, and the canister of HLW unloaded, inspected, overpacked, if necessary, and prepared for repository storage. This waste will come from the West Valley facility and will not contain any spent nuclear fuel.

The R&H Building contains a large, remote handled maintenance cell for repairing the consolidation equipment, canyon cell equipment, and other items that have become contaminated during the consolidation process. Two liquid radwaste systems for low-level and high-level waste and a remote handled air filtration cell will operate in conjunction with the process cells and maintenance cell. These areas will become contaminated during the lifetime of the facility and will be of concern when it is time for decommissioning.

## DECONTAMINATION AND DECOMMISSIONING GENERAL PHILOSOPHY

The MRS Facility has been designed for decommissioning at the end of its useful life in accordance with 10 CFR 72.76. Provisions have been made to facilitate the decontamination of structures and equipment, minimize the quantity of radioactive waste and contaminated equipment and to facilitate the removal of radioactive waste and contaminated equipment at the time of decommissioning.

The design philosophy for equipment and utility arrangements for decontamination is based on operational experience and procedures used for other existing radioactive material handling facilities. The capability has been provided to control the buildup of contamination on surfaces and equipment and to decontaminate the onsite-generated waste packages during operations to minimize the contamination levels before decommissioning. The design incorporates single and multiple decontamination stations and agents in the areas where the highest levels of contamination are expected to occur.

### Design Features

Many features have been provided in the design that will assist in decontamination and decommissioning activities. The fuel rod consolidation equipment has its own air filtration system to collect the potential loose radioactive contaminants during consolidation operations and expendable vacuum cleaners are located in the process cells to clean up the loose particulates that are not collected by the equipment filters. Portable decontamination equipment can deliver foamed decon solution, or 10,000 psi water, to high-pressure spray wands in the cells when necessary. All of the process cells, maintenance cells, canyon cells, decon rooms, and radwaste treatment cells have a stainless steel liner plate on the floors and walls that is seal welded without cracks or crevices and the unlined surfaces, including the ceilings, in these areas are covered with a decontaminable protective coating. Other areas of the building, where the potential for the spread of contamination exists, are covered with surfaces that are resistant to decontamination agents. The normal operations of the facility do not permit personnel entry into the cells, but if this becomes necessary for emergency repairs, air locks with breathing air manifolds and anti-contamination stations are provided at the entry points.

The decontamination stations located throughout the R&H Building are equipped with an aqueous cleaning solution manifold that is connected to either the central decontamination solution makeup system or a portable decontamination system, depending on the location. Curbs are provided to contain the liquids and with sumps that are connected directly to the liquid radwaste treatment system.

The proper application of the decontamination equipment and procedures towards good housekeeping during the facility's lifetime will help to minimize the exposure levels to the decommissioning personnel. All of the equipment in the process cells has been designed in modules for ease of removal by remote manipulators and cranes for decon and maintenance during normal operations as well as for permanent removal at decommissioning. Detailed operational documentation will be maintained so that decommissioning personnel will be familiar with successful techniques and the extent of contamination in various areas.

The decommissioning work will not begin until detailed administrative controls and procedures are developed and approved. The procedures will

include the specific techniques to be used, delineation of responsibilities, safety considerations, as well as normal and emergency plans for all subtask levels. All of these steps will be developed so that radiation exposure levels for onsite and offsite personnel will be kept As Low As Reasonably Achievable (ALARA).

### Decommissioning Options

The four decommissioning methods that have been considered for the MRS Facility are DECON, SAFSTOR, ENTOMB, and DISMANTLE. The DECON method involves the immediate removal of all radioactive material down to residual levels that are acceptable for release of the property for unrestricted use. The SAFSTOR method calls for immediate partial decontamination followed by a period of time that will allow short-lived radionuclides to decay to acceptable levels for release of the facility to unrestricted use. The third alternative is ENTOMB, or entombment of the entire structure after some cleanup and decontamination has been completed. The DISMANTLE method involves the complete decontamination and dismantling of the entire facility.

Because the basic design philosophy for decommissioning of the MRS Facility was to restore it to unrestricted use as soon as possible and at the lowest cost, ENTOMB and DISMANTLE were not considered as feasible options. SAFSTOR, which is basically deferred decontamination, was also rejected because the ongoing maintenance and monitoring costs would exceed the DECON costs. The DECON method has been chosen for developing a decommissioning plan for the facility.

### Decommissioning Plan

Decommissioning has been organized into four phases of work. Phase I includes the decommissioning of the contact handled transuranic (CHTRU) waste storage facility, the concrete storage casks or drywells, transportable metal casks and the onsite transporters. Concurrently with the emptying of the storage field and the shipment offsite of the consolidated fuel assemblies, and related radioactive waste packages, the in-cell equipment will be decommissioned and the cells decontaminated. The second phase includes the decommissioning of the storage facilities. The R&H Building will be used for utility monitoring and laboratory support. Decommissioning of the R&H Building will also begin. During Phase III, the remainder of the protected area will be decommissioned as will the R&H Building radwaste systems and laboratories. The final phase involves the releasing of the entire MRS Facility for unrestricted use.

### DECOMMISSIONING PROCEDURES

#### Concrete Storage Casks

Decommissioning of the storage casks or drywells is expected to be relatively simple. All materials stored are packaged in sealed canisters or drums that have been decontaminated prior to insertion into the cask/drywell. The sealed concrete storage casks will be decommissioned according to the following procedures. The cask design provides the capability to monitor the interior air space and the temperature of the liner. Before a cask is removed from the storage field, the cask will be monitored and the results compared to historical data to determine if any special precautions will need to be observed during the unloading operation. After the cask gas samples have been

analyzed and approval to move and unload the casks has been given, the cask transporter moves the cask to the transfer/discharge corridor within the R&H Building and the outer lid is removed. The cask is then moved into the loadout and decon room where it is mated to a port in the floor of the overpack/discharge area of the canyon cell and the interior cask shielding plug is removed. The waste packages are then removed, inspected and loaded into a repository shipping cask for shipment offsite. The loadout port is then closed and the interior of the cask is surveyed for contamination. If the surface contamination level is acceptable for unrestricted release of the cask, the shielding plug is put back in place and the cask is returned to the field for permanent storage. If the contamination level is too high for unrestricted release, the cask is moved to a temporary decon station in the transfer discharge corridor, where it is decontaminated. If an acceptable level cannot be achieved, the cask will be destroyed and the debris packaged and shipped to a waste disposal site. After all of the casks are emptied and returned to the field or destroyed, the cask transporter will be surveyed, decontaminated, if required, and shipped offsite.

### Drywell

The drywell waste storage area will be decommissioned according to the following procedures. As in the case of sealed concrete storage casks, the drywell storage design provides the capability to monitor the interior space for gaseous releases as well as the temperature casing. Before waste package removal, these measurements will be taken and compared with historical data to see if any special precautions should be taken during the unloading operation. If special precautions are required, a temporary enclosure with a self-contained HVAC system and with special personnel protective systems will be provided during the drywell unloading. After approval to open the drywells, the shielded transporter moves the waste package and shield plug to the loadout and decon room of the R&H Building where it is mated to a loadout port in the floor of the overpack/discharge area in the canyon cell. The waste package is removed, inspected, and placed in a repository shipping cask for transfer offsite.

After the waste package is removed, the transporter and shield plug are surveyed. If an acceptable level of radiation is measured, the shield plug is returned to a drywell that has been surveyed and decontaminated, if necessary. If the shield plug has been found to be above acceptable levels for unrestricted use, it is decontaminated in a temporary station. If the shield plugs or drywells cannot be decontaminated to a level that is acceptable for unrestricted use, they will be packaged and shipped to a waste disposal site. The drywells that are acceptable will be sealed with the plug in place and covered with dirt. After all of the drywells are emptied and decontaminated, if necessary, the transporter will be decontaminated and shipped offsite.

### Transportable Metal Cask

The transportable metal cask decommissioning procedures are described in the following. These casks are spent fuel shipping containers which have been sent to the MRS Facility for storage from various utilities. These casks are capable of having gas samples taken during their storage lifetime. Before removal for shipment offsite, the cask is monitored and the results compared with the historical data to see if any special precautions need to be taken during final cask inspection.

After approval to move the cask, it is transported into the cask unloading and decon room of the R&H Building, where it is pressurized and leak tested. If these tests are passed, the cask is shipped to the repository. If the tests are failed, the cask is unloaded and the fuel is placed in a new cask for shipment offsite. The old cask is then repaired for future use at a designated repair shop.

#### CHTRU Storage Facility

The CHTRU Storage Facility will be decommissioned as described herein. This facility is a compartmentalized, near-surface structure with the capability for monitoring the interior environment of each compartment. Before waste unloading and final packaging, a temporary structures with a self-contained HVAC system will be placed over the facility. Each compartment will be monitored and the results compared with historical data to determine whether any special precautions need to be taken before removing shield plugs.

After the compartment is permitted to be opened, the shield plug is removed and the pallets containing the waste containers are removed, surveyed, and if required, decontaminated. The containers and pallets are prepared for shipping inside the temporary structure and sent to a waste disposal site.

Once the compartment is empty, it is surveyed and decontaminated, if necessary, to a level acceptable for unrestricted release. The decontamination fluids are routed to the R&H Building liquid radwaste system by tank truck.

#### R&H Building

The R&H Building will require the greatest decommissioning effort. A proposed work sequence is described in the following. As previously discussed, the decommissioning of the shielded process cells and their equipment will be done during the waste loadout phase--Phase I. The decontamination operations will be accomplished by using the existing systems, with the overpack/discharge areas in the canyon cells as the primary loadout stations.

Preliminary decontamination of the consolidation, secondary scrap volume reduction, lag storage rack, and other shielded process cell equipment is accomplished remotely within the shielded process cell. After initial decontamination, all equipment is transferred to the remote handled equipment maintenance room, decontaminated, volume reduced, surveyed, packaged, and outloaded either for unrestricted use or as low-level or high-activity waste.

Upon completion of the initial equipment decontamination and equipment removal, preliminary shielded process cell decontamination is initiated. This consists of remote vacuuming and liquid decontamination of cell surfaces. All decontamination solutions generated are routed to the high-activity radwaste treatment system. After the preliminary decontamination operation, the cell interior is surveyed, and a final contact decontamination operation is accomplished until the cell interior reaches acceptable levels as defined in 49 CFR 173.443.

After all contaminated equipment has been decontaminated and packaged, the remote handled equipment maintenance room equipment is decontaminated, packaged, surveyed, and outloaded for unrestricted use or as high-activity waste or low-level waste.

Because all spent fuel handled within the in-cell lag storage vault is contained within sealed canisters, it is not anticipated that this area will be highly contaminated. Consequently, manned entry for final decontamination is assumed for this area. To gain easy access to the storage vault area, a portion of the vault roof will be removed. All areas within the vault are surveyed and, if required, decontaminated by contact methods. All decontamination solutions generated are routed to the low-level radwaste treatment system.

Upon completion of the MRS Facility loadout phase, the decommissioning of the weld/test/decon and the overpack/weld/discharge areas is initiated. Similar to the shielded process cells, preliminary decontamination of the equipment is accomplished remotely before removal to the remote handled equipment maintenance room. Final volume reduction, surveying, and packaging of this equipment are accomplished in the remote handled equipment maintenance room. All decontaminates generated are routed to the high-activity radwaste treatment system. Final decontamination is accomplished as described for the shielded process cells. The remote handled equipment maintenance room and service gallery are decontaminated to acceptable levels, using the existing decontamination stations and systems. All decontamination solutions generated are routed to the high-activity radwaste system.

Other equipment associated with the in-cell operations, cranes, manipulators, etc., are decontaminated to acceptable levels, dismantled when possible, and removed from the site. All other major facility equipment is surveyed, deconned if necessary, and removed from the site. All other systems, piping, electrical, etc., are surveyed and, if found to be at acceptable levels, sealed and left in place. If not found to be at acceptable levels, they are removed and shipped offsite.

All high-level radwaste streams are contained within equipment or piping systems. Except for leaks or equipment failure, the interior of the high-activity radwaste treatment cell should not become contaminated. It is assumed that, when spills are encountered during operations, the area is remotely decontaminated to an acceptable level. Before equipment, piping, and component removal, the systems are flushed with decontamination solution and recycled through the system until manned entry is permissible. Further decontamination of the system components and cell area is accomplished by contact methods. The equipment and piping are removed where possible, volume reduced, surveyed, packaged within the cell, and shipped to a waste disposal site. All decontamination solutions generated during the final decontamination phase are routed to the low-level radwaste treatment system.

Before disassembly of the facility HVAC exhaust system, a temporary HVAC exhaust system (HEPA filtration) is installed to service the low-level radwaste areas and the analytical laboratory. The facility system will continue to operate until little, if any, radioactive particulate buildup occurs in the first- and second-stage HEPA filters. The facility exhaust system is deactivated and the ductwork and accessories between the shielded process cells, remote handled equipment maintenance room, service gallery, weld/test/decon area, and overpack/weld/discharge area are decontaminated in place and dismantled. The dismantled items are surveyed and, if required, the dismantled items are further decontaminated, packaged, and surveyed in the low-level solid radwaste area before shipment for disposal. Filter plenums are decontaminated in place, and will remain in the MRS Facility. The remaining facility HVAC exhaust and supply systems are surveyed and decontaminated to acceptable levels, if required. If acceptable levels are

achieved, the equipment, ductwork, and components remain in place. If acceptable levels cannot be achieved, the items are removed and shipped offsite.

All radwaste resulting from the decontamination operations will be routed to and treated in the existing low-level radwaste system and the analysis of samples and swipes will be performed in the analytical laboratory. Therefore, these two areas will remain operational until all other areas of the MRS Facility are decontaminated to an acceptable level. Before decommissioning of these final two areas, a survey of all areas and remaining components of the Receiving and Handling Building will be conducted to ensure their release for unrestricted use. At the same time, the low-level liquid and solid radwaste and the analytical laboratory equipment will be flushed until the contamination level of the effluent is below acceptable standards. When this level has been achieved, the analytical laboratory will be disassembled, and the equipment decontaminated, if required, and packaged for disposal in the low-level solid radwaste area.

The low-level liquid radwaste equipment is flushed with decontaminants and the liquid processed in the cementing station of the low-level solid radwaste treatment area. Final decontamination of the equipment and area is accomplished by use of portable decontamination equipment. The temporary HVAC exhaust system is operated until little or no radioactive particulation buildup occurs on the HEPA filter. A survey is conducted to ensure that hands-on disassembly can be accomplished with acceptable personnel exposure. When found to be acceptable, the low-level solid radwaste equipment and temporary HVAC system are disassembled, packaged, and shipped to waste disposal.

Decommissioning of the non-process elements of the MRS Facility will be conducted concurrently with other decommissioning activities, with the Receiving and Handling Building providing services to support the general decommissioning. Therefore, the decommissioning operation must be completed before the decommissioning of the Receiving and Handling Building radwaste system and analytical laboratory.

The decommissioning of the site requires a complete survey of all areas within the protected area fence. If any contamination is found, the soil will be removed, packaged in drums, and shipped to disposal.

## SUMMARY

It is estimated that the decommissioning of the MRS Facility will cost in 1985 dollars, \$79 million for the sealed storage cask concept or \$73 million for the drywell concept. This estimate is based on data from the conceptual design as well as the decommissioning plan outlined in this report. The final decommissioning of the R&H Building will take 4.8 years after all the stored waste has been shipped from the site. The decommissioning plan, schedule, and costs are based on leaving structures and storage components in place after the removal of contaminated materials and equipment.

## REFERENCES

All the information presented in this report was taken from the Conceptual Design Report of the Integral Monitored Retrievable Storage Facility, the Ralph M. Parsons Company, Contract No. DE-AC06-84RL10436, for the U.S. Department of Energy, July 1985.

A SIMPLIFIED PATHWAY ANALYSIS APPROACH  
FOR ESTABLISHING LIMITS FOR SOIL CONTAMINATION

John E. Till, Ph.D.  
Radiological Assessment Corp.  
Neeses, S.C. 29107

Mark S. Whittaker, C.H.P.  
Chem-Nuclear Systems, Inc.  
Columbia, S.C. 29210

Robert E. Moore, Ph.D.  
Moore Technical Associates, Inc.  
Oak Ridge, TN. 37830

ABSTRACT

Release from a byproduct material license requires demonstration that radioactive contamination has been removed to the extent practicable. An approach requiring only minimal data inputs was developed which accounts for the potential pathways of exposure from soil contamination and yields a conservative estimate of the annual dose. A limit for soil contamination meeting the extent practicable requirement can then be calculated by assuming an annual dose limit. A soil contamination limit, developed by this method, was applied by Chem-Nuclear Systems, Inc. to the decommissioning of an irradiation facility and was accepted by the NRC.

INTRODUCTION

Nuclear Regulatory Commission (NRC) byproduct material licensing regulation, 10CFR30.36, requires the licensee to "Remove radioactive contamination to the extent practicable" prior to termination of the license or release of the facility for unrestricted use. In situations when soil contamination has occurred, the guidance provided by the NRC concerning release limits is often insufficient to determine the practicable level of decontamination. For these cases, a limit for soil contamination must be established.

REGULATIONS AND DOCUMENTED METHODS FOR DERIVING ACCEPTABLE LEVELS OF DECONTAMINATION

Limited guidance is available on the unrestricted use of contaminated facilities and equipment and the decommissioning of certain nuclear facilities (AEC 1974, NRC 1976, Federal Register 1981, Federal Register 1983). Although criteria have been developed for soil contamination, Kennedy et al. (1983) concluded that a comparison of soil standards was extremely difficult since different units and standards of exposure were used. It was also reported that most of the criteria were derived for specific rather than general applications.

Several pathway analysis methodologies have been developed for deriving acceptable concentrations in soil for a mixture of radionuclides (Eckerman and

Young 1980, Healy 1974, Healy et al. 1979, Kennedy et al. 1979, and Napier 1982). Kennedy et al. (1983) prepared an assessment of allowable residual contamination levels in soil for decommissioning of the Shippingport reactor. The calculations were based on unrestricted use of the site and derived "Allowable Residual Contamination Levels" for radionuclides remaining in the soil. Their approach was based on a scenario exposure pathway analysis and compliance with an annual dose limit for a member of the public. The methodology presented here uses a pathway analysis approach similar to that selected by Kennedy et al. (1983); however, special consideration is given to the exposure buildup due to scatter of high energy gammas in soil.

## DEVELOPMENT OF METHODOLOGY

### Basic Assumptions

#### Limit of Exposure

An allowable dose limit of 25 mrem/a whole body effective dose equivalent was chosen as the basis for decontamination levels derived in this study. This limit was selected since it is well below standards recommended by the ICRP, 500 mrem/a (ICRP 1979a), and NCRP, 170 mrem/a average population dose (NCRP 1971), and below the values used by the U.S. NRC, 500 mrem/a (NRC 1981), in developing its methodology for classification of low-level radioactive waste. This exposure rate is similar to that already recommended by the U.S. EPA for the nuclear fuel cycle, 25 mrem/a (40CFR190), and the U.S. NRC for effluents, 25 mrem/a (NRC 1981).

#### Pathways Considered

Since the desired decontamination end point is immediate unrestricted use, all major pathways that potentially contribute to dose must be considered. The pathways include consumption of food grown on the contaminated soil, surface exposure, inhalation of resuspended materials, and drinking water pumped from a well drilled through the contaminated strata of earth. Although other minor pathways exist, only these pathways were incorporated into this assessment. Initial dose calculations are made for an individual residing permanently at a site, growing food products on the contaminated soil, and drinking water from a well drilled through the contaminated strata of earth. Dose equivalents are estimated assuming a unit concentration of radionuclides in the soil. Once a hypothetical dose has been calculated for a unit concentration in soil, an allowable level of contamination can be determined by applying the limit of exposure to the resident of 25 mrem/a.

Calculations are made assuming the individual establishes residence at the site immediately following decontamination. Initial doses are determined for a one acre area uniformly contaminated with one pCi/g of the appropriate radionuclide(s). The depth of contamination is assumed to be 15 cm for the terrestrial food pathways. Doses for the surface exposure are determined for depths varying between 10-100 cm. The drinking water pathway is evaluated assuming uniformly contaminated soil is in equilibrium with soil water. Other assumptions used in the calculation are that the resident remains on the site 100% of the time and obtains 100% of his sustenance, i.e. plant diet, milk, meat, and water from the contaminated area.

These initial calculations, known to be conservative, identify the important pathways that could exist for the soil contamination. Once important pathways are known, more realistic estimates of dose are obtained by applying modifying factors that account for the size and shape of the source

as well as the possible limited use of the area once it becomes unrestricted.

In order to keep the methodology as simple as possible, neither radioactive decay nor environmental removal are considered. This assumption increases the conservatism of the approach for short lived radionuclides or those possessing high environmental mobility.

#### Description of Methodology

The pathway analysis in this assessment incorporates the use of several models to calculate an annual effective dose equivalent. The total effective dose rate,  $D_{tot}$ , is defined as the sum of the effective dose rates from food, external exposure, inhalation, and drinking water.

The effective dose equivalent is calculated by assuming a unit contaminate concentration in the soil. Once this value is known, the allowable soil concentration can be determined by applying the 25 mrem/a standard accepted as the basis for this study.

#### Ingestion of Contaminated Food

The ingestion of food products produced on the contaminated soil is broken into four types of food: leafy vegetables, produce, milk, and meat. Calculation of the effective dose equivalent for each of these food products is performed (NRC 1977). The total effective dose equivalent from ingestion of contaminated food grown on the contaminated soil is calculated with the following expression.

$$D_{ing} = (L_p + L_l + L_m + L_f) DF_{ing} \quad \text{Eq. 1}$$

where,

- $D_{ing}$  = the total dose due to ingestion from all pathways (mrem/a),
- $L_p$  = the annual ingestion of radionuclide from consumption of produce (pCi/a),
- $L_l$  = the annual ingestion of radionuclide from consumption of leafy vegetables (pCi/a)
- $L_m$  = the annual ingestion of radionuclide through consumption of milk (pCi/a),
- $L_f$  = the annual ingestion of radionuclide through consumption of beef (pCi/c), and
- $DF_{ing}$  = the dose factor for ingestion of radionuclide (mrem/pCi).

#### External Exposure

The effective dose equivalent due to exposure to contaminated ground is calculated by the equation

$$D_{ext} = t C_v R_f DF_{ext} \quad \text{Eq. 2}$$

where,

- $D_{ext}$  = the external dose equivalent (mrem/a),
- $t$  = the depth of contamination (cm),
- $C_v$  = volume concentration (pCi/cm<sup>3</sup>),
- $R_f$  = reduction factor (unitless), and
- $DF_{ext}$  = effective dose equivalent conversion factor (mrem/a per pCi/cm<sup>2</sup>).

The reduction factor,  $R_f$ , is the ratio of the effective surface activity,  $C_a$ , at the surface of an infinite slab of soil with a uniform

concentration of contamination to the surface activity,  $C_s$ , that would result if all of the radionuclides in the slab were residing on the surface, i.e.  $C_s = C_{vt}$ .

The effective surface activity,  $C_a$ , is calculated by integrating the following differential equation, similar to that reported by Cember (1969):

$$dC_a = C_v B e^{-ux} dx \quad \text{Eq. 3}$$

where,

- $dC_a$  = the infinitesimal contribution to the effective surface activity of a slab resulting from radioactivity in the layer  $dx$  at a depth  $x$  ( $\text{pCi/cm}^2$ ),
- $u$  = linear attenuation coefficient ( $\text{cm}^{-1}$ ),
- $B$  = buildup factor (unitless), and
- $C_v$  = volume concentration of uniformly distributed activity in slab ( $\text{pCi/cm}^3$ ).

If  $B$  is assumed to be a linear function of  $x$  over a limited range of  $x$ , then  $B = a x + b$ . Integrating Eq. 3 over the limited range from  $x_1$  to  $x_2$  results in:

$$C_a = C_v a \left[ x_1 e^{-ux_1} / u - x_2 e^{-ux_2} / u + e^{-ux_1} / u^2 - e^{-ux_2} / u^2 \right] + C_v b \left[ e^{-ux_1} / u - e^{-ux_2} / u \right] \quad \text{Eq. 4}$$

Adding together a series of  $C_a$  values from  $x = 0$  to  $x = t$  gives a value for the total effective surface activity.

Based on the methodology described above, the effective dose equivalents resulting from external exposure are calculated. This calculation assumes the individual resides continuously and unshielded on the contaminated area. Modifying factors are then incorporated to adjust for the conservatism of the model. Inhalation

Inhalation of radionuclides could occur from material resuspended to the atmosphere and taken in by the resident. The importance of this pathway depends significantly on the type of soil, vegetation cover, and average wind velocity. The following expression was used to estimate the effective dose equivalent from soil uniformly contaminated with one  $\text{pCi/g}$ .

$$D_{inh} = C_v A U_b DF_{inh} / p \quad \text{Eq. 5}$$

where,

- $D_{inh}$  = the effective dose equivalent from inhalation ( $\text{mrem/a}$ ),
- $C_v$  = the volume concentration of uniformly distributed radionuclide in soil ( $\text{pCi/cm}^3$ ),
- $A_1$  = the atmospheric loading of soil due to resuspension ( $\text{g/m}^3$ ),
- $U_b$  = the annual breathing rate for the resident ( $\text{m}^3/\text{a}$ ),
- $DF_{inh}$  = the dose equivalent conversion factor for inhalation of radionuclide ( $\text{mrem/pCi}$ ), and
- $p$  = the soil density ( $\text{g/cm}^3$ ).

#### Drinking Water

The contribution to effective dose equivalent can be conservatively

estimated by assuming a saturated condition exists in the contaminated soil and that drinking water is taken from the saturated zone. The fraction of radionuclide that is available to the water is estimated with the following expression:

$$C_{dw} = 1000 C_v / K_d p \quad \text{Eq. 6}$$

where,

- $C_{dw}$  = the concentration of radionuclide in ground water (pCi/l),
- $C_v$  = the volume concentration of radionuclide in soil (pCi/cm<sup>3</sup>),
- $K_d$  = the distribution coefficient (ml/g), and
- $p$  = the soil density (g/cm<sup>3</sup>).

The dose equivalent due to drinking contaminated water is therefore

$$D_{dw} = C_{dw} U_{dw} DF_{ing} \quad \text{Eq. 7}$$

where,

- $D_{dw}$  = the annual effective dose equivalent from consumption of drinking water (mrem/y),
- $C_{dw}$  = the concentration of radionuclide in ground water (pCi/l),
- $U_{dw}$  = the annual consumption of drinking water (l/a), and
- $DF_{ing}$  = the effective dose equivalent conversion factor for ingestion (mrem/pCi).

## APPLICATION OF METHOD

### Contamination Event

An irradiation facility, employing approximately 50,000 Ci of <sup>60</sup>Co in stainless steel encapsulated sources, became contaminated when one or more of the sources ruptured in the water shielding tank. While attempting to decontaminate the shield tank, a leak in a transfer line released approximately 5,800 liters of contaminated water onto the floor of the building. This contaminated water flowed over the floor and seeped through wall mortar joints, contaminating the walls, floor, and soils beneath the floor and adjacent to the foundation walls.

### Dose Conversion Factors

The dose factors for external exposure ( $DF_{ext}$ ) were calculated by Kocher (1983). The dose factors for inhalation and ingestion are taken from ICRP publication 30 (ICRP 1979 b). The dose conversion factors for <sup>60</sup>Co and associated parameters assumed for the calculation are listed in Table 1.

Table 1 List of Metabolic Parameters and Dose Factors Selected for <sup>60</sup>Co

<u>Inhalation</u>	
Fraction from GI tract to blood ( $f_1$ )	= 0.05
Inhalation dose factor ( $DF_{inh}$ )	= 3.31 E-5 (mrem/pCi)
<u>Ingestion</u>	
Fraction from GI tract to blood ( $f_1$ )	= 0.05
Ingestion dose factor ( $DF_{ing}$ )	= 1.02 E-5 (mrem/pCi)
Clearance class	= "W"
<u>External Exposure</u>	
Exposure rate per areal contamination	= 2.31 mrem/a per pCi/cm <sup>2</sup>

## Results

### Ingestions of Food

Assuming soil uniformly contaminated to a depth of 15 cm at a concentration of one pCi/g, the calculated effective dose equivalent from food using the methodology described above is  $8.54 \times 10^{-2}$  mrem/a. The contribution to total effective dose equivalent from each pathway is shown in Table 2.

Table 2 Contribution to Effective Dose Equivalent from Food Pathway

Type of food	Dose Equivalent (mrem/a)	% of Total
Produce	$6.63 \times 10^{-2}$	77
Leafy vegetables	$6.78 \times 10^{-3}$	8
Milk	$1.32 \times 10^{-3}$	2
Meat	$1.10 \times 10^{-2}$	13
Total	$8.54 \times 10^{-2}$	100

### External Exposure

The calculated effective dose equivalent to an individual residing on soil layers of increasing thicknesses each uniformly contaminated at one pCi/g varies from 26.9 to 92.7 mrem/a. The dose equivalents for each of the layers is shown in Table 3.

Table 3 Calculated Effective Dose Equivalents from External Exposure to an Individual Residing on Soil Uniformly Contaminated at One pCi/g to Various Depth

Depth (cm)	Effective dose equivalent* (mrem/a)
0-10	26.9
0-15	37.8
0-20	47.1
0-30	61.1
0-40	68.9
0-50	74.6
0-70	79.0
0-100	92.7

\*These effective dose equivalents were calculated using equation 2 where  $C_v = 1$  pCi/g  $\times 1.3$  g/cm<sup>3</sup> (to account for soil density).

### Inhalation

Assuming an atmospheric loading factor of  $1 \times 10^{-4}$  g/m<sup>3</sup> (Napier 1984) and a breathing rate of 8030 m<sup>3</sup>/a (Rupp 1979), the effective dose equivalent for one pCi/g contamination of <sup>60</sup>Co is  $2.7 \times 10^{-5}$  mrem/a. This conservatively assumes the individual continuously breathes the contaminated air.

## Ingestion of Water

Assuming a value for  $K_d$  of 1600 ml/g (Seeley and Kelmers 1984) and a consumption rate of 511 l/a (Rupp 1979), a one pCi/g concentration of  $^{60}\text{Co}$  in soil yields an effective dose equivalent of  $3.1 \times 10^{-3}$  mrem/a.

## Analysis of Pathways

The total dose and relative importance of all pathways can now be determined using the effective dose equivalents calculated for one pCi/g contamination in soil. These data are summarized in Table 4. These data clearly point out that surface exposure is the only pathway of significance; therefore, the calculations of allowable soil concentration and modifying factors which reduce the conservatism of the estimated effective dose equivalent pertain to only this pathway.

Table 4 The Total Effective Dose Equivalent and Percentage Contribution from all Pathways Assuming A Soil Concentration of One pCi/g

<u>Pathway</u>	<u>Effective Dose Equivalent</u>	<u>Contribution</u>
Food	$8.5 \times 10^{-2}$	0.30%
Surface exposure (10 cm)	$2.7 \times 10^{+1}$	99.97%
Inhalation	$2.7 \times 10^{-2}$	0.01%
Drinking Water	$3.2 \times 10^{-3}$	0.01%

## CALCULATION OF ALLOWABLE SOIL CONCENTRATIONS

### Unshielded and Continuous Exposure

Concentrations of  $^{60}\text{Co}$  in soil were calculated using the data in Table 3 showing effective dose equivalents as a function of soil depth and assuming a limiting exposure of 25 mrem/a. These values are shown in Table 5. The effective dose equivalents listed assume exposure is continuous and unshielded.

Table 5 Uniform concentrations of  $^{60}\text{Co}$  in Soil to Various Depths to Produce an Effective Dose Equivalent of 25 mrem/a Assuming Unshielded and Continuous Exposure

<u>Depth (cm)</u>	<u>Concentration in soil (pCi/g)</u>
0-10	0.93
0-15	0.66
0-20	0.53
0-30	0.40
0-40	0.36
0-50	0.33
0-70	0.32
0-100	0.27

### Description of Exposure Scenario

Due to the dominance of the surface exposure pathway, the scenario used in this study assumes that an individual establishes residence on the site immediately following decontamination. Since the soil containing the  $^{60}\text{Co}$  will be covered with uncontaminated dirt as a part of the cleanup, the most

likely mode of exposure would occur from construction of a home with a basement. The earth beneath the basement is contaminated with  $^{60}\text{Co}$ .

#### Application of Modifying Factors

Since many of the assumptions used in calculating the values in Table 5 are more conservative than are appropriate considering the exposure scenario, modifying factors were developed to adjust for the conservatism of the model. Occupancy of the basement was assumed to be only 50%. The basement floor was assumed to reduce the gamma flux by 50%. The horizontal extent of the contamination (shape) is assumed to reduce the time in proximity to the contamination by 50%. The distribution of  $^{60}\text{Co}$  in soil is not uniform. In order to account for this uneven distribution, it is assumed that the average soil concentration is one third the peak value resulting in a dose reduction of 67%. These modifying factors are summarized in Table 6.

Table 6 Modifying Factors

<u>Description</u>	<u>Reduction in dose</u>	<u>Modifying factor</u>
Occupancy	50%	2
Shielding	50%	2
Shape	50%	2
Activity distribution	67%	3

The combined modifying factor for this case, a product of the modifying factors for occupancy, shielding, shape, and activity distribution, is 24. This modifying factor is applied to the soil concentrations in Table 5 to give the results in Table 7. These values are used to establish a decontamination limit.

Table 7 Concentration of  $^{60}\text{Co}$  in Soil to Various Depths to Produce an Effective Dose Equivalent of 25 mrem/a for All Exposure Pathways

<u>Depth (cm)</u>	<u>Concentration in soil (pCi/g)</u>
0-10	22
0-15	16
0-20	13
0-30	9.8
0-40	8.7
0-50	8.0
0-70	7.6
0-100	6.5

Based on the profile of contamination a value of 10 pCi/g was selected as the decontamination limit for soils at the irradiation facility. This limit of 10 pCi/g was approved by the NRC as an acceptable residual soil contamination level for release of the facility for unrestricted use.

#### CONCLUSION

The methodology presented in this paper is usable for other contamination occurrences with proper application of nuclide specific data and modifying factors based on the exposure scenerio. The values presented, however, apply only to the site for which they were calculated.

## REFERENCES

- AEC (U.S. Atomic Energy Commission). 1974. "Termination of Operating Licenses for Nuclear Reactors", Regulatory Guide 1.86.
- Cember, H. 1969 Introduction to Health Physics. Program Press, Oxford
- Code of Federal Regulations. 40 CFR 190. "Environmental Protection Standards for Nuclear Power Operations."
- Eckerman, K.F. and Young, M.W. 1980. "A Methodology for Calculating Residual Radioactivity Levels Following Decommissioning," NUREG-0707.
- Federal Register. 1981. "Nuclear Regulatory Commission, 10 CFR Parts 30, 40, 50, 70, and 72, Decommissioning Criteria for Nuclear Facilities; Notice of Availability of Draft Environmental Impact Statement."
- Federal Register. 1983. "Environmental Protection Agency, 40 CFR 192, Standard for Remedial Actions at Inactive Uranium Processing Sites-Final Rule."
- Healy, J. W., Rodgers, J.C., and Wienke, C.L. 1979. "Interim Soil Limits for R&D Projects," LA-UR-79-1865-Rev.
- Healy, J.W. 1974. "A Proposed Interim Standard for Plutonium in Soils," LA-5483-MS..
- ICRP (International Commission on Radiological Protection). 1979a. Recommendations of the International Commission on Radiological Protection, Publication No. 26, Pergamon Press, Oxford.
- ICRP (International Commission on Radiological Protection). 1979b. "Recommendation of the International Commission on Radiological Protection. Limits for Intake of Radionuclides by Workers," ICRP Publication No. 30. Pergamon Press, Oxford.
- Kennedy, W.E., Jr., Napier, B.A., and Soldat, J.K. 1983. "Allowable Residual Contamination Levels in Soil for Decommissioning the Shippingport Atomic Power Station Site," PNL-4801.
- Kennedy, W.E., Jr., Watson, E.C., Hoenes, G.R., and Napier, B.A. 1979. "A Method for Determining Acceptable Residual Radioactive Contamination Levels," presented at the 24th Annual Meeting of the Health Physics Society, July 8-13, 1979, Philadelphia, PA.
- Kocher, D.C. 1983. "Dose Rate Conversion Factors for External Exposure to Photons," Health Phys., 45(3): 665.
- Napier, B.A. 1982. "A Method for Determining Allowable Residual Contamination Levels of Radionuclide Mixtures in Soil," PNL-3852.
- NCRP (National Council on Radiation Protection and Measurements). 1971. "Basic Radiation Protection Criteria," NCRP Report No. 39, NCRP, Bethesda, MD.
- NRC (U.S. Nuclear Regulatory Commission). 1981. "Draft Environmental Impact Statement on 10 CFR Part 61-Licensing Requirements for Land Disposal of Radioactive Waste," NUREG-0782.

NRC (U. S. Nuclear Regulatory Commission) 1977. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

NRC (U.S. Nuclear Regulatory Commission). 1976. "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material."

Rupp, E.M. 1979. "Dietary Intake and Inhalation Rates, Uap," in F.O. Hoffman and C.F. Baes, III, (eds.), "A Statistical Analysis of Selected Parameters for Predicting Food Chain Transport and Internal Dose of Radionuclides," ORNL/NUREG/TM-282.

Seeley, F.G. and Kelmers, A.D. 1984. "Geochemical Information for the West Chesnut Ridge Central Waste Disposal Facility for Low-Level Radioactive Waste," ORNL-6061.

# LEGAL AND ETHICAL ISSUES RAISED IN CONSIDERING RESIDENTIAL DECONTAMINATION OPTIONS FOR TECHNOLOGICALLY-ENHANCED RADIOACTIVE CONTAMINATION

Larry Jensen (USEPA, 230 S. Dearborn, Chicago, IL 60604) and  
Joyce Feldman (USEPA, 26 Federal Plaza, New York, NY 10278)

## ABSTRACT

A large number of residential and commercial properties have been identified as having radioactive contamination resulting from disposal of radioactive residues from ore extraction operations, usually radium, thorium or uranium. The U.S. Department of Energy has addressed many of the sites associated with the Manhattan Project through its FUSRAP program. But now, the U.S. Environmental Protection Agency is faced with removal of residues from other operations under its Superfund program. High removal and disposal costs are often difficult to balance against the comparatively low value of properties affected. Soil screening techniques must be rapid and present a relatively high level of assurance that criteria are not exceeded. Yet, private lives are affected. Considerations in making cost-benefit decisions are discussed here.

## INTRODUCTION

Residential and commercial properties radioactively contaminated by activities and waste disposal operations of former radium, uranium and thorium operations are increasingly being addressed through the U.S. Environmental Protection Agency's (USEPA) Superfund program. Although normal State and Federal regulatory mechanisms can handle most contemporary problems, sites contaminated before enactment of the Atomic Energy Act of 1954 or before active registration and licensing structures were implemented are usually out of the hands of the States or the Nuclear Regulatory Commission (NRC). The U.S. Department of Energy (DOE) will address many of the sites associated with the Manhattan Project through its Formerly Utilized Sites Remedial Action Program (FUSRAP). The remaining sites will probably require remediation by the USEPA through Superfund under the broad powers of the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) (1). Under CERCLA immediate removals can be undertaken in emergency situations and planned removals can be undertaken when there is sufficient time to weigh options. CERCLA can be applied to both NRC licensees and to DOE. However, because it was designed to address chemical contamination it has limitations that have yet to be corrected.

The USEPA has additional regulatory powers under the Clean Air Act (CAA) (2) and limited regulatory powers under the Safe Drinking Water Act (SDWA) through the Primary Interim Drinking Water Standards (3). Under the CAA, NRC, DOE and non-DOE Federal facilities must meet annual dose limits for air emissions. While the standards have been promulgated, the implementation procedures have not. Drinking water standards apply to most radionuclides but exclude radon and uranium and omit potassium-40. They do not apply to individual users. A more thorough treatment of these options and their limitations follows. Because the few legal tools set so few standards, ad hoc consensus criteria have been developed in some USEPA Regional offices to apply to identification and decontamination efforts.

## LEGAL FOUNDATIONS

The USEPA has three primary legal tools to deal with radioactively contaminated sites.

### Superfund

Broad power to decontaminate radiation sites exists in CERCLA. Although the original intent of this act was to focus on chemical contamination problems, it has great applicability for radiation sites where normal mechanisms for Federal or State supervised decontamination are inappropriate, where cleanup costs are beyond the normal resources of Federal, State or Municipal agencies, of corporate bodies, or of individuals, or where no responsible party can be identified. Superfund can be applied to NRC licensees and may be applicable to certain FUSRAP sites.

CERCLA remediation mechanisms include both emergency and long-term remedial response actions. In emergency actions costs are generally limited to \$1 million or to cleanups requiring less than 6 months. No state cost sharing is required. Long-term remedial actions require listing on the National Priorities List (NPL) and cost sharing by the affected state. Initially, long-term actions begin with an evaluation of available data using the Hazardous Ranking System (HRS) where known releases to the air, groundwater and surface water are numerically evaluated and an overall ranking score assigned to the site. Additional hazards such as fire, explosion and direct contact can also be scored but do not figure in the final site score. Sites scoring above an arbitrary cutoff number qualify for inclusion on the National Priorities List (NPL). This qualifies a site for a Remedial Investigation, a thorough data collection and evaluation, and for a Feasibility Study, including consideration of mitigation options with a recommendation for the most feasible solution. Implementation of the most feasible solution can occur through a legal order to an identified responsible party to perform the decontamination at their expense or money can be utilized from the Superfund to implement the solution. When the Superfund is used, legal cost recovery actions may be taken against the responsible party in an attempt to replenish the fund. All remedial actions require a contractual agreement between the USEPA and the affected State and may include cost sharing on the State's part. DOE sites which score above the minimum ranking score must also be remediated at DOE's expense, with USEPA oversight.

### Clean Air Act

Radionuclides were listed as hazardous air pollutants under Section 112 of the Clean Air Act in November, 1979. Subsequently, National Emission Standards for Hazardous Air Pollutants (NESHAPS), specifically for radionuclides, were issued in February, 1985. Applied to DOE, NRC, and non-DOE Federal facilities, they limit emission dose equivalents for the public to 25 millirem (mrem) per year to the whole body and 75 mrem per year to the critical organ. Doses from radon-220, radon-222 and their progeny were excluded. Provisions for alternative dose equivalent standards of 100 mrem per year for continuous exposures and 500 mrem per year for noncontinuous exposures were included. In the same promulgation, the NESHAP for elemental phosphorus plants was set at 21 curies per year for polonium-210.

The CAA and CERCLA are linked. Because radionuclides are hazardous air pollutants under the CAA they fulfill the definition of a hazardous substance under CERCLA. This is the primary justification for the application of CERCLA to decontamination of radioactive sites.

## Safe Drinking Water Act

Radionuclide concentrations in community water systems are controlled by the NIPDWS developed for the SDWA. The definition of community water system requires 15 service connections used by year-round residents or the service of 25 year-round residents. The regulations limit the maximum contaminant levels for

- (1) combined radium-226 and radium-228 to 5 picocuries per liter (pCi/L),
- (2) gross alpha particle activity (including radium-226 but excluding radon and uranium) to 15 pCi/L, and
- (3) beta particle and photon radioactivity from man-made radionuclides to an annual dose equivalent for the total body or any internal organ to 4 mrem per year.

These regulations are in the process of revision. It is expected that uranium will be added to the new standards.

The NIPDW standards exclude individual users and specifically exclude standards for radon and uranium. The latter are contaminants of primary concern in abandoned site actions. Even potassium-40, which might be enhanced in the environment by some industrial operations, is omitted. Revision of the present set of standards is underway and it is hoped these will fill the needed gaps in the present standards.

In summary, the substance of the legal basis of USEPA decontamination efforts is that CERCLA is a versatile mechanism for immediate and planned removals, the CAA limits doses due to air emissions and serves as the justification for CERCLA intervention, and the NIPDW standards can be imposed where community drinking water systems are contaminated by most radionuclides.

## Limitations of the Laws

CERCLA is not a law specifically designed for cleanup of radioactively contaminated sites. It is aimed at correcting problems attributable to nonradioactive chemicals. Consequently, there are limitations. First, the Hazard Ranking System poorly quantifies radiation risks. For instance, radioactive materials at concentrations generally encountered for site problems are seldom acutely hazardous. The lowest level ranking for waste quantity on the HRS is 1 to 10 tons, much too large for radioactive materials that may present substantial problems at the milligram or less level. Moreover, direct contact, a major concern for health physicists, is not assessed except in a supplementary way. The mixed radioactive-chemical waste problem is also not addressed. Pacific Northwest Laboratory has developed a modified HRS for DOE sites, although it does not have any formal status under CERCLA. USEPA Office of Radiation Programs is also working on a modified HRS that could be introduced into CERCLA.

Risk assessment for the purposes of developing an HRS score or justifying an immediate removal is performed by the Centers for Disease Control. This stems from power delegated to the Surgeon General under the Superfund act to head the Agency for Toxic Substances and Disease Registry within the Public Health Service.

Until the entire package for the CAA is codified, compliance and enforcement of this act is clouded. Inhalation doses from radon-222 and radon-220 are excluded for DOE and NRC sites. Non-NRC and non-DOE regulated sites, such as coal-fired power plants, are not covered. Nevertheless, the law is in effect and its standards must be met.

## ETHICAL ISSUES AT SITES

When decontamination efforts have actually been performed, ethical problems have arisen that tangle the strict application of professional judgments. In a more personal tone, we will give some insights into problems we have encountered.

### 1. Seek and Find

The first critical task in decontamination projects is to find all of the contaminant. If the focus is on a single site then the prospects of uncovering information are better. Records, even if meager, might exist; former employees may still be alive and in the area; neighbors may share memories with you. An aerial gamma survey is possible. Nevertheless, in the case of radium, cottage industries were numerous, and wastes were freely sold or given away. If the focus is on this type of industry the problems are more difficult. Thomas' Industrial Register, State incorporation records, Federal industrial publications and scientific journals are useful to collect company names and addresses, even for decades back; but, unfortunately "radium" was often used then like "environmental" or "high tech" is now, to impress target audiences. Radium might not be associated with the product in any way. In one case where an attempt was made to locate old companies with "radium" in their names, the search led to the old jewelry district but almost all the addresses were for multi-story high rises now used for other purposes. Should 30 floors of tenants be disturbed, maybe unnecessarily frightened, on the remote chance one room is contaminated?

### 2. How Long, How Far the Search?

In one investigation, uranium contaminated ion-exchange resins were sold to a distributor who naively resold them for home water softener use. The search covered the continental United States and spread into Canada. This effort was extraordinary and normally not possible. In another case every lawn in a town of 13,000 was surveyed for thorium wastes from an old gas mantle operation. Even with this exceptional effort, the wastes, which were good construction materials, might also have been used in neighboring towns. A fundamental problem with searches of this type is that they are effectively open-ended. There is little certainty that another case won't turn up after the decontamination or remedial efforts have presumably been completed.

### 3. What's Not Dirt?

Once the general region of concern is identified then comes the problem of separating dirt from more than dirt. Without comprehensive guidelines, the job is difficult. The authors have listed a home-grown table of criteria they have found reasonable elsewhere in this paper. These criteria have varying degrees of sanction. Sometimes they're acceptable simply because they're not unreasonable. Hopefully, they're low enough now that they won't be lowered in the future, thereby discrediting work already done.

### 4. How Clean is Clean?

A very good job can be done if you set low cleanup criteria, take a very long time and spend immense amounts of money. Each retreat from pristine involves increasing the risk of future problems.

If criteria are too lenient work may need to be redone. The same applies to cutting corners on time, labor and expenses. Yet years and years and millions and millions of dollars can't be spent on every problem. Where do you draw the line for a conscientious, defensible job?

## 5. Can a Clean Cleanup be a Disaster?

Yes. If you devote yourself to the philosophy As Low As Reasonably Achievable (ALARA), if you are meticulous in your efforts, if you have an immense fund of money, you will pretty much solve your problem. The new problem you created, however, may be just as big because you have tens of thousands of cubic yards of dirt, truck loads of contaminated pipes, brick and furniture and barrels of contaminated coveralls, booties and gloves. Existing low-level waste sites won't take much of this. The community with the problem wants it out of town--NOW! Wherever you choose to move it, there will be a community ready to fight to keep it out. Is it astute or ethical to impose these wastes on communities because it is viewed to be in the national interest? If you don't, what do you do with this decontamination garbage?

## 6. How do You Get Down to Clean?

Methodology is very important. An aerial gamma survey will identify major size problems - those bigger than a bread truck. A walkover will be better if the grid's not too large. Gamma radiation levels will be the easiest to measure--you can do it walking upright and it's quick - but beta + gamma surveys might greatly increase your sensitivity. Criteria applied at one meter are idealistic because they ignore the geometry of lumpy excavations. Contact criteria are needed to remove small pockets of high concentration, but what's a good criterion?

## 7. What About Our Citizens?

How does one address interactions with the public? The very act of approaching a property owner/occupant may create a high level of stress and anxiety. You may get the door slammed in your face. Where children are involved the reaction is more likely to be fearful, strident and aggressively demanding. With older citizens, especially those who have some knowledge of the facility's operations, indifference or outspoken skepticism may be more likely. These are the people who will refuse surveys and refuse cleanups because they feel they beat the odds. All these diverse people are our constituency. How do we do the best public health job for all while respecting their personal positions?

## 8. Are There Limits to Cleanups?

Every decontamination operation has its dilemmas and sticky decisions. In an ongoing cleanup, the trees near the factory in question had for decades acted as flat plate collectors of airborne emissions; the rain had washed the leaves and brought the radioactive dust to the ground below. Worms had distributed the contaminated material through the top 12 or so inches of soil. It was impossible to decontaminate the root system without taking out the whole tree. This was painful to homeowners who quite naturally had grown attached to these beautiful old trees. In the same operation an outdoor fireplace had been built over contaminated soil by the husband of a woman now widowed. She refused to allow decontamination under the fireplace. Legalities aside, should people's rights to beautiful old trees or cherished possessions force us to set aside our decontamination objectives? If we do, we also set aside our public health commitments. We cannot say unequivocally that the cleanup was complete. We have not eliminated unnecessary exposures. The owners or their heirs will eventually have to pay economically when the property goes up for sale, either in lost equity or in personal decontamination expenses. The community's image will always remain tainted.

## 9. What About the Economics?

Simply the suggestion of radiation problems in a community can decimate property values. Those who want to sell and get out can only do so by taking great financial losses or by being deceptive. Those who can't get out get bitter or disheartened.

Cleanups are immensely expensive. The removal of only 10,000 cubic yards of radium contaminated soil and furniture in one Superfund remedial action cost \$600,000. Yet a thorough cleanup will do much to restore property values. A fair compensation program for property owners can be an extremely cost-effective public relations tool. The removal of psychological stress and anxiety is important though harder to quantify monetarily. Thorough and fair programs probably reduce the chances of costly legal battles.

Paying for cleanups is tricky. Responsible parties must assume cleanup costs. If this is not possible or quicker action is required the Superfund is used. States or municipalities in the affected areas must pay 10% (or 50% if they were responsible parties). Legal action to recover costs can be used to replenish the Superfund. Technically, a property owner who naively accepted or took what they believed to be innocuous fill and now has a radioactively contaminated yard is still a responsible party and could be assessed cleanup costs in the kilo or mega buck range.

## 10. Is "Done" Ever Done?

If we must accept historical searches with gaps, if our surveys are limited by time, labor and money, if criteria become more conservative, if we encounter litigious companies or citizens, to name a few problems, will the need for remedial efforts ever really go away? Is every major cleanup action a lifelong assignment?

## CONCLUSIONS

When radioactively contaminated sites can't be addressed through normal State or Federal regulatory mechanisms or through special programs such as DOE's FUSRAP program, the USEPA has the ability to intercede through its regulatory powers in the Comprehensive Environmental Response, Compensation and Liability Act, the Clean Air Act, and the Safe Drinking Water Act. CERCLA especially has broad utility, both legally and financially. Unfortunately, there are no comprehensive criteria or guidelines for assessing contamination levels and setting decontamination objectives. The USEPA Region V has, for itself, collected its own ad hoc criteria over many years of work that it feels are reasonable. These are provided in the accompanying table.

As site assessments and site decontaminations have actually been performed a number of difficult ethical problems have presented themselves. They seem to be inherent aspects of the cleanup task. Some problems may be resolved as our experience develops. Some issues will remain as quandries. Nevertheless, the ethical problems, like the legal and technical issues, must be addressed whenever decontamination actions are performed.

\*\*\*

Although the information described in this article has been funded wholly or in part by the United States Environmental Protection Agency it does not necessarily reflect the views of the Agency and no official endorsement should be inferred.

## BIBLIOGRAPHY

1. 96th Congress, Public Law 96-510, "Comprehensive Environmental Response, Compensation and Liability Act of 1980," United State Code, Title 42, Part 9605, et seq.
2. 95th Congress, Public Law 95-95, "Clean Air Act," United States Code, Title 45, Section 1857, et seq.
3. U.S. Environmental Protection Agency, "National Interim Primary Drinking Water Regulations," Code of Federal Regulations, Title 40, Part 141.
4. U.S. Environmental Protection Agency, "Standards for Remedial Actions at Inactive Uranium Processing Sites," Code of Federal Regulations, Title 40, Part 192.
5. U.S. Environmental Protection Agency, "Environmental Standards for Uranium and Thorium Mill Tailings at Licensed Commercial Processing Sites," Code of Federal Regulations, Title 40, Part 192.
6. U.S. Nuclear Regulatory Commission, "Disposal or Onsite Storage of Thorium or Uranium Wastes from Past Operations," Federal Register, Volume 46, Number 205, October 23, 1981, pp 52061-52063.
7. W.H. Ellett and N.S. Nelson, "Informal Technical Note on Thoron Risk," U.S. Environmental Protection Agency, May, 1984.
8. Centers for Disease Control, "Public Health Advisory - Glen Ridge and Montclair, New Jersey," December 6, 1983.
9. U.S. Nuclear Regulatory Commission, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material," July, 1982.
10. C.R. Cothorn, W.L. Lappenbusch, and J.A. Cotruvo, "Health Effects Guidance for Uranium in Drinking Water," Health Physics, 44, Supplement No. 1, pp 377-384, 1983.

Ad Hoc Criteria Found to be Reasonably Applicable in U.S. Environmental Protection Agency Region V

<u>Nuclide</u>	<u>Criteria</u>	<u>Basis</u>	
SOILS			
Radium-226	5 pCi/gm averaged over 100 m <sup>2</sup> * and the top 15 cm of soil, 15 pCi/gm for subsequent 15 cm depths - applied over background	40 CFR 192.12(a)(1,2)†	(4)
Radium-228	Same criteria as radium-226	40 CFR 192.41(c)	(5)
Natural Thorium (Th-232 + Th-234)	10 pCi/gm **	46 FR 205, pp 52061-52063 ††	(6)
Natural Uranium (U-238 + U-234)	10 pCi/gm **	46 FR 205, pp 52061-52063	
AIR			
Radon-222	0.02 working levels (WL), including background, in an occupied or habitable building	40 CFR 192.12	
Radon-222	20 pCi/m <sup>2</sup> /sec from exposed soil	40 CFR 192.32	
Radon-220	0.05 WL including background in an occupied or habitable building	40 CFR 192.12 and USEPA "Informal Technical Note On Thoron Risk" May, 1984	(7)
Radon-220	20 pCi/m <sup>2</sup> /sec from exposed soil	40 CFR 192.41	
	* 100 m <sup>2</sup> is generally too large for residential remedial actions		
	** all daughters present and in equilibrium		
	† Code of Federal Regulations, Title 40, Part 192		
	†† Federal Register, Volume 46, Number 205, pages 52061-52063		

(CONTINUED)

<u>Nuclide</u>	<u>Criteria</u>	<u>Basis</u>	
SPECIAL RADON CRITERIA			
	>0.5 WL	Immediate Action (1-2 days) Restrict occupancy to 2 hr/day, prohibit smoking in high level areas  Within 2 weeks Reduce levels to as far below 0.5 WL as feasible using temporary measures	Public Health Advisory (8) Glen Ridge and Montclair New Jersey - Centers for Disease Control, December 6, 1983
	0.1 - 0.5 WL	Prompt Action (1-3 months) Temporary remedial action to reduce levels to 0.1 or less	
	0.02 - 0.1 WL	Permanent Remedial Action (1-2 years) Reduce exposures from non-natural sources below 0.02 WL	

The above Special Radon Criteria can be modified to apply to radon-220 by dividing the daughter concentrations in working levels by 0.4

USEPA "Informal Technical  
Note on Thoron Risk" -  
May, 1984

#### SURFACE CONTAMINATION

U-natural, U-235, U-238, and associated decay products	5000 dpm alpha/100 cm <sup>2</sup> , average 15000 dpm alpha/100 cm <sup>2</sup> , maximum 1000 dpm alpha/100 cm <sup>2</sup> , removable	Nuclear Regulatory (9) Commission, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source or Special Nuclear Material" - July, 1982
Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227	100 dpm/100 cm <sup>2</sup> , average 300 dpm/100 cm <sup>2</sup> , maximum 20 dpm/100 cm <sup>2</sup> , removable	

(CONTINUED)

<u>Nuclide</u>	<u>Criteria</u>	<u>Basis</u>
Th-natural, Th-232, Ra-223 Ra-224, U-232	1000 dpm/100 cm <sup>2</sup> , average 3000 dpm/100 cm <sup>2</sup> , maximum 200 dpm/100 cm <sup>2</sup> , removable	
Beta-gamma emitters (nuclides with decay modes other than alpha emission except those noted above	5000 dpm/100 cm <sup>2</sup> , average 15000 dpm/100 cm <sup>2</sup> , maximum 1000 dpm/100 cm <sup>2</sup> , removable	
DRINKING WATER		
Radium-226 + 228	5 pCi/L	National Interim Primary Drinking Water Regulations
Gross alpha (excluding radon and uranium)	15 pCi/L	
Beta particle and photon radioactivity from man-made radionuclides	4 mrem/yr	
Uranium-natural	10 pCi/L	"Health Effects Guidance for Uranium in Drinking Water," Health Physics Society Journal, Vol 44, Supplement No. 1, pp 377-384, 1983

(10)

PUBLIC INFORMATION EXPERIENCE IN THE URANIUM MILL  
TAILINGS REMEDIAL ACTION (UMTRA) PROJECT

H.R. Meyer and D.J. Skinner, Chem-Nuclear Systems, Inc.  
P.O. Box 9136, Albuquerque, NM 87119

ABSTRACT

To enhance public perception of the UMTRA Project remedial actions at some 23 sites in the United States, an information program has been designed to demonstrate the small risk associated with the work in each community. An unusual aspect of this information program has involved a specific effort to educate listeners concerning the Health Physics terminology being used. The information program has been successful in reducing the level of public anxiety associated with UMTRA remedial action at the first Project site, and is currently being applied at several new sites.

INTRODUCTION

An important aspect of any radioactive materials decontamination and decommissioning project involves work to ensure the cooperation of the public. Without such cooperation, routine operations at the decommissioning site can be made much more difficult, and scheduling can be impacted. In extreme cases, misunderstandings can lead to severe animosity on the part of the public, with consequent picketing and other disruptive events. Because the UMTRA Project remedial action involves decontamination of private residences, as well as large uranium tailings piles located in or near communities, daily interaction with private citizens as well as more formally organized concerned citizen's groups is necessary. A firm background of trust and understanding, based on early public information work at each site, has been found to be an effective approach to minimizing problems.

CANONSBURG

The specific case study used as the basis for this paper involves the first UMTRA Project site to be decontaminated, the former Vitro processing site located in Canonsburg, Pennsylvania, near Pittsburgh. The work to be performed in Canonsburg involved stabilization-in-place of the large tailings pile located in Canonsburg, and remedial action on more than 100 homes and other nearby vicinity properties contaminated with associated radioactive materials. Chem-Nuclear Systems, Inc. (CNSI) is an integrated subcontractor to Morrison-Knudsen, Co. (M-K), contracted by the US Department of Energy to perform remedial action at some 23 UMTRA mill tailings sites and numerous vicinity properties around the country. Chem-Nuclear is responsible for all radiological measurements in the Project, including worker and environmental protection, and final verification/certification monitoring. The team of Morrison-Knudsen and Chem-Nuclear Systems is collectively identified as the Remedial Action Contractor (RAC).

The RAC began preparatory work in Canonsburg in the Fall of 1983, but significant remedial action did not commence until Spring, 1984. By that time, several public meetings with school and community groups had been held, and an atmosphere of distrust and hostility toward the project had been noted in the community. By early Fall of 1984, groups in the local community had voiced their intention to picket, with the object of closing down local schools, which were perceived as being impacted by radioactive materials being released to the environment from the tailings site. In response, the USDOE prepared for a series of public meetings to be held in the community, and CNSI staff were requested to prepare presentations designed to thoroughly describe the environmental protection program implemented by the RAC in the Canonsburg community. Health Physicists from Oak Ridge National Laboratory, involved in other segments of the Project, were also called in to make presentations at the scheduled meetings.

### Training

At earlier public meetings, MK and CNSI staff had noted that many of the problems developing between Project staff and the local Canonsburg community simply involved misunderstandings of information being presented. It was therefore decided to depart from previous public information formats, and to spend a significant amount of presentation time in "training" audiences in the unfamiliar health physics terminology being used. Such terms as activity, concentration and dose were emphasized and described in more familiar terms, and the common units including the Curie, pCi, pCi/l, rem and mrem were covered in detail. Comparisons were made to more ordinary units: for example, the pCi/l was compared to more familiar units of concentration. Figures 1 and 2 display viewgraphs used in this context. Viewgraphs were designed for minimum complexity, particularly for that early portion of the presentation designed to make the audience comfortable with the terms and units to be used later.

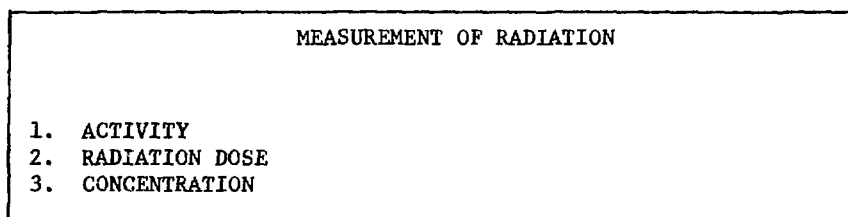


Figure 1. Basic Terminology

Questions were encouraged, and the atmosphere was made as informal as possible. The presenter stood just in front of the audience, not remote on a raised pedestal. The presentation was made by a trained health physicist, with good presentation skills, in place of a professional public relations speaker. The reasoning behind this emphasis should be clear: the presentation was similar in content and format to a classroom lecture, and questions from the audience had to be handled from a firm basis of health physics/radiation biology knowledge.

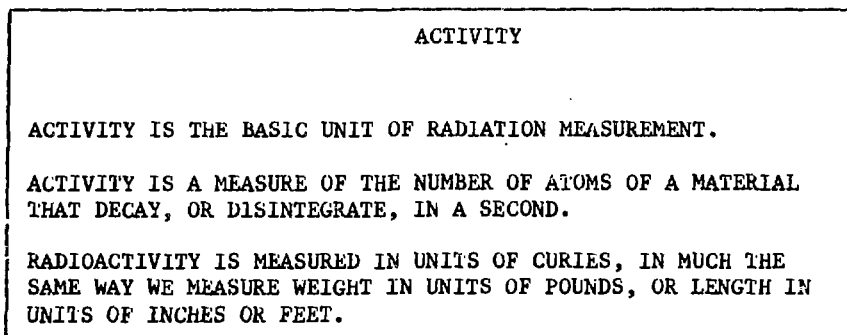


Figure 2. Basic Terminology

#### Safe Levels of Exposure

Once sufficient time had been spent in acquainting the audience with these basic terms, a few minutes were spent in outlining the basis for currently defined safe exposure levels. Figure 3 is an example of one of the viewgraphs used in this segment of the presentation. The purpose of this information was to convince the audience that there are safe levels of exposure, i.e., levels below which we expect essentially no hazard. An effective argument in this context involved the fact that the speaker, as a radiation worker, was allowed radiation exposures an order of magnitude larger than allowed for the general public, and was comfortable with the risks of such exposures.

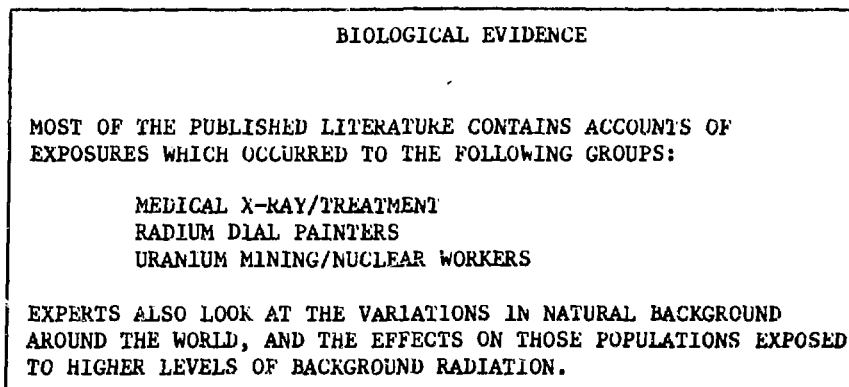


Figure 3. Determining Safe Exposures

## Background Levels

Given this preliminary information, the presentation then focused on levels of radiation commonly found in the natural environment. This discussion of background radiation levels was particularly appropriate in the context of the UMTRA Project, since the only radionuclides encountered in the Project are uranium chain isotopes. Figures 4 and 5 illustrate the approach taken in the discussion of background radiation; note that comparisons are made between exposures encountered in Pennsylvania, site of the presentations, and other states.

WE ARE CONTINUALLY SURROUNDED BY NATURAL RADIATION.  
WE CALL THIS NATURAL RADIATION BACKGROUND RADIATION.  
WE USE THE LEVELS FOUND IN OUR NATURAL ENVIRONMENT  
AS INDICATORS OF ACCEPTABLE LEVELS OF RADIATION.  
IN A SENSE, BACKGROUND IS THE NATURAL STANDARD FOR MINIMAL  
LEVELS OF RADIATION IN OUR ENVIRONMENT  
THIS PHILOSOPHY GIVES THE RADIATION PROTECTION PROFESSION  
A STARTING POINT FOR WRITING STANDARDS.

Figure 4. Background Exposures

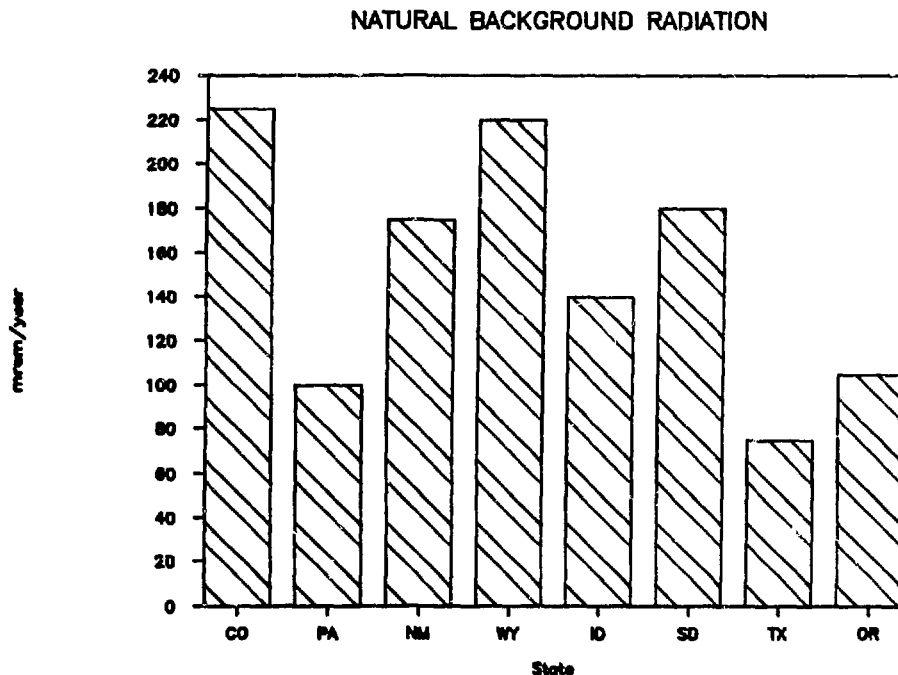


Figure 5. Background

This focus on local conditions compared to levels in other communities was emphasized throughout the remainder of the presentation. Figure 6 represents a viewgraph developed after repeated presentations of the information program. A number of specific questions recurred at the meetings; preempting a question through the format illustrated in this Figure was found to be an effective approach. Once background radiation levels were covered in sufficient detail, the question of relative risk could easily be handled graphically, in the format illustrate in Figure 7.

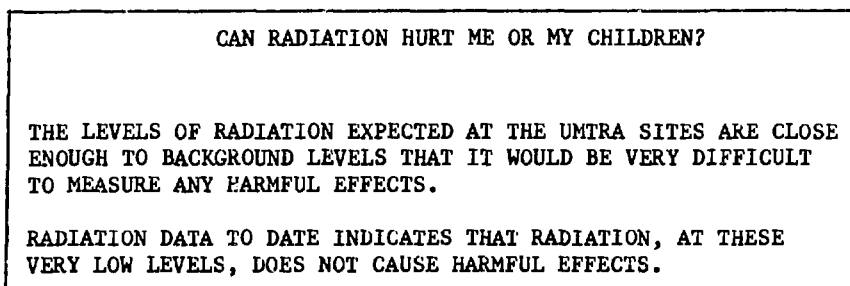


Figure 6. A Recurring Question

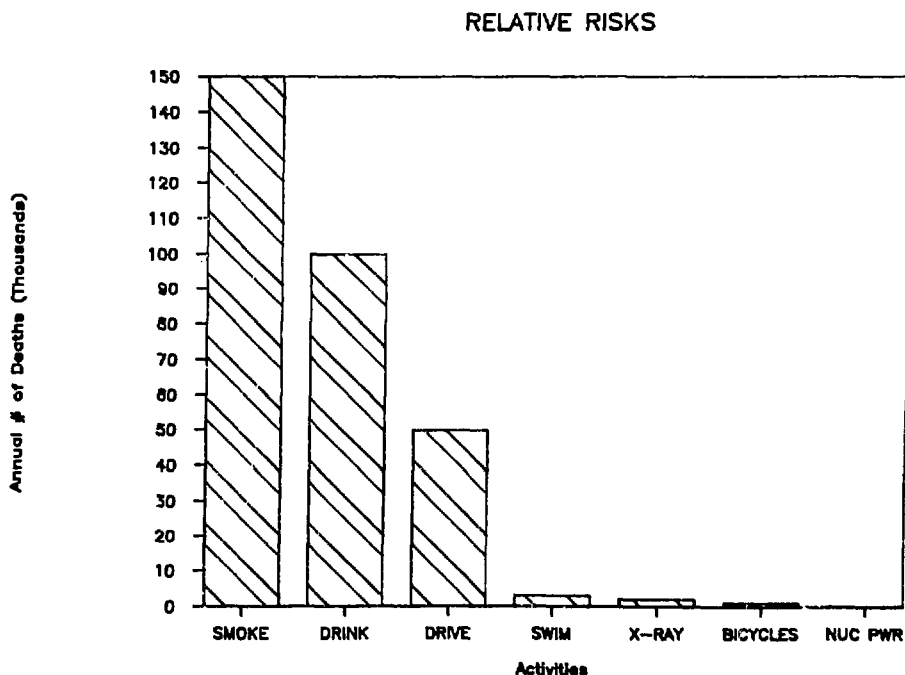


Figure 7. Risks

## Radon

In the Canonsburg, Pennsylvania community, the public had become particularly concerned with the presence of radon gas emitted from the mill tailings pile. For this reason, a great deal of time was spent discussing levels of radon in the community, comparing these levels to concentrations measured elsewhere for radon and daughters, and in defining the program of radon monitoring developed for use in the community. Because the RAC had been operating this system for nearly a year at this time, summaries of data were presented specific to Canonsburg, and these measured values were compared to values in other communities around the world. Based on this information, a discussion of USDOE standards for environmental concentrations of radon gas was initiated. Figure 8 illustrates a set of points made during this discussion. Figure 9 presents some typical environmental radon data taken during remedial action.

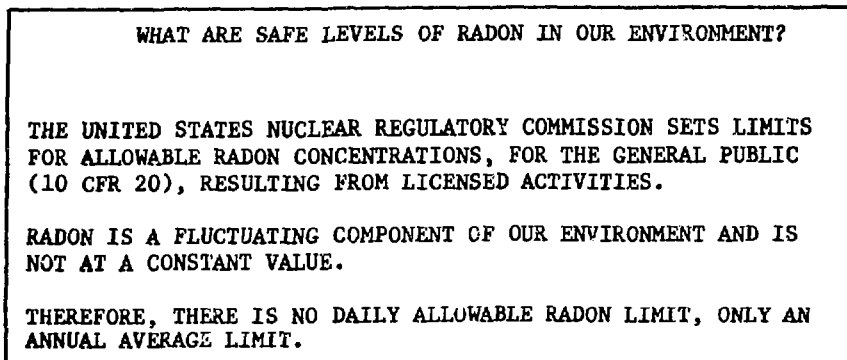


Figure 8. A Specific Question on Radon

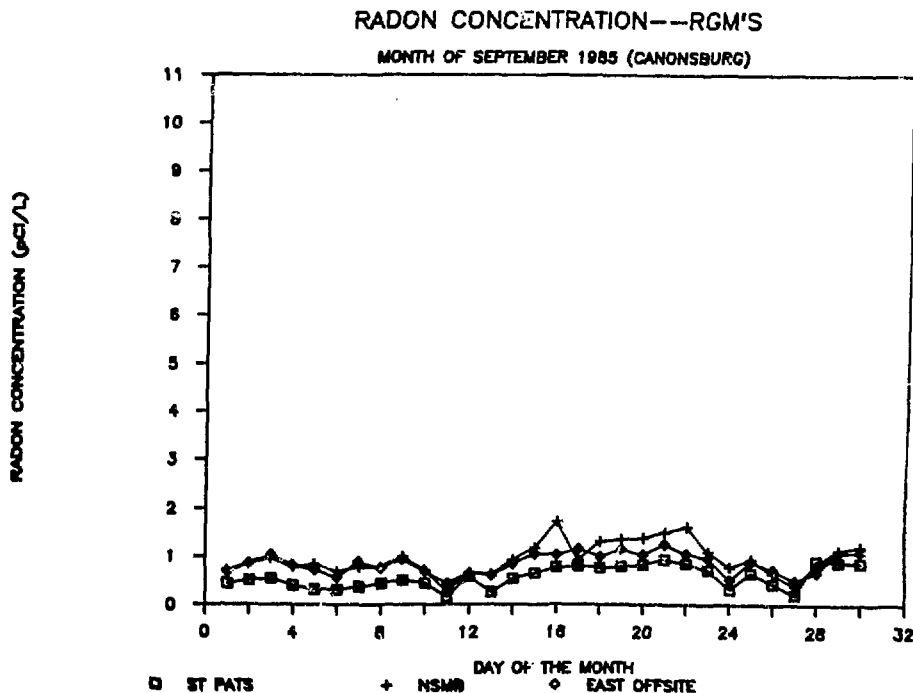


Figure 9. Community Radon Data

In this context, data derived from additional studies performed by Mound Laboratories were also presented, and compared to local values measured by the RAC. It should be noted that, for other communities in which this program has been presented (in New Mexico, Arizona, Colorado, Wyoming, and Oregon), community emphasis is more typically on radioactive particulates or groundwater contamination, and this section of the presentation contains location-specific information emphasizing these concerns. The applicable standards were discussed, emphasized the standards associated with the community's principal concern. The UMTRA Project remedial action standards were also discussed, to familiarize the community with the purpose of the project.

An interesting problem in communicating information on radon concentrations and risks was uncovered during the meetings, and a series of viewgraphs including Figures 10 and 11 were developed to handle this problem. The concept of dispersion and dilution of radon leaving the vicinity of the tailings pile was found to be a difficult one, with a number of individuals convinced that radon from the pile was reconcentrating in their homes and schools, and exposing themselves and their children to abnormally high levels.

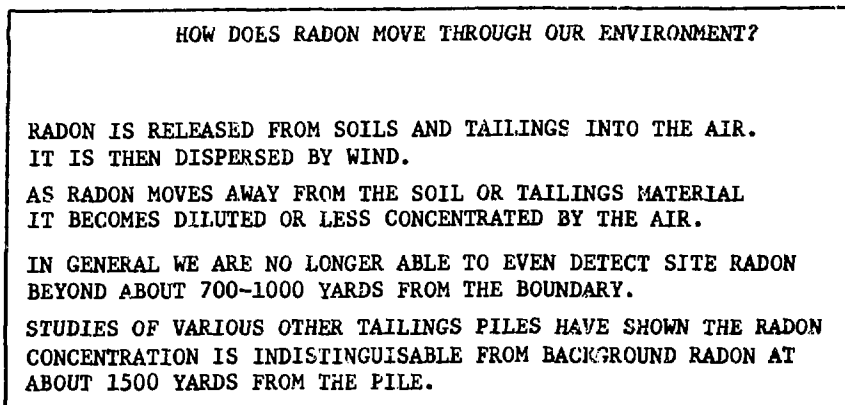


Figure 10. A Specific Question on Radon

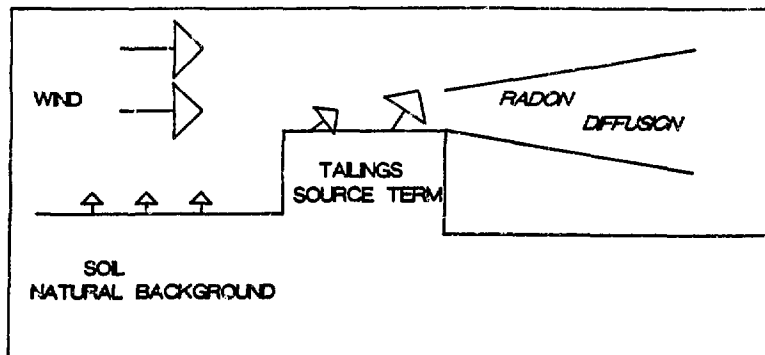


Figure 11. Dispersion/Dilution of Radon

Figure 12 in particular was developed to demonstrate the source of observed, increased levels of radon in some local homes; an example of smoke dispersing from its source was used to illustrate the principal of radon diffusion after leaving the site. Even after these discussions, enough concern remained in the community to justify developing a set of radon-response procedures, involving defined control measures to be employed if community radon levels ever reached certain values. These control measures included watering or covering the tailings, and modifying the construction plan and schedule to reduce the quantity of exposed tailings. While these controls were never employed during remedial action, their development and publication were a positive factor in community relations, and were seen as a reasonable response to voiced concerns. To assure the community that the procedures were being enforced, weekly distribution of data from the community radon monitors was initiated. Community representatives also were encouraged to witness the collection of radon concentration data from the community monitors. Actual handling of the calibrated monitoring units was, of course, not permitted for any but trained technicians employed by CNSI or the calibrating laboratory.

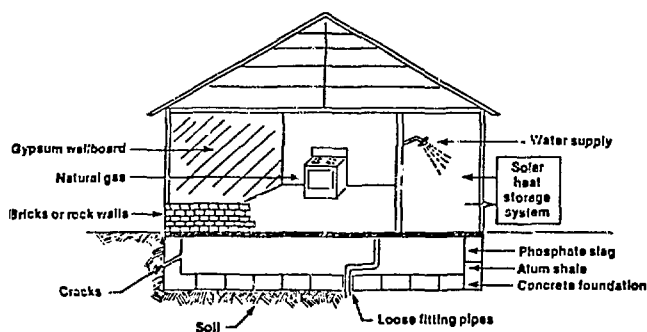


Figure 12. Radon Sources

### Coverage

A series of 15 presentations, each an hour in length, was made in the Canonsburg area during the Fall of 1984 using this step-by-step, lecture approach. Presentations were made to local governmental and civic groups, high school, junior high school and elementary school students, the entire faculty and administration of the local school district, realtors, nurses, maintenance personnel and others. Response to the series was surprising, with a number of previously concerned individuals stating that they finally understood the risks involved in the remedial action, and were no longer worried.

The threatened school picketing action was cancelled, and the remedial action work, both at the tailings site and in homes throughout the community, proceeded on schedule. Health physics and engineering personnel located at the site have continued to attend public and local government meetings during the last year, to inform interested citizens as to the work's status and current levels of radon in the environment. Weekly distribution of radon concentration information (Figure 9) has continued.

#### SUMMARY

The success of this effort seems to be related to a few key elements in the Public Information lectures:

1. Presentation, by an expert, of material geared to the level of the audience, with visual aids simplified through the use of graphics and short phrases, and made as community-specific as possible;
2. Familiarization of the audience with the key radiological units to be employed in the lecture.
3. Discussion of natural background levels of radiation, for reference;
4. Comparisons of local, measured values for key radionuclides with values measured in other communities around the country;
5. Discussion of the basis for radiation protection standards, and comparison of the risks associated with these exposures to other commonly accepted risks,
6. Use of an informal atmosphere allowing the audience to ask questions without embarrassment, and;
7. Demonstration of the influence of community concerns by responding to questions, when appropriate, with actual modifications to plans or schedules.

While preparation and delivery of such a community-specific program requires a great deal of effort, the benefits of cooperation by the local community can be of extreme importance in terms of overall project efficiency.

FINANCIAL ASSURANCE FOR DECONTAMINATION AND DECOMMISSIONING:  
A TEXAS PERSPECTIVE

C. D. Rao, Stephen D. Etter, and Timothy W. Dziuk  
Bureau of Radiation Control, Texas Department of Health  
1100 West 49th Street  
Austin, Texas 78756-3189

ABSTRACT

The Texas Department of Health (TDH) has the regulatory responsibility to ensure that funds are available for decontamination, decommissioning, and reclamation of uranium recovery facilities in Texas. Uranium recovery licensees are required to post financial security with the Agency for that purpose.

Texas uranium facilities include (1) conventional surface mining and milling plants, including tailings ponds, and (2) in situ solution mining plants, each with somewhat different cost elements for decontamination, decommissioning, reclamation, and closure.

Cost estimates for decontamination, decommissioning, and reclamation, along with a facility closure plan, are initially submitted to the Agency by the licensees. These are verified and compared with detailed independent cost estimates prepared by Agency staff. Significant differences between the two estimates are examined and resolved by negotiation and/or recalculation to the satisfaction of the state.

The Texas philosophy for maintaining financial security permits flexibility in the closure plan without jeopardizing or compromising the ultimate long-term objectives of closure. Review of closure plans incorporates new technological developments. In contrast, financial security is established expeditiously by applying the best available cost data to necessarily conservative estimates of the work involved. Financial security cost estimates are subject to annual review and adjustment.

INTRODUCTION

Uranium recovery in Texas began about 25 years ago. Texas regulations require that uranium recovery facilities, which have a useful life of 5 to 15 years, be properly decontaminated and decommissioned. This paper presents the major features of Texas procedures for ensuring decontamination and decommissioning, and addresses questions encountered in the process.

URANIUM INDUSTRY IN TEXAS

Two basic forms of uranium recovery operations are conducted in Texas: conventional uranium mining and milling and in situ leaching operations. Conventional recovery involves the excavation and transport of ore to a central facility for processing and disposal of the depleted ore (tailings) in surface impoundments. At in situ operations, a fluid is injected into the ore zone to solubilize uranium, which is then transported from well fields to a surface recovery facility.

Before Texas became an agreement state in 1963, one conventional facility was licensed by the Atomic Energy Commission. This facility, which includes seven tailings ponds on 146 acres, has since been declared a Title I site (i.e., eligible for 90% federally-funded remedial action). Texas has licensed four additional conventional facilities. The first three (licensed in 1970, 1971, and 1977) have tailings impoundments of 46 acres, 260 acres, and 150 acres. The latest (licensed in 1983) would have utilized below-grade disposal of tailings in the original excavation; construction of this facility was not started, and the company has recently requested termination of the license.

The first in situ uranium recovery license was issued in 1968 to Union Carbide to operate their Palangana Dome Project, where a 24-acre well field was developed. To date, 17 in situ uranium recovery facilities have been licensed with a total of 690 acres under production and a total of 2610 acres authorized for production. Thirteen of these facilities were licensed between 1977 and 1981. Well fields currently under production range from 3 acres to 152 acres in size.

Conventional uranium recovery begins with the removal of as much as 300 ft of overburden to expose the uranium ore zone. The ore is then extracted and transported to a central facility, where the uranium is leached out, concentrated, and subsequently dried to produce "yellowcake." The tailings, rich in radium-226 and its daughters, are then deposited in a tailings pond. Radioactive wastes result primarily from contamination of buildings, equipment, and surface soils. The principal sources of radioactivity are particulates from ore handling, yellowcake dust from drying and packaging operations, and radon and particulates from tailings ponds. The main termination activities are ore pit reclamation; process plant decommissioning and decontamination of the surrounding surface environs; and stabilization of tailings ponds. All areas are to be released for unrestricted use except the sites of tailings ponds and buffer zones, which will be deeded to the state. (Below-grade tailings disposal sites will also be released.)

In situ mining requires the construction of production and injection wells that tap a uranium ore zone located in an aquifer. In the production circuit, an oxidizing fluid is injected into the ore zone, where it solubilizes uranium (and other elements). The fluid is then pumped to the surface, passed through ion exchange columns to remove the uranium, and reinjected. "Loaded" columns are then diverted into a recovery circuit, where the uranium is eluted and concentrated for shipment either as a slurry or as dried yellowcake. Liquid waste sources include a "bleed stream" (which produces a hydraulic sink to control excursions), laboratory wastes, and a restoration stream. Solid wastes include contaminated equipment, resins, filters, and surface soil contaminated by leaks and spills. The primary source of radioactivity is radon offgassing from circulating fluids. Surface reclamation is similar to that for conventional mills. Subsurface (aquifer) restoration involves reducing groundwater constituents to pre-mining levels, usually by circulation/recirculation of uncontaminated native groundwater.

#### STATE REGULATORY AUTHORITY

The Texas Department of Health's authority to regulate all sources of ionizing energy is detailed in the Texas Regulations for Control of Radiation, promulgated in response to the Texas Radiation Control Act (Article 4590f, V.T.C.S.). These regulations are compatible with those of the U. S. Nuclear Regulatory Commission and, as such, form the basis for the contractual agreement to regulate specific sources of ionizing radiation in Texas.

The Railroad Commission of Texas (RRCT) has exclusive jurisdiction for the

regulation of surface mining operations in Texas. Statutory authority specific to uranium recovery stems from the Texas Uranium Surface Mining and Reclamation Act. The TDH and RRCT have overlapping responsibilities in the disposal of uranium-depleted ore tailings.

The Texas Water Commission (TWC), formerly the Texas Department of Water Resources, has primary jurisdiction under the Texas Water Code to prevent contamination of water in the state and to regulate wells other than those regulated by the RRCT. The delegated authority for control of underground injection originates under the Federal Safe Drinking Water Act of 1974.

Financial security is mandated to ensure that funds will be available to the state should it be required to assume responsibility for the reclamation of an open pit mine, aquifer, uranium recovery facility, or tailings impoundment.

The RRCT maintains financial security for all conventional (surface mining) operations. The TDH maintains security for decommissioning the recovery facility proper. The two agencies maintain overlapping security instruments for reclamation of tailings impoundments. In addition to these refundable security instruments, licensees are required to pay into the TDH Radiation and Perpetual Care Fund sufficient funds to cover Agency costs of long-term care and maintenance once an impoundment is deeded to the state.

The TWC is responsible for financial security associated with the plugging and abandonment of injection and disposal wells at in situ facilities. All other security for these facilities is maintained by the TDH. The TWC and the TDH jointly determine the security requirements for aquifer restoration. The TDH assumes primary responsibility for determining the security associated with decommissioning of the facility and reclamation of the land areas to unrestricted release levels.

Acceptable security arrangements include cash or negotiable securities, irrevocable assignment of savings, certificates of deposit, or letters of credit (an instrument executed by the operator and a corporate surety or financial institution with the Agency as beneficiary). Bonding or self-insurance of the operator may be accepted if the operator can demonstrate the existence of a responsible party, an acceptable history of financial solvency, and a high probability of continued operation.

## FINANCIAL SECURITY

The starting point for providing financial security is the license application submitted to the TDH. Each application includes (1) a closure plan for decontamination, decommissioning, and reclamation of the facility at the termination of operations, and (2) cost estimates for closure. These are evaluated by TDH staff, and any needed modifications are made in consultation with the applicant. A financial security instrument based on the mutually agreed cost is executed by the licensee. The security is reviewed annually thereafter by TDH staff. The annual review affords an opportunity to make adjustments for progress made or changes in the closure plan and to account for inflation. Upon satisfactory completion of closure, the Agency releases financial security in conjunction with license termination. Funds deposited with the TDH as security are refunded with accrued interest.

Review of initial security submissions showed clearly that cost-estimating procedures and their accuracy varied greatly. After an extensive study of closure activities for each type of facility, TDH staff developed detailed procedures for cost estimation in late 1982. They also prepared independent closure cost estimates for each licensed facility (excluding aquifer restoration

costs). The chief utility of those estimates was to verify licensees' submissions. The review process subsequently led to the issue, in May 1984, of TDH Regulatory Guide 1.2, entitled "Procedures for Estimating Reclamation Costs for Conventional and In Situ Uranium Recovery Facilities." This guide currently forms the Agency's basis for cost estimation. While strict compliance with the guide is not mandatory, licensees and new applicants are encouraged to follow it. Most current licensees had provided securities before the issuance of the guide; a number of them have started using the guide in submitting revisions as part of the annual review.

#### Technical Basis

Texas regulations specify basic technical requirements for closure of uranium facilities. Significant differences in processing features exist between conventional milling and in situ mining facilities. The following discussion therefore considers each type separately.

##### (a) Conventional Facilities

The initial stage of conventional uranium recovery requires the construction of large open pits (up to 160 acres in size, and 300 ft deep) to expose the ore for subsequent extraction and transport to the processing facility. During reclamation, the operator is required to restore the affected land to a condition resembling the pre-existing one or to a substantially beneficial condition related to previous land uses.

A unique feature of the conventional process is the tailings pond. The usually large tailings pond (up to 260 acres) serves as a receptacle for all solid and liquid wastes generated at the facility. The solid wastes consist primarily of powdered uranium-depleted ore, known as tailings, which nearly equal the quantity of ore mined in volume. This waste is pumped to the tailings pond in the form of a slurry. Tailings contain most of the radium (and thorium) present in the ore, and hence pose a potential long-term radiological hazard.

The rest of the facility includes ore storage pads, ore transportation systems, process plant, drying and packaging buildings, and smaller process ponds or waste ponds. All these require specific decontamination and decommissioning techniques, which are equally applicable to the above-ground facilities at in situ mining projects. In every case, security cost estimates based on a closure plan must be accompanied by supporting site information and drawings, such as dimensions, areas, volumes, distance to waste disposal facility etc., to enable independent verification by TDH staff. Closure requirements include (1) dismantling and decontamination of all equipment, piping, and interconnections; (2) demolition of buildings, foundations, concrete pads, and other fixtures; (3) removal of contaminated soil (and uncontaminated soil, if required) and its disposal at an approved disposal facility; (4) adding topsoil where needed, leveling, and contouring; (5) surface soil preparation and revegetation; and (6) conducting appropriate radiological surveys, analyses, and chemical tests. Salvage of usable equipment or property (for possible resale) is permitted; however, such items must be surveyed by TDH inspectors to ensure complete decontamination. Basic decontamination criteria used for reclaimed surface soils are those specified in 40 CFR 192 for radium (Ra-226): 5 pCi/g averaged over 0-15 cm depth and 100-m<sup>2</sup> area, and 15 pCi/g averaged over succeeding 15-cm layers.

Tailings pond reclamation constitutes a major part of the total closure activity in terms of time required, expense, and potential environmental consequences. Federal regulations (10 CFR 40) require that reclamation to control radiological hazards "be effective for 1000 years to the extent

reasonably achievable, and, in any case, for at least 200 years." Tailings pond reclamation involves the following actions: (1) neutralizing pond fluids as needed and dewatering the pond; (2) covering the tailings with an adequate thickness of carefully selected cover material (layered, compacted soils, or riprap, or combinations); (3) backfilling, contouring, or leveling as necessary to produce a surface configuration that ensures smooth storm water runoff and long-term stability against erosion and water infiltration into the tailings; (4) rebuilding the embankments to provide gentle slopes to ensure long-term stability and erosion resistance; (5) revegetating reclaimed areas with a self-sustaining cover to ensure long-term ecological stability; (6) providing features to limit human and animal access; and (7) providing monitor wells, bench marks, or levels to facilitate long-term monitoring of the site.

Materials used to cover the tailings must be thick enough to ensure radon (Rn-222) emanation rates below the 40 CFR 192 limit of 20 pCi/m<sup>2</sup>-s. Suitable dust suppression measures must also be taken during soil hauling.

Because of the potential hazards associated with tailings ponds, the responsibility for long-term care and maintenance of the reclaimed sites will be assigned to the state upon termination of the license. Licensees will be required to transfer sites to the state free of cost.

#### (b) In Situ Mining Facilities

Acceptable closure plans for in situ operations must cover the process plant site and equipment; above-ground equipment associated with the well fields, waste disposal wells, and aquifer restoration (pumps, piping, tankage, etc.); waste retention/evaporation ponds; well field soils; and aquifer restoration. The cost factors considered and data required for state verification of plans for closing out the plant site and decommissioning other above-ground equipment or facilities are essentially the same as those for conventional mines discussed above.

Close-out of waste retention/evaporation ponds must include removal and disposal of contained fluids and solids (sediments, precipitates, etc.), pond liner, contaminated soil below the liner, and associated equipment (pumps, pipes, leak detection system, etc.). Pond excavations must be backfilled with uncontaminated material (including a layer of topsoil adequate to support plant growth), leveled or appropriately contoured, and revegetated. Proper reclamation of in situ well fields basically involves decontamination, removal and disposal of above-ground equipment, and removal of contaminated soil.

Engineering drawings and other supporting data sufficient to enable the TDH staff to verify a company's pond closure and well field reclamation cost calculations should be submitted as part of the closure plan. Supporting data typically include pond dimensions and capacities, fill and waste material volumes, equipment lists, disposal costs, and revegetation costs. For well field reclamation, results of field surveys, monitoring programs, and spill investigations are useful to establish the actual extent of surface contamination.

The factors considered in cost estimates for aquifer restoration include restoration of the mining aquifer groundwater quality to levels prescribed by the state, decontamination and/or disposal of associated surface equipment, and disposal of restoration-generated byproduct materials. The decontamination/disposal of surface equipment is straightforward, but factors related to aquifer restoration are complex. Therefore, cost estimates for aquifer restoration are the most difficult to verify and probably the least accurate made for in situ mining operations. There are several reasons for this situation.

In the first place, several restoration techniques are available, and selection of a particular strategy depends on site-specific geology, hydrology, and geochemistry. The most commonly used technique in Texas is groundwater sweeping (also called pore volume flushing), in which contaminated water is withdrawn from the well field and uncontaminated water from the surrounding area flows in to replace it. Most other techniques involve modifications of groundwater sweeping or procedures used in combination with sweeping. Possible variations include injection of clean water around the periphery of the well field; injection of chemicals or chemically treated water to reduce contaminant concentrations; treatment of withdrawn water at the surface (as by reverse osmosis or electrodialysis) and injection of the clean water stream.

Data available when initial restoration cost estimates are made (ideally before mining) are limited. The selection of a restoration technique is therefore based on incomplete exploratory data and perhaps on results of pilot-scale restoration "demonstrations." As more experience is gained during mining and the initial restoration attempts, the restoration technique may well change from that considered in cost estimates and covered by financial security.

Last, the level of restoration required is known only in a general sense prior to actual restoration, when cost estimates are made. Current TDH and TWC policies require that groundwater quality in a mined aquifer be restored at least to the extent that the water can be used for any purpose to which it was suited prior to mining. The exact contaminant levels to be achieved and the length of time restoration efforts must continue are set by the state only after a substantial restoration effort has already been made. The initial cost estimates, therefore, are really a best guess by the state of effort that will be required and of how effective and efficient the chosen restoration method will prove to be.

Virtually all restoration techniques generate large volumes of radiologically contaminated groundwater which must be safely disposed of. Historically, the most popular disposal options have been the use of a deep disposal well and/or evaporation ponds. More recently, however, emphasis is shifting toward surface discharge or irrigation, both requiring some level of pre-treatment. The anticipated costs for these latter options are directly related to the degree of required pre-treatment. The state is currently reviewing the potential environmental impacts of these approaches but has yet to define pre-treatment requirements. Accordingly, cost estimates for fluid disposal may also be uncertain.

Certain information is required for evaluation of aquifer restoration cost estimates regardless of which restoration and disposal options are chosen. The most significant requirements are the physical dimensions and porosities of production zones; number of pore volumes required and documentation supporting that estimate; cost of a monitor well sampling program to track restoration progress and verify final restoration; cost of chemicals used in treatment; initial and maintenance costs for special equipment such as reverse osmosis units; fluid disposal costs; labor costs; and power consumption costs.

#### Cost Estimation Basis

The TDH Regulatory Guide 1.2 outlines procedures, quantitative measures, physical constants, and other assumptions to be used in cost estimation. These were derived from published and solicited information, discussions with licensees, informed assumptions relating to decontamination and decommissioning requirements, and average physical properties. The procedures and assumptions were designed to fulfill the basic regulatory criteria relating to radiological decontamination.

All cost estimates must be based on using an independent contractor for carrying out closure. No credit is allowed for salvage or resale of any equipment or property. No credit is allowed for potential earnings from sale of any products generated during or after closure, for example, have recovered from reclaimed land. The base year to which the costs refer must be specified along with projections accounting for inflation. Cost data and sources used must be provided. Site-specific information that warrants changes in assumptions or procedures must be justified.

#### (a) Conventional Facilities

The general quantitative assumptions and requirements for surface facilities (including those at in situ locations) state that (1) concrete pads are 1 ft thick under the process plant and 4 inches thick under other structures; (2) a minimum of 0.5 ft of soil is contaminated under the process plant pad and 1 ft under the ore storage pad; (3) soil extending 0.5 ft below a pond liner (excluding tailings pond) is contaminated; and (4) the average density of soil is 100 lb/ft<sup>3</sup> and that of concrete, 180 lb/ft<sup>3</sup>.

There are special requirements for tailings pond closure cost estimation: (1) The tailings should be covered with a minimum of 3 meters of uncontaminated earth material, including a layer of topsoil to support vegetation, and any reduction in cover thickness should be justified. (2) The embankment should be leveled and contoured to minimize erosion, with final slopes not exceeding 5:1 horizontal to vertical; steeper slopes should be justified; and a rock (riprap) cover should be provided in embankment areas where vegetation growth is unlikely and where storm water runoff is directed.

#### (b) In Situ Mining Facilities

Although all adequate in situ mining closure plans must consider the general elements outlined above, each plan is site-specific and unique. In order to ensure some basis for comparison, however, TDH Regulatory Guide 1.2 specifies certain assumptions, physical constants, calculation methods, etc., that must be used to determine closure costs for all sites. Assumptions regarding reclamation of the process plant or similar in situ facilities (satellite plants) are generally the same as for conventional operations. Reclamation of evaporation/waste retention ponds assumes that 0.5 ft of soil beneath the pond liner is contaminated and that necessary topsoil and backfill material are stockpiled nearby. The TDH recognizes that actual well field soil contamination will not be uniform, but will vary in both location and depth. It is assumed, however, that the volume of soil to be removed is equivalent to the removal of the top 2 cm over the entire well field area. Soil density is again considered to be 100 lb/ft<sup>3</sup>, and no additional topsoil is necessary.

For aquifer restoration, fluid volume calculations are critical. A pore volume is defined by:

PV = CATP, where

PV = pore volume in gallons

A = acreage of production area + 10%

T = average contacted thickness of production zone

P = average production zone porosity, and

C = conversion constant as appropriate to convert volume to gallons.

Additional assumptions are that achievement of restoration is verified according to sampling procedures specified by the TWC, and that estimated costs cannot include credit for capital generated from the use of restoration byproduct materials.

## Alternatives

The TDH has adopted a flexible attitude towards evaluation of facility closure plans and cost estimates. Alternatives are acceptable if they satisfy the regulatory criteria for decontamination mentioned earlier and are adequately supported. Alternatives encountered so far generally fall into four classes or levels shown in Table 1.

Table 1 Alternative Considerations in Decontamination and Decommissioning

Class or Level	Typical Example	Typical Alternative Proposed
Overall Reclamation Concept	Reclaimed surface of conventional mill tailings impoundment	(1) Well drained vegetated surface (2) Partially ponded surface
	All reclaimed sites	Final disposition of land, ponds
Treatment Technology	Treatment of conventional mill tailings fluid	(1) Neutralization with caliche (2) No treatment
	In situ well field aquifer restoration	(1) Groundwater sweep (2) Combinations with partial recirculation after treatment
	Radium removal from restoration wastewater at in situ mines	(1) Barium complexing (2) Ion exchange
Waste Disposal	Ultimate disposal of restoration wastewater from in situ mining (with or without radium removal)	(1) Deep underground injection (2) Surface discharge (3) Land application (irrigation)
Determination of Details	Conventional mill tailings impoundment	Selection of both cover and final embankment slopes
	All reclaimed sites	Vegetation selection

Selection of an alternative requires in-depth technical evaluation to minimize long-term care and maintenance. It may require specifying decontamination limits based on the ALARA principle at control levels that are below the regulatory limits. In some instances cost benefit analysis may be used.

At many sites, excepting mill tailing ponds, the nature of final reclamation will be influenced by expectations of landowners from whom the land was leased. Technological choices typically cover (1) processes for uranium tailings neutralization, (2) method for restoration of in situ mined aquifers, and (3) methods of wastewater treatment. The choice of wastewater disposal methods has raised questions requiring equitable compromise between direct cost benefits and the broader interest of minimizing environmental contamination.

Certain alternatives do not yet demand attention, but will need to be addressed in the future. An example is the method of long-term disposal of

solid wastes generated at in situ facilities, i.e. whether to allow continuation of the current practice of disposal in existing uranium mill tailings ponds, or to find other, perhaps out-of-state, disposal alternatives.

## Annual Review

The chief purpose of an annual review is to keep the security amounts current, so that full decontamination and decommissioning can be accomplished in the event of an operator default. The costs are adjusted for inflation by using a factor derived from the annual implicit price deflator for the gross national product. Security reductions and appropriate releases can be made only after affected areas have been inspected and formally released by the respective regulatory agency.

## TEXAS EXPERIENCE

The three existing state-licensed conventional mining and milling facilities in Texas are covered by a total financial security of \$15.4 million (1985): \$7.2 million, \$5.2 million, and \$3.0 million. One of these is still in production, and the other two are at different stages of closure. A fourth conventional facility, which terminated operations in the early 1970's, falls under the federal Uranium Mill Tailings Remedial Action program. Currently, nine in situ uranium facilities are in active production with a total security of \$65.2 million (1984 and 1985). Four in situ facilities are engaged full time in aquifer restoration, accounting for \$3.2 million in security (1984). Two more are being closed out, but their security of about \$2 million has not yet been released. The largest security for an in situ facility is \$22.1 million, while the smallest is \$0.5 million. A breakdown by the type of security instrument includes ten performance bonds, two personal bonds, one trust agreement, and two letters of credit. (In two instances, a security instrument covers more than one facility.)

For the most part security estimates submitted for uranium recovery operations have been in accord with the framework suggested in Regulatory Guide 1.2. There have, however, been a few problems.

By far the most common problem has been the use of inaccurate data, especially regarding the physical dimensions of well fields and production zones and hence the size of a pore volume. A few operators based security calculations on pore volumes that were much smaller (in some cases five times smaller) than values on file with both the TDH and the TWC. These differences are significant, of course, because the most costly elements of in situ mine closure, aquifer restoration and fluid disposal, are directly related to pore volume size. In such cases the TDH requires detailed mapping, cross sections, and well data to verify pore volume calculations. A few companies used a very low total pore volume requirement without providing adequate test data or other supporting documentation.

Several companies have insisted on claiming credit for the use of restoration-generated byproduct material. One operator initially submitted a zero cost estimate based on the fact that oil income generated by a secondary recovery operation using restoration water would totally offset costs of aquifer restoration. The credit was not considered, and to date the project has served as a means of disposing of waste fluid, but oil recovery has been unsuccessful.

A number of operators changed their restoration technique, disposal option, or both in their security estimate submissions. This presents a problem when

the new options have yet to receive authorization from state regulatory agencies. In those cases, operators are requested to re-submit two estimates, one based on the new options, and one assuming that state authorization of these is denied.

With two conventional milling facilities, a large difference was observed between the licensees' closure cost estimates and those prepared by TDH staff, the latter being higher in each case by over \$1 million. This discrepancy was traced in large part to a single step in the cost calculation: the cost of bulk soil hauling necessary for filling and reclaiming tailings ponds. The basis for the two estimates were different, although both assumed the use of wheel tractor scrapers of 32-yd<sup>3</sup> capacity. The TDH staff had assumed three scraper trips per hour over an assumed average hauling distance of one mile. This assumption was made before definitive information was available on the scraper time cycle data, and resulted in a soil hauling cost of about \$1.60 per bank cubic yard (BCY). The licensees used a more realistic time cycle of five to six trips per hour over the same distance, resulting in an average cost of about \$1.25/BCY. After examining the actual performance data of scrapers over several months and inspecting the actual hauling operations at both sites, the Agency accepted the lower cost.

## CONCLUSIONS

The Texas Department of Health allows considerable flexibility in developing closure plans for facility decontamination, decommissioning, and reclamation. The Agency has encountered many technical and policy questions, most of which have been resolved. Some alternatives that pose short-term and long-term policy questions requiring final resolution include disposal alternatives for aquifer restoration fluids, and the means of ultimate disposal of solid wastes generated at in situ uranium recovery facilities. Regardless of these issues, the financial security itself may be developed from rather fixed closure cost estimation bases. The TDH Regulatory Guide 1.2 facilitates timely estimation of closure costs on a conservative and equitable basis.

## ACKNOWLEDGEMENTS

The opinions expressed in this paper substantially represent those of the technical staff of the Environmental Assessment Branch, Division of Environmental Programs, Joseph F. Thiel, Director. The TDH Regulatory Guide 1.2 and the closure cost estimates referred to in the paper were prepared by one of the authors (SDE) along with Marilyn J. Preusse, Ph.D., Environmental Surveillance Branch, of the Division. Technical assistance was provided by Mary P. Shannon, Ph.D.

# THE USE OF SITE GROUNDWATER INVESTIGATIONS IN THE DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR FACILITIES

Stephen A. Molello  
GPU Nuclear Corporation  
Environmental Controls Department  
P. O. Box 388, U. S. Route #9  
Forked River, New Jersey 08731

## Abstract

A major component of the environmental analysis process during the decontamination and decommissioning of nuclear facilities is the assessment of the subsurface for radiological contamination. The maximum value for efforts and dollars expended during such assessments is achieved when the site groundwater investigation is conducted in a carefully planned and systematic manner. Guidelines and instructions for the accomplishment of these investigations are presented, including: 1. An introduction explaining the needs for and uses of such investigations, 2. A "step by step" explanation of a desirable format for the design of site groundwater investigation work plans, and 3. The utilization of data obtained in the implementation of remedial or corrective actions, if necessary. Several topics related to these three areas are also discussed, including public relations and litigation.

## INTRODUCTION AND PURPOSES

A major component of the environmental analysis process during the decontamination and decommissioning (D & D) of nuclear facilities is the assessment of the subsurface for radiological contamination. This requires high quality sampling and analysis of soil and groundwater underlying the site, as well as an integration of data obtained with the regional hydrogeologic picture. The maximum value for efforts and dollars expended is achieved when the groundwater investigation is carefully planned and systematically accomplished. A work plan is presented to explain some of the major phases of such projects.

There are many potential contamination pathways to groundwater from nuclear facilities. Some of these, such as a slowly leaking buried pipe network, may be unknown to site operators during facility history. Others, such as lagoons and shallow land burial sites, have been commonly used for storage and disposal of materials with low levels of radioactive contamination. A site groundwater study performed during D & D operations will determine the nature and extent of subsurface radiological contamination. This information is useful in defining remedial and corrective actions and as an aid in determining future activities and uses for the site. In addition, the data obtained will be used to determine the dose to the affected populace from using groundwater for drinking or irrigation. The legal liability for the underlying groundwaters should not be overlooked. A recent study by the Edison Electric Institute (1.) has noted that groundwater pollution may be the major environmental issue of coming decades.

## DESIGN OF ONSITE GROUNDWATER INVESTIGATIONS

### Prerequisites

There are a number of prerequisites that should be fulfilled prior to the actual initiation of work.

A complete understanding of the regional groundwater flow system should be obtained. This type of information is generally available from regional offices of the United States Geological Survey, state geological surveys or local environmental departments. The implications of site specific data can only be fully known when the rates, directions, and uses of groundwater in the surrounding region are understood.

Early in the D & D process, budget consideration should be given to groundwater related work. Costs will be determined by the specific geology and facilities of the site. A general "rule of thumb" for unconsolidated materials such as sand, silt and clay, is to allow a total finished cost of 150 dollars per foot of well needed. Thus, a site in a sandy geologic unit that required fifty monitoring wells averaging 30 feet deep would cost about 225,000 dollars.

It is important that the planning and execution of the project be properly timed relative to the D & D operation as a whole. If, for example, an entire building or storage tank is to be disassembled and removed, it may be necessary to time some of the well installations to allow them to be placed in the vacated areas after these structures are gone.

Public and media relations efforts should be accomplished before initiating the project. It is especially important to make the public aware of the program prior to the discovery of any contamination. Personnel in the public and media relations departments should be notified of the study and kept informed as it develops. Periodic public meetings and media briefings on the D & D operation should also include updates on groundwater.

Communications channels within the organization that is coordinating the project should be established prior to beginning the work. Key personnel that need to be aware of major issues, such as the locations of wells or the discovery of contamination, should be notified and made aware of the overall plan.

Personnel involved in the project should be aware of the need for top quality documentation. The framework for a complete and retrievable data and information base should be established and implemented when the project begins.

The last prerequisite is the preparation of a scope of work document. This would take the form of a request for proposal if the project is to be subcontracted. The specific goals and objectives that are dictated by the unique aspects of the site should be written into this document. Also, the minimum expected information products should be specified. These may include groundwater flow maps, aerial photography, topographic mapping, interim and final reports, well logs, specialized data analysis, the use of computer models to verify flow paths and so on.

## Steps Toward Accomplishment

The first step in the accomplishment of this type of project is the analysis of historical data. This is primarily a research task and the source documents include all groundwater information previously generated in construction specifications, environmental reports, safety analysis reports, and so forth. This information will be diverse and may include well log data sheets. In addition, any local dewatering or injection operations should be considered as these can markedly influence site groundwater flow. After collection, these data should be interpreted by a project geologist to preliminarily define site hydrogeology.

There is an abundance of literature available to provide project management and staff with information to assist them in coordinating the task at hand. A recent groundwater text by Bowen (2) provides a background in fundamental concepts. In addition, several detailed field guides have been published by organizations with extensive groundwater experience such as General Electric Company (3.) and the United States Air Force (4.). These guides provide a variety of instructional case studies. An introduction to the overall national picture on groundwater contamination by Pye, Patrick and Quarles (5.) will help put your situation into perspective, while an analysis of naturally occurring isotopes in the hydrosphere by Ferronsky and Polyakov (6.) may assist in identifying radioactivity that is not related to your facility's operations. NCRP 76 (7.) and NUREG 3332 (8.) contain information that will be useful for dose assessment from the groundwater pathway.

The second step is the site survey. It is important to use the most current topographic maps available that have been constructed by photogrammetric means. Maps or drawings created from sources other than aerial photography are often in error. It may be necessary, and is probably generally advisable prior to D & D operations, to produce new aerial photographs for use in generating a site topographic map. Once updated maps are available, an actual touring of the site and interviews with key site personnel can occur. During site visits, it is essential to learn as much about historical operating practices as possible. Some potential sources of groundwater contamination are: process buildings with stored liquids, under or above ground tanks and piping systems, outdoor decontamination pads, loading pads, temporary storage areas, shallow land burial sites, lagoons and settling ponds, plant roads and rail sidings, areas of heavy stack emission deposition, injection wells, and prior spill sites. Older and more complex sites may require the use of nonintrusive investigative techniques such as ground penetrating radar and resistivity surveys to inventory potential sources of contamination underground.

The third step is the preparation of an interim report. This report will integrate all of the site historical hydrogeologic data with that of the region, as well as identify and discuss the specific areas of concern at the site. A recommended well and lysimeter network should be provided (a lysimeter is a device to sample moisture in the unsaturated zone above the water table). The main goal of the well and lysimeter network is to define the nature and extent of subsurface contamination. This interim report should receive a rigorous review by appropriate personnel which may result in minor modifications and adjustments to the proposed network.

The installation of wells and lysimeters is the fourth step. Extreme care must be taken to locate and avoid areas with buried utility lines such as electrical cables. Generally, continuous core sampling is used in this type of investigation because it reduces the chances of inadvertently missing areas of contamination. Groundwater samples are also collected, and it is important to address conditions specific to the contaminants to be expected. For example, Mackay, Roberts and Cherry (9.) recently described a variety of possible flaws in monitoring well networks that resulted in the failure to sample those portions of aquifers containing organic contaminants. Cutting fluids or oils bearing radionuclides act as organics when transported by groundwater. The density and miscibility of these fluids will determine their relative elevation within the aquifer. During the analysis of water and core samples for radiological contamination, initial efforts should cover as many possibilities as would be expected from prior activities at that site. For most nuclear facilities, this means a preliminary scan for gross alpha activity, gross beta activity, gamma emitting isotopes, and tritium. More detailed measurements and analyses will follow, as necessary. Geotechnical analyses on core samples should also be conducted to obtain information on permeability, porosity, and geochemistry. In addition, pump, slug and dispersion tests on newly installed wells will refine the analysis of site hydrogeology provided in the interim report. It is useful to finish well and lysimeter installations with crush proof casings and caps that will allow for safe passage by vehicles and heavy equipment. In addition, air and watertight sanitary seals should be placed inside the casings to prevent debris from the surface from contaminating the groundwater.

The last step is the final report assembly. Adequate review time should be permitted for the appropriate personnel to comment on a draft of this report prior to approval. The final report should include a summary of site hydrogeology and its relationship to that of the surrounding region, groundwater flow maps, presentation and analysis of radiological and geotechnical data, well logs, maps of the subsurface geology, and recommendations for further sampling. The final report will provide data to be used in other phases of the D & D project. For example, groundwater flow information coupled with radionuclide concentrations can be used to calculate dose to the affected populace. The degree or absence of environmental impact can then be verified.

Remedial actions or decontamination efforts can be recommended. For example, heavily contaminated soil that may act as a leachable long term source can be removed for recycling or disposal as radioactive waste.

All or part of the preceding approach can be useful in planning for a site groundwater investigation. In any case, this part of the environmental analysis process deserves the same care, precision and attention to detail as the rest of the D & D process.

#### REFERENCES CITED

1. Edison Electric Institute, Groundwater: Emerging Issues for the Electric Utility Industry, Edison Electric Institute, Washington, D. C., 1985.
2. Robert Bowen, Groundwater, Halsted Press Division of John Wiley and Sons, New York, 1980.

3. L. G. Everett, Groundwater Monitoring, General Electric Company, Schenectady, New York, 1980.
4. J. H. Guswa, W. J. Lyman, A. S. Donigian, Jr., T. Y. R. Lo, E. W. Shanahan, Groundwater Contamination and Emergency Response Guide, Noyes Publications, Park Ridge, New Jersey, 1984.
5. V. I. Pye, R. Patrick, and J. Quarles, Groundwater Contamination in the United States, University of Pennsylvania Press, Philadelphia, Pennsylvania, 1983.
6. J. I. Ferronsky and V. A. Polyakov, translated by S. V. Ferronsky, Environmental Isotopes in the Hydrosphere, John Wiley and Sons, New York, 1982.
7. National Council on Radiation Protection and Measurements, NCRP Report No. 76, Radiological Assessment: Predicting the Transport, Bioaccumulation, and Uptake by Man of Radionuclides Released to the Environment. NCRP, Bethesda, Maryland, 1984.
8. J. E. Till and H. R. Meyer Radiological Assessment A Textbook on Environmental Dose Analysis (NUREG/CR 3332) USNRC, Washington, D. C., 1983.
9. D. M. Mackay, P. V. Roberts, and J. A. Cherry, "Transport of Organic Contaminants in Groundwater," Environmental Science and Technology, Vol. 19, No. 5. pp. 384-392, May, 1985.

#### SUPPLEMENTAL REFERENCES

1. T. F. Zimmie and C. O. Riggs, Permeability and Groundwater Contaminant Transport, American Society for Testing and Materials (ASTM Special Technical Publication 746), Philadelphia, Pennsylvania, 1981.
2. F. van der Leeden, Geraghty and Miller's Groundwater Bibliography, Water Information Center Incorporated, Syosset, N. Y., Third Edition, 1983.

COMPARISONS OF PASSIVE ENVIRONMENTAL RADON MONITORS (PERMs) AND  
TRACK ETCH (R) RADON DETECTORS DURING REMEDIAL  
ACTIONS AT THE NIAGARA FALLS STORAGE SITE

William C. Borden and Mark E. Kaye \*  
Bechtel National, Inc., P.O. Box 350, Oak Ridge, TN 37830

ABSTRACT

During recent remedial actions for the U.S. Department of Energy at its Niagara Falls Storage Site, in excess of 6,000 curies of radioactive materials (thorium-230 and radium-226 in equilibrium with its particulate daughters) were slurry transferred approximately one mile to an engineered waste storage facility. During this transfer, radon-222 was released to the atmosphere. Off-site concentrations of radon-222 were monitored throughout a year's period with 30 PERM and 29 Track Etch (R) detectors. Of these 59 monitoring devices, 38 were arranged as paired stations providing 266 data points for comparative assessments. Detailed attention is given to environmental and topographical conditions that significantly effect detector performance, and that are unique to the north eastern United States. Experimental data is presented that quantifies effects of climate changes, as well as detector response and integration capabilities. Conclusions point to the need for significant programmatic and siting changes.

ENVIRONMENTAL MONITORING REQUIREMENTS DURING REMEDIAL ACTION  
AND STABILIZATION OF URANIUM MILL TAILINGS (UMTRA PROJECT)

D.J. Skinner, H.R. Meyer and L.G. Hoffman, Chem-Nuclear Systems, Inc.  
P.O. Box 9136, Albuquerque, NM 87119

ABSTRACT

The Department of Energy is conducting remedial action at uranium mill sites which supplied uranium to the U.S. government under the Manhattan Project. There are requirements to conduct an environmental monitoring program during actual remedial action. The purpose of the environmental monitoring program is to provide information and assurance to the DOE and the public that environmental releases, due to remedial action activities, are within applicable regulations and guidelines.

A discussion of the generic program requirements including: radon monitoring, radionuclide particulate monitoring, water monitoring and environmental gamma dose assessment are reviewed. Specific results for the Canonsburg, PA and Shiprock, NM UMTRA sites, are briefly presented.

INTRODUCTION

During the 1940's through the 1960's the Federal government contracted with many private uranium mining/milling companies. The uranium product, (yellow cake,  $U_3O_8$ ) was utilized by the government, after refinement, for nuclear weapons development, national defense programs (Manhattan Project) and research programs. The milling operations produced a waste product termed tailings or residual radioactive material in copious quantities, since only about four pounds of usable uranium product were produced for every ton of ore processed. The tailings material contains almost all the same radioactive elements as the naturally occurring uranium ore, except that the uranium has been extracted and concentrated as a usable product.

When a majority of the Federal contracts were phased out (late 1960's), most of the milling operations shut down. At that time, little was known about the long term health effects of low level residual radioactive materials, consequently the 24 tailings piles were abandoned (Figure 1). No significant attempt was made to protect the public or the environment because there was no clear, epidemiological evidence that a potential health problem existed.

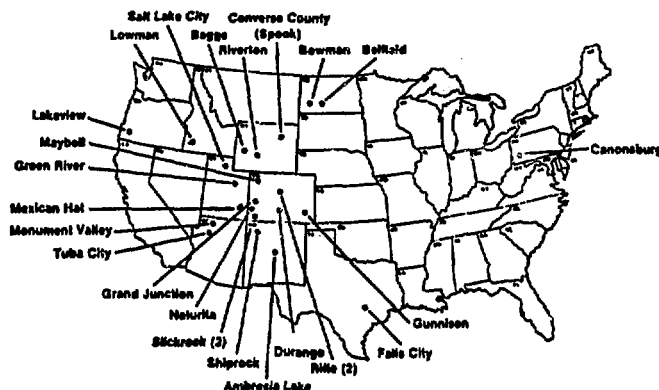


Figure 1.0 Locations of UMTRA Inactive Uranium Mill Sites.

The UMTRA Project is a program administered by the USDOE to stabilize these 24 designated tailings sites. The DOE has environmental monitoring guidelines which apply to all DOE facilities and have been implemented on the UMTRA project. Since environmental releases are minimal on this project, the environmental data collected represent a data base for public information as well as a demonstrated compliance with DOE environmental orders.

#### Generic Environmental Monitoring Program

The generic UMTRA environmental monitoring program includes requirements for radon monitoring, gross alpha air particulate monitoring, water monitoring and external dose monitoring. The general guidelines are published in a program document (DOE 85a). Implementation of the general guidelines involves evaluation of population distribution, meteorological data, availability of electrical power, hydrogeological considerations and surface water drainage patterns.

At an early stage in the planning process a local community task force becomes involved in vocalizing concerns of the community. Discussions between the DOE and the task force typically include items such as: locations of monitoring stations, information and data to be available to the community and frequencies of reporting the data.

#### Radon monitoring

Radon monitoring is accomplished with a real time radon gas monitor (RGM). These Lucas-George cell type instruments are available from Eberline Corporation. The RGM's provide 24 hour average radon concentration values (pCi/l) with hourly radon concentrations printed out on thermal tapes. The RGM's are calibrated, annually, at Mound Laboratories in a radon chamber. Instrument background determinations are made at the mill site, on a monthly frequency, to track potential radon daughter build-up and/or potential particulate contamination during the construction activities.

The general requirements are for one radon monitoring station upwind and one downwind at the mill boundary. In addition, a background (upwind and distant) and a community monitoring station are selected based on population density and proximity to the site. An RGM is also placed at the nearest downwind residence for real time maximum probable dose assessment. The total number of RGM's required is five. Additional RGM's may be placed at the discretion of the DOE, based on task force input and availability of funds.

The RGM provides 24 hour radon concentration data which can be utilized to indicate potential excessive radon releases into the community. Since the DOE orders allow annual averaging of radon values, an action limit of 3.0 pCi/l for one week can be utilized as an action level to assure the community that no excessive releases will occur during the remedial action. If radon concentration values approach this weekly limit, construction activities could be modified by: 1) reducing the amount of tailings materials exposed or 2) wetting the tailings material with more water. Either of these actions would result in a lower diffusion rate of radon gas. In fact, however, no such actions have been necessary to data at either the Canonsburg, PA or Shiprock, NM sites.

## Air particulate monitoring

Air particulate sampling is conducted on a continuous basis, during periods of major remedial action activities, with an Eberline RAS-1 air sampling pump. Samples are collected on a frequency of 3-4 days (due to MDA requirements for Th-230) and counted, onsite for gross alpha activity, with an FC-2 gas flow proportional counter. The gross alpha (24-48 hour decay) results determined onsite, are compared with the Th-230 soluble concentration guide limits in air ( $8 \times 10^{-14}$  uCi/ml). On a quarterly basis, air samples are composited, by station, sent to an offsite radiochemistry laboratory and analyzed for Th-230 and Ra-226.

The locations of air sampling monitors are determined by evaluation of wind direction, speed and frequency data and population distribution around the mill site. In general, four (4) boundary monitoring stations are chosen with one upwind, one downwind and the other two in major cross wind directions. Also three (3) air particulate stations are located as follows: one at the nearest downwind residence (maximum probable dose), one at a community center and one at a background location (upwind and distant).

## Water monitoring

All of the mill tailings sites on the UMTRA project have been well characterized and studied (DOEb and DOEc). As part of the characterization program ground water wells (deep and shallow aquifers) were installed. Seven to nine ground water monitor wells are selected for pre-operational, quarterly and post-operational sampling. Selection of the wells is based on hydrological considerations including direction of ground water movement with respect to the mill site. In general, one up gradient, one down gradient, one background and four to six other wells at the site boundary are chosen for routine sampling.

In addition to ground water well sampling, surface water likely to be influenced by remedial action activities, is also sampled (pre-operationally, quarterly and post-operationally). A duplicate set of surface water samples is collected upstream and downstream from the mill site, for radiological analyses.

Radiological parameters are utilized as indicators of changes in ground water contamination levels, which may be potentially caused by remedial action activities. Water samples are analyzed pre and post-operationally for gross alpha, Beta-gamma U-Nat, Ra-226, Th-230, Pb-210 and Po-210. On a quarterly frequency, Ra-226 and Th-230 are routinely analyzed. EDA Laboratories (Denver, CO) is utilized for all radiochemical analyses.

## Environmental gamma dose monitoring

Since large volumes of contaminated materials are being redistributed, moved and encapsulated on the UMTRA project, thermoluminescent dosimeters ( $\text{CaF}_2$ ) are utilized to measure integrated gamma dose for the duration of remedial action. Five replicate chips per location are utilized to generate a (mean) dose value. Usually TLD's are located at each air particulate and/or radon monitoring station with boundary locations supplemented by 3-4 additional TLD locations. In addition, a control TLD is kept offsite at the HP manager's home, to give an indication of background dose for the area.

## RESULTS

### Air Particulate and Radon

Specific locations of air particulate (AP) and radon (RGM) monitoring stations are shown in Figures 2.0 A and B (Shiprock, NM and Canonsburg, PA respectively).

Preliminary (1984) air particulate and radon concentration results have been previously published (SK85 and ME85) for the Canonsburg, PA mill site. The 1985 data for the Shiprock and Canonsburg sites are presented in Figures 3.0 and 4.0 A and B. Average to date values are shown in all radon plots. Monthly average air particulate and gross alpha concentrations are shown in Figures 4.0 A and B.

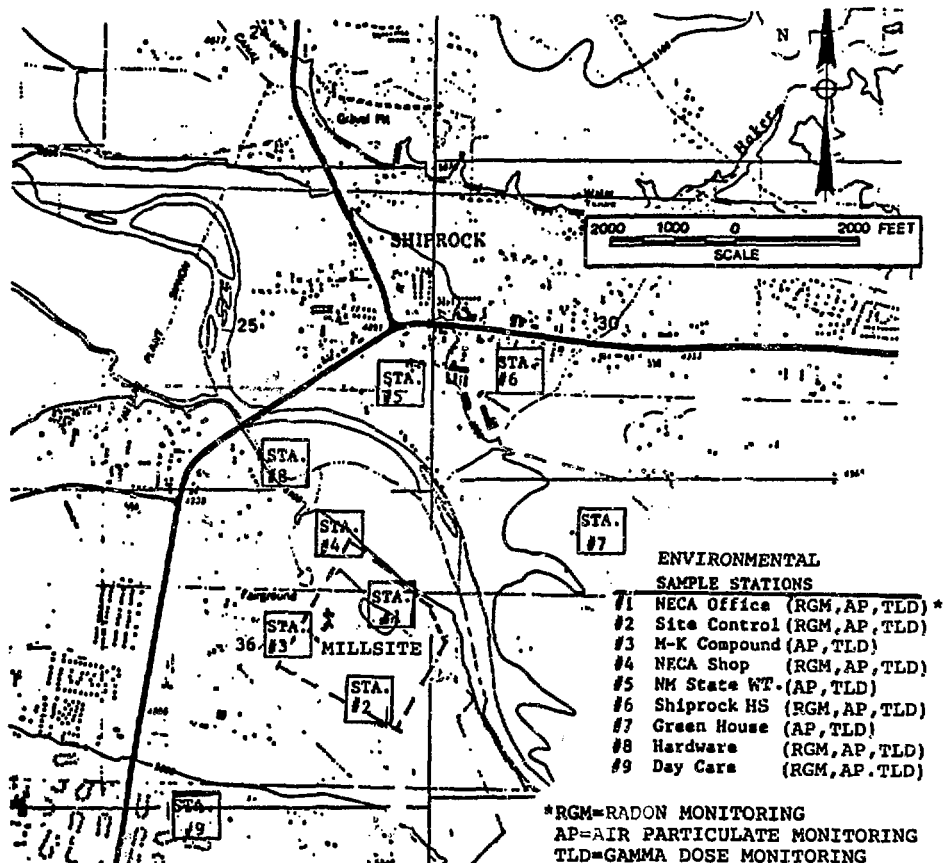
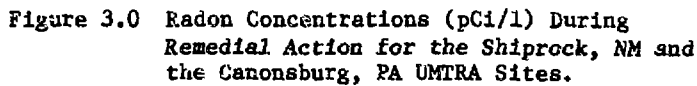
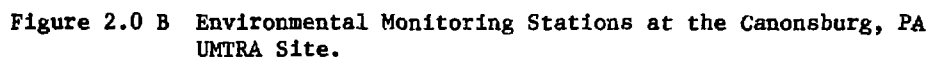
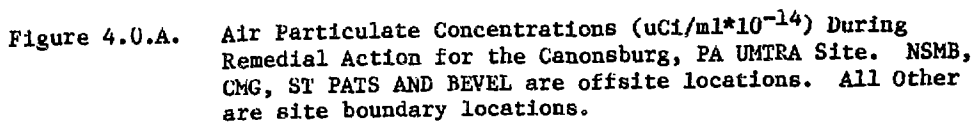


Figure 2.0.A Environmental Monitoring Stations at the Shiprock, NM UMTA site.





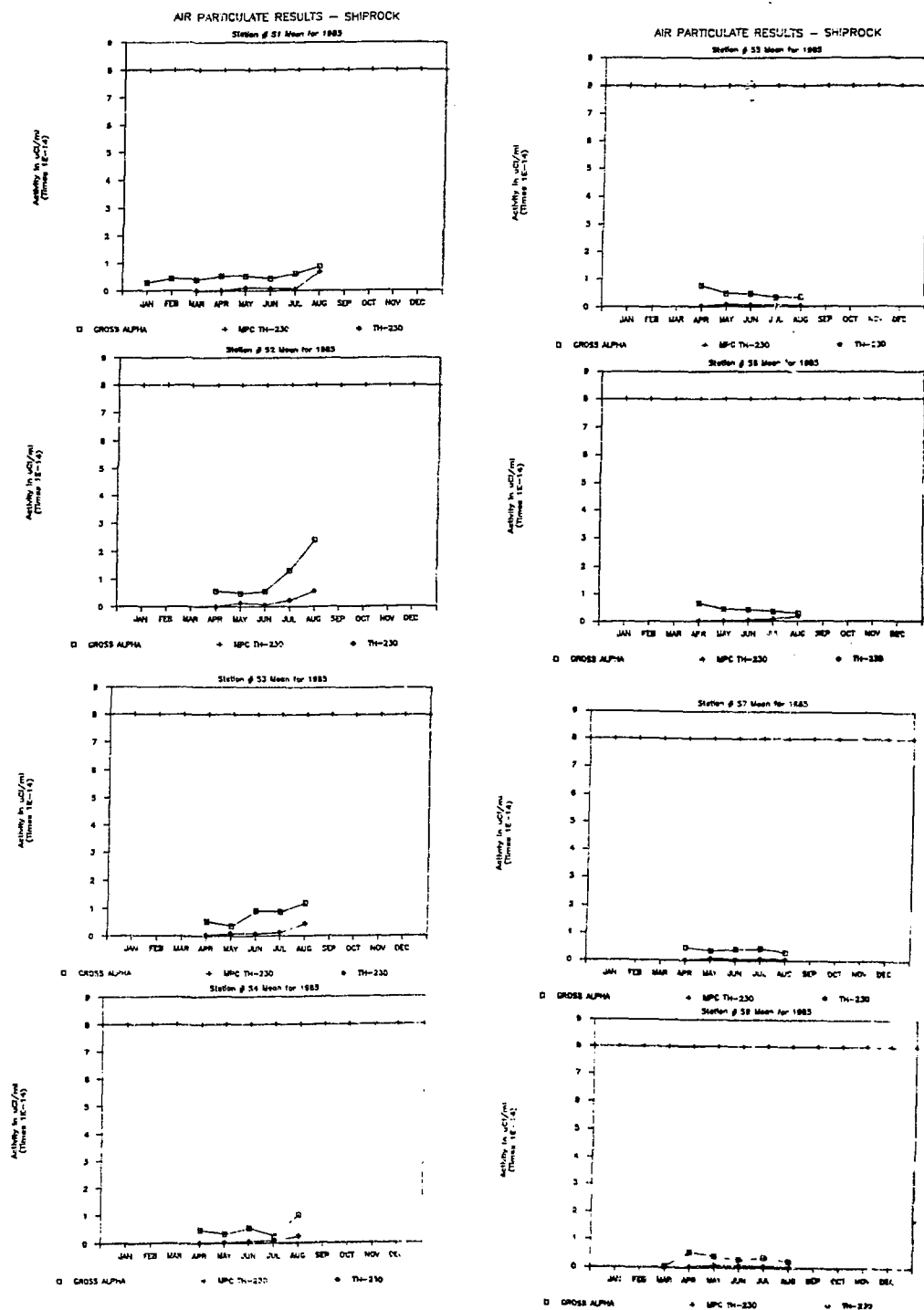


Figure 4.0.B. Air Particulate Concentrations ( $\mu\text{Ci}/\text{ml} \times 10^{-14}$ ) During Remedial Action for the Shiprock, NM UMTRA Site. Stations 5-9 are offsite locations. All others (1-4) are boundary or onsite locations.

Radon concentration guide limits require annual average values, for unrestricted areas, to be controlled to levels below 3.0 Ci/l.

Figure 3.0 shows that even during periods of heavy construction activities, DOE radon concentration guides were continually met for both locations.

Th-230 is one of the most hazardous radionuclides in the uranium U decay product chain. It is anticipated that Th-230 will be the significant inhalation (particulate) hazard in the tailings material. An onsite analysis for Th-230 in air (isotopic radiochemistry) is not practical on the UMTRA project, so onsite gross alpha air particulate concentrations were determined (5 day decay), as an estimator of potential Th-230 environmental releases. Since no information was available regarding the form or solubility of Th-230 in the tailings material, the conservative soluble limit concentration guide ( $8 \times 10^{-14}$  uCi/ml) has been utilized as the standard.

Both gross alpha onsite and Th-230 (laboratory) results are plotted for the Shiprock and Canonsburg sites (Figures 4.0.A and 4.0.B). The average monthly values indicate gross alpha air concentrations were always well within the DOE Th-230 concentration guides, even when compliance was based on gross alpha onsite results.

#### Water Concentration

The locations for ground water well monitoring at Shiprock and Canonsburg are presented in Figure 5.0 A and B.

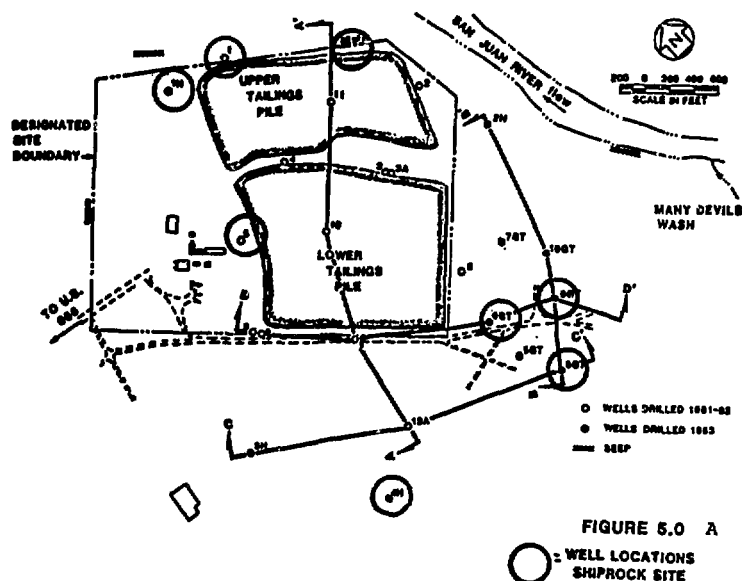


Figure 5.0.A. Groundwater Monitor Well Locations at Shiprock, NM.



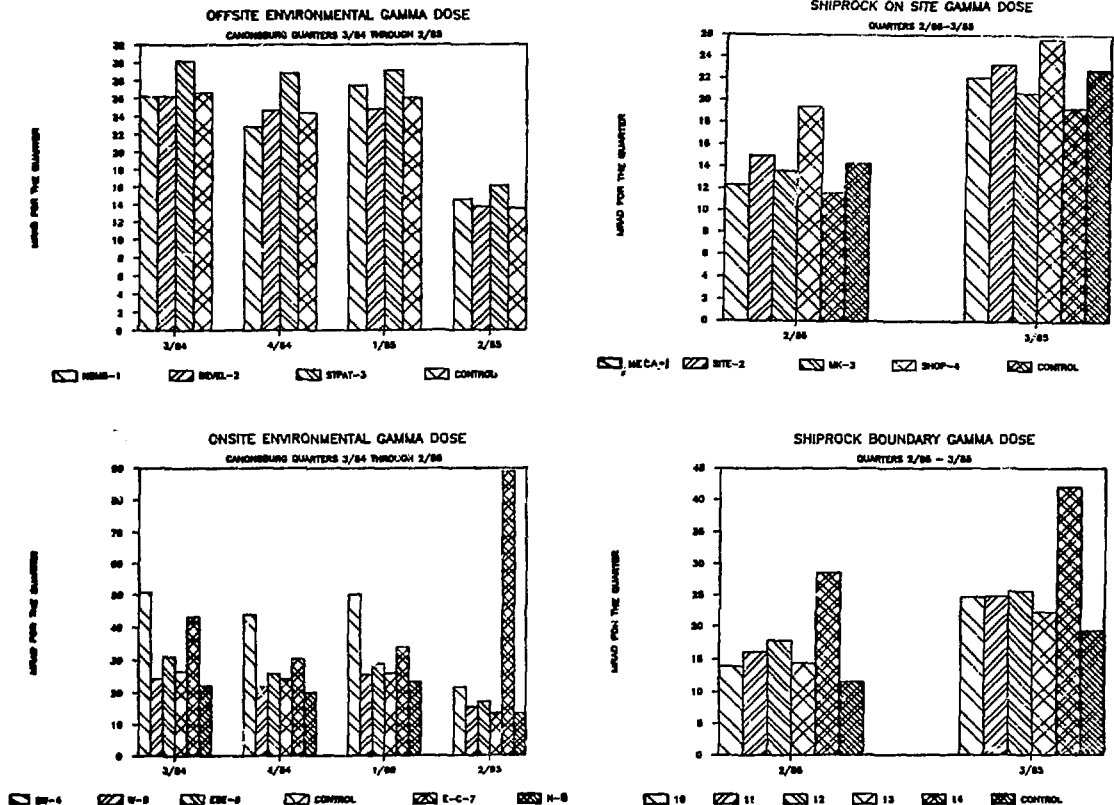


Figure 6.0 Canonsburg, PA and Shiprock, NM Environmental Gamma Measurements (MRAD/Quarter). First Quarter Shown was Pre-Operational Measurements.

Both onsite (boundary) and offsite (community) data are presented (Figure 6.0). In general, the boundary and onsite data are significantly greater than the offsite and control data. An anomaly in quarter 2/85 for the Canonsburg site was noted. Very large increases in integrated dose were observed due to construction activities in this area, during this quarter. Construction activities (of Berms B and C) uncovered large volumes of highly contaminated ( $> 100 \text{ pCi/g Ra-226}$ ) tailings materials.

Only two quarters of data are available for the Shiprock UMTRA site. Quarter 2/85 represents pre-operational environmental gamma dose measurements. Quarter 3/85 represents the first quarter during which remedial action is taking place. Not much change is noted at this time.

## CONCLUSIONS

The UMTRA Project environmental monitoring data are an excellent source of information. DOE and members of the general public interested in environmental releases as a result of remedial action, can be assured that all pathways monitored show concentrations well within the allowable concentration guide limits. Radon concentrations, air particulate concentrations, water concentrations and environmental gamma dose all fall below DOE guidelines.

## References

DOE 85a United States Department of Energy, Aug. 1985, "UMTRA Project Environmental, Health and Safety Plan" UMTRA-DOE/AL-150224.0006.

DOE 83b United States Department of Energy, Oct. 1983, "Remedial Action Plan for Stabilization of the Inactive UMTRA Site at Canonsburg, PA" UMTRA-DOE/AL-140.

DOE 85c United States Department of Energy, June 1985, "Remedial Action Plan for Stabilization of the Inactive UMTRA Site at Shiprock, NM" UMTRA-DOE/AL-050504-0039.

SK 85 D.J. Skinner, H.R. Meyer, "Demonstration of 10 CFR 20 Air Particulate Compliance Requirements on the UMTRA Project" in Environmental Radiation '85. A Topical Symposium of the HP Society at Colorado Springs, CO; Jan. 6-10, 1985.

ME 85 H.R. Meyer, D.J. Skinner, "Environmental Monitoring in the UMTRA Program" in Environmental Radiation '85, A topical symposium of the HP Society at Colorado Springs, CO; Jan 6-10, 1985.

RADIOLOGICAL CHARACTERIZATION OF THE WELDON SPRING QUARRY  
WELDON SPRING, MISSOURI

C. R. Hickey, R. D. Glenn, K. L. Christian  
Bechtel National, Inc., Oak Ridge, Tennessee 37830

ABSTRACT

Bechtel National, Inc., performed a radiological survey of the Weldon Spring Quarry (WSQ), Weldon Spring, Missouri in 1984 and 1985 as part of the U.S. Department of Energy's Surplus Facilities Management Program. The WSQ is a 9-acre site that is presently used for storage of low-level radioactively contaminated soils, rubble, and equipment. The WSQ received these thorium-, radium-, and uranium-contaminated materials including rubble, equipment, and soils from 1959 to 1968.

This survey consisted of beta-gamma and gamma measurements made at the ground surface, external gamma and radon rate measurements, and subsurface gamma loggings. In addition, surface and subsurface soils, surface water, groundwater, and sediment samples were collected and analyzed for thorium-230 and -232; radium-226, and -228; uranium-234, -235, and -238; and lead-210. Analyses of samples indicated concentrations of thorium-230, and -232; radium-226; and uranium-234, -235, and -238 in excess of the U.S. Department of Energy remedial action guidelines.

Several problems were encountered during this survey, including surveying on rough terrain, drilling into rubble, and severe weather conditions. This paper discusses the monitoring equipment and procedures used during the survey. In addition, solutions to several problems encountered will be described.

INTRODUCTION

The Weldon Spring Site is a U.S. Department of Energy (DOE) Surplus Facility located in St. Charles County, Missouri. The site comprises three separate facilities approximately 4 miles apart and their surrounding vicinity properties. The three facilities are the 52-acre raffinate pit facility and adjoining 169-acre chemical plant, and the 9-acre Weldon Spring Quarry (WSQ). The WSQ was designated for remedial action under DOE's Formerly Utilized Sites Remedial Action Program (FUSRAP), responsible for the cleanup of low-level radioactive contamination at former Manhattan Engineer District/Atomic Energy Commission (MED/AEC) sites. This paper discusses the radiological characterization of the WSQ.

The WSQ was a limestone and sand quarry that was previously part of the Department of the Army's Weldon Spring Ordnance Works; during the 1940's it was used for disposal of rubble contaminated with trinitrotoluene (TNT) (Ref. 1). In 1958, the AEC acquired title to the property and took possession of the site from the Army. The WSQ was first used for storing radioactive waste in 1959 when an estimated 200 yd<sup>3</sup> of

drummed 3.8 percent thorium residues were dumped there. In 1963 and 1964, the Mallinckrodt Chemical Works' Destrehan Street Feed Plant in St. Louis was demolished and an estimated 50,000 yd<sup>3</sup> of uranium- and radium-contaminated soils, building rubble, and equipment were deposited in the WSQ (Ref. 2).

In 1966, an estimated 600 yd<sup>3</sup> of drummed and bulk 3 percent thorium residues were deposited in the WSQ. Later that year, these residues were covered with an unspecified volume of TNT-contaminated rubble (Ref. 3).

The WSQ last received radioactive wastes in 1968 when the Army decontaminated several buildings at the Weldon Spring Chemical Plant. This decontamination effort produced an estimated 6,000 yd<sup>3</sup> of uranium- and thorium-contaminated building rubble and unrecoverable processing equipment that were disposed of in the WSQ (Ref. 3 and 4).

Bechtel National, Inc. (BNI), the Project Management Contractor for FUSRAP, performed an engineering evaluation to assess the alternatives for handling wastes at the Weldon Spring Site. The results of this assessment and of other documentation required by the National Environmental Policy Act will provide the basis for a DOE decision regarding final disposition of the site.

BNI and its radiological support contractor, Eberline Analytical Corporation (EAC), conducted a radiological survey of the WSQ from October to December 1984 and in May 1985 to provide detailed information necessary to supplement the engineering evaluation. The major objectives of the survey were to determine the areal boundaries and depths of contamination. This paper describes the equipment and procedures used in performing the survey and the results of the survey.

## FIELD MEASUREMENTS

A primary measurement grid of mutually perpendicular lines spaced 50 ft apart was established over the site by a civil survey. The field survey crew established a smaller grid spacing of 12.5 ft to facilitate the contamination boundaries. All measurements and results presented in this paper are gross readings; background has not been subtracted.

## METHODS OF MEASUREMENT

Measurements of near-surface gamma radiation levels across the WSQ were made 12 in. above the ground at 12.5 ft intervals within the grid system, using a 2-in. x 2-in. sodium-iodide (NaI) detector. This detector (EIC model SPA-3) was mounted in a probe assembly surrounded with a conical lead shield to reduce the gamma intensity through the sides, thus producing a downward directional response. This detector was coupled to an EIC PRS-1 ratemeter/scaler. Near-surface gamma radiation measurements were also measured at three background locations located at distances ranging from 1 mile to 7 miles from the WSQ.

Gamma radiation measurements on the bottom of the Quarry pond were made at 20-ft intervals within the grid system using a NaI scintillation detector (SPA-3) inside a waterproof, nonshielding container. This detector was supported by a chain with its links marked to indicate the depth of the detector below the water surface.

The gamma exposure rate was measured with a pressurized ionization chamber (PIC) at 11 locations across the site and at the three background locations. The PIC registers gamma radiation in roentgens. At 58 locations, including the 11 PIC measurement locations and the three background locations, the gamma radiation level 3 ft from the ground surface was also measured using an unshielded EIC model SPA-3 detector mounted on a tripod. A conversion factor for SPA-3 measurements was established by correlating these two measurements at the 11 locations. SPA-3 readings were then used to estimate gamma exposure rates.

Boreholes of 6-in diameter were drilled to the bedrock interface (or to refusal). Each hole was temporarily sleeved with a closed-end, 4-in.-diameter PVC pipe to allow gamma logging. A 2-in. x 2-in. NaI(Tl) gamma-scintillation detector (SPA-3 NaI crystal in a modified probe used specifically for borehole logging) was lowered into the pipe to obtain a profile of the depth of contamination. Timed gamma measurements were typically made at 1-ft vertical intervals; measures were made at shorter intervals near the boundaries of contaminated areas. When the auger was obstructed, an attempt was made to drill through the obstruction to characterize soil below it. Where bedrock was reached, the PVC pipe was removed after the borehole was logged and a 2-ft core of bedrock was taken. The corehole was also gamma logged. To determine a correlation between the radioactivity measured in cpm and the radionuclide concentrations measured in pCi/g, gamma measurements were compared with the results from laboratory analysis of subsurface soil samples obtained from the borings.

Five existing monitoring wells located in the contaminated area of the WSQ floor were gamma logged to the top of the water table (9 to 30 ft from ground surface with an average elevation of 462 ft m.s.l.). The water table was typically above the elevation of the rubble/soil interface.

Surface radon emanation rates (radon flux) were measured at locations selected on the basis of near-surface gamma measurements in order to give an adequate representation of the average radon flux of the quarry. Additional radon flux measurements were made in radioactively contaminated areas to define maximum radon emanation rates. Emanation rates were also measured at the three background locations by placing charcoal canisters in direct contact with the ground surface and exposing them for 2 days before analyzing them by gamma spectrometry (Ref. 5).

#### SAMPLE COLLECTION, ANALYSIS, AND RESULTS

Systematic surface (0 to 6 in. deep) soil samples were collected from 50-ft grid intersections. Biased samples were collected from selected areas to more accurately determine the areal extent of contamination. Each sample was placed in a 0.5-liter plastic container, capped, and labeled. The samples were analyzed for radium-226, uranium-238, and thorium-232 using gamma spectrometry techniques. This analysis consisted of a 10-minute count using an intrinsic germanium detector housed in a lead counting cave lined with cadmium and copper. Selected samples were shipped to the EAC laboratory in Albuquerque, New Mexico for thorium-230 analysis using basic radiochemical techniques.

Buried rubble and other obstructions rendered subsurface soil sampling extremely difficult. Shelby tube sampling was attempted, but an insufficient amount of soil was recovered. Split-spoon samplers were used instead, although even their use was hampered in many places. Attempts were made to core through obstructions to obtain complete soil profiles,

but these were not successful in all cases. Gamma logs of the western portion of the WSQ, an undisturbed area, indicated no contamination; thus subsurface samples were not collected in this area. In addition to gamma spectrometry analysis for radium-226, uranium-238, and thorium-232, several subsurface soil samples were radiochemically analyzed for uranium-234, -235, -238; radium-226 and -228; thorium-230 and 232; and lead-210.

The primary contaminants detected were separated uranium and daughter products from the uranium-238 chain, with contaminants from the thorium-232 chain also present. The maximum uranium-238 concentration observed in the soil samples was 1,600 pCi/g, compared with the DOE guideline of 150 pCi/g (Ref. 6). The maximum thorium-232 concentration observed in the soil was 806 pCi/g, compared with the subsurface limit of 15 pCi/g (Ref. 6).

To test for the presence of materials defined as hazardous by the Resource Conservation and Recovery Act (RCRA) and for others that might necessitate special industrial safety precautions during remedial action, several soil samples were collected, packaged, preserved, and analyzed for nonradiological parameters in accordance with Environmental Protection Agency (EPA) procedures (Ref. 7). One surface and six subsurface soil samples were collected and tested. Analyses of these samples showed that RCRA hazardous substances were not present.

Sediment samples collected from the bottom of the Quarry pond were analyzed by basic radiochemical techniques for uranium-234, -235, and -238; radium-226; thorium-230 and -232; and actinium-228. Although guidelines for radionuclide concentrations in sediment have not been established, the guidelines for concentrations in soil may be used for comparative purposes. Uranium-234, -235, and -238 concentrations ranged from 765 to 1,180 pCi/g, 31 to 227 pCi/g, and 735 to 1,170 pCi/g, respectively. Thorium-230 and -232 concentrations ranged from 220 to 405 pCi/g and 0.2 to 3.9 pCi/g, respectively. Actinium-228 concentrations varied from 2.3 to 2.7 pCi/g, and radium concentrations varied from 3 to 11 pCi/g.

Analyses of groundwater samples collected from boreholes showed uranium-234 concentrations ranging from 124 to 4,000 pCi/l, uranium-235 concentrations from 5 to 230 pCi/l, and uranium-238 concentrations from 130 to 4,000 pCi/l. The DOE guideline for total uranium in water for release to uncontrolled areas is 600 pCi/l (Ref. 8). Concentrations for radium-226 and thorium-232 were below their respective guidelines of 30 pCi/l and 2,000 pCi/l (Ref. 8).

Composite water samples from two locations in the Quarry pond were analyzed for the presence of priority pollutants, pesticides and polychlorinated biphenyls (PCBs), and asbestos. Analyses were conducted at the EAL laboratory in Richmond, California. Organic priority pollutants, pesticides, and PCBs were not detected in these composites. Chromium and mercury were detected, but did not exceed the limits of 0.05 mg/l and 0.0002 mg/l established in 40 CFR 141 (Ref. 9). Zinc was detected at 0.01 and 0.02 mg/l.

Asbestos was detected at a concentrations of  $1.9 \times 10^6$  fibers per liter. This concentration of fibers in water is consistent with levels noted in beverages and drinking water (Ref. 10). Concentration limits have not been established by DOE or EPA for either zinc or asbestos.

Exposure rates taken on-site ranged from 8 to 286 uR/h. The overall area background rate measured by BNI was 8 uR/h. The DOE guideline for the level of gamma radiation at any location on a site to be released for unrestricted use is 20 uR/h above background (Ref. 6).

Twenty-nine radon emanation measurements were taken at the WSQ. Results ranged from 0.06 to 42.9 pCi/m<sup>2</sup>/s, compared with the criterion of 20 pCi/m<sup>2</sup>/s (Ref. 8).

Beta-gamma dose measurements ranged from 0.02 to 38.5 mrad/h. The DOE guideline is 0.2 mrad/h averaged over one square meter (Ref. 8).

## CONCLUSION

The radiological survey conducted in 1984 and 1985 by BNI defined the boundaries and extent of contamination at the WSQ.

In many cases, it was difficult to ascertain the exact depth of contamination because boring was hampered by buried rubble, drums, and other obstacles. For this reason, the assumption was made that wherever subsurface contamination was identified and bedrock could not be reached, the contamination extended to bedrock. The depth of contaminated sediment in the Quarry pond was not determined during this survey; however, it was estimated that the average depth was 1 ft. Estimates based on the assumed depths indicate that approximately 95,000 yd<sup>3</sup> of radium- and thorium-contaminated soil, sediment, and rubble are present in the WSQ.

These findings are in general agreement with the Lawrence Berkeley survey (Ref. 1). Both surveys found the bulk of the radioactive materials to be located on the main quarry floor, covering a surface area of approximately 30,000 ft<sup>2</sup> and extending to depths of up to 40 ft.

## REFERENCES

1. Berkeley Geosciences Associates. Characterization and Assessment for the Weldon Spring Quarry Low-Level Radioactive Waste Storage Site, DOE/OR-853, Berkeley, CA, September 1984.
2. National Lead Company of Ohio. Environmental Monitoring Program for DOE Weldon Spring, Missouri Site, NLCO-009EV, August 1981.
3. National Lead Company of Ohio. Weldon Spring Storage Site Environment Monitoring Report for 1979 and 1980, NLCO-1176, April 1982.
4. National Lead Company of Ohio. Weldon Spring Decommissioning Study Quarry Supplement, NLCO-1121 Supplement 1, Cincinnati, OH, September 1975.
5. George, Andreas. "Passive Integrated Measurements of Indoor Radon Using Activated Carbon." Health Physics, Vol. 46, No. 4, April 1984, 867-872.
6. Department of Energy. U.S. Department of Energy Guidelines for Residual Radioactivity at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites, July 1985.
7. Code of Federal Regulations. 40 CFR 261, "Identification and Listing of Hazardous Waste," Washington, DC, July 1984.

8. U.S. Department of Energy. Order 5480.1A, "Environmental Protection, Safety, and Health Protection Program for DOE Operations," Washington, DC, August 1981.
9. Code of Federal Regulations, 49 CFR 141, "National Interim Primary Drinking Water Regulations," Washington, D.C., July 1984.
10. Cunningham, H. M. and R. Ponterfract. "Asbestos Fibers in Beverages and Drinking Water." Nature, Vol. 232, July 30, 1971, 332-333.

ENVIRONMENTAL IMPACTS OF THE RELEASE OF A TRANSURANIC ACTINIDE,  
AMERICIUM-241, FROM A CONTAMINATED FACILITY

J. Wang and P. Merry-Libby  
Environmental Research Division  
Argonne National Laboratory  
9700 S. Cass Avenue  
Argonne, Illinois 60439

ABSTRACT

Americium-241 is widely used as a radiation source, but it also has some potential risk if taken into the body because of its high dose conversion factor. Although the radiotoxicity of americium-241 is small compared to other transuranic actinides, its effects on the reproductive system and on development of the placenta are more damaging than the effects of plutonium-239. In Ohio, a gemologist's laboratory was contaminated with americium-241. Prior to decontamination of the laboratory, potential radiological impacts to the surrounding environment were assessed. A hypothetical fire accident resulting in a unit release (1 curie) was assumed. Potential radiological impacts were simulated using an atmospheric dispersion and dosimetry model with local meteorological data, population census data, and detailed information regarding the neighborhood. The results indicate that there could have been a significant impact on nearby residents from americium-241 via atmospheric dispersion if a major catastrophic release had occurred prior to decontamination and decommissioning of the laboratory.

INTRODUCTION

During the 1970s, a laboratory facility in a house near Newark, Ohio (Figure 1), was used for irradiation of diamonds and other gemstones to induce color changes for better market value. Americium-241 (powdered oxide form) was the principal source of radiation. Operations authorized under a license of the U.S. Nuclear Regulatory Commission (NRC) were discontinued in the early 1980s. A radiological survey of the facility was conducted in 1983 (1). Major americium contamination (fixed and removable) was located in the hood of the old glove box in the restricted laboratory area. Some minor contamination was found on the floor, on equipment surfaces of the restricted area, in the sink drain system, in the holding tank, and in the ventilation system filter (1). A previous estimate indicated that the amount of americium contamination in the laboratory was approximately 150 mCi (2). Later, after decontamination operations began, 22 to 24 curies of americium-241 were seized from the gemologist's facility, and the confiscated radioactive material was sent to a nearby U.S. Department of Energy (DOE) facility (3).

Prior to decontamination, the gemologist's house was occupied by tenants. The laboratory area where the americium was used was locked and designated as a restricted area. However, a fire or other accident might have caused a release of americium to the environment.

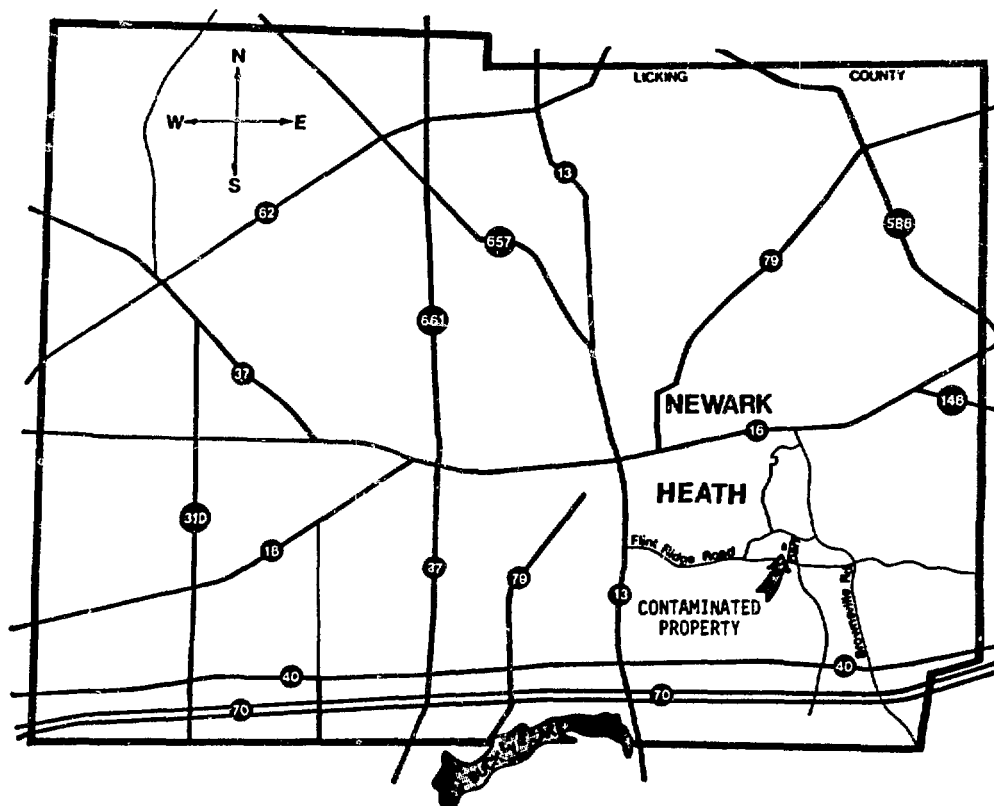


Figure 1. Map of Licking County, Ohio, Indicating the Location of the Americium-Contaminated Property.  
Source: Modified from Reference 1.

#### CHARACTERIZATION OF AMERICIUM-241

Americium exists in all oxidation states from II to VII, but the trivalent (III) state is most common. Most trivalent compounds can be converted to highly oxidative tetravalent americium compounds by ignition. The density of americium (III) oxide (hexagonal) is  $11.75 \text{ g/cm}^3$  and that of americium (IV) dioxide (cubic) is  $11.68 \text{ g/cm}^3$ . The value of  $11.7 \text{ g/cm}^3$  is used for the dose commitment evaluation in this study.

Americium-241 has a physical half-life of 458 years and an effective half-life in bone of approximately 140 years. The assumed effective half-lives in the whole body and liver are 100 and 40 years, respectively (4). Americium-241 transforms to neptunium-237 by emitting alpha particles of two distinct energies: 5.49 MeV (85%) and 5.44 MeV (13%). The major photons emitted by americium-241 are gamma rays of 60 keV (36%) and 26 keV (100%) and conversion L X-rays of neptunium-237<sub>m</sub> with energies centered at approximately 18 keV.

The major pathways of concern for exposure of humans to americium are inhalation, ingestion, and skin wound. The critical organs are liver, gonads,

red bone marrow, and bone surface (5). Americium-241 differs somewhat from other transuranium actinides in its metabolism and radiotoxicity; it has a smaller radiotoxicity and a shorter biological turnover time (faster biological clearance). However, the effects of americium-241 are more damaging than plutonium-239 with regard to the reproductive system and development of the placenta, especially in causing intra-uterine death of embryos and post-natal detrimental effects (6).

Americium oxide powder can be readily dispersed in the atmosphere because of its fine particle size (7-9). If no significant particle agglomeration occurs, the particle size can be approximated by a log-normal distribution with limited variance. In this analysis, a representative activity median aerodynamic diameter (AMAD) of 2.2 microns, with a geometric standard deviation of 2, is assumed (7,8).

#### POTENTIAL DOSES TO THE GENERAL POPULATION AND INDIVIDUALS AT NEARBY LOCATIONS IN THE EVENT OF A FIRE

##### Assumptions and Methodology

A severe fire in the laboratory area could cause the release of americium-241 particles to the atmosphere. The dispersion of americium would depend on the severity of the fire, the particle sizes, and the meteorological conditions at the time of the fire. Doses to the general public and to individuals at nearby locations would depend on the amount and pattern of the dispersed americium, the population distribution, and the locations and activities of the nearby individuals. The following assumptions are made to estimate the potential risk to humans in the event of a fire:

1. Release of americium-241 to the atmosphere during the fire is normalized to be 1 curie. The average particle density is assumed to be  $11.7 \text{ g/cm}^3$  (see previous discussion).
2. Dispersion in the atmosphere is estimated based on the yearly average meteorological conditions at Columbus, Ohio, which is located about 50 km from the gemologist's laboratory.
3. The population distribution of 1.6 million people living within 80 km of the contaminated property is based on 1980 census data.
4. Some nearby individuals for whom radiological doses are calculated were identified by NRC based on onsite observations and a local tax map (Figure 2).

A computer program developed at Argonne National Laboratory [a modified version of the UDAD computer code (10)] is used to estimate dispersion of the americium and potential radiation doses to the general public and to individuals at nearby locations. Four potential pathways of americium-241 dispersion are evaluated: (1) inhalation of americium particles, (2) direct radiation from immersion in a cloud of americium particles, (3) direct radiation from americium deposited on the ground, and (4) ingestion of americium via the food chain. Doses to the general public are reported as the 100-year dose commitment, and doses to individuals are reported as the 50-year dose commitment. For simplicity, the term "dose" is used as an interchangeable term for "dose commitment" in this analysis.

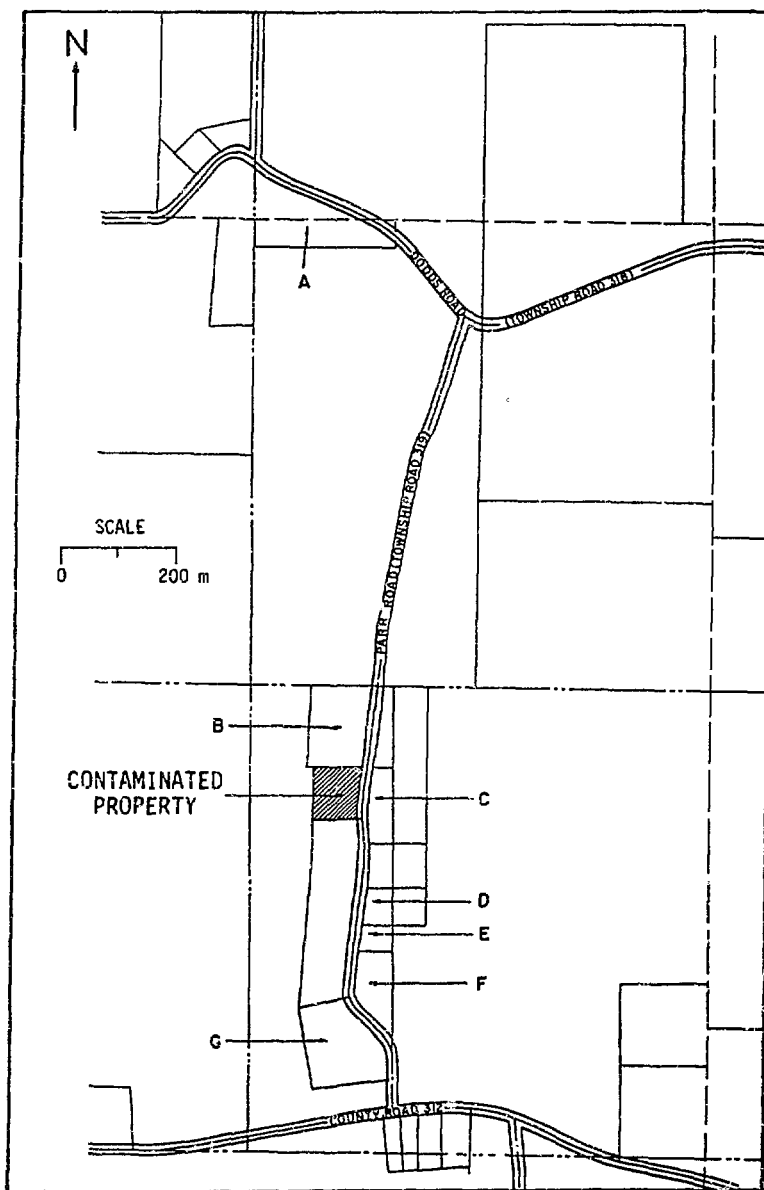


Figure 2. Location of Some Nearby Residents to the Contaminated Property for Whom Radiological Doses Are Calculated. Source: Tax Map, Franklin Township (Twp. 1; Rg. 11), June 1, 1970 (Revised November 4, 1983), and Onsite Observations, December 7, 1984.

## Estimated Doses to the General Public

The inhalation and ingestion pathways would account for more than 99% of the potential doses to the general public (population doses) (Table 1). The highest doses would occur in the skeleton (bone surface), red marrow, and liver. The effective population dose would be  $1.1 \times 10^4$  person-rem (110 person-Sv) per curie release of americium-241, which is approximately equal to 7% of the natural background radiation dose that the same 1.6 million people would receive in one year. However, distribution of the population dose would not be uniform throughout the area analyzed (within 80 km of the source). Most of the population dose would be concentrated near the source. For example, 50% of the whole-body population dose via the inhalation pathway is predicted to occur within 0.5 km of the gemologist's property.

The projected americium-241 concentrations in air (Figure 3) show a pattern of decreasing concentration with distance from the source. Due to the compartmental approach of the model, the spatial distributions of all other pathways are similar to the inhalation pathway (see Figure 3) but differ by a concentration factor. Also, the decreases are fairly uniform in all directions. This can be expected because the annual frequency of wind occurrence in all 16 directions is very uniform, with an average of 6.25% and a standard deviation of 2.43%. For brevity, the air concentrations are presented for only four directions (N, E, S, and W) in Figure 3. On the log-log plotting, all lines have approximately the same slope of -1.9, fitting closely to an inverse square law. This means that doubling the distance would decrease the americium-241 concentration in air by about 73%.

## Estimated Doses to Individuals at Nearby Selected Locations

Potential doses per 1 curie release of americium-241 at seven selected nearby locations are given in Table 2; for comparison, the regulatory limits (11) are also given. The doses to the skeleton (bone surface) and red bone marrow are the highest among the selected organs. At all seven locations, individuals would receive a red marrow dose exceeding the NRC regulatory limit of 500 mrem/yr (5 mSv/yr). Except for location A, farthest from the source (about 1 km), all others would have skeletal doses exceeding the regulatory limit. The highest bone doses (34.6 rem [346 mSv] to the red marrow and 33.0 rem [330 mSv] to the skeleton) would be received at location C, just across from the road from the source. At most selected locations, the doses to other organs would also exceed the regulatory limits if there were 1 curie of americium released during the fire.

## Discussion

The analysis presented herein is constrained by the assumption that there is 1 curie of americium-241 released from the contaminated laboratory during a fire. In fact, this source term is a product of two independent factors, the amount of americium-241 present at the laboratory and the fraction of that americium-241 released into the air during the fire. As discussed previously, the amount of americium-241 could be more than 20 curies. However, any variation in the two factors--amount of americium-241 present and fraction released during a fire--could directly affect the amount of pollutant available for atmospheric transport and would modify the assessment of radiological impacts.

The estimated doses are actually estimated "risks" of doses based on the release and dispersion of 1 curie of americium-241, assuming the "probability"

Table 1. Estimated 100-Year Dose Commitments to the General Public<sup>#a</sup>

Organ	Dose Commitment (organ-rem/curie)				
	Inhalation	Ingestion	External		Total
			Ground	Cloud	
Whole body	$1.01 \times 10^3$	$3.63 \times 10^3$	$4.61 \times 10^0$	$9.07 \times 10^{-6}$	$4.63 \times 10^3$
Liver	$5.14 \times 10^3$	$1.93 \times 10^4$	$2.64 \times 10^0$	$5.21 \times 10^{-6}$	$2.44 \times 10^4$
Gonads	$2.65 \times 10^2$	$9.60 \times 10^2$	$4.05 \times 10^0$	$7.93 \times 10^{-6}$	$1.23 \times 10^3$
Lung	$1.90 \times 10^2$	$1.47 \times 10^3$	$3.25 \times 10^0$	$6.35 \times 10^{-6}$	$1.91 \times 10^3$
Red marrow	$2.37 \times 10^4$	$7.27 \times 10^3$	$8.00 \times 10^0$	$1.56 \times 10^{-5}$	$3.10 \times 10^4$
Skeleton	$2.49 \times 10^4$	$9.07 \times 10^4$	$8.60 \times 10^0$	$1.71 \times 10^{-5}$	$1.15 \times 10^5$

<sup>#a</sup> Dose commitments to the population within 80 km of the contaminated property assuming a release of 1 curie (see text).

Table 2. Dose Commitments to Individuals at Nearby Locations<sup>#a</sup>

Organ	Regulatory Limit <sup>#b</sup> (mrem/yr)	Dose Commitment (mrem/Ci) <sup>#c</sup>		
		A	B	C
Whole body	$5 \times 10^2$	$3.55 \times 10^2$	$1.09 \times 10^4$	$1.4 \times 10^4$
Liver	$5 \times 10^2$	$1.82 \times 10^3$	$5.59 \times 10^4$	$7.13 \times 10^4$
Gonads	$5 \times 10^3$	$9.33 \times 10^1$	$2.87 \times 10^3$	$3.67 \times 10^3$
Lung	$5 \times 10^2$	$5.31 \times 10^2$	$2.06 \times 10^4$	$2.64 \times 10^4$
Red marrow	$5 \times 10^2$	$6.80 \times 10^3$	$2.58 \times 10^5$	$3.30 \times 10^5$
Skeleton	$1.5 \times 10^3$	$8.80 \times 10^3$	$2.70 \times 10^5$	$3.46 \times 10^5$
-----				
Organ	Dose Commitment (mrem/Ci)			
	D	E	F	G
Whole body	$3.71 \times 10^3$	$2.89 \times 10^3$	$1.85 \times 10^3$	$1.27 \times 10^3$
Liver	$1.89 \times 10^4$	$1.47 \times 10^4$	$9.80 \times 10^3$	$6.53 \times 10^3$
Gonads	$9.73 \times 10^2$	$7.60 \times 10^2$	$4.87 \times 10^2$	$3.36 \times 10^2$
Lung	$7.00 \times 10^3$	$5.45 \times 10^3$	$3.49 \times 10^3$	$2.41 \times 10^3$
Red marrow	$8.73 \times 10^4$	$6.80 \times 10^4$	$4.36 \times 10^4$	$3.01 \times 10^4$
Skeleton	$9.20 \times 10^4$	$7.13 \times 10^4$	$4.59 \times 10^4$	$3.17 \times 10^4$

<sup>#a</sup> See Figure 2.

<sup>#b</sup> Based on Reference 11 (10 CFR 20) and Reference 12 (Table 4.15).

<sup>#c</sup> Reported as 50-year dose commitments per curie release of americium-241.

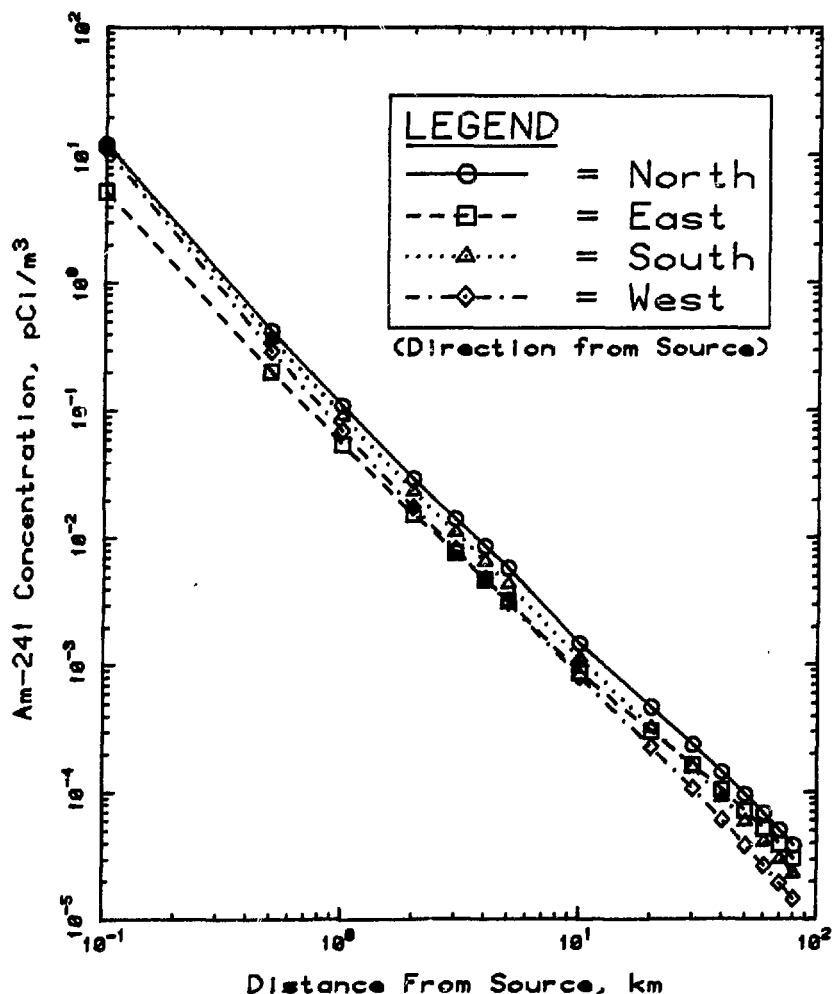


Figure 3. Americium-241 Concentration in Air from Release of One Curie.

of various weather conditions at Columbus, Ohio. Doses--especially individual doses--could be more or less, depending on weather conditions at the time of the fire, and actual weather conditions at the site may differ from those at Columbus. For example, there are a small hill and a small ditch behind the contaminated laboratory (on the west side) that may cause significant micro-meteorological variation. This local effect may alter the potential doses, especially for nearby individuals. However, site-specific meteorological data would be needed to make a more accurate dose assessment.

Firefighters or other persons who might be closer to the fire could possibly receive even higher doses than the nearby individuals. This would depend upon how much time such a person spent near the fire, whether or not the person was downwind of the fire, and whether or not the person used a protective breathing apparatus.

During a fire, americium (III) oxide would be oxidized to americium (IV) dioxide. Tetravalent americium is easily hydrolyzed to form complex ions (13) and experimental work has shown that americium dioxide is somewhat soluble (7). Airborne americium can be washed out substantially by rain. If it is assumed that (a) it rains during or immediately after the hypothetical fire, (b) most of the released americium is confined within 1 km of the contaminated property and is dissolved in rain water, and (c) there is approximately a 1.8-mm (0.07-in.) rainfall, the rainwater would have an average concentration that is at the maximum permissible concentration of 4 pCi/mL. [Note: A 1.8-mm rainfall is approximately 2% of the monthly average rainfall of 78 mm at Columbus, Ohio (14).]

Because the computer model used in this analysis is basically an air pollution model incorporating radiological metabolic models, the water pathways (groundwater and surface water) are not analyzed in this study. If the water pathway were significant, the ingestion dose would be higher. Further study is required to determine the significance of the water pathway.

#### ACKNOWLEDGMENT

Work supported by the U.S. Department of Energy under Contract W-31-109-Eng-38.

#### REFERENCES

1. M.W. Stafford, Radiological Survey of the J.C. Haynes Property, Newark, Ohio, SMPB-2, Prepared for U.S. Nuclear Regulatory Commission, Region III Office, by Radiological Site Assessment Program, Oak Ridge Associated Universities, Oak Ridge, TN, Final Report, May 1984.
2. J. Berger, Personal communication (Oak Ridge Associated Universities) to P. Merry-Libby (Argonne National Laboratory), October 15, 1984.
3. A.R. Benedetto, "The 'Don't Thumb Your Nose at the NRC' Department," Health Phys. Soc. Newsl. XIII(6), p. 14, 1985.
4. International Commission on Radiological Protection, The Metabolism of Compounds of Plutonium and Other Actinides, ICRP Publ. No. 19, Pergamon Press, Elmsford, NY, 1972.
5. International Commission on Radiological Protection, Limits for Intakes of Radionuclides by Workers, A Report of Committee 2 of the International Commission on Radiological Protection, Adopted by the Commission in July 1978, ICRP Publ. No. 30, Part 3, Pergamon Press, Elmsford, NY, 1978.
6. E.P. Ovcharenko, "An experimental evaluation of the effects of trans-uranic elements on reproductive ability," Health Phys. 22, p. 641, 1972.
7. B.A. Muggenburg and J.A. Mewhinney, "Removal of inhaled  $^{241}\text{Am}$  oxide particles of various sizes from beagle dogs using lung lavage and chelation treatment," Health Phys. 41, pp. 123-133, 1981.

8. J.A. Stanley, A.F. Edison, and J.A. Mewhinney, "Distribution, retention and dosimetry of plutonium and americium in the rat, dog and monkey after inhalation of an industrial-mixed uranium and plutonium oxide aerosol," Health Phys. 43(4), pp. 521-530, 1982.
9. Park, L. "Cleanup of home to cost EPA \$150,000," Columbus Citizen-Journal, April 17, 1985.
10. Momeni, M.J., Y. Yuan, and A.J. Zielen, The Uranium Dispersion and Dosimetry (UDAD) Code, NUREG/CR-0553; ANL/EIS-16, Prepared for U.S. Nuclear Regulatory Commission, Washington, DC, by Argonne National Laboratory, Argonne, IL, 1979.
11. U.S. Nuclear Regulatory Commission, Title 10, Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation".
12. B. Shleien and M.S. Terpilak (comp.), The Health Physics and Radiological Health Handbook, Nucleon Lectern Associates, Inc., Olney, MD, 1984.
13. Kirk-Othmer, Encyclopedia of Chemical Technology, 3rd ed., John Wiley & Sons, New York, Vol. 1, pp. 456-480, 1978.
14. Gale Research Company, Climates of the States, National Oceanic and Atmospheric Administration Narrative Summaries, Tables, and Maps for Each State with Overview of State Climatologist Programs, 2nd ed., Vol. 2, 1982.

A REVIEW OF CURRENT RESEARCH INTO SOIL DECONTAMINATION AND VOLUME  
REDUCTION TECHNIQUES FOR PURPOSES OF TRANSPORTATION AND DISPOSAL

Fred F. Haywood  
Eberline Analytical Corporation, 800 Oak Ridge Turnpike,  
Oak Ridge, TN 37830

ABSTRACT

In recent years, a number of soil decontamination studies have been carried out by investigators in the United States and Canada. These investigations have included the segregation of potentially hazardous chemicals from uranium mill tailings, the use of various leaching agents for the selective removal of radium, thorium, and uranium from mine and mill tailings, the study of neutralization, fixation, and conditioning processes for tailings, the separation and removal of transuranium elements from contaminated soil and sediment, and finally studies of the cost effectiveness of various decontamination approaches. Remedial action programs for which soil decontamination may be an attractive alternative are those which include the cleanup of large open land areas since thousands of cubic yards of material are generally involved. The practice of decontaminating or otherwise segregating contaminated and uncontaminated soil fractions before shipment to a final disposal site can only be done if it is economical to do so. This paper considers several techniques which have been reported in unclassified literature and includes a comparison of the authors' conclusions regarding the effectiveness of their investigations. The manuscript will include a comprehensive bibliography of references to techniques which appear to be feasible alternatives in remedial action programs.

## **TMI-2 POST-ACCIDENT DOSE REDUCTION THROUGH COATING REMOVAL**

**S. Scott Hinds and E. D. Pavelek, II**  
Bechtel National, Inc.  
P. O. Box 72, Middletown, PA 17057-0072

**Kenneth R. Hoyt**  
GPU Nuclear Corporation  
P. O. Box 480, Middletown, PA 17057-0072

### **ABSTRACT**

This paper describes the approaches used to define sources and to achieve both effective decontamination and dose reduction through coating removal from surfaces in the TMI-2 auxiliary and fuel handling building and reactor building. The testing, production rates, and lessons learned during surface removal operations from 30,000 ft<sup>2</sup> of epoxy coated concrete floors are discussed. Finally, the resulting post-decontamination exposure levels and recommendations for future work are presented.

### **INTRODUCTION**

The 1979 accident at Three Mile Island Unit 2 (TMI-2) dispersed fission products throughout the plant. Radiological surveys taken in the reactor building and the auxiliary and fuel handling building (AFHB) in 1982 indicated that a substantial portion of that contamination was embedded in the nuclear-grade coatings used to paint structural surfaces. By the fall of 1982, the corridors on the 305' and 328' elevations of the AFHB had been decontaminated to a stable level that permitted personnel access in minimal or no anti-contamination clothing. However, the corridors of the 282' elevation were still contaminated to levels of 50K to 150K dpm/100 cm<sup>2</sup>, requiring radiation work permits (RWPs), anti-contamination clothing, and respirators.

Initially, methods used to attempt to clean and stabilize the 282' elevation corridors included vacuuming, mopping, and scrubbing. Surveys after these operations showed that an initial reduction of contamination levels was followed by an increase to approximately the original level, presumably from particulate deposition or leaching of embedded contaminants from coatings. Further studies proved that leaching of radionuclides from contaminated coatings was indeed the primary recontamination source.

This report describes the methods and equipment used to achieve effective decontamination and dose reduction through nuclear-grade coating removal from the AFHB and the reactor building floors.

### **DESCRIPTION OF WORK**

A literature/vendor search was conducted to determine candidate coating removal techniques; the following were considered:

1. Chemicals - dissolve coatings
2. Surfacers - concrete finishing tools
3. Scarifiers - use blades to strip surfaces

4. Scabblers - shatter concrete surfaces via concussion
5. Grinders - typical industrial power grinders
6. Abrasive pads/mechanical scrubbers - scrubbing via manual labor.

These methods were tested in clean areas to determine the waste form generated, production rate, power requirements, and effectiveness for total coating removal. This testing established that scarifiers and scabblers were the most satisfactory candidates and would undergo further testing on contaminated coatings.

Before testing in contaminated areas, an evaluation was performed to determine the most dose effective work sequence. This evaluation showed that to maintain personnel exposures as low as reasonably achievable (ALARA), the best work sequence would be: 1) removal of coatings in large open areas; 2) trimming to within 1 in. of the walls; and 3) trimming to the walls.

#### SCARIFIER TEST RESULTS AND CONCLUSIONS - AFHB

The floor area selected for the scarifier test was adjacent to an open hatch on the 305' elevation of the AFHB. The hatch had been used for removing contaminated equipment from the 282' elevation, resulting in some contamination on the floor around it. Maximum radiation readings in this area were found to be approximately 10 mR/h gamma and 500 mrad/h beta.

The test required two workers to operate the scarifier for approximately 30 minutes to remove a 25- to 30-mil epoxy coating and the concrete substrate to a maximum depth of 1/16 in. Because the original hardened steel scarifier blades wore out before the test area was covered, tungsten carbide-tipped blades were substituted for future tests. Also, airborne activity increased temporarily during the scarification process.

The major conclusions reached from these tests were that surface removal to a depth of 1/16 in. could be achieved at an average rate of 96 ft<sup>2</sup>/h and that decontamination factors of approximately 20 for maximum gamma source and 1000 for maximum beta source were possible.

After the test, the treated area was scrubbed and vacuumed to remove remaining debris and dust. A radiological survey, conducted after the test area dried, indicated that the area was below the radiological release limit of 1000 dpm/cm<sup>2</sup>. The test area was covered with a protective plastic sheet for several days and another radiological survey was then performed. No contamination was found, thereby confirming that removal of the coating eliminated that source of recontamination. The treated floor was subsequently recoated.

A contaminated cubicle on elevation 282' of the AFHB was used for the balance of the coating removal tests because of the radiological safety and convenience inherent in an enclosed cubicle. The equipment for trim removal could also be tested in this enclosed area without the risks of recontamination encountered in an open clean area.

Scarification in the cubicle reduced the general area gamma radiation levels from approximately 35 mR/h to about 10 mR/h and beta radiation levels on the floor surface were reduced from a range of 18 to 40 mrad/h at contact to below the limit of detection, approximately 0.2 mrad/h. Smearable contamination levels were reduced from a range of 150K to 300K dpm/100 cm<sup>2</sup> to less than 500 dpm/100 cm<sup>2</sup>.

## SCABBLER TEST RESULTS AND CONCLUSIONS - AFHB

A modified three-piston wall scabblers was tested next. The major conclusions from a clean area test were that minimal dust and small particles escaped during the unit's operation and that circular patterns using a seven-point cutter bit worked best for removing the coating. Contaminated area scabbling tests were then conducted in the "A" reactor coolant bleed tank cubicle. These tests demonstrated that no adverse radiological conditions were encountered during or after operations.

The modified scabbling equipment removed approximately 70 percent of the surface gamma radiation and 98 percent of the surface beta radiation with the coating. Initial smearable contamination levels of 150K to 300K dpm/100 cm<sup>2</sup> were reduced to less than 500 dpm/100 cm<sup>2</sup> upon completion of scabbling and one vacuum and scrubbing operation. The waste volume generated was approximately 1 ft<sup>3</sup> per 150 ft<sup>2</sup> of surface treated.

Based on waste samples, auxiliary building coating debris radionuclide composition by activity was calculated to be:

<u>Isotope</u>	<u>% (Curies)</u>
Mn- 54	0.6
Co- 60	4.2
Cs-134	5.3
Cs-137	85.5
Sr- 90	<u>4.4</u>
	100.0

The average curie content of a 55-gallon drum of auxiliary building scabble debris was 150 mCi.

As a result of scabbling and recoating operations on the 282' elevation of the AFHB, recontamination due to leaching from coatings has been eliminated and the floor area, with a few exceptions, has remained at less than 500 dpm/100 cm<sup>2</sup>. All coated areas treated with this device were released as radiologically clean after the scabble debris was collected and scrubbing and wet vacuuming were performed. No significant airborne contamination nor spread of surface contamination was encountered when using this device. In all instances, the resulting floor condition was acceptable for repainting without any further finishing. The quality of the floor surface after scabbling justified use of a larger floor scabblers to take advantage of the increased rate of coating removal. The relative ease of operation as well as the longer life and increased coating removal rates made the three- and five-bit scabblers a clear choice over the scarifier.

The five-bit U-5 floor scabblers was tested using nine-point cutter bits in a clean area. With a two-person crew it exhibited a production rate of 100 ft<sup>2</sup>/h at available air pressures and flow volumes. The contaminated coating was totally removed and the resultant surface was acceptable for recoating. No visible escaping dust or debris was seen, and residual debris was examined and determined to pose only a minimal potential for airborne contamination. The depth of coating and surface removal was less than the maximum permitted (3/16 in.) after total coating removal.

Testing of the modified U-5 scabblers in a contaminated area was conducted in the same cubicle as the earlier tests in the AFHB. Because the scabblers has a lower and narrower profile, the scabblers could be used in areas inaccessible to the scarifier. The initial smearable contamination levels

were 150K to 300K dpm/100 cm<sup>2</sup> and the initial beta radiation readings at contact with the floor were 18 to 40 mrad/h. The gamma radiation levels were not included in the test data because of variable gamma readings resulting from close proximity to a tank containing contaminated water.

The contaminated area tests using the U-5 scabblers resulted in post-scabbling smearable contamination levels that were less than 500 dpm/100 cm<sup>2</sup> upon completion of scabbling, dry vacuuming, scrubbing, and wet vacuuming operations. Post-scabbling beta radiation readings at contact with the floor were less than 0.2 mrad/h. Some airborne activity was detected, which required that negative pressure filter respirators be worn during scabbling operations. The production rate for coating removal was 100 ft<sup>2</sup>/h for a two-person team. Waste was generated at a rate of 1 ft<sup>3</sup> per 100 ft<sup>2</sup> of treated area, and the waste activity was the same as reported for the three-bit scabblers.

#### SCABBLER TEST RESULTS AND CONCLUSIONS - REACTOR BUILDING

Having established scabbling as an effective coating removal method through auxiliary building testing, dose reduction via scabbling was evaluated in the reactor building. A 3 ft by 3 ft area was marked off for initial testing on the 347' elevation. Scabbling of this small area accomplished a 93 percent beta reduction with 1/8 in. concrete surface removal. Tests demonstrated that greater than 90 percent of the floor source was contained within this layer. The isotopic composition of the small-scale scabbling test debris was 94 percent Cs-137 and 6 percent Cs-134, and the total activity was 150 uCi/cc.

Theoretical calculations of exposure rates from coatings before and after removal indicated that a maximum average reduction of 7 to 8 mR/h was possible through large-scale scabbling. The tests exceeded this projection. Models based on this test data indicated a potential 50 to 60 percent exposure rate reduction at 3 ft above the 347' elevation floor through coating removal. The demonstrated exposure rate reduction and projected potential dose savings justified a large-scale dose reduction through coating removal program.

A large test area was selected based on the highest personnel occupancy zones, as projected by the recovery technical plans, in order to provide the maximum potential ALARA benefit for the project. The test area covered approximately 500 ft<sup>2</sup> on the 347' elevation of the reactor building. Upon completion of the coating removal with the U-5 scabblers, the general area gamma exposure rates within the test area at 3 ft above the floor, as measured by thermoluminescent dosimeters (TLDs), were reduced from 65 mR/h to an average of 36 mR/h; a reduction of 45 percent.

The waste volume generated was approximately 1 ft<sup>3</sup> for every 65 ft<sup>2</sup> of floor surface treated. The increase in waste volume per unit area over the auxiliary building was due to greater coating thickness and to the strippable coatings that had been applied over existing epoxy paint. The isotopic composition of a drum of reactor building scabbling debris was as follows:

<u>Isotope</u>	<u>Activity (Curies)</u>
Co- 60	7.59E-5
Ni- 63	6.94E-5
Sr- 90	2.68E-2
Tc- 99	3.98E-4
Cs-134	3.20E-2
Cs-137	<u>6.64E-1</u>
	7.23E-1

The total activity was 723 mCi/drum.

Upon completion of the reactor building test areas, further scabbling work was planned. Using both the three-bit and five-bit scabblers, accessible areas of the 347' and 305' elevations in the reactor building were scabbled with excellent results. Approximately 10,000 ft<sup>2</sup> of reactor building floor surfaces were treated with the scabblers and recoated, with a dose expenditure of approximately 40 person-rem. General area gamma exposure rates were reduced 15 to 70 percent, while beta radiation was not detectable in the majority of areas after scabbling. Contamination levels after scabbling, vacuuming, and scrubbing averaged less than 5000 dpm/100 cm<sup>2</sup>.

#### RECOMMENDATIONS AND CONCLUSIONS

Scabbling and scarifying were both found to be effective decontamination and dose reduction methods for contaminated coated surfaces. Scabbling was chosen as the primary coating removal method at TMI-2 because the machines were more cost effective, generated less airborne contamination, and could gain access to tighter quarters than scarifiers. The coating removal via scabbling program at TMI-2 has been very successful. Overall dose reductions of approximately 50 percent gamma and 90 percent beta have been achieved in the AFHB and reactor building. Coating removal by scabbling has proven to be a safe, efficient, and effective means to reduce exposure as well as contamination.

The original nuclear-grade coatings were effective in preventing migration of contaminants into the concrete substrate; however, the coatings themselves became highly impregnated with various contaminants. The nuclear industry should work to improve nuclear-grade coatings to provide greater resistance to absorption of soluble radioactive contaminants. Coating removal and replacement programs in nuclear facilities should also be developed and integrated into maintenance and ALARA programs where applicable.

**THE USE OF URETHANE FOAM IN THE DECONTAMINATION AND  
DECOMMISSIONING OF NUCLEAR FACILITIES**

**Rudy E. Hermetz  
Monsanto Research Corporation  
Mound\*  
Miamisburg, Ohio 45342**

**ABSTRACT**

Urethane foam is being used in decontamination and decommissioning work in radioactively contaminated areas at Monsanto Research Corporation's Mound facility. Used in a two-step method, the foam is first sprayed onto the interior surfaces of contaminated gloveboxes, fixing residual contamination beneath the urethane. The foam is then used to package and stabilize gloveboxes inside standard transuranic shipping containers. The procedure reduces health and safety risks and has proven cost effective.

**\*Mound is operated by Monsanto Research Corporation for the U. S. Department of Energy under Contract No. DE-AC04-76DP00053.**

## INTRODUCTION

The decontamination and decommissioning (D&D) of nuclear facilities described here is taking place at Monsanto Research Corporation's Mound facility, which is operated for the Department of Energy. The purpose of D&D at Mound is to safely clean and/or dispose of surplus radioactively contaminated facilities and equipment in order to reduce safety and health risks to Mound employees, the public, and the environment. Another goal is the reuse of existing contaminated facilities, if possible. The D&D is under way in four areas at Mound, and each area will be returned to a condition suitable for reallocating it to other efforts. Started in 1978, this multimillion-dollar effort is scheduled to continue through 1996.

## DISCUSSION

Some years ago, the National Aeronautics and Space Administration (NASA) was searching for a reliable source of electrical energy for its space programs. NASA was planning probes of the moon and distant planets, where the weight of solar cells would be unacceptable. Radioisotopic thermoelectric generators (RTGs) were chosen for their compactness and projected reliability. The fuel source for the RTGs was plutonium-238, which gives off about  $\frac{1}{2}$  watt of heat per gram of mass and has a half-life of 87 years.

Mound was given the mission to process plutonium-238 into a number of special fuel forms, then encapsulate them in several layers of exotic metals for containment and protection.

In 1975 regulations were changed for handling unencapsulated plutonium, and Mound was eliminated as a processor of unencapsulated plutonium. We could, however, continue to do the assembly work on materials already prepared for assembly. Currently, Mound is producing RTGs for the Solar Polar Mission -- a probe of the sun's polar regions.

Because Mound no longer processes the fuel forms, the Plutonium Processing Building and associated work areas, where work had been done, were of no use as they were. Planning for their decontamination and decommissioning was begun in 1977. The work started in 1978 in the Research Building and the Plutonium Processing Building.

Mound had developed some experience over the years with D&D operations of facilities involving various types of radioactive materials. The type of material and its level of activity dictate the means used to clean it up. For example, the isotope polonium-210 presents few problems because its half-life is short (approximately four months). Residual quantities of material decay completely within a few years. Plutonium-238, on the other hand, has a half-life of about 87 years and even very small residual quantities are considered unsafe. Some previous D&D of plutonium-238 facilities had indicated we would face some serious problems in preparing, handling, packaging, and shipping the contaminated equipment for burial.

There are many methods and techniques for decontaminating and decommissioning radioactively contaminated facilities. The method described here uses urethane foam to control contamination.

The planning and estimating for D&D were based on procedures for spraying urethane foam on the interior surfaces of gloveboxes to "fix" residual contamination to the surfaces. The procedure is outlined below.

1. Remove, prepare, and package the interior services, equipment, and any noncritical interior structural members of the gloveboxes.
2. Clean all interior surfaces within the enclosure (glovebox) of gross quantities by sweeping and scrubbing.
3. "Fix" residual quantities of contamination on interior surfaces with the application of sprayed urethane foam.
4. Prepare, remove, and package the exterior piping, valves, conduit, and support equipment.
5. Loosen gloveboxes from their positions and either size-reduce, if too long, or prepare for packaging.
6. Load gloveboxes into transuranic (TRU) type shipping containers.
7. Package and stabilize loads within shipping containers by the use of sprayed urethane foam.

Procedures 3 and 7 call for the use of sprayed urethane foam in two different applications: one on the interior surface of gloveboxes (Procedure 3) and the other for packaging in waste shipping containers (Procedure 7).

Procedure 3 uses the urethane foam on the interior surfaces of the gloveboxes to "fix" any residual quantities of contamination in place. This eliminates some cleaning operations on interior surfaces without sacrificing safety in any respect. Figure 1 shows a typical large glovebox foamed and ready for size reduction.

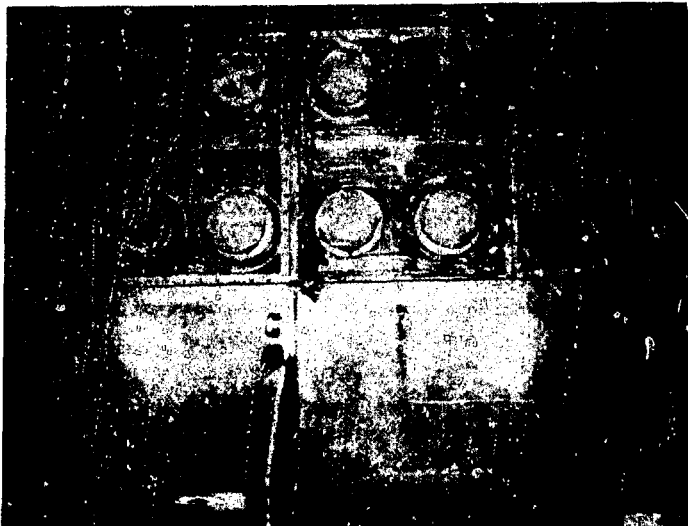


Figure 1 - Typical long stainless steel glovebox foamed and ready to be cut as indicated. Methods were developed to cut through the glass panels where necessary.

Implementing this procedure presented two problems. First, the contaminated interior environment must remain separated from the uncontaminated outside. The method worked out to solve this problem adapted a bagging technique commonly used in radioactive work. The foam gun being used was a Model D Gusmer which has an air cap that blows off foam from the gun tip. The cap was used to secure the plastic bag in place around the gun. Then, the open end of the bag was put on a gloveport and clamped on securely. The bag can be made any convenient length so that a long glovebox can be foamed from one port location. The coating operation is worked back to the entry port. A minimum of foam is sprayed onto the interior surfaces. When all surfaces are coated, the gun is withdrawn from the bag, the cap removed, the small hole taped over, and the bag secured.

Second, the Model D gun was constructed in a manner that made access through a gloveport difficult. We talked with our supplier about this, and they modified the gun by remotely locating the trigger. This reduced one dimension of the gun so that it would easily pass through the gloveports. The remote triggering also made the gun more maneuverable for our uses. The modified gun is shown in Figure 2.

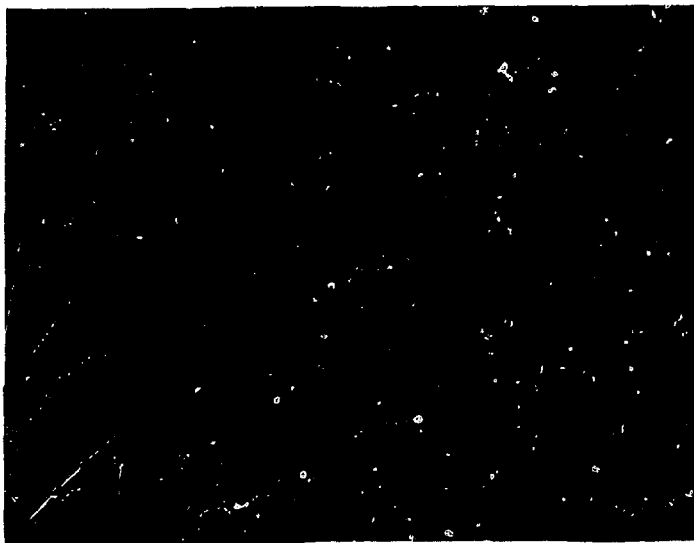


Figure 2 - Modified Gusmer Model D gun with a remotely located triggering device to allow greater mobility.

The foaming solved a related problem by allowing us to cut gloveboxes that are too long for any of our waste shipping containers. Some gloveboxes are up to 20 feet long. We can cut these contaminated boxes apart without a problem with proper preparations. These practices related to Procedure 3 have saved us many manhours and greatly minimized personnel exposures.

Procedure 7 involves the use of sprayed urethane foam in packaging. Because our range of sizes (both shipping containers and loads) is so wide, void sizes range widely too. The standard TRU shipping container is 4 feet wide by 4 feet high by 7 feet long, and the largest TYPE I (TRU) container is 6 feet wide by 9 feet high by 11½ feet long. Due to this wide range, sprayed urethane foam seemed to be the most versatile packaging system for our needs. It protects against damage during shipment by truck and/or rail. Because the gloveboxes are stripped of equipment and services, their weight-to-surface-area ratio is very low. A typical shipping container with foam in place is shown in Figure 3. The gross weight is kept low by the prudent use of the 1½-pound density foam. The four corners and a pad on top are foamed between the load and the container, except on longer loads where foam is applied at the sides. In our first attempts at packaging with foam, the exothermic heat buildup caused us some problems. These were overcome with some procedural changes and additional operator training.

Our experiences so far with sprayed urethane foams have been very favorable. To date, more than 560 linear feet of gloveboxes have been foamed and removed using this method. The savings in time and labor will help cut costs and schedule times considerably.

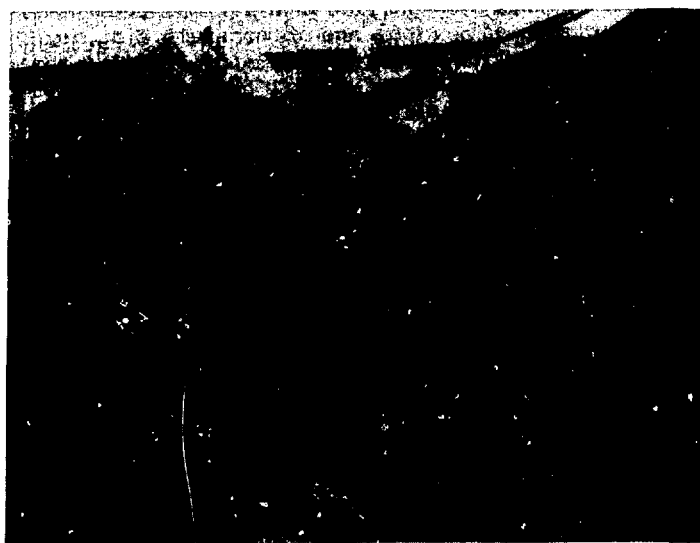


Figure 3 - Shipping container with test load foamed in place.

REMOTE METHODS  
FOR  
DECONTAMINATION AND DECOMMISSIONING OPERATIONS

J. R. DeVore  
Oak Ridge National Laboratory  
P.O. Box X, Building 3026-C  
Oak Ridge, Tennessee 37831

ABSTRACT

Three methods for the decontamination and decommissioning of nuclear facilities are described along with operational experience associated with each method. Each method described in some way reduces radiation exposure to the operating personnel involved.

Electrochemical decontamination of process tanks is described using an in-situ method. Descriptions of two processes, electropolishing and cerium redox decontamination, are listed. A method of essentially smokeless cutting of process piping using a plasma-arc cutting torch is described. In one technique, piping is cut remotely from a distance using a specially modified torch holder. In another technique, cutting is done with master-slave manipulators inside a hot cell. Finally, a method for remote cutting and scarification of contaminated concrete is described. This system, which utilizes high-pressure water jets, is coupled to a cutting head or rotating scarification head. The system is suited for cutting contaminated concrete for removal or removing a thin layer in a controlled manner for decontamination.

This work is sponsored by the U.S. Department of Energy as part of the Surplus Facilities Management Program under Contract No. DE-AC05-84OR21400.

## INTRODUCTION

The decommissioning of nuclear facilities at times requires the decontamination of and equipment removal from areas which can have high levels of radiation or contamination. This paper presents several methods which have been used at the Oak Ridge National Laboratory (ORNL) for decontamination for the purpose of reducing radiation levels in equipment thus reducing personnel exposure during the decommissioning of two hot cell facilities. This work is sponsored by the U.S. Department of Energy as part of the Surplus Facilities Management Program under Contract No. DE-AC05-84OR21400.

## ELECTROCHEMICAL DECONTAMINATION

At ORNL, two methods of electrochemical decontamination have been employed in the decommissioning of the Fission Product Development Laboratory (FPDL). The FPDL was a full-scale processing facility for separating megacurie quantities of strontium-90, cesium-137, and cerium-144 for a variety of source applications and operated at full capacity from 1958 to 1975. Part of the facility has been designated as surplus, and that part is being decontaminated on a "for re-use" basis. Both electrochemical decontamination methods used have been "in-situ" methods, that is, the objects were left in place while being decontaminated rather than removing them.

Electropolishing is a method which is well documented as a decontamination process. (1) It is a process in which the object to be decontaminated is made to be the anode in an electrolytic cell. The passage of an electric current results in the dissolution of a thin layer of surface material, and radioactive contamination which is adhered to or otherwise entrapped in surface imperfections is removed and released into the electrolyte. At the FPDL, electropolishing has been used to decontaminate two 127-gallon tanks located in a stainless-steel-lined cell. In the original fission product purification process, these tanks had been used in the crystallization of cesium-137 oxalate. The tanks were highly contaminated to radiation levels on the order of hundreds of R per hour. When the FPDL was shut down in 1975, the tanks were decontaminated using the more conventional means of flushing with acidic and basic solutions. Later, high-pressure spraying and more chemical decontamination had been used to reduce the tank levels to 8 to 10 R/h at contact with isolated locations reading up to 100 R/h.

The electropolishing equipment consisted of a 3200-amp dc power supply coupled with a stainless steel electrode, as shown in Fig. 1. Since all FPDL process cells have a stainless steel liner to which the tanks and piping are attached, all tanks and piping are grounded. Thus, only one set of leads was required. The first tank was filled with 80% phosphoric acid which was heated to 85°C. The electrode was inserted and the electric current was applied for approximately 15 minutes. The resulting data are presented in Table 1.

The current density achieved was below that required for true electropolishing but resulted in a net reduction in the tank radiation reading to 3 R/h. Because the solution could not reach the tank hot spot, which was located in a filter leg on the tank top, the radiation reading of 100 R/hr at that spot was not affected. The solution was then pumped to the other identical tank and the process repeated, giving similar results.

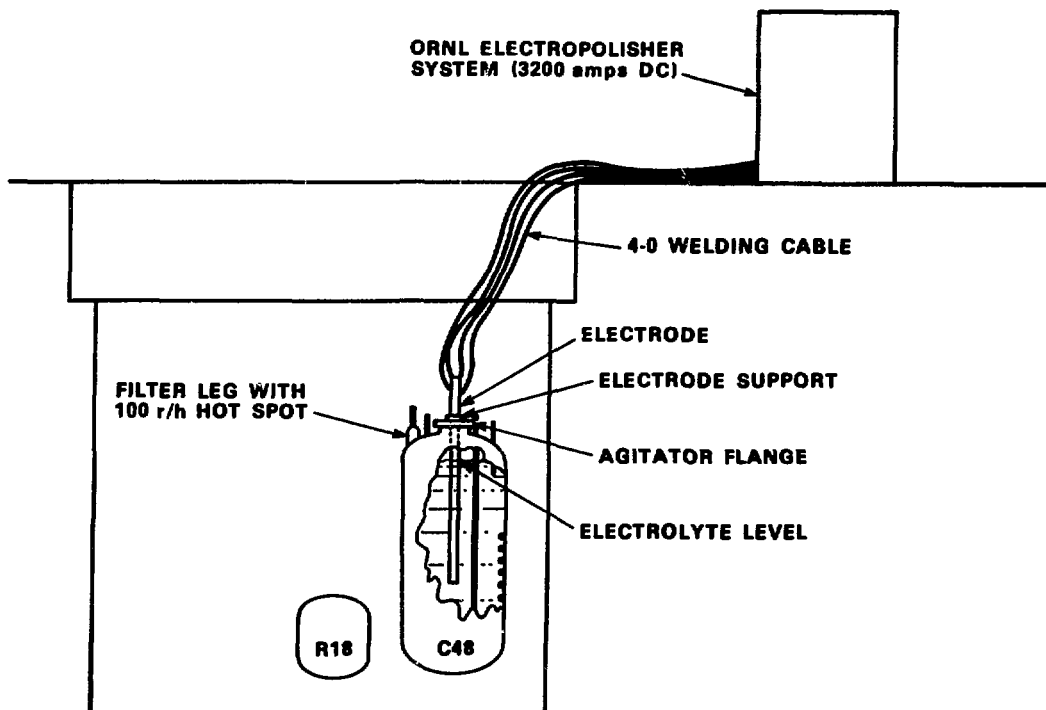


Figure 1. Electropolishing Setup.

Table 1. Electropolishing Data

Tank No.	Area ft <sup>2</sup>	Current A	Current Density A/ft <sup>2</sup>	Time m	Radiation	
					Initial R/h	Final R/h
C-38	55	1600	29.1	15	10	3
C-48	55	1600	29.1	45*	10	3

\* Longer time due to not pre-heating solution.

Several problems were encountered in using the electropolishing method. First, pre-heating the solution proved unnecessary as the power dissipated by the electropolishing current provided adequate heating. Second, the phosphoric acid solution was too dense to be easily pumped by steam jet, and no suitable submersible pump could be found. Therefore, solution was transferred by siphon from one tank to another and required makeup with fresh phosphoric acid. Third, the hot spot on one tank which was located in a filter leg on top was not affected as the solution could not contact it. This represents the most serious limitation of the method. The radiation was reduced at this point by burning a hole into the housing which surrounded the filter leg and washing out the contamination with a high-pressure spray.

Another method which proved successful for reducing the radiation reading in a tank was cerium redox decontamination.(2,3) This technique involves the use

of a solution of 4-molar nitric acid with 0.1-molar cerous nitrate  $[\text{Ce}(\text{NO}_3)_3 \cdot 6\text{H}_2\text{O}]$  present, together with an electric current. In this process, the  $\text{Ce}^{+3}$  ion is oxidized by the current to  $\text{Ce}^{+4}$  which, in the acidic medium, attacks stainless steel and is reduced back to the +3 valence state. The electric current then re-oxidizes the  $\text{Ce}^{+3}$  back to the +4 valence state. This electrochemical reaction continues as long as the electric current is applied. A smaller 40-gallon tank located in the same cell as the two tanks discussed above was decontaminated using this method. This tank was used as a vacuum receiver for decanted supernatant from the two crystallizer tanks and had no agitator. Due to this operating history, much higher radiation levels of >100 R/h were encountered since the conventional decontamination methods used before were not as effective.

The equipment and setup used were the same as were used in electropolishing the other two tanks. Since the tank was totally closed, a hole was drilled in the top using a holesaw. This provided an opening both for solution addition and for the electrode. Since the electric current only provided for the chemical reaction, a much lower current was required. An average current of 59 A was applied for eight hours resulting in the reduction of the radiation reading from >100 R/h at 12 in. to 10 R/h at contact.

#### PLASMA-ARC CUTTING TORCH

Another method for reducing the exposure of personnel during decommissioning operations is the use of a plasma-arc torch for cutting pipe. This technique has been extensively used at the former Nuclear Fuel Services, Inc., fuel reprocessing plant at West Valley, New York.(4) The same technique has now been demonstrated and used at ORNL and has been proved reliable.

Before the plasma torch was developed, the standard method of pipe cutting for contaminated equipment removal at ORNL was sawing either with a sabre saw or a rotary abrasive saw. Because of the high levels of both contamination and radiation, either saw had to be used in conjunction with extension poles to reduce personnel exposure. An abrasive saw produces excessive airborne contamination, and therefore, its use is limited. The sabre saw produces limited airborne contamination, but it is slow cutting and requires several blade changes during a single cut which, of course, must be done in a high radiation field.

Subject to some limitations, the plasma-arc torch has eliminated many of these exposure problems. Instead of fifteen minutes or longer to make a single cut with the sabre saw, the plasma-arc torch can make the same cut in five seconds. The savings in both effort and exposure are obvious. At ORNL, since crafts are used to perform the actual equipment removal, the required craft has been shifted from pipefitter to welder. This has caused some problems in supplying enough gas-mask-certified welders. The Union work rules have also required the use of a pipefitter in addition to a welder, so that two people are receiving exposure instead of one, but the plasma-arc torch has produced significant reductions in personnel exposure anyway.

The equipment used is a standard, off-the-shelf plasma-arc cutting torch manufactured by the Thermal Dynamics Corporation. The PAK-45 torch, illustrated schematically in Fig. 2, is used with an argon-hydrogen mixture as the plasma gas and  $\text{CO}_2$  as the shield gas. The system can also use other gas mixtures and can be made to operate under water. The present mixture was selected because it

gives the least amount of smoke during operation. The only modification made was to the torch itself and consisted of equipping it for remote operation. This was composed of taping up the start switch and fabricating a bracket to keep the torch a constant distance (0.25 in.) from the work. A later addition was that of replacing the 70° torch with a machine torch, thus eliminating the need for any modification. Both torches were operated from the end of a ten-foot-long pole.

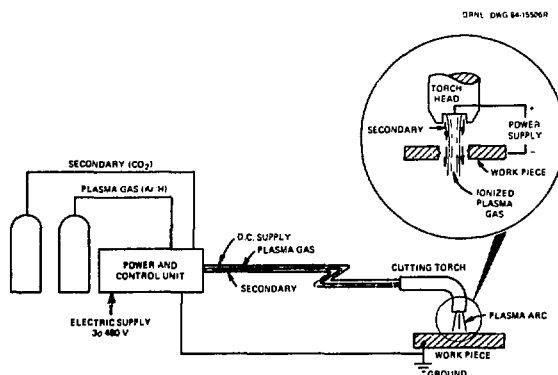


Figure 2. Schematic of Plasma-Arc Torch.

The main limitation of this technique is preventing vaporization of any hazardous materials which may be contained inside of the pipe when it is cut. Because of the high temperature associated with the plasma-arc torch, there is the potential for volatilization of materials which are contained within the pipe. Care must be exercised in order to minimize the quantity of material which could be volatilized. At the FPD, the cells are top opening, and containment is provided by maintaining proper airflow direction. Only pipe reading less than 25 R/h is allowed to be cut with the plasma-arc torch which has been calculated to be the equivalent of less than one curie of cesium-137. These controls, together with tested HEPA filters located on the ventilation discharge from the building and observation of the building particulate monitors during cutting, have been successful in controlling activity releases.

Table 2 compares four cells which were decommissioned between August 1983 and May 1985. The equipment in the first cell listed was removed using conventional cutting methods. The equipment in the latter three cells utilized plasma-arc cutting. Direct comparisons between the results of Table 1 are difficult because of the fact that different materials were processed in each cell. Cell 1 was a load-in, load-out cell and never had any processing done in it. Cells 2 and 3 were used for strontium-90 processing, and as mentioned before, Cell 8 was used for cesium-137 processing. It should be expected that radiation levels and their resulting exposures would be lower for the beta radiation present in Cells 2 and 3.

However, had the data been taken from like cells, there still would have been a significant reduction in personnel exposures by using the plasma-arc torch because of the reduction in working time. It has been estimated that, had the plasma-arc torch been used on Cell 8, the total exposure would have been approximately 6 rem.

Another application of the plasma-arc torch to the removal of equipment from a cell was using the torch remotely in a manipulator cell. The equipment in a stainless-steel-lined manipulator cell was removed in this manner. A

Table 2. Comparison of Cells Decommissioned with Mechanical Cutting and Plasma-Arc Cutting

Cell No.	Radiation Level R/h*	Manhours	Method	Personnel Exposure man-rem
8	1-2	215	saw	15.05
1	1-2	55	torch	0.92
2	3-4	180	torch	1.25
3	6-7	170	torch	1.68

\* At top of cell after decontamination, unshielded. Does not indicate hot spots up to 100 R/h.

smaller version of the PAK-45 called the PAK-10 was procured, and using the same cutting conditions and restrictions, the in-cell equipment was removed without entering the cell. The only modifications required are illustrated in Fig. 3. These were to the torch itself and consisted of manipulator grips and a vertical plate to keep the optimum cutting distance from the work. The cell false floor and all in-cell piping were cut out and sectioned into pieces of a maximum size of six inches square. These pieces were then decontaminated by acid soaking to a level requiring only contact handling. All of this was done without significant personnel exposure, except that received during waste loadout.

ORNL-DWG 85-17718

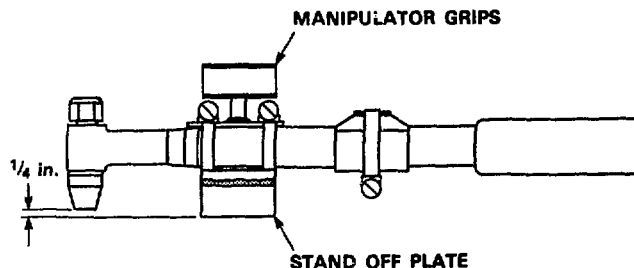


Figure 3. Modifications to PAK-10 Torch for Manipulator Cell Use.

#### HIGH-PRESSURE WATER JET SCARIFICATION AND CUTTING SYSTEM

In certain facilities, where contamination controls were either not available or not closely adhered to, contaminated concrete has become a problem. A widespread need throughout the nuclear industry is that of a proven method for concrete decontamination. A method which has shown promise both at the DOE Hanford reservation and at ORNL has been scarification of the concrete with high-pressure water. This technology has been developed for the mining and construction industries. The method of moving a lance containing two jets of water pressurized to 35,000 psi in a controlled manner over a surface thus scarifying it utilizes existing equipment which has been specially modified for decontamination purposes. The system can also be operated as a concrete cutting tool by using a different lance. At ORNL, the system has been used only for cutting uncontaminated concrete.

The primary equipment for this work is supplied by Flow Industries, Inc., of Kent, Washington. The high-pressure water jet scarification system now in

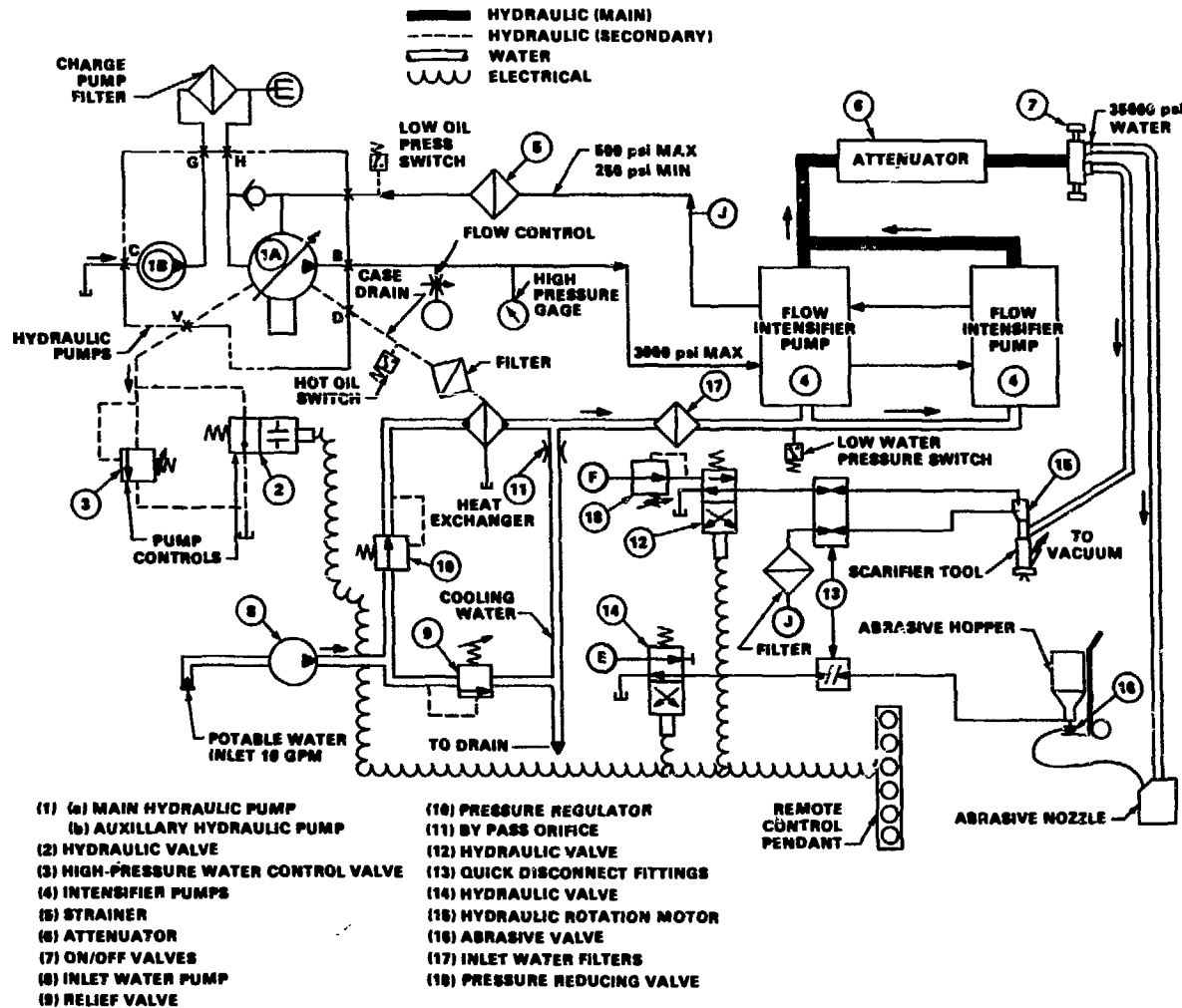


Figure 4. Schematic of High-Pressure Water Jet System.

use at ORNL is shown schematically in Fig. 4. The main power unit consists of a hydraulic oil pump which drives a special high-pressure pump called an intensifier. The ORNL system is driven by a 75-HP electric motor. Inlet water is supplied to a conventional pump and is pressurized to 80 psi. The water is then fed to the intensifier which pressurizes it to up to 35,000 psi. This is controlled by regulating the hydraulic oil pressure from 0 to 3000 psi by means of a manual control valve. The pressurized water is fed through an accumulator to the scarification lance and scarifies the concrete on which it impinges. Surrounding the scarification lance is a shroud which collects the spent water and the concrete particles which are removed. This stream flows into a waste collection system as illustrated in Fig. 5.

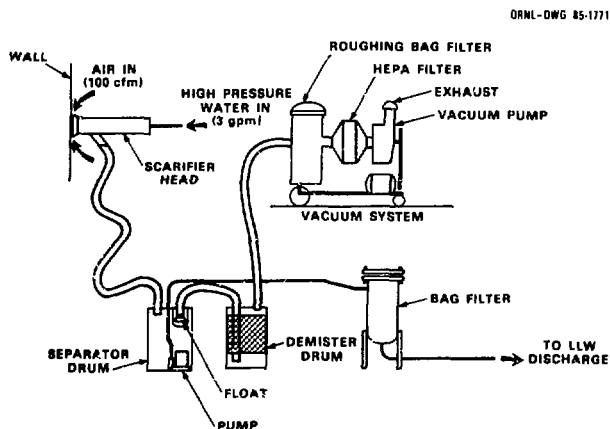


Figure 5. Waste Collection System.

The effluent first flows through settling drums where the entraining air and large concrete particles are separated. The water and concrete fines flow through a filter before being discharged as low-level liquid waste. The air flows through a demister drum before being HEPA filtered and discharged. The filter system has not been field tested.

A method for moving the scarification lance and shroud over a surface in a controlled and predictable manner is a necessary feature for using the system for decontamination. A system has been conceptualized to perform this task and is illustrated in Fig. 6.

The system is fabricated out of commercially available automatic welding equipment and consists of three sections of track to which are attached three tractors. The scarification lance is attached to the center tractor as shown.

The operation of the cutting lance is essentially the same as the scarification lance, except there is no waste collection system and the addition of abrasive to the high-pressure water is required. The recommended abrasive is garnet, but at ORNL, sand has proven to be a satisfactory substitute that is considerably less expensive.

At ORNL, the only field use of the high-pressure water jet system has been that of cutting noncontaminated concrete. A project to cut a new viewing

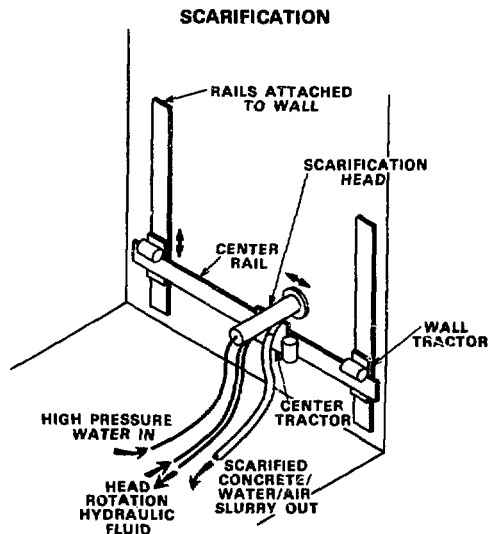


Figure 6. Wall Tracking System.

window in a hot cell facility was undertaken and completed; this involved making a blind cut through a 24-inch-thick concrete wall. This is a greater thickness than the system will cut on one pass, so multiple passes were required in making the cuts. There was a significant problem in cutting the last six-inch thickness of wall due to the fact that the water jet would not focus and tended to wander out of its intended path. This was solved in the following manner. First, a wedge was cut out toward the inside of the work. Then, the cutting lance was positioned inside the plane of the wall. Instead of using the standard length of focusing nozzle (4 in.), a 12-in. nozzle was procured and used. These techniques produced satisfactory results. Another problem encountered was that of the tracking mechanism clogging up with grit from overspray. The tracking equipment used was similar to that described above for the scarification system but had only one-dimensional travel. This problem was never solved in a satisfactory manner. Periodic cleaning of the track and tractor proved an acceptable but less than optimum fix.

A development program is planned for testing the scarification, waste handling, and tracking concepts on uncontaminated concrete. After optimizing the system operation, further development of decontamination techniques on contaminated concrete will be undertaken prior to initiating cell decontamination operations in the Metal Recovery Facility.

#### REFERENCES

1. R. P. Allen, H. W. Arrowsmith, L. A. Charlot, and J. L. Hooper, "Electro-polishing as a Decontamination Technique", USDOE Report PNL-SA-6858, Battelle Pacific Northwest Laboratories, Richland, Washington, 1978.
2. R. E. Lerch, "Corrosion of Stainless Steel in Solutions of Cerium(IV)-Nitric Acid", USAEC Report BNWL-CC-1646, Pacific Northwest Laboratory, Richland, Washington, May 1968.

3. Jerry A. Partridge, Ronald E. Lerch, Gail P. Bosuego, "Decontamination of TRU Contaminated Metals", Pacific Northwest Laboratories, August 1979.
4. E. Michael Blake, "Cleaning Up West Valley", in Nuclear News, Vol. 28, No. 9, July 1985.

**JOHNSTON ATOLL SITE CLEANUP - SOME PRACTICAL CONCERNS**

**Lt. Bruce E. Vesper,  
Lt. T.H. Mohaupt,  
Cpt. M.A. Dooley,  
Armed Forces Radiobiology Research Institute  
Bethesda, MD 20814-5145**

**and**

**Dr. E. Bramlett, Field Command  
Kirtland AFB, NM 87116**

**ABSTRACT**

Johnston Island in the Pacific Ocean was contaminated as a result of weapons testing in the early 1960's. A recent cleanup project removed approximately 540 tons of actinide contaminated debris from a missile launch site. The site removal effort consisted of monitoring all debris for contamination levels, fixing contamination in place and packaging materials to meet DOT regulations for both land and water transportation. This report presents details of this recent practical project in site decontamination and details decommissioning of an actinide only contaminated site.

## POST-ACCIDENT RECOVERY OPERATIONS AT TMI-2

Dale J. Merchant  
GPU Nuclear Corporation  
Middletown, PA

James E. Tarpinian  
Bechtel National Inc.  
Oak Ridge, TN

### ABSTRACT

The accident at Three Mile Island Unit 2 resulted in severe radiological conditions in many areas of the station. Personnel entry into some of these areas was required for plant stabilization and radwaste management although the radiation levels were extraordinarily high. At the time of the accident remotely operated equipment was not immediately available that could perform the required tasks in lieu of manned entries. Because of the plant layout, physical constraints, and required versatility, the type of remote equipment needed would have been extremely sophisticated. In fact, attempts were made to find equipment to do this work but none was found that possessed the required level of sophistication or versatility. Personnel radiation exposures may have been significantly reduced had such devices been available for immediate use and designed to perform the necessary tasks. This paper reviews the immediate post-accident recovery tasks with a perspective on the functional criteria for the development of remotely operated equipment which could be used in similar circumstances in the future. Also, recommendations are made for the possible development of plant installed equipment which could be designed to be compatible with remote equipment operations and thus reduce the degree of sophistication needed.

Although post-accident recovery operations are significantly different from decommissioning operations the functional requirements for mobile teleoperators can be very similar. Remote equipment designed to meet the functional criteria described herein will possess the necessary capability for access and data acquisition for most applications during the decommissioning of a nuclear facility.

### INTRODUCTION

The Three Mile Island Nuclear Generating Station (TMI) is a two unit, pressurized light water reactor facility located 10 miles from Harrisburg, Pennsylvania. On March 28, 1979, Unit 2 experienced a reactor shutdown due to a loss of feedwater to the "B" steam generator and subsequent turbine trip. The resulting series of events culminated in a loss of reactor coolant and a partially uncovered core. The fuel cladding failed, due to the dramatically increased core temperatures, which introduced large amounts of radioactive fission products into the reactor coolant system.

Following the turbine trip, the open pressure-operated relief valve (PORV) on the pressurizer permitted reactor coolant to fill the reactor coolant drain tank. Fifteen minutes after the turbine trip the reactor coolant drain tank rupture disc failed due to overpressurization and primary coolant flowed to the reactor building sump causing the basement to begin filling with water. The reactor building sump pumps started automatically and transferred approximately 8100 gallons to the auxiliary building sump tank before pumping was secured. Since the available capacity of the auxiliary building sump tank was only 700 gallons and the rupture disc was broken, liquid overflowed to the auxiliary building sump, causing water to back up through the floor drains in both the auxiliary and fuel handling buildings (1). This resulted in several inches of water on the auxiliary building basement floor and very high radiation fields.

At the same time, very high fission product activity and even fuel debris circulating in the primary coolant were introduced into piping systems in the auxiliary and fuel handling buildings. The systems included makeup and purification, seal injection, reactor coolant liquid radwaste, and radwaste disposal. These piping systems, which were designed to normally contain and process reactor coolant and associated liquid wastes, created radiation fields far in excess of the normal plant operating conditions. Despite the extreme radiological conditions, manned entry into the auxiliary and fuel handling buildings was required for several weeks after the accident in order to maintain the operating status of certain plant systems and to attempt to bring the plant into a stable configuration.

Manned entry into the reactor building was not attempted for over a year following the accident. High radiation fields were caused by the presence of fission products that were dispersed by steam and circulating air currents and by fission gases, notably Kr-85. The reactor building basement contained 600,000 gallons of water with fission product concentrations of 160 microcuries per milliliter and attendant radiation fields of 20 to 40 rem per hour. The radiological conditions were difficult to assess because direct measurements were not possible. The reactor building radiation monitors had failed during the accident and only limited access to the reactor building could be obtained through penetrations. Although entry into the reactor building was not required to maintain the plant status, it was important to try to determine the structural condition of the building and plant. As determined a couple of days later, hydrogen gas was vented to the reactor building through the reactor coolant drain tank rupture disc and was ignited on March 28, 1979 causing a pressure spike of 28 psig. This elevated pressure activated the containment spray system which introduced about 17,500 gallons of borated water and sodium hydroxide solution into the reactor building (2). The combined effects of the hydrogen burn, alkaline sprays, elevated building temperature, sustained high radiation fields, and prolonged condition of 100% humidity were unknown. Attempts to survey the buildings for damage could only be attempted through penetrations in the reactor building walls.

At the time of the accident remotely operated equipment was not immediately available that could perform the required tasks in lieu of manned entries. Because of the plant layout, physical constraints, and required versatility, the type of remote equipment needed would have been extremely sophisticated. In fact, attempts were made to find equipment to do this work but none was found that possessed the required level of sophistication or versatility. However, if such equipment had been available it would

have certainly been used. This paper describes the tasks which were required to be performed with a perspective on the functional criteria for the development of remotely operated equipment which could be used in similar circumstances in the future. Also, recommendations are made for the possible development of plant installed equipment which could be designed to be compatible with remote equipment operations and thus considerably reduced the degree of sophistication needed.

#### POST-ACCIDENT RECOVERY OPERATIONS

Despite the radiological conditions and associated constraints to personnel access, certain tasks were required to be performed for plant stabilization and for the protection of the health and safety of the public. The tasks required plant operations personnel to enter the auxiliary and fuel handling building (AFHB) to manipulate valves and to direct the transfer of liquid and gaseous radiological waste from a master control panel located in the auxiliary building. Chemistry personnel were required for sampling and analysis of the reactor coolant system (RCS) and other plant systems. Radiological controls technicians were needed to obtain information about radiological conditions and maintain the station airborne effluent monitoring system. Each of these tasks involved repeated entries into high radiation areas.

The tasks involving chemistry sampling and operations valve lineups were the most exposure intensive. An RCS sample taken the day after the accident was greater than 1000 rem per hour on contact with the sample container and 400 rem per hour at a distance of one foot. The chemist who obtained that sample received 4.1 rem whole body exposure during sampling operations (3). Exposures between 0.5 rem and 1.5 rem per entry were frequent. Radiological controls technicians were required to strictly limit the stay times of personnel entering high radiation areas, prescribe pathways to and from the work site, and provide radiation monitoring devices in order to keep exposures as low as reasonably achievable. Communications were essentially limited to the plant paging system. Contamination protection for personnel included cloth and plastic coveralls, boots, gloves, hoods, and respiratory protection devices. One consideration not normally encountered in power plant operations was the incremental reduction of worker mobility with each layer of protective clothing, and the subsequently increased time required to perform a task. Because certain entries could have caused a person to receive his quarterly dose limit in just a few minutes, reduced protective clothing was prescribed in order to reduce radiation exposure.

#### Sampling the Reactor Coolant System

Reactor coolant samples were required frequently for several weeks after the accident. This operation required access to sampling locations, operation of sample valves, and transporting and handling small sample containers.

Because frequent sampling resulted in such high individual and collective worker doses the NRC regulations for post-accident sampling systems have been upgraded. The "lessons-learned from TMI" plant modifications included a revised system design to allow obtaining the highly radioactive liquid samples more efficiently and with lower doses for personnel.

## Processing Liquid Radioactive Waste

Continuous movement and processing of liquid radioactive waste was necessary due to limited tank storage capacity. These operations were conducted from the radioactive waste master control panel in the auxiliary building. The tasks included accessing the control panel; reading gauges, meters, indicator lights and alarms; manipulating switches and push buttons; and maintaining communications with the plant control room.

Initially, the dose rates in front of the master control panel were about 8 rem per hour. Several weeks later they were reduced to 0.05 rem per hour. Although these radiation levels were not high compared to other areas in the plant requiring access, the number of hours spent in this area resulted in high collective personnel doses. Establishing a decontaminated pathway to this area, and lowering the dose rates in the area, became the top priority for the initial recovery of the auxiliary building.

## Maintaining the Plant Status

Certain operations were critical to maintaining the plant in cold shutdown condition. Regardless of high radiation levels, operators had to access virtually all areas of the plant to perform valve lineups and verifications. An example of such a critical operation is an entry into the seal injection valve room a week after the accident in order to restore seal water to a reactor coolant pump. Dose rates of 400 rem per hour were measured adjacent to the valve. In this radiation field a person could reach his quarterly dose limit in about thirty seconds. Access to the valve was made difficult by narrow passageways and interferences. The entry team members were also encumbered by protective clothing and by self-contained-breathing-apparatus respiratory equipment. Despite these adverse conditions the job was successfully accomplished within established dose limits.

## Radiation Monitoring and Sampling

A large percentage of AFHB entries made by radiological controls personnel during the first week after the accident involved the replacement of the roll-type filter paper on the plant ventilation system monitors. The radioactivity collected on the filter paper very rapidly, requiring frequent filter changes in order to keep the detector and meters within their operating range. In addition, the only available access for obtaining reactor building air samples was located in a high radiation area in the AFHB. These operations required access to most areas of the plant; manipulation of small clips, screws, and switches; retrieval of filters; and the reading of meters and gauges.

The need for radiation surveys in the AFHB was limited for two reasons:

- 1) installed plant area radiation monitors provided an indication of radiation levels in many areas, and
- 2) plant operations personnel were allowed to carry their own radiation monitoring devices because this was the most dose effective means of conducting building entries. This is contrary to normal plant operating practices because radiological controls technicians are usually used to perform radiation surveillances. As a rule, radiological controls technicians did not enter the affected areas of the plant for

the sole purpose of radiation surveys, rather surveys were performed in conjunction with other tasks.

#### Initial Reactor Building Entry

The first entry into the reactor containment building occurred on July 23, 1980, sixteen months after the accident. This event marked a milestone in the recovery of the damaged reactor and containment building. Until that point, information about the physical and radiological conditions of the building was sparse as measurements had to be obtained remotely.

The first two entries consisted primarily of visual and radiological surveillance. The initial entry occurred after the venting of about 40,000 curies of Kr-85 from the reactor building. The venting allowed some relief from protective clothing requirements because of the reduced beta radiation dose rates. The general area radiation levels were on the order of 0.7 rem per hour. Lighting was generally good, but not available in all areas. The building did not appear to have suffered significant structural damage. The most extensive damage was apparently due to surface rusting of exposed steel surfaces to moisture. The reactor building temperature was generally around 90°F, and a constant 100% humidity caused condensation in the dome region and a subsequent "rain forest" environment. Some minor damage occurred as a result of the hydrogen burn and there was some evidence of chemical damage due to the sodium hydroxide sprays (4).

The availability of remotely operated equipment to perform the visual and radiological surveillance much sooner in the program would have considerably eased the concern over the damage to the building. Also, earlier knowledge of the building's radiation levels would have greatly aided in recovery and decontamination planning efforts.

#### FUNCTIONAL REQUIREMENTS FOR TELEOPERATORS

The operations for which teleoperators would have been useful in the recovery of TMI-2 immediately after the accident are varied and complex. The functional criteria for the design of teleoperators are dictated by the operational needs for plant access, valve and switch manipulation, equipment changeout, sample acquisition, and radiation monitoring.

##### Access Requirements

Unfortunately, access to most of the areas of the plant was made difficult by obstructions. Ideally, a teleoperator would have to have the flexibility to access the same areas as a man. Such flexibility would require cableless signal/power transmission in order to negotiate airlocks, travel to distant areas of the plant, or negotiating complex routes.

Physical constraints that may restrict the movement of remote vehicles include:

- 1) doors and gates - unlocking, opening, and closing of doors are necessary.
- 2) piping obstacles - field run pipe and supports, along with valve reach rods, will hinder the vehicle's path. The ability to negotiate these obstructions requires extreme dexterity because pipe can run along the floor at heights ranging up to 24".

- 3) corridors and labyrinths - due to the many cubicle access labyrinths designed for radiation shielding purposes, and narrow valve and pipe corridors, the vehicle must be sized small enough to proceed through these to reach its destination. In addition, its turning radius must allow for the difficult maneuvering involved.
- 4) water - the remote teleoperator must be able not only to move through standing water, but to remain functional after being sprayed or rained on by leaking valves or pipes. Electronic equipment, camera lenses, meter covers, and lighting would need protection.
- 5) stairways - the ability to climb and descend stairs is essential. Elevators are generally unreliable following an accident due to possible power failures or flooding of the pits, as occurred after the TMI accident.

#### Manual Dexterity

Many operations involved tasks of varying complexity ranging from simple valve operation to removing small clips or handling glassware. Valve sizes ranged from 1" to 8". Because the requirements are so varied, a teleoperator system must be flexible and versatile. The versatility could be supplied by modular components or by a series of different machines.

Force reflective feedback is essential for most of the described operations because of torque limitations on valves and switches and the delicate nature of sample containers. Precise programming may be a suitable alternative to force reflective feedback for identical and repetitive operations such as RCS sampling. However, consideration must be given to changing differential pressures when tuning valves during system operation.

Some operations, such as filter paper changeout, may be so complex and require such a high degree of manual dexterity that it may be more practical to redesign the plant equipment to be more easily serviced by remote teleoperators. In the case of plant ventilation radiation monitors, this could be accomplished by designing a cartridge-type of filter holder as opposed to designed models using small pins, clips, and screws.

#### Communications

The teleoperator must be able to perform visual surveillance of areas, requiring wide angle viewing, as well as to read dials and gauges, requiring high resolution. Obviously, the camera equipment must be capable of giving enough of a field-of-view to allow the equipment operator to guide the machine to its destination. Portable illumination is also required since the availability of lighting may be uncertain following an accident.

The remote signal capability must allow two way communication with the operators. This may be accomplished with an umbilical cord, however this system is limited, as previously discussed. Wireless transmission may be made difficult within the plant, however, because of interference and shielding due to thick, reinforced concrete walls.

Radiation surveillance would require the remote transmission of radiation data. Radiation instrumentation should be capable of monitoring conditions over a wide range of dose rates. Instruments should be capable of detecting gamma (penetrating) and beta (non-penetrating) radiation separately.

## General

In a nuclear plant environment, an instrument must be capable of withstanding assault from radiation fields caused the both gamma and beta radiation. The most sensitive components of a teleoperator would be in the electronics or the camera lenses. However, most radiation environments in nuclear power stations will not be sufficiently high to cause concern for even these components. Also, the decontamination of the equipment will be required. Wherever possible, external surfaces should be made smooth and not subject to damage by water. Internal surfaces should be sealed.

The reliability of the equipment must be very high, especially the transporter. A breakdown in a high radiation area would make retrieval difficult and may require a manned entry. In the same light, remote vehicles would require high capacity batteries to sustain long entries.

A very important consideration for the use of remote teleoperators is their availability. Shortly after an accident, or any event requiring access to high radiation areas, remote teleoperators can be invaluable dose saving devices. The highest radiation dose rates and some of the highest individual exposures at TMI occurred within the first few days of the accident. With this constraint there is inadequate time for the development of new equipment. Thus, prompt delivery of appropriate equipment and the availability of trained operators is an essential feature for remote equipment if it is to be used for maximum benefit. Yet the cost of such equipment and operator training would probably be sufficiently large that individual utilities would not consider their purchase during construction or even normal operations simply to have as part of their emergency equipment. A DOE sponsored activity to develop and provide this type of equipment in the event of a nuclear emergency would be a more feasible option, since the prospects for commercial applications may not be encouraging. If teleoperators meeting the previously described functional requirements had been available to the recovery staff at TMI, collective personnel doses may have been considerably reduced.

Under the current recovery program, teleoperators and remote equipment are being developed to perform specific tasks in some remaining high radiation areas. Although these work tasks are different from those described in this paper, the functional requirements are similar in many cases. Thus, the development of teleoperators to aid in TMI-2 recovery may lead to future development of equipment which can be used in the event of nuclear emergencies.

## REFERENCES

1. Electric Power Research Institute, "Characterization of Contaminants in TMI-2 Systems," EP-2922. EPRI, Palo Alto, CA, 1983.
2. Bechtel Power Corporation, "Planning Study for Containment Entry and Decontamination," Gaithersburg, MD, 1979.
3. Nuclear Regulatory Commission, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG 0600. NRC, Washington, DC, 1979.

4. Tarpinian, J.E. and Hollenbeck, P., "A Summary of the Radiological and Physical Conditions of the Three Mile Island Unit-2 Reactor Containment and Their Impact on Recovery," in Energy, Environment, and the Economy edited by Shymal K. Majumdar, pp 219-228, The Pennsylvania Academy of Science, Easton, Pennsylvania, 1981.

## THE IMPLICATION OF REMOTE TECHNOLOGY FOR DECONTAMINATION ACTIVITIES

B. G. Kniazewycz  
KLM Technologies, Inc  
2700 Ygnacio Valley Road, Suite 160  
Walnut Creek, California 94598-3454

J. T. Irving, D. C. Long, and H. Wagoner  
Niagara Mohawk Power Corporation  
Nine Mile Point - Unit 1  
P.O. Box 32  
Lycoming, NY 13093

### ABSTRACT

The use of robotics or remote technologies to remove man from hostile environments has long been a driving objective for technology development within the nuclear power industry. The Niagara Mohawk Power Corporation (NMPC) Nine Mile Point Nuclear Station is taking an organized approach to integrating remote technology into routine decontamination activities, non-processed radioactive waste disposal and special decontamination projects.

Preliminary tests of an underwater robotic device for scrubbing and vacuuming was conducted at Nine Mile Point Unit 1 (NMP-1). The results of the test successfully provided "proof of concept" and identified improvements which would render the device acceptable for nuclear power station applications.

Niagara Mohawk Power Corporation is undergoing a major radwaste retrofit project involving almost the entire Radwaste Building at NMP-1. The pre-demolition decontamination scenario was determined to be beneficial at reducing the project radiation exposure. Applications of remote technologies were identified to be cost beneficial for several project tasks including inspections, radiation survey mapping, debris sampling and removal, and video documentation.

The purpose of this presentation is to discuss the remote technologies application program at Nine Mile Point and present in detail results of the projects mentioned.

### INTRODUCTION

The purpose of this presentation is to describe the efforts made by personnel at the Niagara Mohawk Power Corporation (NMPC) Nine Mile Point Nuclear Station to become involved with the emerging technology of robotics and remote technology. The word robot has taken on a variety of definitions and misconceptions, however, the key point in any definition is that once programmed, a robot acts autonomously without a human intervening in the control loop. The only problem is no such system exists today for applications of interest to the nuclear industry. The proper terminology used to discuss the machines presently available is "teleoperated mobile manipulators" since they are remotely controlled by a human operator. This terminology is somewhat cumbersome, thus the word robot will be used in this paper. The robotics and remote technology activities to be discussed are funded by the NMPC Research and Development Department.

In September 1984 the multi-departmental Nine Mile Point site ALARA Committee recognized robots and remote technologies as a viable means to reduce personnel radiation exposure and thus warranted further investigation. The ALARA organization contacted the NMPC R&D Department to discuss funding and engineering assistance for developing a program to introduce robotics to Nine Mile Point Unit 1 (NMP - 1). The proposed project fit the R&D criteria of having a high potential benefit and probability of success, but with more risk than justifiable for Corporate capital expenditure. Therefore, it was decided that the site ALARA Coordinator and the R&D representative would jointly provide overall program leadership and for each particular project a site "sponsor" would serve as that project coordinator. KLM Technologies, Inc. served as the R&D representative for this program.

The first order of business for the robotics program was to investigate the available technology and determine applicability to NMP-1 short and long term needs. The mobile robots examined were propelled by either wheels, tracks or legs and were powered through either a tether or an on-board battery pack. The particular applications of interest to NMP were those robots capable of underwater decontamination, surveillance, in-service inspection, maintenance and possibly operational support. The following discussion briefly describes a few of the mobile robots examined and then returns to the discussion of the NMPC program specifics, accomplishments and future plans.

## MOBILE TELEOPERATED ROBOTIC DEVICES

This section briefly reviews the information available regarding specific mobile teleoperated robotic devices which are employed in or proposed for hazardous environments. These devices can be used to inspect, monitor, locate, identify and/or maintain components and items as well as act as mobile surveillance devices while monitoring various constituents in the hazardous environment such as temperature, humidity, noise and radiation.

Over 50 devices have been identified by various sources as mobile teleoperated robotic devices. These range from one of a kind specialty devices, to experimental mobile vehicles, to proof-of-concept robots as well as a limited number of "commercial" offerings. Many of these devices have been developed by U.S. national laboratories, government funded projects, foreign entities for general hazardous environmental applications as well as specialty applications such as ordinance disposal. Based upon a variety of surveys, a number of sensory capability and design requirements for nuclear mobile robots have been identified. These include:

1. Autonomous navigation
2. Radiation (and chemical) detection
3. Collision avoidance
4. Heat source detection
5. Temperature and humidity detection
6. Capability of manipulating various loads
  - light (0-5 kg)
  - moderate (5-25 kg)
  - heavy (greater than 25 kg)
7. Object detection
8. Smoke detection
9. Sound detection
10. Vision capability

Sources utilized to support this effort included numerous robotic and remote technology meetings, symposiums and vendor contacts; various government sponsored reports; EPRI sponsored research; and, various articles in trade magazines including Nuclear News, Electric World and Spectrum.

A brief description of various mobile robotic vehicles follows. These and others were reviewed as potentially useful in support of the NMPC robotic and remote technology program.

### AUTOMATION TECHNOLOGY CORPORATION, COLUMBIA, MD.

**SURVEYOR™** - This 2-tracked, remotely controlled tetherless device is used to conduct surveillance and inspection missions in nuclear power plants. Surveyor's™ relatively light weight of less than 330 lbs, including a telescoping manipulator arm, enables it to be easily transported manually or driven from location to location. The device is able to climb 45 degree stairs, and has a maximum speed of 90 ft/min. The lifting capability of the arm is 5 lbs. Surveyor™ can traverse through 6 inch deep water and over 9 inch high obstacles. The power supply is provided by on-board lead acid 24v batteries which can operate up to 4 hours before recharging. The maneuver-ability is enhanced by 3-D stereo images produced by a pair of on-board color CCD cameras which are communicated back to the teleoperator's control panel via a high radio frequency, along with the other sensory and operating data. The vehicle is able to be decontaminated. The supervisory control panel contains 3 TV monitors and a microcomputer which monitors all performance and sensory data. This is a commercial offering.

### BATTELLE - COLUMBUS LABORATORIES, COLUMBUS, OH

**ROCOMP** - This tetherless, remotely controlled 2-tracked device has been designed for nuclear applications and chemical/explosives handling and hazardous environments. The radio controlled platform can climb stairs and has two speed ranges: 0-90 and 0-150 ft/min. Its two tracks can be replaced by wheels to facilitate rapid movement over flat terrain. The platform is designed to perform 5 tasks: 1) detect radiation; 2) collect air samples; 3) obtain smear samples; 4) perform mechanical tasks and 5) act as a tool caddy. In addition to a 4-axis manipulator arm, the device will have a video camera. The on-board power source of two 12v batteries permits continuous operation of 2-4 hours between battery charges. The ROCOMP platform dimensions are 54 x 28 x 18 (lwxh) inches; it can carry loads of up to 250 lbs. On-board computers will enable ROCOMP to support some form of autonomous navigation. This was apparently designed for internal use.

**BLOCHER - MOTOR GMBH & CO. KG, METZINGEN, WEST GERMANY - CMS TECHNOLOGIES, INC. FT. LEE, NJ US DISTRIBUTOR**

**MF3** - This device is a remotely controlled, tethered 4-tracked vehicle which is used in the nuclear industry and other hazardous environments. It was initially conceived and developed at the KFA Julich Research Laboratory in West Germany. Its single, light duty electric powered manipulator arm can lift up to 44 lbs; the heavy duty arm can lift up to 176 lbs. Both arms have 6 axes of movement and possess infinitely rotating tong openings. The MF3 is remote controlled from a portable control desk located up to 100 m away. The MF3 dimensions are 89.1 x 28.3 x 15.7 (lwxh) inches; with track adjustment, the length and height are, respectively, 37 and 42.5 inches. It can climb stairs with a gradient of up to 45 degrees, turn on a 47.2 inch radius, and can surmount 23.6 in. high obstacles, and traverse 3 ft wide chasms (gaps). Its maximum speed is 99 ft/min. Optional accessories are video cameras, TV monitor at the control desk, headlights, noise transmissions system, X-ray unit with mounting arm, and alternate grippers. Power (220v, 50 Hz) and communications is made through an umbilical cord (cable). This is a commercial offering.

#### **CARNEGIE - MELLON UNIVERSITY, PITTSBURGH, PA**

**RRV-1 (Remote Reconnaissance Vehicle)** - This device is a tethered, remotely controlled 6-wheeled vehicle which has been designed to operate in the Three Mile Island Unit 2 (TMI-2) nuclear power plant. The power supply (110v, 60 Hz), the video/sensory data (radiation level monitoring), and the command instructions are transmitted to and from the vehicle by a 400 m long cable from the control console. The cable retractor located on-board the RRV is remotely operated by one of the two teleoperators. Two of the three on-board video cameras are used for navigation for the vehicle and the third is used for general surveillance and to monitor the cable retraction. The dimensions of the Standard Manufacturing Co. supplied base are 50 x 29 x 19 (lwxh) inches. The total height and the approximate weight of the vehicle, which includes a lift cage, is 5 ft and 1000 lbs, respectively. The maximum speed of the vehicle is 44 ft/min. The device can travel through 12 inch deep water and it has been designed for quick wash-downs and decontamination procedures.

#### **CYBERMATION, INC., ROANOKE, VA**

**KLUGE** - This teleoperated, three-wheeled, tetherless device is designed to act as a mobile material and tool transporter in factories and other industrial facilities. Its three extendible wheels and coupled working platform are omnidirectional, thereby enabling the device to rotate on its own axis (zero turning radius). The working platform has a 20 in. height and a 33.5 in. width; it can support a load of 250 lbs. The device can climb a 30 degree slope. Sensory information available on the KLUGE includes video and two audio communications. Power to the KLUGE is supplied by on-board batteries.

#### **GCA/PAR CORPORATION. ST. PAUL, MN**

**HERMAN** - This device is a remotely controlled, tethered, 2-tracked vehicle which is used in the nuclear industry. HERMAN is located at the Oak Ridge Y-12 plant, operated by Martin Marietta for the U.S. Department of Energy, since 1966. The vehicle carries two video cameras, along with their individual pan-and-tilt mechanisms and support platforms, and is designed to operate at distances up to 700 ft. from the control console. The manipulator arm can lift 160 lbs. and can drag up to 500 lbs. The vehicle dimensions are (with the manipulator arm in a stowed position): 45 x 29 x 62 (lwxh) inches. This device is expected to be replaced in 1986-1987 timeframe.

#### **ODETICS, INC., ANAHEIM, CA**

**ODEX-1** - This tetherless, remotely controlled 6-legged device was designed to serve as a proof-of-concept mechanism for other similar devices to be employed in a variety of industries. The device is not yet commercially available but illustrates substantial mobility and high strength to weight ratio.

#### **VIKING ENERGY COMPANY, PITTSBURGH, PA**

**ROD** - This tethered, remotely controlled, 6-wheeled device was designed to assist in the decommissioning of nuclear facilities. The on-board industrial robotic arm supplied by the American Robot Company is able to lift up to 50 lbs, depending upon the model of the arm. The working envelope of the arm is 40 or 60 in. depending upon the arm model. Various end-of-arm tooling can permit ROD to conduct specific missions and tasks, such as sand-blasting, pipe cutting, and vacuuming. In addition to being able to clear 4 inch high obstacles, ROD can climb slight inclines and pull up to 1300 lbs on a dry pavement. Both the tether and the wheels are capable of being decontaminated by spray cleaning or wiping. Video signals, radiation monitor readings, and power supply are transmitted along a 200 ft. long cable. This is a commercial offering.

#### **REMOTEC, OAK RIDGE, TN**

**SURBOT (Surveillance Robot)** - This device is under development as a tethered teleoperated surveillance robot for the nuclear industry. This three (3) wheeled vehicle has a master-slave manipulator capable of lifting 25 lbs. and is supplied with video, sound, temperature, and relative humidity sensors. This device is capable of climbing 15° slope, traveling through 3 inch deep water and can turn on its radius. Commercial-ization is planned for early to mid 1986 after a test program.

## **NINE MILE POINT ROBOTIC PROGRAM**

Many other devices were examined during the two month data gathering period and information about other devices is still being sought and reviewed. After sufficient information was gathered concerning the present state of technology, a plan to implement a robotics program at NMP-1 was formulated in December 1984.

It was recognized early in the program development that the critical element necessary for a successful remote technologies program was the input and support of station operating personnel. Therefore, a task force jointly sponsored by the site ALARA organization and Corporate R&D and consisting of interested representatives from Maintenance, Radwaste Operation, Inservice Inspection, Instrument and Control, Technical Support and Nuclear Engineering was formalized. The purpose of the task force was to analyze station operations to identify beneficial robotic applications. The first task force meeting was held in January 1985. During the next two months task force members reviewed the information and vendor supplied video tapes. A brain storming session was held to list the possible applications and a prioritization based upon the available technology and practicality was performed.

In March 1985, a survey form was distributed to solicit ideas and recommendations from other members of the operating staff. The response to the survey, like any other survey, was not overwhelming, however the ones returned were very good and proved an interest from people not otherwise considered.

### **Dolphin-Nuclear Test Program**

The first robotic endeavor was the test of Automation Technology Corporation's (ATC) Dolphin-Nuclear robot. Dolphin-Nuclear is a modified version of an underwater tethered device designed to scrub and vacuum swimming pools. This device was tested at Nine Mile Point Unit 1 in April 1985.

ATC proposed to conduct the activities for this evaluation project under the sponsorship of two nuclear utilities. Niagara Mohawk Power Corporation (NMPC) and Pennsylvania Power and Light Company (PP&L) funded and actively participated in this program. NMPC Research and Development offered the use of the Nine Mile Point Unit 1 Nuclear Station. PP&L Nuclear Support provided technical support and test personnel during the testing and report phases.

The project was based upon the current off-the-shelf model of Dolphin which was designed to function cost effectively in a commercial swimming pool environment. The participants recognized the limitations of Dolphin-Nuclear, but they believed this device could serve as "proof-of-concept" for a remote underwater cleaning device. The goals of this project were to identify the decontamination applications, perform a field test, and, if successful, identify the necessary modifications for next-generation devices.

These specific goals were to be attained by completing the following tasks:

- 1) Select a test/evaluation facility with project participants.
- 2) Identify applications for evaluation programs - to include use of video/charged coupled device (CCD) and radiation monitoring surveillance devices.
- 3) Conduct an evaluation program using an unmodified unit, including examination of operation, maintenance and decontamination procedures.
- 4) Identify and define system design requirements.
- 5) Prepare and issue final report.

Dolphin-Nuclear is an underwater device designed to scrub and vacuum the horizontal and vertical surfaces of pools, tanks, cavities and sumps. The equipment has autonomous operation with the power supplied via a tether. Dolphin-Nuclear was lowered into the NMP-1 Waste Surge Tank as a "test of concept" that a modified design of the Dolphin swimming pool cleaner could be adapted to nuclear applications. Dolphin-Nuclear was easily lowered through the man-way and submerged. When the power was turned on, Dolphin-Nuclear could be seen swimming toward the side of the tank, climb the wall, descend the wall and then become lost in a pile of sludge. Dolphin-Nuclear has successfully demonstrated "proof-of-concept." During the June-July 1985 period, Dolphin-Nuclear was tested for ease of decontamination. Several decontamination methods including manual scrubbing, ultrasonics, and freon were evaluated. The results of the tests provided the data necessary for an improved design. The data gathered through observations, photographs, and video is being used to design a nuclear production grade Dolphin-Nuclear called SCAVENGER which is shown in Figure 1.

SCAVENGER is equipped with two brushes that scrub the surface debris beneath the unit where an internal pump provides suction to draw up the debris for filtration. The internal filter bag collects the debris



Figure 1

ATC Scavenger - An underwater  
decontamination robotic device

for disposal. The positive pressure created by the discharged filtrate maintains SCAVENGER contact with the floor and walls.

The goals identified for the Dolphin-Nuclear were fully met as follows:

1. Select a test/evaluation facility with project participants.

Response: The NMPC Nine Mile Point Unit 1 served as the test/evaluation facility.

2. Identify applications for evaluation programs -- to include use of video/charged coupled device (CCD) and radiation monitoring surveillance devices.

Response: In the course of this project, the participants identified and discussed various potential applications including:

- Tank and sump decontamination
- Tank and sump maintenance
- Other applications include the need for radiation mapping, video inspection, autonomous operations, obstacle avoidance, and potential for untethered operations.

The potential for an entire family of devices appears logical, especially when waste loading is considered.

3. Conduct an evaluation program on an unmodified unit, including examination of operation, maintenance and decontamination procedures.

Response: The evaluation of the unmodified Dolphin-Nuclear underwater robotic device was concluded to the satisfaction of all participants. The device demonstrated that it can be readily adapted to the nuclear power plant environment. Minimal or no extraordinary preparation or precautions were required for Dolphin-Nuclear to perform its intended function. Further operational assessment indicates that the unit is easily adaptable to any tank configuration with a standard manway opening. Unit operation by untrained personnel demonstrated ease of operation. The decontamination methods used demonstrated Dolphin-Nuclear durability.

Maintenance of Dolphin-Nuclear was not required during the evaluation project, however, disassembly for decontamination demonstrated acceptability. Full protective clothing, including two sets of rubber gloves was required for decontamination. Disassembly under these conditions was readily performed by untrained individuals with average mechanical abilities.

Decontamination of Dolphin-Nuclear proved to be difficult; however, the requirements were not outside the abilities of an operating nuclear power facility. Since Dolphin-Nuclear was not modified to suit nuclear applications, complete decontamination was not expected.

4. Identify and define system design requirements.

Response: A series of observations were developed which clearly identify the experience and design features which are desirable for an underwater robotic cleaning device.

NMP-1 continues to test different configurations of Dolphin-Nuclear in the hopes of developing an autonomous device that will replace the need for emptying process tanks and using men for cleaning these radioactive tanks.

An observation made during the initial Dolphin-Nuclear testing was that the bag filter filled quickly under normal waste tank conditions. A recommendation was made to modify Dolphin-Nuclear so that the mobile system could carry the suction end of a hose, connected to an external diaphragm pump. This configuration essentially rendered Dolphin-Nuclear a tank bottom desludger.

A simple modification to Dolphin-Nuclear was made. An aluminum 2.5" flange was mounted to the pump-impeller discharge located on top. The flange was threaded to accept a 1" male quick disconnect fitting. Tests to the modified Dolphin-Nuclear are planned for early 1986. The preliminary plan is to connect the male fitting to a 1" hose connected to a diaphragm pump. Dolphin-Nuclear would be submerged under water and tested to see what affects, if any, there were on the mobile system.

#### RADWASTE APPLICATION PROGRAM

A major radwaste retrofit project at NMP-1 was initiated in late 1985. As a result, the project was examined to see if any robotic applications would be of benefit. In June 1985, NMPC reviewed with DOE personnel the potential for the use of HERMAN for several decontamination sequences. HERMAN was the first choice for this application because of its proven history. This option proved impractical as discussed below.

During July, Inservice Inspection and Radwaste Operation personnel were consulted about possible applications of a surveillance robot in their area of responsibility. Applications were determined and test criteria were established.

The interest in robotic applications has not decreased at NMP-1 and applications are continually being suggested. The application list and interest was high enough that plans were made in August 1985 to lease a robotic device for trial use. The purpose of the lease was to perform the previously identified tasks and also to allow NMPC personnel to gain hands-on experience with a mobile robot. The choice of technology was based upon a detailed analysis of the performance requirement for a robotic vehicle system to support the radwaste system refurbishment project.

Based upon surveys and extensive plant support, performance requirements for a robotic vehicle system were developed. These requirements reflect, to the extent practical, the minimum requirements identified to support the NMP-1 refurbishment plan and reflects the best available information on the expected conditions to be encountered by a remote vehicle as well as any restraint imposed by the current operations, procedures and practices at Nine Mile Point Unit 1.

The performance requirements are as follows:

1. A mobile autonomous robotic vehicle is required.
2. The vehicle must be capable of descending and ascending stairs at Nine Mile Point Unit 1.
3. Remote control, video and sensor data transmissions on a non-interfering communication frequency is required.
4. The vehicle must be capable of climbing over obstacles including a 7 inch dam.
5. The vehicle must be capable of traveling through a 6 inch depth of water, and water and sludge mixtures.
6. The vehicle must be capable of traveling across concrete flooring, fiberglass and steel grating, wood, grass, gravel, soil, mud and sludge.
7. Obstacle avoidance either autonomously or via teleoperations is required.
8. The robotic vehicle must be capable of entering labyrinths as encountered at Nine Mile Point.
9. The robotic vehicle must be able to enter and exit various cubicles at Nine Mile Point including the ability to turn on its axis.
10. The robotic system must be capable of remote video recording, including provision of necessary lighting.
11. The robotic vehicle must be capable of either tethered or untethered operations.
12. The vehicle must be capable of radiation mapping including 3-degrees of freedom.
13. Wide angle telephoto vision is required with video taping capability.
14. Inspection optics with 3-degrees of freedom is required.
15. Supplemental optics such as stereovision are desirable, including the capability to confirm equipment and component locations.
16. 360 degrees video coverage in the horizontal plane is required; 270 degrees or greater video coverage in the vertical plane is desirable.
17. The robotic system must be capable of monitoring and surveillance of activities at all areas within NMP-1.
18. It is highly desirable that the vehicle be capable of moving cylindrical non-stationary components such as overturned 300 lb. drums.
19. It is highly desirable that a liquid, solid and possibly gaseous sampling capability be provided.
20. It is desirable that the robotic vehicle be capable of taking surface smear samples.
21. It is highly desirable that the robotic vehicle have a tool capability to bore, puncture and sample waste drums.
22. The robotic vehicle and any ancillary equipment must be capable of being decontaminated.
23. The robotic vehicle must be capable of being recovered in case of various single component failures.
24. It is desirable that the vehicle be capable of transporting a pumping/transfer station (vehicle) to various areas.
25. It is desirable that the robotic vehicle be capable of transporting and utilizing a hydrolaser to support facility decontamination.
26. It is desirable that the vehicle be multiple use in nature.
27. It is desirable that the device should require minimal modifications of NMP-1 facilities.
28. The device should be capable of continuous performance when subjected to 130°F, 100% relative humidity and 100 R/hr radiation environment.
29. The device should be capable of movement between plant areas without releasing contamination into hallways, etc.

30. The device should be capable of reliability equivalent to industrial robot system with 300 hours meantime between failure (MTBF).

Nine robotic devices were described above which were considered, to varying degrees for use at Nine Mile Point. It should be noted that other devices were reviewed but either were not available for loan, lease or procurement or were totally inappropriate for consideration at NMP-1.

Table 1 was prepared to evaluate the final candidate robotic vehicles against the NMP-1 performance requirements. Based upon this evaluation, the Automated Technology Corporation SURVEYOR™ was recommended for use at Nine Mile Point for continuation of the robotic and remote technology R&D project.

During fall 1985 and early 1986, NMP-1 will continue developing the robotics application program with projects scheduled for Dolphin-Nuclear and SURVEYOR™.

The next long term robotic project at NMP-1 will be the test of the SURVEYOR™ robot in a real work environment. As described earlier, SURVEYOR™, shown in Figure 2 is a teleoperated, tetherless device designed for nuclear power plant applications. Surveyor will be used in conjunction with a major radwaste retrofit project where it will have to overcome the obstacle inherent to any construction site and transverse stairs. The SURVEYOR™ project and test period is expected to last approximately six months.

## CONCLUSION

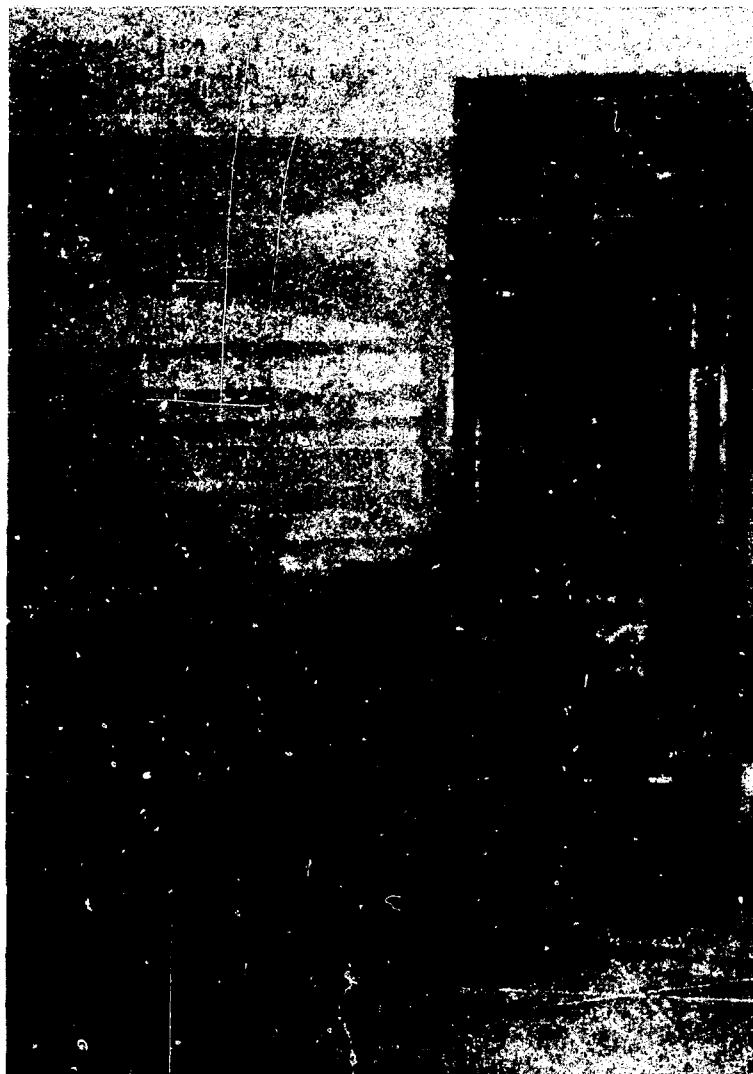
In conclusion, some key aspects of the NMP-1 robotics program are as follows:

1. Plant advocates are needed. In order to support any robotics project, one must have the support of the particular work group for which the robot is intended to assist. Without this support, the project may be burdened with a negative attitude, and the credibility for other projects will be damaged.
2. Planning is critical for robotic test and applications. In order to maintain a positive attitude among supporters, the robot test must be carefully planned. Do not let expectations exceed the robot's capabilities. Plan the work and work the plan.
3. Small successes are important. People who are generally skeptical of an emerging technology will keep a success/failure score card. Do not overlook even the smallest objective and its successful achievement.
4. Technology is in its infancy. Maintaining a positive attitude among advocates is extremely important, and the attitude to keep is that the technology is new, it has a great potential for improving station operation, and there is a lot yet to be learned.
5. Most importantly, a dedicated project manager is critical to the success of a robotic application. I don't need to tell anyone here that we are in a hectic business and keeping our daily schedule full is the easiest job we have. Plan the robotic test projects so that one individual is dedicated to completing the task. Delays create confusion, loss of objectivity and poor data acquisition.

Table 1

## NMPC Robotic Vehicle Evaluation

Performance Requirement	Surveyor™	Device MF-3	Herman	Surbot
1. Mobile autonomous vehicle	Y	Y	Y	Y
2. Climb/descend stairs	Y	Y	N	N
3. Remote control	Y	N(tethered)	N(tethered)	N(tethered)
4. Climb 7 inch dam	Y	Y	N	N
5. Travel 6 inch				
- Water	Y	Y	Maybe	N
- Water and sludge	Probably	Y	Maybe	N
6. Travel multi-floorings	Y	Y	Y	Maybe
7. Obstacle avoidance	Y	Y	Y	Y
8. Labyrinth travel	Y	Probably	N	Y
9. Turn on axis	Y	N	N	Y
10. Videorecording	Y	Probably	Y	Y
11. Tethered and/or untethered	Y	N(untethered)	N	N
12. Radiation Mapping	Y	Probably	Probably	Y
13. Telephoto Vision	Y	Y	Y	Y
14. Inspection optics	Y	N	N	Y
15. Stereo optics	Y	N	N	N
16. Vision planes	Y	Maybe	N	Y
17. Surveillance	Y	Y	N	N
18. Move drum	Maybe	Probably	Y	N
19. Sampling	Y	Maybe	Y	Maybe
20. Smear samples	Y	N	Maybe	Maybe
21. Tool capability	Probably	Maybe	Probably	N
22. Decontamination	Y	Y	Y	Y
23. Recovery	Probably	Unknown	Y	Maybe
24. Transporter	Y	Y	Y	N
25. Hydrolaser	Y	Y	Y	N
26. Multiuse	Y	Y	Y	Y
27. Minimal plant mods	Y	Y	N	N
28. Environmental conditions	Y	Probably	Y	Y
29. Contamination control	Probably	Maybe	N	Y
30. Reliability	Y	Unknown	Probably	Unknown
Present Purchase Cost	\$200,000	\$300,000- \$600,000	Not Applicable	\$175,000- \$200,000
Lease/Rent	Y	N	Y	N
Availability	60-90/Days	Unknown	Negotiated	Unknown (1986)
Other Factors	EPRI tested NDE Center and Catawba	Extensive European experience	Extensive DOE experience	EPRI tested and to be tested at Browns Ferry



**Figure 2**  
**ATC Surveyor™ - A remotely  
operated mobile surveillance system**

THE DECONTAMINATION AND DECOMMISSIONING  
OF EXTRACTION CELL 3 AT THE  
WEST VALLEY DEMONSTRATION PROJECT

E. D. JONES  
WEST VALLEY NUCLEAR SERVICES CO., INC.  
P.O. BOX 191  
WEST VALLEY, NEW YORK 14171

ABSTRACT

This paper describes the decontamination and decommissioning (D/D) of Extraction Cell 3 (XC-3) at the West Valley Demonstration Project. XC-3 is one of several cells in the former reprocessing plant required for future use in support of the solidification of high-level waste. The cell became radioactively contaminated during nuclear fuel reprocessing from 1966 to 1972. This report describes the work performed to accomplish the D/D objectives of removing existing piping and equipment from XC-3 and of reducing radiation and contamination levels to allow installation of equipment for the Liquid Waste Treatment System (LWTS). Contaminated debris and equipment inside the cell were removed, packaged and stored for future disposition. Interior surfaces (walls, floor, and ceiling) of the cell were then decontaminated to a radiation level that allowed entry without the use of protective clothing or respiratory protection.

INTRODUCTION

The West Valley Nuclear Services Company, Inc., a subsidiary of the Westinghouse Electric Corporation has been contracted by the U. S. Department of Energy (DOE) to perform a High-Level Radioactive Waste Management Demonstration Project at West Valley, New York.

Approximately 560,000 gallons of liquid high level waste, currently stored underground in a steel tank, will be solidified in a glass form. In support of this work several shielded cells in the existing facility were identified for use. Decontamination and decommissioning (D/D) programs are being implemented to prepare these cells for the installation of new systems. This paper describes the D/D work which has been completed on one such cell, Extraction Cell 3 (XC-3).

PHYSICAL DESCRIPTION

The cell measures 15 feet x 21 feet and is 60-feet high. Entry is via a personnel door at ground level and a 6-foot square access hatch through the ceiling. The floor has a stainless steel lining extending 18 in. up each wall. Concrete walls and ceiling are coated with a phenolic paint. The cell contains no remote handling equipment and there are no windows.

Piping, Vessels, Columns, and Support Structures

Table 1 shows a list of equipment that was removed from the cell during the D/D campaign.

Table 1  
List of Equipment Removed from XC-3

<u>Equipment</u>	<u>Size</u>	<u>Quantity</u>	<u>Weight (lbs.)</u>
Piping	1/8" - 6" dia.	7,332 linear ft.	11,800
Tanks	6" - 5' dia.	13	17,890
Columns	16" dia. x 40' long	8	14,770
Box Beams	Various up to 12" sq.	440 linear ft.	8,760
Platforms	Various	187 sq. ft.	1,100
		Total Weight	54,320

#### RADIOLOGICAL CONDITIONS

Initial radiation surveys indicated that the floor, which was covered with a mud like debris, was the primary radiation source with readings as high as 150 mR/hr. Radiation levels in higher elevations of the cell averaged 5 mR/hr.

After final decontamination and painting, smearable (loose) contamination was generally less than 50 dpm alpha and less than 500 dpm beta per 100 cm<sup>2</sup> while whole-body radiation levels were less than 1.0 mR/hr.

#### FISSILE MATERIAL IN VESSELS AND PIPING SYSTEMS

A method was devised to test each vessel and system to determine the presence of liquid containing fissile material. Each vessel and low point of a piping system was drilled and if liquid was detected a sample was collected and analysed. Based on the results of the analyses, it was determined that the fissile concentrations were relatively low (less than  $1 \times 10^{-2}$  grams per liter). Consequently the total amount, 300 liters of process liquid, was drained to the sump for eventual transfer to a liquid waste storage vessel.

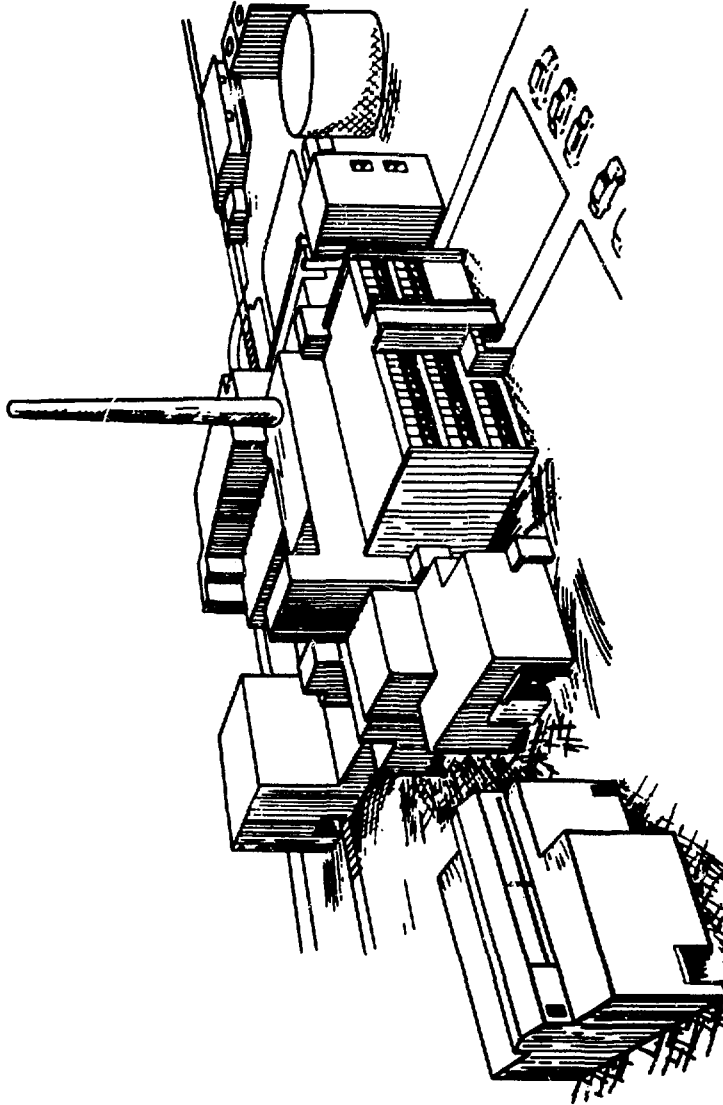
After all of the residual liquid was drained, sample coupons (from vessels) and rings (from pipes) were cut and analysed. Based on the sampling plan, it was shown that the total fissile material found in all vessels and piping, was less than 30 grams. It was further determined that, based on plutonium mass and weight of components, each vessel and all piping could be handled as non-TRU contaminated material (see footnote).

#### CLEANING THE FLOOR OF DEBRIS AND TRASH

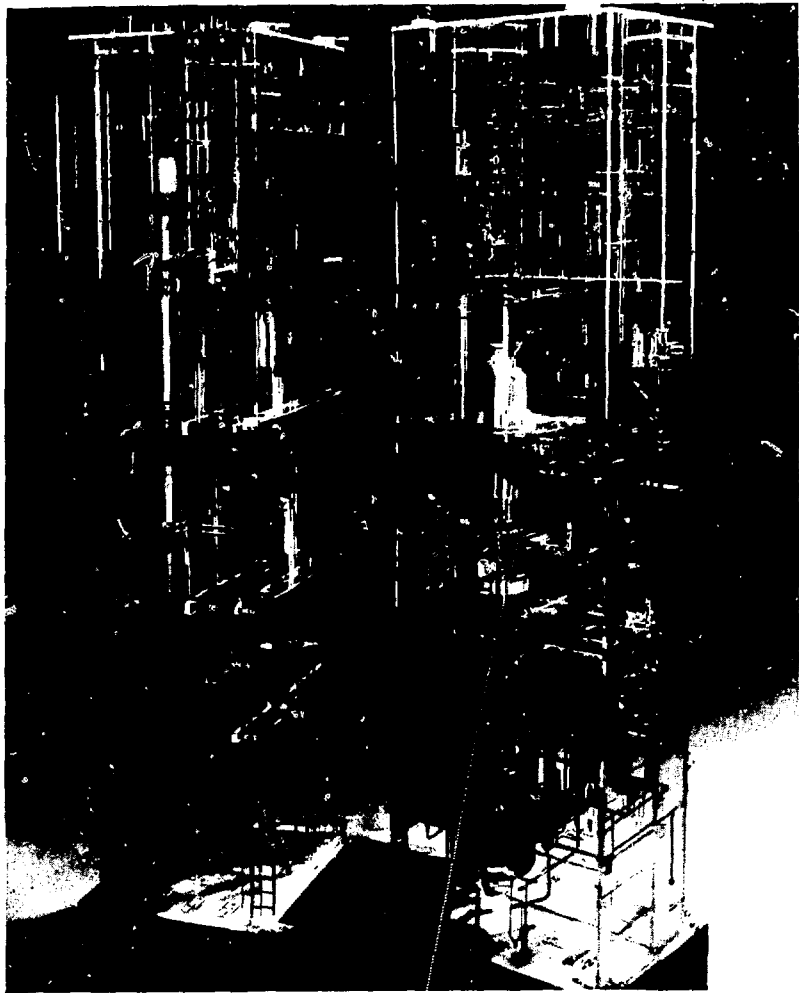
An inspection of the cell floor indicated that it was littered with discarded anti-C clothing, air lines and tools from prior entries. More significantly though it was noted that a mud-like debris covered the stainless steel lined floor. Radiological surveys indicated an average exposure rate of 5 mR/hr with hot spots to 150 mR/hr.

---

\*Footnote: Transuranic (TRU) contaminated material is material that without regard to source or form, is contaminated with alpha-emitting radionuclides of atomic number greater than 92 and half lives greater than 20 years, in concentrations greater than 100 nCi/gram.



**FIGURE 1.**  
**West Valley Demonstration Project Process Building**



**FIGURE 2.**  
**Scale Model of XC-3**

The following tasks were performed to clean up the floor area:

- o Sampling debris to determine fissile content
- o Collecting debris and trash from the floor
- o Solidifying the debris in cement
- o Removing the solidified debris and containers of trash from the cell
- o Flushing and wiping the floor, lower walls, accessible piping and vessels.
- o Sampling and analyzing the sump contents (to assure low fissile material inventory)
- o Transferring liquid to a waste storage tank via the sump.

Samples of debris were collected using a vacuum cleaner with a long handled probe. Analysis of the samples indicated estimated mass levels of 104 grams Pu, 2 grams Am-241 and 40 grams U235. The mud-like debris was collected, again by vacuuming, into 5 gallon containers. The containers were modified so that fissile mass was limited to 25 grams per container. Once in the containers and prior to removal from the cell the debris was solidified with cement and water.

The final radiological survey indicated an average whole-body exposure rate of less than 1 mR/hr with hot spots to 5 mR/hr around the sump.

#### SUMMARY OF OPERATIONS

Initially entries were made at floor level to perform surveys and to clean up the floor area. Almost all subsequent operations were performed by lowering workers into the cell, through the 6 feet square hatchway in the cell ceiling. The room above the cell, the Extraction Chemical Room, (XCR) was used as a work staging area after it had been cleared of its vessels and piping. A large containment tent was erected surrounding the hatchway to prevent the spread of contamination. Lifting equipment was staged inside the containment. First the hatch covers were removed after which piping, tanks, extraction columns and support structure followed. After the cell was cleared of all equipment, the walls, floor and ceiling were decontaminated and painted. The cell was then ready for the installation of new equipment in support of the Low Level Waste Treatment System (LWTS).

#### WORK CREW

A total of nine D/D operations personnel were required for each cell entry. This number does not include Radiological Safety support personnel. The duties of the nine operators were as follows:

- |                                      |             |
|--------------------------------------|-------------|
| 2 - In cell workers                  |             |
| 2 - Life line tenders                |             |
| 1 - Air hoses and power cable tender |             |
| 1 - Staging person                   |             |
| 1 - Communications and record Keeper | Outside     |
| 2 - Suit-up and equipment expeditors | Containment |

The average time each operator spent in-cell, per 8 hr. shift was approximately 2 hours and was generally limited by fatigue. Some significant causes of the fatigue problem were the extensive preparation time, the bulkiness of the anti-C clothing, manual requirements of the tasks in-cell and the cell ambient temperature, generally > 80°F.

## IN-CELL RADIOLOGICAL SAFETY MEASURES

Respiratory protection and anti-C clothing were routinely required for all in-cell workers. Continuous air monitoring and personnel surveys were also mandated to protect the workers.

### Respiratory Protection

The primary respirator used was a positive pressure face piece device. In addition and worn over the primary respirator, in-cell workers wore a positive pressure supplied air hood, in combination with a vinyl suit. A back-up air supply, to supplement the plant system, was available. The air supply system was designed to automatically transfer to bottled air if the plant compressor failed. An alarm system alerted support personnel who were in voice communication with workers in the cell at all times. Back-up air provided four workers with emergency air for ten minutes; sufficient to effect an orderly evacuation of the work area, if necessary.

### Anti-Contamination Clothing

The anti-C clothing used by the workers in the cell included the following:

#### Coveralls

Two pairs shoe covers

Plastic bags taped over shoe covers

Paper suit

Second plastic bag taped over feet

Second pair coveralls

Cotton (or skin) gloves

Anti-C gloves taped to coveralls

Second pair Anti-C gloves taped to outer coveralls

Vinyl suit

Airhood

Cloth cap

Third pair coveralls

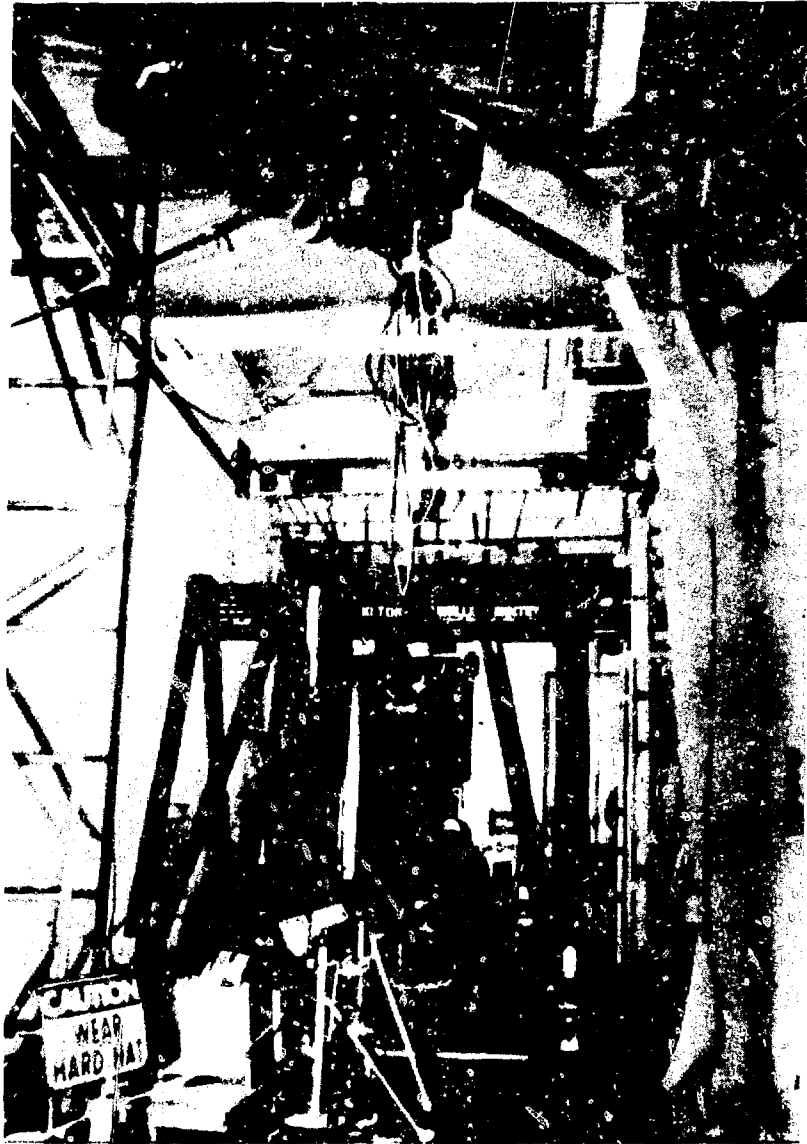
Rubber overshoes

## MATERIALS HANDLING EQUIPMENT

To facilitate the removal of equipment from the cell, through the top access hatch, a five-ton capacity jib crane and hoist was installed adjacent to the hatch opening. Consideration was given to the available head room from the XCR floor to containment roof and the height of the largest tank, in the selection of a low-head hoist. To supplement the jib crane, a three-ton capacity chainfall attached to a tripod, was mounted above core drilled holes in the cell ceiling. The holes were strategically located relative to vessel and equipment pick points. Once in the XCR, an air pallet (four-tons capacity) moved the wrapped vessels and waste containers through the containment to a monorail mounted hoist. The monorail transported the load through large doors to the southwest corner of the plant building where it was lowered to ground level and transported to interim storage.

## SPECIAL TOOLING

The rationale behind tool selection was to provide tooling that was



**FIGURE 3.**  
**Jib Crane in XCR**

effective, low cost, and readily available. To satisfy these requirements, basic hand power tools were selected for the major portion of the pipe and structural box beam cutting, and vessel sampling. Frequently, the standard power tool was modified for remote or extended reach work.

Portable power band saws were modified with extensions to cut pipe and box beams below the worker platform.

Hydraulic pipe shears were modified to operate from a suspension cable lowered through the roof hatch and controlled from the floor area over the cell.

Hydraulic-driven milling cutters were used to reduce the length of columns prior to packaging for later disposal. Reciprocating hack saws were used to cut box beams and pipe that were too large for the hand-held band saws. Standard portable electric drills were used to cut plug samples from vessels for assay specimens and to drill holes in pipe for liquid sampling.

## CELL ENTRY

### Description of Hatch Opening and Covers

The roof of the XC-3 cell is 3 feet thick. The hatch opening is 7 feet square at the top tapering to a 6 feet square opening into the cell.

The hatch cover is made up of three layers. Each layer is a steel frame filled with reinforced concrete. The top and middle sections are 9 inches thick and each weighs 6,000 lbs.; the lower section is 18 inches thick and weighs 10,000 lbs.

### Work Platform

To gain access to all elevations of the cell, through the 6 ft. square hatch in the ceiling, a two-man work platform was used. The work platform was suspended from an overhead gantry beam and was controlled by the operator. However, an additional control unit was added so that, in case of an emergency, the platform could be operated from outside the cell. Each cell worker wore a parachute type full body harness which was attached to a winch operated life line.

### Removal of Piping

Pipe was removed from XC-3 by two methods. A description of both methods follows:

The first entries for pipe removal were made at floor level from the Cell Access Aisle (CAA). Pipe was removed up to approximately eight feet above the floor. Portable band saws were used to cut out long lengths of pipe, and an hydraulic shear was used to reduce the cut pipe into lengths to fit into a 55-gallon drum. Initially, cut pipe was sleeved with plastic prior to placing in the drums. It was later determined that fixative sprayed onto the pipes was satisfactory to reduce airborne contamination and sleeving was discontinued. Full drums were smear surveyed as they were taken from the tent. TRU waste drums (white) were used. Analyses of pipe samples however, indicate that the pipe will be found to be non-TRU, by neutron interrogation analyses.

After completion of all work that could be performed at floor level, entries were made from the top of the cell through the open hatch utilizing the suspended work platform discussed previously.

Long tray-like disposable boxes, into which the cut pipe was laid, were used to remove pipe from the cell. The pipe boxes were lowered into the cell and placed on a work platform. Cut pipe was then placed in the boxes and covered with plastic sheet. When full, the pipe boxes were lifted up and out of the cell and placed in a large waste container. Eight such boxes fit into one waste box and were designed to carry 500 pounds of pipe each. A detachable four point spreader beam was used for lifting the boxes.

#### Removal of Support Beams and Platforms

Structural tubes, rectangular in cross-section, used to support tanks, columns, piping, and personnel platforms were removed in parallel with the removal of vessels. Most of these support beams were oriented in the east-west direction and spanned the cell. The beams were bolted to inserts that were cast into the concrete walls. Permanent expanded metal platforms as well as temporary platforms made by laying aluminum planks on existing support beams, were used as work areas during the decommissioning of the cell. As the equipment was removed and the support beams and platforms were no longer needed, they were cut into manageable lengths and removed, to provide access to lower regions of the cell.

#### Removal of Vessels

A total of 21 vessels (tanks and columns) were removed from XC-3. Table 2 provides a characterization of the vessels with diameters exceeding six-inch diameter. Since none of the vessels had lifting eyes or trunnions, each of the larger vessels had to be fitted with a lifting bail or clamp to remove it from the cell. For the smaller diameter vessels, a sling in a choker hitch located below a protruding pipe stub or mounting bracket, was used for lifting.

Table 2  
XC-3 Vessel Characterization

Equipment Number	Elevation (ft)	Overall Height	O.D.(s)	Comments
4D-12	127	12'	3'7"	Contains 4,000 lb. boron glass rings
13D-6	104	9'8"		5'
13D-3	118	7'9"		4'
5D-2	112	6'8"	3'1"	Pu tank, contain 2,500 lb. boron stainless steel ring
13D-19	122	6'8"		2'2"
5D-1	127	4'10"	4"	Slab tank for Pu solutions
4Y-11	122	4'9"	2'	Contains 15 boron stainless steel plates
4Y-12	136	4'9"	1'6"	Contains 11 boron stainless steel plates
4C-11	102	4'3"	1'10", 11", 2'	
4C-10	109	4'10"	1'10", 11", 1'6"	
4C-12	109	4'10"	1'10", 11", 1'6"	
4C-12A	106	3'3"	9"	
13C-3	102	28'6"	11"	Contains 800 lbs. stainless steel rings
4E-5	112	17'8"	9"	
13C-7	114	15'6"	1'4", 8"	
13C-6	102	15'4"	1'8", 11"	
4Y-22	105	10"	8"	
4Y-28	149	4'	7"	
13Y-9	143	2'6"	1'4"	
4Y-57	137	2'6"	6"	
13E-1	147	2'	1'6"	

Before removing a vessel, all pipes to the vessel were drained, the pipes were then removed and stub ends capped with heat shrinkable plastic caps. The vessel was then sprayed with a fixative to reduce the potential for airborne contamination. The jib crane hook was lowered through the sleeve and attached to the vessel rigging. If the vessel center was not under the hatch opening, it was necessary to use an auxiliary hoist along with the jib crane hoist. A core drilled hole in the XC-3 ceiling directly over the vessel was used for the auxiliary hoist to access the vessel. One or more core drilled specimens were then removed from the vessel wall for analysis of fissile material. The vessel was rigged to the monorail hoist in a horizontal position and lowered into a waste container at ground level and transported to the Lag Storage Building.

Removing the chemical process columns presented a greater challenge than the other equipment in XC-3 in that their lengths exceeded the head room and handling room available in the containment.

There were seven chemical process columns and a heat exchanger which was handled as a column because of its length. The three largest columns will be described in this section. These columns were nearly identical in size and were located near the south wall of the cell. The longest column was 43-feet 3-inches long including pipe stubs at top and bottom. The column diameter was 10-inch nominal pipe and had internal perforated plate decks through its entire length. Both top and bottom ends of the columns had disengaging heads with the largest outside diameter being 22 inches. All columns were supported at the their base and had "U" bolts holding them to wall brackets at several elevations.

The columns were sprayed with fixative to reduce airborne contamination. Each column was rigged for lifting with two slings in choker hitch. The U bolts supporting the columns and the stud bolts at the base were then removed.

A work platform was placed across the hatch opening, designed to support the column as it was being cut into manageable lengths of approximately 7 feet. Circumferential cuts were made through the column wall using a cutter designed to automatically travel around the work piece.

A total of five cuts were made on each of the largest columns. All the cuts were located at girth welds, and this resulted in one-piece cut-off sections (i.e., no internal plate decks were cut free to be handled as separate pieces).

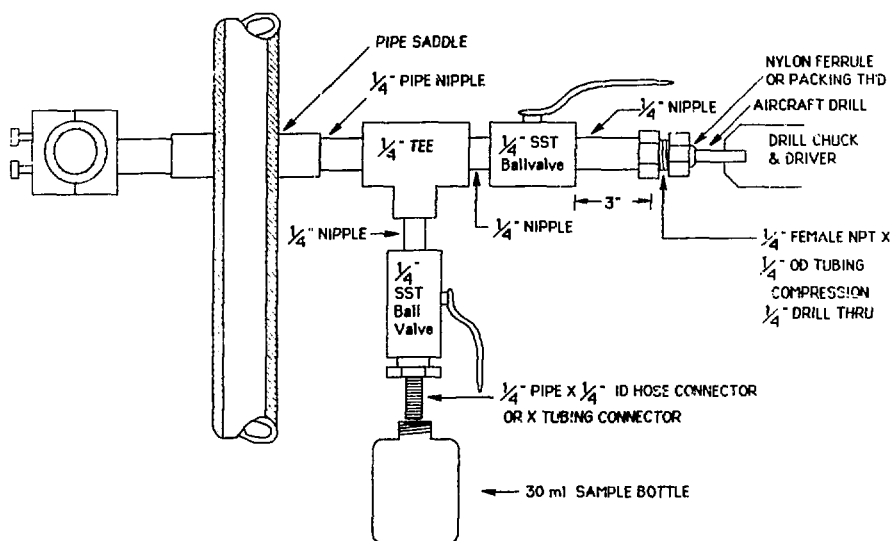
Two of the seven columns contained loose stainless steel raschig rings at preferred cut locations. To prevent rings from being released, insulating foam was injected through drilled holes in the column wall in the area to be cut. After cutting the column wall, the foam retained the rings in the cut-off top section. The heat exchanger, which has 122 tubes inside a 8-inch nominal pipe section was also foamed and cut as described previously however, the internal tubes had to be cut with a reciprocating saw with longer reach than the circumferential cutter. The cut lengths of all the columns were then handled in the same manner as described for the tanks.

#### WASTE CONTAINERS

Waste containers were procured which complied with U.S. Department of Energy requirements for Low Level Waste (LLW) and Transuranic (TRU) waste.



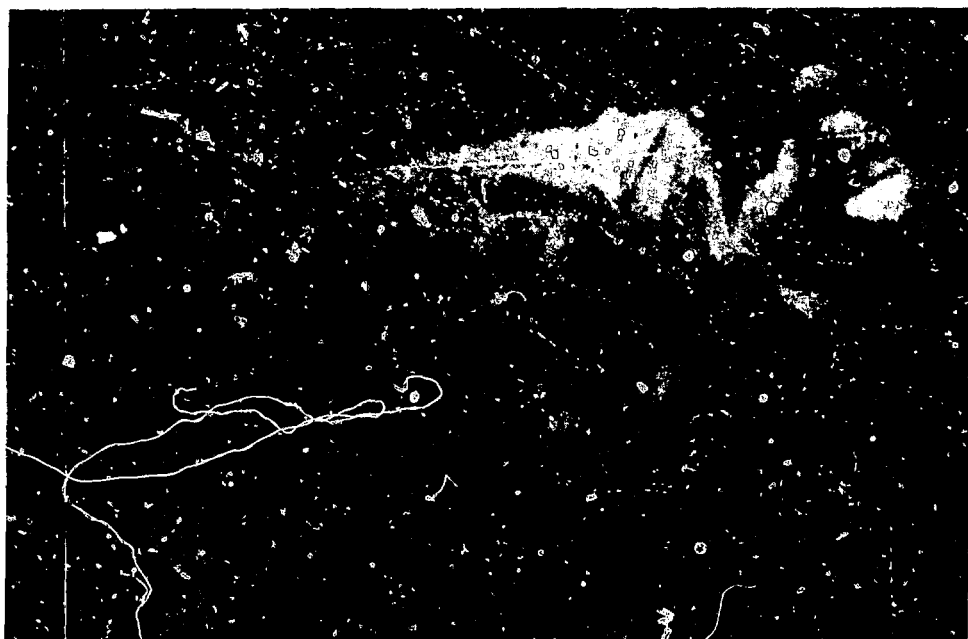
**FIGURE 4.**  
**Hatch Cover Wrapped and Rigged for Removal**



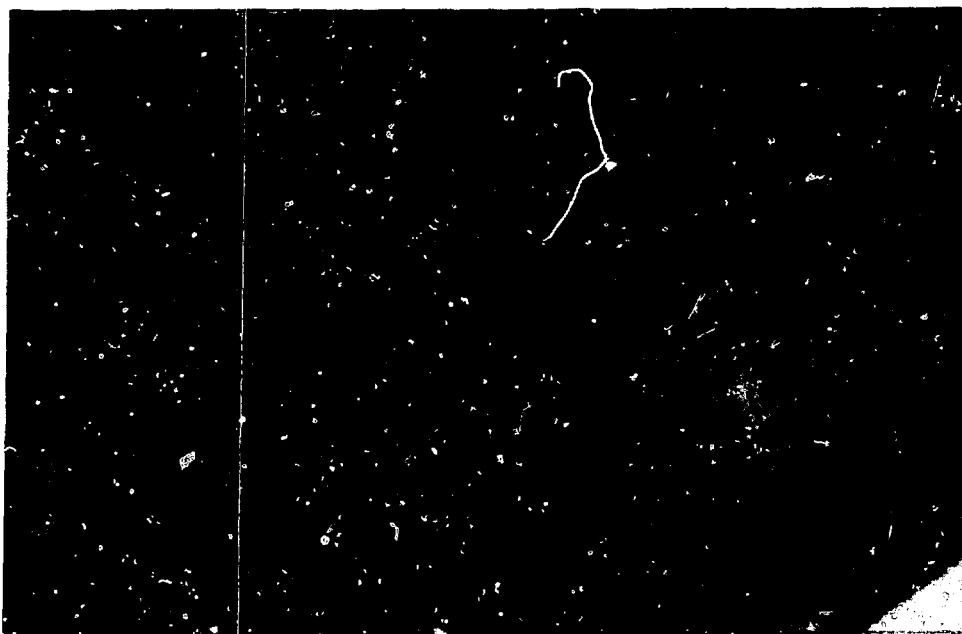
**FIGURE 5.**  
**Device for Removing Liquid Samples (Telltale) from Pipes**



**FIGURE 6.**  
***Gathering and Tying Sleeve Below Bottom of Vessel 13D-3***



**FIGURE 7.**  
***Wrapping Vessel 13D-3 in Horizontal Position on Cribbing***



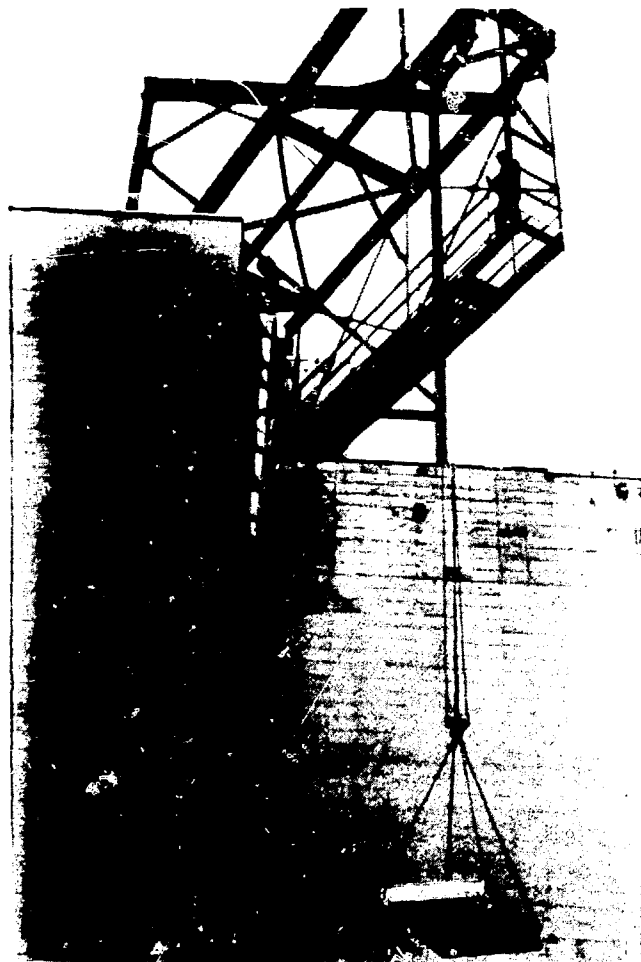
**FIGURE 8.**  
**Column Rigged for Removal from XC-3**



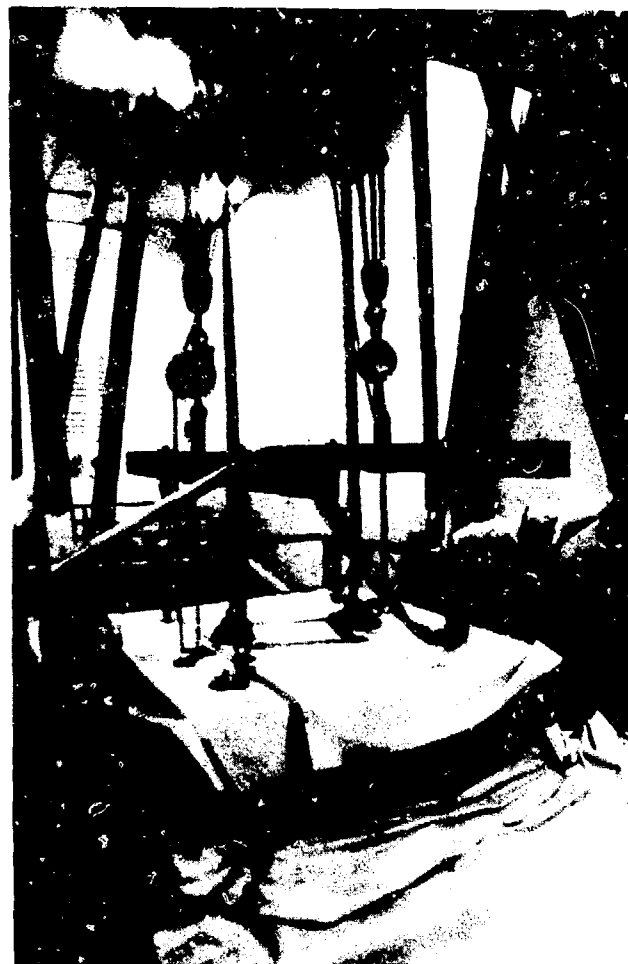
**FIGURE 9.**  
**Rigged Column Being Lifted to XC-3 Hatch**



**FIGURE 10.**  
***Attaching Hydraulic Milling Cutter to Column***



**FIGURE 11.**  
*Monorail Outside of XCR*



**FIGURE 12.**  
*Gantry Crane in XCR Lifting the XC-3 Hatch Cover*

The containers, 55 gallon drums and 90 cubic feet rectangular boxes, are color coded to differentiate between low level and TRU waste. In addition, rectangular containers of various sizes were procured for the interim storage of process vessels. Table 3 shows the volume of waste (low level and TRU) removed from XC-3 and the types of containers used.

Table 3  
Volume of Waste Removed from XC-3

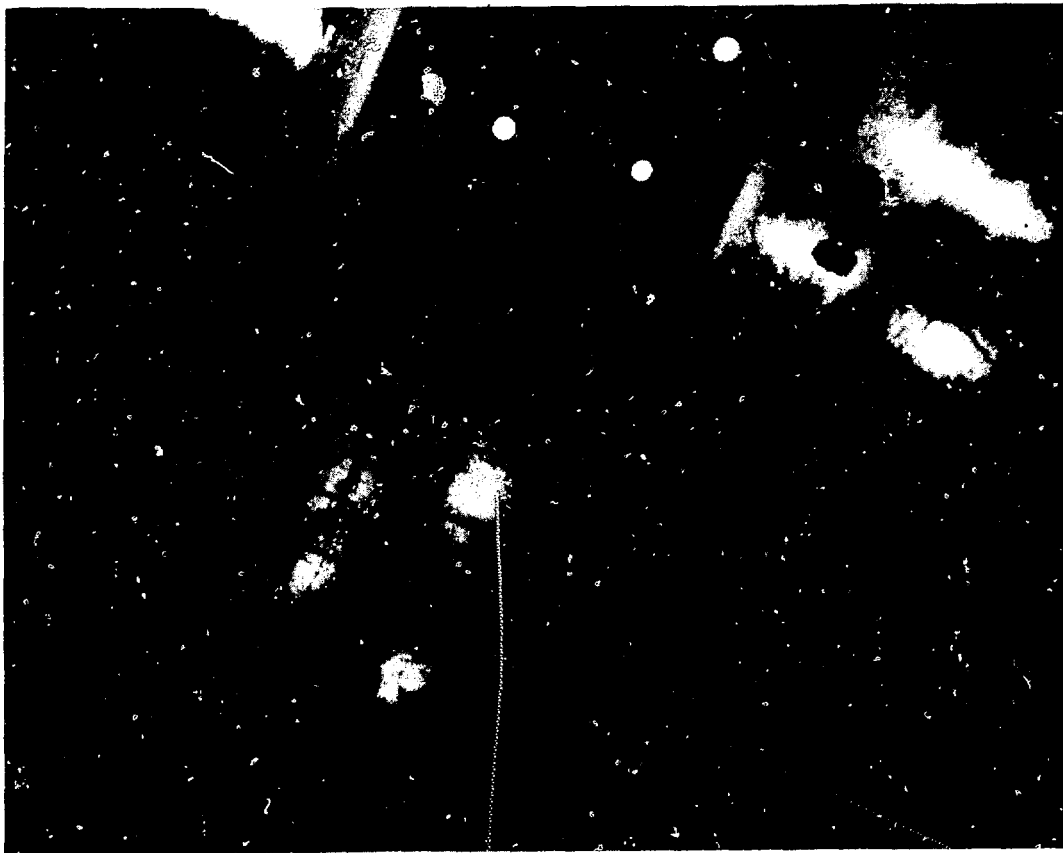
<u>Low-Level Waste</u>	<u>Number of Containers</u>	<u>Volume (Cubic Feet)</u>
55-gallon drum (black - for non-compactible waste)	290	2,131.5
55-gallon drum (yellow - for compactible waste)	26	191.1
*55-gallon drum (white)	229	1,671.7
Rectangular container (B25) (6 x 4 x 4 ft)	1	101.0
Rectangular container (S144) (12 x 6 x 6 ft)	5	2,678.3
Rectangular container (Type A) (67 x 53 x 34 in.)	20	1,615.0
TRU Waste		
**55-gallon drum (white)	30	220.0
	Total	8,608.6 cubic feet

\*It should be noted that white containers were used for "suspect" TRU waste. Subsequent sampling data and assay results show the contents to be in fact low level waste and not TRU. The white container is then color coded with a black stripe.

\*\*Confirmed TRU waste by individual assay.

#### FINAL DECONTAMINATION

After all the vessels, structures, and specified piping had been removed from the cell, various decontamination techniques were tested in-cell and evaluated. The colorless fixative used to minimize airborne contamination during dismantling tasks was not easily removed using high-pressure water (2,200 psi), acidic or mildly alkaline cleaners. Subsequent tests indicated



**FIGURE 13**  
***View of Cell after Final Decontamination***

that a dilute caustic solution (less than 0.5 weight percent) was effective in dissolving the fixative and dislodging contaminated material when used with moderate agitation.

Once the fixative had been satisfactorily removed, further testing with water, nitric acid, paint strippers, and alkaline detergents on wall and floor surfaces was performed. Alkaline detergents yielded the best results. Decontamination factors of approximately 10 were obtained especially when the detergent was incorporated into a foam (to extend residence time) and used with moderate agitation. A low-pressure water stream was effectively used to rinse the spent decontamination solution off the treated surfaces. Survey data before and after testing was evaluated to determine decontamination efficiencies. Prior to final decontamination and painting, the respective cell smear data for alpha and beta contamination levels ranged up to  $1.1 \times 10^6$  and  $1.7 \times 10^5$  dpm/100 cm<sup>2</sup>. After foam cleaning, the smear data indicated relatively low average contamination levels < 500 dpm alpha/100 cm<sup>2</sup> and < 2,000 dpm beta/100 cm<sup>2</sup> on the ceiling and upper walls while higher values (800 to 85,000 dpm alpha/100 cm<sup>2</sup> and 900 to 70,000 dpm beta/100 cm<sup>2</sup>) were noted for the lower walls (20 feet above floor level) and the floor. After final painting, smearable contamination was generally less than 50 dpm alpha/100 cm<sup>2</sup> and less than 500 dpm beta/100 cm<sup>2</sup>.

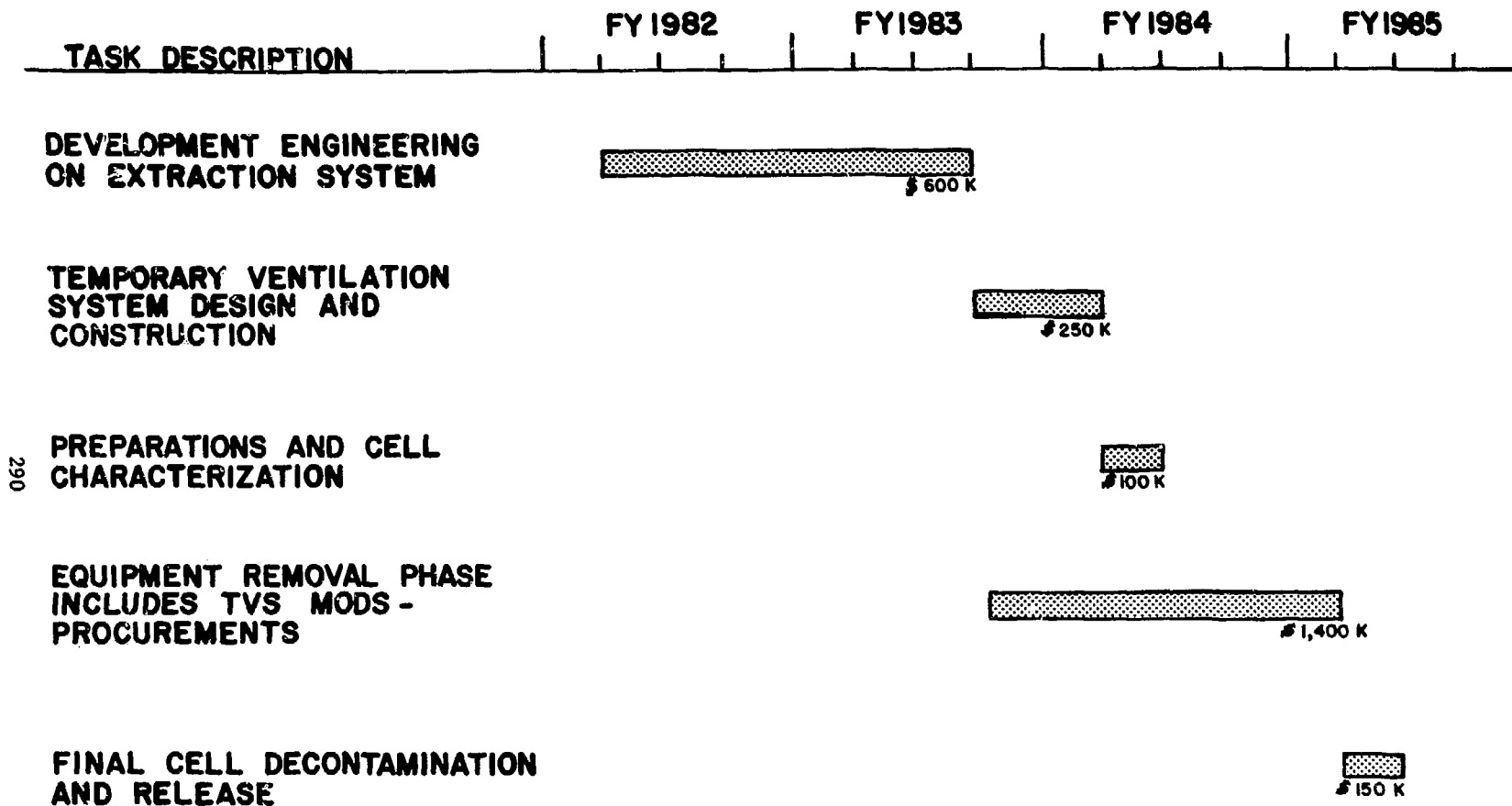
#### COST AND SCHEDULE

Table 4 shows a breakdown of the cost and schedule of the work performed for the decontamination and decommissioning of XC-3. The total cost was \$2.5 million.

#### IMPORTANT OBSERVATIONS

During and after the XC-3 D/D effort, the following significant observations were made:

1. Less time-consuming alternatives to liquid sampling collection must be investigated and implemented to reduce cumulative radiological dose rates and to improve general productivity.
2. In order to preclude problems associated with double-rigging loads, hoisting equipment should be able to reach loads located on the cell floor (60 feet from the XCR floor). The hoist on the XCR jib crane had a maximum lifting height of 46 feet.
3. Air conditioning and/or personnel cooling systems must be evaluated and implemented to reduce operator fatigue and permit longer stay-time in cell. Increased productivity, higher operator morale, and decreased costs (less anti-Cs will be needed and contaminated waste volume will decrease) are the expected benefits.
4. Care must be used with high-pressure liquid systems to prevent the recontamination of previously cleaned areas with overspray containing contamination
5. To sample and collect solid debris effectively, a telescoping vacuum probe should be developed to reduce average personnel exposure rates and enhance sampling and collection efficiencies.



COST AND SCHEDULE DATA FOR XC-3

TABLE 4

6. Existing structures such as beams and platforms could be used to facilitate D & D Operations and to support scaffolding, equipment, and men.
7. Coloring should be added to contamination fixatives so that the extent of the initial covering and the subsequent removal during final decontamination can be visually verified.
8. The utilization of high-pressure water (approximately 2,200 psi) was not an effective method to remove fixative material covering contaminated surfaces. A dilute caustic solution was required with moderate agitation.
9. Acids and alkaline cleaners were effectively used to decontaminate the stainless steel floor while alkaline cleaners incorporated into foams (to increase residence time) were effective on the walls and ceiling.
10. The replacement of wheeled carts with an air transporter effectively reduced operator fatigue, increased maneuverability, and increased the overall height of loads that can be moved through containment.
11. Vessel assaying for fissile material could be safely performed out of cell.
12. Access to the cell through a 6' x 6' opening in the ceiling was very restrictive. Future similar exercises would benefit by increasing the opening.
13. Improved audio and video communication systems must be investigated.
14. Insulating foam was effectively used to retain loose rings and tube bundles in column during cutting operations.

#### REFERENCES

1. West Valley Nuclear Services Company, Radiological Controls Manual.
2. West Valley Nuclear Services Company, Industrial Hygiene and Safety Manual.
3. West Valley Nuclear Services Company, Hoisting and Rigging Manual.
4. Topical Report - Decontamination and Decommissioning of the Extraction Chemical Room (XCR) at the West Valley Demonstration Project.
5. History of Decontamination - NFS, Inc. - G. E. Riethmiller, June 1981

#### ACKNOWLEDGMENT

Substantial input to this paper was provided by D. F. Burke

DECONNING DURING THE 1984 H.B. ROBINSON STEAM GENERATOR  
REPLACEMENT PROJECT

Bruce A. Meyer, C.H.P.  
Carolina Power & Light Company  
Harris Energy & Environmental Center  
Route 1, Box 327  
New Hill, North Carolina 27562

ABSTRACT

In 1984 the H.B. Robinson Nuclear Plant replaced the nuclear supply system steam generators. Many deconning techniques and applications were used during this major year-long maintenance outage.

Some of the deconning techniques used included an alumina grit blasting system; a high-temperature steam cleaning/vacuum cleaner; an ultrasonic Freon degreaser; a sandblasting glove box; and soap, water, and "elbow grease."

Some of the specific deconning applications included the steam generator channel heads, the reactor refueling cavity, a variety of tools and equipment, and the complete general area of the reactor containment vessel.

These and other deconning techniques and applications reduced the need for respiratory protection, reduced the exposure rates in work areas and, subsequently, workers' dose, decreased the personnel contamination events, and reduced the amount of radwaste generated.

DETERMINATION OF U-238, Ra-226, AND Th-232 BY GAMMA-RAY  
SPECTROMETRY IN UNPROCESSED SOIL SAMPLES

Katherine H. Dendahl, Jeffrey A. Brown  
Eberline Analytical Corporation, 800 Oak Ridge Turnpike,  
Oak Ridge, TN 37830

ABSTRACT

Following the excavation of open land areas during remedial action programs, it is frequently necessary to analyze soil samples which have not been processed. Typically, soil is placed in containers such as 0.5 liter plastic jars or 2 liter metal cans at the time it is collected. An analysis procedure for these samples has been developed with reference to counting standards in the same types of container. Information from the initial analysis is used in conjunction with near-surface in situ gamma-ray measurements over excavated areas to support the decision to backfill an area. Following this initial analysis, each sample is dried, pulverized and packed into a Marinelli beaker for final counting. A comparison of the results from both analyses shows the first analysis to be low by as much as a factor of 2. The principal reason for this effect is the presence of moisture in the original samples. The difference is most noticeable for low energy photons from Th-234 (U-238), at least for photons from Th-232 daughters. This paper will address the development of these techniques for analyses.

AN IN SITU MONITORING TECHNIQUE FOR DETERMINING POST REMEDIAL ACTION  
CONCENTRATIONS OF RADIUM IN SOIL

Nels R. Johnson  
Eberline Analytical Corporation, 5635 Kircher N.E.,  
Albuquerque, NM 87109

R. Dan Glenn

Bechtel National, Inc., 800 Oak Ridge Trnpg, Oak Ridge, TN 37830

ABSTRACT

There are several federal programs where action is being carried out to clean up or otherwise control residual radioactivity, at levels above current guidelines, which exists as a result of activities carried out during the early years of this nation's atomic energy program. At most of the sites which are candidates for clean up, the principal radionuclide of concern is Ra-226. Following excavation of open land areas, measuring the concentration of Ra-226 in soil samples from discrete locations or by indirect monitoring techniques. For application in remedial action programs, as in situ monitoring technique has been employed with favorable results. Instrumentation for this technique consists of a 2" x 2" NaI probe mounted on a wheeled dolly so that the detector is approximately 12" above the ground. Radiation levels are read out using a portable digital ratemeter/scaler. Readings with this device are representative of photons which reach the detector from a ground area subtended by projection of the cone shield side wall. This system was calibrated using facilities at the Department of Energy Technical Measurements Center, Grand Junction, CO. A series of measurements at this facility revealed a conversion factor which relates the response for Ra-226 in dry soil:  $\text{Ra-226 pCi/g} = 1.06 \times 10^{-3} \text{ (cpm)}$  - 2.94. The correlation coefficient was 0.99. This technique has been used extensively to predict the Ra-226 soil concentration averaged over individual 100 m<sup>2</sup> areas.

# IN SITU DETERMINATION OF U-238 CONCENTRATIONS IN SURFACE SOIL

Jeffrey A. Brown  
Eberline Analytical Corporation, 800 Oak Ridge Turnpike,  
Oak Ridge, TN 37830  
Richard C. Robertson  
Bechtel National, Inc., 800 Oak Ridge Turnpike,  
Oak Ridge, TN 37830

## ABSTRACT

The manufacture of depleted uranium metal products in facilities designed for this purpose can result in stack releases of uranium oxide to the atmosphere. In some cases, real property in the vicinity of these facilities has become contaminated with uranium above current guidelines. Assessments of contamination boundaries require special monitoring techniques because of the low energy photons emitted from U-238. An in situ soil-assay technique has been developed for the rapid and qualitative analysis of U-238 in surface soil. The field technique consists of a FIDLER detector coupled to a portable ratemeter/scaler (Eberline PRS-1). Methods of calibration, determination of field correlation factors, and detection limits will be discussed. A comparison of U-238 concentrations estimated from field measurements and corresponding U-238 concentrations in soil samples determined from gamma-ray spectrometry will be presented. Utilization of this technique for effectively directing excavation operations at remedial action sites will also be discussed.

## RADIOLOGICAL FIELD SURVEY PROBLEMS & SOLUTIONS (?)

E. J. Deming and A. J. Boerner  
Oak Ridge Associated Universities  
P. O. Box 117  
Oak Ridge, Tennessee 37831-0117

### ABSTRACT

Situations often arise during radiological field surveys which require the health physicist to improvise and/or make spot decisions. At times these situations can be humorous, but they can also present hazards more serious than normal radiological considerations.

This presentation will depict various problematic situations encountered by Oak Ridge Associated Universities Radiological Site Assessment Program in the course of performing field environmental surveys. Detailing these potential hazards can alert other field survey groups to problems they may encounter.

MEASUREMENT OF RADON AND RADON PROGENY IN RADIUM CONTAMINATED  
RESIDENTIAL PROPERTIES IN NEW JERSEY

J. Eng and A. D. White  
New Jersey Department of Environmental Protection  
380 Scotch Road  
Trenton, New Jersey 08628

R. Hopper  
U.S. Environmental Protection Agency/ORP Las Vegas Facility  
PO Box 18416  
Las Vegas, Nevada 89114

ABSTRACT

As part of a radiological investigation and remedial action program, 45 homes affected by radium contaminated soil were monitored quarterly since Fall 1983. Elevated levels of radon and radon progeny were found in homes built on former radium contaminated landfill areas. Radon progeny levels in excess of 0.1 Working Levels (WL) were lowered by installing ventilation systems in the basement. The remaining homes with radon progeny levels less than 0.1 WL were not remediated. Both sets of homes were monitored quarterly through 1985. Radon measurements were made over 3-5 days with the activated carbon canister manufactured by the NJ Department of Environmental Protection based on a US Department of Energy's Environmental Measurements Laboratory design. Radon progeny measurements were made over 5-7 days with the Radon Progeny Integrating Sampling Unit (RPISU) designed by the US Environmental Protection Agency.

This paper will evaluate the impact of seasonal and home use factors on radon, radon progeny and equilibrium factor.

INTRODUCTION

As a result of the New Jersey Department of Environmental Protection's (DEP) program to identify former radiation facilities which may continue to be a public health or environmental problem, three residential areas were found to have radium contaminated materials. Review of historical photographs and records as well as the recollections of longtime residents indicate that these residential areas were former landfills or open areas in the first quarter of the twentieth century. Housing was constructed on these areas in the 1930s through the 1940s. With federal resources provided by the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), 511 residences were sampled for radon or radon progeny during the period October 1983 through February 1984. Those houses with radon levels greater than 4 pCi/l (pico-curie per liter) by carbon canister sampling over several days or radon progeny levels greater than 0.01 WL (Working Level) by Kusnetz grab sampling were identified for additional sampling. The additional sampling was performed with the Radon Progeny Integration Sampling Unit (RPISU) designed by Colorado State University. The RPISU is the primary integrating air sampling unit for radon progeny

used by the U.S. Environmental Protection Agency (EPA). The RPISU sampling performed during December through February confirmed that there were 45 houses with radon progeny levels exceeding 0.02 WL.

It was necessary to obtain integrated radon progeny concentrations in order to reduce diurnal variations and to compare with available limits for risk estimates of lung cancer due to exposure to radon progeny. In early December 1983, state and federal public health and environmental officials met to develop a risk assessment and management plan. This plan provided a schedule for reducing elevated radon progeny concentrations.

Based on the RPISU results, the 45 houses were categorized into three tiers. The 2 houses in Tier A with radon progeny concentration greater than 0.5 WL are to have the exposures reduced within days or weeks by temporary remediation. The 14 houses in Tier B with radon progeny concentration between 0.1 and 0.5 WL are to have these exposures reduced within months by temporary remedial action. Finally, according to the schedule in the risk management plan, those residences in Tier C (25 houses) with levels between 0.02 and 0.1 WL will be remediated by permanent measures within 1-2 years.

As a temporary remedial action measure, 22 houses received ventilation systems in their basements to reduce the elevated radon progeny levels. The ventilation systems funded by CERCLA were designed by an EPA contractor, ARIX. The operation and maintenance (O&M) costs were covered by the New Jersey Spillfund. The systems cost approximately \$10,000 per unit with monthly O&M costs of \$160 for electricity and \$40 for maintenance. The ventilation system was designed to provide 200-300 cfm (cubic feet per minute) of outdoor air which would be conditioned, i.e. heated or dehumidified, before being ducted into the basement. To prevent stagnant air pockets, an exhaust fan was installed to maintain airflow. As part of the design of the ventilation system, ARIX also identified cracks in walls and floors to be sealed and doors to be installed in order to isolate the basement from the first floor. The ventilation systems did reduce the radon progeny levels to below 0.1 WL.

#### MONITORING RESULTS

In order to obtain information on the annual exposure to radon and radon progeny levels, EPA and DEP began quarterly monitoring in 1984 in the 45 houses identified with radon progeny levels greater than 0.02 WL. Quarterly, integrated monitoring of 5 day duration would obtain information on seasonal variability as well as provide a means to assure that the ventilation systems were operating properly.

Of the 45 houses, 44 are owner occupied, 41 are single family dwellings, and 44 have basements. Although 45 houses were monitored, this paper will discuss the results for 33 houses. The remainder had insufficient quarterly data for adequate analysis due to access denial, vacations, or missed appointments.

Quarterly monitoring of radon progeny was performed with the RPISU unit provided by the EPA's Office of Radiation Programs' Las Vegas

Facility. The radon progeny in air are collected on a filter by an air pump. The collected particulates irradiate the thermoluminescent dosimeter (TLD) situated in close proximity upstream from the filter. Filter heads are provided by EPA and the exposed heads returned to the EPA Las Vegas Facility for analysis. The RPISU units are installed and retrieved by EPA's Field Investigation Team (FIT) contractor, NUS Corporation. At the same time, DEP performed quarterly monitoring of radon gas with its activated carbon canisters. The canisters consist of 80 grams of activated carbon charcoal. By partial pressure, radon gas diffuses into the canister. The subsequent radon decay in the canister is measured at the end of the sampling period with a sodium iodide gamma spectroscopy system. Canisters are prepared, installed, retrieved and counted by DEP.

Table 1 summarizes the 5-day integrated radon and radon progeny sampling data for the summer (June through August) and winter (December through February) quarters for 17 houses with ventilation systems installed in the basements. The ventilation systems were effective in reducing both the radon and radon progeny levels, although the radon gas concentration remains higher in the basement than first floor. However, the radon progeny concentration did not necessarily decrease proportionately. In 13 of the houses, radon progeny concentration is higher in the first floor than the basement for the winter monitoring. Of the 17 houses, 16 exhibited maximum radon gas concentration in the winter and minimum concentration in the summer. Radon measurements from all floor levels followed a similar seasonal pattern. Based on the quarterly data, 50% of the ventilated houses could meet the 0.02 WL annual average.

Table 2 summarizes the summer and winter quarterly periods for 16 houses with no ventilation system in the basement. These houses had radon progeny levels between 0.02 and 0.1 WL. For the unventilated houses, there was no strong seasonal pattern for radon concentrations other than basement levels are highest. Since no additional ventilation was introduced into these houses, the radon progeny concentrations are highest in the basement level.

In four of the houses, the highest radon concentration was in the winter quarter, however an equal number had the highest radon value in the fall (September through November) quarter. The remainder of the houses did not exhibit any distinctive seasonal pattern for radon levels.

In both the ventilated and unventilated houses, the radon progeny levels in the basements were suppressed in the winter. Figure 1, a graph of quarterly data for one ventilated house, illustrates this phenomena. This is likely due to increased air circulation when the house heating furnaces are on. Table 3 summarizes the working level ratio or equilibrium factor for the summer and winter measurements. As shown, the winter equilibrium factor is lower than the summer factor. The basement factor is lower than the first floor.

For the unventilated houses, it is surprising to see the basement working level ratio lower than the first floor ratio since one would expect basements to have lower air change rate thus have higher equilibrium factor. As expected, for the ventilated houses, the working

**Table 1: Seasonal Radon and Radon Progeny Data for 17 Houses With Ventilation Systems In Basements**

LOCATION	SUMMER QUARTER		WINTER QUARTER	
	Radon (pCi/l)	Radon Progeny (Working Level)	Radon (pCi/l)	Radon Progeny (Working Level)
First Floor	1.7	0.003	10.8	0.030
Basement	10.3	0.025	20.4	0.020
First Floor	0.4	0.002	4.6	0.010
Basement	0.8	0.002	10.8	0.015
First Floor	1.7	0.010	14.6	0.054
Basement	9.6	0.015	30.7	0.032
First Floor	2.5	0.007	8.8	0.013
Basement	3.3	0.005	9.1	0.006
First Floor	1.6	0.005	30.0	0.055
Basement	7.6	0.024	70.4	0.072
First Floor	1.9	0.004	13.4	0.034
Basement	3.9	0.006	13.5	0.014
First Floor	0.5	0.002	13.0	0.033
Basement	7.7	0.022	28.2	0.047
First Floor	3.6	0.014	11.5	0.024
Basement	6.5	0.011	17.4	0.015
First Floor	7.7	0.036	10.3	0.030
Basement	3.7	0.006	6.0	0.007
First Floor	0.5	0.002	12.4	0.026
Basement	1.1	0.004	17.6	0.020
First Floor	3.6	0.015	38.8	0.065
Basement	11.1	0.030	45.8	0.050
First Floor	1.0	0.005	25.8	0.054
Basement	1.5	0.004	17.6	0.022
First Floor	3.4	0.019	54.2	0.126
Basement	8.0	0.032	58.0	0.052
First Floor	0.7	0.003	15.2	0.027
Basement	1.1	0.004	17.0	0.011
First Floor	.5	0.005	3.4	0.014
Basement	1.0	0.005	2.4	0.004
First Floor	2.0	0.006	4.6	0.005
Basement	5.6	0.004	10.8	0.007
First Floor	0.8	0.004	19.2	0.036
Basement	30.9	0.004	114.0	0.075

**Table 2: Seasonal Radon and Radon Progeny Data For 16 Houses With  
No Ventilation Systems In Basements**

LOCATION	SUMMER QUARTER		WINTER QUARTER	
	Radon (pCi/l)	Radon Progeny (Working Level)	Radon (pCi/l)	Radon Progeny (Working Level)
First Floor	0.5	0.002	15.8	0.040
Basement	3.5	0.024	14.0	0.026
First Floor	0.8	0.006	20.5	0.055
Basement	4.3	0.026	33.2	0.052
First Floor	0.6	0.004	3.8	0.013
Basement	3.8	0.024	9.0	0.025
First Floor	0.4	0.002	7.4	0.029
Basement	8.8	0.040	17.0	0.028
First Floor	0.3	0.004	2.8	0.002
Basement	9.0	0.048	10.8	0.022
First Floor	0.6	0.003	3.5	0.010
Basement	3.8	0.022	6.7	0.021
First Floor	0.6	0.003	7.6	0.017
Basement	14.0	0.035	15.2	0.017
First Floor	8.8	0.040	12.8	0.034
Basement	38.6	0.127	28.8	0.040
First Floor	1.0	0.005	7.5	0.022
Basement	20.6	0.062	20.1	0.026
First Floor	0.6	0.003	4.0	0.011
Basement	13.2	0.045	26.0	0.049
First Floor	0.6	0.006	11.4	0.034
Basement	31.2	0.122	32.3	0.052
First Floor	0.5	0.002	0.5	0.002
Basement	6.2	0.022	12.7	0.007
First Floor	2.1	0.010	3.4	0.013
Basement	22.1	0.082	17.6	0.050
First Floor	0.8	0.004	3.0	0.015
Basement	2.3	0.007	13.3	0.076
First Floor	1.2	0.008	4.5	0.018
Basement	4.2	0.032	7.4	0.024
First Floor	0.5	0.003	3.7	0.011
Basement	7.0	0.024	9.1	0.027

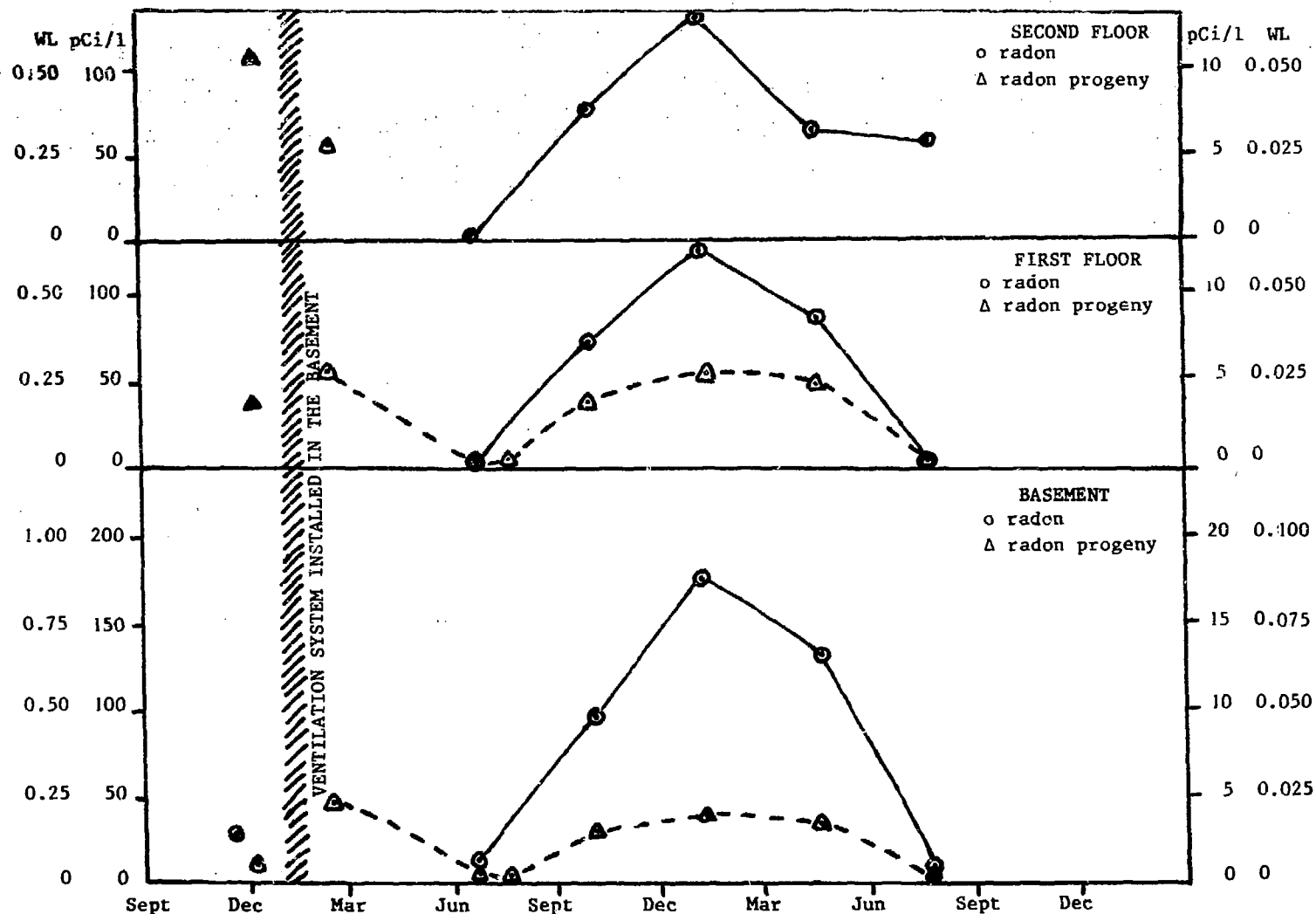


FIGURE 1: INTEGRATED FIVE-DAY MEASUREMENTS OF RADON AND RADON PROGENY. LEFT ORDINATE IS FOR PRE-INSTALLATION OF A VENTILATION SYSTEM IN THE BASEMENT. RIGHT ORDINATE IS FOR QUARTERLY MONITORING DURING VENTILATION OF THE BASEMENT.

**Table 3: Seasonal Working Level Ratio Or Equilibrium Factor\***

	Summer (June-Sept)	Winter (Dec-Feb)
<b>Ventilated Basement</b>		
First Floor	0.44 + 0.19	0.23 + 0.08
Basement	0.28 $\pm$ 0.16	0.10 $\pm$ 0.03
<b>Unventilated Basement</b>		
First Floor	0.56 $\pm$ 0.15	0.31 + 0.10
Basement	0.45 $\pm$ 0.15	0.22 $\pm$ 0.12

$$\text{*Working Level Ratio or Equilibrium Factor} = \frac{\text{Radon Progeny (WL)}}{0.01 \times \text{Radon (pCi/l)}} = \text{WL-R}$$

level ratios for both the first floor and basement are lower due to the deliberate introduction of outdoor air into the basement.

With respect to the impact of home use on integrated radon and radon progeny measurements, insufficient data were obtained since most of the sampling was performed over a 5-day period covering the weekend in order to obtain radon and radon progeny levels in occupied houses. Thus it was not possible to observe differences between weekday and weekend measurements for professional households.

#### SUMMARY

Forty-five houses with elevated radon and radon progeny levels as a result of radium contaminated areas were monitored for a two-year period for radon and radon progeny with week-long integrated samples taken on a quarterly basis. For the ventilated houses, which had the highest radon progeny levels (greater than 0.1 WL) prior to installation of temporary active remedial ventilation system, the highest radon and radon progeny values were measured in the winter months with radon levels highest in the basement and radon progeny levels highest on the first floor. For the unventilated houses, the highest radon and radon progeny levels tend to be found in the basement and when compared to the ventilated houses there appears to be a weaker association of high radon and radon progeny levels to the seasons. For these houses, the equilibrium factor is higher in the summer (0.4-0.6) than in the winter (0.2-0.3).

#### ACKNOWLEDGEMENT

The authors would like to express their appreciation to F. Rauch from DEP and the Field Investigation Team (FIT) from NUS Corporation who installed the quarterly monitoring units.

A PORTABLE WIDE RANGE INSTRUMENT FOR BETA RADIATION MONITORING  
WITH MINIMUM ANGULAR DEPENDENCE

J. Milakovic, G. W. Frank, and  
M. J. Slobodien, CHP  
GPU Nuclear Corporation  
Three Mile Island Unit 2  
P.O. Box 480  
Middletown, PA 17057

ABSTRACT

Most commercially available instruments used for radiation monitoring have great directional dependence when used in beta radiation fields. This directional dependence can result in underestimating the beta radiation fields from areas with large amounts of surface contamination. Such conditions exist at the Three Mile Island Unit 2 plant as a result of deposits of primary coolant on floors, walls, pipes, cable trays, and the like.

The Radiological Instrument Group at TMI developed a directionally independent beta/gamma ionization chamber detector for use with a Eberline Instrument RO-7. The probe is suitable for use in radiation field intensities from 0-20 krad per hour.

The ionization chamber is an inexpensive adaptation of commercially available equipment. Tests using beta radiation from a Sr-90/Y-90 plaque indicated that the angular response varied from 75% to 100% of true dose rate compared with 0-100% with the conventional RO-7 probe. The modification did not significantly alter the beta correction factor and still provided adequate buildup for accurate evaluation of gamma radiation exposure rates at 0.66 MeV (Cs-137).

This paper provides construction details and results of tests for angular dependence, axial symmetry, and beta energy response using the isotopes Pm-147 ( $E_{\max}$  .224 MeV), Tl-204 ( $E_{\max}$  .76 MeV), and Sr-90/Y-90 ( $E_{\max}$  2.3 MeV).

INTRODUCTION

The accident at Three Mile Island Unit 2 created areas in which there were depositions of large amounts of high energy beta emitting fission products - notably Sr-90/Y-90. The resulting beta radiation fields were non uniform and particularly difficult to measure accurately due to angular response limitations of conventional survey instruments. This paper describes the development of a high-range beta sensitive field survey instrument with minimal angular dependence.

Essentially, several prototypes of the instrument were developed and tested. The data show a logical progression from the off-the-shelf directional design to the final omnidirectional configuration. The results of the project demonstrate that an omnidirectional design has been developed and can be employed in field applications when directionality is undesirable.

TABLE 1

Detector Construction

Normal Detector (N)	Aluminum Housing Phenolic Liner Mylar End Window Side Wall Thickness-981 MG/CM2 Window Thickness-7 MG/CM2 Active Volume-7 CM3
Detector #1	Cast Acrylic Wall Thickness-81.4 MG/CM2
Detector #2	Cast Acrylic (Threaded Section) Polypropylene (Active Section) Wall Thickness-84.6 MG/CM2
Detector #3	Cast Acrylic (Threaded Section) Polypropylene (Support for Active Wall) Rubber (Active Section) Wall Thickness-approx. 9.4 MG/CM2
Detector #4	Cast Acrylic (Spacer Inside Area of Old Chamber) Acrylic Bubble (Sheer Acrylic Vacuum Formed Over Mold) Wall Thickness-10 MG/CM2 Max

Densities

Aluminum	2.699 G/CM3
Phenolic	.860 G/CM3
Acrylic	1.123 G/CM3
Polypropylene	.9 G/CM3
Rubber	1.101 G/CM3

TABLE 2

Source Response

Gamma Calibration  
Performed Using - 400 Ci Cs-137  
All Readings in rem/hr

Calibration Points	N	#1	#2	#3	#4
40	40.1	40.1	40.9	39.8	40.1
80	80.6	81.2	82.1	81.8	81.8
162	161.0	163.5	164.5	160.5	160.0

## Beta Response

Beta Correction Factor (BCF)  
Performed Using Sr-90, Y-90 #483, 270, & 220  
All Readings in rad/hr

Calibration Points		4.55	77.16	253.97
N	Reading	4.5	65.9	> 199.9
	BCF	1.01	1.17	NA
1	Reading	2.6	41.0	126.5
	BCF	1.75	1.88	2.01
2	Reading	2.3	37.9	120.0
	BCF	1.98	2.04	2.12
3	Reading	3.9	58.6	197.0
	BCF	1.17	1.32	1.29
4	Reading	3.3	54.6	163.0
	BCF	1.38	1.41	1.56

TABLE 3

Beta Correction Factor Vs Beta Energy

All Readings in rad/hr

Isotope Ave. Energy (KEV)	Sr-90/Y-90 934	Tl-204 244	Pm-147 62
Dose Rate	77.16	93.86	12.47
N Reading	65.9	50.8	2.4
BCF	1.17	1.85	5.2
1 Reading	41.0	2.0	0
BCF	1.88	46.9	N/A
2 Reading	37.9	3.5	0
BCF	2.04	26.82	N/A
3 Reading	58.6	48.1	1.1
BCF	1.32	1.95	11.34
4 Reading	54.6	40.5	.35
BCF	1.41	2.30	35.6

TABLE 4

Symmetry Check - Axially

The detector was rotated 90 degrees on its axis to check symmetry around chamber.

Reading in rad/hr

Rotation	N	#1	#2	#3	#4
0 Degrees	3.2	24.1	34.2	41.7	43.0
90 Degrees	3.2	24.5	36.0	47.1	44.0
180 Degrees	1.9	21.7	36.0	49.1	46.6
270 Degrees	2.3	20.4	34.1	42.5	43.5

TABLE 5

Directional Response

## Plane Source

Performed with Sr-90/Y-90 #483. 270 at 77.16 rad/hr.

Source was placed 100 mm from detector center (along axis). Detector was then rotated from 0-112.5 in each direction in 22.5 degree increments and readings were taken.

	N	#1	#2	#3	#4
112.5	.6	8.6	23.8	23.0	32.8
90	3.2	21.7	34.3	41.7	43.0
67.5	19.7	29.2	42.4	54.7	52.6
45	42.2	35.7	42.4	58.3	56.2
22.5	61.5	39.9	39.5	57.9	56.3
0	65.9	41.0	37.9	58.6	54.6
22.5	54.0	39.4	39.5	59.2	56.4
45	34.4	36.5	42.8	60.0	56.8
67.5	11.3	32.9	43.0	58.5	54.6
90	1.9	24.1	36.0	49.1	46.6
112.5	.3	8.6	24.1	33.0	31.9

## Point Source

Performed with Sr-90/Y-90 at 4.55 rad/hr. Performed as for plane source.

	N	#1	#2	#3	#4
112.5	0	.7	1.5	1.8	2.1
90	.1	1.4	2.4	2.9	2.9
67.5	1.1	2.0	2.8	3.5	3.5
45	2.5	2.3	2.6	3.7	3.6
22.5	4.0	2.5	2.4	3.8	3.5
0	4.5	2.6	2.3	3.9	3.3
22.5	3.7	2.7	2.4	4.0	3.4
45	2.2	2.4	2.7	4.0	3.3
67.5	.6	2.1	2.8	4.0	3.5
90	0	1.7	2.8	3.2	3.0
112.5	0	.9	1.7	2.0	2.0

TABLE 6

Detector #3	<u>End Cap Shield Results</u>		77.6 rad/hr. Gamma Only
	200 mm Sr-90/Y-90		
	Non Directional	Directional	
112.5	23.0	1.0	.9
90	47.1	1.8	1.2
67.5	54.7	3.7	1.0
45	58.3	15.5	.5
22.5	57.9	40.4	.2
0	58.6	51.0	.1
22.5	59.2	43.7	.2
45	60.0	19.2	.4
67.5	58.5	4.7	.9
90	49.1	1.8	1.2
112.5	33.0	1.0	.8

FIGURE 1  
Detector Construction Details

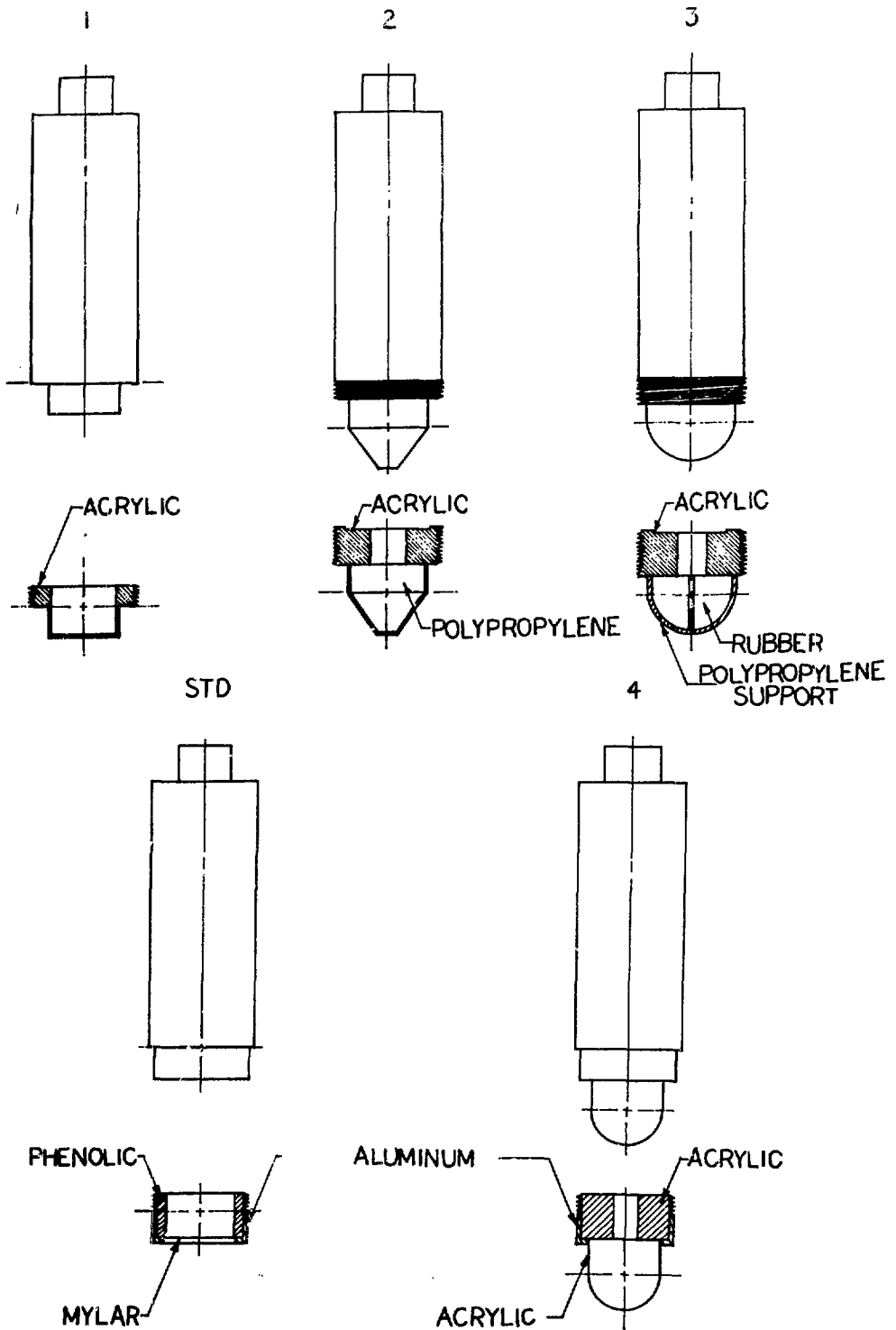


FIGURE 2  
Axial Symmetry

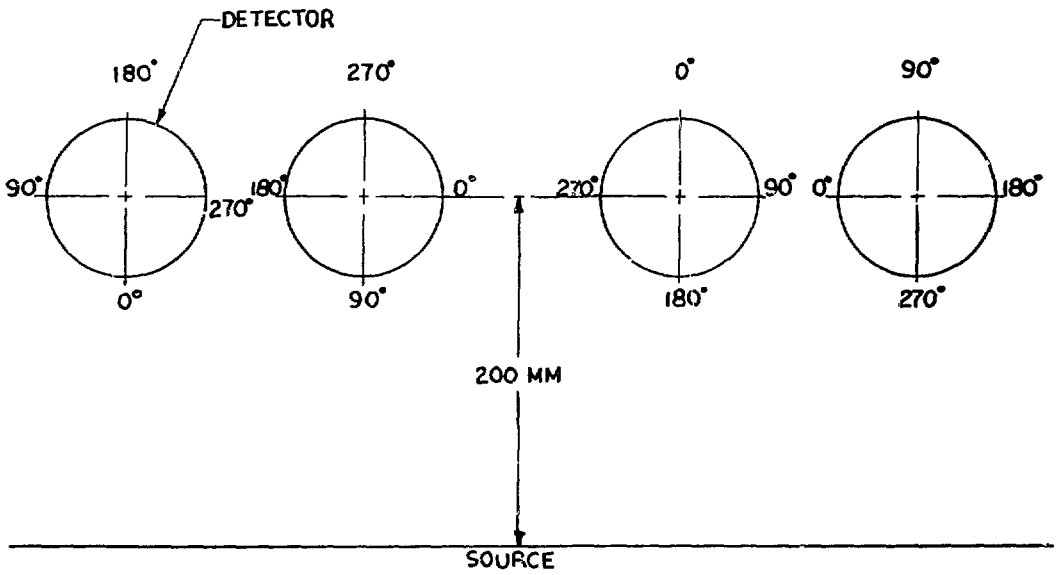


FIGURE 3  
Directional Response

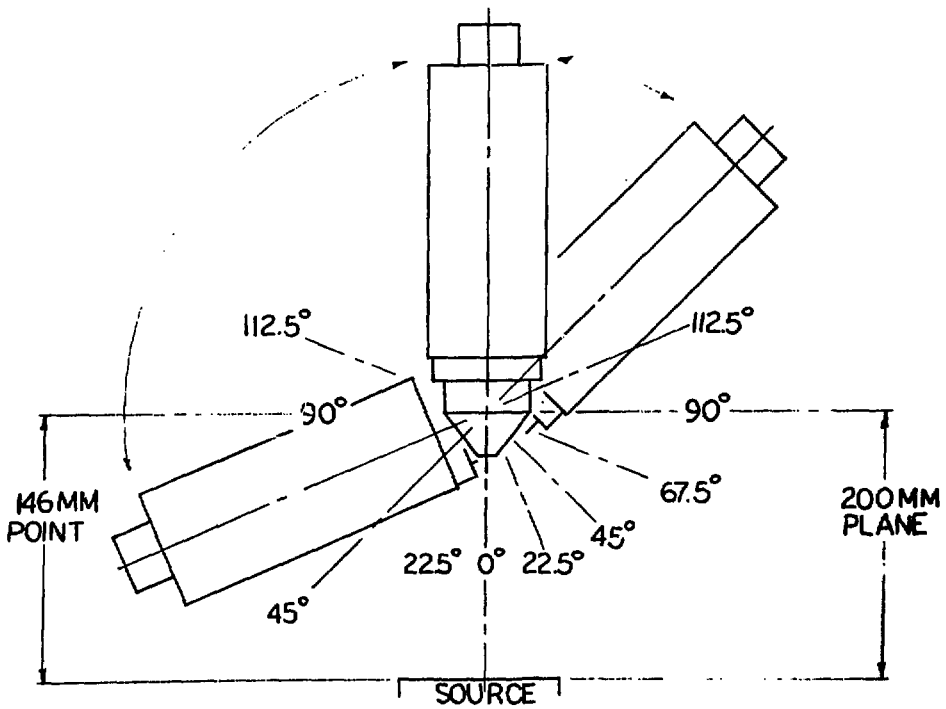
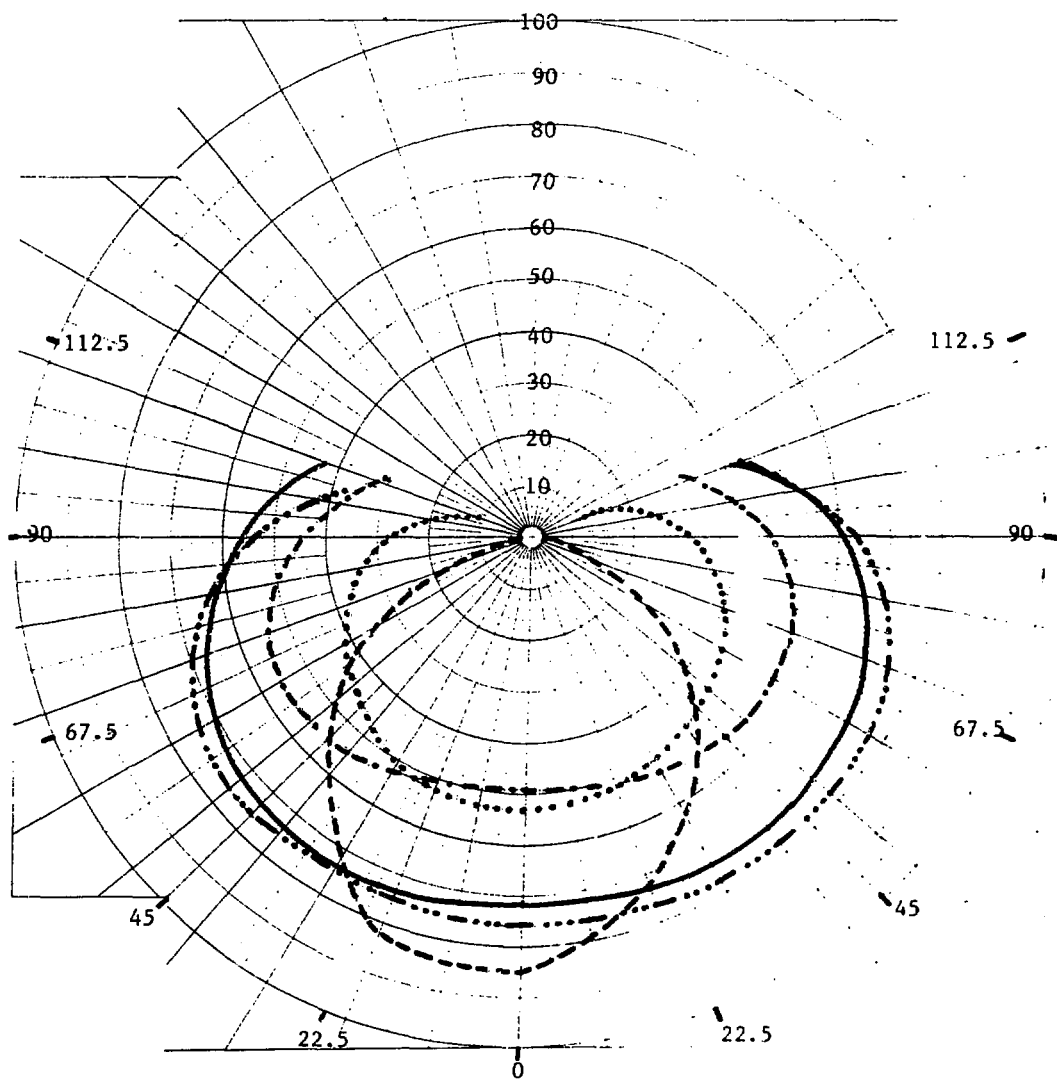


FIGURE 4

Relative Response

Relative Response (%)



Directional Response to Plane Source

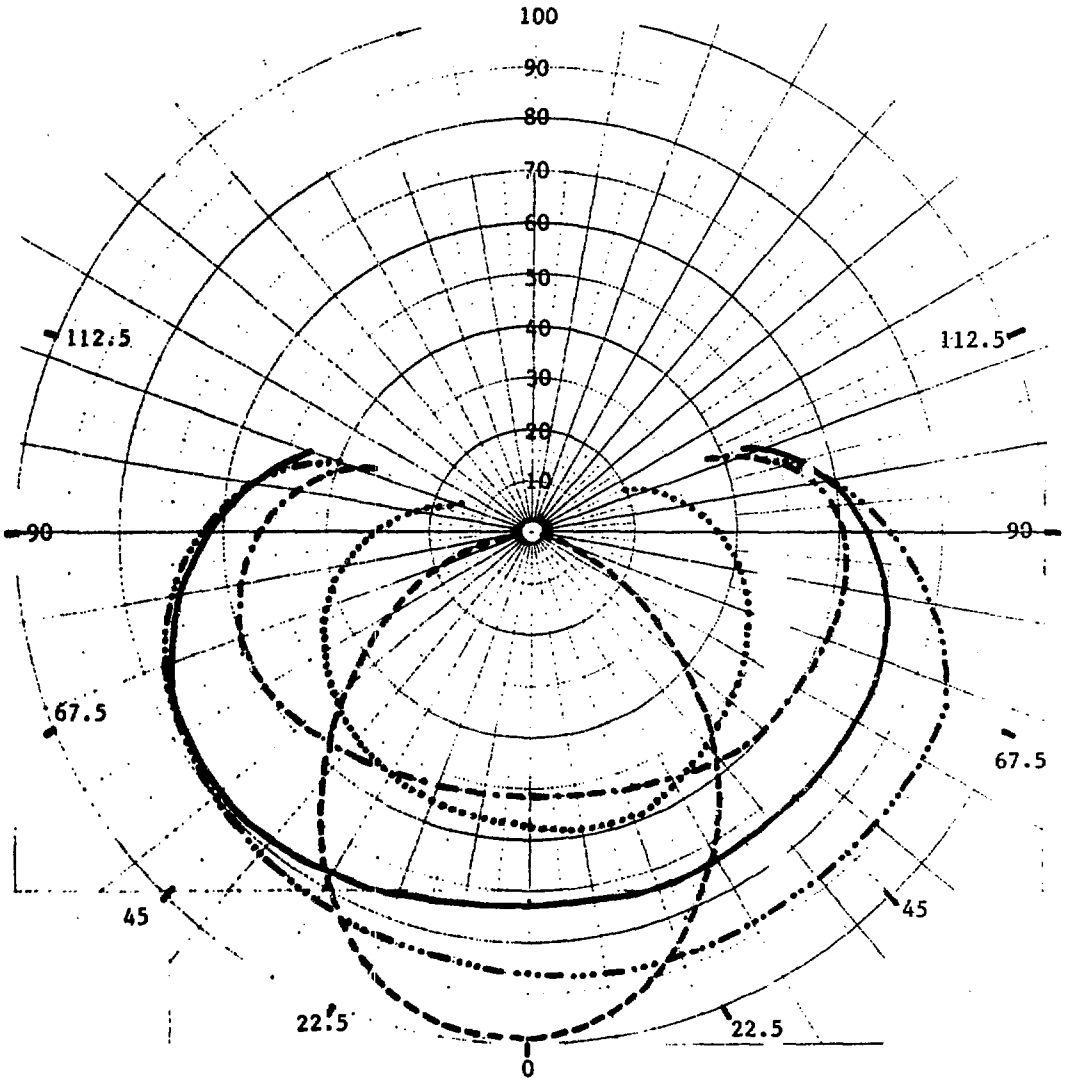
STD Detector

- |    |           |
|----|-----------|
| #1 | -----     |
| #2 | .....     |
| #3 | - . - . - |
| #4 | —————     |

FIRUGE 5

Relative Response

Relative Response (%)



Directional Response to Plane Source

STD Detector

- |    |             |
|----|-------------|
| #1 | .....       |
| #2 | -----       |
| #3 | - . - . - . |
| #4 | —————       |

**FIGURE 6**  
**Shield Configurations**

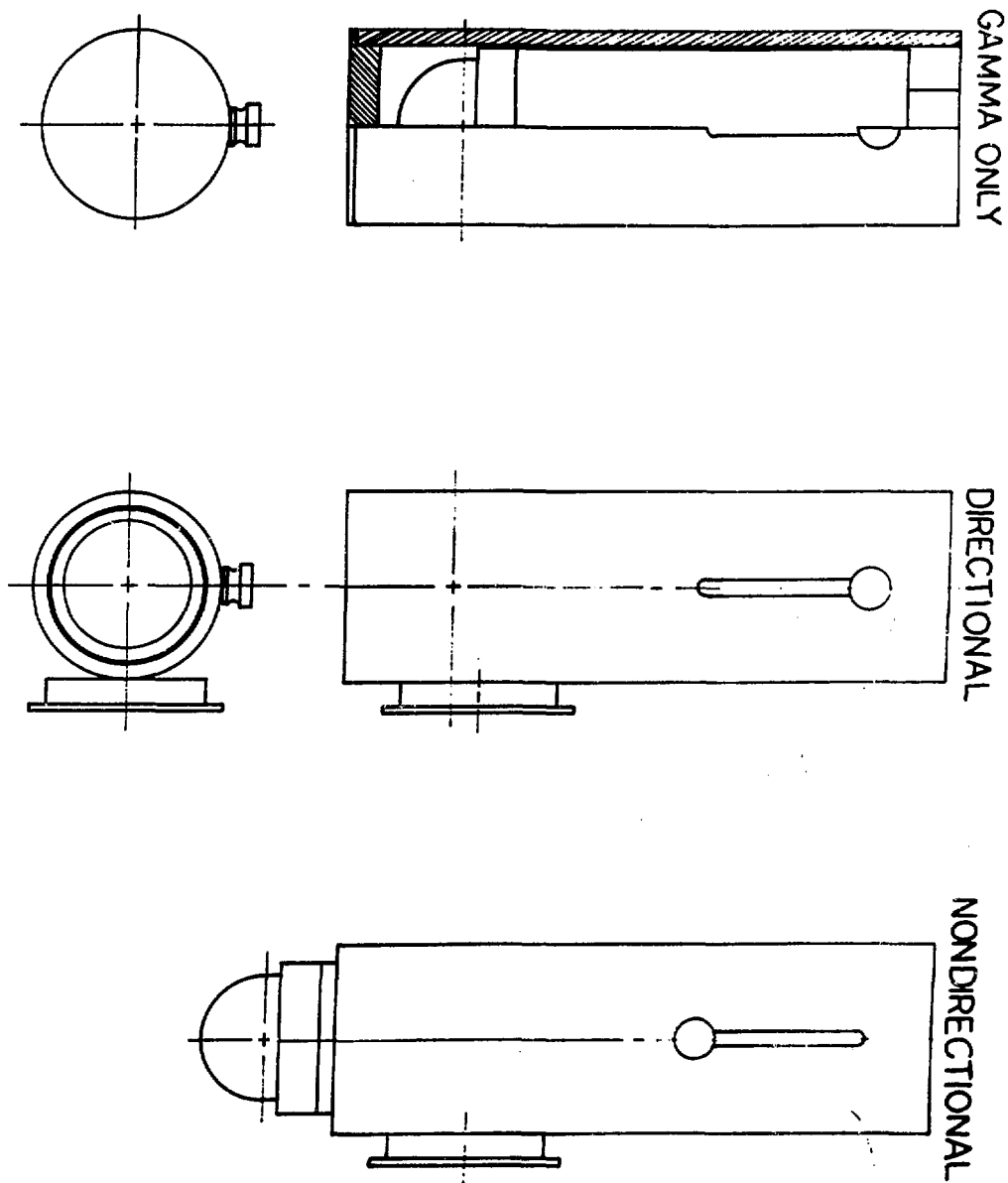
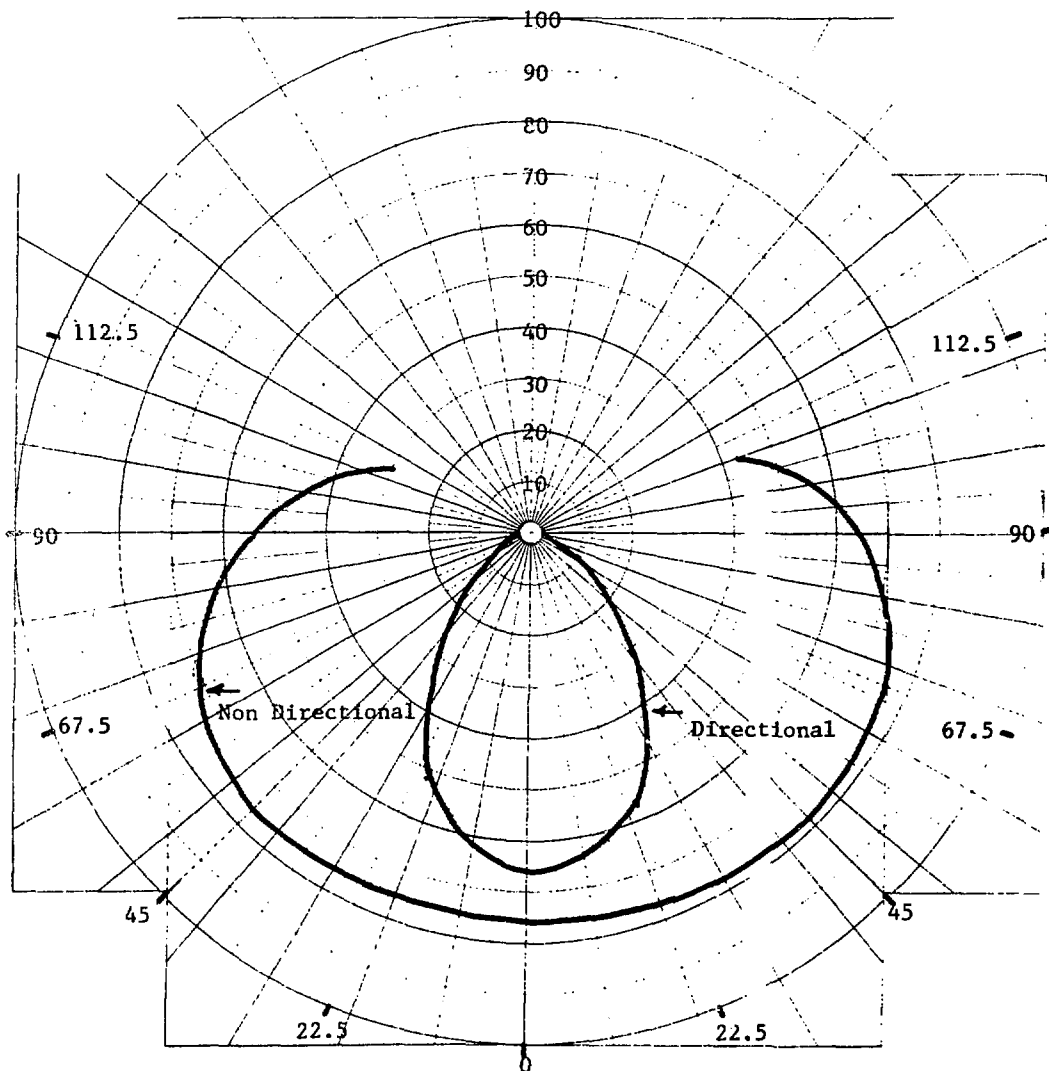


FIGURE 7

Relative Response  
Relative Response (%)



Directional response to plane source with shield in directional and non directional configurations.

JOHNSTON ATOLL CLEANUP - ACCURATE MONITORING FOR ACTINIDES

Lt. Bruce E. Vesper

Lt. T.H. Mohaupt

Cpt. M.A. Dooley

Armed Forces Radiobiology Research Institute

Bethesda, MD 20814-5145

and

Dr. E. Bramlett, Field Command

Kirtland AFB, NM 87116

ABSTRACT

A recent cleanup of actinide contamination was conducted at a site on Johnston Island in the Pacific Ocean. Extensive effort was necessary to quantify specific nuclide concentrations and to determine the total actinide levels as required for shipment to a burial site. Initial monitoring of site debris was made with a FIDLER probe connected to a log-linear rate meter to roughly determine the hot spots. Counting was then done for quantitative actinide levels via a high intrinsic Germanium detector combined with a multichannel analyzer. Isotopic analysis and actinide levels were then determined by feeding MCA data into a computer. This report reviews the approaches to some of the unique challenges encountered in this cleanup effort.

# **EVALUATION OF USE OF THE "NNC GAMMA-12" PORTAL MONITOR FOR INTERNAL DOSIMETRY SCREENING DURING DECOMMISSIONING**

**G. WILHEIM**

Atomic Energy of Canada Limited, CANDU Operations  
and

**J. YOUNG**

London Nuclear Ltd.

## **ABSTRACT**

The implementation of the Radiation Management Program, during decommissioning of Gentilly-1 NPP, includes a requirement to control occupational doses to ALARA levels. The target, to approach zero dose equivalent commitment due to internal uptakes, was perceived as realistic, provided the frequency of in-vivo monitoring of all personnel was sufficient to detect any problem early. The usual health physics control measures would require whole body counting which is both time consuming and expensive if carried out too frequently.

An alternative solution has been found in using a portal monitor, with enhanced sensitivity, for pre-screening of personnel instead of whole-body counts. The National Nuclear Corporation Monitor, Model Gamma-12 was selected mainly for the sensitivity and flexibility, allowing fast processing of personnel being screened. In selecting this method, certain conditions had to be met; in particular, the composition of site main contaminants had to be reasonably well understood. Fortunately, decommissioning which involves a certain "waiting" period, allows for the decay of short lived radionuclides, hence leaving principally only a small number of long lived gamma emitters.

Testing data is given, permitting the evaluation of the sensitivity of the monitor in "internal" mode and its potential in detecting doses related to the limits prescribed by regulatory agencies.

In a final discussion, the advantages and constraints from using this method are presented. Depending on specific circumstances, the use of portal monitor for pre-screening of personnel was found both cost-effective as well as a reliable means of controlling the doses.

## **1. INTRODUCTION**

An essential part of an effective radiation protection program on a nuclear site is the implementation of an adequate internal contamination monitoring method. The method has to be informative enough to ensure the basis for further action and its application should allow early detection of active traces. The equipment selected as well as the way it was used, the scheduling of the involved personnel to the contamination checks are all subject to these objectives.

The means available to attain these objectives are primarily dependent on the actual site conditions; in the context of the Decommissioning of Gentilly-1 nuclear plant, it was considered opportune to use a high sensitivity Portal Monitor to survey plant personnel for both external and internal contamination. It is an approach that, for the particular conditions of Gentilly-1 site, was considered satisfactory responding both to regulatory/monitoring requirements as well as to economic constraints.

## **2. INTEGRATION OF INTERNAL/EXTERNAL CONTAMINATION MONITORING**

### **2.1 Objectives and Criteria**

In the early stages of planning the radiation protection program for the Gentilly-1 site (1), it was necessary to find an effective and economical way to ensure a timely detection of an internal contamination uptake.

In order to do this, several criteria had to be met:

- High level of confidence in detecting the major contaminants at site (all gamma emitters);
- Assuring detection of beta emitting contamination;

- Using equipment of minimal complexity, not requiring highly specialized operators;
- Capital costs within restricted budgetary limits of G-1 project.

Also several specific site conditions were taken into account:

- Duration of project: 12-18 months;
- Site personnel: changing between a low of 50 and a high of 150;
- Critical group exposed to internal contamination: 15-30 persons;
- Degree of risk to uptake: moderate \*
- Close proximity of an operational 600 MW nuclear station (Gentilly-2), having WBC and urinalysis facilities.

## 2.2 Options

A cost-benefit analysis of several possible options was made considering the above characteristics. The estimates were based on an annual need of 650 counts (25 persons screened every two weeks).

1. To have WBC services provided by the nearby Gentilly-2 NPS at a cost of \$100 per count; total annual cost \$65 000.

2. AECL to own and operate WBC system at site this would include the cost of equipment at about \$60 000 (for the least sophisticated system), a full time operator to operate it, perform QA functions, maintain it and generate QA reports - at a yearly cost of about \$40 000 and costs incurred to purchase, calibrate and set-up the WBC at about \$15 000; total cost for this option would amount to \$115 000.

3. Outside Contractor a typical US contractor could provide a WBC on site with an operator; the costs for this would be approximately \$55 000 yearly equipment rental, equipment relocation cost of \$2 000 and operator cost of \$37 hour plus per diem for a total cost of approximately \$130 000.

None of these alternatives seemed to satisfy both the flexibility required and the relatively moderate budget available for the R/P program. Option 1 while less expensive in direct costs, required each individual to be tested off-site, with considerable time expenditure; option 2 and 3 assured a better flexibility but at an unacceptable price.

During that time, the procurement of a portal monitor to survey the traffic at the main site entrance for external contamination was pursued; this led to the suggestion of using the portal monitor for both external/internal contamination monitoring.

Several factors showed to be in favour of this approach:

- The high sensitivity featured by the latest offerings from the contamination monitors manufacturers;
- The possibility of having it installed and operated at the site gatehouse, in a fairly low background environment;
- The relatively low traffic of people at the gatehouse.

In selecting the actual monitor, several criteria had to be satisfied: high enough sensitivity, gamma detection capabilities, possibility of extra-shielding, and flexibility to operate in either external or internal contamination mode.

The final selection decided on was a National Nuclear Corporation "Gamma-12" Model \*\*, essentially a "Gamma-10" type (used as a standard for external contamination) with additional facilities for shielding and switching from one mode to another.

To estimate the effectiveness of the monitor to detect internal contamination, the listed sensitivity in internal contamination mode was compared to the regulatory requirements:

- Listed sensitivity for  $^{60}\text{Co}$  \*\*\* (in a  $15\mu\text{R/h}$  environment/10 s counting time): 555 Bq (15 nCi).
- Annual Limit of Intake (ALI) for  $^{60}\text{Co}$ , according to recommendations from ICRP-30:  
 $1 \times 10^6 \text{ Bq}$ .

\* Considered for a site with no active high pressure/temperature systems and with only hazards coming from contamination of room surfaces, tools, etc. or from manipulation/storage of irradiated fuel.

\*\* The selection was done after a market research limited by time constraints and does not exclude the possibility of using other manufacturers products to achieve equivalent results.

\*\*\*  $^{60}\text{Co}$  was selected as being the most restrictive radionuclide present at the Gentilly-1 site.

This way a threshold of sensitivity of 0.055 of 1% of ALI could be obtained. Considering 1% of ALI an acceptable limit of investigation, leads to the conclusion that the selected equipment will comply with the requirement for an early detection.

### 2.3 Description of Selected Option

The "Gamma-12" unit (see Fig. 1), is an upgraded version of the "Gamma-10" model. It consists of a two side and foot detector assembly, with the microprocessor-based electronics in a detached mini-bin unit (2)

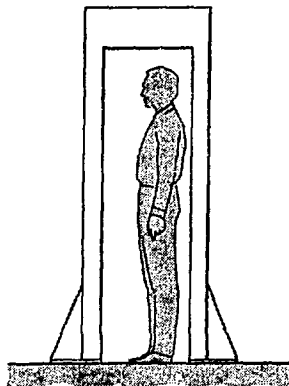


FIGURE 1 PORTAL MONITOR WITH PERSON FOR INTERNAL CONTAMINATION SCREENING

The detector assembly consists of plastic scintillators, 200 cm high, 28 cm wide coupled to a 5 cm photomultiplier tube operating at 1500 Volts.

The electronic unit consists of a mini-bin and power supply, high voltage power supply, Amp/Single Channel Analyzer, ratemeter and alarm module. External controls include adjustments to the amplifier gain, width and lower level of the SCA window, ratemeter range and damping; also occupancy detector is included (ultrasonic detection) which activates the external contamination alarm in presence of a person.

The alarm unit has provision for warning signals (red light and buzzer) whenever radioactive contamination is detected on a person passing through; also an alarm is provided if either high or low background limits are exceeded.

To also detect internal contamination new features were added:

- Additional lead panel shielding: instead of the standard shielding of 1 cm lead backing the side detectors and the top, the manufacturer installed 2 cm lead plates and added two other complete side/top identical lead "dummy" portals to extend the original portal in both directions; the end result is a tunnel 1.1 m long for better background control;
- Switching from "Walk-thru" mode (external contamination monitoring/alarm) to "Body scan" mode; the latter mode entails keeping only one side detector active while the opposite side and the foot detector are disconnected; the individual tested will be facing the active side;
- A counter with a six-digit display and printer for recording the testing results;
- A timer with selector switch to set the counting time span (ten spans ranging from 1 s to 10 min. and one manual position).

This set-up allows for a convenient selection of the type of operation as one single mode-switch and selection of counting time are sufficient to set the system ready for internal contamination checks.

### 3. DATA ANALYSIS

To interpret the data obtained from counting sessions, a "true" count rate was extrapolated from the measured count rate. Background variability was tested first. The background was continuously sampled over a twenty four hour period using five minute counting periods, data is plotted in Fig. 2. The background is generally constant except for several largely deviating points, which a Poisson Distribution would not statistically predict. No viable explanation was found for these outlying points although site activities at G-2 could have been responsible. However, this is hard to ascertain as no air sampling was done during the experiment.

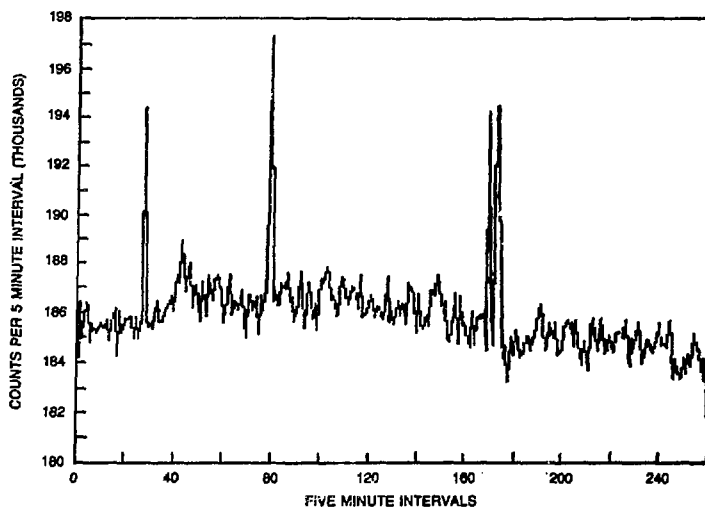


FIGURE 2 BACKGROUND AS A FUNCTION OF TIME

For simplicity, it has been assumed the background is constant over short intervals of time such as those taken while doing body counts. The total test time for an average group of 5 persons is about twenty minutes. To eliminate the possibility of using unreasonable background rates an objective criteria based on a Shewart Chart was implemented. The accepted mean background is 37152 cpm, a background reading outside the outer action limits (3 sigma from the mean) or two consecutive readings outside the inner warning limits (2 sigma from the mean) constitutes a change in the mean. When this happens a background investigation is carried out to determine if this fluctuation can be explained by site activities such as fuel transfer. If no explanation is found and the background subsequently returns to "normal" then the extraneous background is disregarded. Any reading falling within the warning limit is used as background without further investigation.

Secondly the variation of efficiency as a function of height was examined. A 911 Bq (25 nCi)  $^{137}\text{Cs}$  source was placed at a constant distance away from the body scan detector and measurements were taken along several points on the vertical axis. The efficiency for each distance was calculated according to the formula:

$$\text{Eff} = \frac{R_o - B}{R_t} \quad (1)$$

where  $R_o$  = observed count rate of source + background (cps)

$R_t$  = true count rate of source (dps)

$B$  = background rate (cps)

The efficiency at various distance is plotted in Fig. 3. The efficiency is a constant function of height except at the extreme bottom and top. This is due to the reduction of detector surface subtended by the relevant solid angle of radiation.

To simulate internal contamination conditions two body phantoms were constructed. The first consisted of five 1 cm plexiglass sheets stacked together to form a 5 cm plexiglass block. A 911 Bq (25 nCi)  $^{137}\text{Cs}$  source was placed behind the block and for a constant height (approximately chest level), the phantom was moved away from the detector. Chest level was chosen since  $^{60}\text{Co}$  tends to accumulate in the lung. A second phantom was constructed using a 25 cm diameter cylindrical plastic container filled with 7.50 kg of water spiked with 740 Bq (20 nCi) of  $^{137}\text{Cs}$ . Once again the phantom was placed at chest level and a series of measurements taken for various source to detector distances.

It should be noted that the source to detector distance was considered to be the distance from the closest point on the circumference of the cylinder to the detector. Equation (1) was used to calculate efficiency.

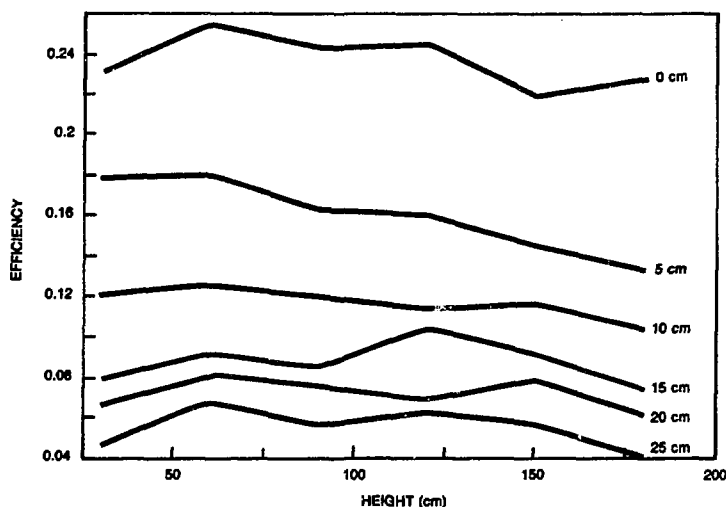


FIGURE 3 EFFICIENCY AS A FUNCTION OF HEIGHT  
(FOR CONSTANT S-D DISTANCES)

The results for both these body simulants are plotted in Fig. 4. The efficiency of the second phantom is slightly lower due to the effect of self shielding. Because this phantom more closely simulated the desired conditions, it was chosen as a suitable body replicant. It also provides a more conservative estimation of activity level. To optimize efficiency the occupant stands as close as possible to the body scan detector. In this position an efficiency of 3.45% has been estimated, corresponding to a 5 cm source to detector distance of the water phantom.

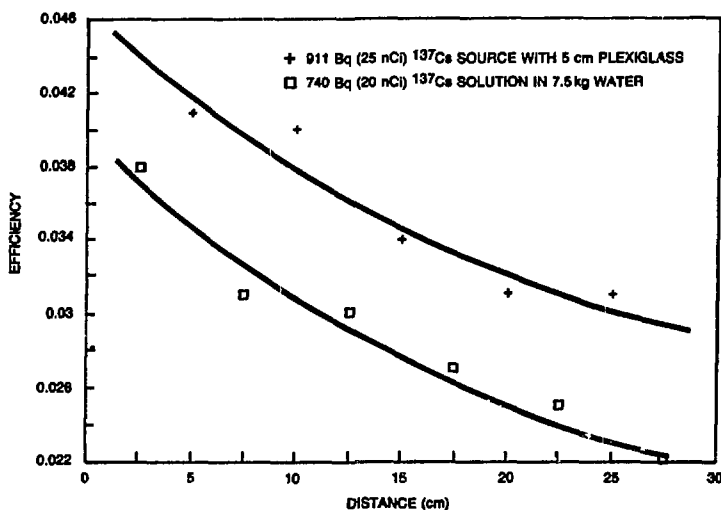


FIGURE 4 EFFICIENCY AS A FUNCTION OF DISTANCE (BODY SIMULANTS)

A correction factor due to the effects of self-shielding by the human body has to be added. This was necessary as for virtually all screening sessions the count rate with an occupant was lower than the background count rate. Hence the correction factor for a standard 175 cm tall man weighing 75 kg was determined. A previous WBC at G-2 showed the occupant to be free of any internal contamination. By taking a series of five minute counts with and without the occupant present the count rate with the occupant was observed to drop an average of 10845 counts in four different trials. The mean background for these trials was 185714 counts while the mean for the occupant plus background was 174869 counts. Consequently the background was corrected for this by simple subtraction.

$$B^1 = B - 36.15$$

(2)

where

$B^1$  is the corrected background rate (cps)

$B$  the measured background rate (cps)

A program was written on a micro-processor to analyze the data, give a hardcopy of the results and store the information for later retrieval.

The final objective was to determine whether a counting period of one minute was sufficiently long to reliably detect one percent of the annual limit intake (ALI) of  $^{60}\text{Co}$ . All tests were done with  $^{137}\text{Cs}$  because of its easy availability. However, it should be noted that the detection capabilities of the Gamma-12 are greater for  $^{60}\text{Co}$  than  $^{137}\text{Cs}$  due to its higher gamma energy, resulting in a conservative estimate of efficiency.

Using a 95% confidence level, the decision limit,  $L_c$ , and the detection limit,  $L_d$ , can be calculated according to the following equations: (3)

$$L_c = (2.32 \sqrt{B}) / \text{Eff} \quad (3)$$

$$= 216 \text{ Bq. (5.8 nCi)}$$

$$L_d = (2.71 + 4.65 \sqrt{B}) / \text{Eff} \quad (4)$$

$$= 434 \text{ Bq. (11.7 nCi)}$$

(Efficiency of 3.45% at 5 cm for water simulant.)

(Average background rate of 37 152 cpm.)

The (ALI) for  $^{60}\text{Co}$  is listed as  $1 \times 10^6 \text{ Bq}$ . (ICRP Publication 30.) A one percent investigation limit of this level corresponds to  $1 \times 10^4 \text{ Bq}$ , which is within the limits of detection of the Gamma-12. Hence, by screening individuals on site and only acting when the investigation limit is reached, numerous expensive WBC at another facility are eliminated.

#### 4. TESTING PROGRAM

In organizing a testing program for the site employees, several factors had to be taken into account:

##### 4.1 Location

The monitor was installed at the main entrance of the site (gatehouse) to monitor all the traffic in and out of the station. This was beneficial for the internal contamination monitoring, as it was operating in an optimum environment due to low background (@ 3-4  $\mu\text{R/h}$ ). However, this created a potential interference between the two functions, as somebody standing by the monitor for a WBC would block the passage way for normal traffic. This was resolved by scheduling the checks for the critical groups during low traffic periods in the day.

##### 4.2 Critical Groups

Most employees at site would normally never be exposed to airborne contamination, therefore, they would not come into consideration for a regularly scheduled screening; this schedule would apply to critical groups, essentially persons working in contaminated areas; the working groups falling into this category were:

- Irradiated fuel manipulation, transfer and storage group
- Decontamination (of rooms, equipment, etc.) group
- Radiation protection group
- Active waste storage group.

##### 4.3 Frequency

Screening of the critical groups every two weeks was considered frequent enough taking into account that the object of detection were long lived radionuclides ( $^{60}\text{Co}$  or  $^{137}\text{Cs}$ ). The supervisors of the working groups were advised that this procedure was available to them anytime and they were encouraged to send their people for testing whenever they deemed necessary.

Also occasionally, other working groups were called for testing whenever it was considered pertinent.

#### 4.4 Counting Procedures

- Each person is counted for a one minute span. Before and after testing a group, a five minute background count is taken. Groups tested at a time are no larger than 10 persons.
- During the one minute count, the person is facing the detector for 30 s and standing with his/her back to it for the remaining 30 s.

The persons taking the count have to wear their street clothing, not working clothing.

#### 4.5 Complementary Measures

In using this method as a screening process, its limitations were recognized and solutions found:

a) It would act as a warning system, without indicating the nature of the problem; an arrangement was established with nearby Gentilly-2 nuclear station to perform a full spectrographic analysis for each individual suspected of a contamination uptake.

b) It would detect only gamma radiation, not being sensitive to beta. The essential components of most of the site contaminants were  $^{60}\text{Co}$  and  $^{137}\text{Cs}$  having significant gamma emission (a typical sample of site contamination would contain 80%  $^{137}\text{Cs}$  and 20%  $^{60}\text{Co}$ ). In order to further limit the possibility of non detection of a beta contaminant, a bioassay program for the critical group was instituted (this would have been necessary with any other WBC method).

c) It was difficult to differentiate between external and internal contamination. To detect the location of the source, empiric or inferential methods had to be used; thus in a typical case, after a suspect count, no external contamination was found by frisking, nor by the subsequent whole body count with the specialized equipment in Gentilly-2. The inference was that the person had some minor contamination absorbed in his street clothing which he used sometimes at work without proper protection.

#### 5. CONCLUSION

The Gamma-12 portal monitor was installed and commissioned at Gentilly-1 Decommissioning Site in July 1985 and since then it has produced beneficial results in monitoring site personnel for both external and internal contamination; the method proved to be cost-effective and efficient; a standard portal monitor with an additional \$3000 worth of shielding proved sufficient to ensure screening for internal uptake; screening which was done in situ, requiring only approximately 2-3 min./person, compared to about 10-15 min. for a full whole body count.

The application of the method was largely due to the favorable factors specific to the Gentilly-1 site

- Good advance knowledge of the contamination components
- Site work force of relatively small size
- Need for the equipment on a relatively short term project
- Installation of the monitor in a low intensity background (site gatehouse)
- Close proximity of a nuclear facility disposing of an available whole body count back-up equipment.

It could be argued that these particularities show the limitations of this method, as they would represent requirements for its application; this is not necessarily so; for instance, in a larger nuclear site, several portal monitors would be required and one of them, installed in a low background/low traffic area could be used for internal contamination screen-up; and even if equipment for whole body counting is already in place, its operator workload could be decreased while in the same time increasing the frequency of screening.

In summary, it can be said that the described method has a definite application for smaller projects with a short term duration typically exemplified by decommissioning of nuclear facilities. However, the method's potential is worth considering also for different situations.

#### 6. ACKNOWLEDGEMENTS

This article was possible due to the encouragement and inspiration provided by Dr. M.A. Maan, assistant plant manager at Gentilly-1 and Mr. S. Schafer, manager of the Health and Safety program at Gentilly-1; the authors also extend their thanks to the plant manager, Mr. B. Gupta for assuring the facilities necessary to complete the research required by this study.

## **7. REFERENCES**

- (1) Michael A. Maan, R.P. Nair, and S. Schafer, "Establishing a Comprehensive Cost-Effective Radiological Protection Program for Decommissioning Project, 19th Midyear Topical Symposium of Health Physics Society, 1986.
- (2) Instruction Manual, Gamma-10 Portal Monitor with Gamma-12 Modification, National Nuclear Corporation, 1984.
- (3) "A Handbook of Radioactivity Measurement Procedures", National Council on Radiation Protection and Measurements, Report No. 58, 1973.

## RADIOACTIVE WASTE ASSAY AT THE WEST VALLEY DEMONSTRATION PROJECT

S. Marchetti, C. W. McVay, and J. R. Stimmel  
West Valley Nuclear Services Company, Inc.  
P. O. Box 191  
West Valley, New York 14141-0191

### ABSTRACT

The decontamination and decommissioning of the West Valley nuclear reprocessing plant is generating a large variety of types of radioactive waste which require classification in accordance with 10CFR61. A radioactive waste assay system has been developed to efficiently assay and classify this radioactive waste. The waste is assayed by segmented gamma counting, passive neutron counting, dose rate conversion, and/or laboratory analysis. The system is capable of handling all the waste from generated at the West Valley Demonstration Project. The system produces a list of nuclides present with their concentrations and determines the classification of the waste based on the criteria contained in 10 CFR 61.

### INTRODUCTION

In March 1983, WVNS received DOE approval to proceed with development of a TRU waste sorting plan with assay techniques including:

- o Assaying 55-gallon drums of low-density waste (about 0.15 grams/cm<sup>3</sup>) using high-resolution gamma spectroscopy.
- o Assaying high-density wastes using large area passive neutron detectors operated in a coincidence mode.
- o Estimating TRU content of large-volume, low-dose rate objects using direct gamma spectroscopic or alpha surveys.
- o Estimating TRU content of high-dose rate equipment using remote sampling and radiochemical assay.

The proposed system was to separate TRU waste from non-TRU waste at the 100 nCi/g level for TRU waste. Equipment procurement was initiated immediately resulting in the purchase of a segmented gamma scanner and the "loan" of a 2  $\pi$  passive coincidence neutron counting system from Los Alamos National Laboratory. The equipment had been modified for TRU waste separation when DOE decided that WVNS would implement DOE Order 5820 by classifying waste according to the criteria contained in 10 CFR 61<sup>1</sup>. The equipment procured was further modified after this decision to perform the waste classification function.

The methodology for waste assay now in use at the WVDP includes:

- o Segmented gamma scanning of low-density, 55-gallon waste drums.

- o Passive neutron counting of suspect TRU waste in containers other than 55-gallon drums and high-density waste in arrays of 55-gallon drums, using both coincidence and total neutron counts.
- o Estimating the curie content of non-TRU isotopes in waste by conversion from dose rate using scaling factors and radiochemical assay data.
- o Estimating curie content of waste by sampling and radiochemical analysis.

## DEVELOPMENT PLAN

Waste areas and streams were characterized based upon existing surveys and the plant operating history. Areas of possible TRU contamination were investigated further by characterization of plant systems to better define wastes. This approach focused evaluation efforts on areas and systems which could contain TRU waste and eliminate extensive characterization and assay work in areas which are free of transuranic isotopes.

Radiochemical assay of areas were characterized based upon existing surveys and the plant operating history. Areas of possible TRU contamination were investigated further by characterization of plant systems to better define wastes. This approach focused evaluation efforts on areas and systems which could contain TRU waste and eliminates extensive characterization and assay work in areas which are free of transuranic isotopes.

Radiochemical assay of areas where transuranic waste was suspected using the following methodology:

- o High-resolution gamma spectroscopy.
- o Separation followed by alpha spectroscopy.
- o Separations followed by beta analysis.

This isotopic data was used to develop:

- o Total neutron yield information to support sorting high-density waste using passive neutron coincidence counting.
- o Ratios of total TRU isotopes to Am-241 or fission product gamma emitters to support gamma spectroscopic assay of low-density waste in 55-gallon drums.
- o Ratios of total isotopic to Cs-137 or other gamma emitters to support gross gamma assay of waste containers and gamma spectroscopic assay of containers.

Once the waste was characterized, current methodology and equipment were evaluated. Factors taken into consideration were sensitivity, range of applications, and cost.

A four-component system was developed based on this information.

- o High-resolution segmented gamma spectroscopy.

- o Passive neutron system.
- o Curie content from gross gamma analysis.
- o Estimating curie content of waste by sampling and radiochemical analysis.

## DEVELOPMENT OF METHODOLOGY IN USE AT THE WVDP

### Segmented Gamma Scanner

The Canberra Model 2200B\* segmented gamma scanner (SGS) purchased by WVDP was designed to detect gram quantities of plutonium and uranium by their gamma rays. The hardware specified including replacing the standard detector with a reverse electrode high-purity germanium detector, which gives improved (CANSCAN)<sup>2</sup> response at low energies, to allow detection of Am-241. The software included with the system was modified to determine microcurie quantities of Am-241, Cs-137, and Co-60 in the waste, and the system was calibrated using NBS traceable Am-241, Cs-137, and Co-60 in a simulated homogeneous source of known density. When the decision was made to implement 10 CFR 61, further modifications were required. The Yb-169 transmission source included with the SGS was replaced with a Eu-152 transmission source which has better attenuation correction lines for the three nuclides assayed and will last for the life of the Project. The software was also modified to report all the nuclides of interest in classification of the waste per 10 CFR 61 and an algorithm was developed which classifies the waste according to the criterion contained in 10 CFR 61. To assay drums with dose rates up to 35 mr/hr, an Mn-54 "dead-time" correction source was added.

### Passive Neutron Counting System

The passive neutron system received from Los Alamos National Laboratory is an approximately one Pi system consisting of two 48-inch x 48-inch x 13-inch He-3 detector arrays, a power supply, amplifier, and a reduced variance logic module. The system was calibrated for neutron detection efficiency using a Cf-252 source in a multipoint calibration and calculating the average efficiency. The neutron production per curie of typical WVDP waste was calculated based on chemical analysis values. Calculations from total neutron count to nanocuries per gram of waste have been written into the waste classification software developed for the SGS system.

### Dose Rate Measurement and Conversion to Total Curie Content

Estimating curie content has been recommended by the U. S. DEpartment of Energy<sup>3</sup>. The standard WVDP Waste container geometries, density values, and average gamma energy were reduced to a computer algorithm. This algorithm developed from data published by Bowman and Swindal<sup>4</sup> was used to write a program which takes dose rate, weight of container, and type of container, calculates nuclide concentrations, and classifies the waste. This program can also handle passive neutron system data for standard suspect TRU waste containers.

### Laboratory Analysis of Waste

The radiochemical laboratory and counting room at WVDP has been equipped to analyze for the nuclides present in WVDP waste which needs to be quantified

is waste for 10 CFR 61 classification with the exception of Tc-99. Standard methodology was verified as useful for the matrices encountered at the WVDP or modified as required.

All calibration sources used in the laboratory are NBS traceable or equivalent, and all instrumentation is verified to be operating properly before use.

## PERFORMANCE OF THE WASTE ASSAY SYSTEM

### Segmented Gamma Scanner

#### 1. Verification of Results

The SGS is calibrated using NBS traceable sources. Each day prior to counting samples, a source is counted to verify proper operation.

The SGS analysis values for four drums containing TRU waste with a total fissile content of 0.5 to 12.5 grams were compared with the analysis values of the passive neutron counter and found to be in good agreement.

The SGS analysis values for 100 drums of low-level plant waste for fission products were compared to the dose rate measurements with estimated curie content and found to be in good agreement.

#### 2. Lower Limit of Detection

The limit of detection for fission products is a factor of 1,000 lower than that required to demonstrate compliance with 10 CFR 61. For TRU isotopes, based on detection of Am-241 and a factor of 5.5 for TRU/Am-241 ratio, 0.6 nCi/g TRU is detectable in a 100-pound drum containing 356  $\mu$  Ci of Cs-137.

### Neutron Counting System

The neutron counting system was calibrated using a Cf-252 neutron source.

#### 1. Verification of Results

The neutron system performance was checked against radiochemical analysis values for floor debris from the XC-3 which was solidified and placed in four drums. The laboratory results are based on four analyses for each drum. Extremely small sample sizes were required. There were six samples taken from each of six areas of the cell with the highest in Am-241 used for the actual analysis, so that if sampling errors or in homogeneous material was present, the resulting estimate of fissile content would be higher than the actual fissile content to provide a safety margin. The analysis results were in good agreement (see Table 2) with the exception of one drum where there was a factor of 30 differences between the two of the aliquots taken from analysis. The value taken ensured a high estimate of fissile material content which is suggested by the neutron count estimates being lower than the radiochemical analysis estimates on drum TD 048.

#### 2. Limits of Detection

Assuming a 3,000-pound box of waste, an efficiency of 1.4 percent and

a 30-minute count time, the detection limit would be 100 nCi/g using total neutron counting. The only limit on mass is the load limit on the container being used to dispose of the waste.

#### Estimating Curie Content from Dose Rate

##### 1. Verification of Results

The dose rate for a drum containing ( $1.71 \text{ Ci/m}^3$ ) of Cs-137 was used to calculate the Cs-137 content of the drum. A value of  $1.79 \text{ Ci/m}^3$  was calculated by the computer program written to perform the waste classification verifying the algorithm. For further verification, 100 drums were chosen at random and were counted on the SGS for comparison assay results using the dose rate conversation method. In 199 percent of the drums counted, the agreement between the dose rate and SGS method was sufficiently accurate to assure good performance in accordance with the 10 CFR 61 criteria.

##### 2. Lower Limit of Detection

The LLD of the dose rate method varies with the background where the dose rate is taken and the weight of the container. In all cases, when container's dose rate measured in a low background area ( $< 0.5 \text{ mr/hr}$ ), the LLD is below the Class A limits.

#### D. Laboratory Analysis

The laboratory calibrations are all performed with NBS traceable or equivalent standards, and precautions are taken to ensure that accurate results are reported. In interlaboratory cross-checks performed to date have verified the accuracy of analysis produced by the WVNS Analytical Chemistry group.

#### COST OF THE SYSTEM

The SGS was purchased from Canberra Industries for approximately \$150,000. Approximately four work-months were required to modify the software for WVNS' particular need with one work-month for verification and calibration.

The neutron system was on loan from LANL at a cost of \$10,000 per year including technical support. Approximately two work-months have been required to calibrate and verify system operation.

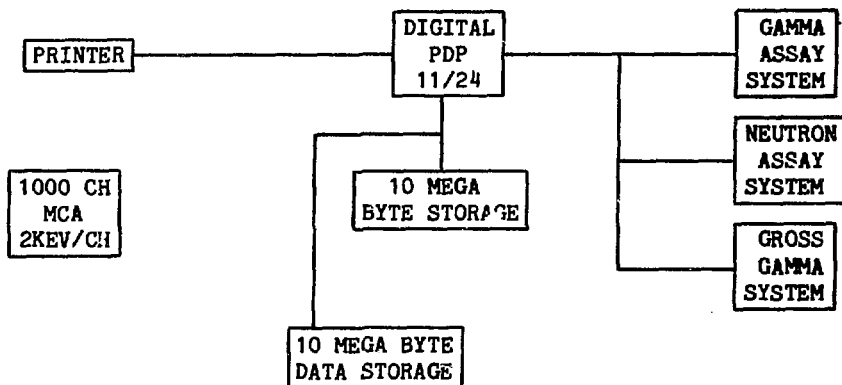
The dose rate to curie content required approximately one work-month of effort to program. The surveys are required of all waste packaged at WVNS so no additional cost is involved for assay of the waste.

Has the following advantages:

- o System has the sensitivity that exceeds requirements needed to meet 10 CFR 61.
- o Follows classification scheme outlined in 10 CFR 61.
- o High benefit to cost ratio.

- o Wide range of applications.
- o Not limited to container types.
- o Ease of operation.
- o Complete record-keeping system.
- o Data storage.

The overall waste assay system:



Block Diagram of Total System.  
Figure 1

Based on 1985 radiation waste disposal records, 12 work-weeks would be required to assay the 1,303 55-gallon drums that were disposed assuming a 15-minute count and 5-minute handling time per drum.

1985 radioactive waste disposal records show that 223 waste boxes were buried. If it is assumed that all waste boxes were TRU assayed using the neutron system, it would require a total of four work weeks to complete the assays assuming a 30-minute counting time and 10-minute handling time per box.

To determine the gross gamma of 50 containers, it would require five work-hours assuming 5 minutes to generate data per drum. Since waste surveys are currently required prior to removing waste from the work area, the only extra labor effort is that each container must be weighed.

In summary, the waste assay system developed. It meets and surpasses the requirement of 10 CFR 61. This system observes the ability to function in any plant facility. It is a highly automated system. Due to its broad range of applications and with the proposed further development, it will become a valuable commodity considering the upcoming waste classification needs required by the 10 CFR 61 implementation. An integrated system to classify reactor waste has been developed by Westinghouse. This system would in five minutes classify a drum of waste in compliance with the radiological requirements of 10 CFR 61.

## References

1. Licensing Requirements for Land Disposal of Radioactive Waste, 10 CFR 61, U. S. Nuclear Regulatory Commission, Washington, DC, December 27, 1982.
2. CANSCAN, Version 4, Canberra Industries, Inc., Meriden CT, Copyright 1979.
3. Radioactive Waste Management, DOE 5820.2, U. S. Department of Energy, Washington, DC, dated February 6, 1984.
4. Determination of Curie Content of Packaged Radioactive Waste Using Measured Dose Rates, Health Physics, Pergamon Press, 1976, Volume 3 (November), pp. 445-450.

**A COMPARISON OF THE RESPONSE OF A NAI SCINTILLATION  
CRYSTAL WITH A PRESSURIZED IONIZATION CHAMBER  
AS A FUNCTION OF ALTITUDE, RADIATION LEVEL, AND  
RA-226 CONCENTRATION**

Richard Provencher, Gregory Smith, T. B. Borak and P. Kearney\*  
Department of Radiology and Radiation Biology  
Department of Physics\*  
Colorado State University  
Fort Collins, Colorado 80523

**ABSTRACT**

The Grand Junction Uranium Mill Tailings Remedial Action-Radiological Survey Activities Group (UMTRA-RASA) program employs a screening method in which external exposure rates are used to determine if a property contaminated with uranium mill tailings is eligible for remedial action. Portable NaI detectors are used by survey technicians to locate contaminated areas and determine exposure rates. The exposure rate is calculated using a regression equation derived from paired measurements made with a pressurized ionization chamber (PIC) and a NaI detector. During July of 1985 extensive measurements were taken using a PIC and a NaI scintillator with both analogue and digital readout for a wide range of exposure rates and at a variety of elevations. The surface soil was sampled at most of these locations and analyzed for  $^{226}\text{Ra}$ . The response of the NaI detectors was shown to be highly correlated to radiation level but not to  $^{226}\text{Ra}$  concentration or elevation.

**INTRODUCTION**

The Uranium Mill Tailings Remedial Action Program (UMTRAP) is responsible for decontamination of inactive uranium mill sites and associated "vicinity properties". Vicinity properties include residences, schools, parks, motels and other public and commercial structures (1). These properties were contaminated with waste uranium mill tailings primarily through the use of tailings as a substitute for sand in construction projects.

The Department of Energy (DOE) was charged with the performance of the UMTRA Program. It identified over 8200 candidate properties (designated properties) of which about 7000 are located in Grand Junction, CO (1).

One of the steps in the remedial action process is a radiological survey of each designated property to determine eligibility for decontamination by UMTRA. These surveys are conducted by Oak Ridge National Laboratory (ORNL) which is designated as the Inclusion Survey Contractor (ISC).

The Pressurized Ionization Chamber (PIC) is considered to be the standard instrument for environmental exposure rate measurements. However, because of its size, weight and cost it is not practical as a portable survey meter. Therefore, the gamma surveys are performed primarily with portable instruments consisting of a NaI scintillator with an analogue ratemeter (scintillometers). The scintillometers are relatively inexpensive, portable, and have a high sensitivity with a rapid response. However, they measure ionizing events in the crystal, not exposure.

Readings made with the scintillometer, in kilo Counts Per Minute (kCPM), are transformed to exposure rates ( $\mu\text{R/h}$ ) using an equation developed through a regression analysis of paired pressurized ionization chamber and NaI scintillometer measurements (2).

It is imperative to understand the response of the portable scintillators as a function of exposure rate. Since both detectors are subjected to the combination of terrestrial and atmospheric radiation, this study was designed to observe the response of each as a function of elevation and concentration of  $^{226}\text{Ra}$  in soil.

## METHODS

Exposure rates were measured using a Reuter Stokes RSS-111 pressurized ionization chamber modified with an integrated display in units of  $\mu\text{R/h}$  (3, 2). The calibration of the PIC was verified at the beginning of the project using two sources of  $^{226}\text{Ra}$  in a procedure recommended by the manufacturer, and field checked daily using a  $^{60}\text{Co}$  source.

A NaI scintillator (3.2 cm dia.; 3.8 cm l) with an analogue rate meter was tested. This system was identical to the configuration used by the ISC survey teams. Estimates of count rate were made by averaging the meter reading over a timing interval of approximately 10 s.

A second NaI scintillator (2.5 cm dia; 2.5 cm l) with a digital scaler was also compared to the response of the PIC. The count rate was determined by integrating over a 60 s time interval.

Each of the NaI scintillometers was field checked daily using a depleted uranium source. All measurements were taken at ground level since the screening criteria for vicinity properties are written for ground level measurements (2).

Data collection was performed in several steps. In order to determine the response of the NaI crystals under background conditions, the city and surrounding areas of Grand Junction were divided into a 2.56 km<sup>2</sup> grid. One hundred and twenty-five locations were measured on this grid. In order to obtain the response at higher levels of radiation, twenty-five measurements were made in the vicinity of the uranium mill tailings pile. A series of measurements were taken from 1500 m to 3300 m along Grand Mesa to determine the response of each detector as a function of elevation.

At each site the top 15 cm of soil was removed using a post hole digger. Approximately 350 g of each sample was dried and analyzed for  $^{226}\text{Ra}$  concentration. This analysis was performed using a NaI spectrometer and a computer algorithm developed by ORNL (2). Response as a function of  $^{226}\text{Ra}$  concentration was then determined.

## RESULTS

Figure 1 shows the paired readings of the PIC ( $\mu\text{R/h}$ ) and the NaI scintillometer (kCPM). The data was plotted on a log-log scale because of the dynamic range of the readings. The solid curve illustrates the result of a regression analysis. The data were separated into two sections in order to simplify the equations.

From 0-50 kCPM the response of the NaI system was a linear function of exposure rate. At higher readings a power function was required. The equations obtained from the regression analysis are:

$$\text{Exposure Rate } (\mu\text{R/h}) = (\text{kCPM}) + 8.3 \quad \text{kCPM} \leq 50 \quad (1)$$

$$\text{Exposure Rate } (\mu\text{R/h}) = 0.53 (\text{kCPM})^{1.2} \quad \text{kCPM} > 50 \quad (2)$$

The  $R^2$  for equation (1) was 95 and that for equation (2) was 99. This combination provided a satisfactory fit to the data but as can be seen from Figure 1, there is a tendency for the equation to overestimate the true values at low count rates. Generally, this distribution of residuals is unacceptable. However, for screening purposes the equations seem to be adequate. The dashed lines in Figure 1 show an interval of  $\pm 20\%$  about the regression lines. It is clear that over 95% of the true readings fall within this interval.

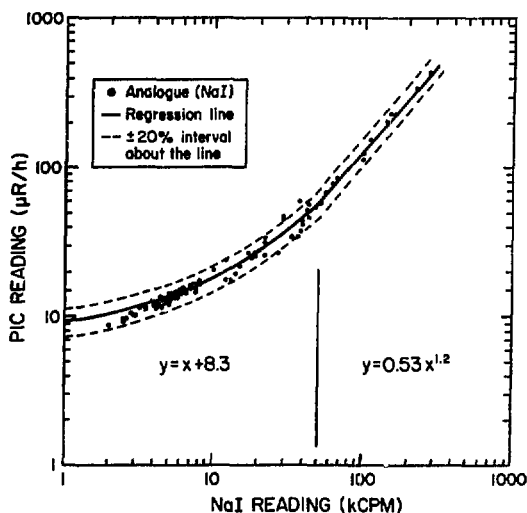


Figure 1. Plot of paired readings for the pressurized ionization chamber and the NaI scintillometer with analogue display showing the two regression equations that intersect at 50 kCPM.

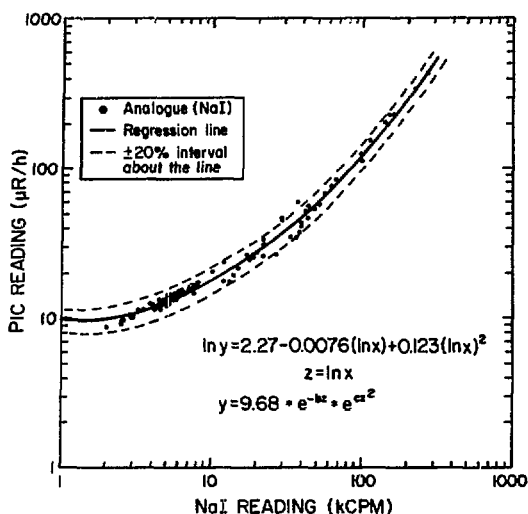


Figure 2. Plot of paired readings for the pressurized ionization chamber and the NaI scintillometer with analogue display with a quadratic function over all exposure rates measured.

Another regression analysis of the data was made using a quadratic function on the log-log scale. This model yields the desired normal distribution of residuals. This curve is shown in Figure 2. The equation describing this curve is:

$$\ln(y) = 2.27 - 0.0076(\ln x) + 0.123(\ln x)^2 \quad (3)$$

Thus,

$$\text{Exposure Rate } (\mu\text{R/h}) = 9.7 (\text{kCPM})^{-0.0076} \{(\text{kCPM})^{0.123}\} \ln(\text{kCPM}) \quad (4)$$

Although this function is statistically superior based on the distribution of residuals, it is quite cumbersome to use and does not vastly improve the confidence interval based on  $\pm 20\%$  variation as shown in Figure 2.

Figure 3 shows the response of the NaI detector for low values of exposure rate. There is an apparent threshold which indicates that the NaI scintillator does not respond until the exposure rate is greater than 8  $\mu\text{R/h}$ .

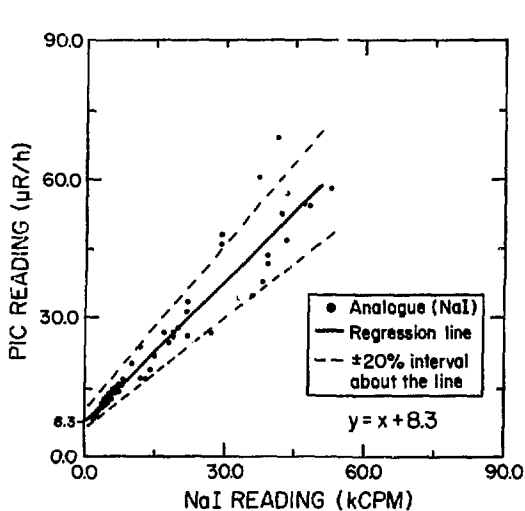


Figure 3. Plot of paired readings for the pressurized ionization chamber and NaI scintillometer with analogue display at low exposure rates which shows a threshold at 8.3  $\mu\text{R/h}$ .

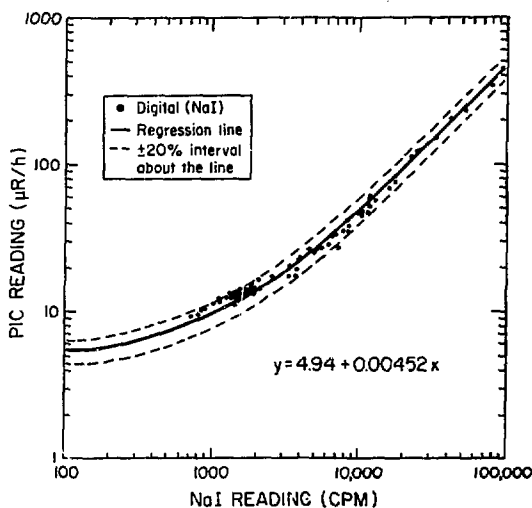


Figure 4. Plot of paired readings for the pressurized ionization chamber and the NaI scintillometer with digital display showing a linear fit over all exposure rates measured.

Figure 4 shows the response of the NaI detector with a digital scaler. In this situation a single linear expression was sufficient to fit the data:

$$\text{Exposure Rate } (\mu\text{R/h}) = 5 + 0.0045 (\text{CPM}) \quad (5)$$

$$\text{Exposure Rate } (\mu\text{R/h}) = 5 + 0.45 (\text{kCPM}) \quad (6)$$

Notice that there is a threshold exposure rate of 5  $\mu\text{R/h}$  for the scintillometer with a digital scaler.

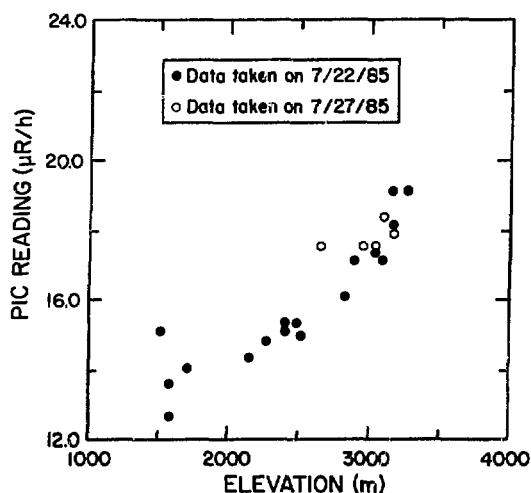


Figure 5. The response of the pressurized ionization chamber vs. elevation between 1500 m and 3400 m along Grand Mesa, CO.

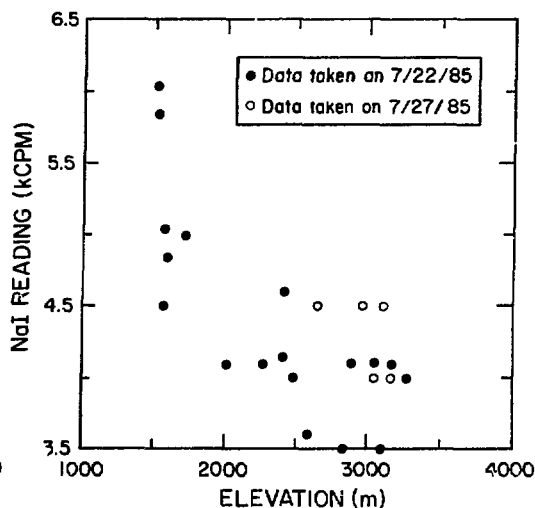


Figure 6. The response of the NaI scintillometer vs. elevation between 1500 m and 3400 m along Grand Mesa, CO.

Figure 5 shows the response of the PIC as a function of elevation. There is a clear correlation between exposure rate and elevation. Figure 6 shows the response of the NaI detector as a function of elevation. There does not appear to be any correlation between the count rate and elevation. Figure 7 shows the paired readings of the PIC and NaI detector for the data previously shown in Figures 5 and 6. Notice that for the data taken at higher elevations the NaI scintillator underresponds when compared to the regression equation obtained at lower elevations.

Figure 8 is a plot of the concentration of  $^{226}\text{Ra}$  in surface soil as a function of exposure rate measured by the PIC. Figure 9 is a plot of  $^{226}\text{Ra}$  concentration as a function of the NaI reading. Neither of these show any apparent correlation. Notice that  $^{226}\text{Ra}$  concentrations at or below the minimum detectable concentration are observed over the entire range of exposure rates and NaI readings.

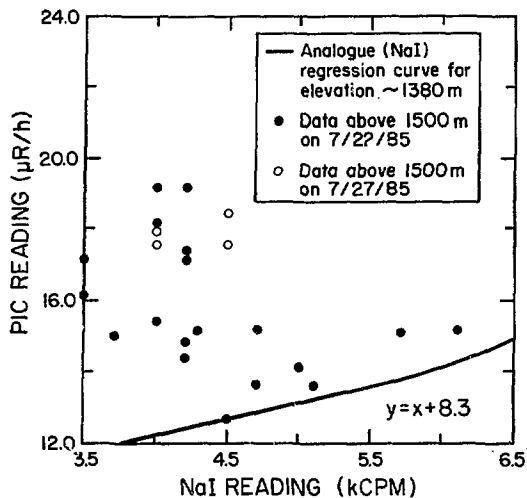


Figure 7. Plot of paired readings taken above 1500 m elevation for the pressurized ionization chamber and the NaI scintillometer. The solid line is the equation obtained from the regression analysis shown in Figure 1.

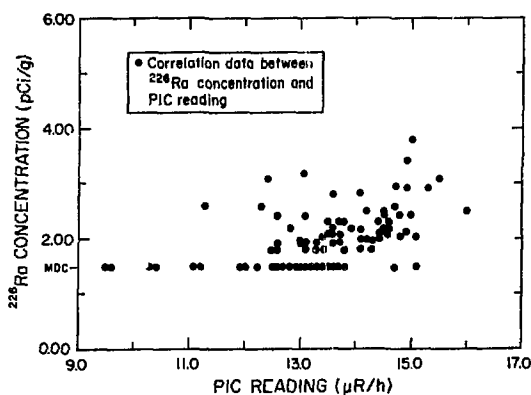


Figure 8. Plot of  $^{226}\text{Ra}$  concentration in soil as a function of pressurized ionization chamber reading. The minimum detectable concentration was 1.5 pCi/g.

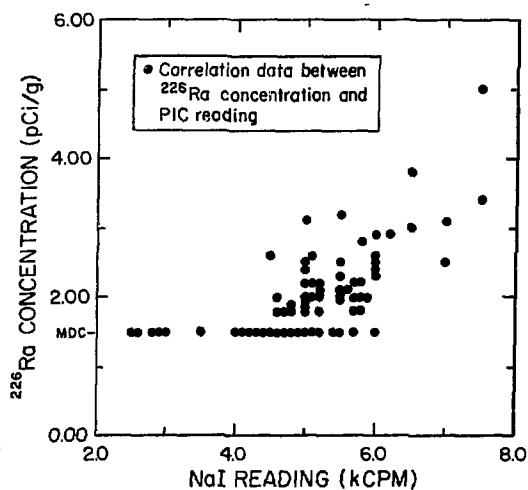


Figure 9. Plot of  $^{226}\text{Ra}$  concentration in soil as a function of NaI scintillometer reading. The minimum detectable concentration was 1.5 pCi/g.

## CONCLUSIONS

At elevations below 1500 m the count rate of portable NaI survey meters was correlated to the external exposure rate measured with a pressurized ionization chamber (PIC). The response was linear for low count rates, but a power function was required at higher count rates when measured with an analogue meter. Both digital and analogue scintillometers displayed thresholds in the sense that they did not respond to exposure rates measured with the PIC below 5  $\mu\text{R/h}$  and 8  $\mu\text{R/hr}$ , respectively. It is suggested that this corresponds to missing pulses generated by atmospheric radiation which are not registered by the rate meters.

The response of the PIC increased with elevation. However, there was no correlation between the NaI detectors and elevation. The response of the NaI scintillometers was probably due to random changes in the terrestrial component. The correlation between the PIC and elevation was preserved even when the output of the NaI was subtracted from the PIC. This is further evidence that the NaI detectors did not respond to atmospheric radiation.

Attempts to correlate the response of the PIC and NaI detectors with the concentration of  $^{226}\text{Ra}$  in soil directly below the point of measurement were entirely unsuccessful.

The insensitivity of the portable NaI scintillometers to atmospheric radiation might be explained by considering the stopping power of high energy muons and the path length distribution through the crystal. The stopping power for minimum ionizing particles in NaI is 4.8 MeV/cm (5). Using Cauchy's theorem, the mean path length through the 2.5 cm  $\times$  2.5 cm cylinder is 2/3 of the diameter which is 1.7 cm. This corresponds to an energy deposition greater than 8 MeV. It is not clear how the amplifier and discriminators process such large signals. For this cylindrical geometry only about 2% of the events would be less than 2 MeV which is in the range of those produced by terrestrial gamma rays (6).

In summary, NaI scintillometers can be used for rapid screening of external exposure rates. Caution must be exercised when large variations in elevation are anticipated. The measured exposure rates were not correlated to the concentration of  $^{226}\text{Ra}$  in soil taken directly below the point of measurement.

## ACKNOWLEDGMENT

Work supported by the Radiological Survey Activities Group, Health and Safety Division, Oak Ridge National Laboratory.

## REFERENCES

1. U. S. Department of Energy, "Vicinity Properties Management and Implementation Manual," Uranium Mill Tailings Remedial Action Program, DOE/Albuquerque, Albuquerque, NM, 1984.
2. Oak Ridge National Laboratories, Health and Safety Research Division, "RASA/UMTRA Procedures Manual (Draft)," Oak Ridge National Laboratory, Oak Ridge, TN, 1984.
3. Reuter Stokes, Operational Manual, "RSS-111, Area Monitoring System," Reuter Stokes Instruments, Inc., 2nd Printing, 1983.
4. P. D. Kearney, N. D. Kretz, and P. W. Zelle, "Gamma Survey Measurement Techniques at UMTRAP Sites," Proceedings from the Symposium on Uranium Mill Tailings Management, Colorado State University, Fort Collins, CO, pp. 97-113, 1981.
5. D. Pines, Reviews of Modern Physics, The American Physical Society, New York, Volume 56, Number 2, pg. 553, 1984.
6. S. E. Merwin, "An Analytical Model of the Response of NaI(Tl) Scintillation Spectrometers to Environmental Gamma Radiation," Master's Research Paper, Colorado State University, Fort Collins, CO, 1985.

USING A HPGe DETECTOR AND MICRO R METER FOR QUICK AND  
SIMPLE ESTIMATE OF RADIONUCLIDE ACTIVITY IN  
LSA SHIPPING CONTAINERS

M. J. Robinet, A. L. Justus, and R. A. Wynveen  
Argonne National Laboratory  
9700 South Cass Avenue  
Argonne, Illinois 60439

ABSTRACT

Shippers of low specific activity (LSA) waste are required by both waste receivers and Department of Transportation (DOT) regulations to provide an estimate of radionuclide activity in each container. In our work, a high resolution HPGe detector was used to measure the  $^{226}\text{Ra}$  and daughters gamma flux from 90 ft<sup>3</sup> LSA waste containers. The detector was calibrated with a radium point source of known activity. To calculate the radium activity in a container, the gamma flux reaching the detector from successive slices of the container was numerically integrated over its dimensions. The counting time for each container initially required several hours. However, by constructing a graph of container radium activity versus average exposure rate ( $\mu\text{R/h}$ ) at container surfaces, a measurement could be made in a few minutes and the activity read from the graph. Results are estimated to be accurate to at least an order of magnitude. Where hundreds of containers are involved, the time and cost saved by this technique can be significant.

INTRODUCTION

The total  $^{226}\text{Ra}$  activity in 90 ft<sup>3</sup> boxes containing low specific activity (LSA) waste was estimated by measuring the count rate with a collimated HPGe detector placed 10 ft from each of four sides of a box. A high resolution HPGe detector was used so that radionuclides could be identified and quantified by discrete photon energies. For measuring the radium activity in LSA boxes, it was established that  $^{226}\text{Ra}$  was in equilibrium with  $^{214}\text{Pb}$  and  $^{214}\text{Bi}$ . The gamma rays counted were the 186.211 keV line from  $^{226}\text{Ra}$ , the 351.92 keV line from  $^{214}\text{Pb}$ , and the 609.31 keV, the 1120.29 keV, and 1764.494 keV lines of  $^{214}\text{Bi}$ .

This approach appears simple and somewhat elegant, but it required at least two hours to count each side of a box. For the situation encountered, there simply was not enough time to count more than a hundred boxes which had to be shipped within a few weeks. Fortunately, it was determined through experimentation that there was reasonable correlation between the measured average activity in boxes and the average exposure rate ( $\mu\text{R/h}$ ) measured at the surface of the sides of the boxes. A graph of container radium activity versus average exposure rate at the container surface was hence constructed for use by field personnel. By measuring the average exposure rate at the center of five sides (average of readings taken at front, back, right, left,

and top) the corresponding approximate activity in a box could be read from the graph. The total time required was usually less than ten minutes per box. The micro R meter used (Eberline PRM-7) contained a 25 mm diameter by 2.5 mm long sodium iodide detector and was exposure-rate-calibrated with radium.

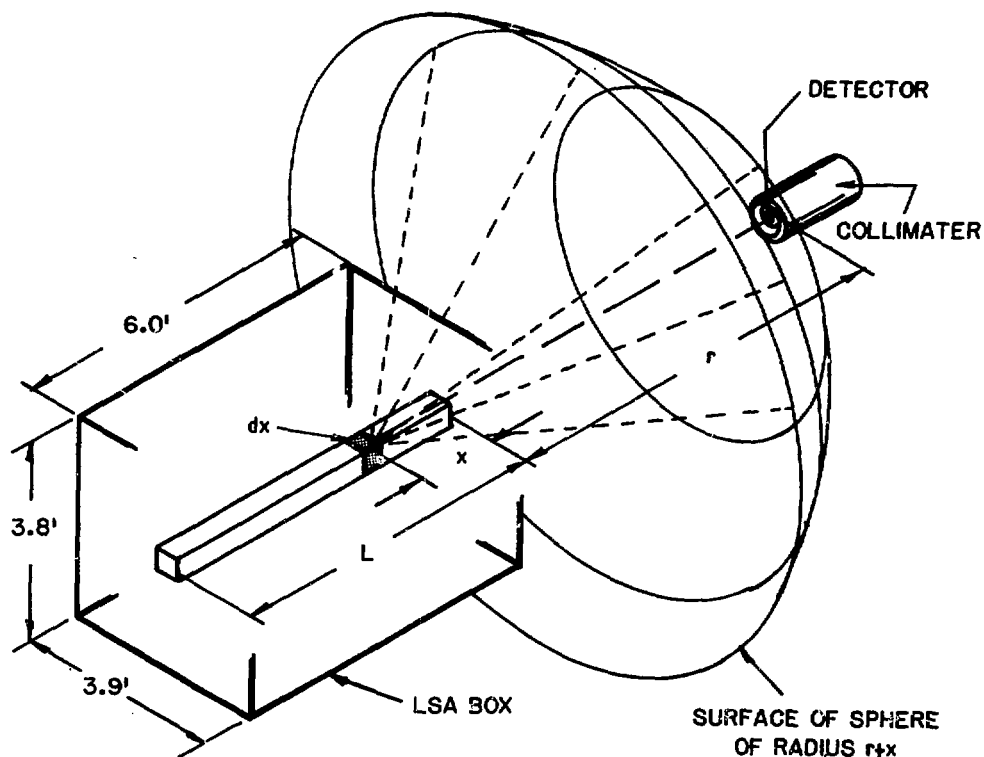


Figure 1 Schematic Representation of Small Rectangular Source Element in LSA Box.

## THEORY

The photon emission rate from an infinitely small rectangular slice of the material in a box (see Figure 1) can be expressed as

$$dS = Y_i dA \quad (1)$$

where

$dS =$  photon emission rate (photon/sec) from activity  $dA$  in the differential slice  $dx$  for photon of energy  $i$ .

$Y_i$  = photon yield (gammas/disintegration), the number of photons of energy  $i$  emitted per transformation of the nuclide of interest.

$dA$  = activity (dis/sec) in the slice  $dx$ .

If we assume that the activity in a box is uniformly distributed, the activity per unit length is  $A/L$  and the activity in the slice  $dx$  is

$$dA = \frac{A}{L} dx \quad (2)$$

where

$A$  = total activity in box (dis/sec or  $\mu\text{Ci}$ )

$L$  = length of the rectangular element, i.e., of the box (cm).

Then, equation (1) can be written as

$$dS = Y_i dA = Y_i \frac{A}{L} dx \quad (3)$$

The photon fluence rate (photon/cm<sup>2</sup>-sec) seen by the detector at a distance  $r + x$  from the slice  $dx$  is

$$d\phi = \frac{dS}{4\pi(r+x)^2} e^{-\mu_i x} \quad (4)$$

which gives upon substitution for  $dS$

$$d\phi = Y_i \frac{A}{L} \frac{dx}{4\pi(r+x)^2} e^{-\mu_i x} \quad (5)$$

where

$d\phi$  = fluence rate (photons/cm<sup>2</sup> sec) at detector on the imaginary surface of a sphere of radius  $r + x$ .

$r$  = distance from detector to surface of box.

$x$  = layer of material in front of slice  $dx$ .

$e^{-\mu_i x}$  = fraction of photons removed by layer of material  $x$ .

$\mu_i$  = effective attenuation coefficient (cm<sup>-1</sup>) of the material at photon energy  $i$ .

The count rate  $dI$  from the detector (see Figure 1) due to fluence rate  $d\phi$  is

$$dI_i = k_i d\phi$$

which upon substitution for  $d\phi$ , gives

$$dI_i = k_i Y_i \frac{A}{L} \frac{dx e^{-\mu_i x}}{4\pi (r+x)^2} \quad (6)$$

where

$dI_i$  = count rate (counts/sec) from a detector which sees a fluence rate  $d\phi$  of photons of energy  $i$ .

$k_i$  = detector efficiency in terms of counts/sec per unit fluence rate (counts/sec per photon/cm<sup>2</sup>.sec).

## CALIBRATION

The value of  $k_i$  was determined by measuring the count rate from a "point source" of known activity (677  $\mu$ Ci of <sup>226</sup>Ra) at various distances  $r$  from the detector. This calibration constant can be written as

$$k_i = \frac{I_{cal_i}}{\phi_{cal_i}}$$

$$k_i = \frac{I_{cal_i}}{\frac{A_{cal} Y_i}{4\pi r^2}} = k_i = \frac{4\pi r^2 I_{cal_i}}{A_{cal_i} Y_i} \quad (7)$$

Let  $\frac{I_{cal_i}}{A_{cal_i}} = \varepsilon_i$  so

$$k_i = 4\pi r^2 \frac{\varepsilon_i}{Y_i} \quad (8)$$

where

$A_{cal_i}$  = disintegration rate of calibration source (dis/sec or  $\mu$ Ci)

$I_{cal_i}$  = count rate due to photon of energy  $i$  from the calibration source

$\varepsilon_i =$  detector efficiency in terms of count rate per unit activity for photons of energy  $i$  (counts/sec per dis/sec or counts/minute per  $\mu\text{Ci}$ ).

$\phi_{\text{cal } i} =$  fluence rate (photon/cm<sup>2</sup>.sec) seen at detector at distance  $r$  from the calibration source.

Equation (6) can then be rewritten as

$$dI_i = 4\pi r^2 \frac{\varepsilon_i}{Y_i} Y_i \frac{A}{L} \frac{dx e^{-\mu_i x}}{4\pi (r+x)^2}$$

$$dI_i = \frac{A\varepsilon_i r^2}{L} \frac{dx e^{-\mu_i x}}{(r+x)^2} \quad (9)$$

The count rate from the entire length of the rectangular element is

$$I_i = \frac{A\varepsilon_i r^2}{L} \int_{x=0}^{x=L} \frac{e^{-\mu_i x} dx}{(r+x)^2} \quad (10)$$

This simple equation can be applied to the entire box by using an average counting efficiency which accounts for the decrease in count rate from off-axis positions across the width of the box. The spacial averaged efficiency  $\bar{\varepsilon}_i$  was determined by computing the average of the efficiencies measured at several different off-axis positions (see Figure 2).

The measurements of efficiencies at off-axis position for five different gamma rays are plotted in Figure 2. Using this approximation, the count rate due to the activity in an entire box can be written as

$$\bar{I}_i = \frac{A\bar{\varepsilon}_i r^2}{L} \int_{x=0}^{x=L} \frac{e^{-\mu_i x} dx}{(r+x)^2} \quad (11)$$

where

$\bar{I}_i =$  count rate of photons of energy  $i$  due to total activity  $A$  in a given size box.

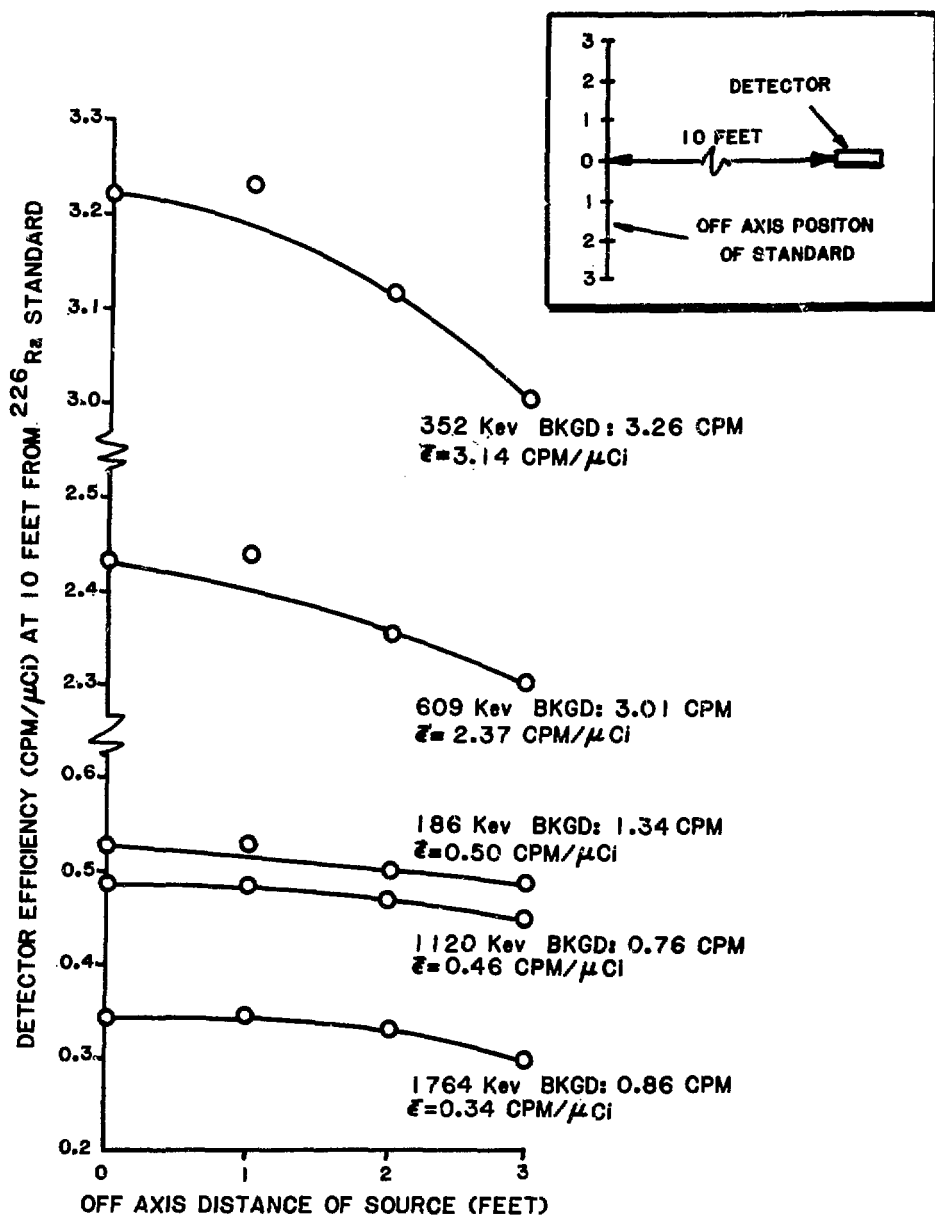


Figure 2. HpGe Detector Efficiency Versus Off Axis Position of  $^{226}\text{Ra}$  Standard (677  $\mu\text{Ci}$ ).

$\bar{\epsilon}_i =$  detector counting efficiency (counts/minute per  $\mu\text{Ci}$ ) averaged over the given maximum width of a box.

The linear attenuation coefficient  $\mu$  was computed from the product of the average value of the mass attenuation coefficient ( $\text{cm}^2/\text{g}$ ) and the effective density of the material in a box. That is,

$$\mu_i = \left( \frac{\mu}{\rho} \right)_i \rho_{\text{eff}} = \frac{\mu}{\rho_i} \frac{m}{V} \quad (12)$$

where

$\mu_i =$  linear attenuation coefficient ( $\text{cm}^{-1}$ ) of material in box for photon of energy  $i$ .

$\rho_{\text{eff}} =$  effective density of material in box ( $\text{g}/\text{cm}^3$ ).

$m =$  mass of material in box (gram).

$V =$  volume of box ( $2.66 \times 10^6 \text{ cm}^3$ ).

$\left( \frac{\mu}{\rho} \right)_i =$  mass attenuation coefficient ( $\text{cm}^2/\text{g}$ ) of material in box for photon of energy  $i$ .

The mass attenuation coefficients for a wide variety of materials (reference 1) are about the same at each photon energy of interest. The minimum and maximum values at five different energies are listed in Table 1. The average of the minimum and maximum values were used to calculate activity A in boxes. The attenuation of the wall of the box was not included in equation (11) because it would imply a degree of accuracy which did not exist.

Table 1  
Mass Attenuation Coefficients  $\mu/\rho$  ( $\text{cm}^2/\text{g}$ )<sup>a</sup>

Photon Energy (KeV)	Min $\mu/\rho$	Max $\mu/\rho$	Assumed <sup>b</sup> Average $\mu/\rho$
186	0.13	0.17	0.15
352	0.10	0.11	0.11
609	0.071	0.085	0.08
1120	0.051	0.065	0.06
1764	0.041	0.054	0.05

<sup>b</sup>Values used to calculate activity in box.

<sup>a</sup>Minimum and maximum values taken from graphs on pages 447 and 448 of reference 1.

## SOLUTION OF EQUATION

Unfortunately, there is no exact analytical solution for equation (11). Therefore, solutions were sought involving numerical methods of integration and the use of the  $E_2(b)$  exponential integral graph on page 372 of reference 1. A simple numerical approximation can be obtained by the use of Simpson's rule:

$$\int_{x_0}^{x_n} f(x)dx = \frac{h}{3} (f_0 + 4f_1 + 2f_2 + 4f_3 + 2f_4 + \dots + 2f_{n-2} + 4f_{n-1} + f_n) \quad (14)$$

where

$$x_n = x_0 + nh, \quad f_n = f(x_n)$$

$$h = \frac{x_n - x_0}{n}$$

$$n = \text{number of subintervals} = 2, 4, 6, 8 \dots$$

In applying Simpson's rule to equation (11), it was found, to several significant figures, that the use of 10 or 20 subintervals would yield the same results as the use of 99 subintervals. In fact, the use of 4 subintervals was sufficient to two significant digits. Moreover, the same solution to equation (11) was confirmed through use of exponential integrals (reference 1).

Applying Simpson's rule to equation (11) for  $n = 4$ ,

$$\bar{I}_i \approx \frac{A\bar{\epsilon}_i r^2}{L} \frac{h}{3} (f_0 + 4f_1 + 2f_2 + 4f_3 + f_4) \quad (15)$$

where

$$h = \frac{x_n - x_0}{n} = \frac{L-0}{4}$$

## SAMPLE CALCULATION

Consider a box of waste with the following parameter and associated data:

$$A = ?$$

$$\bar{\epsilon}_i = 2.37 \text{ counts/minute per } \mu\text{Ci}$$

$$\bar{I}_i = 43.44 \text{ counts/minute at } 609 \text{ keV}$$

$$r = 10 \text{ ft} = 305 \text{ cm}$$

$$L = 6 \text{ ft} = 117 \text{ cm}$$

$$\text{Then, } h = \frac{L-0}{4} = 29.25 \text{ cm and } f = e^{-\mu x_n / (r + x_n)^2}$$

$$\begin{array}{ll} x_0 = 0 & f_0 = e^{-\mu x_0 / (r + x_0)^2} = 1.08\text{E-}5 \\ x_1 = h = 29.25 & f_1 = e^{-\mu x_1 / (r + x_1)^2} = 5.18\text{E-}7 \\ x_2 = 2h = 58.5 & f_2 = e^{-\mu x_2 / (r + x_2)^2} = 2.58\text{E-}8 \\ x_3 = 3h = 87.75 & f_3 = e^{-\mu x_3 / (r + x_3)^2} = 1.31\text{E-}9 \\ x_4 = 4h = 117 & f_4 = e^{-\mu x_4 / (r + x_4)^2} = 6.63\text{E-}11 \end{array}$$

and finally

$$\bar{I}_i \approx \frac{A \bar{\epsilon}_i r^2}{L} \frac{29.25}{3} (1.29 \times 10^{-5}) \text{ and } A \approx 183 \text{ } \mu\text{Ci}$$

#### DATA, AND RESULTS

Samples of the measurements made on two of the four analyzed LSA boxes are given in Table 2. The graph of activity versus  $\mu R/h$  at box surface is given in Figure 3. The calculation was done with a personal computer, but could also be done with a programmable calculator.

Photographs of the equipment during use are shown in Figures 4 through 7.

Table 2.  
Samples of Activity and  $\mu R/h$  Data From LSA Boxes

LSA Box #3		Weight = 2000 Pounds Density = 0.36 g/cm <sup>3</sup>				
Box Side	186.211 keV	$\mu\text{Ci Measured at Listed Energies}^a$				Net <sup>b</sup> $\mu R/h$
		351.92 keV	609.31 keV	1120.29 keV	1764.494 keV	
Front	41.73	32.96	38.25	31.83	38.16	10
Back	21.38	33.12	38.90	39.13	31.37	8
Right	88.32	27.05	35.48	34.97	38.10	7
Left	--	25.65	33.60	39.95	32.56	5
Top	--	--	--	--	--	5
Average	--	29.64	36.56	36.47	35.05	7

LSA Box #18

Weight = 6800 Pounds

Density = 1.21 g/cm<sup>3</sup>

Box Side	<u>µCi Measured at Listed Energies<sup>a</sup></u>					Net <sup>b</sup> µR/h
	186.211 keV	351.92 keV	609.211 keV	1120.29 keV	1764.494 keV	
Front	134.03	157.50	175.92	215.67	162.77	92
Back	--	144.12	152.30	173.92	143.80	92
Right	--	170.35	205.29	244.24	252.65	82
Left	--	144.75	162.40	203.77	198.46	62
Top	--	--	--	--	--	82
Average	--	154.18	173.98	209.40	189.27	82

<sup>a</sup>Two sigma counting errors were less than 10%.

<sup>b</sup>Measured at the surface of the center of each side. Typical background was 8 µR/h.

## CONCLUSION

The activity of gamma emitting nuclides in LSA waste containers can be quickly estimated after the containers are sealed by obtaining sufficient data with an HPGe detector to construct a graph of activity versus exposure rate. Since the exposure rate could be the same from very different mixtures of gamma emitting nuclides, this approach is valid only if the ratios of gamma emitters are the same in all boxes. Radium was a simple case because the exposure rate from all of the boxes was known to be due only to radium and its daughter products. The degree of equilibrium between radium and its daughter products was assumed to be the same in all boxes. The data show that, with an HPGe detector system, one can measure the activity of each individual gamma emitting radium daughter in a box. This approach could be applied to mixtures of other gamma emitting radionuclides if it could be established or assumed that the radionuclide activity ratios were about the same in all containers.

## REFERENCE

1. T. Rockwell III, Editor, Reactor Shielding Design Manual, Naval Reactors Branch, Division of Reactor Development Report TID-7004, National Technical Information Service, Springfield, VA. 1956.

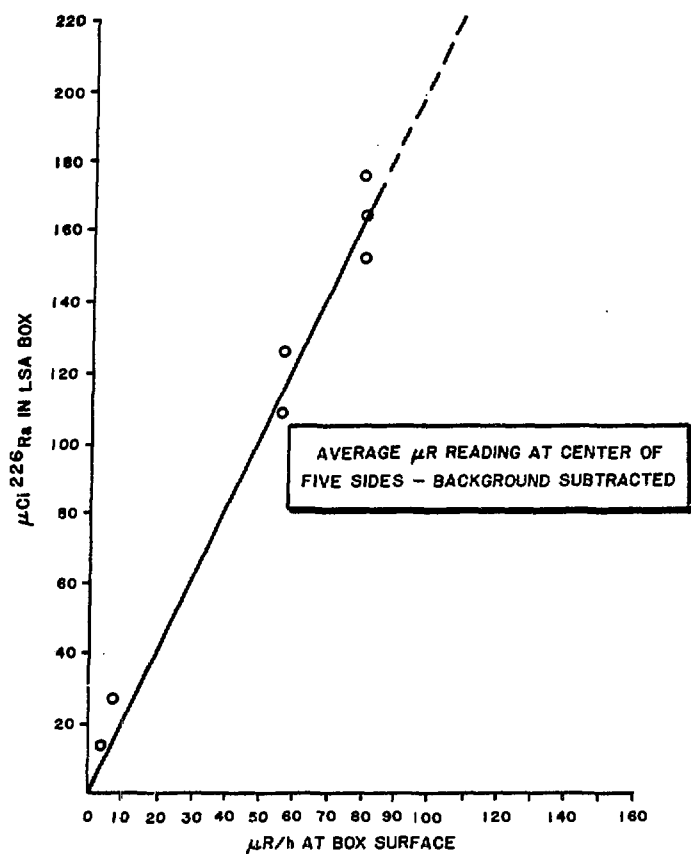


Figure 3.  $\mu Ci$   $^{226}Ra$  in 90 cu ft LSA Box Versus  $\mu R/h$  at Surface of Box.



Figure 4. Calibration Source With Collimated HPGe Detector in Background.

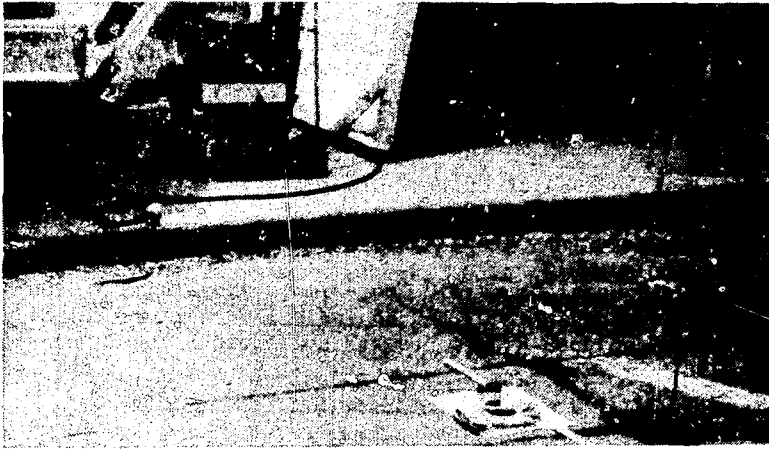


Figure 5. Counting Calibration Source.



Figure 6. Placing LSA Box in Counting Position.  
Multichannel Analyzer is in Vehicle on Left.



Figure 7. Measuring Exposure Rate at Box Surface  
With  $\mu\text{R}$  Meter.

## **RAPID ESTIMATION OF $^{226}\text{Ra}$ IN SOIL FOR THE GRAND JUNCTION RASA/UMTRA PROJECT**

**J. B. Kark, T. B. Borak, P. D. Kearney\***  
Department of Radiology and Radiation Biology  
Department of Physics\*  
Colorado State University  
Fort Collins, Colorado 80523

**A. S. Rood**  
Oak Ridge National Laboratory  
Grand Junction Office  
P.O. Box 2567  
Grand Junction, Colorado 81502

### **ABSTRACT**

The Radiological Survey Activities (RASA) Group of the Health and Safety Research Division at Oak Ridge National Laboratory (ORNL) is an Inclusion Survey Contractor (ISC) for the Uranium Mill Tailings Remedial Action Program (UMTRAP). The purpose of the ISC is to survey designated sites potentially contaminated with radioactive material originating from the 24 inactive uranium mill sites and make recommendations as to whether the site should be included in or excluded from further consideration by UMTRAP. An important aspect of the program is a prompt and inexpensive estimation of Radium-226 ( $^{226}\text{Ra}$ ) concentration in soil samples. A large sodium iodide (NaI) well crystal coupled to a multichannel analyzer is used to count soil samples. Count data are currently analyzed with an algorithm that utilizes three regions of interest (ROI). A lack of agreement was observed when samples were also analyzed with lithium-drifted germanium (GeLi) spectrometers. The average estimate of  $^{226}\text{Ra}$  obtained using the current algorithm was 19% greater than the GeLi determination. Some possible reasons for these differences were examined. In 8.5% of the samples, the relative concentration of Cesium-137 ( $^{137}\text{Cs}$ ) was highly correlated to the extent of error. Using alternative analysis techniques, the error for  $^{226}\text{Ra}$  estimations may be reduced by a factor of 2 for randomly selected samples and by a factor of 4 for samples containing high concentrations of  $^{137}\text{Cs}$  relative to the concentrations of  $^{226}\text{Ra}$ .

### **INTRODUCTION**

Public Law 95-604, the Uranium Mill Tailings Radiation Control Act of 1978, requires the Federal government to perform remedial action on 24 inactive uranium mill tailings sites and their associated vicinity properties. In accordance with standards recommended by the Environmental Protection Agency (EPA), a property contaminated by mill tailings may be included in UMTRAP if the concentration of  $^{226}\text{Ra}$  in land averaged over any area 100 m<sup>2</sup> exceeds 5 pCi/g above background averaged over the first 15 cm of soil below the surface or 15 pCi/g above background averaged over 15 cm thick layers of soil more than 15 cm below the surface (1).

Soil samples taken during inclusion surveys performed by the RASA Group of the Grand Junction Office of ORNL are analyzed for  $^{226}\text{Ra}$  using a NaI counting

system. In a group of 1500 randomly selected soil sample analyses reviewed, 8.5% were denoted as "Cs-flagged." Such a flagged sample was indicative of analytical interferences due to the presence of additional radionuclides in the soil, thus resulting in false overestimations of  $^{226}\text{Ra}$  content. The radionuclides considered were  $^{137}\text{Cs}$  due to global fallout as well as naturally-occurring Potassium-40 ( $^{40}\text{K}$ ) and Thorium-232 ( $^{232}\text{Th}$ ).

This paper addresses the limitations of the currently used algorithm that utilizes three ROI, and three alternative methods that reduced the error inherent in that algorithm.

## **SAMPLING AND PREPARATION PROCEDURES**

During a radiological survey, soil sampling is required if a property is not included based on sufficiently elevated indoor or outdoor gamma exposure rates. Soil sampling and preparation procedures are described in detail in the RASA/UMTRA Procedures Manual (2). Generally, a surface soil sample, 0-15 cm in depth, is taken at a background location. If necessary, surface and subsurface samples are taken at the highest outdoor gamma location. Additional samples may be taken to further characterize the extent of on-site contamination. Soil is removed with a post-hole digger, mixed for homogeneity, and a sample of approximately 500 g is deposited into an aluminum pan.

Soil samples are oven dried at  $43^{\circ}\text{C}$  for a minimum of twelve hours. Dried samples are subsequently crushed to a maximum particle size of  $0.6\text{ cm}^3$  and placed in 0.5 L plastic jars to a specified fill level (approximately 400  $\text{cm}^3$ ). Net weights of the samples are recorded; then the jars are sealed with tape to prevent the escape of radon. Samples are stored for at least twelve days prior to analysis to allow for ingrowth of radon progeny.

## **COUNTING SYSTEM**

Three sodium iodide-thallium activated crystals 15 cm by 23 cm in size with 8.3 cm by 8.9 cm wells are surrounded by copper and cadmium liners and housed in 8 cm thick "pickle barrel" lead shields. The crystals are coupled to an ND-66 multichannel analyzer via an ORTEC #113 pre-amplifier, ORTEC #490B linear amplifier, and ND-575 analog to digital converter. High voltage is supplied by an ORTEC #456 power supply. The ND-66 is interfaced with an IBM microcomputer and line printer.

Gross count data are acquired in 512 channels are calibrated at approximately 6 keV/channel. Data for background, standards and soil samples are acquired for five minutes in the ND-66 spectral display groups. A computer program written in BASIC transfers data from the ND-66 to floppy disks from which  $^{226}\text{Ra}$  concentration estimations are determined. Uncertainty is based solely on counting statistics.

## **ANALYTICAL METHODS**

Concentrations of  $^{226}\text{Ra}$ ,  $^{137}\text{Cs}$ ,  $^{40}\text{K}$ , and  $^{232}\text{Th}$  in randomly selected soil samples were determined by the ORNL RASA Group using GeLi spectrometers that had been calibrated with an NBS standard. A group of samples was also analyzed by Bendix Field Engineering Corporation in Grand Junction, Colorado using GeLi

spectrometers for the purpose of quality assurance. The comparison between these two GeLi analyses was satisfactory although the  $^{226}\text{Ra}$  concentrations averaged about 10% higher than ORNL GeLi determinations.

The algorithm currently used to obtain estimations of  $^{226}\text{Ra}$  from NaI spectra is based on the addition of net counts in three ROI (3ROI method). ROI1, ROI2, and ROI3 correspond to Bismuth-214 ( $^{214}\text{Bi}$ ) gamma energies of 609 keV, 1120 keV, and 1764 keV, respectively. These three ROI are demarcated on the NaI spectrum of a  $^{226}\text{Ra}$  reference in Figure 1. Table 1 contains a summary of NaI spectral groups which indicates the lower and upper limit energy levels (at approximately 6 keV/channel) for each ROI utilized by the 3ROI method and the three alternative methods referred to as THS, RAK, and WESTON.

For the 3ROI method, a value of (cpm/g) is determined using the following formula:

$$\text{cpm/g} = \frac{(\text{ROI1} + \text{ROI2} + \text{ROI3})}{\text{weight(g)} \times \text{time(min)}} \quad (1)$$

This value (cpm/g) is inserted into an equation which ultimately yields an estimation of  $^{226}\text{Ra}$  concentration in pCi/g. The method was developed by ORNL based on GeLi and NaI data (3). The average ratio value for ROI2/ROI1 was established to be 0.41. Since some of the counts in ROI1 can be contributed by the Barium-137m gamma from  $^{137}\text{Cs}$  at 662 keV, the sample analysis is flagged for possible  $^{137}\text{Cs}$  contamination if the ratio value is less than 0.41. These samples will be referred to henceforth as "Cs-flagged".

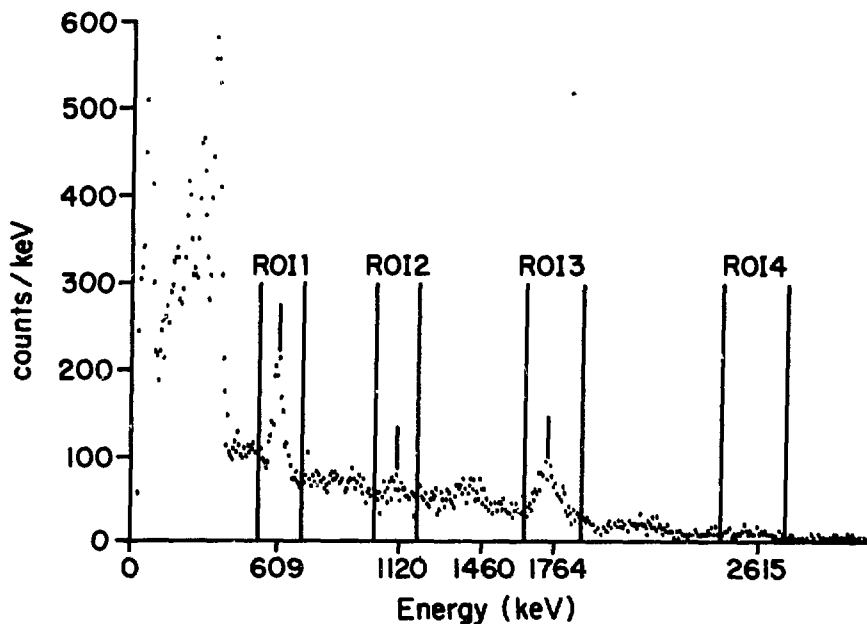


Fig. 1. NaI differential spectrum of  $^{226}\text{Ra}$  reference (5.1 pCi/g).

Table 1. Lower and upper channel numbers which correspond to energy levels (approximately 6 keV/channel) of each ROI used for the four NaI analytical methods. The ROI utilized by each method are indicated

Method	ROI1 90-120 (channel)	ROI2 170-200 (channel)	ROI3 274-314 (channel)	ROI3' 274-413 (channel)	ROI4 410-455 (channel)	ROI4' 426-455 (channel)	ROI4'' 427-512 (channel)
3ROI1	X	X	X				
THS			X		X		
RAK			X			X	
WESTON				X			X

The study undertaken during the summer of 1985 involved re-analysis of randomly selected soil samples in an effort to reduce the disagreement in  $^{226}\text{Ra}$  estimations using the three ROI method compared with GeLi results. The three alternative analytical methods did not utilize ROI1 and ROI2 because of the probable interference of  $^{137}\text{Cs}$  and  $^{40}\text{K}$  in environmental soil sample analyses. An attempt was also made to eliminate the effect of  $^{232}\text{Th}$  on  $^{226}\text{Ra}$  estimations.

One alternative analysis was a simple  $^{232}\text{Th}$ -stripping algorithm that utilized  $^{226}\text{Ra}$  and  $^{232}\text{Th}$  reference materials supplied by the Technical Measurements Center of the U.S. Department of Energy (4,5). This method, THS, used net counts in ROI3 and ROI4: ROI3 corresponds to the 1764 keV  $^{214}\text{Bi}$  gamma from  $^{226}\text{Ra}$ , and ROI4 to the 2615 keV Thallium-208 ( $^{208}\text{Tl}$ ) gamma from  $^{232}\text{Th}$  (6). In Figure 2, ROI3 and ROI4 are demarcated on the NaI spectrum of the  $^{232}\text{Th}$  reference. The counts in ROI3 due to the presence of  $^{232}\text{Th}$  were presumed to be eliminated by subtraction of the counts that appeared in ROI3. The fraction determined was unique to this counting system at this location. The following formula was implemented:

$$^{226}\text{Ra}(\text{pCi/g}) = \frac{N(\text{sample})}{N(^{226}\text{Ra ref})} \times \frac{\text{Wt}(^{226}\text{Ra ref})}{\text{Wt}(\text{sample})} \times \text{pCi/g}(^{226}\text{Ra ref}) \quad (2)$$

where:  $N = [\text{ROI3} - (0.818 \times \text{ROI4})]$

Further investigation warranted a revision of the THS method. Essentially, the RAK method was the same as THS except that ROI4 was reduced in size to eliminate counts from two low-yield photons with energies of 2204 keV and 2447 keV from  $^{214}\text{Bi}$ . ROI4' is demarcated on Figure 3. This modification dramatically improved the estimate of the contribution from  $^{232}\text{Th}$  into ROI3. The following formula was used for the RAK method:

$$^{226}\text{Ra}(\text{pCi/g}) = \frac{N(\text{sample})}{N(^{226}\text{Ra ref})} \times \frac{\text{Wt}(^{226}\text{Ra ref})}{\text{Wt}(\text{sample})} \times \text{pCi/g}(^{226}\text{Ra ref}) \quad (3)$$

where:  $N = [\text{ROI3} - (1.1 \times \text{ROI4}')] ]$

The WESTON analysis uses a complex routine to strip  $^{232}\text{Th}$ . ROI4 was enlarged to include not only the 2615 keV  $^{208}\text{Tl}$  peak, but also counts contributed by Compton scattering from  $^{208}\text{Tl}$  sum peaks. ROI3' was an enlargement of ROI3 that encompassed the low-yield  $^{214}\text{Bi}$  peaks. Additional computations were performed. The  $^{226}\text{Ra}$  contribution to ROI4'' was determined, and the fraction of those that appeared in ROI3' were subtracted from ROI3' (7).

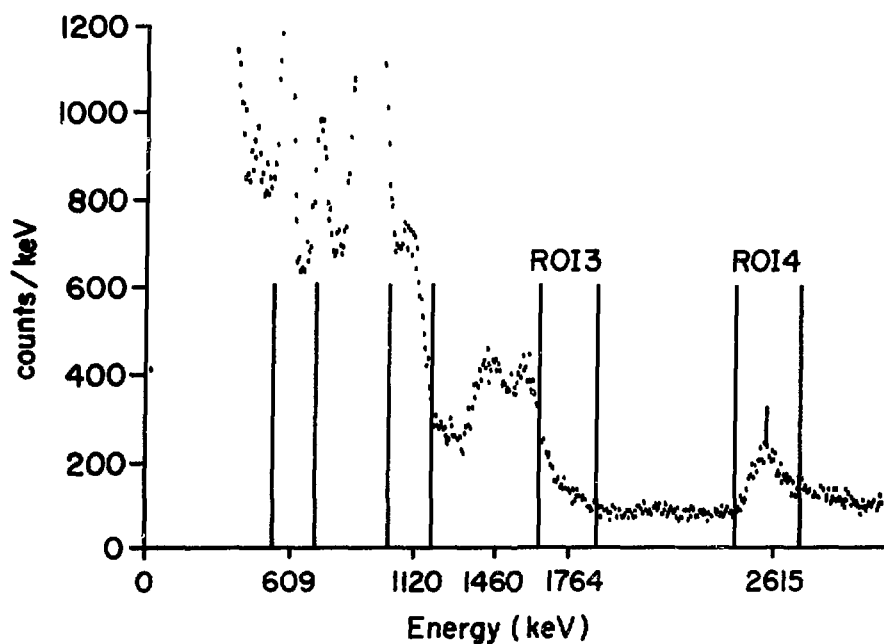


Fig. 2. NaI differential spectrum of  $^{232}\text{Th}$  reference(70.2 pCi/g).

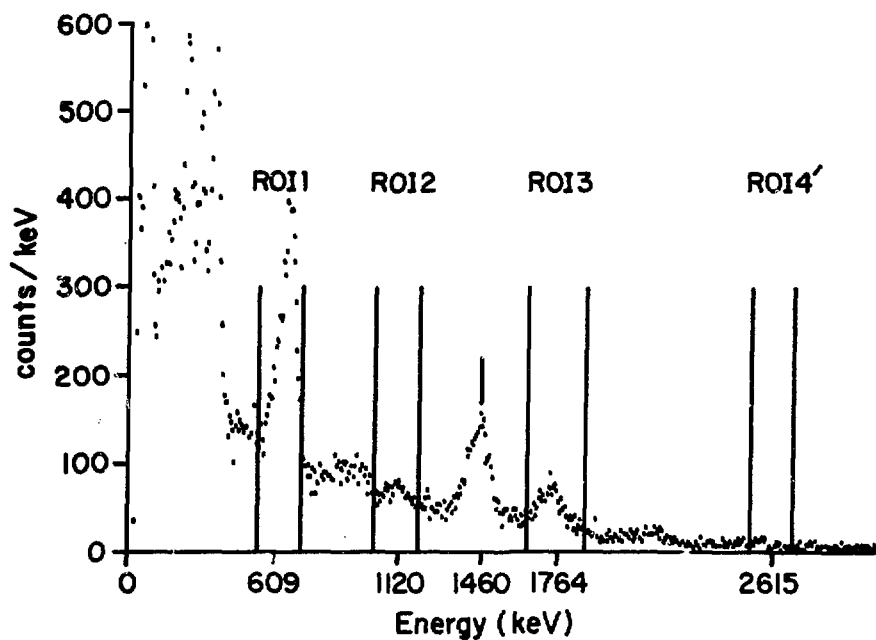


Fig. 3. NaI differential spectrum of "Cs-flagged" soil sample containing:  $^{226}\text{Ra}$ (4.1 pCi/g),  $^{137}\text{Cs}$ (2.6 pCi/g),  $^{40}\text{K}$ (23 pCi/g), and  $^{232}\text{Th}$ (0.81 pCi/g).

## RESULTS AND DISCUSSION

A group of 64 samples that had previously been analyzed using the ORNL GeLi spectrometers was also analyzed for  $^{226}\text{Ra}$  using the 3ROI method and the three alternative analysis methods: THS, RAK, and WESTON. In order to assess the error for  $^{226}\text{Ra}$  concentration estimations using the various NaI methods, the results were compared with  $^{226}\text{Ra}$  concentrations determined by GeLi analysis. The % difference for each  $^{226}\text{Ra}$  estimation from the GeLi concentration was calculated in the following manner:

$$\% \text{ Difference} = \left( \frac{\text{NaI estimation} - \text{GeLi determination}}{\text{GeLi determination}} \right) \times 100 \quad (4)$$

Figure 4 contains box plots depicting the distribution of the % differences between NaI  $^{226}\text{Ra}$  estimations and GeLi determinations for both groups of 64 randomly selected and 64 "Cs-flagged" soil samples. In a box plot, the spread of the bulk of the data (the central 50%) is seen as the length of the box. The median is portrayed by a horizontal line segment within the rectangle. Vertical lines extend from the ends of the box to adjacent values. The upper adjacent value is defined as the observed value that is less than or equal to the upper value of the box plus 1.5 times the range of values defined by the box. The lower adjacent value is defined similarly for values falling below the box. Outside values, which fall beyond the adjacent values, are plotted as individual points (8).

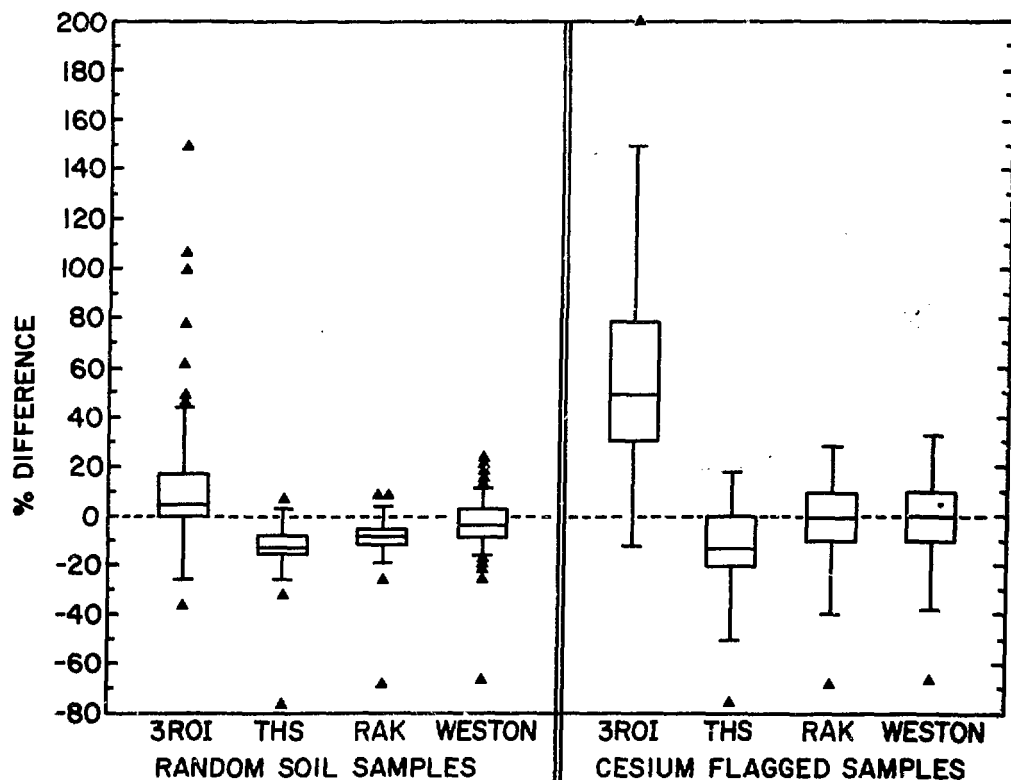


Fig. 4. Box plots showing the distribution of the % difference between estimates of  $^{226}\text{Ra}$  in soil using NaI analysis techniques and GeLi determinations.

All four NaI methods resulted in estimations of  $^{226}\text{Ra}$  which were in fairly good agreement with GeLi determinations for random samples. Excluding outside values, the range of errors for the 3ROI method was 70% compared with about 26% for the three alternative methods. The median value for the 3ROI method corresponds to an overestimate of 5%, while the other algorithms yielded a value corresponding to an underestimate of  $^{226}\text{Ra}$ . It should be noted that 6 of 25 outside values indicated on the box plots for random soil samples were for "Cs-flagged" samples in that group.

The ranges of % differences for "Cs-flagged" samples were more than twice as large as those for randomly selected samples. Nearly all of the estimates of  $^{226}\text{Ra}$  concentration using the 3ROI method were substantially higher than the GeLi determinations. The THS method yielded underestimations that averaged 12% lower than corresponding GeLi results. Analyses using the RAK and WESTON methods resulted in estimations of  $^{226}\text{Ra}$  which were in excellent agreement with GeLi determinations; 50% of the estimations were within 10% of the  $^{226}\text{Ra}$  concentrations determined by GeLi analysis.

In order to determine if the errors in  $^{226}\text{Ra}$  concentrations using the four NaI analytical methods were directly related to the concentrations of additional radionuclides present in the soil samples, correlation coefficients were computed for the % difference and the concentration ratios of ( $^{226}\text{Ra}/^{137}\text{Cs}$ ), ( $^{226}\text{Ra}/^{40}\text{K}$ ), and ( $^{226}\text{Ra}/^{232}\text{Th}$ ). The use of a correlation coefficient implies that there is an association between two variables, however, it is not a cause and effect relationship. When dealing with multiple variables, the correlation coefficient between any two might be low, nevertheless, the correlation could be both appreciable and significant. Table 2 contains a summary of correlation coefficients for a group of 15 random samples and 15 samples that were "Cs-flagged". Confidence in the correlation coefficients can be determined using  $r_{\text{test}}$  values. If the correlation coefficient is greater than the  $r_{\text{test}}$  value, one can be certain (to the degree selected) that the calculated correlation coefficient was not due to chance alone (9). Table 3 contains applicable  $r_{\text{test}}$  values.

Table 2. Correlation coefficients (r values) calculated for % differences of NaI  $^{226}\text{Ra}$  concentration estimations from GeLi  $^{226}\text{Ra}$  concentration determinations and soil sample concentration ratios of ( $^{226}\text{Ra}/^{137}\text{Cs}$ ), ( $^{226}\text{Ra}/^{40}\text{K}$ ), and ( $^{226}\text{Ra}/^{232}\text{Th}$ )

Sample Group	Concentration Ratio	3ROI (r)	THS (r)	RAK (r)	WESTON (r)
Random	( $^{226}\text{Ra}/^{137}\text{Cs}$ )	-0.199	-0.0170	-0.0580	-0.134
Random	( $^{226}\text{Ra}/^{40}\text{K}$ )	-0.453	-0.195	-0.431	-0.359
Random	( $^{226}\text{Ra}/^{232}\text{Th}$ )	-0.480	-0.0910	-0.359	-0.309
"Cs-flagged"	( $^{226}\text{Ra}/^{137}\text{Cs}$ )	-0.957	-0.342	-0.517	-0.428
"Cs-flagged"	( $^{226}\text{Ra}/^{40}\text{K}$ )	-0.275	0.143	-0.179	-0.0886
"Cs-flagged"	( $^{226}\text{Ra}/^{232}\text{Th}$ )	-0.126	0.288	0.00469	0.171

Table 3. Correlation coefficient test values ( $r_{\text{test}}$ ) that may be compared with  $r$  values to establish the degree of certainty with which positive or negative correlation coefficients may be due to chance alone

certainty (80%)	certainty (90%)	certainty (95%)	certainty (99%)	certainty (99.5%)
0.351	0.441	0.514	0.641	0.760

There was no significant correlation between % error and relative concentration of  $^{137}\text{Cs}$  for randomly selected soil samples using any analytical method. With 90% certainty, a negative correlation existed between 3ROI error and the relative concentrations of  $^{40}\text{K}$  and  $^{232}\text{Th}$ . With a lesser degree of certainty (80%), the % differences correlated with ratios of ( $^{226}\text{Ra}/^{40}\text{K}$ ) and ( $^{226}\text{Ra}/^{232}\text{Th}$ ) for the RAK method. Similarly, there was a negative correlation between WESTON error and the ratio of ( $^{226}\text{Ra}/^{40}\text{K}$ ), but not with ( $^{226}\text{Ra}/^{232}\text{Th}$ ). With exception to the THS method, approximately 16% of the % differences for the estimations of  $^{226}\text{Ra}$  could be described by correlations with the relative concentrations of  $^{40}\text{K}$  or  $^{232}\text{Th}$  present.

The % differences and relative concentrations of  $^{137}\text{Cs}$  were negatively correlated with nearly 100% certainty for "Cs-flagged" samples analyzed using the 3ROI method, and to lesser degrees using alternative methods. About 90% of the  $^{226}\text{Ra}$  estimations determined by the 3ROI method in "Cs-flagged" samples were directly influenced by relative concentrations of  $^{137}\text{Cs}$ . Less than 20% of the estimations using the alternative methods were influenced by the presence of  $^{137}\text{Cs}$ .

There were no significant correlations between % differences and relative concentrations of  $^{40}\text{K}$  and  $^{232}\text{Th}$  in "Cs-flagged" samples, although approximately 8% of soil samples analyzed using the 3ROI and THS methods have errors which may be described by the concentration ratios of ( $^{226}\text{Ra}/^{40}\text{K}$ ) and ( $^{226}\text{Ra}/^{232}\text{Th}$ ), respectively.

Figures 1-3 provide illustration of the possible effects of  $^{137}\text{Cs}$ ,  $^{40}\text{K}$ , and  $^{232}\text{Th}$  on  $^{226}\text{Ra}$  concentration determinations. When comparing Figure 3 (spectrum of "Cs-flagged" soil sample containing 4.1 pCi/g of  $^{226}\text{Ra}$ ) to Figure 1 (spectrum of reference containing 5.1 pCi/g of  $^{226}\text{Ra}$ ), it may be observed that not only is the peak in ROI1 about twice as tall, but also it is skewed to the right. Those phenomena are mostly due to the presence of  $^{137}\text{Cs}$  in the soil sample. However, the baseline shift evident on Figure 3 as compared with Figure 1 is mainly due to the presence of  $^{40}\text{K}$  in the sample. Figure 2 ( $^{232}\text{Th}$  reference spectrum) shows that  $^{232}\text{Th}$  does indeed contribute counts to lower energy ROI. Utilization of the 3ROI method may therefore erroneously attribute count contributions to  $^{226}\text{Ra}$  which are actually due to  $^{40}\text{K}$ ,  $^{232}\text{Th}$ , and especially  $^{137}\text{Cs}$ .

## SUMMARY

Soil samples collected during radiological surveys performed for the RASA/UMTRA Project were analyzed for  $^{226}\text{Ra}$  using NaI spectrometers. Of 1500 randomly selected analyses determined by the currently used 3ROI algorithm, 8.5% were flagged for possible  $^{137}\text{Cs}$  contamination ("Cs-flagged").

Results using the 3ROI algorithm were compared with results using three alternative analytical methods by calculating the % difference between the  $^{226}\text{Ra}$  estimated by each NaI analysis and  $^{226}\text{Ra}$  GeLi determinations. For random samples, estimations of  $^{226}\text{Ra}$  were fairly accurate using all four methods although the range of error for the 3ROI method was twice as large as those for the other methods. For "Cs-flagged" samples, the 3ROI method generally overestimated the concentration of  $^{226}\text{Ra}$  by about 50%. The three alternative methods yielded more accurate results. The RAK and WESTON methods yielded estimations that were within 10% of the GeLi determinations in 50% of the "Cs-flagged" samples.

A statistical analysis using correlation coefficients indicated that for random soil samples, about 13% of the error for all four NaI methods could be related to the concentration ratios of ( $^{226}\text{Ra}/^{40}\text{K}$ ) and ( $^{226}\text{Ra}/^{232}\text{Th}$ ).

The % differences from  $^{226}\text{Ra}$  GeLi results when using the 3ROI method were strongly correlated to the ratio of ( $^{226}\text{Ra}/^{137}\text{Cs}$ ) for "Cs-flagged" samples.

The RAK and WESTON methods yielded an improvement in the accuracy of estimated  $^{226}\text{Ra}$  concentrations when compared with the 3ROI method. Using these alternative methods, 97% of the analyses for  $^{226}\text{Ra}$  in soil samples were within 20% of the GeLi determinations.

## ACKNOWLEDGMENTS

This work was funded by the Radiological Survey Activities Group, Health and Safety Research Division, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, Tennessee, 37831, under DOE contract #5-36387. The authors wish to thank the following ORNL personnel for their support and assistance: Barry Berven, Craig Little, Rick Doane, Betty Ellis, and Carla Miller. The authors also appreciate the expertise given by Roger Nelson of the Jacobs/Weston team and George Angleton of Colorado State University.

## REFERENCES

1. U. S. Environmental Protection Agency, "Standards for Remedial Action at Inactive Uranium Mill Processing Sites," Federal Register, Code of Federal Regulations, Section 40, Part 192, Wednesday, January 5, 1983, pp. 590-606.
2. S. J. Ramos, B. A. Berven, and C. A. Little, RASA/UMTRA Procedures Manual, ORNL/TM-8600 (draft).
3. R. W. Doane, B. A. Berven, and M. S. Blair, "A Computer Controlled System for Rapid Soil Analysis of  $^{226}\text{Ra}$ ," Computer Applications in Health Physics, Proceedings of the Seventeenth Midyear Topical Symposium, pp. 7.173-7.178, February 5-9, 1984. Health Physics Society, 1340 Old Chain Bridge Road, McLean, Virginia, 22101.

4. U. S. Department of Energy, Division of Remedial Action Projects, Technical Measurements Center, Grand Junction Area Office, Colorado, "Development of Solid Ra-226 Reference Materials," GJ/TMC/10(83), UC-70A (November 1983).
5. U. S. Department of Energy, Division of Remedial Action Projects, Technical Measurements Center, Grand Junction Area Office, Colorado, "Development of Solid Th-232 Reference Materials," GJ/TMC-22, UC-70A (May 1985).
6. P. D. Kearney, H. R. Meyer, and B. T. Binde, "Field Measurements of Soil Radium Concentrations: The Opposed Crystal System," Abstracts of papers presented at the Thirtieth Annual Meeting of the Health Physics Society, p. 53, May 26-31, 1985.
7. Roger A. Nelson, Jacobs/Weston team, 5301 Central Avenue NE, Suite 1700, Albuquerque, New Mexico, 87108.
8. John M. Chambers, William S. Cleveland, Beat Kleiner, and Paul A. Tukey, GRAPHICAL METHODS FOR DATA ANALYSIS, Wadsworth International Group, Belmont, California, and Duxbury Press, Boston, 1983.
9. William Mendenhall and Lyman Ott, UNDERSTANDING STATISTICS, Third Edition, Duxbury Press, North Scituate, Massachusetts, 1980.

## **A FAST SORTING MEASUREMENT TECHNIQUE TO DETERMINE DECONTAMINATION PRIORITY**

**C. H. Distenfeld\*, B. Brosey, and H. Igarashi\*\***  
GPU Nuclear Corporation  
P.O. Box 480  
Middletown, Penna. 17057

**\*Bechtel National Inc.**

**\*\* Hitachi, Ltd.**

### **ABSTRACT**

The method used to select decontamination priorities for the Three Mile Island Unit 2 (TMI-2) reactor building (RB) is systematic, but costs in personnel exposure and time must be borne. One way of minimizing exposure is to define and treat the one or two surface sources that are important contributors to the collective dose of the recovery personnel. Surface characteristics can then be determined and decontamination techniques developed to match the removal requirements. At TMI-2, a fast sorting technique was developed and used to prioritize surfaces for exposure reduction. A second quick sort can then be used to determine the next generation of surface characterization, decontamination method selection, and performance.

The quick-sort method that was developed is based on the Eberline HP 220A directional survey system. The angular response of the HP 220A probes approaches 2 pi steradians and allows toward-away type measurements. Sources distributed over 4 pi steradians are hard to define with this system. Angular differentiation was improved to about  $\pi/2$  steradians by redesigning the probe shield. The change allows unambiguous six-direction measurements, such as up, down, front, rear, right, and left with practically no angular overlap or exclusion. A simple, light-weight stand was used to establish an angular reference for the rectangular packaged probe. The six surface planes of the rectangle work with the angular reference to establish the six viewing angles.

### **INTRODUCTION**

Recovery of large contaminated buildings, such as the TMI-2 RB, are complicated by ceilings that can be 12-13 meters high. Much of the overhead spaces are filled with conduits, pipes, cable trays, ventilation ducts, and steel structures. The total complex surface can greatly exceed the total surface of walls and floor. Concrete pedestals, heavy steel stands, embedded steel rails, refueling mechanisms, and other similar structures complicate normally accessible areas and impede exposure reduction efforts.

Initial recovery of contaminated spaces tends to involve treatment of hot spots and accessible spaces such as floor and wall surfaces. Subsequent decontamination may be less efficient since untreated surfaces, such as in over-head spaces, may be beyond the reach of ordinary decontamination tools.

To conserve radiation exposure of recovery personnel, it is important to prioritize the effort so that early work provides maximum exposure reduction. That way subsequent exposure reduction can be carried out with less total exposure to recovery personnel. This favorable scenario depends on identification of key surfaces that most effect the exposure rate. Two related approaches were considered; the first used directional measurements with a high angular resolution detector, and the second used directional measurements with low angular resolution detectors.

#### INSTRUMENT EVALUATION

The Chalk River Laboratory gamma camera (Reference 1) was evaluated for use in the RB as a high resolution detector system. This detector system is based on a heavily shielded 35 mm camera body with a "pin-hole" lens. Chalk River used a specially designed conical tungsten collimator for the lens. The camera can be used to take normal optical photographs by opening the shutter. Gamma images are formed through the closed shutter on a previously exposed film frame containing a latent optical image or on a separate unexposed film frame. The camera has a field of view of about 45 degrees. Source identification is performed by locating gamma images on the same optical field. Normal photographic image quality depends on light-dark contrast. Gamma images are similar, but are not resolved nearly as well. This is due to energetic gamma rays streaming through the shielding near the pin-hole effectively increasing the hole size and decreasing image resolution. Intense point sources strong enough to establish the exposure field are easily imaged by this high resolution system. Unfortunately, point sources lose contrast when superimposed on fields produced by distributed contamination. In these situations the exposure rate is dominated by the distributed contamination. Field tests suggest that the gamma camera is not the most rapid or convenient method for locating key surfaces where broad fields and modest intensity differences are expected. Future electronic image enhancement and digital conversion of raster images may improve the practicality of the gamma camera.

The US Navy prompted development of a tungsten shielded GM detector that could be used for directional surveying. A commercial version, designated as the HP 220A probe (Reference 2), is an example of a low resolution detector. Sample results of a survey taken with the probe are shown on Table 1, and the geometry of the measurement is shown on Figure 1. The objective was to determine where effective shielding could be placed to provide exposure rate reduction from the contaminated RB air cooler units. Measurements shown on Figure 1 were made according to the scheme shown on Figure 2. Directional survey results are listed on Table 1 for the HP 220A probe and for coupled data from conventional gamma survey instruments. The accuracy of differential

Table 1  
Air Cooler  
Directional Survey Results  
mR/h

Instrument	Total	Toward	Away	Right	Left	Up	Down
RO-2	80 $\pm$ 10						
HP 220A	190	55	15	30	25	55	10
RO-2	50 $\pm$ 6						
Modified	56	5	6	10	13	20	2

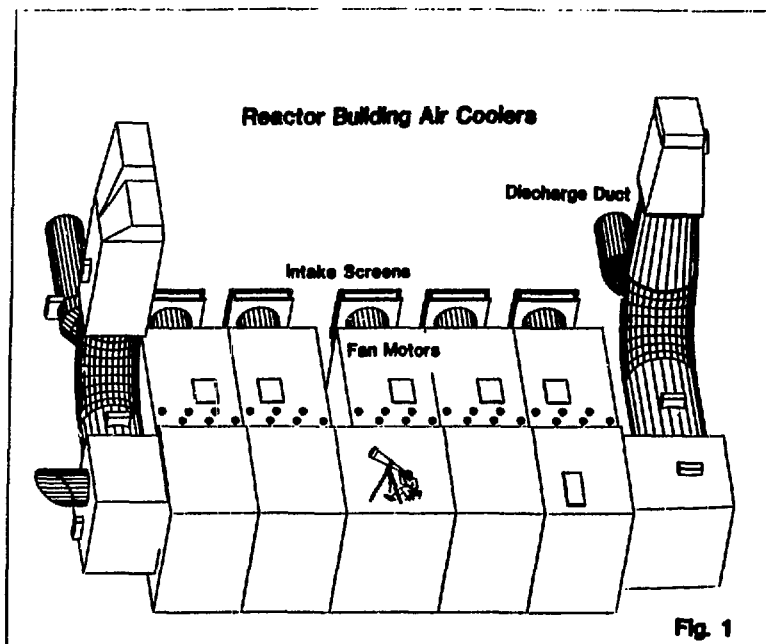


Fig. 1

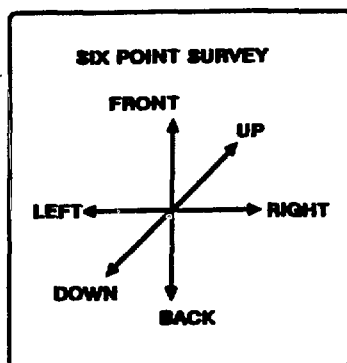
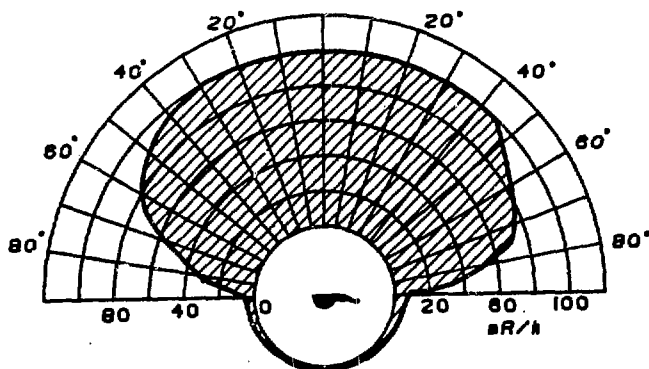


FIGURE 2

measurements can be tested by comparing the summed result with an independent measurement of the whole. Referring to Table 1, the summed results are about twice as large as the conventional 4 pi steradian instrument. Figure 3 depicts the angular response of the HP 220A probe to a Cs-137 source. The response extends about 130-140 degrees in both planes. Attempting to use the detector to differentiate 4 pi steradians into six views as indicated on Figure 2, causes overlapping fields. Thus the same sources are counted more than one time; clearly the probe has too little angular resolution for the six view protocol. Three horizontal determinations would include about 115% of the solid angle in that plane, but with conical exclusions up and down along the vertical axis. For task definition, the utility of six independent views motivated development of an instrument with the appropriate angular response.



ANGULAR RESPONSE OF HP220A  
FIGURE 3

#### 1/2 pi DETECTOR ASSEMBLY

A conical collimator was added to the HP 220A probe as shown on Figure 4. Care was taken to position the sensitive volume of the GM detector within the apex of the cone. Figure 5 depicts the angular response of the new instrument in the plane indicated. Nearly identical response was experienced for measurements made perpendicular to this plane. Response due to "leakage" through the finite shield thickness was limited to about 2% of the forward field of view.

#### CUT-AWAY VIEW OF SPECIAL HP220A PROBE WITH MODIFICATIONS

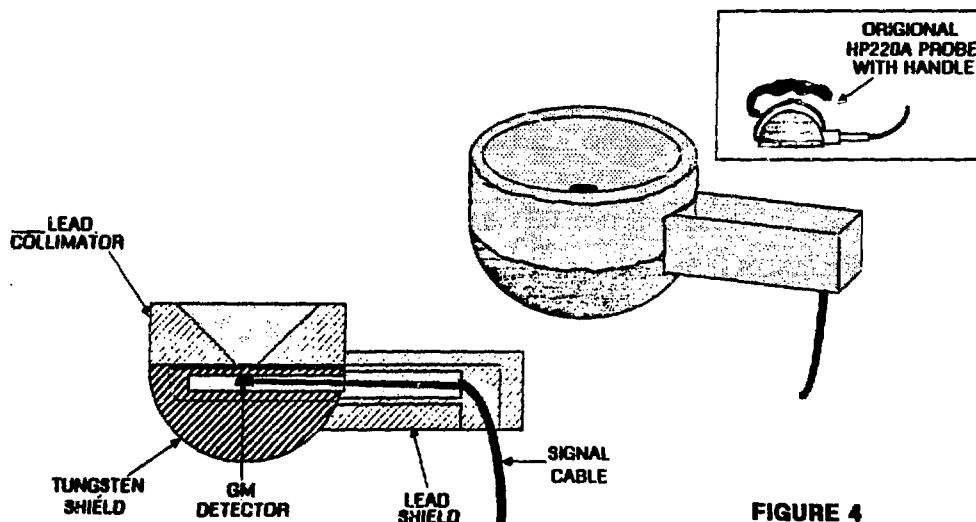
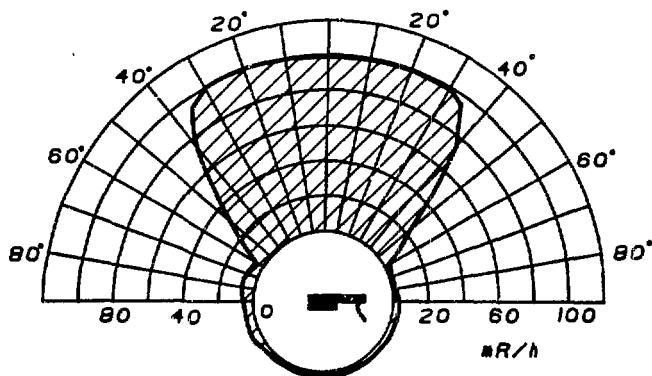


FIGURE 4



ANGULAR RESPONSE OF MODIFIED HP220A  
FIGURE 5

Detector orientation was important so that each field of view was unique. This was accomplished by packaging the detector assembly in a thin walled aluminum box. The rectangular box provided six surfaces that exactly correspond to the angular orientation needed for six independent fields of view. A light aluminum support frame was adapted from a 1.22 meter ladder. All unnecessary material was removed and a clearance hole was cut to allow minimum obstruction to the "down" field of view. As shown on Figure 6 the horizontal top of the ladder supported the detector assembly. Rotation of the detector assembly around a reference edge or line provided the six non-overlapping fields of view.

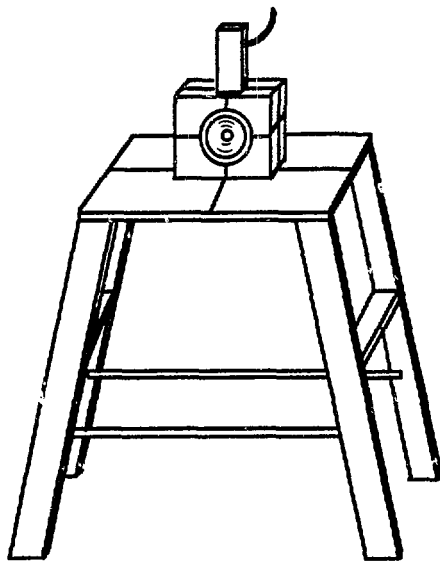


FIGURE 6 BOX--ALIGNMENT

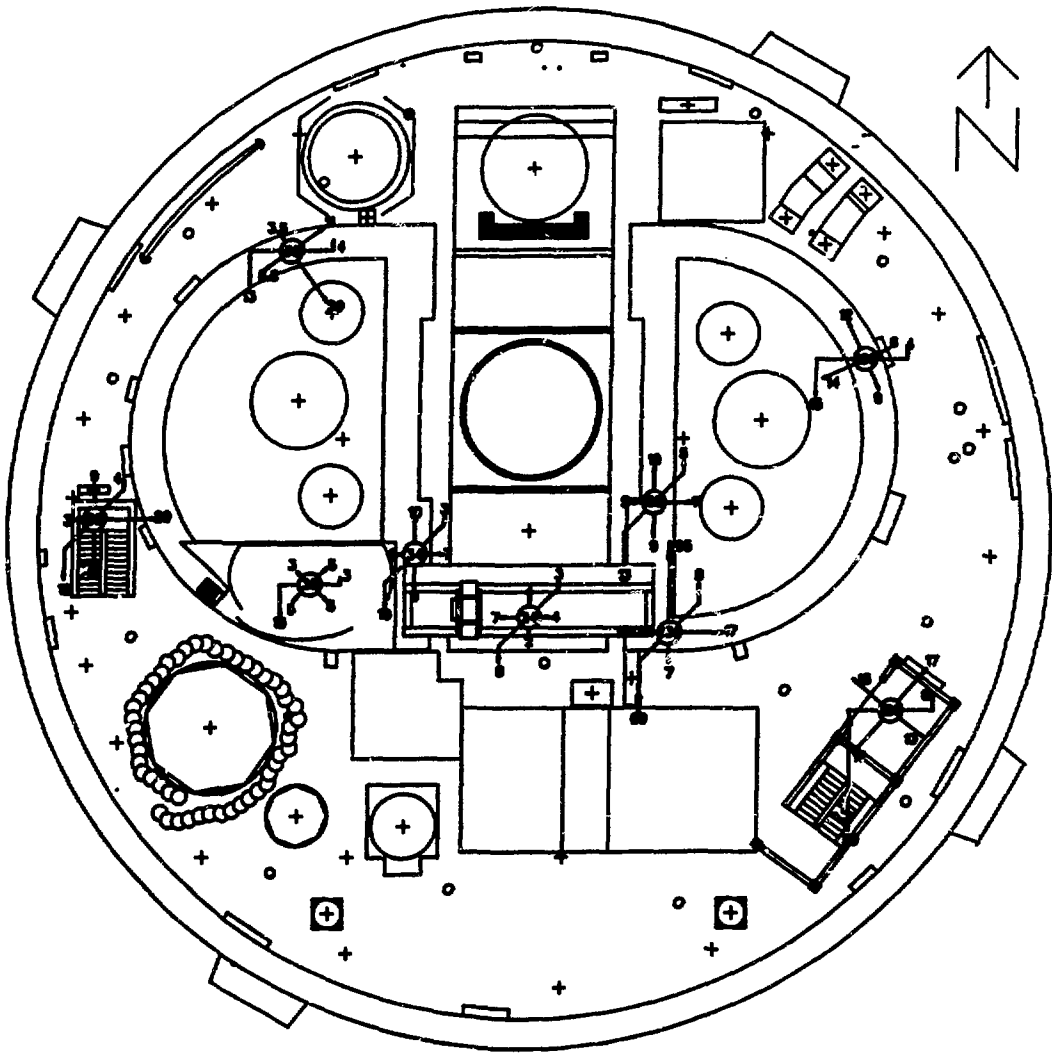
## MEASUREMENT RESULTS

Figures 7, 8, and 9 are plots of the hexa-directional measurement results from the 367', 347', and 305' elevations in the RB. The data are presented as a series of vectors emerging from a central bulls-eye. The numerical value recorded in the bulls-eye was the 4 pi steradians general area exposure rate; the four solid line vectors represented horizontal measurement results; and the up-down vector lines start at the ends of the dashed line extenders. In all cases, vector lengths were drawn to be proportional to the exposure rate value.

The mean value of the ratio of all 4 pi steradians general area values to the corresponding sums of the hexa-directional measurements was 1.03 with a standard deviation of the mean ratio of 4%, indicating an excellent match between the angular response and the hexa-directional protocol.

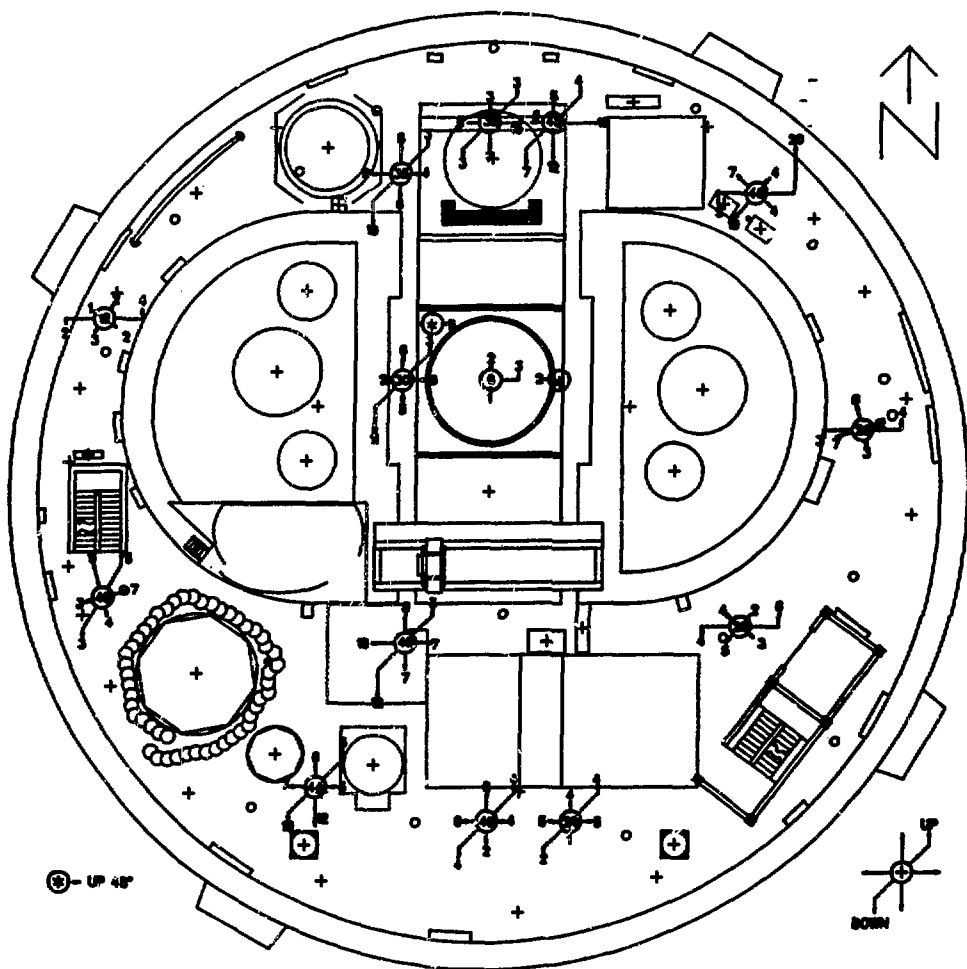
Directional data includes contributions from all sources within view. The objective was to determine exposure reduction priorities for various surfaces. Some effort may be needed to interpret the measurements for contributing surfaces. For example, a down looking measurement made 1.22 meters above the floor will view a disc shaped pattern on the floor with a radius of 1.22 meters since the probe half angular response was about 45 degrees. For many down-looking measurements the interpretation is simple; the measurements directly relate to the floor surface except when walls or equipment intrude into the field of view. The other fields of view must be considered in a similar way.

Table 2 is a summary of results interpreted for major obvious surfaces. The data and interpretation presented are examples of the capability of the system. Specific TMI-2 results are site specific and are considered to be of less general interest than the main conclusion. In summary, an urgent need was experienced to make rapid and accurate directional measurements in contaminated spaces. An instrument was developed to satisfy the requirement and to help make exposure reduction efforts ALARA.



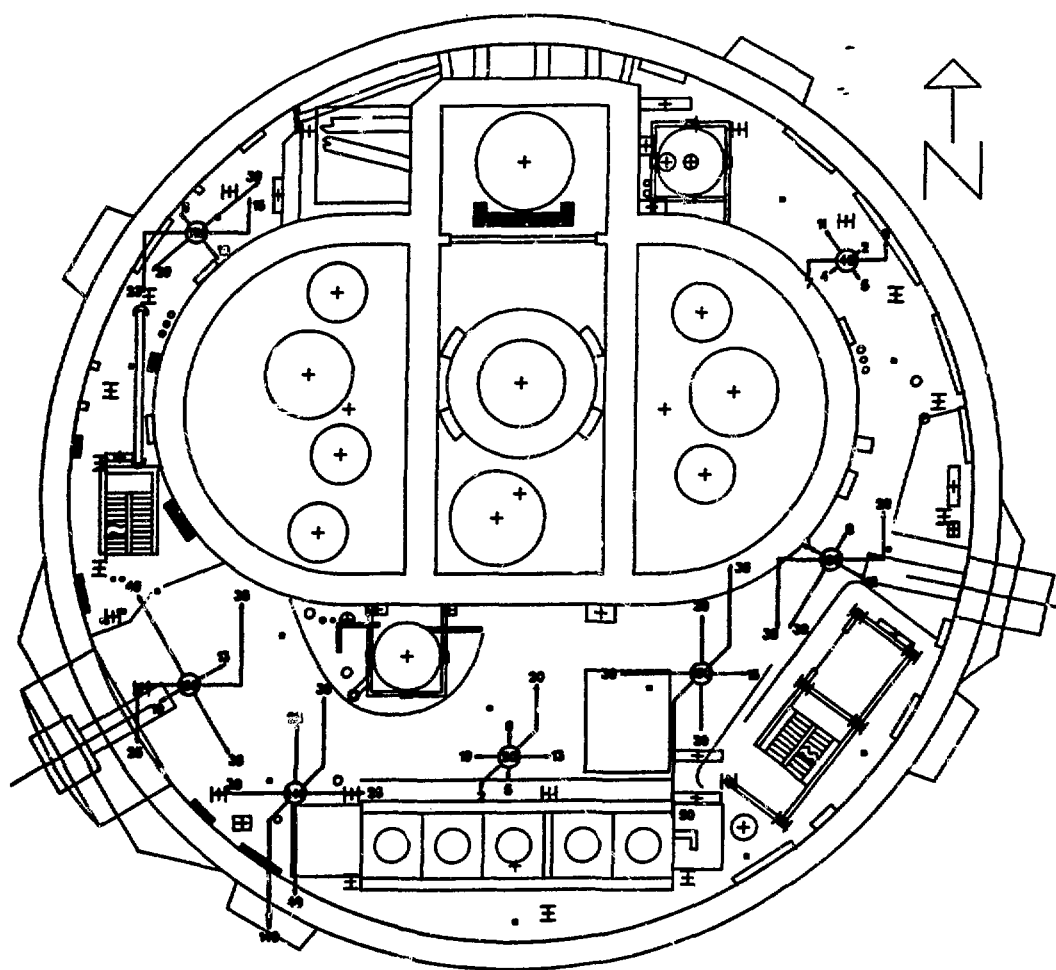
**Reactor Building Elevation 367 ft.**

**Fig. 7**



Reactor Building Elevation 347 ft.

Fig. 8



Reactor Building Elevation 305 ft.

Fig. 9

Table 2

## Results of Six Vector Survey

<u>Responsible Surface</u>	<u>Location</u> (Elevation)	<u>Exposure Rate</u> (mR/h $\pm$ $\sigma$ )
Dome and Upper Liner	367'	2.9 $\pm$ 0.7
	347'	2.9 $\pm$ 0.9
$\bar{x}_i = Up_i - \left( \frac{1}{1-6} \sum - Up_i \right) + 50$		
Overheads	305'	21.0 $\pm$ 9.0
Steel Liner Wall	356'	2.2 $\pm$ 1.4
	347'	2.1 $\pm$ 1.0
	305'	4.0 $\pm$ 3.0
Concrete D-ring Walls	347'	2.5 $\pm$ 1.1
	305'	5.0 $\pm$ 2.2
Air Cooler	347'	22.0 $\pm$ 4.0
Clean 347 Floor	347'	2.9 $\pm$ 0.8
Clean 305 Floor	305'	25.0 $\pm$ 5.0
West Canal Walkway	347'	11 to 15 (range)

## REFERENCES

1. R.J. Beal, "A Remotely-Controlled Pinhole Gamma Camera," AECL-1801, Nov. 1962
2. Eberline Instrument Corporation, Box 2108, Santa Fe, NM 87504

# HEALTH PHYSICS EXPERIENCE IN THE DECONTAMINATION AND DECOMMISSIONING OF THE 203-S, 204-S, AND 205-S URANIUM NITRATE HEXAHYDRATE PURIFICATION AND WASTE TANK CAR UNLOADING FACILITIES.

O. R. Rasmussen  
Rockwell Hanford Operations  
P. O. Box 800  
Richland, WA 99352

## ABSTRACT

The obsolete 203-S, 204-S, and 205-S uranium nitrate hexahydrate purification and storage facility and the radioactive waste tank car unloading equipment in the 200 West Area of the U.S. Department of Energy Hanford Site were decontaminated and decommissioned. Aboveground facilities were dismantled, and the area was backfilled with clean fill over underground concrete structures. The site was released as a radiologically clean area with underground contamination. The 13-month project was completed by Rockwell Hanford Operations decontamination and decommissioning forces three months ahead of schedule and \$275,000 under budget.

Important health physics aspects of the project included hydraulic sluicing of vessels to reduce radioactive material inventories, intact transport, fill and burial of large vessels to minimize hands-on dismantling, and the use of special containment techniques for pipe dismantling, including pipe drainage with explosive-activated tees, and pipe cutting and sealing with hydraulic crimping equipment.

## INTRODUCTION

The 203-S, 204-S, and 205-S Decontamination and Decommissioning (D&D) project represents the first major D&D operation carried out by the recently established D&D team in the U.S. Department of Energy Hanford Site Chemical Processing Areas. The 203-S, 204-S, and 205-S facility (fig. 1) consisted of a small aboveground uranium nitrate hexahydrate (UNH) tank farm (203-S), a larger aboveground UNH/thorium nitrate/mixed fission product waste tank farm with associated rail tank car unloading facilities (204-S), and an underground process cell with an attached two-story chemical makeup building (205-S). An extensive aboveground piping system tied the facilities into an operating unit. Selection of the site for D&D was prompted largely by health physics considerations. The aging and obsolete facility represented an increasing environmental radiological risk. At the same time, the lack of high radiation levels provided an opportunity for the new D&D team to acquire skills and experience without undue radiological risks.

## BACKGROUND

The 203-S, 204-S, and 205-S facilities were constructed in the early 1950s for the removal of trace fission products from the UNH produced by the Redox (202-S) chemical separation plant. Fission products were extracted by a silica gel-filled column in the 205-S process cell. The UNH solutions

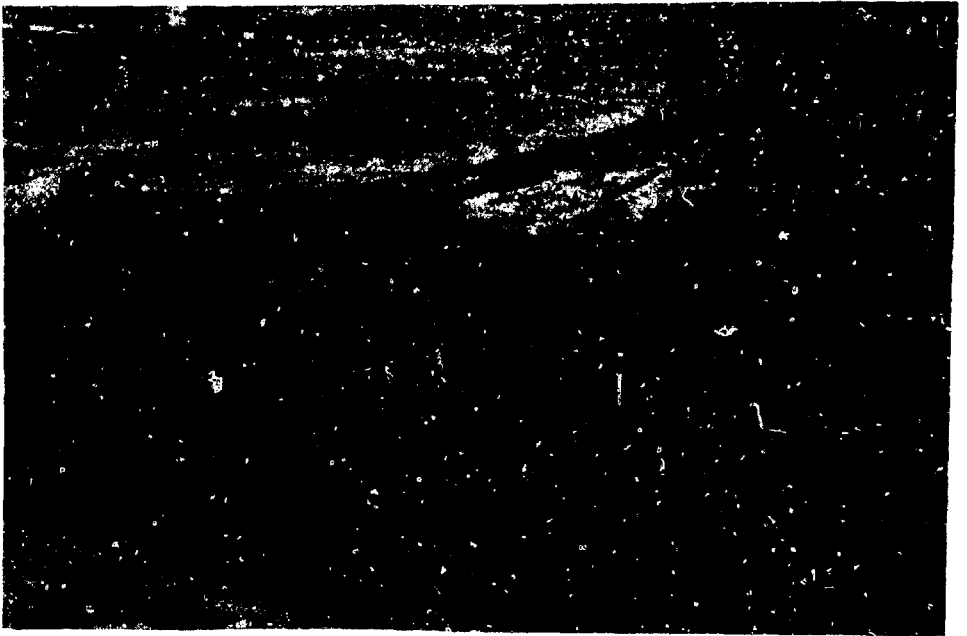
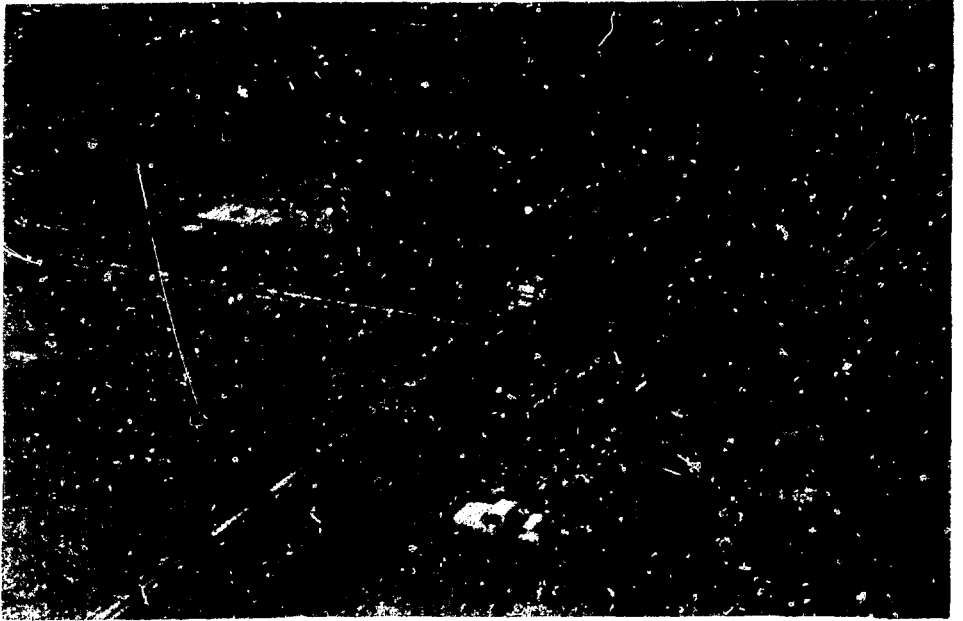


Figure 1. The 203-S, 204-S, 205-S Site Before and After Decontamination and Decommissioning. The top photograph shows the 203-S tank farm at lower center, the 204-S tank farm at top center, and the 205-S underground process cell and chemical makeup building at top left. The partially dismantled waste tank car unloading facility is visible at top right. The only aboveground facility remaining at the site, in the lower photograph, is the main Redox Plant steam line.

were stored in the 203-S and 204-S aboveground tank farms. In later years three of the 50,000-gal 204-S tanks were used to store thorium nitrate solutions, while the fourth vessel was used as a lag storage tank for a new liquid-waste rail tank car unloading station built adjacent to 204-S. Rail tank cars carrying neutralized decontamination slurries from N-Reactor and chemical process wastes from the 300 Area research laboratories were initially pumped to the 204-S-4 tank for lag-storage, although a direct routing to the under-ground tank farms was utilized in later years. The waste tank car unloading operations brought the bulk of the fission products and associated radiological hazards to the site. The primary radionuclides were cobalt-60 in N-Reactor waste, and strontium-90 and cesium-137 in laboratory waste. Tank car contact dose levels of 3-5 rem/h were routine, with similar dose levels found on transfer piping and the 204-S-4 tank.

## SELECTION FOR DECONTAMINATION AND DECOMMISSIONING

By the 1970s, it became clear that outdoor unloading of rail tank cars was impractical, especially in view of high tank car radiation levels, the inability to remove tank car sludges, winter freezeup difficulties, and contamination of the tank basins from pump seal leakage. During its latter years of operation, the 204-S tank car unloading operations caused more than 50% of the occupational exposure absorbed by the entire tank farm staff. Design and construction of a new tank car unloading building was expedited, and the new facility, 204-AR, was placed in operation in the spring of 1981. The new facility was designed with health physics considerations a top priority, and included a totally enclosed operating area, a shielded control room, a high-level shielded access platform, equipment for sluicing radioactive sludge residues from tank cars, and a leak-proof mechanical pump-seal system. Tank car unloading now accounts for a very small fraction of the tank farm occupational exposure. Transfer of operations to the new facility still left an aging and potentially hazardous outdoor facility requiring considerable surveillance and maintenance. Before D&D operations could be funded and initiated, steps were taken for immediate reduction of radiological hazards. The most radioactive piping was removed, the tank basins and pumps were extensively decontaminated, and the four 204-S tanks were sluiced with a commercial tank cleaning machine to remove the bulk of sludge in these vessels. A large net was placed over the entire 204-S complex to prevent bird and animal entry into the basins. While sluicing was very successful in the removal of silica gel and thorium nitrate residues, 10 tons of mixed fission product sludge, a fourth of the original inventory, remained in tank 204-S-4. The material resisted cold water sluicing and left 2-5 rem/h exposure fields in the 204-S-4 vicinity. Although shutdown/standby placed 203-S, 204-S, and 205-S into a relatively safe standby condition, interest in total dismantling remained. Decontamination and decommissioning was authorized in February 1983 and funded for \$1.3 million.

## HEALTH PHYSICS CONSIDERATIONS

From the outset D&D planning was based on the principle that radiological and environmental protection would be a top priority and would guide the course of dismantling operations. Health physics considerations led to the following key concepts.

- o Dismantling work would be phased in by increasing levels of radiological hazard to allow the new D&D team to gain on-the-job experience and to work up to the more difficult and demanding D&D tasks.

- o Size reduction and cutting of radioactive equipment would be held to an absolute minimum to keep radiological hazards and costs under control. It was considered more prudent to spend money on elaborate transport arrangements for bulky equipment than to attempt unnecessary and costly size reduction prior to burial.
- o The high dose levels from the fission-product sludge inventory in tank 204-S-4 would be reduced to the lowest practical level before attempting D&D of 204-S.
- o Maximum use of hydraulic crimping would be made in pipe dismantling to speed pipe removal and to provide effective containment while cutting.

## UTILITY AND PIPING REMOVAL

The initial work phase consisted of the severance of utilities inside the D&D work area by maintenance forces, and the dismantling of isolated nonradioactive utility systems by D&D workers.

Maintenance forces cut overhead and underground electric power, air, water, and chemical services outside the 203-S, 204-S, and 205-S fence line to assure the absence of live utilities in the area. New temporary electrical outlets were installed just inside the fence gate. Steam line branches were blanked. Portable self-contained eye and body wash stations were purchased for use at the work area. The dismantling of nonradioactive systems provided a valuable training opportunity for the new D&D work force.

The second work phase initiated radioactive process equipment dismantling, and cleared the area of most piping, poles, and miscellaneous small equipment.

All piping was verified to be empty by opening drain valves in the low points of pipe systems. Where further verification was necessary, holes were drilled through the pipe. Holes were also used to assess the radiological conditions inside pipes. A great deal of effort was focused on the safe cutting and handling of asbestos materials, since the bulk of piping in the area was asbestos lagged. Standard procedure was to remove a short section of lagging in the area where a cut was to be made. A plastic bag was taped around the piping to catch asbestos fines and a constant water mist was sprayed on the cutting area to minimize airborne fines. After cutting out an asbestos section, the cut faces were securely sealed with tape and plastic for the actual pipe-cutting operation (fig. 2).

Excellent results were obtained with the hydraulic pipe crimper (fig. 3) for cutting piping 2 in. or smaller in diameter. The crimper flattened and sealed the pipes before cutting the metal, leaving only a small pinhole in each corner of the flattened ends. The pinholes could be quickly sealed with tape to contain any loose contamination. The relatively small amount of contaminated piping larger than 2 in. in diameter was cut with a saw, since the weight of the available 4 in. crimper would have required use of a small crane.

In rare instances, where required by radiological conditions inside the pipe, the line was filled with foam-producing chemicals to fix and immobilize radioactive contamination. Cut piping sections were wrapped in plastic and securely sealed with tape for transport to the burial ground.

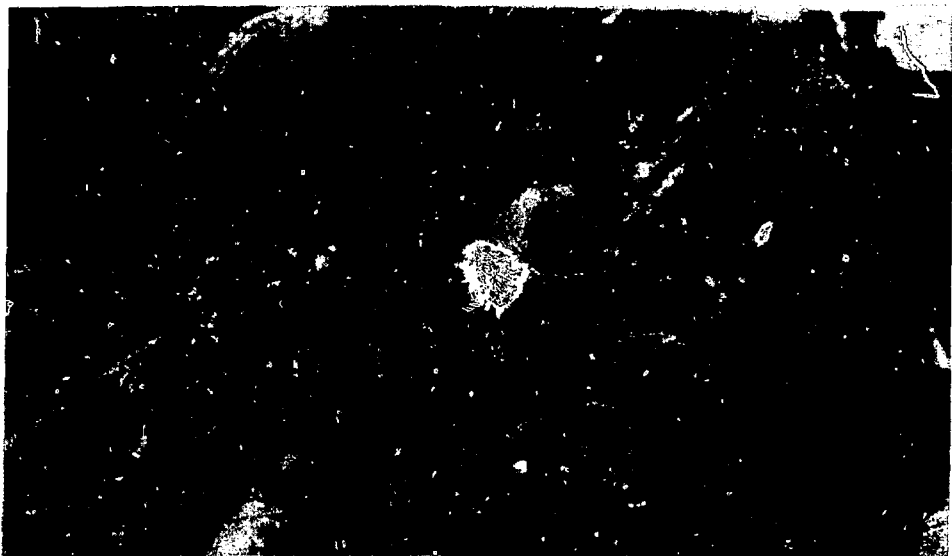


Figure 2. Removing Asbestos Lagging with a Reciprocating Saw. Note the plastic bag used to catch fine particles. The decontamination and de-commissioning worker at left is applying a fine water mist to prevent airborne suspension of particles. Workers were trained for asbestos work and were required to wear protective clothing and respiratory protection at all times.

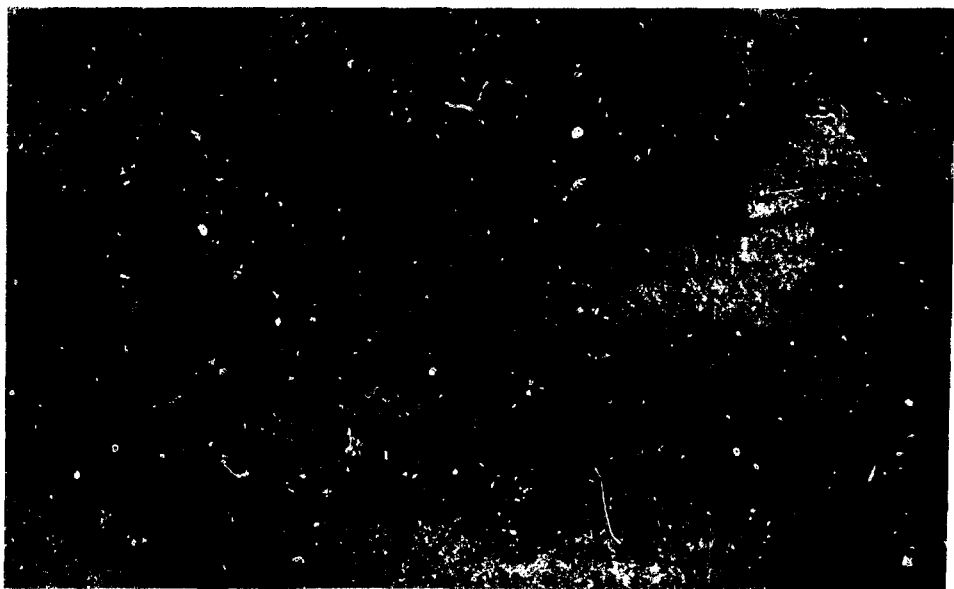


Figure 3. Cutting a Pipe With the Hydraulic Crimper. The crimper was the preferred cutting tool for piping 2 in. or smaller in diameter because it sealed the pipe ends except for two small pinholes.

## DISMANTLING AND SALVAGE OF THE 203-S TANK FARM

The excellent condition and low level of contamination (uranium) of the two 5,000-gal stainless steel tanks made regulated salvage a practical option. Following the removal of all piping and accessories, a greenhouse (fig. 4) was installed over the basin for removing the uranium-soaked lagging. Both tanks had been grouted into their base pads during construction, and D&D workers used jackhammers to free the tanks within the confines of the greenhouse enclosure. A hole was punched into the sump floor to prevent any chance of future liquid accumulation. Following thorough external decontamination and sealing of the tank openings, the greenhouse was removed, and a crane hoisted the tanks out of the pit to a tractor trailer for transport to the regulated spare equipment yard. The concrete basin was backfilled with gravel and soil.

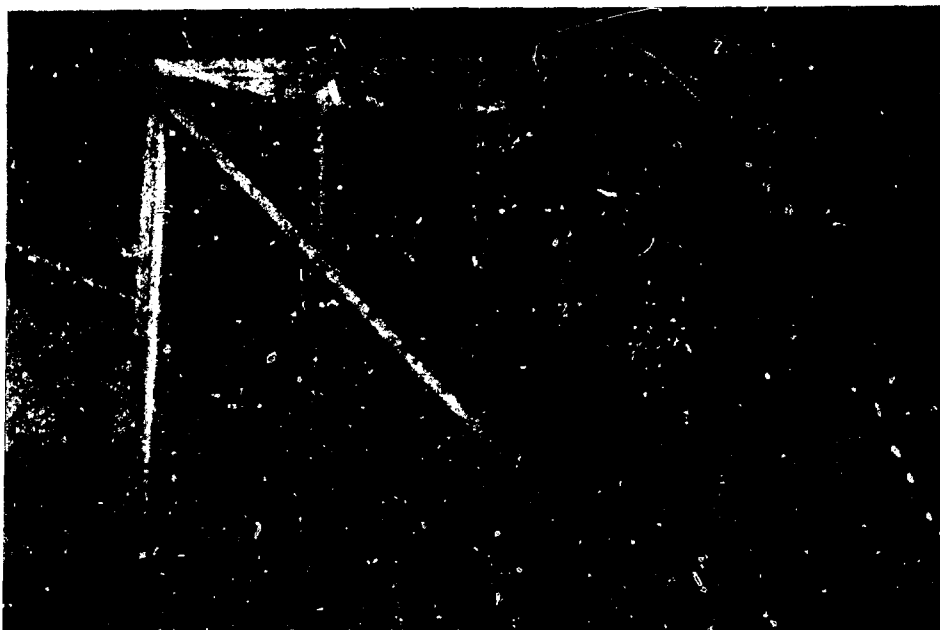


Figure 4. Containment Greenhouse for Removing Contaminated Lagging and Concrete at 203-S.

## DISMANTLING AND SALVAGE OF 205-S

The 205-S facility consisted of an under-ground concrete process vault and an above-ground chemical makeup and control building. The two chemical makeup tanks in the building were empty, but the process vault, the silica gel column, and the neutralizer tank contained a substantial amount of liquid. A submersible electric pump was installed, and low-level contaminated liquid was first pumped from the vault itself, and, after removal of an agitator, from the neutralizer tank. The sealed, all-welded construction made drainage of liquid from the silica gel column more difficult. The column was lifted to grade level by a crane, and a Delta-T Plus explosive-activated self-tapping valved tee was clamped to a recirculation pipe. The column was then lowered part way back into the vault, and the tee was activated by hitting the activating pin with a long-handled hammer. The valve, attached to the tee, was opened with a cable

tied to the handle, and the liquid was allowed to drain. The explosive-activated self-tapping tee provided a very satisfactory means of breaching the pipe under full radiological control. The tee has been used extensively in later D&D efforts such as in the Hot Semi-Works B-cell, where dozens of these units were used. All liquids were pumped to a special regulated tank truck for transport to the double-shell tank farm system.

The neutralizer tank and the two chemical makeup tanks in the 205-S building were salvaged, while the silica gel column was buried. After removal of piping, the two-story chemical makeup building was demolished with a crane and conventional tools. The building floor pad was covered with concrete after sealing all underground pipe penetrations. The underground vault was filled with soil to just below the pipe nozzle level, and the upper part of the vault was filled with concrete.

#### DISMANTLING AND BURIAL OF 204-S

The 204-S decontamination, dismantling, and burial operations presented the most difficult tasks of the D&D project. The 204-S facility contained by far the largest vessels (25 ft dia., 15 ft tall) for disposal. The 204-S facility also held more than 99% of the radioactive fission products stored in the complex. Total decontamination and dismantling of these large, thin-walled (1/4-in. plate) vessels was considered impractical from the personnel radiological exposure as well as the containment standpoint. Transport and burial was chosen as the only practical disposal alternative.

The sluicing operations carried out during the pre-D&D shutdown/standby phase had removed the bulk of the sludge in the four 204-S tanks and left all but tank 204-S-4 in a transportable condition with a solids heel of 1 ton or less. Removal of most of the 10-ton sludge heel in tank 204-S-4 was essential for transport of the vessel, since engineering calculations indicated that an internal tank load of more than 2 tons could lead to unacceptable stresses in the welds joining the tank bottom to the side. The laboratory tested samples of the sludge material and found that although no practical means of chemical sludge dispersion or dissolution was available, a significant amount of sludge softening could be obtained with hot water.

The tank was sluiced with 190 °F water obtained from the 200-West steam plant by tank truck. A pump delivered the water at 150 lb/in<sup>2</sup> pressure from the truck to a commercial tank-cleaning machine inserted via a gasketed flange in the top of the vessel (fig. 5). The resulting slurry was pumped out with an air-driven diaphragm pump to the double-shell tank farm system. Mist and vapors were contained by drawing 2,000 ft<sup>3</sup>/min air from the tank with a 7,000 ft<sup>3</sup>/min-capacity double-HEPA filtered exhaust. The deentrainer and heater ahead of the HEPA filters were adequate to prevent filter wetting or pluggage, and the exhaust operation was entirely uneventful.

The hot water sluicing operation successfully removed more than 90% of the sludge heel. Sluicing was terminated when liquid level, sludge level, and photographic data indicated the 1-ton level had been reached, and the sludge was relatively evenly distributed in the tank. Approximately 113,000 gal of hot water were used in the operation. The large volume of water was required by the intermittent nature of the sluicing operation, by the heavy consistency of the sludge, and by the fully automatic orbital spray pattern of the tank cleaning machine.



Figure 5. Automatic Tank Cleaning Machine Inserted Through the Tank 204-S-4 Roof Near the end of Sluicing Operations.

Most of the sluicing operations were performed remotely, and little personnel radiation exposure was incurred. Dose levels were reduced but were still 2-3 rem/h near the vessel bottom after sluicing. The tank drain line was embedded in the concrete tank base pad and had to be removed. Multiple layers of lead sheeting were attached to the exterior of the tank to protect workers during the necessary concrete-chipping operations. The use of a heavy jackhammer suspended by cable from the top of the vessel allowed the concrete removal to proceed rapidly and to be completed without excessive radiation exposure to personnel.

The 12.5 ton vessels were lifted with a specially designed 6-point spreader bar on to a 13-ft wide, 100-ton capacity truck (fig. 6). The bottoms of the vessels were sprayed with a fixative to hold any loose contamination. The truck bed was prepared with plastic and Sisal Kraft paper. The paper and plastic were taped to the lower 2 to 3 ft of the vessel sides. The vessels were transported approximately 2 mi to a specially constructed burial trench. A new road was built across the Redox railroad cut specifically for this haul, because all existing roads lacked the necessary clearances. Numerous lines had to be raised, and a linemen's truck preceeded all shipments. After unloading from the truck, the lower 5 ft of the vessels were backfilled with dirt to shield workers from the radiation emitted by sludge remnants.

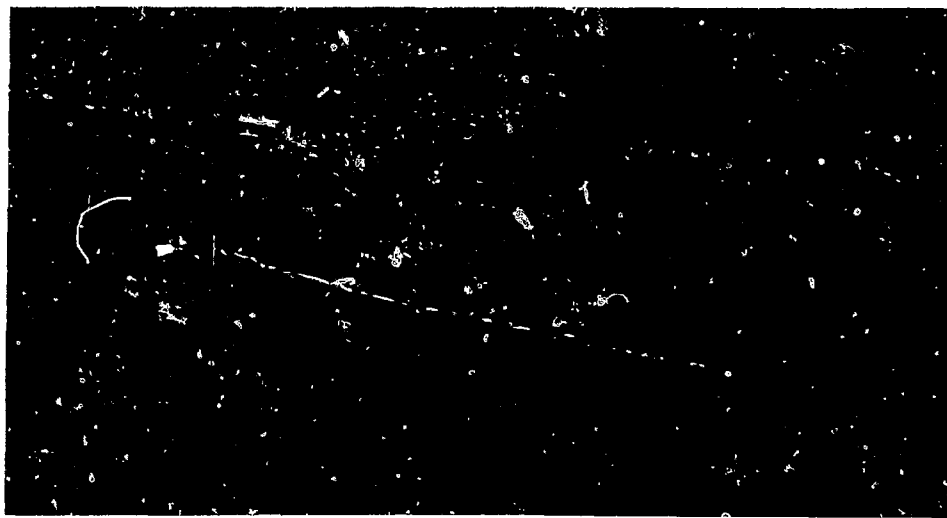
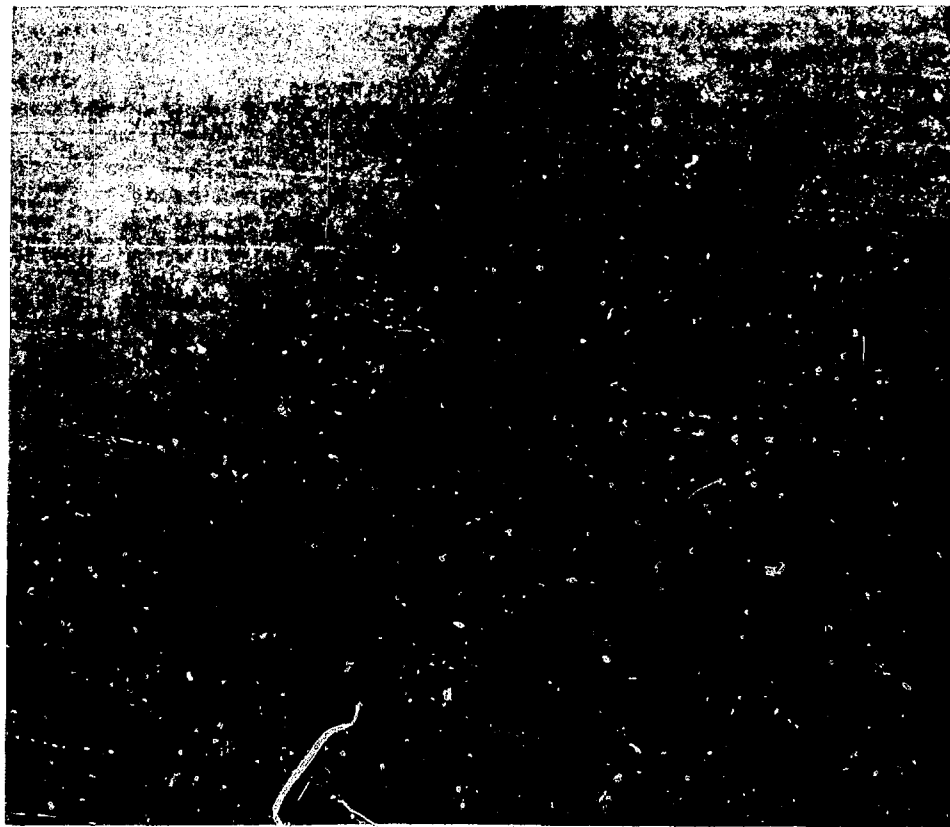


Figure 6. Hoisting a 25-ft Diameter Tank out of the 204-S Basin With a Special 6-Point Spreader Bar (top), and Transporting the Tank on a 100-ton Capacity, 13-ft Wide Trailer Over a Specially Built Haul Road (bottom).

The vessels were filled with concrete and sand to prevent future collapse. A layer of very fluid super-plasticized concrete was first pumped into each tank to seal all sludges remaining on the vessel bottoms. A loader and conveyor then filled slightly moist, dust-free sand into each vessel through six roof openings. A drum-mounted air-ejector HEPA-filtered vacuum cleaner exhausted about 100 ft<sup>3</sup>/min of air from the vessel to maintain a small vacuum. Upon completion of internal fill, the tanks were backfilled with 15 ft of soil cover.

#### FINAL BACKFILL

Final work at the site included spraying the 204-S basin walls with a fixative, demolishing the walls with a wrecking ball (fig. 7), removing the railroad tracks and tank car drain pad, and backfilling the site with 2 to 10 ft of clean soil. Other close-out activities focused on removal of fencing, marking the site, and planting perennial wheatgrass to stabilize the soil cover.

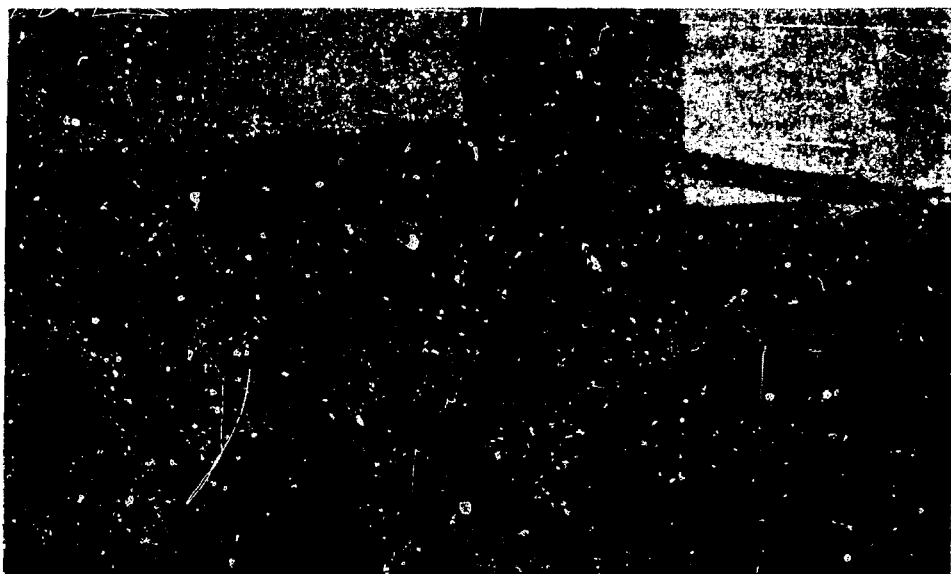


Figure 7. Initiating 204-S Tank Basin Demolition with a Wrecking Ball.

Personnel radiation exposure was kept well within Rockwell's established exposure guidelines of 300 mrem/wk and 1.25 rem/calendar quarter. Most exposure was encountered during the sluicing and concrete chipping operations at tank 204-S-4. Exposure guidelines were approached on only a few days and were never exceeded. Approximately 9 Ci of mixed fission products were removed from the site and buried. An estimated 0.2 Ci of mixed fission products remain in buried piping, concrete, and soil in the 203-S, 204-S, 205-S area.

Total cost for the D&D project was \$1.025 million. This cost figure includes all charges for D&D workers, craft support, transportation, earth moving, burials, engineering support, radiation monitoring, laundry, overhead charges, and rentals. Originally projected costs totalled \$1.3 million. The work was completed three months ahead of schedule.

PLACEMENT OF THE RADIOCHEMICAL PROCESSING PLANT  
AT OAK RIDGE NATIONAL LABORATORY INTO A SAFE STANDBY CONDITION

D. W. Holladay, C. D. Bopp, A. J. Farmer, J. K. Johnson,  
C. H. Miller, B. A. Powers, and E. D. Collins  
Oak Ridge National Laboratory  
Post Office Box X  
Oak Ridge, Tennessee 37831

ABSTRACT

Extensive upgrade, cleanup, and decontamination efforts are being conducted for appropriate areas in the Radiochemical Processing Plant (RPP) with the goal of achieving "safe standby" condition by the end of FY 1989. The ventilation system must maintain containment; thus, it is being upgraded via demolition and replacement of marginally adequate ductwork, fans, and control systems. Areas that are being decontaminated and stripped of various services (e.g., piping, ductwork, and process tanks) include hot cells, makeup rooms, and pipe tunnels. Operating equipment that is being decontaminated includes glove boxes and hoods. Replacement of the ventilation system and removal of equipment from pipe tunnels, cells, and makeup rooms are accomplished by contact labor by workers using proper attire, safety rules, and shielding. Removal of contaminated ductwork and piping is conducted with containment enclosures that are strategically located at breakpoints, and methods of separation are chosen to conform with health physics requirements. The methods of cutting contaminated piping and ductwork include portable reciprocating saws, pipe cutters, burning, and plasma torch. Specially designed containment enclosures will be used to prevent the spread of radioactive contamination while maintaining adequate ventilation.

INTRODUCTION

Work is under way to place the RPP into a "safe standby" condition by the end of FY 1989. This condition is similar to that defined as a type of decontamination and decommissioning (D&D) by the DOE Surplus Facilities Program, except that packaged, solidified radioactive materials will continue to be stored in the RPP in shielded, isolated storage wells. Services are being upgraded or replaced to maintain the plant in a safe mode while other areas are being decontaminated and isolated.

The RPP (Fig. 1), which is operated by the Pilot Plant Section of the Chemical Technology Division, includes the laboratory, glove box, shielded hot cell, and shielded storage facilities that are contained in Building 3019, as well as adjacent service facilities, which include the 3020 Ventilation Stack and the Building 3100 Storage-Vault. Decontamination and decommissioning of the High Radiatic Level Analytical Facility (HRLAF), located at the western end of Building 3019, is not discussed in this paper. This facility is now inactive and is the responsibility of the Analytical Chemistry Division. It is scheduled for decontamination and decommissioning starting in FY 1986. All facilities within the RPP are fully safeguarded such that multikilogram amounts of the fissile isotopes of uranium and plutonium may be stored and processed. Equipment in the shielded hot cells is operated remotely but is directly maintained. Many of the facilities have been in use for over 40 years. During this time, several well-known methods for nuclear fuel reprocessing (Purex, Thorex, Fluoride Volatility, etc.) have been developed and operated.

Since 1963, the RPP has been designated as the national repository for  $^{233}\text{U}$ . Currently, in addition to facilities for processing high-purity  $^{233}\text{U}$ , the RPP contains the Consolidated Edison Uranium Solidification Program (CEUSP) Facility, which is being used to convert a special batch of uranium (containing both  $^{235}\text{U}$  and  $^{233}\text{U}$ ) from a nitrate solution to a solid oxide form. Also, the RPP contains the Plutonium-Uranium Microsphere Preparation (PUMP) Facility, which was used recently to prepare kilogram quantities of mixed oxide microspheres. In addition, several chemical laboratories and glove box facilities in the west end of Building 3019 were operated until recently by the Analytical Chemistry Division and were used to perform a variety of radiochemical analyses.

The piping, equipment, and hot cells are still contaminated with residual fission products (primarily  $^{137}\text{Cs}$  and  $^{90}\text{Sr}$ ). However, the major contamination sources are alpha emitters, particularly plutonium, uranium, and the decay daughters of  $^{232}\text{U}$  (Fig. 2). The key daughters governing most safety considerations are  $^{220}\text{Rn}$  and  $^{208}\text{Tl}$ . The  $^{220}\text{Rn}$  is a gaseous decay product (half life =  $\sim 1$  min) that creates a continuing source of airborne contamination. The  $^{208}\text{Tl}$  emits a 2.6-MeV gamma ray, which dictates the degree of shielding that is needed. Also, the laboratories, hoods, and hot cells that were used for analytical chemistry purposes are contaminated with transplutonium actinides as well as fission products. The off-gas ducts and liquid drains from these facilities have been corroded extensively by the variety of chemicals that have been handled over the years.

This paper will describe the decontamination, decommissioning, and equipment replacement activities that are being carried out simultaneously with the solidification and storage of all radioactive materials now in liquid form. Replacement of the ventilation system and removal of equipment from pipe tunnels, cells, and makeup rooms are accomplished by contact labor using proper attire, safety rules, and shielding. Removal of contaminated ductwork and piping is conducted with containment enclosures strategically located at break-points, and methods of separation are chosen to conform with health physics requirements. Containment enclosures are designed to prevent the spread of radioactive contamination while maintaining adequate ventilation. The ventilation system, which is marginally adequate and must be improved to maintain containment with minimal surveillance, is being upgraded via demolition and replacement of ductwork, fans, and control systems. Hot cells, makeup rooms, and pipe tunnels are being decontaminated and stripped of various services (e.g., piping, ductwork, and process tanks). Operating equipment that is being decontaminated includes glove boxes, sample blisters, and hoods. Methods of cutting contaminated piping and ductwork include portable reciprocating saws, pipe cutters, burning, and plasma torch.

## BASIC CONSIDERATIONS

Three activities are necessary for all decontamination and contaminated-equipment removal efforts: (1) detailed strategic planning of jobs before action is initiated; (2) design and construction of temporary containment, if needed; and (3) worker dress-out and preparation. The purpose of the intensive planning is to provide procedures sufficiently detailed to accomplish the job while preventing release of radioactive materials or excessive exposure of personnel. Breaching of any pipe, ductwork, or equipment requires extensive preparations to determine the condition of the equipment. Temporary containment enclosures are used to prevent spread of contamination when a contained area or piece of equipment must be opened. Worker dress-out is designed to give maximum protection from the intake of radioactive materials, particularly alpha contamination, which constitutes the major hazard within the plant.

## Strategic Planning

For most nonroutine decontamination, a sequence of work is used to direct the work personnel. When there are high levels of either contamination or penetrating radiation, or under difficult working conditions, workers follow a written procedure that (1) gives step-by-step details specifying how and when the work is to be done and (2) tells workers what to expect and what action to take if problems arise.

The system for doing work on contaminated equipment in the RPP includes description of the work to be done, surveys and permits, work sequences, precautions, and procedures for abnormal conditions. The steps are as follows:

1. A work request is issued giving details for the proposed work to the Maintenance Coordinator (MC). (The MC is the liaison between operations and maintenance groups.)
2. If the work is complex, a conference is arranged. Input from appropriate sources is solicited, and a copy of the proposed work plan is provided to concerned parties.
3. The participants meet and reach a consensus on the adequacy of the work plan. If the plan is acceptable, the work may begin.
4. If the plan is not acceptable, changes are suggested by the participants, and the requested revisions or changes are then made by the originator. Any controversies are resolved by management.
5. A final review meeting is usually held to ensure that all participants are aware of the work plan.

## Temporary Containment

Provisions for temporary containment are required when contaminated systems are opened, such as during replacement of glove box windows or when pipes or ducts are cut. The function of temporary containment is to prevent the spread of contamination that may be exposed during the operation. Control is achieved by proper construction, so that there are no significant paths of leakage to the outside, and by providing for a negative pressure inside the containment relative to the surrounding area, so that ventilation flow is always into the containment enclosure. In some applications, HEPA or roughing filters are built into the walls of the tent or containment structure to prevent backflow of contamination. An adequate flow of air into the work area also enhances protection of the working personnel.

Temporary containments are constructed according to standard designs, varying from single-layered plastic tents constructed on a wooden framework to multiple-layered, heavy-gauge plastic tents constructed on heavy frames, to plywood-framed enclosures. During demolition and replacement of the ventilation system ductwork outside the building, multiple containment enclosures were built 15 m off the ground on scaffolding that extended continuously for 30 m (see Fig. 3). Figure 4 shows a typical containment enclosure constructed for contamination control during piping work.

## Worker Dress-out and Preparation

The basic dress-out procedure has been developed over a period of years and is designed to provide two layers of protection between the worker and the source of contamination. This procedure allows the outer layer of protection to be removed when exiting the contaminated area and thus limits the spread of contamination. The general procedure for dress-out is achieved by adding articles of clothing in the following order:

1. An inner pair of coveralls with contamination zone (C-Zone) shoes. Personnel pocket dosimeters and badge dosimeters are worn on the inner coveralls; however, in some cases, a self-reading pocket dosimeter may be worn on the outer layer when it is necessary to monitor the gamma radiation dose as the job progresses.
2. Inner shoe covers, which are taped to inner coveralls.
3. Outer coveralls.
4. Outer shoe covers, which are taped to the outer coveralls.
5. A pair of rubber gloves worn over a pair of thin cotton gloves; the rubber gloves are taped to the inner coveralls.
6. An outer pair of rubber gloves, which may be taped to the outer coveralls; however, for jobs where gloves will become contaminated rapidly, the outer coverall sleeves are taped to the inner coveralls and the gloves are left free.
7. A respirator.
8. A hood, which is taped on. Care must be taken to give enough space for freedom of movement.

The following items may be worn for specific jobs:

1. a vinyl suit over coveralls for short-term (<30-min) work or self-contained air-supplied suits for longer periods, in instances where water or other liquid may be encountered or very high levels of surface and/or airborne contamination are involved;
2. a safety harness and safety line outside of coveralls; and
3. overshoes, which are placed over shoe covers in cases where work on ladders or elevated platforms is conducted.

#### UPGRADING OF THE VENTILATION SYSTEM

Prior to the decision to move toward solidification and storage of all radioactive materials and placement of the processing equipment in standby condition, the ventilation system in the RPP was in need of upgrading. The ductwork in the system was predominantly carbon steel and had been corroded significantly, particularly in the western branches through which the Analytical Chemistry Division facilities were vented. Moreover, the emergency backup system for the two electrically driven fans was a single, steam-driven fan, whose operation required switching a series of electrically controlled, mechanically operated damper valves. Thus, the entire backup system was subject to failure if either the steam system, the turbine fan, the electrical controls, or the mechanically operated dampers failed. In addition, if an electrical power outage occurred, no further backup capability existed and the return to normal operation would require action by operating personnel who were not present during off-shift hours. The multifold dependency of the backup system mandated a high degree of surveillance and maintenance.

Thus, the decision was made to install a new ventilation off-gas system. This is now in progress. The replacement (Fig. 5) is to be more simply operated and will have a much higher degree of backup capability. In normal operation, it will have two electrically driven fans, each of which will have a backup. Any one of the four will be able to maintain safe conditions (negative pressures in the containment areas) if necessary. Emergency power for the fans will be provided by two diesel-powered generators, with each generator serving two fans. Rather than controlled dampers, only a simple backflow preventer damper in each fan exhaust line will be used. The new electrical control system will allow repeated switching of fans, as necessary, to provide continued backup capability. The new fans and ductwork will be of stainless steel construction.

The ductwork routing inside the plant is a patchwork of additions made over the years and does not provide optimum ventilation for all primary and secondary containment areas. Therefore, some rerouting will be necessary. Also, holes through which inleakage of unconditioned air occurs must be sealed to increase the negative pressure to desired levels.

#### Replacement of Contaminated Equipment

The ductwork being replaced in the ventilation system upgrading is not classified as transuranic waste. Most of it is contaminated internally with 30 to 700 Bq/g of transuranic plus nontransuranic radionuclides. Contamination levels in portions of the ductwork upstream of the filters range to >1500 Bq/100 cm<sup>2</sup>. These levels dictate that, during the demolition procedures, the following dress-out is required: two pairs of coveralls, full mask, shoe covers, gloves, etc., as described earlier. Additionally, temporary containment enclosures are required to prevent the spread of contamination to the environment and to protect the workers from excessive internal contamination.

Much of the ductwork being demolished is in an elevated position and hence requires the construction of elaborate scaffolding to provide the workers easy access to the ductwork. Containment enclosures are constructed at each place along the scaffolding where the duct is to be separated. Separation is by flange disconnection or by cutting with a reciprocating saw; no burning is allowed. Containment enclosures are constructed not to be airtight but, rather, to be of sufficient integrity to minimize the probability that any dust reaches the environment. The tops are constructed to be removable so that, once the ductwork is broken free and capped, it can be lifted by a crane from the scaffold area to a truck below.

During demolition, it is necessary for all ductwork that is removed to be immediately capped at all ends, double wrapped in plastic, and taped. Ventilation is maintained through the ductwork until its removal. This is done by beginning the disassembly at the farthestmost point from the fan (vacuum source), and proceeding toward the fan. The end of the duct at which the last flange break or saw cut took place is capped so that the inrush of air that occurs when the next flange is broken or the next cut is made will retain the contaminated dust inside the duct.

#### CLEANOUT OF HOT CELLS, GLOVE BOXES, AND CHEMICAL LABORATORIES

After completion of the processing operations to solidify and store all radioactive materials now in liquid form, the hot cells, glove boxes, and chemical laboratories will be cleaned and decontaminated sufficiently to permit secure containment by means of periodic custodial surveillance and maintenance.

#### Hot Cells

The RPP contains a bank of seven heavily shielded hot cells in which remotely operated and directly maintained radiochemical processing operations have been performed for many years. Most of the cells have square floor areas (~6 m x ~6 m) and are ~8 m high. One cell has only half the normal floor area, and two others are interconnected to form a large work space. All are shielded by 1.5-m-thick concrete walls and a 1.5-m-thick roof.

Most of the hot cells have personnel entry doors at the floor level and 2.7-m<sup>2</sup> equipment hatches in the roof. Equipment is transferred through the hatches to and from the high bay area (called "the penthouse") above the cells

by means of a 10-ton bridge crane located in the penthouse. The chemical processing equipment consists of a variety of tanks (with a typical capacity of  $\sim 1 \text{ m}^3$ ), columns (5 to 15 cm in diameter and up to 11 m tall), pumps, and piping. There are numerous piping penetrations in the cell walls and roof which connect to accessory tanks, pumps, instrumentation, and utility services located above and adjacent to the cells. Ventilation air enters through the bottoms of most of the cells and is exhausted through ducts at the tops of the cells.

The cleanout that is planned for the hot cells will consist of (1) removing selected equipment, tanks, and piping from the cell floors and walls; (2) thorough flushing of the cell floors with water and detergents; (3) thorough flushing of the internals of the process equipment with nitric acid solutions and water, followed by purging with air to dry the equipment; (4) conducting a final radiation and contamination survey and mapping; and (5) disconnecting unnecessary utilities and process instrumentation. Cell lighting, fire prevention sprinklers, floor sump liquid level instrumentation, and cell ventilation air flow will continue in-service. The sump levels and cell pressures will be periodically monitored by surveillance personnel. Annual cell inspections are planned.

Over the past years of operation, numerous decontaminations of the RPP have been performed. These have involved surface cleaning, equipment removal, piping disconnection and removal, etc. Usually, the goal was to improve background radiation levels and to decrease surface contamination so that the operations could be continued or new equipment could be installed.

In one of the more extensive efforts, Cells 6 and 7 were decontaminated by a factor of  $10^3$  by removal of loose equipment, debris, and shielding blocks and by flushing with about  $400 \text{ m}^3$  of various types of decontaminating reagents. This was accomplished during a 5-month period. The remaining traces of trans-ferrable contamination were then immobilized. These efforts reduced the general beta-gamma background to 30 mR/h and the long-lived alpha contamination in the air to  $8 \times 10^{-13} \text{ } \mu\text{Ci/mL}$ .

Flushing of the cell surfaces was accomplished by means of a manually applied Seller's "super booster" hydraulic jet. Reagents (detergents, degreasing solutions, acids, caustics, etc.) were prepared and heated in feed tanks and jetted through a wand of 1.3-cm-diameter pipe fitted with a nozzle. Using high-pressure steam metered through the hydraulic jet, a flow of 10 L/s was delivered at a pressure of 1.4 MPa. The Seller's jet is still used periodically to clean the cell floors and will be used again in the planned cell cleanout.

### Glove Boxes

The RPP contains numerous glove boxes, which are used for a variety of operations with plutonium, depleted uranium, and freshly purified  $^{233}\text{U}$ . The most significant group is the  $^{233}\text{U}$  Oxide Conversion Facility, which is used to produce  $^{233}\text{U}_3\text{O}_8$  or  $^{233}\text{UO}_2$  powder by the ammonium diuranate precipitation-calcination process. After solidification and storage of all  $^{233}\text{U}$  have been completed, the glove boxes and the processing equipment contained in them will be cleaned thoroughly and placed into standby condition. The glove ports will be sealed with metal covers, and the ventilation system, which maintains negative pressure inside the boxes, will be left intact. Liquid supply pipelines and all unnecessary process instrumentation will be disconnected. However, instrumentation to measure the negative pressure inside certain strategic boxes will be maintained and kept under surveillance.

Glove boxes that are deteriorated, or are otherwise no longer useful, will be decontaminated (if possible) to the extent that the remaining contamination can be classified as nontransuranic waste. These boxes will then be disconnected, packaged, and sent to the low-level-waste burial ground. If a box cannot be decontaminated sufficiently, it must be packaged in specially designed metal boxes for shipment to the Waste Isolation Pilot Plant in Carlsbad, New Mexico.

#### Chemical Laboratories

The chemical laboratories in the RPP have been used primarily for performing radiochemical analyses on many nuclides and solutions. Most of the preparative work, such as the separation of  $^{90}\text{Y}$  from  $^{90}\text{Sr}$ , was performed in fume hoods through which the laboratory room ventilation is exhausted. The analytical chemistry operations have been transferred to newer facilities at ORNL. Other low-activity-level process development work may be done in the RPP laboratories; therefore, the laboratory hoods have been cleaned and decontaminated. To perform the decontamination, direct contact methods were used by workers dressed in protective attire (as previously described) and guided by health physics surveys and monitoring. Liquid drain pipelines from the laboratory facilities have become contaminated and deteriorated; therefore, their replacement is being planned. A phased replacement, such as that being used for contaminated ducts in the ventilation system, will be used for the liquid drains.

#### REMOVAL OF CONTAMINATED SAMPLE BOTTLE CONVEYOR

During the peak years of the development of nuclear fuel reprocessing flowsheets at the RPP, a shielded conveyor system was installed to transport sample bottles from the processing area to the analytical chemistry facilities. The conveyor consists of a series of sample bottle carrier cups mounted between two chains that are moved by means of an electric drive motor from one terminal to the other. This configuration is surrounded by steel shielding (see Fig. 6) that is 10 to 50 cm thick.

During the years when the conveyor was used, several sample bottles fell out of the cups during transport and were broken inside the conveyor housing. Thus, the inside of the conveyor housing is highly contaminated with a variety of radionuclides, including actinides and fission products. In addition, rain water has leaked into certain areas of the housing and has caused the generation of contaminated rust debris. A high degree of health physics surveillance is required for the conveyor housing, and intensive decontamination efforts have been required on several occasions. This will not be practical after standby conditions have been reached; therefore, dismantling and removal of at least part of the conveyor system are being planned. Five methods of cutting the steel plates have been considered: plasma torch, portable hack saw, portable band saw, acetylene torch, and abrasive wheel. Current planning is focused on the portable hack saw or a portable acetylene torch; the hack saw is the more desirable choice because it will be much less likely to spread contamination to the air. Other provisions that will be needed to accomplish the job include (1) a 30-m-high, 20-ton crane; (2) extensive containments inside and outside the penthouse; (3) significant structural alterations to the supports for one corner of the penthouse; (4) removal and replacement of sheet metal from the sides of the penthouse corner; and (5) coordination of riggers, crane operators, iron workers, sheet metal workers, carpenters, health physics surveyors, and operations personnel.

## SUMMARY

In summary, work is under way to put the RPP at ORNL into a safe standby condition by the end of FY 1989. This project requires that all radioactive materials now in liquid form be solidified and stored and that all processing facilities in the shielded hot cells, glove boxes, chemical laboratories, and accessory areas be decontaminated and cleaned to the extent that continued surveillance and maintenance activities are minimized. The ventilation system, which maintains a negative pressure in all primary and secondary containment zones, is being upgraded and will be maintained. In addition to special decontamination tools, basic considerations for the D&D activities include detailed strategic planning in advance of actual work, provision of temporary containment when needed, and careful attention to the details of worker dress-out and preparation.

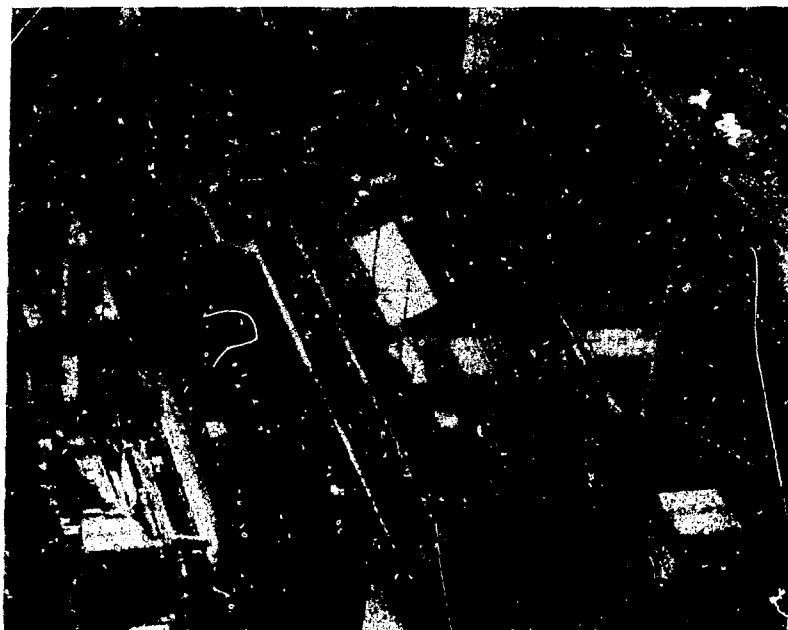


Figure 1. Aerial View of the RPP.

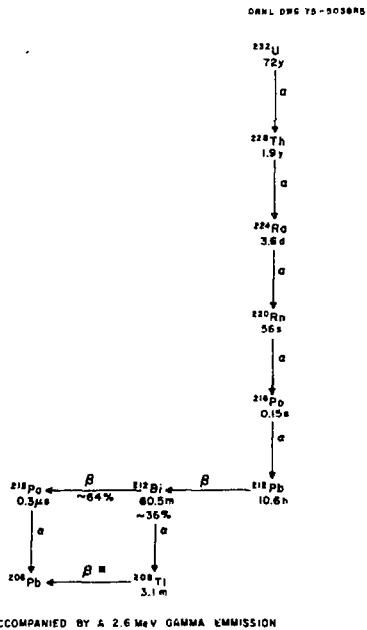


Figure 2. Decay Chain for  $^{232}\text{U}$ .



Figure 3. Scaffolding for Ductwork Replacement.

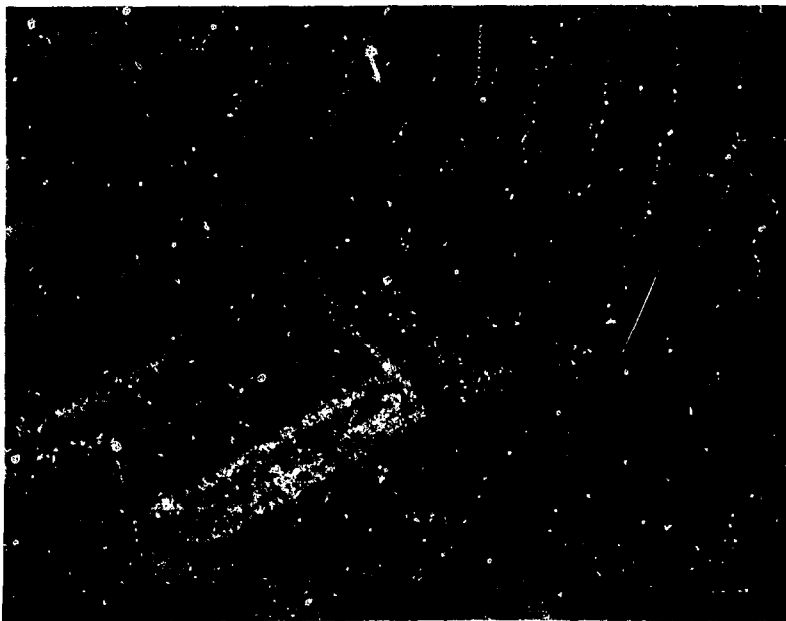


Figure 4. Typical Containment Tent for Hot Work.

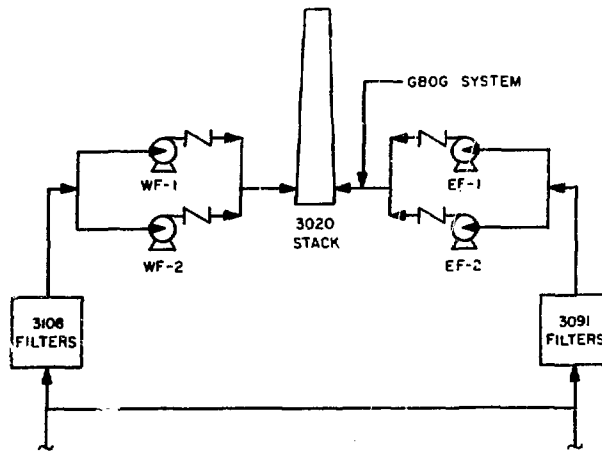


Figure 5. Schematic of New Ventilation System.

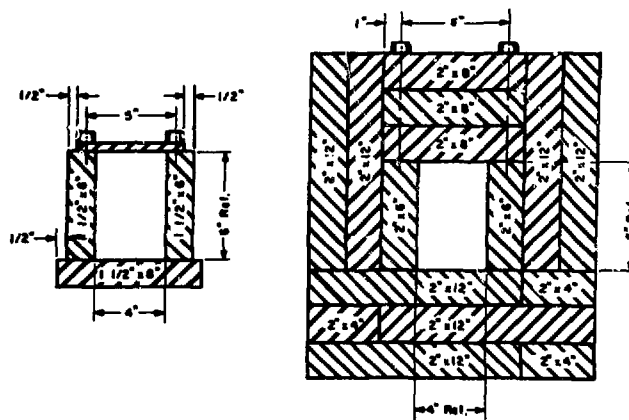


Figure 6. Typical Cross-Sections of Sample Conveyor.

399/400

MEASUREMENTS AND STATUS OF A DECOMMISSIONED  
EXPERIMENTAL REACTOR - SEFOR

L. West, C. Ragland, and T. Welty  
University of Arkansas  
Mechanical Engineering Department  
Fayetteville, AR 72701

ABSTRACT

The Southwest Experimental Fast Oxide Reactor (SEFOR) was decommissioned (by mothballing) in 1972 after two years of operation. Originally funded by the Southwest Atomic Energy Associates and the AEC, the Na-cooled reactor was operated by General Electric under contract. Although fuel was removed during decommissioning, a considerable amount of radioactive material (primarily Co-60 in activated steel) remains. Several years after decommissioning, ownership was transferred to the UA for long-term health physics monitoring and security. Recent GM counter surveys and smear analyses indicate no significant problems with old liquid and gaseous storage tanks. Residual Na is being examined for activation product inventory. Gamma spectroscopy of soil samples in 1984 raised concern of possible ground contamination in one area, prompting further gamma spectroscopy work. Results of surveys, smear analyses, and gamma spectroscopy will be presented for the old reactor vessel, waste storage tanks, soil samples, and sodium heat exchangers. SEFOR represents an unusual situation since the present owner (UA) assumed responsibility (at the request of the Arkansas State Health Department) after the facility was mothballed and after the original owner (SAEA) became defunct.

401/HO2

## HEALTH PHYSICS OR INDUSTRIAL HYGIENE

Jere K. Allen, Vice President  
VIKEM Industries, Inc.  
12 Mulliken Way  
Newburyport, Massachusetts 01950

### ABSTRACT

With the continuing addition to the list of substances exhibiting carcinogenicity, the working health physics individual is being increasingly faced with situations where knowledge of industrial hygiene practices is necessary to insure protection of workers under his control. The problems will continue to multiply as events such as decommissioning of entire facilities and the advent of new decontamination and radwaste processing methods introduces a variety of new substances to the inventory of hazardous materials present in the working environment.

A variety of specific situations will be discussed illustrating scenarios where industrial hygiene protocols must be implemented to insure complete safety for workers.

### INTRODUCTION

A six inch valve in one of the main flow paths for condensate water movement at plant X develops a packing leak. When maintenance goes down to repair it, they discover that the insulation covering the valve contains asbestos. Using correct procedure, the asbestos is removed, the valve repaired, and new, non-asbestos insulation is installed.

During initial health physics survey prior to the start of work loose contamination at the level of 200 pCi/100 cm.<sup>2</sup> was found on the surface of the insulation. When the material was brought out following removal further examination found fixed contamination of 45.9 Becquerels per 100 cm.<sup>2</sup>. No contamination was found other than on the originally exposed surfaces. The total volume of the material is 3.7 ft.<sup>3</sup>.

Question #1: How should the maintenance personnel have been dressed while performing the work? Based on the minimal radioactive contamination levels cited, many health physics departments would require no protective clothing whatsoever, or, at best, coveralls, gloves and shoe covers. Certainly no respirators are called for.

Application of Environmental Protection Agency guidelines for worker protection mandates "Special whole body clothing, and head, hand and foot coverings". Further, "workers are to be provided with respiratory equipment", selection to be based on 8-hour time weighted average of fibers per cc of air.

Question #2: How should the area have been prepared prior to work? Radiologically, spill control is the only requirement. The industrial hygienist will require significantly more, including:

1. Isolation of the work area ventilation system to prevent contamination and fiber dispersal to other areas of the building during the work phase.
2. All moveable objects present in the proposed work area transferred to a new location outside the proposed work area. Anything remaining in the work area must then be sealed with polyethylene sheeting.

3. The asbestos work area be isolated from the rest of the building, and access restricted.

Question #3: How do the workers exit the work area? Again, from a radiological standpoint, and assuming the use of protective clothing, the workers will undress at the step-off-pad, drop cotton coveralls into a laundry drum, disposables into the hot trash, frisk their hands, and be on their way. In contrast, OSHA requires that the worker:

"Remove the disposable coveralls, head covers, and footwear in the work area before leaving the work area. Still wearing their respirators, proceed to the showers and remove their respirators while showering with soap and water."

Additional OSHA regulations concerning asbestos removal detail area posting requirements, packaging and labeling specifications, and area cleanup procedures, most of which resemble nuclear methods but which technically must be complied with in addition to the quote "normal", unquote activities.

The focus on asbestos may appear to be unnecessary since most power stations do not have the material present on the site. Consider, however, the number of plants constructed prior to the mid 70's when use of asbestos ceased. Particularly, the early units such as Dresden I, Indian Point I, Yankee Rowe and Shippingport, which are now in planning or active decommissioning, are those most likely to have it. In fact, the first major contract put out for bids at Shippingport was for asbestos removal. The specifications requested the removal and disposal of an estimated 4500 cubic yards, some of it highly contaminated, and with a portion of the work to be performed in radiation and high radiation areas.

Concern for worker safety was evident in the original Request for Proposal in that procedures for work performance were detailed. It was also evident, however, that in this instance ALARA concerns were not as important as industrial hygiene. To illustrate, the prospectus called for the removal of lead shielding, stripping of asbestos, and then replacement of shielding, all to be accomplished in an area where the contact exposure rate on the affected piping was 800 mR/hour through the shielding. I suggest this requirement was written by an individual familiar with OSHA but not NRC regulations.

A simple lack of awareness was responsible for the next incident. In this case a decontamination technician was sent to a storage area to obtain some TSP to use as a cleaning agent. He was instructed to open one of the bags on a pallet and bring back a small quantity in his pail. Having used the material before, he knew it was a milky crystalline solid. This is in fact what he came back with. Unfortunately, when the material was mixed with water and used to perform the cleaning it quickly ate its way through plastic pants and gloves and proceeded to attack the skin underneath. Only quick flushing with water saved several workers from rather severe burns.

Obviously the technician brought back the wrong material. Subsequent investigation found several different chemicals, all in bags on pallets, stored in the same place. Additionally, the chemical names were similar and the packaging material and printing were nearly identical. Who's fault was it? The stores person who put very similar items close together? The technician for not positively identifying the substance? The supervisor for not being more explicit in his instructions? I don't think it is important to level blame; but rather to learn a lesson. Since health physics often directs decontamination work, those involved would be wise to establish tight controls over the storage and availability of materials, or at least to be aware of conditions as described above in order to be able to prevent unpleasant surprises.

In the last few years several nuclear stations have elected to purchase and operate their own special decontamination units rather than rely on contractors services. The

consequence of this is that station personnel, again often under the direction of the health physics department, must transport, mix, utilize and dispose of substances classified as hazardous materials. An example is phosphoric acid solution, used as the bath in electropolishing systems. The Association of American Railroads Emergency Action Guides describe phosphoric acid as: —"although not flammable, it may react with certain metals to evolve flammable and potentially explosive Hydrogen gas." Further, —"it reacts with water with some evolution of heat, but reactions with strong caustics can cause violent spattering and considerable heat release. It is a corrosive substance which may irritate or burn bodily tissues upon contact. Products of decomposition may include acid fumes." Also, —"may attack some plastics, rubber and coatings."

The last sentence introduces another factor - that normal nuclear protective clothing may not be sufficient to protect the worker from chemical hazard. Since an emphasis in radworker training is to impart a feeling of safety to the uninitiated worker, a false sense of security could lead to dangerous activity.

How many health physics technicians know the chemical hazards of machine shop degreasers, dry cleaning chemicals, freon, asphalt or other solidification media, LOMI and other internal systems decontamination solutions, resin regeneration acids and caustics or water purification materials? Is there a need for them to know? The writer feels very strongly that there is. As previously recounted scenarios illustrate, potential exists for injury to workers through simple lack of information. The basis for the health physics profession is personnel protection. It behooves us to be the very best we can.

## TRAINING REQUIREMENTS FOR HEALTH PHYSICISTS IN THE DECONTAMINATION/DECOMMISSIONING FIELD

E. D. Copenhaver and D. C. Parzyck  
Environmental and Occupational Safety Division  
Oak Ridge National Laboratory  
P.O. Box X  
Oak Ridge, Tennessee 37831

### ABSTRACT

While a significant decrease in the number of new facilities requiring health physics surveillance has occurred in the past decade, there has been a tremendous increase in the need for health physicists to fill regulatory requirements at existing facilities and the Decontamination and Decommissioning requirements of older facilities nearing the end of their operational lifetime. There is a continuing long-term need to provide trained health physicists with the special skills to meet these requirements.

Decontamination and decommissioning programs require both basic and specialized health physics activities to be performed (1) to evaluate the radiation environment of the facility under consideration, (2) to establish the standards to which cleanup activities must be pursued, and (3) to adequately protect the personnel involved in the cleanup. Performance-based training, based on job task analysis, is an appropriate way to define the different types of health physics expertise required for D&D programs.

Materials have been developed to describe potential job requirements in the radiation protection field, and the appropriate training goals to meet these requirements.

### NEED FOR HEALTH PHYSICISTS IN DECONTAMINATION & DECOMMISSIONING

Though the number of new facilities requiring health physics surveillance has been declining significantly during recent years, there has not been a corresponding decrease in the demand for health physicists in the workforce. More stringent regulatory requirements for existing facilities account for part of the increased demand for radiation protection personnel. Additionally, there is further demand caused by radiation protection requirements in the decontamination and decommissioning (D&D) of older facilities nearing the end of their operational lifetime.

A recent study of manpower trends and training requirements for radiation protection personnel in the DOE contractor system revealed some interesting patterns in manpower projections and training program content (1). The study defined the position of health physics technicians in 34 DOE contractor organizations according to the functional character of the referenced organization. The categories of employment areas used in the study were reactors, fuel fabrication, fuel reprocessing, fuel enrichment, weapons fabrication and testing, waste processing and management, accelerators, and general research and development. An "other" category was also used to

address the 5% of the surveyed health physicists in types of activities other than those listed. During the period covered by the study, fiscal year 1980 through fiscal year 1983, the size of the total surveyed workforce remained essentially constant.

Table 1 shows the relative distribution of health physics technicians and individuals with assigned health physics responsibilities in each of the four years of the study. The category of health physics technician refers to those individuals who have radiation protection activities as their primary responsibility while individuals designated "assigned operators" have health physics activities as an assigned portion of their jobs. An examination of Table 1 reveals that the largest portion of assigned operators in the organizations surveyed are employed in waste processing and management. Further, radiation protection personnel in this area are far more likely than health physicists employed elsewhere to be performing radiation protection duties as a collateral responsibility. The study shows that growth in radiation protection manpower has occurred in those facilities whose functions are primarily waste processing and management, fuel reprocessing and weapons fabrication, and testing. In addition, the workforce in the area of waste processing and management is increasing in proportion to the total operating workforce while the proportion in other areas may be decreasing. The waste processing and management area demonstrates one of the highest rates of growth of any of the areas surveyed.

#### GENERAL HEALTH PHYSICS TRAINING PROGRAMS

The Oak Ridge Associate Universities (ORAU) survey also summarized those technical elements included in the formal training programs for health physics technicians and assigned operators in the various organizations surveyed. Table 2 summarizes information derived from the survey. Twenty-three facilities had formal health physics training programs. All facilities teach elements of plant radiation safety policies, basic units, and respirator use. A very large number of the facilities with formal training programs provide technical training in contamination assessment, decontamination methodology and waste management. The importance of these skills to health physicists active in D&D cannot be overemphasized, and specialized training in these areas is desirable.

Table 1

**Health Physics Technicians and Assigned Operator Workforces  
By Facility Function, FY 1980-1983  
(As Percentages of Total Workforces)**

<u>Facility Functions</u>	<u>Health Physics Technicians</u>				<u>Assigned Operators</u>			
	<u>FY 80</u>	<u>FY 81</u>	<u>FY 82</u>	<u>FY 83</u>	<u>FY 80</u>	<u>FY 81</u>	<u>FY 82</u>	<u>FY 83</u>
Reactors	13.7	14.4	13.6	13.7	0	0	0	0
Fuel Fabrication	5.2	5.1	4.4	4.0	.1	.4	.5	.5
Fuel Reprocessing	14.7	14.2	15.7	13.3	2.5	4.4	11.5	13.4
Fuel Enrichment	2.9	3.7	3.5	2.8	28.1	28.3	24.6	21.7
Weapons Fabrication and Testing	21.5	21.1	22.9	24.2	16.7	17.8	16.4	16.0
Waste Processing/ Management	11.0	11.3	11.7	14.2	38.7	35.4	33.7	37.8
Accelerators	5.4	5.5	5.2	4.7	9.0	9.2	8.3	7.4
General Research and Development	19.5	19.1	17.6	18.2	1.5	1.6	1.4	1.3
Other	6.1	5.7	5.3	5.0	3.4	3.1	2.4	2.0

---

Source: J. Trice, 1984

Table 2

Technical Elements Included in Formal Training Programs  
for Health Physics Technicians and Assigned Operator Workforces

Technical Elements	Number of Facilities	
	Health Physics Technician N=23*	Assigned Operator N=8*
Basic Math	18	2
Basic Nuclear Physics	22	6
Radiation Protection Standards, Guides, and Limits	20	6
National/International Organizations (ICRP, NCRP)	8	0
Biological Effects of Radiation	22	6
Basic Units and Terminology	23	7
Fundamentals of Bioassay	16	2
Fundamentals of Detection	22	6
Respirator Use, Test, and Maintenance	23	5
Protective Clothing	22	6
Personnel Contamination Assessment	20	4
Air Sampling Technology	20	4
Surface Contamination Assessment	21	5
Dose/Stay-time Calculation	16	4
Radioactive Source Control	20	5
Shielding	18	3
Decontamination Methodology	21	4
Personnel Dosimetry	19	4
Alpha Monitoring	21	7
Beta, Gamma Monitoring	22	6
Neutron Monitoring	17	2
Instrumentation (e.g., Testing, Maintenance, and Calibration of Portable Survey Equipment)	17	2
Standardization and Application of Lab Counting Equipment	15	2
Plant Radiation Safety Policies and Procedures	23	8
On-Site Emergency Preparedness	20	6
Off-Site Emergency Preparedness	10	1
Criticality Safety	17	3
ALARA	19	4
Recordkeeping	20	3
Waste Management	14	2
Posting and Labeling	21	5

---

\*N = number of facilities responding

Source: J. Trice, 1984

## TAILORING D&D TRAINING NEEDS

Decontamination and decommissioning programs require both basic and specialized health physics activities to be performed (1) to evaluate the radiation environment of the facility under consideration, (2) to establish the standards to which cleanup activities must be pursued, and (3) to adequately protect personnel involved in the cleanup. Performance-based training, based on job task analysis, is an appropriate way to define the different types of health physics expertise required for D&D programs.

Without systematic definition of required job functions, responsibilities, and performance standards, training programs have generic deficiencies: they do not provide complete training of required skills and knowledge (2). The implementation of efficient performance-based training derived from and referenced to job performance criteria has become a major objective for nuclear trainers in the 1980s.

A methodology developed for setting decommissioning priorities in ORNL's Surplus Facilities Management Program (3) is useful in identifying criteria and factors pertinent to job tasks. These criteria include: potential for environmental impact, compliance with standards, potential for on-site and off-site health impacts, safety, control, and public reaction. Factors used to evaluate these criteria are: environmental contact, activity level, type of activity, structure, location, mixed hazardous materials, regulatory factors, and others such as maintenance.

Using these criteria, we find that the elements of a health physics training program for D&D workers involve the usual radiation protection fundamentals but with increased emphasis on certain subject areas that are particularly germane in D&D: environmental monitoring, control of exposures in high radiation areas, waste management, and sample/data management. Increased training in these areas can be based on the detailed manuals that the Department of Energy and others have developed in recent years. For example, the Decommissioning Handbook (4) describes all aspects of the decommissioning process, particularly for mothballing, in-place entombment, dismantling, or converting nuclear sites to new uses. The estimation of occupational exposures, environmental assessment, and disposition of wastes are also covered. On the other hand, there are different procedure handbooks to cover the survey and cleanup of contaminated lands as well as facilities. The Remedial Action Survey and Certification Activities Program Manual (5) exemplifies this type of source document. It covers field measurements, sample collection, laboratory analysis, equipment inventory, control and maintenance, monitoring and decontamination, measurement and test equipment calibration, and computer data management.

### Environmental Monitoring

Environmental monitoring should receive increased attention in a D&D training curriculum because of the need to restore contaminated areas to a condition acceptable for further use. Field measurements of alpha, beta and gamma radiation and of radon and radon daughters are often poorly characterized (5). Proficiency in sampling and evaluation techniques and use of a wide range of equipment is necessary (6).

A strong training program that recognizes the difference in strategy, monitoring methodology, and, quite likely, instrumentation for ongoing operational site surveillance and remedial action to clean up sites no longer active and to certify such sites as free of contamination would be extremely useful. The health physicist active in environmental monitoring as a part of a D&D program is likely to have need of additional skills in field-testing and calibration of equipment, packaging and shipping environmental samples, and even field photography (5).

### Exposure Assessment

Control of exposure to workers in areas where high levels of radiation exist is of particular importance in D&D activities. Often the most dangerous exposure situations in ongoing operations center on maintenance activities, when normal control systems may be interrupted to correct system failures or upgrade operations. Therefore, it is feasible to expect that persons engaged in the decontamination and dismantling of facilities dealing with radioactive materials are subject to high exposure risks. A very important task in this regard is the evaluation of site historical records and data to identify the liquid and gaseous control systems, the radioactive storage and waste disposal areas onsite and the residual radioactivity left in the systems being decontaminated and decommissioned. Special attention on how to evaluate such historical data is essential in D&D work (5). In addition, the health physicist may have a broader range of special work environments to survey and evaluate during D&D; protective limits, equipment, and ALARA strategies need to be very different in dismantling a reactor, replacing a forty-year old waste treatment transport system, dealing with a uranium mill tailings pile, or decontaminating plutonium production facilities. While general training may not deal exhaustively with all the different work environments, it can address the criteria to be used to evaluate the workplace (4, 7-12).

### Waste Management

Management of waste generated during D&D operations is significant because many of the sites to be cleaned up are old waste management systems and/or areas and because the D&D process itself may result in significant amounts of radioactive waste materials to be entombed or stored. Cleanup of old waste management sites/systems may often involve waste relocation when other decommissioning procedures are not adequate (13). As noted earlier, public health risks are much less than the risks of higher-level exposures to workers during waste cleanup/relocation, but again the need to contain such wastes and halt any migration into public air, water, or land requires additional training in decontamination, environmental transport, and waste stabilization and engineering techniques. There are several sources of pertinent information on these issues, particularly in a bibliography on cleanup and treatment of radioactively contaminated land including areas near nuclear facilities (14).

Knowledge of waste control engineering and procedures that prevent the loss or migration of radioactive materials is necessary to properly dispose of radioactive components of any site being decontaminated and decommissioned. Dismantling procedures and disposition of waste are detailed in the Decommissioning Handbook (4).

#### Sample and Data Management

Finally, sample and data management is vital to the success of D&D projects in that sites cannot be certified free of contamination and released for other uses without a competent system of sample and data management to validate such judgments. Additional training in sample collection and laboratory analysis and in data management (utilizing computer techniques) can prepare health physicists to carry out these validation and recording functions in a systematic and retrievable fashion. Proper procedures in sampling and recordkeeping may be particularly important if court challenges of site restoration are raised.

#### SUMMARY

Because the areas of expertise needed to provide adequate radiation protection in D&D are many and vary from one work environment to another, it is difficult to define in depth the scope of a training program. Table 3 summarizes the major health physics task areas that should receive increasing emphasis in any training agendas for health physicists engaged in D&D activities. The references noted are appropriate for use in developing a training agenda based on the additional skills needed for decontamination and decommissioning.

The critical need for radiation protection staff with necessary skills and knowledge in these areas of D&D make it mandatory to continue to strive toward more structured and better defined training programs.

#### ACKNOWLEDGEMENT

This research is sponsored by Office of Energy Research, U.S. Department of Energy, under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

Table 3  
Increased Emphases for Major Health Physics Tasks  
in Decontamination and Decommissioning

Tasks	Increased Emphases	References
Environmental Monitoring	Better field measurements	5
	Field-testing and calibration of wide range of equipment	5
	Packaging and shipment of environmental samples	5
	Proficiency in sampling and evaluation techniques	6
	Field Photography	5
Exposure Assessment	Evaluation of site historical records and data	5
	Evaluation of broad range of work environments	7, 8, 9 10, 11, 12, 14
	High-level exposure potential	3, 4, 5, 14
Waste Management	Waste relocation methods and occupational exposures	4, 13
	Decontamination methods	4, 13, 14
	Environmental transport	4, 14
	Waste stabilization	14
Sample and Data Management	Proper sample collection and laboratory analysis	4, 5
	Computer data management and recordkeeping	5

## REFERENCES

1. J. Trice, Manpower Trends and Training Requirements for Radiation Protection Personnel in the DOE Contractor System, ORAU-225, Manpower Education, Research, and Training Division, Oak Ridge Associated Universities, 1984.
2. L. T. Davis and R. W. Spinney, "Performance-Based Training: From Job and Task Analysis to Training Materials," in The Fifth Symposium on Training of Nuclear Facility Personnel, Gatlinburg, Tennessee, April 25-28, 1983.
3. H. M. Braunstein, W. F. Ohnesorge, K. L. Daniels, R. K. Owenby, M. A. Montford, S. F. Huang, and T. W. Oakes, Preliminary Assessment for Decommissioning Surplus Facilities at ORNL: Environmental and Health Risk, ORNL/CF-84/310, Oak Ridge National Laboratory, 1984.
4. W. J. Manion and T. S. LaGuardia, Decommissioning Handbook, DOE/EV/10128-1, 423p., Nuclear Energy Services, Inc., Danbury, Connecticut, 1980.
5. W. A. Goldsmith, T. E. Myrick, B. A. Berven, W. D. Cottrell, and F. F. Haywood, Procedures Manual for the ORNL Remedial Action Survey and Certification Activities (RASCA) Program, ORNL-6000, Oak Ridge National Laboratory, 1982.
6. Institute of Nuclear Power Operations, Guidelines for Radiological Protection Technician Qualification, INPO 82-006, 1982.
7. E. F. Conti, Residual Radioactivity Limits for Decommissioning, NUREG-0613, Office of Standards Development, U. S. Nuclear Regulatory Commission, Washington, D. C., 1979.
8. E. F. Conti and P. B. Erickson, "Radiological Criteria for Release of Land and for Decommissioning Facilities," Trans. Am. Nucl. Soc. **26**, pp. 120, 1977.
9. H. W. Dickson, Standards and Guidelines Pertinent to the Development of Decommissioning Criteria for Sites Contaminated with Radioactive Material, ORNL/OEPA-4, Oak Ridge National Laboratory, Oak Ridge, Tennessee, August 1978.
10. W. A. Goldsmith, F. F. Haywood and D. G. Jacobs, "Guidelines for Cleanup of Uranium Tailings from Inactive Mills," in Operational Health Physics, Proceedings of the 9th Topical Symposium, Health Physics Society, Denver, Colorado, Central Rocky Mountain Chapter, Health Physics Society, Boulder, Colorado, pp. 735-741, 1976.
11. M. M. Osterhout (ed.), DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR FACILITIES, PROCEEDINGS OF CONFERENCE, SUN VALLEY, IDAHO, September 16-20, 1979, 810 p., Plenum Press, New York, 1979.
12. R. K. Paschall, Decontamination and Decommissioning Criteria for Use in Design of New Plutonium Facilities, ERDA-13156, 123 p., Rockwell International, Atomics International Division, Canoga Park, California, 1975.

13. E. S. Murphy and G. M. Holter, Technology, Safety and Costs of Decommissioning a Reference Low-Level Waste Burial Ground, NUREG/CR-0570, Volume 1, Pacific Northwest Laboratory, Richland, Washington, 1980.
14. C. S. Fore, R. A. Faust, and R. H. Brewster, Cleanup and Treatment of Radioactively Contaminated Land Including Areas Near Nuclear Facilities. A Selected Bibliography, ORNL/EIS-199; NVO/AEIC-243, Oak Ridge National Laboratory, 1982.

SITE CLEAN-UP REQUIREMENTS:  
WHERE DOES ONE DISPOSE OF "DIRT"?

Joyce Feldman  
U.S. Environmental Protection Agency  
26 Federal Plaza  
New York, NY 10278

ABSTRACT

Under its Superfund program the U.S. Environmental Protection Agency (EPA) has committed itself to remediate certain residential and commercial properties impacted by an accumulation of indoor radon or thoron resulting from deposition of contaminated dirt. A case in point: The State of New Jersey now has a number of sites contaminated with the residues of radium or thorium extraction operations. The residues ("dirt") were removed from the original operations areas and used as landfill, often in locations which are now around houses or other buildings. EPA's soil contamination limits for radium or thorium dictate that remediation efforts consist of removal of that fill. This solution presents the problem of disposing of large volumes of marginally radioactive materials. Options which recognize the present lack of low-level radioactive waste disposal facilities are examined here.

INTRODUCTION

Even prior to the passage of the Comprehensive Environmental Response, Compensation and Liability Act of 1980 (CERCLA, or "Superfund"), federal and state authorities had been working to decontaminate properties which had at some earlier time residues of ore processing operations, including those which involved extraction of thorium, radium or uranium. The processing of uranium was covered by the Department of Energy (DOE) under its Formerly Utilized Sites Remedial Action Program (FUSRAP). This project was designated to deal with properties which became contaminated during the Manhattan Project or as a result of other government contractor activities, mostly for the Department of Defense. The processing of thorium was, to a large extent, performed for the purpose of manufacturing the thorium gas mantles still used in propane lanterns. About three years ago, two of the facilities which had handled this material under government contract were targeted, through Congressional action, for remediation under the FUSRAP program. Other sites, where radium tailings were found, have been designated for Superfund action by inclusion on the National Priorities List (NPL). These sites are shown on the map (Fig. 1).

One common feature of all these sites is that none has been completely decontaminated to date, despite the infusion of federal and state funds. In each case, no final disposition of the materials has been possible, due to the lack of an appropriate disposal facility. In several of the DOE's sites, the contaminated soils have been temporarily stored on-site or consolidated at one location, generally within the boundaries of the locality where the problem was initially identified. Each disposition by temporary disposal was permitted by a special Memorandum of Understanding (MOU) with the affected locality. In some cases, there have been problems later, due to the fact that the host location was assumed by the local officials to be housing only soils found within the

boundaries of the single town. In nearly all cases, political jurisdiction was not the limiting factor in utilizing the contaminated soils. Apparently, human nature has changed little over the years: People enjoy getting "something for nothing," even if that includes free dirt. Whatever the mechanism, the contaminated soils were transported from the original site to a number of locations where landfill was needed, often not in the same town (at least not in the same town now). Tracing these landfill locations is not simple. Relocating them is proving to be very nearly impossible.

## THE PROBLEM

Within the State of New Jersey the EPA has a number of sites contaminated with radioactive residues from the processing of radium. These are included on the NPL. The initiation of a Remedial Investigation for the US Radium Site in Orange, NJ marks the start of the remedial process for the last (hopefully) of the sites requiring remediation for this type of problem. Due to the fact that radium-contaminated soils generate radon and its progeny, often in concentrations above those permitted in the workplace, some of the sites have received high priority for remediation. Assessment by the Centers for Disease Control (CDC) dictated that EPA utilize a corrective action promptly. Initially, a number of houses in Montclair, Glen Ridge and West Orange, NJ received close attention, aimed at reducing the levels of radon gas by ventilation. Those houses, and a large number of others, were evaluated and found to have contaminated soils nearby. To effect a permanent solution, the soils must be removed. Any other action, especially if it requires maintenance or periodic attention, cannot be considered permanent remediation. Whatever sealants can be applied will be effective only so long as no cracks appear. Ventilation systems must have periodic maintenance to assure effectiveness. Even placing vents or systems of French drains beneath the structures cannot assure that the diversion of the radon will be continued beyond some finite period.

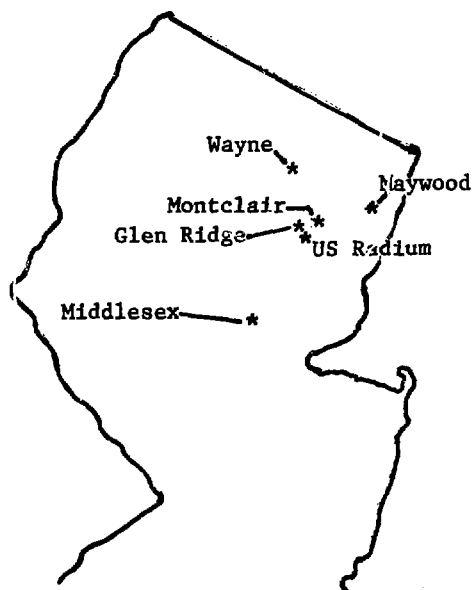


Fig. 1. New Jersey Sites Contaminated with Radioactive Processing Residues.

With this situation in mind, the EPA initiated a Feasibility Study to determine which of a number of possible alternatives would be most appropriate for the remediation of these mostly residential properties. Figure 2 lists the options considered. Many of these options are, in fact, proven technologies; however, the residential nature of the area, as well as the relatively small size of the individual parcels of land became major factors. Hence, the use of in-situ extraction technologies does not appear to be a highly promising option, for example.

The range of options for these sites runs the gamut from source control by capping in place or placement of subsurface barriers to prevent migration of both the contaminated soil and the radon gas through total removal of the source material. In this situation where there is an easily identified contaminant, removal is relatively simple: There is no need to determine

whether the concentration limits are met; since the soils are landfill--and visibly different from the native material--removal can be readily accomplished and certification can be assured by sampling from the remaining native material.

Fig. 2: Remedial Technology Options

On-Site Control and Containment:	Capping / Subsurface Barriers Shielding / Sealants / Passive Collection System / Active Collection System / Ventilation and Air Cleaning Systems Relocation In-Situ Solution Mining / Vitrification
Removal and Off-Site Treatment/ Disposal Technologies:	<u>Excavation:</u> Conventional / Hydraulic Mining <u>Volume Reduction:</u> Chemical Recovery of Radionuclides <u>Physical Separation:</u> By Size, Density / Ion Exchange / Bulk Separation / Bulk Mixing / Dilution
Immobilization of Radionuclides:	<u>Vitrification:</u> Electric Furnace Fusion/ Rotary Kiln <u>Matrix Isolation:</u> Bitumen or Asphalt / Cement / Resins

---

As noted above, the in-situ treatments such as solution mining and vitrification were eliminated from consideration due to the proximity of the houses. The basements would probably be adversely affected by the solutions. Further, there was some doubt that the relatively high removal efficiencies attributed to this methodology (approximately 90%) could be achieved with the relatively low concentration of radium present in the soils here. Adverse environmental impact was also of concern here.

Vitrification--the conversion of the waste into a glass-like material--was similarly rejected. While radon reductions up to 1,400 times that for untreated sands have been established, the soils found here are up to 40% fines, which can be expected to yield lower reduction rates. In addition, there are a number of major drawbacks to utilization of this technology:

- Off-gases produced must be treated to remove both radioactive and non-radioactive pollutants.
- There are large electric power requirements, especially given the high soil moisture in this area. Contractor estimates for treatment of the removal volume 122,000 cubic yards were 54,000 mega watt hours.
- Effects of vitrification have not been established for mixtures which include the ashes and organic materials found here.
- The residential location offered a number of disruptive conduits, such as underground utilities, fuel oil storage tanks, septic tanks and large tree root systems, which would discourage the electric path necessary to complete the process.
- Verification of completion of the process would be very difficult.

The extraction of radium and uranium from the soils by chemical methods, such as carbonate, sulfuric acid or phosphate leaching, would require removal of the soils from their present location to an impermeable pad or some other type of bulk container for actual treatment. The method was considered technically unfeasible for this case since the proven aspects of the technology are based on large volumes of materials containing more homogeneous materials with higher concentrations of the radium and uranium than are found here. Further, the presence of non-radioactive contaminants unique to these sites would necessitate a pilot investigation to establish the efficiency of the process. In addition, the presence of a large volume of acidified residue would require an additional treatment prior to final disposal of residues.

Ion exchange, like the liquid-base extraction technology, is not appropriate for this residential location.

The various forms of matrix isolation have both advantages and disadvantages. The goal of immobilization of the radionuclides, and especially the radon, is ultimately achievable. Leachability of the radioactive components is reduced. The solid product is more easily handled with relatively less chance of dispersal. As noted above, the high power requirements are adverse factors. In the case of rotary kiln sintering, large amounts of coal would be required, with resultant generation of fly ash and scrubber sludges. This would only serve to increase the volume of waste to be disposed.

Asphalt, which could provide good isolation for the soils, has disadvantages as far as process requirements and safety: Heated, it releases fumes which pose potentially toxic inhalation and dermal hazards. It is also a fire hazard. The wastewater from the process produces organic contaminants which require additional processing. Use of cement instead has the advantage of diluting the source. There is some question on the quality of cement resulting from incorporation of the particular soils being handled here, however. It is not likely to reduce significantly the radon emanations or potential leaching of the radionuclides. Polymerization of the materials into a resin is a final possibility. In this case, the organic resin and catalyst are mixed with the materials, incorporating them into the voids formed by the polymerized long chain molecules. While the process is established, well-defined parameters, such as use of a well-dried soil and specific pH requirements, must be met. The technique has the advantage of being amenable to any type of container, so it allows mixing in individual 55-gallon drums, if necessary.

#### THE BIG PROBLEM

Regardless of the chosen methodology, the big problem still ahead is one of location. Whether soils must be removed for treatment, for storage or for final disposal, a site is required to accommodate the activity and to store and/or dispose of the final product. To a large degree, this becomes a societal issue, rather than a technical one. While there are no explicit regulations for disposal of radium from a federal perspective, at this time, the State can make a determination on requirements. However, the soils also contain quantities of other radionuclides, which may or may not meet regulatory minimum levels for radioactive waste disposal. In all likelihood, there may be EPA standards for radium disposal before this project is complete.

The DOE has addressed a number of similar problem sites, including three in New Jersey. However, in each case, these have received only temporary remediation, on-site storage pending designation by the State of a final disposal site. The DOE has, in fact, completed remediation at a similarly troubled site

in Canonsburg, PA, where the entire volume of contaminated soil was permanently encapsulated in an on-site disposal facility. Based on this experience, the capability has been established. However, the question of using an area of ten or more acres within an active community for permanent disposal raises serious ethical questions, especially when the same size property may be available elsewhere in the general area. As part of the Feasibility Study, options were identified which included on-site (either one location or one in each town) storage of a temporary or permanent nature. These options have been met with a high degree of resistance from within the communities. One of the biggest problems faced is that a number of the other options require use of some significant piece of property on an interim basis, either for storage prior to transport or for treatment of the soils prior to disposal. Even this approach has been greeted with hostility.

Suggestions of using the materials--of marginal concentrations, radiologically--in slurry walls or roadbed construction have been precluded to some extent due to the fact that materials needed would require instant availability, which demands storage at some convenient site. In addition, there is relatively little roadbed development in the state now, with the few new roads resulting in a net surplus of materials. Mine burial has received similar rejection, due to the fact that mines available within the state have water problems which would likely increase leachability of the materials above the level that exists at their present locations.

Ocean disposal, an apparently attractive and technically feasible option, now appears doomed to total rejection, due to recent developments on the international level.

#### THE SOLUTION?

While the apparently best solution is disposal in a permitted low-level radioactive waste facility, there have been legal stumbling blocks imposed, even when a permit was issued for a site (Beatty, Nevada). The only two existing facilities besides that in Nevada do not, apparently, have capacity or local desire to accept the large volume of relatively innocuous material which must be disposed. In addition to consideration of the permitted facilities, DOD and DOE facilities have been considered as well. Both policy and regulatory requirements appear to preclude the use of these sites for disposal. The State of New Jersey, which had originally suggested a local piece of State-owned property in West Orange, has not identified other potential candidate locations.

The Low-Level Radioactive Waste Disposal Act of 1980 requires that each state be responsible for radioactive low-level waste generated within its borders. The deadline, January 1986, may soon be extended. However, this is not necessarily a saving effect. With such a large volume of material, the EPA must include fund balancing in its decision-making process. Transporting these large volumes of minimally radioactive materials across country, even if Nevada can be induced to accept the material, is not cost-effective, when compared with the alternative of in-state disposal, or even nearby disposal. New Jersey has joined a compact with Maryland, Delaware and Connecticut. This group of states has not yet designated a site. Following designation, development will likely require a minimum of two years. This means that, to accomplish its goal of remediating the residential problems within a shorter period than that, the EPA must identify a temporary storage location. This process will, of course, require active participation by both State and local officials. Without approval of any selected alternative by these groups, EPA cannot proceed.

## SUMMARY & CONCLUSIONS

Summarizing this problem is relatively simple: There is no room anywhere for this volume of "dirt" which is acceptable to all participants in the process of decisionmaking. From a regulatory standpoint, much, if not most, of the residues are hardly radioactive. The Department of Transportation regulations indicate that a minimum activity of 2 nanocuries/gram defines their requirement for placarding. With an average of 100 pCi/g of radium-226, the total activity of the residues is not likely to meet the DOT level. However, that concentration of Ra-226 is sufficient--demonstrably so--to yield levels of radon and its progeny which exceed 0.02WL within buildings built on or near the materials.

The second part of the problem is that any interim or final solution must meet the acceptance--however grudging--of all parties affected. This includes federal, state and local governmental bodies as well as residents or occupants of the affected properties and neighbors of the disposal or storage location. With so large a group of participants who are also reading of THE RADON PROBLEM in the State's Reading Prong, is it any wonder that there is a sense of near-hysteria regarding choice of a location for even temporary storage?

The Conclusion: Until a repository for all low-level radioactive waste is sited within the compact area, there probably can be no peace when these problems are raised.

----- Although the information described in this presentation has been funded wholly or in part by the U.S. Environmental Protection Agency, it does not necessarily reflect the views of the Agency, and no official endorsement should be inferred.-----

## BIBLIOGRAPHY

- Camp Dresser & McKee, Inc., "Draft Final Feasibility Study for the Montclair/West Orange and Glen Ridge, New Jersey Radium Sites, Volume I," September 13, 1985, developed under EPA Contract No. 68-01-6939.
- US Department of Energy, "A Background Report for the Formerly Utilized Manhattan Engineer District/Atomic Energy Commission Sites Program," September, 1980, DOE/EV-0097.
- US Department of Energy, "Radiological Guidelines for Application to DOE's Formerly Utilized Sites Remedial Action Program," March, 1983, ORO-831 (Rev.).
- US Environmental Protection Agency, "Low-Level Radioactive Waste Management: Proceedings of the Health Physics Society Twelfth Midyear Topical Symposium, February 11-15, 1979, Williamsburg, Virginia," May, 1979, EPA 520/3-79-002.

DECONTAMINATION/DECOMMISSIONING  
OF MIXED HAZARDOUS AND RADIOACTIVE WASTE DISPOSAL SITES

R. R. Ferguson  
D. G. Jacobs  
H&R Technical Associates, Inc.  
575 Oak Ridge Turnpike  
Oak Ridge, TN 37831

C. W. Kimbrough  
Martin Marietta Energy Systems, Inc.  
P.O. Box Y  
Oak Ridge, TN 37830

ABSTRACT

The production and research facilities operated for the U.S. Department of Energy have generated a variety of solid and liquid wastes that contain both chemically hazardous and radioactive materials. In the past, regulations and waste management practices for radioactive wastes were developed independently from those for chemically hazardous wastes. Before implementation of the Resource Conservation and Recovery Act, the emphasis placed on management of hazardous wastes at facilities that handled both types of wastes was significantly less than the emphasis placed on management of radioactive wastes. Thus, inactive disposal sites that may be releasing inconsequential amounts of radioactivity to the environment may be releasing potentially significant amounts of chemically toxic heavy metals and organic compounds. In many instances, past disposal practices may dictate that the wastes associated with these sites be removed from current locations, treated to reduce toxicity, and disposed of in a more stable environmental setting. This paper outlines and describes the regulatory framework for site remediation and emphasizes how the concerns for health physics and occupational safety and health influence implementation.

INTRODUCTION

Based on an April 13, 1984 decision by the United States District Court for the Eastern District of Tennessee (LEAF v. Hodel, 586 F. Supp. 1163), the U.S. Department of Energy (DOE) must evaluate its waste management practices and obtain the necessary permits for continued operation or closure and decommissioning of its hazardous waste management facilities. DOE has developed Orders 5480.2, 5820.2, and Draft Order 5480.14 to provide direction to its facilities for complying with the Resource Conservation and Recovery Act (RCRA), the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), and other legislation applicable to the management of radioactive and hazardous wastes. Although DOE will request permits for continued operation of many of its mixed hazardous and radioactive (co-contaminated) waste treatment, storage, and disposal facilities, many of them are likely to be closed and decommissioned. Several of the facilities to be decommissioned have received wastes of varying composition for over forty years.

Several alternatives are available for closing a disposal site. These alternatives range from securing and stabilizing the wastes in place to complete removal of the wastes. "Clean" closure is generally referred to in this paper as the removal of liquid and solid wastes and contaminated soils such that pre-determined decommissioning standards for residual levels of specific hazardous or radioactive constituents are met. The excavated wastes and soils would then be disposed of in an environmentally secure setting which meets the current regulatory requirements. Clean closure may include treatment of the wastes and soils removed before placement in an ultimate disposal location. Obviously, workers involved in the closure operation will have to be protected from exposure to the wastes during site investigations, waste removal, treatment, packaging, transportation, and ultimate disposal.

## DESCRIPTION OF CLOSURE ACTIVITIES

In order to evaluate the necessity for and magnitude of the occupational safety and health programs necessary during closure activities, a significant amount of information regarding the types and characteristics of deposited wastes must be developed and the activities associated with physical closure of the facility must be well defined. The following subsections provide a general description of the types of hazards that might be encountered during such closure activities and the general progression of decommissioning activities required by environmental legislation and DOE Orders.

### Waste Types and Characteristics

Land treatment, storage, and disposal operations at DOE facilities have generally consisted of storage (and treatment) in surface impoundments and disposal in below-grade trenches or landfills. These operations were typically initiated during a period in which native clay liners were considered suitable for and capable of impeding the migration of contaminants into surrounding soils and ground water. Thus, at a typical DOE installation, ground water may have been contaminated with a wide range of materials, including oils and coolants, uranium, solvents, heavy metals, and nitrates. In many instances, these wastes were disposed of indiscriminately; thus the potential for mixture of incompatible wastes is high. The hazards associated with the mixing of these incompatible wastes include reactions which can: 1) produce heat, fire, or explosions; 2) form toxic fumes, flammable gases, or reaction products of greater toxicity than the reactants; and 3) cause dispersal of toxic dusts, mists, and particles. (1)

### Closure Strategy

DOE issued Order 5480.14 to provide instructions to its contractors for implementing a decommissioning program at its inactive hazardous, radioactive, or co-contaminated waste disposal sites. Figure 1 illustrates a strategy for effecting clean closure at a facility in accordance with Order 5480.14. This closure option was chosen for discussion in this paper because it encompasses a broad range of health and safety concerns. The paper focuses on the implementation phase of the remedial action program (Phase IV), even though protection of health and safety considerations will

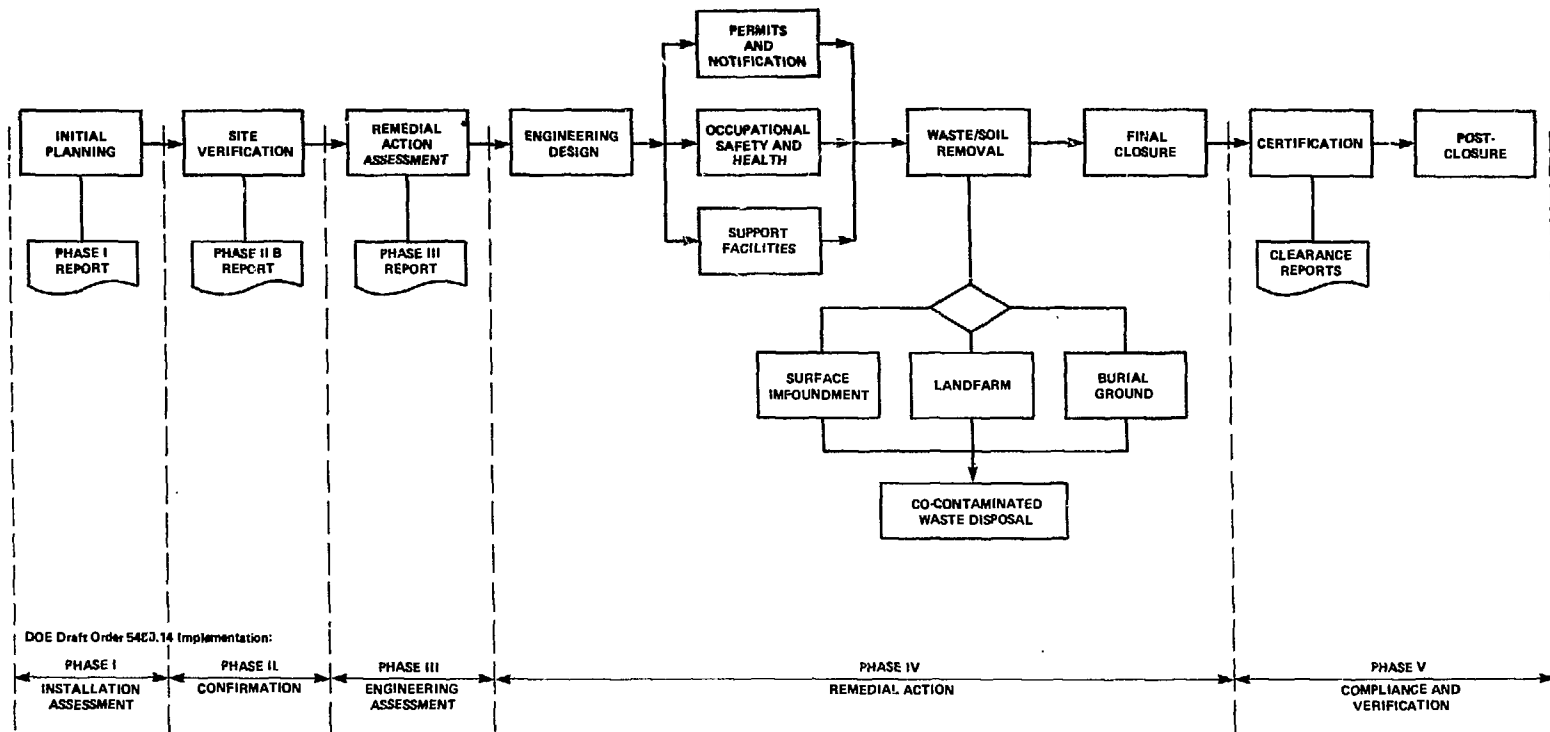


Figure 1 - Clean Closure Strategy

Momentum dynamics affect the boiling water reactor (BWR) neutronic stability by coupling steam void perturbations and core-inlet coolant flow. Computer simulations<sup>1,2</sup> have shown that proper modeling of the recirculation loop, which shares the upper and lower plena pressures with the reactor core, is essential for accurate stability calculations. The purpose of the present work is to show experimental evidence, obtained from a recent series of stability tests performed at the Browns Ferry-1 BWR,<sup>3,4</sup> demonstrating the important role of momentum dynamics in BWR neutronic stability.

The results of the Browns Ferry stability tests<sup>3,4</sup> confirmed the stability of this reactor and showed that the sensitivity of the decay ratio (DR) to variations in power and flow followed the same trends during two-loop and single-loop operation (SLO). SLO measurements, however, exhibited a significant noise increase (300%) in most process signals. The source of this higher noise level was determined to be related to increased turbulence in the downcomer due to crossflow between active and inactive pump loops. This determination was made by comparing the reactor transfer functions (obtained from noise measurements without external perturbations) to the results of computer simulations. Figure 1 presents the transfer function (TF) between active-loop flow and average reactor-power for test BFTP3. The remarkable agreement observed between the calculated and measured TFs implied that the noise source was external to the core and was included in the flow signal.<sup>3,4</sup>

Given the relative simplicity and low cost with which noise measurements can be performed, this well-known technique might be used

and closure scenario. Workers will have to be protected from the hazards presented by the various wastes and potentially incompatible waste mixtures that are likely to be encountered during characterization and waste excavation, treatment, packaging, shipment and disposal. The hazards associated with exposure to radioactive materials arise primarily from direct exposure to ionizing radiation, inhalation of radioactive particulates, and direct skin contact with radioactive materials. Ingestion of radioactive materials is not likely to be of major concern. The exposure modes for chemically toxic wastes are very similar to those for radioactive materials.

Exposure to ionizing radiation or to reactive, ignitable, or flammable chemicals can be controlled with proper shielding or remote handling. Shielding may range from limited physical barriers to heavily shielded or remotely controlled equipment. Respirators can be used to prevent inhalation of radioactive or hazardous particulates or vapors. Similarly, protective clothing can be used to prevent direct skin contact with either type of waste. As an additional precaution, shower facilities should be provided to reduce inadvertent ingestion or spread of contaminants.

Based on a thorough records evaluation of the types, amounts, and locations of buried materials, and in association with sufficient site characterizations, a detailed occupational health and safety program plan can be developed to be instituted during the closure operation. The plan should include provisions for personnel training, emergency preparedness, and personnel protection during all phases of decommissioning. All operational personnel must be sufficiently trained in the proper handling of radioactive and chemically hazardous materials, the use of protective clothing and devices, the use of personnel and work area monitoring equipment, and general occupational safety.

#### Personnel Protection

Personnel working in areas with elevated levels of radioactivity or hazardous chemicals should be issued appropriate individual dosimeters or exposure meters. Suitable equipment should be provided for work area monitoring of toxic and combustible gases. Protective clothing can be used to maintain exposure levels within acceptable limits in highly contaminated work areas. When workers leave areas with elevated levels of contamination, they should remove potentially contaminated protective clothing and devices and then shower, shampoo, and exit through a controlled access point. Workers should change out and wash before eating to reduce the probability of inadvertent ingestion of contaminants. Access to contaminated areas should be limited to individuals who have received proper training and have been issued appropriate monitoring equipment. Work assignments should be restricted when the individual has reached or exceeded applicable dose/exposure limits.

#### Support Facilities and Services

Detailed records should be maintained to document personnel training, personnel access to contaminated areas, personnel doses and exposures, and work area monitoring. Analytical services will be required to evaluate personnel dosimeters and exposure meters, work area monitoring samples, respiration filters, and field instrument calibration. Laundry facilities

may be required for cleaning protective clothing used in contaminated areas.

#### REGULATORY JURISDICTION

Virtually all aspects of the decommissioning process are regulated by state and Federal law, as referenced through the appropriate DOE Orders. The regulations and orders of primary concern are listed in Table 1. Since this paper only references a generic closure scenario, state regulations are not cited. Any contractor closing a disposal site will have to apply for the necessary permits required by the various statutes.

Table 1 - Primary Regulatory Authorities

<u>Closure Activity</u>	<u>Federal Authority</u>	<u>DOE Orders</u>
General Environmental	40 CFR 1500-1508 (NEPA) 40 CFR 702-792 (TSCA)	5440.1C 5480.1A  5480.1A, 5480.4 5484.1
Air Releases	40 CFR 50 (Ambient Air Quality Standards)  40 CFR 61 (National Emission Standards for Hazardous Air Pollutants)	
Water Releases	40 CFR 122 (NPDES)	
Hazardous Wastes	40 CFR 260-265 (RCRA) 40 CFR 300 (CERCLA)	5480.2 5480.14
Radioactive Wastes		5820.2, 5480.2
Packaging and Transportation	10 CFR 71 (Radioactive Materials) 40 CFR 263 (RCRA) 49 CFR 171-178 (Hazardous Materials)	1540.1, 5820.2, 5480.1, 5480.2, 5480.3
Occupational Safety and Health	29 CFR 1910 (OSHA)	5480.3, 5480.4, 5480.1A, 5480.10, 5481.1B, 5483.1A, 5500.2
Records Management	10 CFR 1008 (Privacy Act) 29 CFR 1904 (OSHA)	1324.2, 5482.1A

A key aspect of the clean closure scenario is that an ultimate disposal facility would have to be permitted to receive both hazardous and radioactive wastes. Currently, there are no government-owned sites that are permitted under the RCRA Part B program for disposal of mixed wastes.

The development and permitting of such a site is anticipated to be a costly and time-consuming process.

#### SUMMARY

This paper outlines the regulatory and operational safety and health constraints under which mixed hazardous and radioactive waste disposal sites will be decommissioned. "Clean" closure is emphasized because of the increased concern it places on workers associated with the conduct of the closure operation. All types of decommissioning and remedial action activities at mixed waste storage or disposal sites will require some degree of occupational safety control. However, this brief analysis implies that the potential hazards associated with "clean" closure or removal of these wastes will require careful consideration and will result in excessive health and safety implementation and monitoring costs. These costs, as well as those associated with regulatory compliance and development of a final disposal facility, ultimately reduce the feasibility of the "clean" closure scenario relative to other in-situ remediations.

#### ACKNOWLEDGEMENT

This paper was developed from work performed by H&R Technical Associates, Inc. under Subcontract No. 41B-38435C for Oak Ridge Y-12 Plant, operated by Martin Marietta Energy Systems, Inc., for the U.S. Department of Energy under Contract No. DE-AC05-84OR21400.

#### REFERENCES

1. H. K. Hatamaya, J. J. Chen, E.R. de Vera, R. D. Stephens, and D. L. Storm, "A Method for Determining the Compatibility of Hazardous Wastes", EPA-600/2-80-076, p. 1, April, 1980

## THE LABORATORY PROTOCOL FOR CO-CONTAMINATED WASTE

D. P. Kharkar, D. Rodgers, L. Penfold,  
G. Dunstan and L. Leventhal  
EAL Corporation  
2030 Wright Avenue  
Richmond, California 94804

### ABSTRACT

The decommissioning of the inactive disposal sites such as uranium mill tailings, fuel processing plants, waste storage areas and decontamination of structures, equipment, laboratories, etc., generate hazardous substances containing varying amounts of radioactivity. The enactment of environmental legislation and regulations requires that environmentally responsible actions be taken to identify and eliminate the potential hazards resulting from the waste materials.

A special handling procedure has been developed for screening, preparing, analyzing and disposing of radioactive samples undergoing hazardous waste characterization, i.e., co-contaminated samples. This paper addresses three principal concerns -- laboratory compliance with State and Federal laws affecting handling and release of radioactivity to uncontrolled areas, health and safety of laboratory workers, and handling practices that insure the laboratory's continued ability to perform environmental level radioactive analyses. The analytical procedures used for co-contaminated samples are also discussed.

### INTRODUCTION

The EPA, under The Resource Conservation and Recovery Act (RCRA), (Ref. 1), requires that waste materials that are radioactive be tested for RCRA characteristics prior to disposal. These tests require sophisticated analytical equipment to perform the organic and inorganic analysis. Due to the radioactive content of these samples, the laboratory performing these tests must be licensed by the NRC, or an agreement state, to receive and possess such samples. These samples require special handling procedures for receiving, screening, preparation, analysis and disposal.

This paper addresses laboratory compliance with state and federal laws affecting handling and release of radioactivity to uncontrolled areas, health and safety of laboratory workers and handling practices that insure the laboratory's continued ability to perform environmental level radioactive analyses. The applicable state regulations for California are contained in California Administrative Code, Title 17, (Ref. 2). The analytical procedures used for co-contaminated samples are also discussed.

### CRITERIA OF RADIOACTIVE LIMITS

In setting limits for processing the radioactive samples in controlled areas, the following three criteria were considered:

- A. External exposure to lab personnel in uncontrolled areas must meet the requirements of CAC, Title 17, Section 30286, <2 mR/hr, (Ref. 1).

This was not a limiting factor since satisfying criterion B and C limited activities to levels far below those allowed in Section 30286 of Title 17.

- B. Release of radioactivity to the air in uncontrolled areas must meet the requirements of CAC, Title 17, Section 30269. A very conservative interpretation of Title 17 was used as a guideline. Title 17 requires the concentration of released activity to the air in an uncontrolled area to be less than the values of Section 30355, Appendix A, Table II, Column I, when averaged over a one year period. As a very conservative approach, the limits calculated here were based on the concentrations in air not exceeding the Title 17 values at any time. For calculation purposes, it was assumed that all releases would be to a 12' x 12' x 9' room and dispersed instantaneously throughout the volume of the room.

The primary mechanism by which radioactivity from these samples could be released to the air is the destructive testing procedures which vaporize the samples for analysis such as gas chromatographs and spectrophotometers. In the atomic absorption spectrometer used for metals analysis, the sample is completely vaporized and vented into a small hood. In the gas chromatograph and other spectrometers, the sample is also vaporized, but most common activities would be retained at the front of the columns or gas lines. The atomic absorption (AA) is the worst case then and the limits were calculated assuming all samples would be processed through the AA and the exhaust hood would be 90% efficient venting the activity out of the room.

- C. Processing of radioactive samples must proceed in such a way as to pose no significant contamination threat that could compromise EAL's ability to perform low level radioactivity analyses.

As in criterion B, the mechanism with the most potential for contaminating the lab areas is the AA venting activity to the air. For contamination considerations, somewhat arbitrary limits as to how much activity can be vented to lab air were set at 1,000 dpm of beta/gamma, 100 dpm of alpha, or 10,000 dpm of tritium. It was assumed that AA would be vented into an absolutely filtered hood with a total collection efficiency of 90%. This, then, implies limiting total amounts to be processed through the AA of 10,000 dpm of beta/gamma, 1,000 dpm of alpha, or 100,000 dpm of tritium.

To calculate the values shown in Table 1, the following procedure was followed. The value for each nuclide given in Title 17, Section 30355, Appendix A, Table II, Column I, was converted from microcuries/mL to dpm/mL and multiplied by the volume of 12' x 12' x 9' room (37,000,000 mL) to yield the total allowable dpm. This value was compared with the appropriate contamination limit (100 dpm alpha, 1,000 dpm beta/gamma, 10,000 dpm tritium), and multiplied, whichever number is smaller by 10 to account for only 10%, actually vented into the room air. The resulting value was used for Table 1.

Table 1 Maximum Amounts of Activity to be Tested in an Uncontrolled Area

Nuclide	Radiation Type	Limiting Criterion (a)	Maximum Activity dpm/Aliquot
Natural Th (b)	Alpha, Beta, Gamma	B	500
Natural U (b)	Alpha, Beta, Gamma	B	600
<sup>226</sup> Ra (b)	Alpha, Gamma	C	1,000
<sup>137</sup> Cs	Beta, Gamma	C	10,000
<sup>60</sup> Co	Beta, Gamma	C	10,000
<sup>131</sup> I	Beta, Gamma	C	10,000
<sup>90</sup> Sr	Beta, Gamma	C	10,000
<sup>239</sup> Pu	Alpha	B	50
<sup>238</sup> Pu	Alpha	B	60
<sup>241</sup> Pu	Beta	B	2,000
<sup>241</sup> Am	Alpha	B	160
<sup>243</sup> Am	Alpha	B	160
<sup>210</sup> Po	Alpha, Beta	C	1,000
<sup>210</sup> Pb	Alpha, Beta	C	1,000
<sup>3</sup> H	Beta	C	100,000
<sup>14</sup> C	Beta	C	10,000
Other Beta, Gamma		C	10,000
Other Alpha		-	See Health Physics
Mixed Activities			Use Formula Below

(a) B = Limiting release to air to Title 17 values.

C = Controlling lab contamination.

(b) Value shown is for the parent only and assumed the presence of daughter activity.

#### Formula for Calculation of Limits for Mixed Activities

For each nuclide of a mixed activity sample, divide the activity by the limit in Table 1. If the sum of these ratios for all nuclides is less than 1.0, then the sample is considered to be under the limit.

## ANALYTICAL PROTOCOL

Since the co-contaminated samples require special handling, EAL Corporation has developed special handling procedures for receiving, screening, preparing, analyzing and disposing the samples. EAL divides the sample processing into three categories.

- A. Samples with no significant activity. These are samples which show a reading on a GM survey meter not greater than a background, or samples with less than 1% of the activity shown in Table 1.
- B. Samples with low activity. These samples which show GM survey meter readings above background, or are samples with activity levels 1 to 100% of the values given in Table 1.
- C. Samples with high activity. These are any samples with greater than low activity. EAL has a total limit of 400 microcuries per sample with a dose rate not to exceed 200 mR/hr. The alpha emitting transuranics should not be greater than 1 microcurie per sample.

The co-contaminated samples are delivered to the control area. The shipping documents are examined before unpacking the samples. All samples are placed in a water proof tray and monitored externally for activity with a GM survey meter. If there is no activity through the box, the sample bottles are checked for external activity and for leaks. If there is no indication of radioactivity as a result of examining the paper work, or as a result of monitoring, then the samples are defined as having no significant activity.

If the sample is radioactive as defined above, the EAL health physicist will determine if sufficient information is available to decide if the samples would be classified as low, or high level activity.

If the information on activity levels is not available, then the samples are presumed to exceed the limits given in Table 1 and are screened for gross alpha, gross beta and gamma activity. Figure 1 gives the sample screening procedure. A segregated laboratory area is used to process higher activity samples. The instruments and supplies, such as atomic absorption spectrometer, PH/selective ion meter, analytical balance, organic extraction manifolds, etc., are dedicated to the segregated lab use, so that constraint on sample size may not be necessary on account of health physics. In the course of time, the segregated lab will be supplemented with other instruments so that required detection limits for various analytical tests may be obtained for higher activity level samples.

The extracts prepared in the segregated laboratory from high activity samples are screened for gross alpha and gross beta activity before being removed from this area. Only quantities of extract that can be classified as low level are transferred out.

A wide range of inorganic and organic analysis are performed on the samples. The atomic absorption spectrophotometer and the inductivity coupled plasma (ICP) instruments are used for trace element analysis, while gas chromatograph and GC/MS are used for organics. The toxicity of the samples is determined by a standard EP toxicity leach test, (Ref. 3). The tests are performed on the extract to determine if the thresholds established for eight elements (As, Ba, Cd, Cr, Pb, Hg, Se and Ag), four pesticides and two herbicides are exceeded. If EP

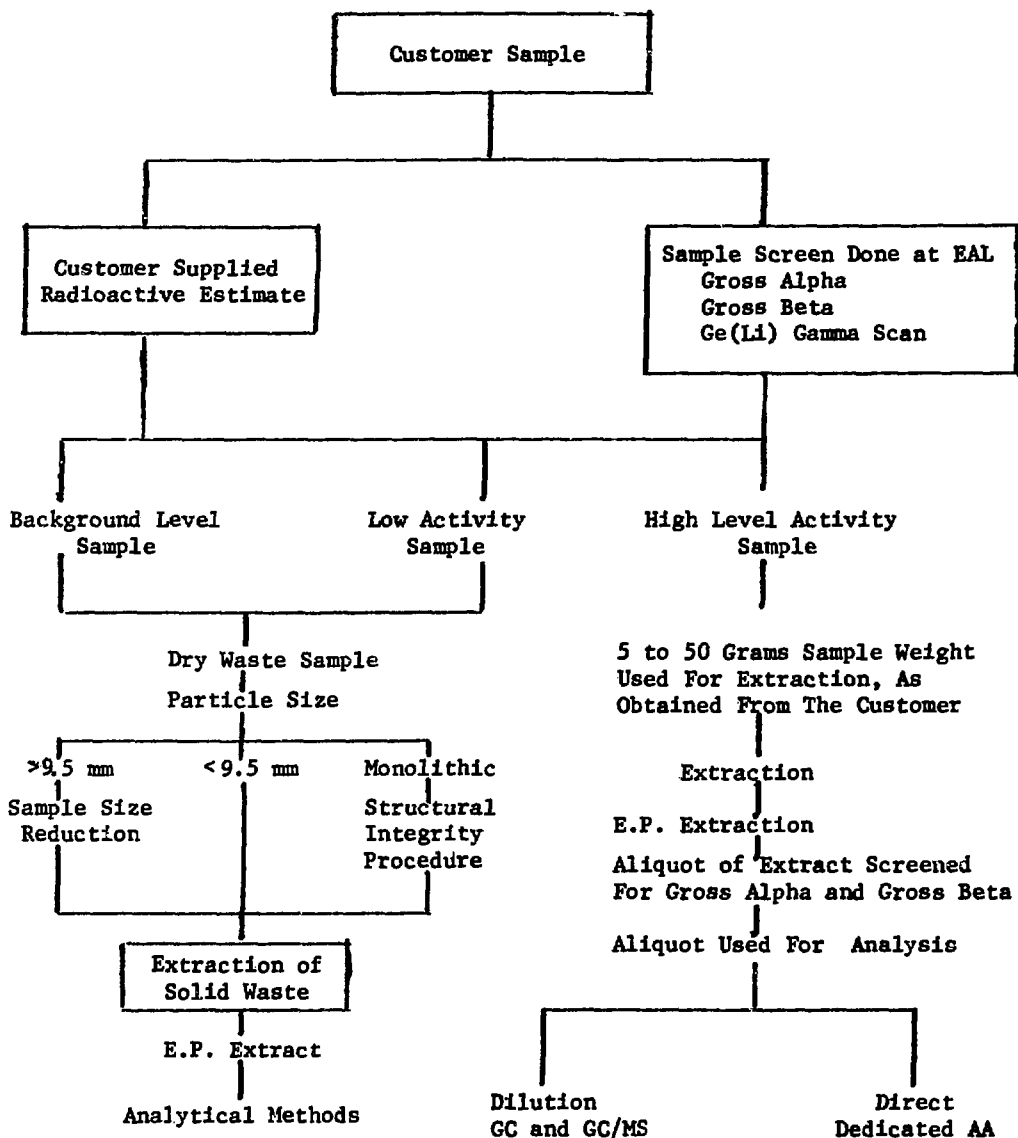


Figure 1 Sample Screening Procedure

extracts contain any one of the above constituents in an amount equal to, or exceeding the levels specified in 40CFR Part 261-24, (Ref. 3), the waste is defined as toxic and is a hazardous waste.

#### DISPOSAL

Disposal procedures are according to State of California Regulations with regard to hazardous and radioactive waste. Radioactive disposal is prescribed by our State of California Radioactive Material License. Samples classified as having "No Significant Activity," are only subject to routine chemical and physical disposal restraints. No special disposal practices are required. "Low Level Activity," liquid samples can be poured down the drain, if the activity is  $<1$  nCi/mL, a very conservative value based on Title 17 regulations. "Low Level" solid samples must be placed in plastic bags and can be disposed of as normal waste, unless chemical toxicities require hazardous waste disposal.

"High Level Activity," samples and wastes will remain in the segregated laboratory and be returned to the client, or disposed of as hot waste.

#### CONCLUSIONS

The analysis of inorganic and organic samples, particularly hazardous samples, is complicated by the presence of radioactivity. A system for screening and dilution of these samples according to radioactive composition and content has been applied which makes it possible to analyze most such samples in a conventional chemical laboratory. Stringent health physics protocols have successfully prevented laboratory contamination and personnel exposure.

#### REFERENCES

1. Public Law 95-580, "The Resource Conservation and Recovery Act of 1976," October 21, 1976, as amended by the "Quiet Communities Act of 1978."
2. "California Radiation Control Regulations," California Administrative Code, Title 17, Health, State of California, DOHS, Sacramento, California, 95814, June 28, 1980, revised.
3. 40CFR Part 261-24, "Hazardous Waste Management System, Identification and Listing of Hazardous Waste," Federal Register, Volume 45, No. 98, May 19, 1980.

## DISPOSAL OF SLIGHTLY CONTAMINATED RADIOACTIVE WASTES FROM NUCLEAR POWER PLANTS

Edward F. Branagan, Jr., and Frank J. Congel  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

### ABSTRACT

Current NRC regulations, with one exception, do not provide minimum levels of radioactivity in solid wastes which may be disposed of in a manner other than as radioactive waste. The exception (Title 10 Code of Federal Regulations Part 20.306) allows licensees to dispose of specified concentrations of tritium and carbon-14 used in liquid scintillation counting and in animal tissue without regard to radioactivity. However, this exception is of minimal use to nuclear power plant licensees. Current NRC regulations require that licensees dispose of even very slightly radioactively contaminated solid materials, with little flexibility as to the small quantities involved, at licensed low-level waste disposal sites or on a case-by-case basis pursuant to 10 CFR 20.302. On February 24, 1983 the NRC staff published Office of Inspection and Enforcement Information Notice No. 83-05, "Obtaining Approval for Disposing of Very-Low Level Radioactive Waste - 10 CFR Section 20.302." Since that time, the staff has reviewed and approved about 20 requests for disposal of slightly contaminated radioactive materials. This paper describes the characteristics of the disposed wastes, the review process and the staff's dose guidelines.

### INTRODUCTION

In its Policy Statement on Low-Level Waste Volume Reduction (Ref. 1), the NRC recognized the need to minimize the quantity of waste generated and shipped to commercial waste-disposal sites. Although current NRC regulations (10 CFR 20.306, Ref. 2) permit the disposal of liquid scintillation fluid and animal tissue containing less than 0.05  $\mu\text{Ci/g}$  of tritium or carbon-14 without regard to their radioactivity, there are no similar minimum levels of radioactivity in solid waste. Consequently, nuclear power plant licensees must dispose of solid waste with any licensee generated contamination at commercial waste disposal sites or on a case-by-case basis under the provisions of 10 CFR 20.302 (Ref. 3). In February, 1983 the NRC staff published Office of Inspection and Enforcement Information Notice No. 83-05 entitled "Obtaining Approval for Disposing of Very-Low Level Radioactive Waste - 10 CFR Section 20.302" (Ref. 4). This information notice encouraged nuclear power plant licensees to apply on a case-by-case basis for alternative methods for disposal of slightly contaminated radioactive materials (i.e., methods other than disposal at commercial wastes sites). Since the incineration of wastes is subject to different requirements, it is not discussed in this paper.

## CHARACTERISTICS OF DISPOSED WASTES

During the past two years the NRC staff has reviewed and approved about 20 requests for disposal of slightly contaminated radioactive materials from nuclear power plants. Table 1 contains examples of the disposed waste applications that illustrate the characteristics of the wastes. The types of waste disposed include the following slightly contaminated materials: soil, sand, sediment from onsite settling ponds, sewage sludge, wood, spent resins used for cleaning the secondary side of pressurized water reactors (PWRs), roofing materials, and scrap metal from feedwater heaters used in the secondary side of PWRs.

The principal nuclides in the disposed wastes are Co-58, Co-60, Cs-134, and Cs-137, with total activity concentrations in the range of about 1 to 50 pCi/g. These nuclides have by and large been in an immobile form.

Disposal methods have included disposal of wood in a municipal landfill, burial of resins on-site, burial of sludge in a chemical waste disposal landfill, mixing of sewage sludge in an anaerobic digester and shipment to a sanitary landfill for burial. Although most of the requests have been for a one-time disposal, the NRC staff has approved a request for disposal of limited quantities of contaminated wood on an annual basis, and a request for disposal of secondary-side resins about every 5 years. In most cases, it is expected that no member of the public will receive more than 1 mrem/yr from the disposal.

## REGULATORY REVIEW PROCESS

Title 10 Code of Federal Regulations Part 20.302 contains only general guidance for obtaining approval of proposed disposal procedures. In addition, there is no Regulatory Guide that describes specific methods acceptable to the NRC staff for obtaining approval of proposed disposal procedures. This section briefly describes the regulatory review process, and the types of information that NRC reviewers look for in an application for disposal of slightly contaminated radioactive materials from nuclear power plants pursuant to 10 CFR 20.302.

Applications should be submitted to the Office of Nuclear Reactor Regulation (NRR). Under 10 CFR 20.302, licensees may request disposal of specific material on a case-by-case basis or licensees may request permission for routine disposal of specific types of wastes on an annual basis using approved procedures and systems. Topical reports describing systems to be used for waste segregation and monitoring may be incorporated by reference in the request when such reports have been reviewed and approved by the NRC. However, at this time no topical reports have been submitted to NRC for review. For disposal of radioactive materials in non-Agreement States, the application is reviewed solely by NRC. For disposal of radioactive material on or off the reactor site in an Agreement State, approval is needed from the Agreement State. After the NRC staff's review of the request, then the staff will grant approval, if appropriate, subject to approval of the disposal by the Agreement State.

Table 1 Examples of Disposed Waste Applications

Nuclear Power Plant	Description of Waste				Disposal	
	Volume or Mass	Total Conc. (pCi/gm or pCi/cc)	Total Activity mCi	Principal Nuclides (%)	Method	Site
1. San Onofre	Sand at bottom of excavated pit, 300 m <sup>3</sup>	0.5/gm	0.05-0.2	Cs-137	Backfill with clean soil	Onsite
2. Oyster Creek	Soil, 480 m <sup>3</sup>	~10	5	Co-60(50) Cs-137(30) Mn-54(15) Cs-134(5)	Relocate the material and cover w/ 6" layer of clean soil, pave over with asphalt	Onsite
3. H.B. Robinson Unit 2	Soil, 1.5 m <sup>3</sup>	7.4/gm	0.014	Co-60(63) Cs-137(23) Cs-134(10)	Stabilize in place along the bottom of a drainage ditch by covering with a minimum 6" layer of clean soil	Onsite, along the bottom of a drainage ditch
4. Humboldt Bay	Sludge, 39 m <sup>3</sup>	12.6/cc	0.49	Co-60(62) Cs-137(36)	Dried bottoms sent to chemical waste disposal landfill, >30" soil cover	Onsite
5. Oconee, Units 1, 2 and 3	Sewage sludge, 120 m <sup>3</sup>	0.63/cc	0.07	Cs-137(45) Co-60(27)	Diluted with other sludge, dried, sent to landfill	Offsite sanitary landfill
6. H.B. Robinson Unit 2	Settling pond sediment, 60,000 m <sup>3</sup>	30/gm wet sediment	1700 (over life of ash pond)	Co-60	Transfer to ash pond, 6" thick soil cover with vegetation	Onsite fossil ash pond
7. R.E. Ginna Unit 1	Roofing materials 14/gm 70 tons	14/gm	1.4	Cs-137 (64) Co-60 (20) Cs-134 (16)	Municipal Landfill	Offsite
8. McGuire 1 & 2	Wastewater - residue sludge, 240 to 380 m <sup>3</sup> /yr	0.24	0.1	Co-58 (50) Co-60 (50)	Landspread, topsoil cover with vegetation	Onsite
9. Oconee 1, 2 & 3	Feedwater heaters: high activity tube bundles, 160 tons	13/gm	6.5	Co-60(79) Cs-137(15)	Burial in 7'-12' deep trenches, 3' clean soil cover	Company controlled area outside security fence
10. Oconee 1, 2 & 3	Sand, 45 m <sup>3</sup>	~50/gm	4.3	Cs-137 Cs-134 Co-60 Mn-54	Burial in 7'-12' deep trenches, 3' clean soil cover	Company controlled area outside security fence
11. Oconee 1, 2 & 3; McGuire 1 & 2; and Catawba 1 & 2	Wood, 12-21 m <sup>3</sup> per plant per year	35/gm	0.4 to 0.7 per year per station	Assumed Co-60 (100)	Compaction & burial in landfill, 6" clean soil cover	Offsite, Sanitary landfill
12. Davis Besse	Secondary-side resins, 150 m <sup>3</sup> every 5 years	8.7/cc	8.5/every 5 years	Cs-137(36) Co-58(34) Cs-134(27) Co-60(3)	Transfer resins to onsite settling basin, dilute with clean waste, land spread dredgings every 5 years.	Company owned land

Licensee's submittals should identify and describe the waste, disposal site, pathways of exposure, and estimate doses from the principal pathways of exposure. The information regarding the waste for each planned request should include: (1) a brief description of the item to be disposed including the approximate volume or mass; (2) identification of the principal nuclides expected to be in the waste; (3) estimates of the concentrations of the nuclides in the waste; (4) estimates of the total activity of nuclides in the waste; and (5) the basis for the estimated concentrations and total activities (i.e., the number of samples measured, the representativeness of the samples, and the appropriateness of the instruments used to measure the activity in samples). Information regarding the disposal site should include: (1) the method of disposal (e.g., diluted with other sludge, burial in deep trenches, land spread and cover with "clean" soil, etc.); (2) the location of the disposal site (e.g., a legible map of the disposal site with compass direction and scale); (3) local land use (e.g., nearby residences, wells, etc.); and (4) any physical or administrative barriers to prevent present and/or future use of the site for other than its intended purpose.\*

The licensee's submittal should briefly discuss the potential pathways of exposure, and estimate doses to individuals from the principal pathways of exposure. Doses should be estimated for both a maximally exposed member of the public, and for a maximally non-occupationally exposed worker. If a particular pathway is not of concern (e.g., groundwater), then this should be stated and the basis for the statement provided (e.g., the nuclides are in an immobile form, and there are no wells within a certain distance of the disposal site). Pathways that are typically of concern include: (1) external exposure from standing or living above the disposal site for a stated number of hours per year; (2) inhalation of resuspended radionuclides if the radioactive material is not covered promptly; (3) external and internal exposure to an inadvertent intruder; (4) external and internal exposure of an individual from assumed recycling of the disposed material at the time the disposal site is released from regulatory control; and (5) internal exposure from ingestion of food grown on the disposal site.

#### DOSE GUIDELINES

In performing its safety evaluations and environmental impact appraisals for these licensee submittals, the staff of the Office of Nuclear Reactor Regulation (NRR) has developed draft dose guidelines for use by NRR reviewers. These guidelines are intended to ensure that potential radiation doses that may result from the proposed method of disposal are maintained as low as is reasonably achievable (ALARA). The waste to which these guidelines are intended to apply are typical solid wastes from reactor facilities slightly contaminated with radionuclides with halflives less than 35 years.

\*Note: For wastes containing mobile radionuclides (e.g., H-3), detailed information on geology and hydrology may be necessary.

The draft dose guidelines were developed based on the following principles: First, the annual dose to a member of the public from exposure to the disposed material should be a small fraction of annual exposure to natural background radiation. Second, the annual dose to a member of the public from exposure to the disposed material should be no greater than the annual dose a maximally exposed individual would receive from exposure to radioactive effluents from normal operations at light water reactors. Third, concentrated sources of radioactivity that might pose a health hazard before or after the time of release of the disposal site from all regulatory controls should not be permitted to be disposed under 10 CFR 20.302.

The draft guidelines follow:

1. The radioactive material should be disposed in a manner such that it is unlikely that the material would be recycled.
2. Doses to the total body and any body organ of a maximally exposed individual (a member of the general public or a non-occupationally exposed worker) from the probable pathways of exposure to the disposed material should be less than 1 mrem/yr.
3. Doses to the total body and any body organ of an inadvertent intruder from the probable pathways of exposure should be less than 5 mrem/yr.
4. For onsite disposal, the dose to the total body and any body organ of an individual from assumed recycling of the disposed material at the time the disposal site is released from regulatory control from all likely pathways of exposure should be less than 1 mrem/yr.
5. For disposal in a sanitary landfill, the dose to the total body and any body organ of an individual from assumed recycling of the disposed material at the time of disposal from all likely pathways of exposure should be less than 5 mrem/yr.

The preceding guidelines are subject to change, and the NRC staff is still considering questions regarding the total quantity (i.e., mass, volume, and activity) that should be permitted to be disposed at one site, safeguards to prevent recycling of disposed materials, and the regulatory interface between the NRC's jurisdiction and Agreement States's jurisdiction. For disposal of radioactive materials in non-Agreement States, the licensee's application is reviewed solely by the NRC. For disposal of radioactive materials in an Agreement State, the NRC staff grants approval, if appropriate, subject to approval of the disposal by the Agreement State.

## REFERENCES

1. U.S. Nuclear Regulatory Commission, "Policy Statement on Low-Level Waste Volume Reduction", 46 Federal Register, 51100, October 16, 1981.
2. U.S. Nuclear Regulatory Commission, Title 10 Code of Federal Regulations Part 20.306, "Disposal of Specific Wastes," January 1, 1985.
3. U.S. Nuclear Regulatory Commission, Title 10 Code of Federal Regulations Part 20.302, "Method for Obtaining Approval of Proposed Disposal Procedures," January 1, 1985.
4. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, IE Information Notice No. 83-05, "Obtaining Approval for Disposing of Very-Low Level Radioactive Waste - 10 CFR Section 20.302," February 24, 1983.

**FEASIBILITY OF CENTRALIZING DISPOSAL AND  
PROCESSING FACILITIES FOR LOW LEVEL BIOMEDICAL WASTE**

T.W. Osborne, M.S. Terpilak, E.A. Tupin  
RSO, INC.  
P.O. Box 1526  
Laurel, Maryland 20707-0953

**ABSTRACT**

Research efforts were designed to examine the feasibility of a centralized processing facility for low level radioactive waste materials. The scope of the proposed facility was to provide a regional commercial processing service to medicine and biomedical research institutions who are unable to perform waste processing or treatment for themselves.

RSO, INC. has conducted an exhaustive literature search to find previous work, data and conclusions in this area. Extensive data was collected from the NRC, the DOE and several of its contractors, and the Conference of Radiation Control Program Directors. Vendors and manufacturers were surveyed to find sources of new technology hardware for compaction, shredding, and other volume reduction systems and methods.

A study of an existing commercial waste processor was made to examine the current technology, its economic viability, and requirements for successful operation.

The results of the literature search has provided an extensive description of low level wastes on a national basis. Sufficient information is now available concerning waste types, physical and chemical forms, radionuclide types and quantities, and volumes on a regional basis to support the operation of a reprocessing facility. Specifically, the total low level biomedical waste volume generated in the U.S. during 1980 was approximately 660,000 cu. ft. Commercial reprocessing operations which could handle approximately 81% of this volume and achieve a gross effective volume reduction ratio of 4 to 1 (including materials totally disposed of by the reprocessor) would be supported by a 15 million dollar annual industry.

New technologies exist which are not currently used in commercial operations. There is an excellent potential for their use to provide an increase in the volume reduction rates on a cost effective basis. Current volume reduction rates of 3-5:1 could be increased to 10:1.

**INTRODUCTION**

RSO, INC. was awarded a Small Business Innovative Research (SBIR) contract to examine the feasibility of a centralized processing facility for low level radioactive waste materials. The scope of the proposed facility was to provide a regional commercial processing service to biomedical research institutions who are unable to perform waste processing or treatment for themselves.

The management of low-level radioactive waste (LLW) has become a topic of intense interest and concern in the past several years. This intense interest

has been exhibited by the public in general and by the Congress through the introduction of various bills relating to LLW, and in particular, by the passage of the Low-Level Waste Policy Act of 1980.

Intense interest in LLW issues has been generated in individual state and local governments as existing commercial sites began to close or reduced the volume of waste being accepted. This situation, which first became critical in the fall of 1979, has made government officials and waste generators recognize the potential unavailability of facilities to dispose of their radioactive waste and the consequences that would result.

Decisive actions concerning waste disposal must now be taken within a reasonably short time period to allow for the uninterrupted use of radioactive materials that have been beneficial to mankind. The decision making process regarding the management of LLW, however, must always keep the public health and safety foremost. Of particular concern are certain types of radioactive wastes generated by institutions (such as biomedical research facilities, hospitals and universities), which in the past have been restricted from land burial by some of the existing commercial disposal sites. Although usually relatively low in radioactivity, these restricted institutional wastes may be chemically toxic or pathogenic or the chemicals may degrade disposal site performance.

The study determined the feasibility of centralizing facilities for processing and disposal of LLW from biomedical generators. Most of these waste generators do little or no in-house processing before shipping waste to a commercial burial ground. The reasons cited are frequently economic or regulatory limitations. Centralization of processing and disposal facilities to serve a number of generators represents a unique approach to the problem. Such a facility would use a variety of state-of-the-art equipment to treat many radwaste forms. Feasibility was studied from several standpoints.

1. Analysis and characterization of the waste stream.
2. Facility and equipment engineering and design.
3. Economic, regulatory and environmental impacts and requirements.
4. Political implications from the federal, state, and local points of view.

## CHARACTERIZATION OF WASTE STREAMS

### Definitions

"Low-Level radioactive waste" is defined only in a negative sense. All radioactive waste which is not high level waste (i.e., spent fuel from nuclear reactors and wastes from reprocessing spent reactor fuel), which is not uranium mine and mill tailing, and which is not transuranic waste, is LLW.

Biomedical wastes are defined as those radioactive wastes that are generated in hospitals, biomedical research facilities and similar institutions. It is waste leftover from biomedical research and various nuclear medicine procedures that use radioisotopes. A large part of the waste is in liquid form, such as the liquid scintillation counting media. (This is usually a toluene, xylene, or pseudocumene based solvent with organic phosphors in which the radioactivity laden materials are placed for counting.) The composition also includes trash - absorbent paper, protective clothing, contaminated plastics, sealed sources such as cesium check sources; biological wastes such as animal

carcasses, bedding and excreta; and other forms.

#### Institutional waste (1)

Institutional wastes are defined as those radioactive wastes generated in hospitals, universities, research centers and similar facilities.

Sources of institutional waste include hospitals, universities, and research centers. Medical institutions generate wastes in diagnosis and therapy procedures that use radioactive materials. An estimated 80 to 100 million nuclear medicine procedures are performed annually in the United States. Most radionuclides used in nuclear medicine have half-lives of less than seven days. Radionuclides are also used in medical research studies to obtain medical data and to develop clinical and diagnostic or therapeutic procedures. Tritium, with a half-life of 12.3 years, is the principal radionuclide found in biological research wastes. Most other nuclides used in biological research have half-lives of less than one year. (Some bioresearch wastes containing carbon-14 and tritium have been classified as nonradioactive according to a recent ruling by the Nuclear Regulatory Commission.)

LLW is generated through research in physics, inorganic chemistry, materials analysis, and geology. Much of that waste is produced through use of charged-particle accelerators or small research nuclear reactors. Some LLW is generated as a result of instructional or classroom use of radioactive materials.

The following types of waste are generated in institutional activities.

Dry solids--Protective clothing, gloves, small tools, plastics, rags, paper, and packaging materials are typical wastes from all institutional sources.

Liquid scintillation waste--Scintillation "cocktails" consist of an organic fluid (usually toluene, xylene, or pseudocumene), the material being analyzed (water, tissue, etc.) and a radiation-sensitive substance. These are contained in a plastic or glass vial for counting. Liquid scintillation waste includes both the liquid and the vial, although sometimes the two are disposed of separately.

Organic liquids--Organic laboratory solvents such as alcohols, aldehydes, ketones, and organic acids (excluding scintillation fluids) are other organic liquid wastes most commonly associated with bioresearch.

Aqueous liquids--Most liquids associated with the medical use of radionuclides are aqueous, that is, in a water solution. These include the washings from contaminated laboratory ware in research facilities.

Biological wastes--These wastes consist mainly of the carcasses of animals used in biological research, animal bedding, and excreta. Labelled culture media are also included.

Gaseous wastes--Hospitals often use gaseous xenon-133 (half-life 2.19 days) or xenon-127 (half-life 36.4 days) for ventilation studies of lung capacity. Most of these gases can be trapped in activated-charcoal filters, which are then disposed of as LLW.

Typical primary radionuclides found in institutional wastes are listed in Table 1. (2) (3)

TABLE 1

## PRIMARY RADIONUCLIDES FOUND IN INSTITUTIONAL WASTES

RADIONUCLIDE	PREDOMINANT HALF-LIFE	IN/WITH WASTE TYPES	USED		PRIMARY (%)	TYPE OF WASTESTREAM	RADIATION
			SHIPPED	PATIENTS			
Calcium-45	165 days	Research fluids	--	--	--	Bioresearch	Beta
Carbon-14	5,730 years	Research fluids	--	--	3	Bioresearch	Beta
Cesium-137	30 years	--	--	Radiation therapy	--		Beta, Gamma
Chromium-51	27.8 days	Research fluids	--	--	2	Bioresearch	Gamma
Gallium-67	77.9 hours	Syringes	Yes	--	--	Medical	Gamma
Hydrogen-3 (Tritium)	12.3 years	Research fluids	--	--	75	Bioresearch 65% Other research 35%	Beta
Indium-111	2.8 days	Bottles/ syringes	Yes	--	--	Medical	Gamma
Iodine-123	13.3 hours	Containers/ syringes	Yes	--	--	Medical	Gamma
Iodine-125	60.2 days	Clinical and research fluids	Tests outside body	11	Medical 40% Bioresearch 59%		Gamma
Iodine-131	8.05 days	Bottles/ syringes	Yes	--	--	Medical	Beta, Gamma
Molybdenum-99	66.7 hours	Depleted sources	--	--	--	Medical	Beta, Gamma
Phosphorus-32	14.28 days	Bottles/ syringes/ research fluids	Yes	5	Bioresearch Medical		Beta
Rubidium-86	18.66 days	Research fluids	--	--	--	Bioresearch	Beta, Gamma
Selenium-75	120.4 days	Bottles/ syringes	Yes	--	--	Medical	Gamma
Sulfur-35	87.9 days	Research fluids	--	3	Bioresearch		Beta
Technetium-99m	6.05 hours	Depleted sources	--	--	--	Medical	Gamma
Xenon-133	5.27 days	Bottles/ collectors	Yes	--	--	Medical	Beta, Gamma
Ytterbium-169	37.8 days	Bottles/ syringes	Yes	--	--	Medical 58% Other research 42%	Beta

### Waste Volume Projections.

Approximately 80,000 cubic meters (2,825,120 cu. ft.) of LLW were handled by the three commercial sites in 1979; 107,000 cubic meters in 1980, 88,000 cubic meters in 1981, 76,000 cubic meters in 1982 and 77,000 cubic meters in 1983. In 1981 Barnwell, South Carolina took 50 percent of the volume, Beatty, Nevada, 4 percent and Hanford, Washington, 46 percent. Barnwell was mandated to reduce to a volume of 34,000 cubic meters annually beginning in October 1981; it accepted about 44,000 cubic meters in all of 1981 so its acceptance of waste is expected to be restricted further. (4)

Total volume from 1980-2000 is projected at 3,620,000 cubic meter with 65 percent from fuel cycle sources, 19 percent institutional, and 16 percent industrial. Volumes from individual states vary widely.

It is quite apparent that the institutional radioactive waste generated will increase quite significantly in the next 15 years.

Figure 1 shows the projected waste volumes through the year 2000 (government and military waste not included).

Table 2 gives the volumes of commercial LLW through the year 2000. The projections in the table are based on the 1979 disposal data. It is expected that at least 184,000 cubic meters of LLW will be generated annually by 1990. By the year 2000, approximately 224,000 cubic meters of waste may be generated annually. (5)

### MANAGEMENT OF LOW-LEVEL RADIOACTIVE WASTE

The proper management of LLW has become a major concern of governmental agencies; federal, state, and some local governments; and of the general public as a whole. The issue is also of much concern to those industries generating radioactive wastes, i.e., primarily the nuclear medicine and associated pharmaceutical manufacturers.

The Conference of Radiation Control Program Directors, Inc. an organization of persons with radiation protection responsibilities at the state and local government levels, has defined institutional radioactive waste and identified and quantified origins and current disposal method.

They identified six (6) alternative, environmentally acceptable methods for managing these institutional wastes.

These alternatives are:

1. Incineration
2. Storage for decay
3. Dilution and dispersion
4. Segregation of waste
5. Reduction in generation
6. Establishment of unlimited release levels for solids

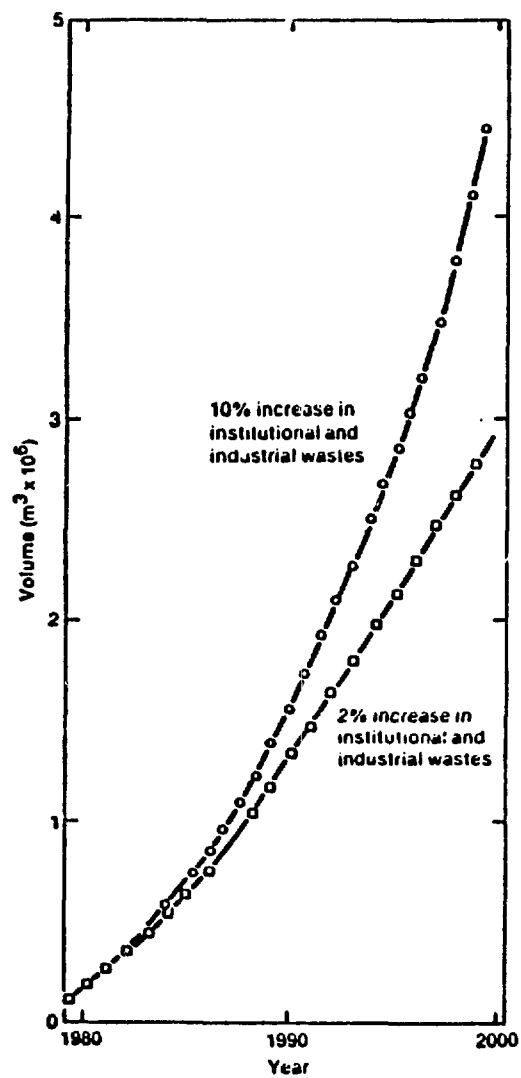


Fig. 1 Projected waste volumes through the year 2000 (government and military wastes not included)

TABLE 2  
ESTIMATED VOLUMES OF LLW REQUIRING DISPOSAL

Year	Volume (thousands of cubic meters)		Total
	Reactor Waste Only <sup>a</sup>	Institutional and Industrial Wastes <sup>b</sup>	
1982	45.3	41.3	86.6
1983	56.0	43.8	99.8
1984	70.3	46.4	116.7
1985	80.2	49.1	129.3
1986	90.4	51.1	141.5
1987	104.3	53.3	157.6
1988	109.4	55.4	164.8
1989	118.7	57.6	176.3
1990	124.3	59.4	183.7
1991	125.1	61.1	186.2
1992	125.1	62.9	188.0
1993	126.4	64.7	191.1
1994	127.8	66.7	194.5
1995	132.0	68.7	200.7
1996	133.5	70.8	204.3
1997	135.9	73.0	208.9
1998	139.0	75.1	214.1
1999	140.1	77.4	217.5
2000	144.1	79.7	223.8

Note: To convert to cubic feet, multiply by 35.314.

- a. It was assumed that the U.S. commercial nuclear capacity would be 54,138, and 180 gigawatts (electrical) at the end of the years 1980, 1990, and 2000 respectively.
- b. The following growth rates were assumed in the calculations: 6 percent for 1982-1985, 4 percent for 1986-1989, and 3 percent for 1990-2000.

Of the identified alternatives, at least three are of such economic, health, and safety significance, that they should be used for all types of waste by all generators. In order of priority, they are:

1. Reduction in generation
2. Segregation of waste to reduce volume
3. Storage for decay

Special emphasis was placed on incineration as a method to reduce the volume of institutional waste. This study concluded that:

Incineration appears to be an effective process for volume reduction of institutional LLW.

Incineration technology has been used with varying degrees of success over the past two decades; new incineration technologies being developed should make this method of disposal more feasible than it has been in the past.

Incineration may create special problems such as increased occupational exposure and more restrictive requirements for handling and disposal.

Incineration does not reduce total radioactivity and may have some environmental impact on land, air, and water resources.

Regulatory requirements of various agencies may create problems for obtaining permits for incinerators.

Proposals for new incinerators meeting regulatory requirements may still meet public oppositions.

In addition, there is an open issue as to what further processing might be required for the ash in order to stabilize it for shipment and burial. Generally it would be expected that solidification might be required. Finally, due to the high use of PVC plastics at facilities, complex off-gas systems would be required on incinerators, often generating a secondary waste stream with the salts from the scrub solution. This may result in incineration being a complicated and less cost effective solution than first thought as an approach to volume reduction.

## ECONOMIC ANALYSIS

### Current versus projected costs for waste disposal

Currently about 50,000 cubic meters (1.77 million cubic feet) of commercial LLW generated annually is due to institutional and industrial sources. To help visualize this volume, picture the 236,000, 55 gallon drums that would be necessary to contain this amount. In 1982, a reported volume of 75,891 cubic meters of radioactive waste was buried in the commercial disposal facilities. The 1982 analysis by the Conference of Radiation Control Program Directors shows that institutions (hospitals, medical labs) account for 14 percent of the total volume.

As disposal costs for LLW continue to escalate and available space decreases, LLW generators must find cost effective solutions to deal with this problem. The Low-Level Radioactive Waste Policy Act of 1980 has complicated the

situation by allowing congressionally approved compacts to restrict use of their land burial disposal facilities after January 1, 1986 to the constituents of member states.

A recent study and proposal for Regional LLW Volume Reduction Services Facility addressed the overall current and projected costs for LLW disposal.

Figure 2 and Table 3 show the economics, cost estimates and projected costs from 1984 through 1989. The projected costs indicate that untreated trash (DRW) will escalate from the present 1984 costs of \$300.00 per drum (7.5 cu. ft., about \$40/cu. ft.) to a value approaching \$546 per drum (7.5 cu. ft., about \$73.00/cu. ft.) in 1989, an increase of almost 80 percent during the five-year period.

#### Facilities Capabilities

A typical facility to handle the Low-Level Biomedical Research waste should have the following capabilities:

- Serve as a regional volume reduction services facility.

- Process Low-Level Biomedical Research Waste; dry solids, medical and research materials.

- Provide a LLW brokerage services.

- Provide the capability for short term storage.

- Utilize state-of-the-art technology and advanced VR methods.

- Provide a more cost/effective technology to save generators money in burial and storage.

#### Operational Capabilities of the Facility

The Low-Level Biomedical Research Facility should be able to provide the following volume reduction capabilities:

##### 1. Incineration

- Burnable materials packaged in burnable (fiber) containers - 55 gal. or 2' x 2' x 3' boxes (max.).
- Limited to 5 percent pure PVC and min metal (desired).
- Ash packaged for burial or storage (55 gal. pending acceptance by Barnwell).
- Volume Reduction about 16:1 over compacted waste or better (Design 33:1).

##### 2. "High-Force" Compaction

- "Non-incinerable" materials - metal, large percent PVC, etc.
- Materials packaged in metal drums (suggest precompact).

## Projected Costs

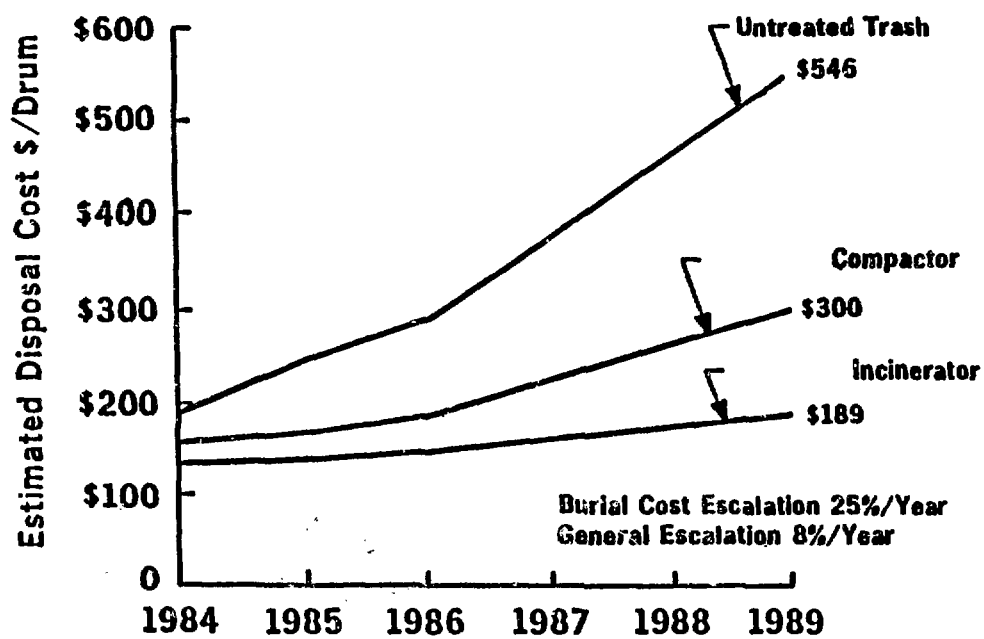


Figure 2

TABLE 3

## ECONOMICS

	COST ESTIMATE			
	1984	1985	1986	1989
Baseline cost \$/drum (Transportation & burial, 20 #/of)	\$198	\$241	\$294	\$546
Incineration service \$/drum (Volume reduction, trans., and burial)	\$132	\$134	\$146	\$189
High-force compaction service \$/drum (Volume reduction, trans., and burial)	\$155	\$170	\$194	\$300
Savings* \$				
200,000 #/yr incineration	\$87,000	\$142,000	\$198,000	\$475,000
100,000 #/yr compaction	\$28,000	\$47,000	\$67,000	\$163,000
Total savings	\$115,000	\$189,000	\$265,000	\$638,000

\*Does not include savings for generator's internal costs.

- c. Metal drum overpack - 70 gal.
  - d. volume Reduction about 3:1 over precompacted (25 lbs./cu. ft.).
3. Sorting
- a. Boxed trash, uncompacted for sorting
  - b. Trash sorted for compaction or incineration.

#### Construction and Capital Costs

Table 4 summarized the overall cost for the facility design, engineering and construction costs assuming a generic incinerator design.

The DAW capacity of the incinerator is assumed to be 250,000 cu. ft./yr. of uncompacted material, which represents current technology of many such units available.

#### REGULATORY ANALYSIS

The process for establishing a centralized disposal and processing facility for Low-Level Biomedical Waste will be time consuming. Depending on the state and location within that state, various federal, state, and local permits and reviews will be necessary and required. It is anticipated that about 1 to 2 years licensing period will be required. The process as established by the NRC and/or state includes the selection of a region of interest, involving the review and evaluation of alternative sites based on the NRC's and/or states site suitability standards. Alternative sites must be identified, reconnaissance data collected, and evaluations conducted prior to submitting an application for a license for a particular site. Evidence of evaluations of alternative sites, along with more detailed submittals on the suitability of a particular site, must be part of the application for a license. The process, also includes public notice, public hearing, environmental assessment reports, and findings of compliance with the siting standards.

#### PUBLIC PERCEPTIONS

By 1979, it had become very apparent that a commercial waste disposal system composed of only three shallow land burial sites, and these not well-matched to waste generating regions, did not make economic or political sense. This realization led national and state policymakers to reconsider governmental roles and responsibilities, leading to a consensus that the states should assume additional responsibility for commercial wastes generated within their borders and take a leadership role in establishing a working waste management system. The Low-Level Waste Policy Act reflects this consensus.

State leadership in the management of LLW is based on the principle that the level of government nearest to the problem and its potential resolution should assume primary responsibility. LLW management is within the resource capabilities of the states and should therefore be addressed at that level.

State leadership can address facility siting issues at the state and local level. This is likely a preferable and more successful approach than a solution mandated by the federal government.

TABLE 4

## COST ANALYSIS

	AMOUNT WITH INCINERATOR	AMOUNT WITHOUT INCINERATOR
1. DIRECT COST		
a. NRC and/or state licensing for the facility	\$100,000	\$100,000
b. Architecture and Engineering consultation fees	\$100,000	\$100,000
c. Incinerator		
1. Incinerator engineering/design	\$500,000	--
2. Incinerator fabrication	\$1,000,000	--
3. Installation, checkout, and start-up	\$200,000	--
d. Site preparation	\$75,000	\$75,000
e. Land cost	\$150,000	\$150,000
f. Building construction (includes solidification, compaction, and drum handling systems)	\$2,000,000	\$2,000,000
TOTAL DIRECT COST	\$4,125,000	\$2,425,000
2. FINANCING		
Investment costs	\$4,125,000	\$2,425,000
Annual cost of investment (financing at 13%)	\$536,250	\$315,250
Annual depreciation (10 yr. life)	\$412,500	\$242,500
TOTAL ANNUAL COST	\$948,750	\$557,750
3. OPERATING COSTS		
Work schedule-staff (3 shifts-7 dys/1 wk)		
a. 4 technician operators		
8000 person hours/yr at \$25/hr	\$200,000	\$200,000
b. 4 technician assistants		
8000 person hours/yr at \$15/hr	\$120,000	\$120,000
Total	\$320,000	\$320,000
c. Utilities	\$75,000	\$75,000
d. Maintenance and technical support	\$300,000	\$300,000
e. Cost for disposal of ash	\$100,000	--
TOTAL ANNUAL OPERATING COSTS	\$795,000	\$695,000
4. TOTAL ANNUAL COST		
a. Financing	\$948,750	\$557,750
b. Operation costs	\$795,000	\$695,000
TOTAL ANNUAL COST	\$1,743,750	\$1,252,750

Planning and implementation of state policy on LLW management should involve the executive and legislative branches of state government, local officials, special interest groups, and the general public. There are many technical, social, and economic factors to consider and incorporate. This inherent complexity requires frequent and clear communications among all the participants. This will help ensure that the proper issues are discussed.

In the long run, the most effective way to minimize the local conflicts in siting hazardous waste facilities is for greater federal and state support for local land-using planning. If local governments are asked to assume responsibility for hosting disposal facilities, they should be familiar with the local and regional needs for such facilities and with the capacity of local land resources to serve as safe waste repositories. Additionally, the local planning that identifies floodplains, wetlands, water resources, or natural hazards may be critical in siting disposal facilities and important in minimizing local opposition to siting decisions. A number of local governments have demonstrated the capacity to plan successfully for hazardous waste facilities.

## CONCLUSIONS

As a result of this study the feasibility and benefits of a centralized disposal and processing facility for low-level biomedical research waste can be identified as follows:

The cost to generators of LLRW will be reduced by an estimated 30 percent over the cost of shallow land burial. This can be shown to represent a nationwide savings of \$18,000,000 per year to non fuel cycle generators.

A centralized facility would provide the capability for an efficient and cost effective service as compared to volume reduction activities conducted by the generator.

The operation of a centralized facility would also insure the continuity of biomedical research at institutions, universities, medical schools and other facilities in the event of a reduction or curtailment of the existing land burial facilities, or the restriction of these facilities to local region or compacts.

The volume reduction techniques would reduce the total volume of LLRW for burial, thus requiring less land area for ultimate disposal of these wastes. As a corollary to this benefit, the existing burial space would be reserved for non reprocessible waste more appropriate to occupy this critical resource.

Regional centralized facilities would produce a smaller shipping volume for cross country transportation of hazardous materials with a corresponding reduction in vehicles on the highway.

Licensed commercial reprocessors will provide, at a national level, a more efficient and uniform system for LLRW disposal.

The establishment and operation of the centralized pilot facility can serve as a model for similar plants in other regional waste compact areas.

As an added economic benefit to the waste generator, the transfer of wastes to a reprocessor constitutes a transfer of title, thus relieving the generator of the long term liability for these materials associated with disposal by means of shallow land burial.

#### SELECTED REFERENCES FOR TEXT

1. Institutional Radioactive Wastes with Restrictions for Land Burial and Environmental Methods to Manage Such Waste, DOE/LLW-ST, National Low-Level Radioactive Waste Management Program, work performed by Conference of Radiation Control Program Directors, Inc. under DOE Grant No. DE-FG07-801D12109, prepared by EG&G/Idaho, Inc. for the U.S. Department of Energy under DOE Contract No. DE-AC07-761D01570, November 1982.
2. Low-Level Radioactive Waste Management Handbook Series. An Introduction, National Low-Level Radioactive Waste Management Program, DOE-LLW-13TA, EG&G, Idaho Falls, Idaho, August 1983.
3. Understanding Low-Level Radioactive Waste, National Low-Level Radioactive Waste Management Program, DOE-/LLW-2, EG&G, Idaho Falls, Idaho, October 1983.
4. G. John Weir, Jr., M.D., Low-Level Radioactive Waste, American College of Nuclear Physicians, Public Education Program, April 1983.
5. Low-Level Radioactive Waste Management: An Update. A Legislator's Guide, Julie M. Jordon, National Conference of State Legislatures, Washington, D.C., October 1984.

## AN OVERVIEW OF THE SURPLUS FACILITIES MANAGEMENT PROGRAM

C. E. Miller, Jr., U.S. Department of Energy and  
B. L. Baumann, UNC Nuclear Industries  
Department of Energy-Richland  
P. O. Box 550  
Richland, WA 99352

### ABSTRACT

In 1978, the Department of Energy established the Surplus Facilities Management Program (SFMP) to provide for the safe maintenance and eventual disposal of nuclear facilities used in early defense and nuclear energy programs. SFMP's current inventory includes approximately 300 shutdown, radioactively contaminated facilities located throughout the United States and Puerto Rico.

SFMP's responsibilities include maintenance and surveillance of facilities awaiting decommissioning, planning, and decommissioning implementation. Some examples of major SFMP decommissioning projects underway include: the Shippingport Atomic Power Station in Shippingport, Pennsylvania; the eight plutonium production reactors on the Hanford Site near Richland, Washington; and the Weldon Spring Site located near St. Louis, Missouri.

This paper presents an overview of the Surplus Facilities Management Program and highlights its major decommissioning projects.

## INTACT DECOMMISSIONING OPTION FOR NUCLEAR POWER PLANTS

S. L. Ostrow, O. I. Oztunali and T. J. Grant  
Ebasco Services, Inc.  
2 World Trade Center  
New York, N.Y. 10048

### ABSTRACT

Intact decommissioning has been proposed as an alternative method for decommissioning nuclear power plants complying with NRC license termination requirements as set forth in 10 CFR 50 and Regulatory Guide 1.86. Under this method, after the fuel and radioactive liquid and wastes are removed, highly radioactive systems and components, such as the reactor vessel and internals and the primary coolant system, would be sealed inside a designated boundary (e.g., the containment building) and left indefinitely. Radioactive materials outside the intact decommissioning boundary would be removed. Minimization of preparatory work, no consolidation of radioactive components in a particular location, and no maintenance or surveillance of the decommissioned facility distinguish this from the entombment option. The associated occupational radiation exposure should be significantly lower, and the public exposure no greater - and quite likely less - than the currently favored decommissioning method, immediate dismantlement. This paper presents the results of a study, sponsored by the National Environmental Studies Project of the Atomic Industrial Forum (AIF/NESP), examining the viability and radiological consequences of intact decommissioning.

### INTRODUCTION

The first systematic and comprehensive study of nuclear power plant decommissioning methods appeared in 1976 as an AIF/NESP report by Manion and LaGuardia.(1) Subsequently, the NRC contracted the Battelle Pacific Northwest Laboratory to produce a series of papers on the decommissioning of various generic fuel cycle facilities. NUREG/CR-0672(2) treated boiling water reactors (BWRs), and NUREG/CR-0130(3) pressurized water reactors (PWRs). These reports discussed methodologies, occupational and public doses, and costs, of the immediate dismantlement, entombment and mothballing decommissioning options. A paper by Lionel Lewis of Duke Power Company, presented at the 1982 Health Physics Society Annual Meeting,(4) first publicly aired the concept of intact decommissioning as a possible attractive alternative to the other methods. This was further developed in a paper(5) presented at a 1985 NRC-sponsored conference by Lewis and Stephen Ostrow of Ebasco Services. Finally, an extensive feasibility and dose assessment study of intact decommissioning was carried out by the authors and others at Ebasco under contract to the Atomic Industrial Forum National Environmental Studies Project (AIF/NESP).(6) Several recent Canadian studies(7,8,9) examined the possibility of long delays in dismantlement of plants in which the fuel and radioactive liquids have been removed and most containment penetrations sealed. These studies stop short of considering permanent storage without maintenance or surveillance, characteristic of intact decommissioning.

Current NRC regulations recognize three decommissioning methods: immediate dismantlement, entombment, and mothballing. Only the first option, however, is viewed as an acceptable permanent decommissioning solution. The new concept of intact decommissioning is simple and appealing. A substantial part of the occupational dose in immediate dismantlement comes from relatively short-lived isotopes. A large portion of the cost of entombment and mothballing lies in protracted maintenance and surveillance. In addition, entombment requires a large amount of work to prepare the entombment structure and to move radioactive components into it. By far the greatest portion of the residual activity after removal of the fuel resides in the activated reactor vessel and internal components. Thus, if the most highly radioactive material could be left in place, the containment structure sealed, the remaining structures removed, and the plant left without maintenance or surveillance, there would be large savings in occupational exposure from elimination of work tasks in radioactive areas, and in the overall cost of decommissioning.

This paper is based on the work performed in Reference 6, which determined the location and preparation of intact decommissioning boundaries, examined the ability of the enclosing structures to safely contain the residual radioactivity, and quantified associated occupational and public exposures. It was found that the intact decommissioning method is technically feasible, and would result in lower occupational, and equal or lower public exposures compared to immediate dismantlement.

#### REFERENCE PLANTS

Representative PWR and BWR plants were chosen as references. The PWR is the Trojan Nuclear Plant, owned by Portland General Electric Company, and located near Portland Oregon. It is the same plant studied in NUREG/CR-0130. (3) The plant contains a 3,500 MW(t) Westinghouse pressurized water reactor, producing a net power of 1,175 MW(e), and went into commercial operation in 1976. Major structures housing radioactivity are the Reactor Containment Building, Auxiliary Building, and Fuel Handling Building. The Reactor Containment Building is a cylinder with a hemispherical dome, and is approximately 210 feet high and 140 feet in diameter with a four foot thick outer wall lined with 1/4" of steel. It houses the nuclear steam supply system, including the nuclear reactor, the primary system piping and pumps, the four steam generators, and the pressurizer.

The PWR reactor vessel, located in the reactor cavity in the central, lower part of the Containment, is surrounded by a six foot thick, reinforced concrete biological shield wall. The vessel is a 44' high by 16' diameter cylinder with hemispherical heads, and is constructed of 8-12" thick carbon steel clad on the inside with 0.156" of stainless steel. It contains the core, thermal shield, core support structure, in-core instrumentation, core barrel, and control rod guide tube assemblies.

The Millstone Unit No. 1 plant, owned by Northeast Utilities and located in Connecticut, was chosen as the reference BWR plant. It began commercial operation in 1970. It contains a 2,011 MW(t) General Electric boiling water reactor, producing a net power of 660 MW(e). The structures housing radioactivity are the Reactor Building, Turbine Building, and the Radwaste and Radwaste Storage Buildings. The Reactor Building is a 142'x108' by 177' high rectangle surrounding the "light bulb" shaped pressure suppression primary containment system. The primary containment system is a Mark I design typical of early BWRs and consists of a drywell containing the reactor vessel and the recirculation system, and the partially water-filled toroidal pressure suppression chamber.

The drywell is a carbon steel pressure vessel with a roughly spherical lower portion approximately 64' in diameter by 62' high, and a 34' high cylindrical upper portion. It is enclosed in reinforced concrete for both radiation protection and mechanical strength. It is shielded on the top by a removable, segmented concrete shield plug. The cylindrical reactor vessel (with hemispherical heads) contains the reactor core and supports, control rod guide tubes, steam separators and dryers, jet pumps, and various safety systems. The vessel has an internal height of 64'-8", and an internal diameter of 18'-8". The carbon steel walls are at least 5-11/16" thick, and are clad on the inside by 1/8" of stainless steel.

#### RESIDUAL RADIOACTIVITY

Dose assessment, particularly in analyzing potential pathways leading to public exposure, requires, in addition to the location and quantity of each radionuclide, knowledge of the form of the radioactivity enclosed in the intact decommissioning barrier. This leads to grouping the residual radioactivity into five categories of different binding matrices:

- (1) Activated Stainless Steel - Reactor internals, composed of Type 304 stainless steel, become activated by neutrons from the core. Radionuclides have very high specific activities, and are immobilized inside the corrosion-resistant metal.
- (2) Activated Carbon Steel - The reactor pressure vessels are made of SA533 carbon steel which becomes activated by neutrons bombardment. The specific activities are considerably lower than in the stainless steel internals, however, the binding matrix is much less corrosion resistant.
- (3) Activated Concrete - Concrete and steel rebar in the reactor cavity become activated from neutrons escaping from the reactor vessel. Significant activation occurs only along approximately 15 feet of the reactor cavity vertically centered on the reactor core, and to a depth of only about 16 inches into the concrete. Radionuclides are relatively long-lived and have low specific activities. The activity is subject to migration only following degradation of, or leaching from, the concrete.
- (4) Contaminated Internal Surfaces - Activated corrosion and fission products from the fuel travel through the reactor coolant water system throughout the radioactive liquid systems in the plant. A portion forms a hard metallic oxide scale on the inside surfaces of pipes and equipment. This contamination is not easily mobilized.
- (5) Contaminated External Surfaces - Plant surfaces become contaminated over the lifetime of the plant from leaks, spills and airborne migration of radionuclides. The specific activity is low, however, the contamination is widespread and easily mobilized.

Radionuclide inventories were calculated for the above five categories as a function of time from plant shutdown to 10,000 years. In order to allow a consistent comparison of intact decommissioning doses to those from other decommissioning methods, the source terms were taken or derived from data in the Battelle decommissioning studies, NUREG/CR-0672(2) for the BWR and NUREG/CR-0130(3) for the PWR. The data in the BWR case were scaled appropriately to account for the different plant sizes in this study and the NUREG which considered a 3,320 MW(t) Mark II containment plant.

The drywell is a carbon steel pressure vessel with a roughly spherical lower portion approximately 64' in diameter by 62' high, and a 34' high cylindrical upper portion. It is enclosed in reinforced concrete for both radiation protection and mechanical strength. It is shielded on the top by a removable, segmented concrete shield plug. The cylindrical reactor vessel (with hemispherical heads) contains the reactor core and supports, control rod guide tubes, steam separators and dryers, jet pumps, and various safety systems. The vessel has an internal height of 64'-8", and an internal diameter of 18'-8". The carbon steel walls are at least 5-11/16" thick, and are clad on the inside by 1/8" of stainless steel.

#### RESIDUAL RADIOACTIVITY

Dose assessment, particularly in analyzing potential pathways leading to public exposure, requires, in addition to the location and quantity of each radionuclide, knowledge of the form of the radioactivity enclosed in the intact decommissioning barrier. This leads to grouping the residual radioactivity into five categories of different binding matrices:

- (1) Activated Stainless Steel - Reactor internals, composed of Type 304 stainless steel, become activated by neutrons from the core. Radionuclides have very high specific activities, and are immobilized inside the corrosion-resistant metal.
- (2) Activated Carbon Steel - The reactor pressure vessels are made of SA533 carbon steel which becomes activated by neutrons bombardment. The specific activities are considerably lower than in the stainless steel internals, however, the binding matrix is much less corrosion resistant.
- (3) Activated Concrete - Concrete and steel rebar in the reactor cavity become activated from neutrons escaping from the reactor vessel. Significant activation occurs only along approximately 15 feet of the reactor cavity vertically centered on the reactor core, and to a depth of only about 16 inches into the concrete. Radionuclides are relatively long-lived and have low specific activities. The activity is subject to migration only following degradation of, or leaching from, the concrete.
- (4) Contaminated Internal Surfaces - Activated corrosion and fission products from the fuel travel through the reactor coolant water system throughout the radioactive liquid systems in the plant. A portion forms a hard metallic oxide scale on the inside surfaces of pipes and equipment. This contamination is not easily mobilized.
- (5) Contaminated External Surfaces - Plant surfaces become contaminated over the lifetime of the plant from leaks, spills and airborne migration of radionuclides. The specific activity is low, however, the contamination is widespread and easily mobilized.

Radionuclide inventories were calculated for the above five categories as a function of time from plant shutdown to 10,000 years. In order to allow a consistent comparison of intact decommissioning doses to those from other decommissioning methods, the source terms were taken or derived from data in the Battelle decommissioning studies, NUREG/CR-0672(2) for the BWR and NUREG/CR-0130(3) for the PWR. The data in the BWR case were scaled appropriately to account for the different plant sizes in this study and the NUREG which considered a 3,320 MW(t) Mark II containment plant.

Finally, Tables 3 and 4, for the BWR and PWR respectively, break down the activity in the stainless steel reactor vessel internals by isotope. For at least the first 50 years after shutdown, the external radiation hazard is dominated by the gamma emitter Co-60, and the internal hazard by the beta emitters Fe-55 and Ni-63. By 100 years, most of the Co-60 has decayed away, and the major direct dose contributor becomes Nb-94, which, although present as only a few curies (less than ten), has a 20,000 year half life. The internal hazard becomes dominated by Ni-59 (75,000 year half life) after about 1,000 years.

Table 3: BWR Radionuclide Inventory of Stainless Steel Vessel Internals

Isotope	Half Life	Activity (Ci) at Time (yr) After Shutdown				
		0	3	100	1,000	10,000
Cr-51	2.77+1 days	2.44+6 (a)	-(b)	-	-	-
Fe-59	4.46+1 days	4.60+4	1.88-3	-	-	-
Co-58	7.08+1 days	3.53+4	7.81-1	-	-	-
Zn-65	2.44+2 days	5.42+1	2.41+0	-	-	-
Mn-54	3.12+2 days	1.43+4	1.26+3	-	-	-
Fe-55	2.70+0 yrs	1.55+6	7.16+5	-	-	-
Co-60	5.27+0 yrs	5.64+5	3.81+5	1.15+0	-	-
Nb-93m	1.36+1 yrs	2.27-1	1.94-1	1.38-3	-	-
Ni-63	1.00+2 yrs	1.47+5	1.44+5	7.37+4	1.44+2	-
Ag-108 -108m	1.27+2 yrs	1.23-1	1.21-1	7.16-2	-	-
Mo-93	3.00+3 yrs	5.48-1	5.47-1	5.35-1	4.34-1	5.43-2
C-14	5.73+3 yrs	1.77+2	1.76+2	1.74+2	1.56+2	5.26+1
Nb-94	2.00+4 yrs	2.52+0	2.52+0	2.51+0	2.42+0	1.69+0
Ni-59	7.50+4 yrs	1.07+3	1.07+3	1.07+3	1.06+3	9.71+2
Tc-99	2.14+5 yrs	5.34-2	5.34-2	5.34-2	5.32-2	5.17-2
Total Activity		4.78+6	1.24+6	7.45+4	1.36+3	1.03+3

Notes: (a) read as  $2.44 \times 10^6$   
(b) less than  $10^{-3}$

Table 4: PWR Radionuclide Inventory of Stainless Steel Vessel Internals

Isotope	Half Life	Activity (Ci) at Time (yr) After Shutdown				
		0	3	100	1,000	10,000
Nb-95	3.50+1 days	2.72+3 (a)	-(b)	-	-	-
Fe-59	4.46+1 days	7.34+4	3.00-3	-	-	-
Zr-95	6.40+1 days	1.65-1	-	-	-	-
Co-58	7.08+1 days	2.28+5	5.04+0	-	-	-
Zn-65	2.44+2 days	1.65+2	7.36+0	-	-	-
Mn-54	3.12+2 days	1.01+5	8.89+3	-	-	-
Fe-55	2.70+0 yrs	2.14+6	9.90+5	-	-	-
Co-60	5.27+0 yrs	1.54+6	1.04+6	3.15+0	-	-
Ni-63	1.00+2 yrs	2.00+5	1.97+5	1.00+5	1.96+2	-
Mo-93	3.00+3 yrs	6.20-1	6.20-1	6.06-1	4.92-1	6.16-2
C-14	5.73+3 yrs	2.49+2	2.49+2	2.46+2	2.21+2	7.42+1
Nb-94	2.00+4 yrs	7.96+0	7.96+0	7.93+0	7.64+0	5.35+0
Ni-59	7.50+4 yrs	1.32+3	1.32+3	1.32+3	1.31+3	1.21+3
Total Activity		4.29+6	2.23+6	1.01+5	1.73+3	1.29+3

Notes: (a) read as  $2.72 \times 10^3$   
(b) less than  $10^{-3}$

## INTACT DECOMMISSIONING BOUNDARIES

The selected intact decommissioning boundaries should contain radioactivity long enough to reduce within acceptable limits any risks to the public. Referring to Tables 3 and 4, it is apparent that most of the hazardous radionuclides would decay to insignificant levels on the order of 100 years. Thus, this duration was chosen as a reasonable zero leakage period required of decommissioning boundaries. In addition, within the spirit of the intact decommissioning alternative: (1) boundary preparation effort should be minimized - precluding entombment schemes; (2) maintenance during the passive phase (following the active phase which puts the plant in the intact decommissioned state) is ruled out; and, (3) surveillance of the facility during the passive phase is likewise precluded.

The various plant structures in the reference plants housing radioactive components were examined to determine which are likely to satisfy the above criteria. Different groups may qualify in different actual plants due to variations in local conditions. However, in this generic study, the Reactor Buildings were chosen for both the BWR and PWR plants. Table 1 shows that over 99% of the total activity at shutdown, including virtually all the long-lived isotopes important to population dose, are in the Containments. Hence, the viability of the intact decommissioning option from a radiological standpoint rests on whether the containments can meet the acceptance criteria.

Part of this study examined the long term integrity of structures exposed to the environment without maintenance, (6) and concluded, with a high degree of confidence, that the Containments would survive for at least 100 years in adequate condition to ensure retention of radioactivity. All penetrations into the Containments, such as for personnel and equipment access, and for HVAC, piping and electrical systems, would have to be sealed, typically, by cutting them on both sides of the Containment, filling them with concrete, and welding them shut with steel cover plates. In addition, since the Reactor Building in a Mark I BWR plant closely surrounds the primary containment and contains a significant amount of radioactivity itself, the concrete part would also be left standing. "Special" penetrations requiring sealing in a BWR would be the concrete shield plug over the drywell, the canals to the fuel storage pool and to the dryer and separator storage pool, and the steam tunnel. This active phase of intact decommissioning entails substantially less effort than is envisioned for entombment.

None of the other buildings in either reference plant containing radioactivity appears to be suitable for inclusion within the intact decommissioning boundaries through a combination of non-Class I seismic construction of parts of the buildings, and the use of metal siding (subject to serious corrosion over a long period of time) on the upper floors. However, other building in other nuclear plants may be suitable for inclusion within the boundaries depending on specific local conditions.

## RADIATION EXPOSURE

Potential public (individual and population) and occupational radiation exposures from intact decommissioning were determined and compared to those from other decommissioning options. Public doses compare favorably, while occupational doses should be substantially less than for immediate dismantlement.

There are many possible mechanisms through which radionuclides contained in the decommissioned facility may be released and transported through the environment, exposing people through various pathways. Exposure may result either through direct contact with the radioactive material (e.g., by inhalation, ingestion, or by direct exposure), or indirectly through a number of pathways involving contaminated vegetation and animals which have come into contact with or consumed radioactive materials. This study examined population and individual exposures during both the decommissioning period and the post-decommissioning period. In addition, the latter was divided into two parts: before and after the postulated loss of integrity of the decommissioning boundary.

Erosion and corrosion rates of structural materials (i.e., concrete, and carbon and stainless steel) exposed to different atmospheres and aqueous compositions were estimated based on a literature search covering a wide range of environmental conditions, and the most conservative values used in the dose assessment for release of radioactive materials from the plant to the environment. All the radiological evaluations performed indicate that intact decommissioning is a viable alternative to the dismantlement option - definitely in the short-term, and probably in the long-term as well.

Potential public and individual exposures for intact decommissioning are lower in the short-term (to 100 years) since: (1) reduced work scope implies reduced routine and potential accident exposures during the active decommissioning phase; (2) exposures during transportation of significantly smaller amounts of radioactive waste to a disposal site would be commensurately reduced; and, (3) during the passive phase of decommissioning, the sealed containment buildings would maintain integrity for at least 100 years, allowing most of the radioactivity to decay away.

Potential long-term (from 100 to 10,000 years) exposures should be no greater in intact than in other decommissioning methods. The principal uncertainty estimating long-term releases stems from the (reasonable) assumption that rubble will cover the reactor vessel internals and shield people from direct exposure to the few curies of Nb-94 in the stainless steel when the carbon steel vessel eventually corrodes away after several thousand years.

Occupational exposures incurred during the active phase of decommissioning are determined by first developing a detailed task list of activities, then associating with each task the number of people required and duration of time, and, finally, the radiation dose rate. Table 5 summarizes exposures for BWR and PWR reference plants for intact decommissioning, compared to those from immediate dismantlement, entombment (both with and without removal of reactor internals), and mothballing. (Data on the non-intact options were extracted from References 2 and 3). As expected from the greatly reduced scope of work, intact decommissioning results in greatly reduced occupational exposures compared to immediate dismantlement. The exposure decreases from 1,347 to 766 man-rem (adjusted for plant size) in the BWR, for a 581 man-rem "saving", and from 1,223 to 668 man-rem in the PWR, for a 555 man-rem saving. In addition, recent experience with actual major activities in substantial radiation fields (such as PWR steam generator repairs and BWR recirculation pipe replacements) indicates that the NUREG(2,3) estimates of exposures for non-intact occupational exposures may be understated, thereby increasing the advantage of intact decommissioning over the other methods.

Table 5: Occupational Exposures <sup>(a)</sup>

Decommissioning Options	Radiation Dose (man-rem)	
	BWR	PWR
Intact	766	668
Immediate Dismantlement	1,347	1,223
Entombment w react internals	1,151	900
w/o " "	1,229	1,000
Mothballing (preparation for passive safe storage)	274	416

Note: (a) from Reference 5.

---

#### REFERENCES

1. W. J. Manion, T. S. LaGuardia, "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives", AIF/NESP-009, November 1976.
2. H. D. Oak, G. M. Holter, W. E. Kennedy, Jr., G. J. Konzek, "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station", NUREG/CR-0672, prepared by Battelle Pacific Northwest Laboratory for the USNRC, Vols. 1&2, June 1980, Addendum 1, July 1983, Addendum 3, September 1984.
3. R. I. Smith, G. J. Konzek, W. E. Kennedy, Jr., "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station", NUREG/CR-0130, prepared by Battelle Pacific Northwest Laboratory for the USNRC, Vols. 1&2, June 1978, Addendum 1, August 1978, Addendum 2, July 1983, Addendum 3, September 1984.
4. L. Lewis, "To Decommission a Nuclear Power Reactor, 'Just Lock the Door'", report presented at the Health Physics Society Annual Meeting, Las Vegas, Nevada, 1982.
5. L. Lewis, S. Ostrow, "Intact Decommissioning: A Dose Assessment", International Conference on Decommissioning, Bethesda, Maryland, July 16-18, 1985.
6. S. Ostrow, O. Oztunali, T. Grant, J. Healey, C. McCaul, I. Romero, "Intact Decommissioning of Nuclear Power Plants: A Dose Assessment", AIF/NESP-033 (to be published).
7. J. M. Liederman, J. I. Saroudis, "The Inherent Advantages of Delayed Dismantling of Decommissioned Nuclear Stations", presented at NEA Workshop on Storage With Surveillance vs. Immediate Decommissioning for Nuclear Reactor Components and Buildings, Paris, France, October 22-24, 1984.
8. P. Denault, I. Kuperman, "Gentilly-1 Decontamination for Decommissioning Program", presented at the International Conference on Decommissioning, Bethesda, Maryland, July 16-18, 1985.
9. C. Roy Bennett, "The Extension of the SWS Period of CANDU Reactors With Particular Reference to Douglas Point", presented at the International Conference on Decommissioning, Bethesda, Maryland, July 16-18, 1985.

#### ACKNOWLEDGEMENTS

The authors wish to express their appreciation to the National Environmental Studies Project of the Atomic Industrial Forum (AIF/NESP) for its support of this work, and to the AIF/NESP Task Force members for their technical suggestions. Particular thanks goes to AIF/NESP Project Manager Dr. A. Scott Leiper for his deft guidance of the study to completion, and for his invaluable comments.

DECONTAMINATION OF THE CHEMICAL CRANE ROOM AND  
DECONTAMINATION AND DECOMMISSIONING OF THE  
EXTRACTION CHEMICAL ROOM AT THE  
WEST VALLEY DEMONSTRATION PROJECT

E. C. Phillips and M. P. Golden  
West Valley Nuclear Services Inc.,  
P. O. Box 191  
West Valley, New York, 14171-0191

ABSTRACT

The West Valley Nuclear Services Company Inc., a subsidiary of the Westinghouse Electric Corporation, has been contracted by the U.S. Department of Energy (DOE) to perform a high-level radioactive waste management demonstration project at West Valley, New York. Approximately 560,000 gallons of high level liquid nuclear waste, stored underground in a steel tank, will be solidified into a glass form. Certain of the existing facilities within the plant have been identified for use in support of the solidification program. Two of these are the Chemical Crane Room (CCR) and the Extraction Chemical Room (XCR). The preparation of these two rooms for their support roles is the subject of this paper.

This paper describes the decontamination of the Chemical Crane Room (CCR) and the Extraction Chemical Room (XCR) from radioactively contaminated conditions to essentially shirt sleeve environments. In both cases, subsequent use re-contaminated the rooms. Prior to decontamination, general exposure rates in the CCR were 50 to 100 mR/hr with hot spots as high as 2,000 mR/hr. Smearable levels on the floor were in the range of  $10^5$  to  $10^6$  dpm per 100/cm<sup>2</sup>. Respiratory protection was mandatory for entry.

In the Extraction Chemical Room (XCR) prior to decontamination and decommissioning (D/D), radiological surveys indicated a maximum radiation field of 5 mR/hr, due to sources internal to the room, and 20,000 dpm beta/100 cm<sup>2</sup> surface contamination. A radiation source external to the XCR caused a hot spot with a 9 mR/hr exposure rate inside the XCR.

The CCR, located at the north end of the Chemical Process Cell (CPC) is for the storage and servicing of two bridge cranes used in the CPC. Decontamination and exposure reduction in the CCR has been completed using vacuum cleaning, damp wipe down, and surface grinding followed by shielding and painting. The decontamination and decommissioning of the Extraction Chemical Room (XCR), located on the fifth floor elevation (160') of the reprocessing plant at the WVDP, has been completed. D/D operations included removal of piping, tanks, supports, and equipment to provide a clean work area of about 3,000 square feet and 17 feet high.

SUMMARY OF RESULTS

With both bridge cranes in the CPC locked out of the CCR, the decontamination of the CCR lowered the whole body exposure rate to 7 mR/hr, which was as low as reasonably achievable (ALARA).

The airborne radiation level was reduced so that respiratory protection was not required.

To satisfy the objective of preparing the XCR to support the work in the Extraction Cells, contaminated and unnecessary equipment was removed to the bare walls. At the completion of decontamination and decommissioning, the XCR provided a clean environment for the support work.

Because of the differences in D/D treatment received by the CCR and the XCR, they are discussed separately. The Chemical Crane Room is discussed first; the Extraction Chemical Room second. Overall conclusions which were drawn from these two projects are discussed at the end of the paper.

## DECONTAMINATION OF THE WEST VALLEY DEMONSTRATION PROJECT CHEMICAL CRANE ROOM

### Introduction

The decontamination program for the Chemical Crane Room (CCR) described in this paper was performed in preparation for the decontamination of the Chemical Process Cell (CPC). With a length of 27'6", a width of 22', and a height of 16', the CCR has direct access to the CPC and the Equipment Decontamination Room (EDR). Personnel access to the CCR is through an airlock, and equipment access to the CPC is through a doorway with a three feet thick shield door and to the EDR through a hatchway. See Figures 1, and 2. Two sets of bridge crane rails extend the full length of the CPC and CCR. The plant ventilation system provides 100 cfm of air flow through the CCR into the CPC.

The purpose of the CCR is to service and store the cranes and the PAR (Power and Remote manipulator). Repeated movement of the cranes in and out of the CCR contributed to the spread of contamination from the CPC to the CCR. Decontamination of the cranes in the CCR spread contamination to the floor, walls, and ceiling of this cell.

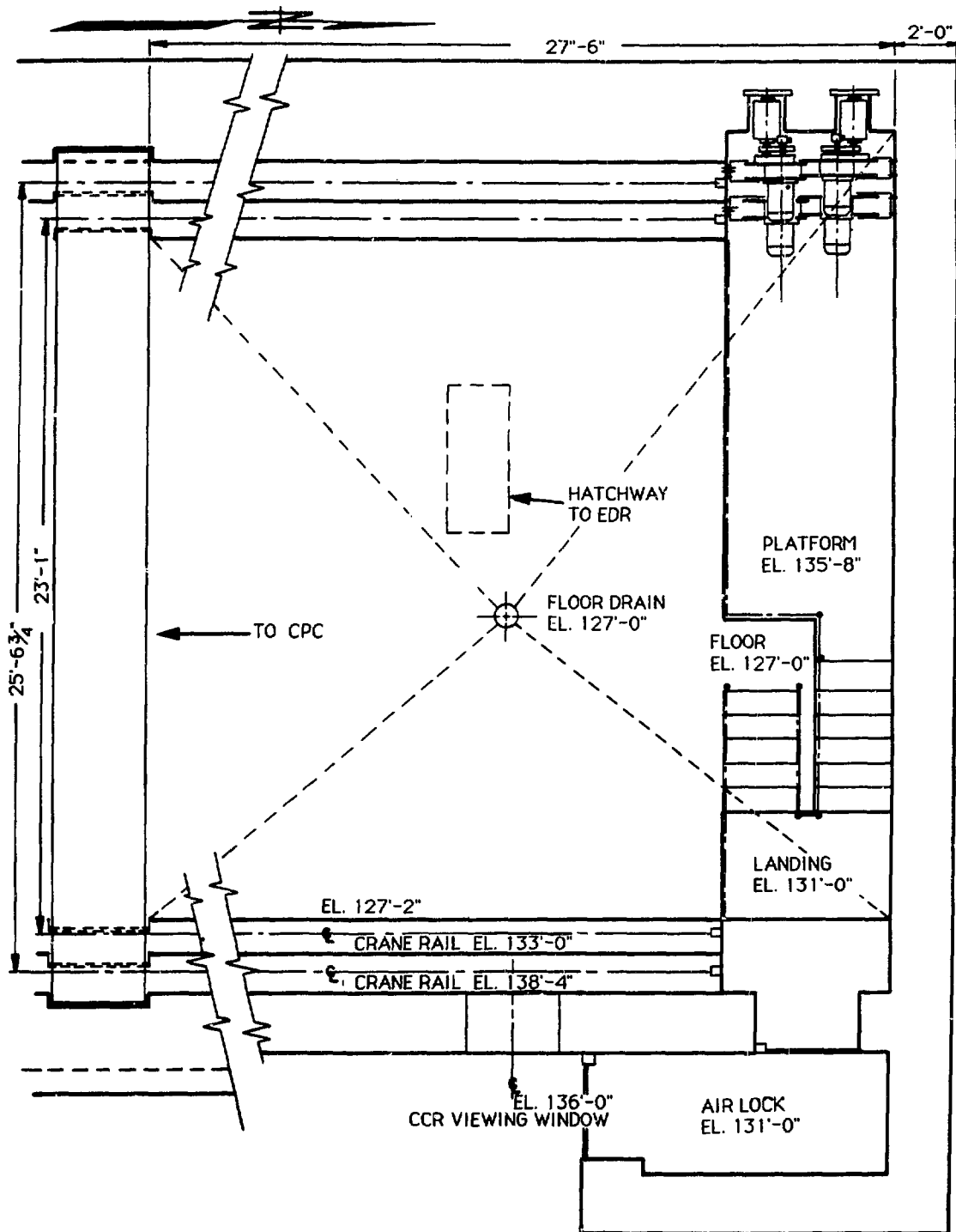
After the last dissolution of fuel in the CPC was completed (April 9, 1972), decontamination operations were conducted in the CCR sporadically for approximately three years. No further decontamination was performed between 1976 and 1982. During that time the CCR was used to train personnel in the use of respirators and anti-C clothing.

At the time of entry for decontamination, with both bridge cranes in the CPC locked out of the CCR, the radiation field in the CCR was approximately 50 mR/hr whole body exposure rate with some hot spots 20 times the whole body exposure rate. Airborne contamination was  $5.10 \times 10^{-10}$   $\mu\text{Ci/cc}$  alpha activity and  $2.45 \times 10^{-9}$   $\mu\text{Ci/cc}$  beta activity.

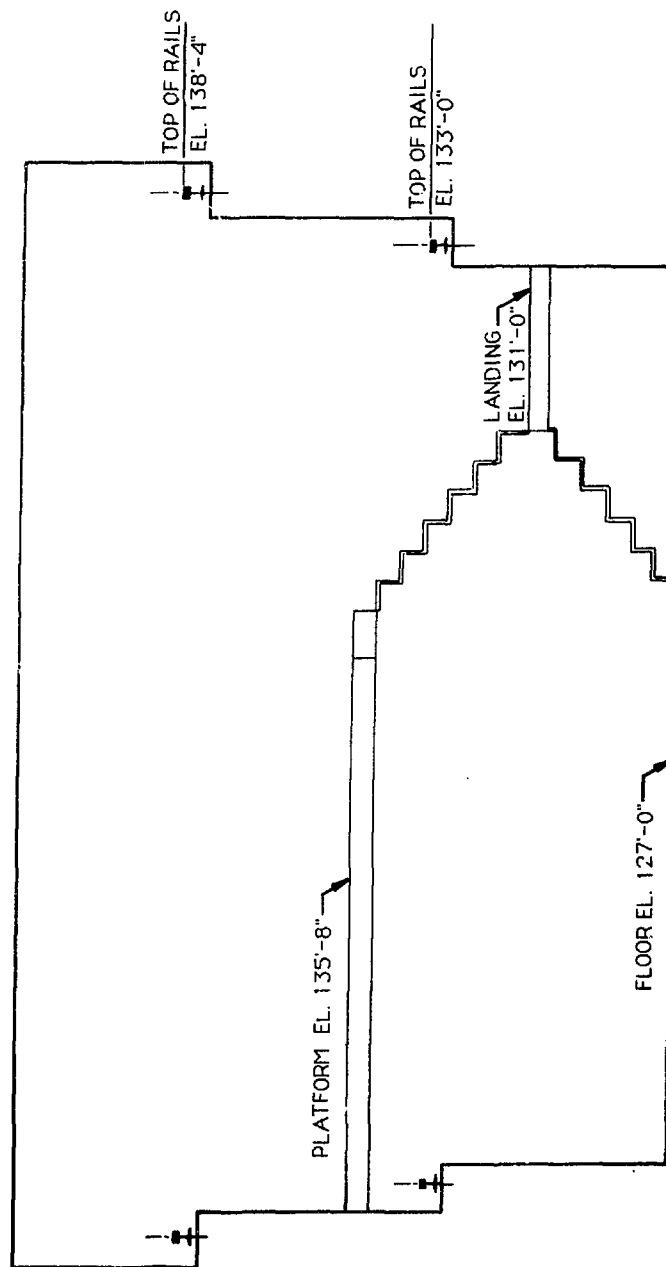
### Objectives for Decontaminating the CCR

The objectives for decontaminating the CCR were as follows:

1. To reduce the whole body exposure rate from 50 mR/hr to 5 mR/hr or as low as reasonably achievable (ALARA).
2. To make the CCR available for refurbishing and servicing the CPC cranes and PAR.



**FIGURE 1**  
**CCR PLAN VIEW**



**FIGURE 2**  
**CCR NORTH ELEVATION**

## Decontamination Method for the CCR

The approach to decontamination was first to establish a containment that attached to the CCR airlock, then to decontaminate the airlock, and finally to decontaminate the CCR itself. Prior to any entry into the CCR airlock or the CCR proper, both bridge cranes were moved into the CPC and locked out of the CCR with the shield door closed.

A two compartment containment enclosure fabricated from reinforced polyvinyl chloride fabric was used to attach to the airlock. The plant ventilation system provided a constant 100 cfm flow of clean air through the containment tent into the CCR to keep the contamination enclosed.

The surface contamination in the CCR airlock was removed by vacuum cleaning followed by damp wipe down. In the CCR, the rail shelves were cleaned by vacuum cleaning. Surfaces of the rails were cleaned by stripping paint and degreasing the tops of the rails with solvent.

Fixed contamination in the floor and walls of the CCR was removed by a surface grinder with an attached vacuum cleaner to capture the contaminated dust generated. A fixture, shown in Figure 3, was used to hold the grinder against the walls. The fixture has a remotely controlled drive unit for propelling it along the crane rails, a vertical positioning adjustment, and a system for adjusting the wall contact pressure. Residual loose contamination on the floor and walls was fixed with epoxy paint.

Grinding and acid descaler were used to remove fixed contamination from the crane rail surfaces. The use of these methods produced no measurable change in the 100 to 200 mR/hr gamma exposure rate and the 200 to 2000 mR/hr beta exposure rate measured before decontamination. Because the rails would be contaminated again as soon as they came in contact with the cranes, additional decontamination was judged to be labor and exposure intensive. Lead blankets 0.25 inch thick were used to shield the rails lowering the measurable gamma exposure rate to 30 to 50 mR/hr.

## Results of the CCR Decontamination

Decontamination of the CCR resulted in a reduction of whole body exposure rates in the CCR and a reduction in the surface contamination levels in both the airlock and the CCR.

In the airlock, prior to decontamination, the whole body exposure rate was measured at 5-10 mR/hr while beta smearable was 50,000 dpm/100 cm<sup>2</sup>. After decontamination, the smearable beta activity was 200 to 1000 dpm/100 cm<sup>2</sup>. With a flow of equipment, people and material through the airlock during decontamination of the CCR, the average beta activity on the airlock floor increased to approximately 16,000 dpm/100 cm<sup>2</sup>.

At the start of D/D operations in the CCR, the airborne contamination level was  $5.10 \times 10^{-10}$   $\mu$ Ci/cc alpha activity and  $2.45 \times 10^{-9}$   $\mu$ Ci/cc beta activity. At the completion of D/D activities, the airborne contamination was less than the MPC (maximum permissible concentration) of  $2 \times 10^{-12}$   $\mu$ Ci/cc for alpha activity and less than  $1 \times 10^{-9}$   $\mu$ Ci/cc beta activity.

Prior to completion of CCR floor grinding, the whole body exposure rate had been reduced from 50 to 20 mR/hr, the smearable beta activity by a factor of 6 from 374,000 dpm/100 cm<sup>2</sup> and the smearable alpha activity by a factor of 12 from 7400 dpm/100 cm<sup>2</sup>.

Efforts to completely decontaminate the rails were ineffective; therefore, lead shielding was placed over the rails which reduced the measurable gamma exposure rate of the rails from 100 to 200 mR/hr to 30 to 50 mR/hr. At the completion of decontamination in the CCR, the whole body exposure rate was 7 mR/hr. This represents a factor of seven reduction.

A comparison of the radiological conditions in CCR before and after decontamination is shown in Table 1.

All waste was removed from the CCR in 55-gallon plastic lined steel drums which meet DOT Specification 17C. A total of forty-nine 55-gallon drums of waste, none of which were classified as TRU (transuranic), was removed from the CCR and stored.

The total overall exposure logged for the decontamination operation was 2228 mR with a total exposure time of 281 man-hours which yields an average overall exposure rate of 8 mR/hr.

#### Discussion

The CCR was successfully decontaminated to a shirt sleeve environment safe for entry without respiratory protection. Decontamination of the floor by grinding met the objective of reducing the whole body exposure rate in the CCR to a level consistent with the ALARA concept. Further floor grinding and subsequent reduction in the whole body exposure rate would have been labor and exposure intensive.

After the decontamination of the floor, the walls became the dominant radiation sources in the CCR. However, after decontamination of the walls, other radiation sources (the rails) became dominant and it was impossible to determine whether further decontamination of the walls would reduce the whole body exposure rate in the CCR. Consequently, additional wall grinding would have been inconsistent with the ALARA concept.

Since decontamination of the crane rails was ineffective, the rails were shielded with lead blankets.

An overall radiation level reduction factor of 7 was achieved with the reduction of the whole body exposure rate from 50 to 7 mR/hr. After completion of the decontamination of the CCR, the cranes were decontaminated and serviced in the CCR, re-contaminating the room.

### DECONTAMINATION AND DECOMMISSIONING OF THE EXTRACTION CHEMICAL ROOM

#### Introduction

During NFS plant operation, the XCR was used to supply chemicals to the extraction cells. Consequently, it was large enough, 81 feet by 41 feet by 17 feet 3 inches minimum height, to contain 27 tanks, 35 drums and the auxiliary equipment necessary for chemical handling. Figure 4 shows a photograph of a scale model of the XCR complete with equipment. Over each extraction cell hatch, there is a hatch in the roof of the XCR of sufficient size to pass the largest hatch cover for the associated extraction cell. See Figure 5 for locations of Extraction Cells in XCR.

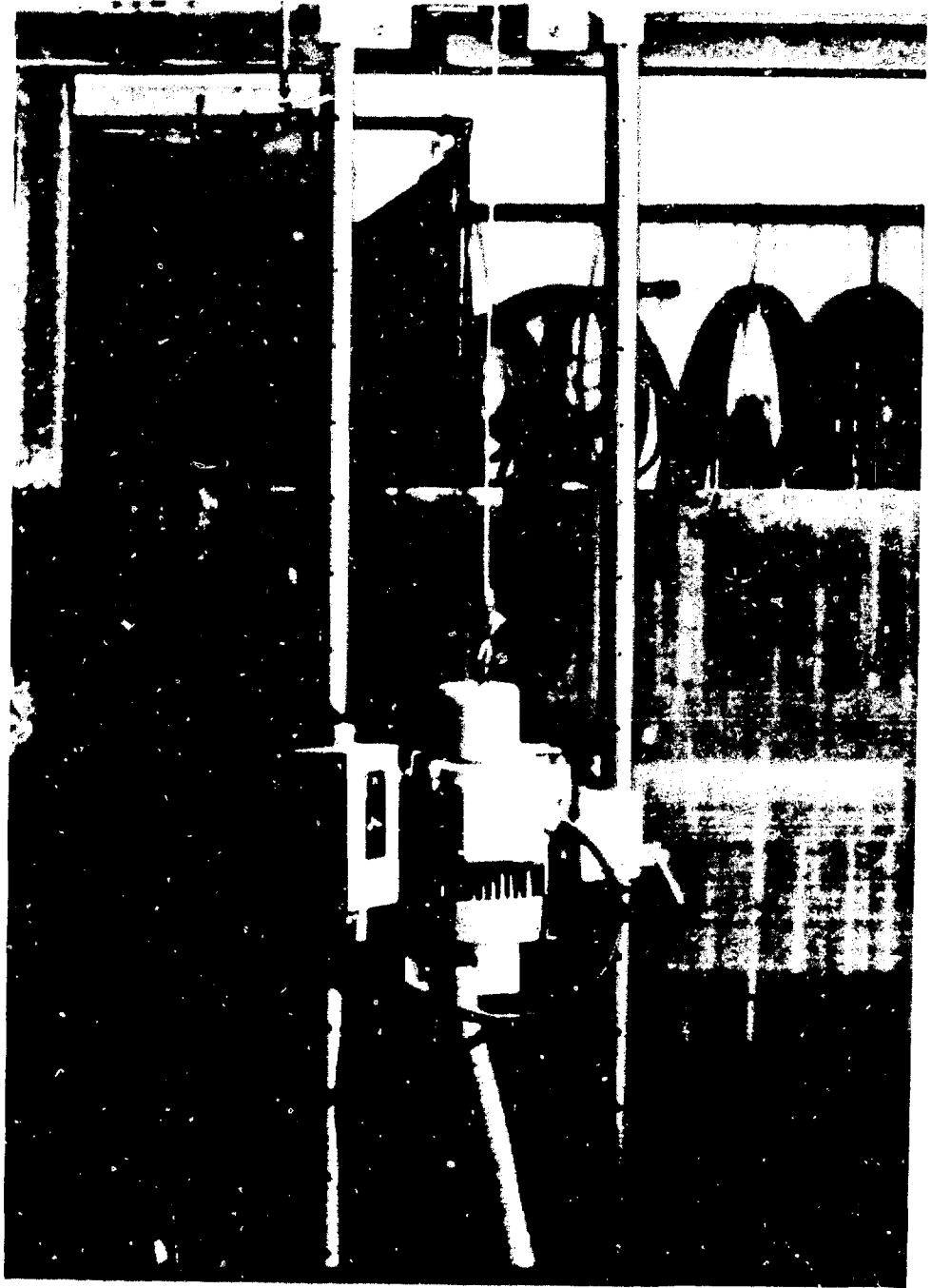
Table 1

Radiological Conditions in the Chemical Crane Room  
Before and After Decontamination

	Airlock		CCR		East Wall		West Wall		Rails	
	Before	After	Before	After	Before	After	Before	After	Before	After
Exposure Rate mR/hr	General 5-10	General 5-10	General 50	General 7	$\beta$ Y Contact 300-1000 Y Contact 30-100	$\beta$ Y Contact 200-500 Y Contact 30	$\beta$ Y Contact 300-2000 Y Contact 30-200	$\beta$ Y Contact 200-500 Y Contact 30	$\beta$ Y 200-2000 Y 100-200	$\beta$ Y 200-2000 Y 100-200 Y* 30-50
Airborne Alpha $\mu$ Ci/ml	NA**	NA	$5.10 \times 10^{-10}$	$< 2 \times 10^{-12}$	NA	NA	NA	NA	NA	NA
Airborne Beta $\mu$ Ci/ml	NA	NA	$2.45 \times 10^{-9}$	$< 1 \times 10^{-9}$	NA	NA	NA	NA	NA	NA
Smearable Alpha dpm/100 cm <sup>2</sup>	NA	NA	7,400	600 (Avg)	NA	NA	NA	NA	NA	NA
Smearable Beta dpm/100 cm <sup>2</sup>	50,000	200 - 1000	374,000	62,000 (Avg)	NA	NA	NA	NA	NA	NA

\* After the addition of .25 inch of lead blanket.

\*\* Not Available



**FIGURE 3**  
**WALL GRINDER EQUIPMENT IN TEST CONFIGURATION**



**FIGURE 4**  
**XCR MODEL FROM THE SOUTH**

In addition to the chemical handling equipment, the XCR also contained platforms and stairs, which formerly provided access to the chemical equipment, plus a small hoist mounted on a monorail. A larger hoist, 1 ton capacity on a monorail, provided access to a 1-ton jib crane on a roof below the double doors shown in Figure 5, which in turn provided access to a paved driveway at ground level. These cranes were the means of equipment removal when the XCR was dismantled. Prior to decontamination, the maximum radiation field within the XCR due to sources internal to the XCR was 5 mR/hr and the surface contamination was 20,000 dpm beta/100 cm<sup>2</sup>. A hot spot with a 9 mR/hr exposure rate in the XCR was caused by a radiation source external to the XCR.

## Objectives

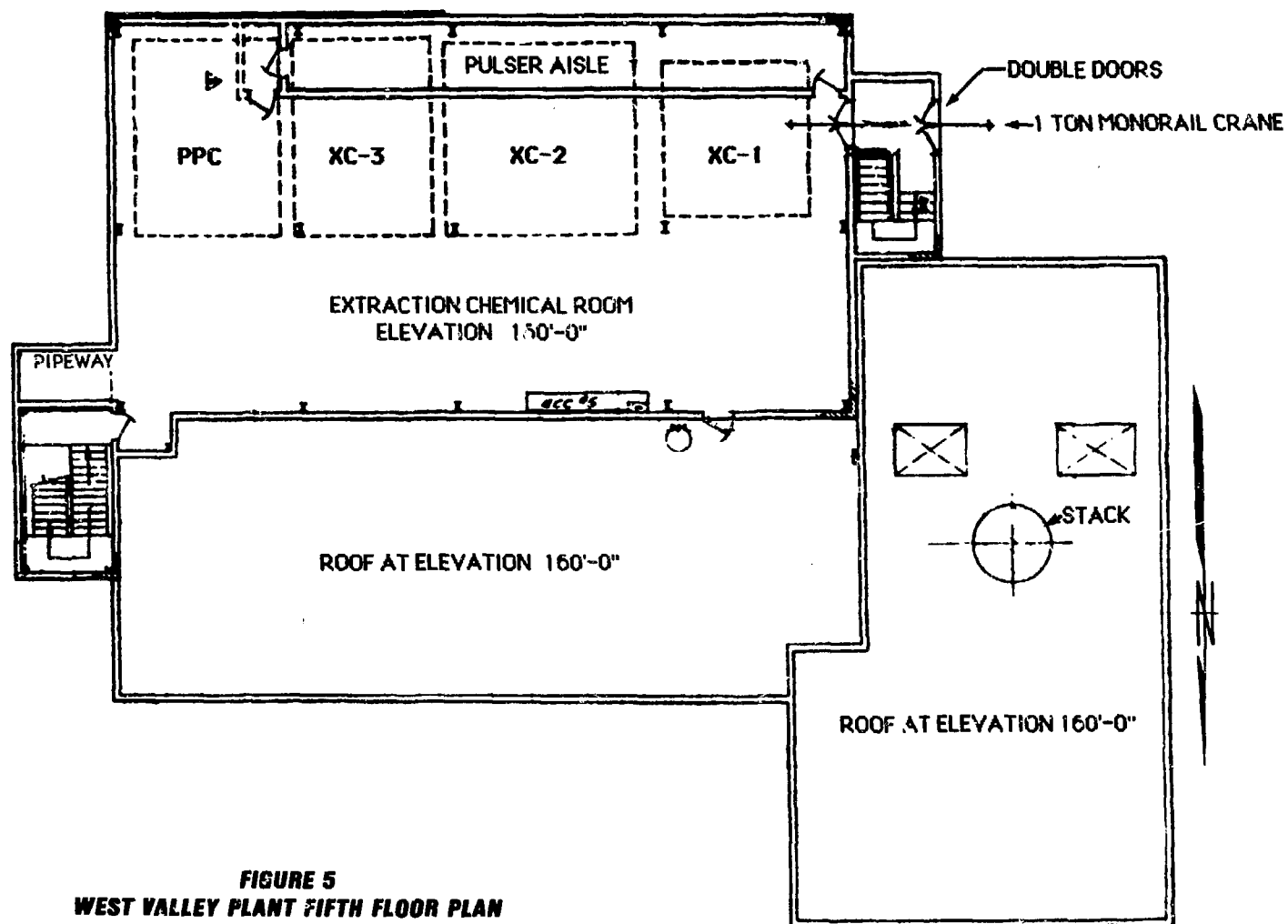
Decontamination of the XCR was performed to satisfy the objective of preparing a work area to support decontamination work in the extraction cells.

## Decontamination Method of the XCR

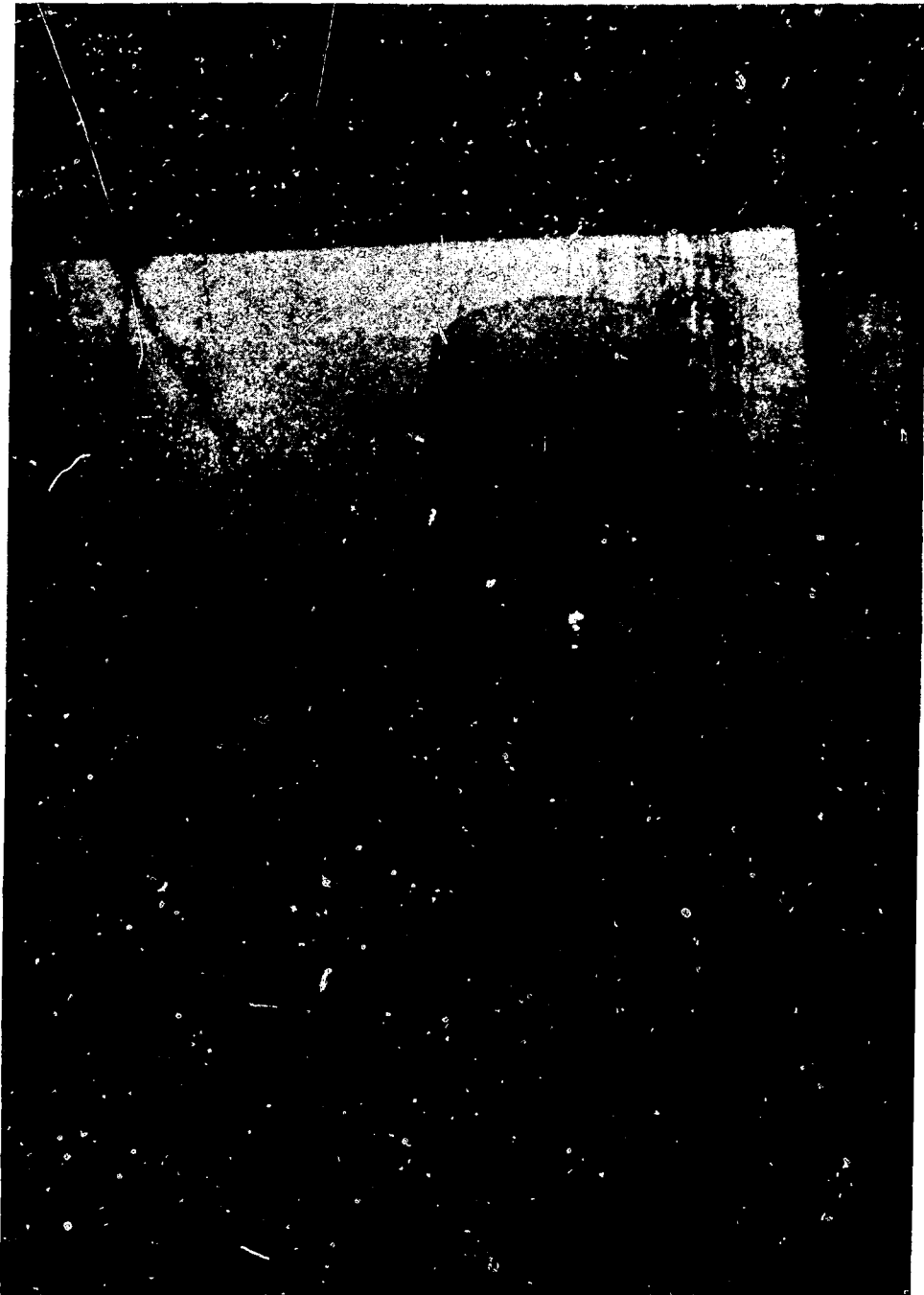
To perform its intended function, it was necessary to remove everything from the XCR to the bare walls. The piping was removed first. Each pipe run was identified by comparison with plant drawings, and a determination was made with regard to its content and contamination level. Each pipe was radiologically surveyed and "telitaled" for liquid. A radiochemical analysis was made of any liquid contents of the pipes. Contaminated pipes were filled with a polyurethane foam to fix the contamination before the pipes were cut out of the system. The cut pipe ends or openings were plugged, sealed, bagged, and taped before placing the pieces in 55-gallon drums for disposal as low-level radioactive waste. Instrumentation, electrical components, supports, etc., were then removed by similar methods. The drums were removed through the west doorway in the XCR by means of the existing 1-ton monorail hoist and a roof-mounted jib crane. The XCR tanks, when determined to be free of external contamination, were then permanently plugged, covered, sealed, and removed through hatches in the XCR roof by means of a ground-level mobile crane to temporary storage. See Figures 6 and 7.

At the time that decontamination of the XCR was initiated, the floor on the north half of XCR was enclosed within an 8 inch high curb to contain spillage from the chemical tanks and equipment. Prior repairs to this floor had fixed contamination in or below the existing surface. It was decided to fix any residual contamination in the floor by pouring concrete to the level of the curb. To prepare the floor for resurfacing, all protruding pipe stubs, electrical conduit, tank feet, pump bases, etc. were cut off as far below the curb level as possible. All drains but one were capped and wire mesh reinforcing was installed. An epoxy based bonding agent was applied to the existing floor, then mesh was laid; immediately followed by pouring fresh concrete. The concrete was continuously poured, and then finished to as smooth a troweled level surface as possible. After curing, the concrete was sealed with a high build (.008 inch) epoxy sealer. This was followed by a clear polyurethane coating for wear and durability. Figure 8 shows the resurfaced floor.

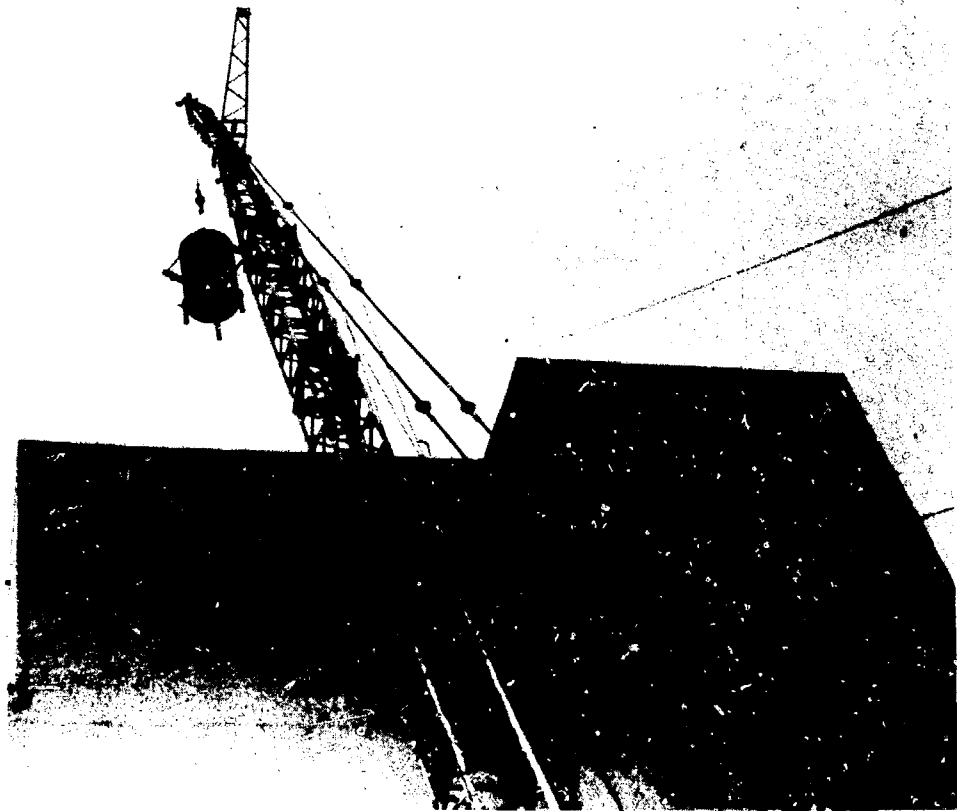
The remaining floor surface in the XCR had contamination fixed in paint. These surfaces were sealed with the high epoxy sealer and coated with the clear polyurethane coating.



**FIGURE 5**  
**WEST VALLEY PLANT FIFTH FLOOR PLAN**  
**Elevation (160'-0")**  
**Showing The Extraction Chemical Room**



**FIGURE 6**  
**REMOVAL OF A VESSEL THROUGH XCR ROOF HATCH**



**FIGURE 7**  
**VESSEL BEING LOWERED TO GROUND LEVEL**



**FIGURE 8**  
**NEW RAISED CONCRETE FLOOR IN XCR**

## Results of the XCR Decontamination

At the completion of D/D operations, the XCR was still considered a radiation area because of hot spots caused by an external radiation source which exhibited an exposure rate of 9 mR/hr, which was not affected by D/D operations inside the XCR. The average exposure rate for the XCR was 1 mR/hr while smearable and airborne contamination levels were less than the lower limits set for contaminated areas.

The D/D operations in XCR took place over a period of 170 calendar days with an estimated expenditure of 6700 man-hours. The total radiological exposure was approximately 3500 mR for an average overall whole body exposure rate of approximately 0.5 mR/hr.

## Discussion

The success of the D/D Operations in XCR was demonstrated by the XCR being a radiation area free of smearable contamination at the completion of operations. The difficulties encountered during the work were primarily a consequence of the elevation of XCR, 60 feet above ground level. This elevation problem, in addition to the location of building walls, made it difficult to remove vessels and heavy equipment from XCR and to pour concrete in the XCR.

An estimated 2,800 cubic feet of material were removed from XCR and disposed of as low-level waste. The activity of liquid from tellitaling was less than  $5 \times 10^{-3}$   $\mu\text{Ci/ml}$  and the liquid was drained to the plant interceptors. Radiological surveys before D/D indicated a radiation field, due to sources internal to the room, of 5 mR/hr maximum, and surface contamination as high as 20,000 dpm beta activity/100  $\text{cm}^2$ . After D/D operations, the XCR was a radiation area with surface contamination below 50 dpm alpha/100  $\text{cm}^2$ , and below 500 dpm beta/100  $\text{cm}^2$ .

## CONCLUSIONS

The XCR decontamination and decommissioning accomplished all the objectives. It was well planned, well executed and successful. The radiation levels in the decontaminated room were 1 mR/hr and airborne contamination was below MPC.

In the CCR, all of the decontamination methods used were effective in producing the desired whole body exposure level or ALARA with the exception of those used on the crane rails. Operator time in-cell and total exposure could have been reduced if the capability had existed to locate and isolate close radiation sources more accurately. The success of the operation was demonstrated by lowering the whole body exposure rate to 7 mR/hr and the airborne radiation level below MPC levels. The successful decontamination efforts in these two rooms will make future decontamination projects at West Valley easier to accomplish.

DECONTAMINATION AND DECOMMISSIONING OF CONTAMINATED  
SURPLUS FACILITIES IN THE 200 AREAS OF THE HANFORD SITE\*

D. R. Speer and W. F. Heine  
Rockwell Hanford Operations  
P. O. Box 800  
Richland, Washington 99352

## ABSTRACT

The Surplus Facilities Management Program (SFMP) of Rockwell Hanford Operations (Rockwell) is currently engaged in decontamination and decommissioning (D&D) of contaminated surplus facilities in and around the 200 Areas of the Hanford Site. These contaminated surplus facilities range in size and complexity from small wood-frame buildings to an 83,000 ft<sup>2</sup> reprocessing facility with monolithic concrete shielding walls and extensive residual radioactive contamination. The current inventory of facilities includes 3 major processing facilities, 8 large support buildings, 14 small support buildings, and 30 tanks, vaults, basins, and stacks.

Completed activities include the decontamination or dismantlement of 9 facilities. Part of this involved the intact transport of four 50,000-gal tanks from a facility to a burial ground with subsequent filling and burial of the tanks.

Work underway includes the decontamination, dismantlement, and entombment of a fuel reprocessing pilot plant and the decontamination of a plutonium processing facility with a goal of returning the processing facility to beneficial use.

Thus far, the decontamination efforts have resulted in a significant reduction in hazards associated with safe storage of the 9 completed facilities as well as cost avoidances associated with returning buildings and equipment to beneficial use.

## INTRODUCTION

The 200 Areas of the Hanford Site in Washington State (figure 1) have been used more than 40 yrs for nuclear fuels reprocessing and radioactive waste management. The development of new techniques for reprocessing and waste management has naturally resulted in the obsolescence of many older facilities. These facilities have been declared surplus to current and future Department of Energy (DOE) needs. The facilities are now being decontaminated and decommissioned under the Surplus Facilities Management Program (SFMP).

The inventory of facilities have been grouped into projects that are scheduled for completion according to their relative potential radiological hazard and relative costs for safe storage prior to decontamination and decommissioning (D&D). This grouping is shown in table 1 and the corresponding schedule is shown in figure 2.

\*Work sponsored by DOE under contract DE-AC06-77RL01030

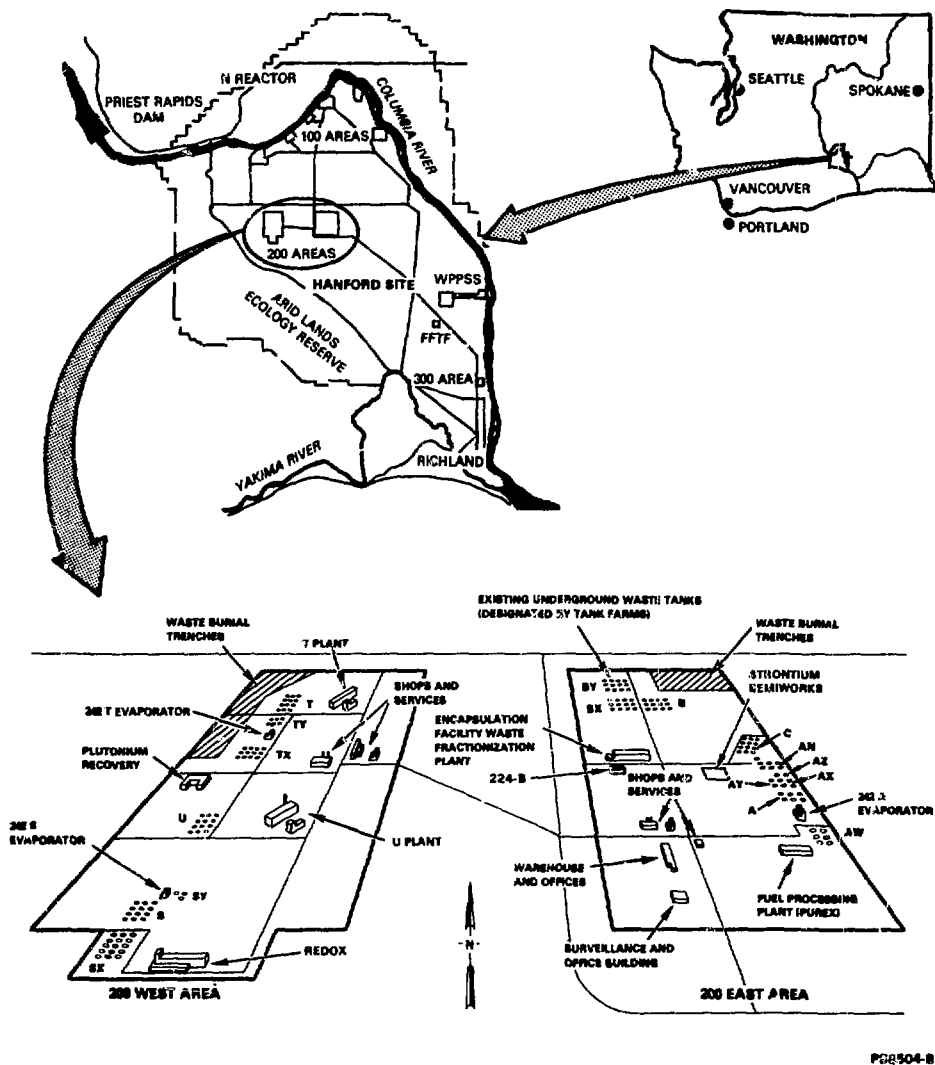


Figure 1. The 200 Areas of the Hanford Site.

Table 1. Project Priorities			
Project	Overall priority	Relative economic priority	Relative radiological priority
Buried Tanks	High	High	High
233-S Facilities	High	Medium	High
Strontium Semiworks	High	High	High
224-B Facility	High	High	Medium
REDOX	High	Medium	High
Weirs and Control Structures	Medium	High	Medium
Vaults	Medium	High	Low
U-Plant	Medium	Medium	High
205-A Facility	Medium	Medium	Low
276-S Facilities	Medium	Medium	Low
200 East Area Tank Farm	Low	Low	Medium
Buildings			
212 Buildings	Low	Low	Low
207-S Retention Basin	Low	Low	Low
200 West Area Tank Farm	Low	Low	Low
Buildings			
232-Z Facility	Low	Low	Medium

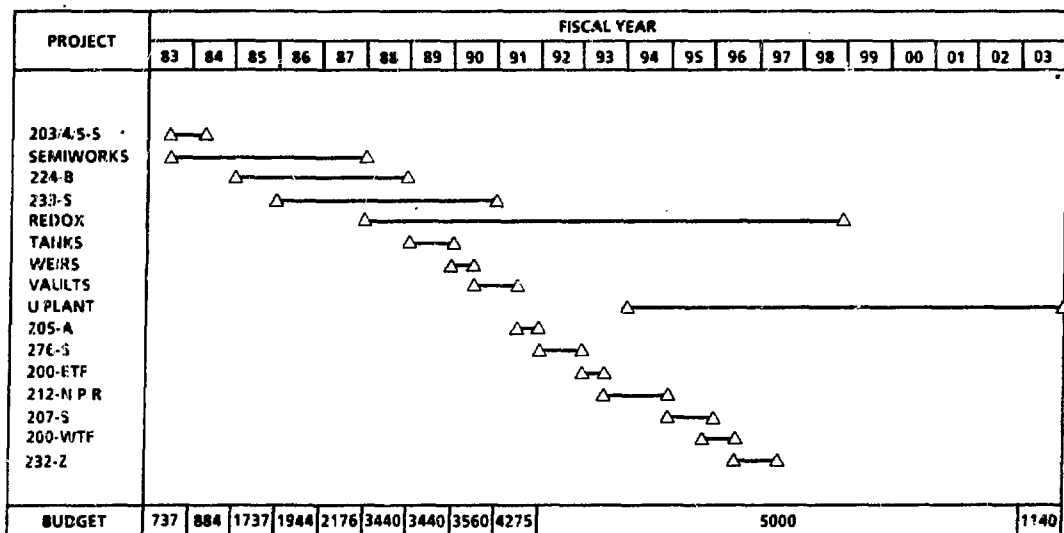


Figure 2. Long Range Schedule for Decontamination and Decommissioning of Contaminated Surplus Facilities in the 200 Areas of Hanford.

## EXPERIENCES

### 203-S, 204-S, 205-S Project

The 203-S, 204-S, 205-S complex (photo 1) was designed and operated as a uranyl nitrate hexahydrate decontamination and lag storage facility in support of fuel reprocessing at the Redox Plant. The 204-S portion of the complex was later converted to a receiving station for liquid radioactive waste from the 100 and 300 Areas.

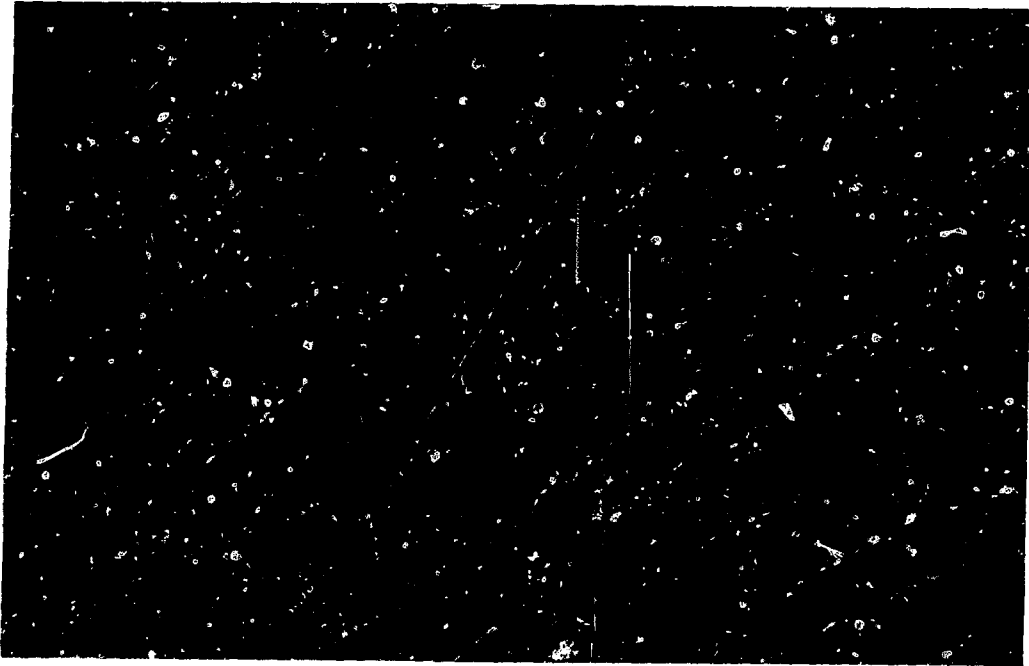


Photo 1. The 203-S, 204-S, 205-S Area before decommissioning.

The D&D of the complex was begun in the spring of 1983 to alleviate potential and actual environmental hazards associated with the facility. The method of decommissioning was the dismantlement of all aboveground structures and the in-place burial of all remaining structures below 2 ft or more of clean soil. The area was seeded with Russian wheat grass and mulched with straw and is now free from access restrictions to the surface (photo 2).

A significant activity undertaken during the D&D of these facilities involved cleaning and removing the four 50,000-gal tanks located at the site. The tanks were welded stainless steel vessels 25 ft in diameter, 17 ft in height and made from 1/4-in. and 1/2-in. plates. One of the tanks contained 10 to 12 tons of heavy sludge that produced a radiation dose rate of 3 to 5 R/h at the edge of the tank. This tank was cleaned out to a remaining sludge level of approximately 1 ton by using a commercial tank sluicer and large quantities of hot water (180-200° F). After cleaning, the 50,000-gal tanks were lifted intact with a crane, specially fabricated six-point spreader bar, and lifting clamps, and placed on a 100-ton low-boy truck for transport to a burial site. The tanks were filled with sand, capped with concrete grout, and covered with soil.

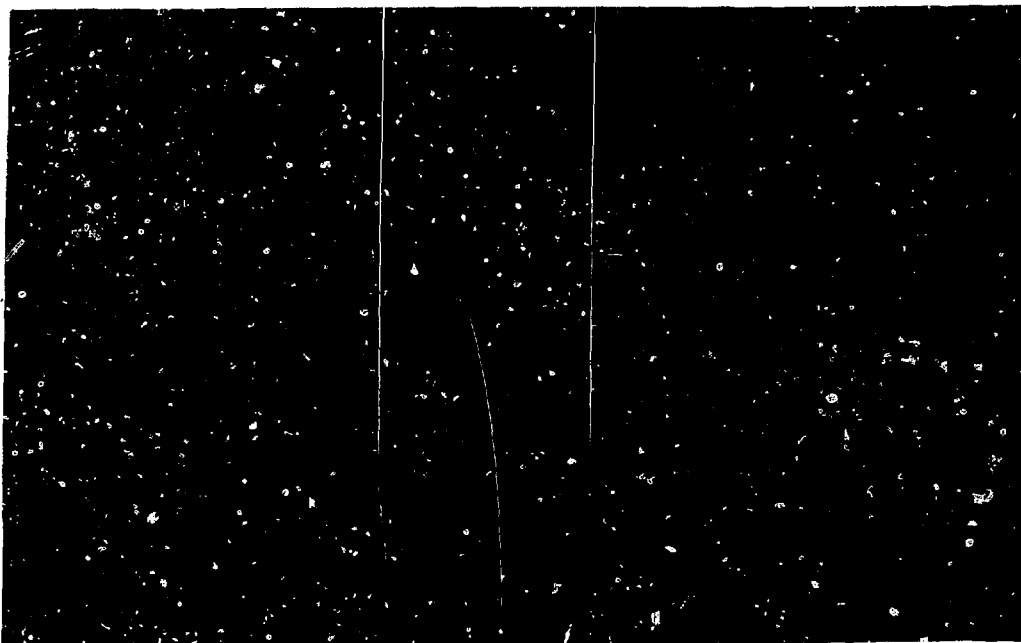


Photo 2. 203-S, 204-S, 205-S Area after decommissioning.

The 203-S, 204-S, 205-S Project was completed in 9 mos at a cost of \$1.025 million. A total of 9 Ci of radioisotopes, including 1.8 Ci of strontium-90, 3.8 Ci of ruthenium-106 and 1.9 Ci of cesium-137, was removed from the site and buried locally. The 9 Ci were contained in approximately 55,000 ft<sup>3</sup> of waste. An estimated 0.2 Ci of mixed fission products remain at the site in buried piping, concrete, and soil.

#### Strontium Semiworks Project

The Strontium Semiworks Project, figure 3, is located in the Hanford 200 East Area. This facility was used to pilot the Redox and PUREX fuel separations processes and to pilot the process for removing strontium from high level radioactive liquid waste. The facility was shut down in 1967 and was held in safe storage until 1984 when decommissioning was started.

The Strontium Semiworks complex consists of 11 structures; the main structure is the 201-C Process Building. The 201-C Process Building is made up of three concrete cells in which radionuclides were processed and two other concrete cells for storage and loadout of the product. These cells extend to 25 ft below ground, as noted in figure 4. Service galleries, a maintenance shop, and treatment facilities for process water and air are included in this building. Other structures in the complex include a solvent handling building, ventilation system, and underground storage tanks. The estimated radiological inventory for the complex includes 10 Ci of plutonium and 9600 Ci of strontium.

Planning for decommissioning of the complex began in 1983 with the preparation of an evaluation of the various decommissioning alternatives for the 201-C Process Building. The alternatives included:

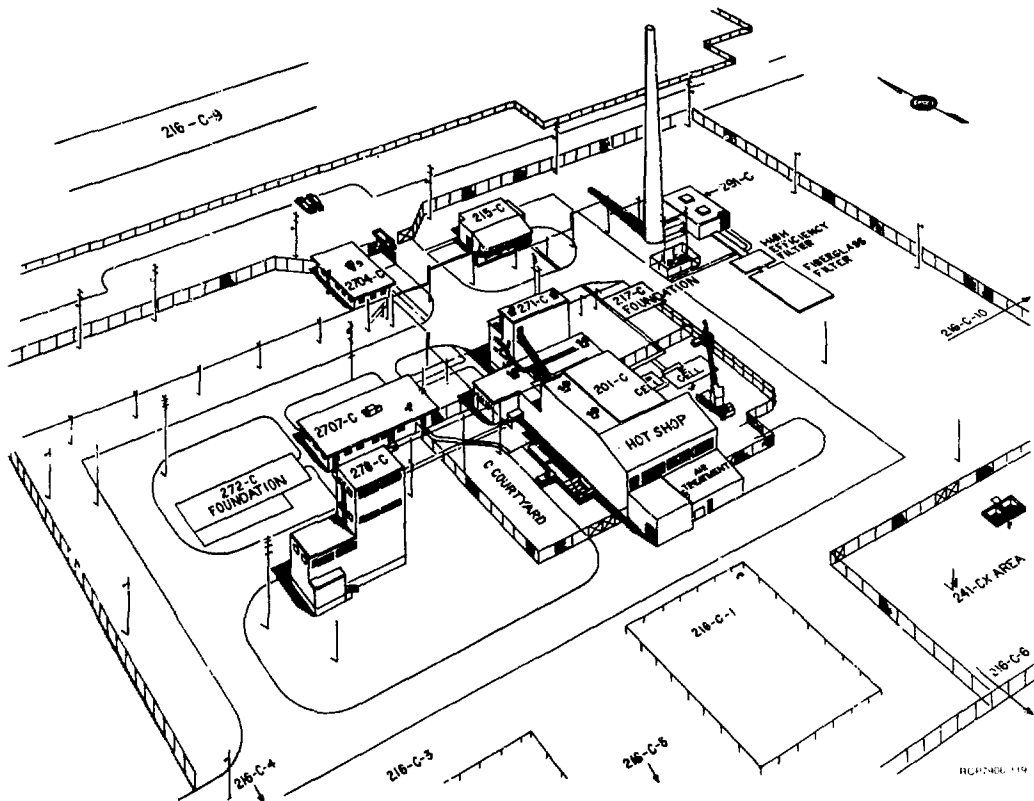


Figure 3. Strontium Semiworks Complex.

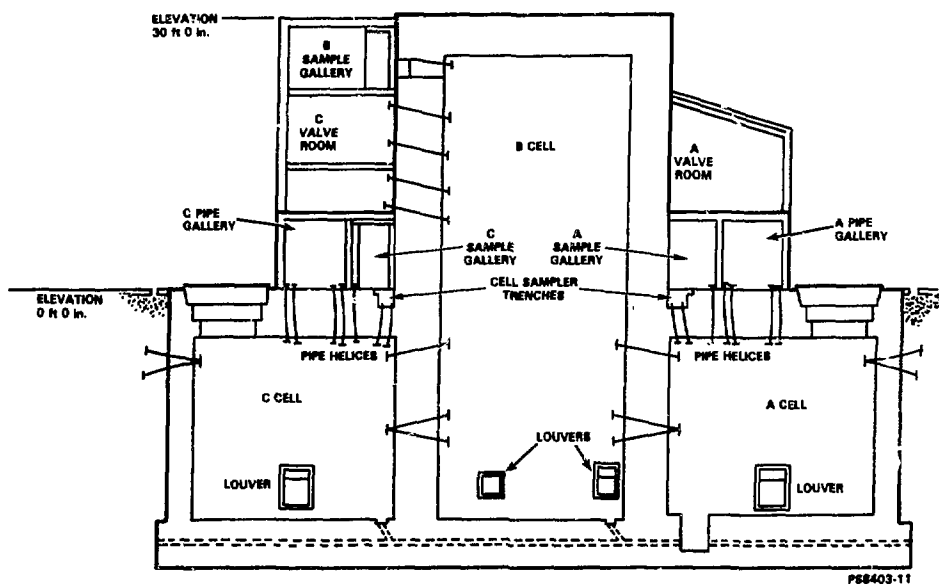


Figure 4. Cross Section View of 201-C Process Building.

- Entombment
- Partial dismantlement followed by entombment
- Razing to grade level
- Total dismantlement and removal
- No action (continued safe storage).

The estimated costs and occupational radiation doses associated with each alternative are listed in table 2.

Table 2. Estimated Doses and Costs for Each of the Alternatives

Alternatives	Occupational Dose (Man-Rem)	Costs (Millions 1984\$)
1) Entombment	2.7-4.1	3.5
2) Dismantlement/Entombment	11-21	4.9
3) Razing	25-58	8.0
4) Total Dismantlement	135-400	18.0
5) No Action (1)	5	10

(1) Continued safe storage; routine annual exposure and costs for the first 100 years. Non-routine exposure and costs not included. These costs would continue indefinitely.

Alternative 2 was selected as the preferred option. The service galleries and B-Cell will be demolished to 10 ft above grade, the remaining service galleries and cells filled with concrete, and the site covered with an engineered earthen barrier as shown in photo 3.

An Environmental Assessment (1), which evaluated the current and future impacts of the project, was prepared and submitted according to the requirements of the National Environmental Policy Act as defined by the Council on Environmental Quality regulation 40 CFR 1501.2. The Environmental Assessment indicated that the projected worst-case hypothetical accident during the decommissioning action could result in an offsite 50 yr whole body dose commitment of 0.18 rem to the maximum individual with a dose to the critical organ (bone) of 0.32 rem. The projected worst-case hypothetical whole body dose to an intruder after entombment of the site was calculated to be 0.55 rem with a critical organ (lung) dose of 7.2 rem. The probability of severe accidents occurring during decommissioning is low. The hands-on decommissioning of the site began in April of 1984 with the decontamination of ancillary facilities. During the period of April 1984 to September 1984, the 2707-C Building (figure 3), the 276-C Building, and the 215-C Building were decontaminated and radiologically released for unrestricted use. In addition, the 271-C Building and the adjacent one-story control room were completely dismantled to the foundation slab.

Work during fiscal year 1985 included the decommissioning of the large process cell (B-cell) and the service galleries of the 201-C Building. The interior of the service galleries above an elevation of 10 ft were dismantled and decontaminated in preparation for their structure dismantlement. The interior of B-Cell was initially decontaminated and 700 yd<sup>3</sup> of concrete was

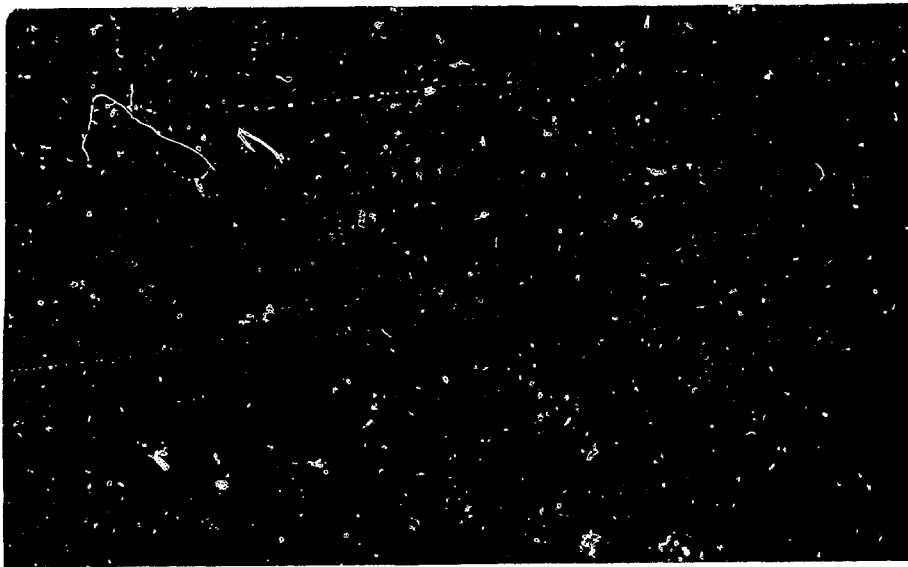


Photo 3. Artists concept of the engineered earthen barrier over the Strontium Semiworks site.

poured into place to entomb the lower 23 ft of the cell. Remaining work to be accomplished includes:

- Dismantlement of all remaining piping and equipment in B-Cell
- Demolition of the B-Cell shielding walls and ceiling
- Entombment of the four remaining cells and the lower service galleries
- Demolition of the 200 ft tall exhaust stack
- Entombment of the three underground waste tanks and the deep bed fiber glass exhaust filter
- Construction of an engineered earthen barrier over the site
- Publication of a project closeout report.

The completion of all D&D activities is scheduled for the end of fiscal year 1987 with the publication of the project closeout report.

#### 224-B Building Project

The 224-B Building is a three-floor structure with overall dimensions of 69 ft long by 197 ft wide by 60 ft tall. The building was used in the 1940s to process plutonium solutions.

The D&D plan for this building is to remove all contaminated surplus equipment and decontaminate all of the building surfaces. This will result in the availability of a structurally-sound, 30,000 ft<sup>2</sup> building for reuse.

All predecommissioning documentation (project planning, safety analysis, and environmental compliance documentation) for the project was completed in fiscal year 1985. The building is now being prepared for decommissioning by disconnecting all live activities (steam, water, and electricity), establishing temporary utilities, and refurbishing required services (elevator and gantry crane). The dismantlement activities are scheduled for fiscal years 1987 and 1988. The total anticipated cost of the project is \$3.1 million.

## LESSONS LEARNED

A prime lesson to be learned by any organization involved in the D&D of large facilities contaminated with significant quantities of radioactive materials is that preproject planning must be accurate and thorough. Sufficient time and effort must be taken to accurately characterize the facility to determine the locations and quantities of radioactive materials. Records of past facility operation must be carefully scrutinized and correlated with the radiological characterization data. Finally, all available personnel who were associated with the operation of the facility should be interviewed to determine if there are any data missing from the facility operation records. This careful preplanning allows for accurate estimates of the time and resources required to complete the job.

## FUTURE PROJECTS

Future D&D projects to be completed at the Hanford Site include numerous small facilities and two large fuel reprocessing plants (Redox and U-Plant).

Redox is a 467 ft long by 161 ft wide by 82 ft tall (60 ft above grade) fuel reprocessing building which was operated from 1952 to 1967. The building consists of a double row of remotely-operated below-grade process cells separated from the operating and service galleries by 5 ft thick shielding walls. The estimated radionuclide inventory is 1500 Ci of fission products and 9,000 Ci of alpha emitters.

The reference decommissioning method is partial dismantlement followed by in-place disposal of all remaining structures. The project is scheduled to begin in fiscal year 1988 with the preparation of an Environmental Impact Statement. Completion is scheduled for fiscal year 1998 at a total estimated cost of \$34 million.

U-Plant is a 810 ft long by 66 ft wide by 77 ft tall (52 ft above grade) facility which was used from 1952 to 1958 to recover uranium from high-level liquid radioactive waste from early fuel reprocessing. The building has a single row of remote processing cells separated from the operating and service galleries by a shielding wall that ranges from 5 to 9 ft thick. An estimated 10,000 Ci of fission products remain in the facility at this time.

The reference decommissioning method is in-place disposal, with work scheduled to take 10 yrs beginning in fiscal year 1994 at an estimated cost of \$27 million.

1. ENVIRONMENTAL ASSESSMENT RELATING TO THE DECOMMISSIONING OF STRONTIUM SEMIWORKS FACILITY, U. S. Department of Energy, P. O. Box 550, Richland, Washington 99352, May 1985.

495/496

## REMEDIAL ACTION OF RADIUM CONTAMINATED RESIDENTIAL PROPERTIES

Duncan White and Jeanette Eng  
New Jersey Department of Environmental Protection  
Bureau of Radiation Protection  
380 Scotch Road  
Trenton, New Jersey 08628

### ABSTRACT

Since November 1983, the New Jersey Department of Environmental Protection (NJDEP) and the U. S. Environmental Protection Agency (USEPA) have been in the process of identifying properties in Montclair, Glen Ridge and West Orange, New Jersey, which were built over radium contaminated soil landfilled areas. Elevated indoor radon concentrations prompted the Centers for Disease Control (CDC) to issue a health advisory which included permanent remediation of radon progeny levels in excess of 0.02 Working Levels within two years of discovery. In order to expedite remedial action, NJDEP undertook a ten million dollar cleanup program.

Remedial Action at the 12 residential properties encountered some unanticipated problems despite the efforts of numerous government agencies and their contractors to characterize the contamination as much as possible prior to remediation. Some of the unanticipated issues include contamination from other radionuclides, underestimation of removal volumes, and controversy over the transportation and disposal of the radium contaminated soil at a commercial facility in Nevada.

This paper will review the approach taken by NJDEP to the remedial action for radium contaminated soil, discuss some of the issues encountered during the remedial action, and provide post remedial action data.

### INTRODUCTION

In July 1983, the New Jersey Department of Environmental Protection (NJDEP) began surveying areas for radiological contamination identified in a 1982 aerial overflight of 12 square miles of Essex County centered about a former radium processing facility in Orange, New Jersey. By November it became apparent from the indoor survey that the extent of elevated radon progeny concentrations in Glen Ridge, Montclair and West Orange was not restricted to a few homes. At this time NJDEP asked the United States Environmental Protection Agency (USEPA) for assistance in remediation of homes with high radon progeny levels and identification of additional properties built on radium contaminated soil.

The presence of elevated radon progeny concentrations, which ranged up to 1.5 Working Levels (WL), prompted the Centers for Disease Control (CDC) to issue a health advisory in December 1983. The advisory (Table 1) provided a schedule for permanent reduction of radon progeny concentrations to less than 0.02 WL within a period of two years from discovery. For 22 homes, temporary reduction of working levels greater than 0.1 was accomplished with the installation of ventilation systems.

In an effort to meet the schedule of December 1985 for the remediation of 45 homes with working levels in excess of 0.02 WL,

representatives from NJDEP and USEPA met in May 1984 to select properties to be remediated as part of a pilot program. The intent of the pilot cleanup program was to include properties where the location of the contaminated soil varied with respect to the house (e.g. under basement, under slab, etc.) in order to develop costs for the remaining properties. Figures 1 and 2 indicate the location of the four groups of properties (12 properties in total) selected for the program.

#### DEVELOPMENT OF STATE REMEDIAL PROGRAM

The implementation of the pilot program by USEPA and NJDEP would require a disposal site. Due to the December 1985 deadline set by CDC, an interim storage site was needed since development of a permanent disposal facility would take several years. The interim storage site selected by NJDEP was the West Orange Armory, a state owned property approximately eight miles from the contaminated properties. Although projected environmental and health impacts were minimal, NJDEP found it impossible to implement this interim storage option because of the intense local political and public opposition.

While NJDEP was looking for an interim storage site, the USEPA proceeded with engineering and radiological characterization work at the twelve properties. When it became apparent that it would not be possible to use the West Orange Armory as interim storage, USEPA decided it could not participate without completing a remedial investigation/feasibility study. NJDEP made the decision to proceed with state funding in order to expedite the cleanup. Eight million dollars was appropriated by the state legislature for the project. Since disposal of the contaminated soil at a commercial facility was the only option available consistent with meeting the CDC deadline, funds for the cleanup had to be sufficient to cover disposal and transportation costs.

#### RADIOLOGICAL CHARACTERIZATION

Despite the efforts of NJDEP, USEPA and their contractors to accurately estimate the volume of contaminated material, it became apparent during excavation that the actual volumes are significantly higher, as shown on Table 2. Since NJDEP contracts were based on a per unit basis, costs for completion of the project are now projected to be 14.5 million dollars, 80% above original estimates.

The reason for the low volume estimate can be traced to the characterization techniques employed. Original surveys of the properties were designed to determine exposure rates in order to estimate health risks. Subsequent subsurface investigations relied on these measurements to guide borehole placement. Buried contaminated soil would be missed with this technique because of the inhomogenous radium concentrations and irregular distribution in soil. This situation was further complicated by attempts to obtain a correlation between soil concentrations and gamma measurements at known depths in the cores. Although the aggregate of data for all the boreholes demonstrated this correlation, comparison of concentrations and gamma measurements on an individual borehole basis revealed large discrepancies. The use of a



Figure 1: Glen Ridge Properties in Remedial Action Program



Figure 2: Montclair and West Orange Properties in Remedial Action Program

pre-determined borehole gamma measurement to determine radiological concentration did not satisfactorily delineate contamination layers at cleanup concentration of 15 pCi/g of radium-226.

In order to rectify the problem, further subsurface investigation was undertaken at one set of properties that would be excavated shortly. Boreholes were placed uniformly over entire properties where no contamination was believed to be present. Areas where borehole gamma logging was not in excess of background were eliminated from excavation engineering considerations. This technique proved to quite successful in delineating uncontaminated from contaminated areas.

A second problem arose during excavation that was overlooked during characterization. All previous soil samples taken by various radiological survey groups showed radium-226 concentration higher and not in equilibrium with uranium-238. These samples revealed only a portion of a radium extraction waste streams. At the radium processing facility, waste streams containing uranium may have been discharged or disposed by other means than in landfills.

During routine analysis of some samples from the excavated areas, elevated uranium not in equilibrium with radium was found. A closer look at soil samples from the same vicinity collected during characterization work revealed that they also contained uranium minus the extracted radium.

Since uranium may be present in higher concentration than radium, different analytical techniques will be necessary to determine that uranium-238 concentration in the final verification soil samples meet the cleanup criteria of 40 pCi/g. Prior knowledge of the excess uranium would have resulted in additional field and laboratory equipment to be available at the beginning of the cleanup project.

#### RADIOLOGICAL EXPERIENCES DURING EXCAVATION

In order to minimize the inadvertent disposal of soil meeting cleanup criteria, radiological control of the excavation was critical. This was not always easy to minimize due to the nature of the excavation. The properties on which excavation was taking place bordered properties that were also contaminated. This problem was compounded when the excavation to depths greater than 6 feet exposed deeply buried contaminated layers in the adjacent properties. Due to gamma shine, the radiation fields were higher than usual in the excavated area, making it difficult to differentiate contaminated from uncontaminated soil.

When excavation was completed, the field verification of the remaining soil layer was complicated by the elevated gamma field. Pressurized ion chambers and to a lesser extent the scintillometers used in the verification process were sensitive to the nearby contaminated layers. As a result, there was a greater dependency on soil samples to verify the cleanup soil criteria.

The shipment of the soil as Radioactive LSA requires that the total activity of each container be stated on the manifest. In order to determine the activity of each container without soil sampling all of

them, a rapid means of determining activity content was needed. This was accomplished by taking a contact exposure rate measurement and representative soil sample from a number of containers. Once the container was weighed and the radium concentration of the soil sample determined, a correlation between the exterior contact exposure measurement and total activity per container was established.

During excavation, random soil samples were taken daily of the filled containers and analyzed in a field laboratory. These results are compiled in Table 3. Comparison of these results with those derived from exterior gamma measurements show a similar range of concentrations for radium. However, the average concentration for the set of drums determined by soil analysis was more than twice the concentration determined solely by gamma measurements. The differences were probably due to analyzing soil samples without preparation to eliminate moisture and reach equilibrium, as well as due to the non-homogeneous nature of the waste in the containers.

#### IMPLEMENTATION OF DISPOSAL OPTION

In New Jersey, a controversy arose between NJDEP and the town of Kearny where the sealed drums were taken for weighing and loading on trailers for rail shipment. The drums were taken to Kearny because the transportation contractor's railyard was located in the industrial portion of town. The contractor intended to consolidate 70 trailers for shipping on a unit train to the disposal site. The town resisted the temporary storing of containers because of the concern over becoming an interim storage area for this material.

NJDEP decision to dispose in Nevada was based on competitive bidding for the transportation and disposal contracts. When it came to attention of the Nevada governor's office and local officials in Las Vegas that New Jersey was shipping a large volume of soil to the Beatty commercial low level radioactive waste site, Nevada opposition against disposal intensified. After an initial State of Nevada attempt to block the shipment failed, both the State of Nevada and Clark County instituted new hazardous material permitting requirements which brought the issue back to court. The case will be heard before the U.S. Supreme Court at a later date after a New Jersey request for an immediate hearing was rejected. This effectively ceased cleanup operations with only five of the twelve properties remediated.

Ironically, during NJDEP's difficulties with disposal, over 1000 cubic yards of cobalt-60 contaminated soil, concrete and equipment was shipped to Barnwell by a private contractor who was decommissioning an irradiation facility in New Jersey.

#### POST REMEDIAL MEASUREMENTS

Restoration has been completed at four of the five properties in which remediation occurred. Table 4 shows the first set of indoor radon and radon progeny taken at 103 Carteret Street one month after the owner returned. For 18 months prior to remediation, the concentrations in this house were temporarily lowered by installation of a ventilation

system in the basement. Post remediation concentrations without the ventilation system are typical of natural background levels in northern New Jersey houses.

#### CONCLUSION

In order to remediate a dozen properties in Montclair, Glen Ridge and West Orange, NJDEP undertook remedial action with disposal at a commercial facility. Despite extensive characterization of the contaminated soil, volume estimates were underestimated. Evaluation of the characterization techniques revealed that a systematic subsurface investigation of the entire properties, using borehole logging to identify above background gamma radiation as an indicator of possible contamination, can accurately limit the areas requiring consideration for excavation. The importance of complete sample analyses during characterization was demonstrated by the unexpected finding of uranium contamination during the excavation phase. Due to the proximity of contaminated material on adjacent properties, there was a greater reliance on soil samples rather than field measurements to verify excavation for this project.

Table 1: Center for Disease Control--Risk Management Plan

<u>Tier</u>	<u>Radon Progeny Level</u>	<u>Time Frame</u>	<u>Action (Exposure Reduction)</u>
A	over 0.5 WL	a) immediately (1-2 days)	Restrict occupancy in high level areas of homes to 2 hours per day. Prohibit smoking in high level areas.
		b) within 2 weeks	Temporary remedial action to get as far below 0.5 WL as feasible using temporary measures.
B	0.1-0.5 WL	Prompt (1-3 months)	Temporary remedial action to get to 0.1 WL or less. (Prioritization of action by exposure level.)
C	0.02-0.1 WL	1-2 years	Permanent remedial action to reduce exposure from non-natural sources to below 0.02 WL (prioritization of action by exposure level).
D	0.02 WL		No Action.

TABLE 2: PROJECTED AND ACTUAL VOLUMES OF EXCAVATED SOIL  
FOR PHASE ONE PROPERTIES

<u>Location</u>	<u>Projected</u>		<u>Actual</u>	
	<u>Surface Area</u>	<u>Volume</u>	<u>Surface Area</u>	<u>Volume</u>
Carteret Street	2615	470	4750	1014
Lorraine Street	1790	320	5030	1217
Virginia/Franklin Avenues	14770	2725	15680(1)	1716(1)
Amelia Street	3290	805	(2)	(2)

Surface Area in square feet

Volume in cubic yards

(1) Excavation not complete

(2) Excavation not started

Table 3: Average Radium Concentration on Excavated Soil

1. By Soil Analysis

<u>Excavation Site</u>	<u>Average Ra-226 Concentration (pCi/g)</u>	<u>Range (pCi/g)</u>
Carteret	110	9.2 - 1264
Lorraine	280	5.4 - 7436
Virginia/Franklin	55.6	3.8 - 814
All Sites	192	3.8 - 7436

2. By Contact Gamma Measurement

All Sites	74.4	25 - 5792
-----------	------	-----------

Table 4: Radon and Radon Progeny Results for 103 Carteret Street

<u>Status</u>	<u>Radon (pCi/l)</u>	<u>Radon Progeny (WL)</u>
1. <u>Pre-Remedial</u>		
a. basement	59.7	0.135
b. first floor	42.6	0.177
c. second floor	32.6	-
2. <u>Ventilation System Installed</u>		
a. basement	17.2	0.020
b. first floor	4.2	0.032
c. second floor	3.3	-
3. <u>Post Remediation</u>		
a. basement	1.6	0.005
b. first floor	0.8	0.003
c. second floor	0.5	-

EXPERIENCE GAINED DURING REMEDIAL ACTION AT RESIDENTIAL PROPERTIES  
CONTAMINATED WITH Th-232 AND Ra-226

William L. Beck, Richard Lynt, Edward N. Lazo  
Bechtel National, Inc., P.O. Box 350, Oak Ridge, TN 37830  
Michael R. McDougall  
Eberline Analytical Corporation, P. O. Box 3874,  
Albuquerque, NM 87190

ABSTRACT

Seventeen residential properties contaminated with Th-232 and Ra-226 were decontaminated and restored by Bechtel National, Inc., during 1984 as part of the U.S. Department of Energy's Formerly Utilized Sites Remedial Action Program. The extent of contamination varied from only a few cubic yards of material in the edge of one lawn to extensive contamination around foundations and under basement floors. Three unusual "case histories" will be described in detail. In the first case, a few cubic yards of contamination were expected, but more than 100 cubic yards were found. In Case 2, contamination was found under the floor of an addition to a house. Case 3 involved a home built on a lot where contaminated soils had been used as fill material to depths of up to 10 feet. The remedial action included the removal of the basement, including the floor slab and foundation.

This paper will provide details about several lessons learned during remedial action at residential properties: a) radiological characterization surveys must be thorough; b) interviews with owners can provide helpful information about unusual or unexpected contamination problems; c) radiological monitoring during construction is necessary for efficient and thorough remedial action; and d) good public relations are an important element of residential remedial action projects. While these lessons may seem to be "common sense" items, the circumstances that require their use are often very subtle. As the case histories show, successful remedial action requires constant vigilance and careful attention to details.

INTRODUCTION

Seventeen residential properties in the Maywood, New Jersey area that were contaminated with Th-232 and Ra-226 were decontaminated and restored during 1984. These properties were contaminated with "residues" or wastes from the former Maywood Chemical Works located nearby. From 1916 through 1956, this plant extracted thorium and/or rare earths from a natural, sand-like ore called Monazite. The slurry waste, or residue, was pumped into large settling basins for disposal. This waste material was not considered as hazardous during most of this time; consequently, some plant workers were allowed to haul the material away and use it as fill dirt or mulch. Contaminated material also washed onto properties adjacent to the settling basin areas during heavy rains and floods. These mechanisms resulted in the contamination of the above mentioned 17 properties. The

plant was licensed by the Atomic Energy Commission in 1954, 2 years before thorium operations ceased. Between 1956 and 1960, several plant areas were decontaminated and restored, and the contaminated materials were buried on the plant site. In November of 1980, a radiological survey along Highway 17, which was built through the Maywood Chemical Works site, was done in search of a lost radiographic source. This survey indicated that several large areas along the highway were contaminated, and this finding prompted the Nuclear Regulatory Commission to request an aerial radiological survey of a 4- by 5-mile area centered over the former Maywood Chemical Works site. The aerial survey confirmed that several areas, ranging in size from a few acres to about 50 acres, were contaminated (Ref. 1). Ground-level surveys of these areas confirmed the need for remedial action (Refs. 2-4).

Remedial action at the Maywood area was added to the U.S. Department of Energy's Formerly Utilized Sites Remedial Action Program (FUSRAP) by the 1984 Energy and Water Development Appropriations Act. Bechtel National, Inc. (BNI) is the Project Management Contractor and, as such, plans and directs the remedial action. BNI has a major subcontract with Eberline Analytical Corporation (EAC) for radiological services, including health physics field personnel, radiological instruments, and analytical and dosimetry services. Local construction contractors were used for the actual remedial action, including excavation, demolition, and restoration.

#### REMEDIAL ACTION CASE HISTORIES

For 14 of the 17 properties decontaminated in 1984, the remedial action was completed with very few changes in design plans. The only difference worthy of note was that excavated volumes were typically 1.5 to 2 times larger than expected. This was probably due to the field monitoring technicians being somewhat conservative in their decisions to continue or stop excavation. This conservatism was not diminished by the fact that at least three organizations (BNI, a DOE Independent Verification Contractor, and the New Jersey Department of Environmental Protection) and possibly others would be reviewing the post-remedial action data to verify the completeness of the cleanup. This conservatism can also be credited to the application of ALARA to this work.

For 3 of the 17 properties, the remedial action varied considerably from the design plans. Each property had rather unique problems, and the review of these "case histories" indicates the types of problems that can arise in this work. In other words, Murphy's laws can and do apply to residential remedial action projects!

#### CASE HISTORY 1

Case History 1 (CH1) was one of nine contaminated residential properties whose backyard was adjacent to a former residue settling basin. However, unlike the others, property CH1 had a somewhat elevated backyard. When surface gamma radiation surveys showed no readings to indicate contamination levels in excess of criteria it was felt that this finding only served to reinforce the belief that the contamination on these properties was the result of migration by rainwater runoff. A limited subsurface investigation was undertaken for these properties consisting of 16 boreholes that were gamma logged and 22 subsurface soil

samples that were analyzed. Only one subsurface sample exceeded the remedial action guideline and only by 25 percent. Thus, it was expected that very few areas would require more than a 6-inch-deep excavation.

Remedial action for property CH1 began at its boundary with the adjacent settling basin property. Monitoring during excavation quickly revealed a subsurface region of contamination about 2 to 3 feet below the surface and extending to about 7 feet deep. Excavation of this "pit" produced approximately 100 cubic yards of contaminated material, including soil, tree stumps, concrete blocks, and other construction debris (Figure 1). After excavation of this "pit", a nearby resident recalled that the area had been used as the trash pit during construction of several houses on the street. The pit appears to have served as a "sump" for contaminated materials for about 30 years prior to the remedial action. Two factors led to the overlooking of a sizeable volume of contaminated material in the remedial action design. First, the uncontaminated overburden prevented detection of this material by surface measurements and, second, the use of a "logical" approach to limiting the subsurface investigation did not allow for the presence of anomalous conditions.

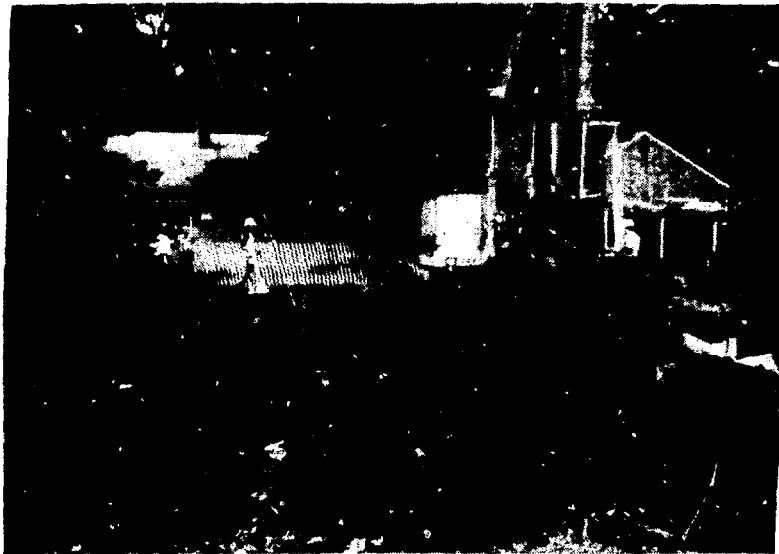


Figure 1. Excavation in the backyard of property CH1

## CASE HISTORY 2

Case History 2 (CH2) was also a property adjacent to the waste settling basin property. However, this property was located such that surface water runoff was not the most likely mechanism for contamination.

The initial radiological survey indicated that four small areas were contaminated as shown in Figure 2. The small sizes and the scattered locations of these areas suggested that they were areas where the contaminated soil had been used as fill dirt. Radon daughter measurements

made inside the house were indistinguishable from background. Gamma measurements made inside the house were also equal to background except for some slightly elevated readings just inside the wall from the number 4 area of contamination. It was assumed that these readings were "shine" from the outside material.

Excavation of contaminated areas 1, 2, and 3 was completed as planned. Area 4 at the rear of the house was also excavated as planned, but gamma readings in the excavated area remained at about 75 microroentgens per hour. A careful survey with a beta/gamma instrument indicated that radiation was coming through the foundation wall. A concrete block was removed from the wall and measurements within the hole ranged from 200 to 1000 microroentgens per hour. A closer inspection of the construction details of the house showed that the contamination was in a confined area that had originally been a porch between the house and the garage. The porch foundation had been filled with contaminated material during remodeling of the porch into an additional room. Analysis of a sample of the contaminated fill material indicated about 2000 picoCuries per gram of Th-232.

Before excavation of the material began, the wall connecting to the house was sealed with plastic to prevent dust and radon infiltration. The 6-inch-plus cement floor was then broken up and removed. This revealed a layer of sand approximately 12 inches deep covering the contaminated fill. The relatively thick cement floor and the sand had formed an effective shield that prevented discovery of the contamination during the earlier radiological survey.

Removal of the contaminated soils was accomplished in about 10 hours by hand labor. When the excavation reached about 95 percent completion, the walls were washed down and the "floor" surface was cleaned last. The walls were surprisingly easy to clean because they had been heavily coated with a sealant during construction. This sealant prevented seepage of radon into the house, accounting for the radon levels in the house being no higher than background. Approximately 10 cubic yards of contaminated soil were removed.

As in CH1, a significant quantity of contamination had been shielded from the initial survey. As in CH1, owner information provided the explanation as to the origin of the contamination. And, although the additional volume of material removed was not large, additional man-hours and materials were spent for both contamination removal and property restoration.

#### CASE HISTORY 3

The Case History 3 (CH3) property was located about 1 mile north of the former Maywood Chemical Works. A former owner of the CH3 property was interviewed. He recalled obtaining the fill dirt from the thorium plant and using it to raise the level of a vacant lot behind his former home. The lot was later sold, and a house was constructed there during the early 1950s.

The initial radiological survey of this property revealed extensive contamination in the yard and under the basement floor. Radon and radon daughter levels were also found to be elevated, particularly in the basement.

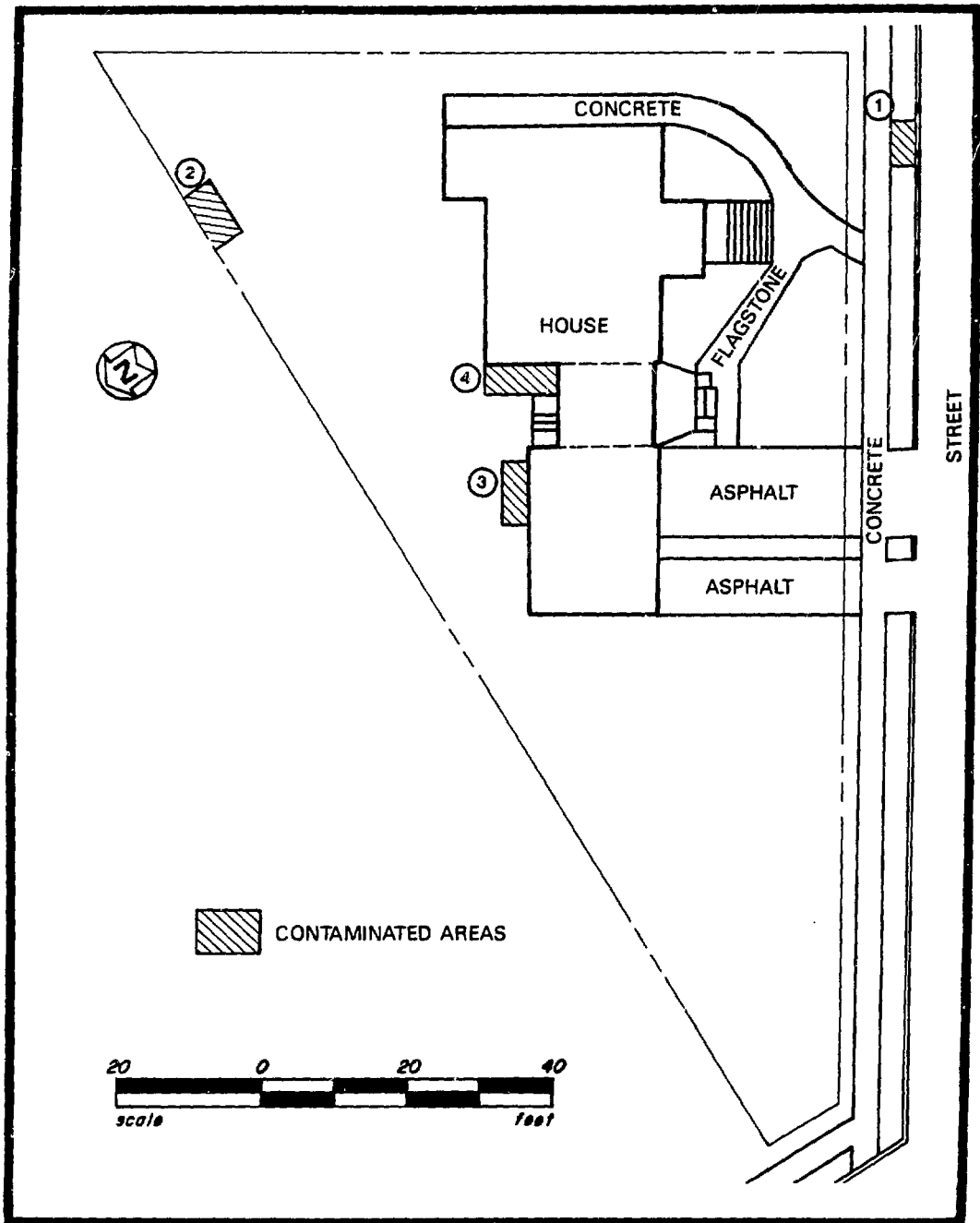


Figure 2. Areas of elevated radionuclide concentrations at Property CH2

The remedial action plan for this property called for excavation of contamination along the outside of the foundation walls, removal of the concrete floor, and excavation of the contamination under the floor and along the inside of the foundation wall. It was assumed, based on initial radiological survey data, that the lower part of the foundation wall and the footings were in clean soil. The remedial action plan included a contingency plan for removing contaminated materials from under the footings.

Excavation began in the front yard. After 2 feet of soil was removed, as called for by construction drawings, soil samples assayed at background thorium concentrations while gross gamma readings were still about 500 microroentgens per hour. An additional 2 feet of excavation revealed that clean fill had been placed over a 3-foot-thick layer of highly contaminated soil: from about 200 to 2000 picoCuries per gram. This proved to be the case throughout much of the front yard such that excavation reached 8 feet below original grade in most of that area. Radiation levels in the excavated areas reached as high as 7 milliroentgens per hour at soil contact.

After removing the basement floor and excavating approximately 2 feet of contaminated soil, another major problem was encountered. The foundation wall was constructed of two different widths of concrete blocks. This allowed the contamination to get inside the foundation wall. Consequently, the house had to be supported and the entire foundation removed in order to complete decontamination. This is shown in Figures 3 through 5.

Again, significant quantities of contamination were not detected in the initial surveys. In this instance, the volume of material removed was about 3 times greater than anticipated.

#### SUMMARY OF LESSONS LEARNED

Much valuable experience was gained from remedial action in 1984 at 17 residential properties. Lesson number one was that radiological characterization surveys should be very systematic and complete. Measurements with a directionalized gamma scintillation instrument should be made on a grid with spacings of no greater than 10 feet over all of the property and inside all buildings. Subsurface measurements should also be done in a systematic fashion with the frequency determined by judgement in the field. Being "logical" and trying to reduce costs by limiting characterization surveys led to the underestimation of excavation volumes and overruns in construction times and costs. This situation also leads to a loss of confidence in the project management contractor by the client, regulatory agencies, subcontractors, and the public.

Another lesson learned was that sitting down with the property owner and discussing the entire remedial action process was time well spent. These discussions with property owners should include questions related to previous construction activities at or near their properties. Answers to these questions may provide useful information about the distribution and amounts of contaminated materials. If told why and where the property is contaminated, what the potential risks are from both a health effects and economic standpoint, how the remedial action is to be accomplished, the projected schedule, and the potential for unusual problems and delays, the property owner gains an understanding of the problem and generally becomes a more willing participant in the remedial action process.



Figure 3. Removing the foundation at Property CH3

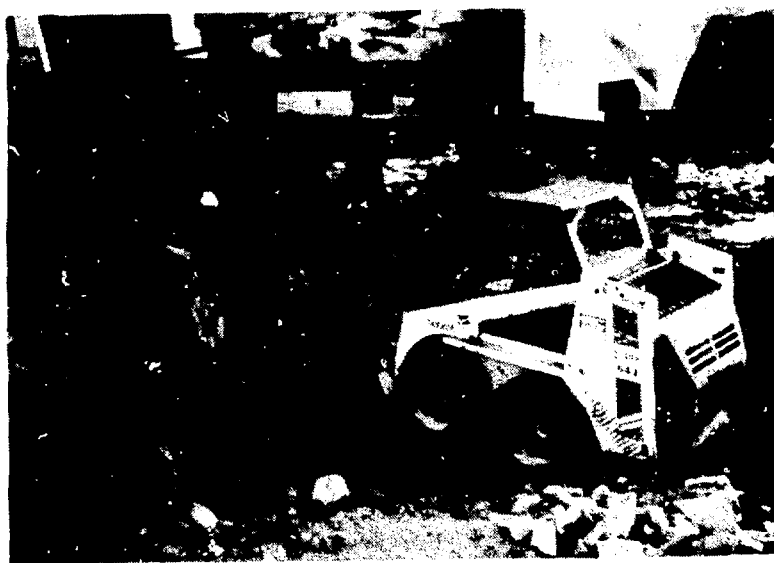


Figure 4. Removing contaminated soil and rubble from under Property CH3



Figure 5. Property CH3 after removal of the basement level including the floor, foundation, and footings.

#### REFERENCES

1. J. R. Mueller and S. A. Gunn, An Aerial Radiologic Survey of the Stepan Chemical Company and Surrounding Area, Maywood, New Jersey, EG&G Survey Report NRC-8109, Maryland, April 1981.
2. Henry W. Morton, Natural Thorium in Maywood, New Jersey, Nuclear Safety Associates, Inc., Maryland, (Undated).
3. L. W. Cole, J. D. Berger, P. Cotten, R. Gosslee, J. Sowell, and C. Weaver, Radiological Assessment of Ballod and Associates' Property (Stepan Chemical Company), Maywood, New Jersey, Oak Ridge Associated Universities, Oak Ridge, TN, July 1981.
4. J. Cirilli, Radiological Study of Maywood Chemical, Maywood, New Jersey, Report R-584-11-83-1, NUS Corporation, November 1983.

DECONTAMINATION AND DECOMMISSIONING OF A LUMINOUS DIAL  
PAINTING FACILITY: RADIOLOGICAL CHARACTERIZATION,  
SEGREGATION AND DISPOSAL OF BUILDING MATERIALS

D. Ed, L. Chu, P. Chepulis, M. Hamel  
Illinois Department of Nuclear Safety  
1035 Outer Park Drive  
Springfield, Illinois 62704

ABSTRACT

The State of Illinois, Department of Nuclear Safety, has decontaminated and decommissioned the defunct Luminous Processes, Inc. facility located in Ottawa, Illinois. The State's overall experience throughout the project is generally described, with particular emphasis given to the radiological characterization (Ra-226+progeny and H-3) and subsequent segregation and disposal of building materials as either "radioactive" or "non-radioactive." Experiences involving direct application of health physics principles (criteria selection, sampling schemes, analytical techniques, data reduction, quality assurance) are discussed. Experiences involving other health physics regimens (personnel protection and dosimetry, environmental monitoring) as well as social sciences and economic considerations (public perception, media relations, political involvement, contractor interactions, fiscal management) are discussed only insofar as they affect the radiological characterization, segregation and disposal processes.

INTRODUCTORY OVERVIEW

Historical Background

Luminous Processes Incorporated, operated a luminous dial painting facility in Ottawa, Illinois from 1934 until 1978. The company has since gone into receivership, leaving a two-story, 17,000 square foot radiologically contaminated building. Until recently, the unoccupied structure had remained essentially "as abandoned" in the midst of downtown Ottawa.

In the spring of 1983, \$2 million was appropriated for the decontamination, decommissioning and physical removal of the facility. (1) Responsibility for this undertaking was assigned to Illinois Department of Nuclear Safety (IDNS).

Responding to this mandate, the Department performed a preliminary investigation of the structure, its contents, and the proximate surrounding property. The building was found to contain a myriad of assorted items, ranging from furniture (tables, chairs, desks, stools, benches) to production machinery (silk screens) to general debris (old paint containers, rags). Almost all of these items were contaminated with both radium (Ra-226+progeny) and tritium (H-3). The level of contamination exhibited by the items was of such a magnitude that in situ radiological characterization of the building surfaces, and surrounding areas was difficult.

IDNS concluded that decontamination, dismantlement, and disposal of the facility was to be anything but a trivial task.

## **Project Description**

The overall project was divided into three distinct phases:

### **Phase I**

In order to simplify the assessment of radioactive contamination of the building itself, all non-structural items were removed from the building during this phase. This was accomplished by decontaminating salvageable items to levels of radiological exposure and radioisotope concentration below pre-determined limits, followed by release for unrestricted use. For instance, the silk screen machines were salvageable. If decontamination was determined to be economically infeasible or otherwise undesirable, items were disposed as radioactive waste in accordance with established criteria. Furniture, general debris and decontamination residuals were handled in this way. At the end of this phase all that remained of the building were the walls, roof, floors, and associated beams, rafters, joists and other entities necessary to insure structural stability. Of course, also remaining were the soils and associated materials immediately under and surrounding the building.

### **Phase II**

During this phase, the building proper was decontaminated, dismantled and disposed. Since decontamination to levels at or below those considered acceptable is not always feasible, (or in certain instances, even desirable) building rubble was segregated into three categories:

- 1) low-level radioactive waste,
- 2) non-contaminated refuse, and
- 3) non-contaminated salvage.

Materials falling into "category one" were shipped to the low-level radioactive waste disposal site near Richland, Washington; "category two" materials were suitable for disposal as ordinary refuse in a sanitary landfill; materials qualifying as salvageable under "category three" were released for re-use without restriction.

At the end of this phase, the LPI property was void of all above-grade building and structural materials; only below-grade materials (i.e., pipe, footings) and soil remained.

### **Phase III**

During this phase, the soil and below grade materials at the site will be radiologically characterized, segregated, and disposed or salvaged in a manner similar to that of Phase II. Phase III is scheduled to be completed by late December, 1985.

The remainder of this paper describes activities undertaken during Phase II of the above-described project. It focuses specifically on the radiological characterization, segregation and disposal of building materials. Mention or description of other activities occurring during this phase are included if particularly interesting or necessary for clarity or completeness.

## **THE STORY OF PHASE II**

### **Organizational Responsibilities**

Crucial to any such undertaking is assembling a team of individuals competent to perform the work required, and the assignment of specific tasks to

those individuals according to the abilities of each. To perform the activities of Phase II, the Department assembled a team of individuals representing three distinct organizations: IDNS, Argonne National Laboratory (ANL) and IT Corporation (IT).

IDNS maintained overall responsibility for direction and management of the project. All procedures were either authored or approved by IDNS. The execution of these procedures occurred under the oversight of IDNS personnel. Always present at the site during Phase II activities were the IDNS project manager, the IDNS principal health physicist, a team of three or more staff health physicists, one or two radiochemists, one electronics technician and one secretary.

ANL functioned as the Department's technical advisor, providing input and assistance in the development of criteria and procedures. Additionally, ANL staff maintained a constant presence during Phase II activities, providing personnel for quality assurance "overview" surveys. The ANL mobile laboratory was at the site for the duration of Phase II, facilitating quality control "cross-check" analysis of laboratory samples.

IT Corporation conducted the day-to-day activities of the project. IT arranged for the necessary manpower (technical and non-technical); provided mechanical, electrical and office equipment (everything from end-loaders to coffee pots); performed rad-waste packaging, shipping and disposal; addressed the seemingly endless occurrence of "minor details" that if not addressed would bring the entire operation to an abrupt halt. IT's staff performed the majority of routine radiological surveys, took and prepared samples, and conducted some of the laboratory analyses. IT provided the laborers who actually performed the decontamination and dismantling jobs.

#### Decontamination Criteria

Decontamination criteria determine "that level of contamination below which something is considered to no longer be contaminated."

Unfortunately, there are no universally accepted decontamination criteria for building materials. While there are criteria for decontaminating a building to levels allowing unrestricted reoccupancy (2), they are perplexingly unclear and ambiguous. Additionally, there are criteria defining acceptable levels of Ra-226 and other isotopes in materials such as mill tailings (3), and for H-3 in waste materials (4), but their applicability to the case under discussion is not straightforward.

IDNS purposely selected "conservative" criteria in order to assure that no radiological problem with decontaminated material could occur (5). The criteria were:

Maximum isotopic concentrations:	Ra-226.....5 pCi/g H-3.....50 pCi/g
Maximum removable surface contamination:	Ra-226.....20 dpm/100 sq cm H-3.....1000 dpm/100 sq cm
Maximum fixed surface contamination:	Ra-226.....100 dpm/100 sq cm H-3.....5000 dpm/100 sq cm
Maximum exposure rate:	.....0.25 mR/hr @ 1 cm

The "concentration" criteria were considered satisfied if average concentrations were less than the specified limit. The average concentration

in a volume could be estimated by measuring the concentration in a single composite of different samples from the volume in question or by computing the average of concentrations measured in individual samples of the volume in question. In either case:

- 1) the samples were statistically representative of the volume being tested;
- 2) the maximum concentration was identified and included in the average;
- 3) the maximum concentration did not exceed ten times the limit; and
- 4) the averaging layer at any depth did not exceed fifteen centimeters.

The "surface" criteria were allowed to be averaged over areas less than one square meter provided that the maximum activity in any one hundred square centimeter area was less than three times the limit.

#### Procedures

After the criteria were adopted, procedures were developed that would determine whether contamination levels exceeded these criteria. These procedures addressed in situ survey techniques, portable instrument calibration, sample collection, sample analysis, laboratory instrument calibration, and quality assurance activities for each.

Procedures for packaging, transport, and disposal of LSA waste materials, as well as the transport and disposal of non-rad waste materials were also prepared. Additional procedures regarding both occupational and environmental health physics practices were devised, as well as those addressing decontamination techniques. Finally, procedures were prepared for training individuals to conduct the above procedures.

The preceding describes only those procedures related to radiological activities; many more were prepared addressing issues such as project management and administration, building dismantlement, safety and industrial hygiene, security and again, training. All such procedures are described in detail by the Phase II "Manual of Standard Operating Procedures." (5)

#### Building Demarkation

Prior to commencement of any radiological measurement or sampling regimen, it was necessary to devise a scheme for identifying the precise locations at which these measurements or samples were taken. To accomplish this, all building surfaces (horizontal and vertical, interior and exterior) were identified by and referenced to a grid system.

Architectural drawings (plans and elevations) of all building surfaces were commissioned. On the drawings, each building surface was demarked into ten foot by ten foot squares called "grids." Each grid was further divided into nine equal subsections called "subgrids." These subgrids equaled approximately one square meter in area (each grid appeared as a large square containing nine smaller squares resembling a "tic-tac-toe" board). Each grid was uniquely numbered; the subgrids were identified alphabetically.

Once the drawings of the building had been entirely demarked by this grid system, the system was duplicated on the building surfaces proper (i.e., the grids were painted on the walls, floors, ceilings, roof, etc.). At this point, the location of any measurement made or sample taken could be uniquely identified and easily documented.

## In Situ Radiological Survey Protocol

This procedure was developed to determine, within reasonable limits of confidence, the levels of Ra-226 and H-3 contamination fixed on the surface of building materials and to detect the presence of Ra-226 (plus progeny) within such materials. All building surfaces were surveyed (many more than once) in accordance with this procedure.

Initially, each subgrid was surveyed utilizing a portable gas-flow proportional instrument (e.g., Eberline PAC-4G) in the alpha+beta mode. Each subgrid was scanned in its entirety at a rate not exceeding one-half linear inch per second and at a distance from the surface not exceeding one-quarter inch.

For each subgrid, the location and intensity of both the highest and lowest count rate were identified and documented. Also estimated for each subgrid was the average count rate. Additionally, the location of the highest alpha+beta reading over the entire grid (not subgrid) was uniquely identified and documented; this facilitated subsequent sampling of the grid for laboratory analysis.

An additional and similar survey was likewise performed over each grid and subgrid with the gas flow instrument configured in the alpha only mode (this survey was also performed utilizing zinc sulfide scintillation instruments). Again, the highest, lowest and average readings were identified and recorded.

These scanning surveys were used to identify locations at which more accurate measurements could later be made; no attempt was made to quantitatively correlate the observed count rate with contamination per unit area.

Following the two "scanning" surveys (alpha+beta and alpha only), a third survey was performed for each grid. With the gas flow proportional instrument configured in the alpha only mode (or with ZnS type scintillation instruments), stationary "contact" readings were taken at the two highest points encountered in each of the two scanning surveys on each grid. For this survey, the detector was held stationary in contact with the surface for a minimum period of one minute. Here the results were recorded in cpm/detector surface area and later converted to dpm/100 sq cm as Ra-226.

Since the release criteria contain provisions for exposure rate determinations, a fourth survey was in order. Each grid was scanned in its entirety utilizing a collimated 2X2 NaI(Tl) detector configured with an appropriate rate meter. The highest count rate identified in each grid was recorded for later conversion to exposure rate. The location exhibiting this reading was also marked, facilitating future sampling of the grid.

A final survey of the materials from each grid was performed after the grid was dismantled. This "rubble" survey was performed with a 2X2 NaI detector as a precautionary, final check of building materials prior to packaging and shipment.

Each instrument combination (detector/rate meter/cable/etc.) utilized in all of the surveys was calibrated on all scales and in all operational modes on a daily basis. This required utilization of alpha and beta sources, both varying in strength by at least four orders of magnitude. The calibration was routinely checked throughout the day (or night) as the instrument was used; any discrepancy resulted in repair, if necessary, and recalibration.

## Survey Data Reduction

Once the field or "raw" data had been collected, the task of converting counts per minute into meaningful units of dpm/100 sq cm or mR/hr was necessary. This was accomplished by applying the appropriate factors for detector efficiency and surface area, electronic amplification, and background. To avoid the complications that accompany performing mathematical calculations in the field, the conversion of data from "raw" to "finished" was done after each survey crew's shift.

IDNS conservatively decided to assume that the dpm/100 sq cm (more correctly alpha/min per 100 sq cm) derived from the alpha contact survey were due to solely Ra-226 nuclear transitions, i.e., any contribution from progeny transitions was attributed to Ra-226. This knowingly biased the results for fixed Ra-226 surface contamination high by a factor as large as four.

It is difficult to measure H-3 in situ. Therefore, an attempt was made to devise a scheme for inferring the H-3 surface contamination from the more easily measured Ra-226 contamination. It was thought that there might be a constant ratio between radium and tritium contamination; then the simple application of a factor to the radium number might estimate the tritium number. Unfortunately, an empirical study at the site proved that there was no uniform correlation between the amount of contamination from each respective isotope. H-3 contamination therefore had to be determined by sampling and laboratory analyses (discussed in a subsequent section of this paper).

Derivation of exposure rate was more straightforward. The NaI detectors were calibrated against Ra-226 (plus equilibrium progeny) sources causing exposure of known magnitude. All scales were calibrated by varying the detector/source distance. The calculated exposure rates assumed that radium was in secular equilibrium with its progeny.

### Sampling for Laboratory Analyses

To determine the concentrations of Ra-226 and H-3 in the various types of materials at the site, a scheme was developed for collecting and analyzing samples of these materials. To meet the intent of the decontamination criteria, this scheme should provide for "representative" sampling as well as accurate and precise laboratory analyses for concentrations at or just below the criteria.

It was soon discovered that to sample the entirety of the materials at the site "representatively" with satisfactory statistical accuracy required an effort of heroic magnitude. Since the project's resources could not support such an effort, a simplified, but conservative, sampling regime was adopted. Each survey (alpha, alpha+beta, exposure) had identified the "highest" respective reading over each grid. The procedure assumed that samples from one or more of these "high" locations should contain the "highest" radioisotope concentrations (excepting of course, the Nemesis tritium).

Two sampling programs were therefore devised: one for radium and one for tritium. For radium, a sampling location for each grid was identified by reviewing the in situ surveys. Comparing the relative intensities of the maxima of these surveys, a sampling location was chosen. Samples (about 2000 gm) were obtained by "coring" through the wall, floor, or roof at these locations. Samples of beams, joists and other such materials were obtained by drilling, planing, scrapping or other techniques as deemed appropriate and practical.

For tritium, IDNS decided to take a large number (about 500) of small volume (about 5 gm) samples from randomly selected locations throughout the building. These were prepared and analysed to determine what percentage, if any, exceeded the criteria. None of these samples exhibited concentrations approaching even ten percent of the criteria (surface or density). This finding, combined with the fact that each of the samples taken for radium analysis (over 1000) were also tested for tritium content, seemed to be an adequate test for tritium contamination.

Since the criteria also addressed removable (i.e., wipeable) contamination, an additional sampling program testing for this type of contamination was devised. Wipe samples were taken from two locations on each grid, one at the point of "highest" contamination as determined by in situ surveys, the other randomly at some point exhibiting non-elevated readings. Two samples were taken because the possibility existed that, through the years, the point of high contamination might have remained so because the contamination was "fixed" and not easily removable; the reverse being true for points exhibiting lower in situ readings.

Many other environmental and occupational health physics samples (air, urine, dosimeter) were collected. Since these do not directly relate to the radiological characterization of building materials, this paper does not discuss them.

### Radioanalytical Techniques

In all of the below-described analytical schemes, the counting time and sample size were adjusted to be of sufficient magnitude to detect activities at or below the release criteria, taking into consideration detection efficiencies, recovery efficiencies, self-absorption, quenching, etc. (6)

The concentrations of Ra-226 in materials such as brick, concrete, wood, steel, and soil were determined through use of high resolution gamma spectroscopy (HPGe). Masonary samples were dried and crushed to a powder-like consistency; wood and steel were analyzed as "shavings"; soil was dried. Aliquots (700 to 800 gm) of these samples were weighed into 500 ml Marinelli beakers and counted for 100 minutes in a low background environment. Standard minimum detectable concentration algorithms clearly show that 750 grams (3750 pCi at the release criteria), counted for this length of time with reasonable detection efficiencies, should clearly indicate the presence of Ra-226 @ 186 kev, Pb-214 @ 351 kev and Bi-214 @ 609 kev.

The amount of removable Ra-226 surface contamination appearing on wipe samples was determined utilizing low background gas-flow proportional counting in both the alpha only and alpha+beta modes. The samples were simply planchet mounted and counted for a period of 10 minutes. In the low background environment of the proportional counter, this provided adequate sensitivity; one drawback was that no determination of self-absorption could easily be made. For this reason, results at or near the release criteria were closely examined, applying the professional judgment of the analyst.

Tritium concentrations were determined by oxidizing an aliquot (1 to 5 gm) of the sample, condensing the resultant tritium oxide and measuring the activity by liquid scintillation techniques. Reasonably complete oxidation of the tritium followed by low background counting of the resultant water, easily enabled detection of tritium at and below the criteria. Determining the accuracy and precision of the oxidation process on such a wide variety of physical matrices presented some difficulty; these were again addressed through use of "conservative" assumptions.

The amount of removable surface tritium contamination was measured by oxidizing the wipe samples (after Ra-226 analysis) and again counting the resultant water using liquid scintillation. Here, the oxidizer efficiency and condensate recovery were more easily determined; since the samples were all of the same physical and chemical composition, precision was more easily maintained. These differences tended to complement the lower level of activity specified by the criteria for removable contamination, providing for easily obtained minimum levels of detection at and below the criteria.

#### Reduction of Laboratory Data

Raw counting data were converted to dpm or dpm per unit area and pCi or nCi per unit weight utilizing accepted algorithms. (7,8) Uncertainties in these results were similarly calculated (6,7,8).

In calculating Ra-226 density concentrations, computer analyses of the gamma spectra were programmed to yield results for Ra-226, Bi-214 and Pb-214. The higher of these three values was interpreted to be the concentration of Ra-226 (in every instance, the value calculated for Ra-226 @ 186 kev was the highest, presumably because of disruption of equilibrium due to radon loss during sample collection and processing).

Removable Ra-226 contamination was calculated assuming that all detected alpha particles were solely attributable to Ra-226, again biasing the results high by a factor of as much as four depending upon degree of equilibrium at counting time. This bias may have been offset to some extent by self absorption, which could not be easily determined.

Tritium contamination levels or concentration amounts were calculated conservatively assuming a combined oxidation efficiency and condensate recovery in the low range of those empirically determined (between 60 and 70 percent). Quenching appeared to remain relatively constant and equal to that observed in calibration standards, regardless of sample type.

#### Waste Segregation and Disposal

After literally thousands and thousands of in situ surveys, samplings, laboratory analyses, instrument calibrations, quality assurance measurements, calculations, and several man-years of data reduction and review, all grids were considered to be radiologically characterized.

The final data describing each grid were compared to the release criteria. If all criteria were "met" (i.e., testing results demonstrated levels of contamination to be below criteria values), all materials from that specific grid were considered to be non-contaminated and were sent for disposal in a local sanitary landfill.

If the data indicated that one or more of the criteria were exceeded for a specific grid, either of two actions could be taken: decontamination and retesting or, designation of all materials comprising the grid as LSA waste. The decision of which pathway to follow was made after re-examining the data in considerable detail and determining the most cost effective mechanism for disposal.

Once the above-described determinations had been made, materials from several like (i.e., contaminated or non-contaminated) grids were segregated into "lots" pending packaging or loading for disposal. Materials declared contaminated were placed in LSA containers (some compacted, some not). The isotopic content of each container was conservatively estimated using the radiological data derived for each grid contributing to the contents of that

container. Final surveys and wipe tests were performed for each container in accordance with federal, state and disposal site requirements. Non-contaminated materials were loaded into standard semi-trailer "dumpsters" and after a final in situ survey (gamma exposure rate), hauled to a nearby landfill for disposal.

## CONCLUSIONS

After about seven months (April-September, 1985), and the expenditure of \$3.2 million, Phase II of the project was brought to a close. Examining the exceptionally large data base amassed during execution of this phase, it can be stated with a high level of certainty that the building materials at LPI were accurately characterized, effectively segregated, and properly disposed.

Furthermore, given the conservatisms built into the criteria and procedures, the health effects, if any, imposed upon the people of the Ottawa area due to Phase II activities were (and will continue to be) inconsequential.

## RECOMMENDATIONS

### Criteria

The ability to conclude the project with such positive statements is indeed professionally gratifying. It is also expensive; conservatism is not "free." Decontamination to the levels prescribed by the Phase II criteria resulted in a large volume of LSA radioactive waste. Disposal of this waste was not only fiscally expensive, it was also costly from a resource utilization perspective. The dollar amount assessed for waste disposal seems to be ever-increasing, while the available space for this type of disposal seems to be ever-decreasing.

Recognizing these limitations, IDNS elected nonetheless to use conservative (i.e., rehabilitation) criteria because no others existed that were universally accepted as assurance against potential radiological problems.

Appropriate and acceptable decontamination criteria need to be developed. These criteria should address the following issues:

- 1) the scope should be broad enough to encompass decommissioning of all types of radiologically contaminated facilities;
- 2) in addition to defining levels of contamination allowing rehabilitation, levels allowing disposal of materials as non-radioactive waste should also be described;
- 3) when contamination involves multiple isotope decay schemes, the criteria should explicitly state whether the limits apply solely to transitions of parent nuclei or to a summation of transitions of parent and progeny; and,
- 4) the relationship of the limits to ambient background should be explicitly stated, the limits should be defined to be either inclusive or exclusive of background and the method for determining background should be described.

The efficiency and economy of future decommissionings will depend upon the timely development of criteria addressing the above issues.

### Procedures

Since radiological characterization procedures are designed to test for "contamination" and, "contamination" is defined by criteria, procedures cannot

be developed until applicable criteria have been identified and adopted. Assuming such criteria are available, to what detail should procedures be developed to test for compliance with the criteria?

The answer depends upon how confident one wishes to be regarding the declaration of materials as "contaminated" or "non-contaminated." During Phase II, IDNS desired a high level of confidence that any materials declared "non-contaminated" were indeed so. All procedures were developed with this level of confidence in mind. This resulted in extensive and multiple surveys scanning of the entirety of the building's surface and sampling for laboratory analyses biased to yield maximum concentrations.

In retrospect, no less of an effort would have been acceptable. IDNS assured the citizens of the Ottawa area that any waste disposed locally had been determined, with confidence, to be non-contaminated. Future decommissioning activities contemplating the local disposal of non-contaminated waste should anticipate similar commitments. Given the price differential between disposing rad-waste vs. non-rad-waste, a considerable effort can be cost-effectively exerted toward confidently determining materials to be non-contaminated.

In summary, facility decontamination and decommissioning activities should be preceded by adoption of applicable and universally acceptable criteria. Procedures should then be devised to determine and document, with a high degree of confidence, whether decontamination to levels below those specified by the criteria has occurred.

#### REFERENCES

1. State of Illinois, Public Act 83-1103, "An Act Making Appropriations of State Funds", 1983.
2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.86 "Termination of Operating Licenses for Nuclear Reactors", 1974.
3. Code of Federal Regulations, 40 CFR 192, "Health and Environmental Protection Standards for Uranium Mill Tailings", 1980.
4. Code of Federal Regulations, 10 CFR 20.306, "Disposal of Specific Waste", 1981.
5. Illinois Department of Nuclear Safety, Request for Proposal, "Luminous Processes Cleanup", July 25, 1984.
6. Illinois Department of Nuclear Safety, Luminous Processes Inc., Decontamination and Decommissioning Project, "Phase II Standard Operating Procedures", 1985.
7. U.S. Environmental Protection Agency, EPA 520/1-80-012, "Upgrading Environmental Radiation Data", 1980.
8. U.S. Environmental Protection Agency, EPA 600/7-79-054, "Application of Germanium Detectors to Environmental Monitoring", 1979.
9. U.S. Environmental Protection Agency, EPA 520/5-84-006, "Eastern Environmental Radiation Facility, Radiochemistry Procedures Manual", 1984.

CORRELATION BETWEEN GAMMA RADIATION LEVELS  
AND SOIL RADIUM CONCENTRATIONS AT THE  
EDGEMONT URANIUM MILL SITE

R. G. Wallace  
Tennessee Valley Authority  
Western Area Radiological Laboratory  
Muscle Shoals, Alabama 35660

R. P. Reed and J. L. Polehn  
Tennessee Valley Authority  
5865 E Lookout Place  
Chattanooga, Tennessee 37402-2801

G. T. Wilson  
Silver King Mines, Inc.  
P. O. Box 49  
Edgemont, South Dakota 57735

ABSTRACT

The Tennessee Valley Authority's uranium mill in Edgemont, South Dakota is being decommissioned. Approximately 4 million tons of contaminated tailings, building equipment, and contaminated soil and debris on the mill site will be removed to the disposal site located approximately 3 kilometers to the southeast. To minimize recontamination of cleaned areas, tailings removal will progress from the northwest corner to the southeast corner of the mill site. As specific areas are cleaned, surveys will be conducted to determine if the concentrations of radium-226 in soil are within the limits outlined in 40 CFR, Part 192. Conformance with the criteria will be demonstrated by a gamma survey of the area employing the differential, or delta-measurement, technique. This technique involves fitting the detector with a base and a receptacle for a removable high-density filter. By making measurements with and without the filter in place, a gamma radiation level proportional to the radium-226 concentration in soil can be determined. This paper describes the results obtained in the development of the correlation between the gamma survey measurements and the soil radium concentrations.

INTRODUCTION

The Tennessee Valley Authority (TVA) purchased the Edgemont Uranium Mill in Edgemont, South Dakota in 1974 with plans to refurbish and operate the mill; however, with the cancellation of a number of nuclear power reactors, those plans were dropped. Subsequently, TVA agreed to decommission the mill and remove the tailings from the site. Approximately 4 million tons of tailings and contaminated soil and debris from the mill site will be removed to the disposal area located approximately 3 kilometers to the southeast. To minimize recontamination of cleaned areas, tailings removal will progress from the northwest corner to the southeast corner of the mill site. As

specific areas are cleaned, surveys will be conducted to determine if the concentrations of radium-226 in soil are within the limits outlined in 40 CFR, Part 192. In this phase of the project, determinations concerning the effectiveness of cleanup operations must be made in the presence of interfering radiation from surrounding uranium mill tailings. This paper describes the results of surveys conducted under these conditions in the summer of 1985 and the corresponding concentrations of radium-226 in soil samples taken from the surveyed areas. Proposed survey techniques for demonstrating compliance with the criteria of 40 CFR, Part 192 are also described.

## CRITERIA

Environmental Protection Agency regulations, as promulgated in 40 CFR, Part 192.32, require the removal of radioactive materials so that remaining concentrations of radium-226, when averaged over any area of 100 square meters, will not exceed (a) 5 pCi/g in the first 15 centimeters of soil, and (b) 15 pCi/g in subsequent 15-centimeter intervals. The study described in this report is designed to establish the relationship between the radium-226 concentrations in soil on the mill site and gamma radiation level measurements. This should permit the "clearing" of a significant portion of the mill site area for compliance with the regulations by means of gamma surveys or rapid onsite soil sample analyses. This approach is designed to reduce the number of the more time-consuming and costly soil sample analyses by an offsite laboratory. A large portion of the site will be excavated and subsequently covered with clean material (that is material containing less than 5 pCi/g radium-226). For these areas, the criterion of 15 pCi/g radium-226 will be applicable; therefore, for this study, emphasis is placed on areas where the soil radium concentration is expected to be between 3 and 20 pCi/g.

## EQUIPMENT

In order to optimize costs, the study was performed utilizing equipment available at the Edgemont site. Consequently, the following instrumentation was employed in the development of the survey procedures:

1. Reuter Stokes, Model RSS 111 Pressurized Ionization Chamber (PIC).
2. Ludlum, Model 19  $\mu$ R Meters (2 each).
3. Ludlum, Model 15 Scaler with a Model 44-10 2" x 2" NaI Gamma Scintillation Detector.
4. Ludlum, Model 2200 Portable Scaler Rate Meter with a Model 44-10 2" x 2" NaI Gamma Scintillation Detector.

The PIC was used as a comparison standard for the  $\mu$ R meters. An initial instrument response test was conducted by exposing the instruments to a 1.039 mg, radium-226 source at various distances. The results are shown in table 1.

Table 1. Calibration of the Pressurized Ionization Chamber and  $\mu$ R Meters

Source-to detector distance, feet	Calculated Exposure rate, $\mu$ R/hr	Measured exposure rate (gross rate less background), $\mu$ R/hr		
		PIC	$\mu$ R meter 15600	$\mu$ R meter 15610
50	3.65	3.2	3.5	3.0
40	5.70	4.5	6.0	6.0
30	10.13	9.8	11.0	9.5
25	14.59	14.8	14.5	13.0
20	22.80	23.2	23.5	21.0
15	40.52	41.4	50.5	48.0
10	91.18	91.4	97.5	86.0

The three instruments showed good general agreement with the calculated exposure rate, and the  $\mu$ R meter readings were consistent with the PIC measurements.

The scintillation detectors were adapted for measuring radium with the delta measurement technique by providing a mechanism for mounting a removable lead filter to the base of the detector. In employing this technique, the detector is placed in a reproducible geometry on or near the ground, and consecutive measurements are made with and without the filter in place between the detector and the sample area. The measurement made without the filter in place will include radiation from the sample beneath the detector and from the surrounding area. The measurement made with the filter in place includes radiation from the surrounding area, with only a fraction of the radiation from the sample penetrating the filter. If a sample is collected from the area directly beneath the detector and analyzed for radium-226 concentration, the difference between the two survey measurements can be interpreted in terms of an equivalent radium-226 concentration. In order to reduce the influence from deposits of mill tailings near the measurement sites, the detectors were surrounded by approximately 5/8 inch of lead, in essence collimating the detectors. This shield has the effect of reducing the background by approximately a factor of four.

The model 2200 portable scaler rate meter with a 2" x 2" NaI detector was used to count the samples in the onsite laboratory. A shield of approximately 3/4 inch of lead with a cavity of approximately 6" x 6" x 10" was constructed for counting the samples. The counting window was adjusted to count all radiation between approximately 300 and 700 keV.

#### SURVEY PROCEDURES

In June 1985, approximately 20 survey points were selected at locations which were influenced by the surrounding tailings and suspected of containing radium-226 concentrations in the general range of 3-20 pCi/g. At each site the radiation level was determined with an unshielded  $\mu$ R meter, and a delta measurement was made with each of the four instruments. For the delta measurements, each instrument was placed on the ground at the designated sample area, and a measurement made with and without the filter in place. After

all measurements were completed, a sample of the soil directly beneath the detector location was collected to a depth of approximately 2 inches. The soil samples were returned to the onsite laboratory, weighed, and counted with the model 2200 system. Following this analysis, the samples were forwarded to TVA's radioanalytical laboratory in Muscle Shoals, Alabama. Here each sample was dried, pulverized, and sealed in an airtight container. The samples were counted for radium-226 content after a 28-day ingrowth period. Preliminary estimates of the radium-226 concentration may be determined by counting a sample at any time after sealing the container. The concentrations of total uranium and potassium-40 were also determined along with estimates of any significant thorium content.

An additional 20 survey measurements and corresponding soil samples were taken at Edgemont in September 1985. These data will serve as additional data points in establishing the procedures for determining equivalent radium-226.

## RESULTS

The results from the June field survey measurements and the laboratory analyses are presented in table 2, while the results from the analysis of the September samples are shown in table 3. The onsite laboratory measurement is the count rate, normalized to a 1000 gram wet weight sample, determined by counting the sample with the Ludlum Model 2200 system at the

Table 2. Field Survey Results and Radium-226 Concentrations  
Determined for the June 1985 Survey (Data Set 1)

Sample Location	Delta Measurement <sup>3</sup>				Onsite Laboratory Measurement cpm/1000g	Radium-226 Concentration, pCi/g	
	15600 µR/hr	15610 µR/hr	Model 15 CPM	Model 2200 CPM		Measured	Predicted
1	4	5	1000	408	152	3.89	4.5
2	6.5	5	5500	543	215	7.97	6.2
3	11	11	9500	1329	191	6.70	5.6
4	11	11	11000	1243	846	34.67	22.8
5	18	14	16000	1826	1432	52.88	38.3
6	4.5	4	4700	393	163	2.43	4.8
7	4	5	5100	324	167	3.24	4.9
8	10.5	11	12500	1985	732	16.50	19.8
9	16	12	15000	1658	877	32.26	23.6
10	6.5	7.5	6500	948	652	17.37	17.7
11	6	6	7000	878	567	12.72	15.5
12	12	9.5	11000	1564	769	26.38	20.8
13	12	10	10000	1501	1021	29.58	27.4
14	4.5	3	4400	622	244	5.56	7.0
15	17	13	16000	2271	2509	74.13	66.6
16	4.5	5.5	5400	579	244	5.44	7.0
17	9	2.5	5500	350	163	1.85	4.8
18	8.5	6	6000	560	463	12.40	12.7
19	8	6	8000	392	278	5.96	7.9
20	11	11	9500	1756	1142	35.61	30.6
21	4	4	4100	467	217	4.17	6.2
22	4.5	6	4800	733	353	10.28	9.8

Table 3. Field Survey Results and Radium-226 Concentrations  
Determined for the September 1985 Survey (Data Set 2)

Sample Location	Delta Measurement <sup>3</sup>				Onsite Laboratory Measurement, cpm/1000g	Radium-226 Concentration, pCi/g	
	15600 μR/hr	15610 μR/hr	Model 15 CPM	Model 2200 CPM		Measured	Predicted
2A	8	8	9000	9028	550	23.18	15.0
2B	10	10	10500	12612	604	25.39	16.4
10A	6	5	8000	1062	649	19.25	17.6
10B	4	4	6000	701	316	10.79	8.9
10C	3	4	5200	741	303	8.68	8.5
11A	4	3	3600	757	172	6.31	5.1
11B	4	4	4400	682	266	7.62	7.5
11C	7	6	7300	1599	519	20.85	14.2
11D	6	3	5500	635	236	6.95	6.7
11E	6	6	6500	1062	433	14.39	11.9
12A	12	8	12000	2549	1262	61.21	33.8
12B	11	7	11000	1880	1311	59.31	35.1
14A	16	12	16000	3303	2030	71.20	54.0
14B	12	9	12000	2465	1369	75.96	36.6
18A	8	5	7000	8085	162	3.29	4.8
18B	12	7	10000	12277	432	13.77	11.9
22A	5	5	6400	6222	369	15.77	10.2
22B	8	6	6000	8824	672	25.66	18.2
22C	7	6	7000	8582	639	27.01	17.4
22D	4	3	4100	4492	201	10.16	5.8

Edgemont laboratory. The measured radium-226 concentrations are the values determined by analysis of the samples at TVA's radioanalytical laboratory.

The data were analyzed in two ways. For the delta measurements, a frequency table presenting the frequency of occurrence of radium concentrations for given delta measurements for each instrument used was constructed. For the μR meters, table 4 indicates that for delta measurements less than 6 μR/hr, only one radium-226 concentration, or about 3 percent of the measurements, was greater than 15 pCi/g. Therefore, this value represents a cutoff point for determining if the radium-226 concentration is greater or less than 15 pCi/g. Similar tabulations produce cutoff points of 6000 and 800 cpm for the Ludlum Model 15 and Ludlum Model 2200, respectively. However, a number of delta measurements on the lower end of the scale would predict a radium-226 concentration below 5 pCi/g when the measured value was in reality above 5 pCi/g. Therefore, although the delta technique may be used for areas being cleared at the 15 pCi/g criterion, another technique, such as soil sampling, will be required in areas for which the 5 pCi/g criterion is applicable.

The count rate from the onsite laboratory measurements are plotted against the measured radium-226 concentrations in figure 1. The subset of data in the general range of the cleanup criteria, i.e., with radium-226 concentrations less than 20 pCi/g, are plotted in figure 2. For this subset of data, a linear regression analysis of the count rates against the measured radium-226 concentrations yielded the equation,

$$y = .5293 + .0263x$$

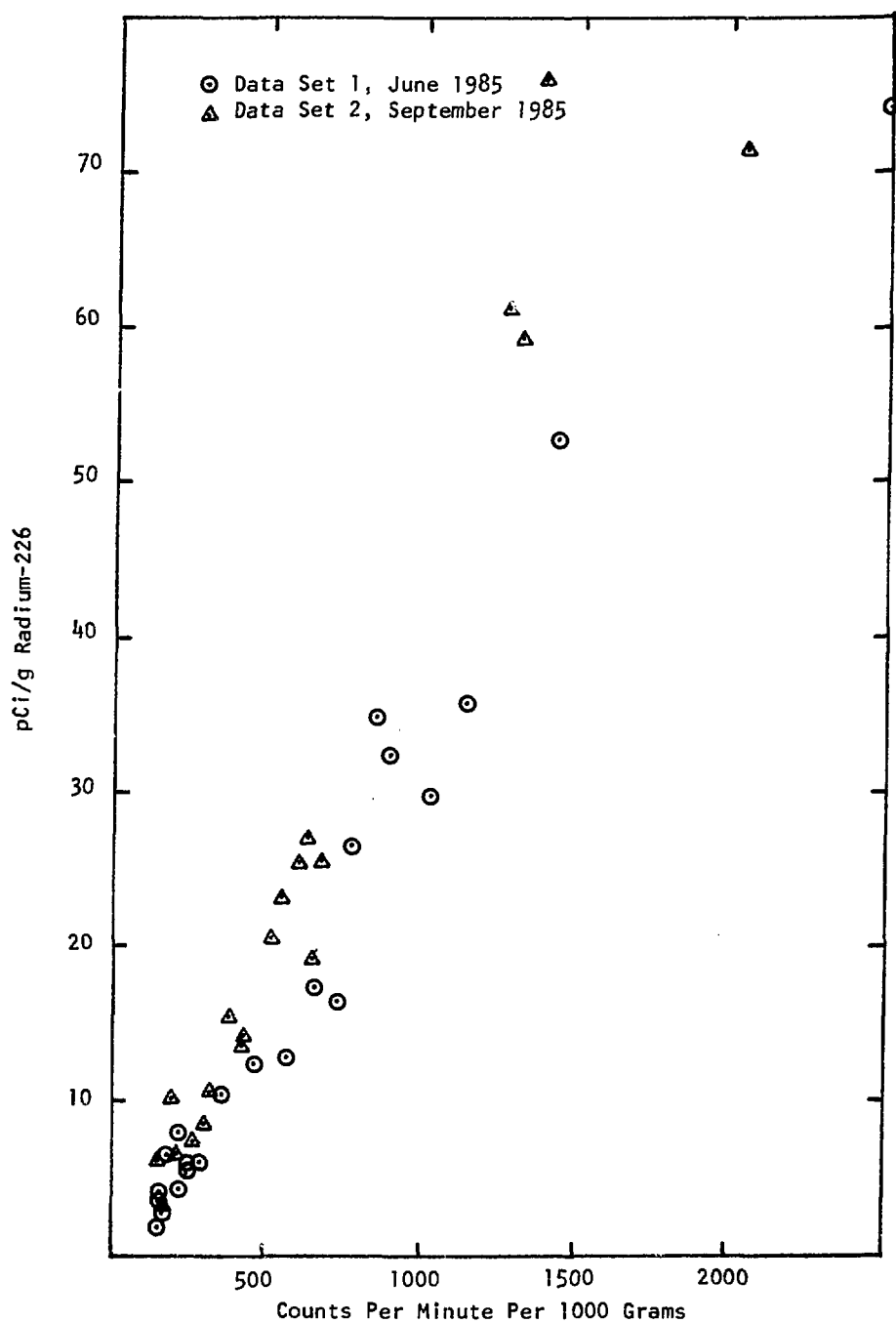


Figure 1. Net Counts Per Minute Per 1000 Gram Sample  
Vs pCi/g Radium-226.

Table 4. Frequency of Occurrence of Radium-226 Concentrations for Various Delta Measurements made with the Ludlum Model 19  $\mu$ R Meters - Number of Occurrences

Radium-226 Concentration, pCi/g	Sample Set*	Delta Measurements, $\mu$ R/hr			
		0 - 4.4	4.5 - 5.9	6 - 12	>12
0 - 4.9	1	6	3	1	0
	2	1	1	1	0
5 - 10.9	1	1	5	6	0
	2	11	0	1	0
11 - 14.9	1	0	0	4	0
	2	0	2	3	1
15 +	1	0	0	8	10
	2	0	1	15	4

\*Set 1 samples collected in June 1985 and Set 2 samples collected in September 1985.

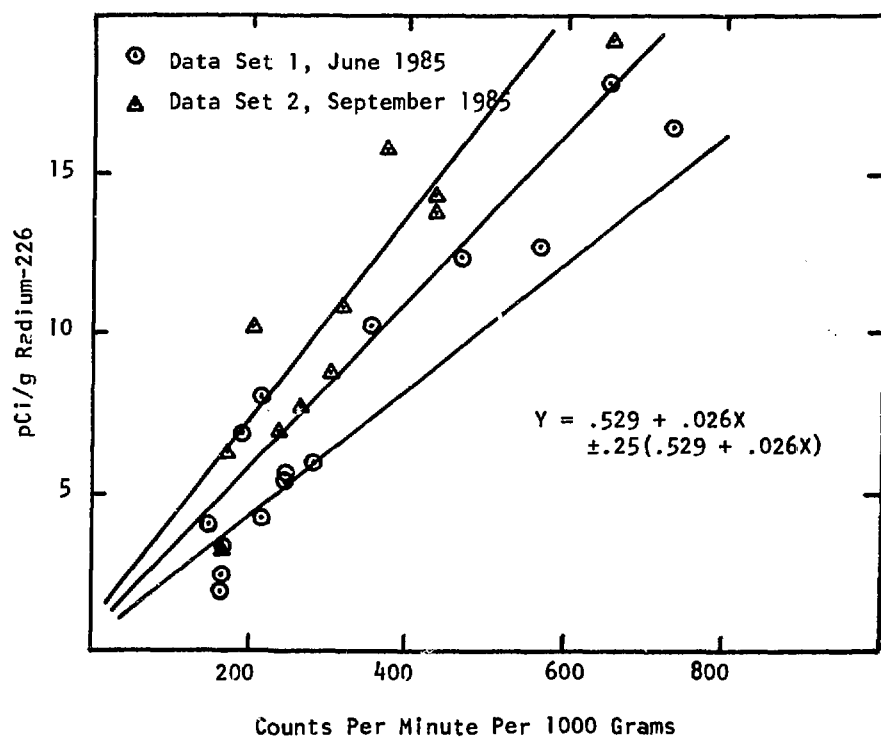


Figure 2. Net Counts Per Minute Per 1000 Gram Sample Vs. pCi/g Radium-226 for Radium-226 Concentrations Less than 20 pCi/g.

with a coefficient of correlation,  $r$ , of 0.90, where  $y$  is the radium-226 concentration for a corresponding count rate,  $x$  (normalized to a 1000 gram sample). Substituting the onsite laboratory measurements from tables 2 and 3 into the generalized equation produces the predicted radium-226 concentrations in the last column of these two tables. For the higher concentrations the predicted values are only about 65 percent of the measured values. However, for radium-226 concentrations below 20 pCi/g, the predicted values provide a reasonable approximation of the measured concentrations. Figure 2 also depicts the line defined by the regression equation and the lines representing values plus or minus 25 percent of the reference line. Nearly 90 percent of the measurements are less than plus 25 percent of the predicted value.

#### PROPOSED CLEARANCE PROCEDURES

The following procedures are proposed for use in determining compliance with 40 CFR, Part 192.

Each area to be cleared shall be marked off on a 10-meter grid. Each 100-square meter area will be surveyed with one of the instruments described in this report, with the model 19  $\mu$ R meter being the primary instrument. If the area is one in which contaminated surface material has been removed and uncontaminated cover material is to be added to level the site, the criterion of 15 pCi/g of radium-226 in the top 15 cm of soil in each 100 square meter area shall apply. If the area is undisturbed surface material and/or no cover material is to be utilized, the criterion of 5 pCi/g shall apply. Each area shall be surveyed and a delta measurement made at the point with the highest reading in each area. If the delta value is less than the value corresponding to the 15 pCi/g concentration as indicated by the appropriate frequency table, the area shall be considered clean. If the delta is greater than the cut-off value, or for areas in which the 5 pCi/g criteria is applicable, a soil sample shall be collected consisting of a composite of 10-20 aliquots over the 100 square meter area. The samples shall be taken to a depth of 15 cm, bagged, returned to the onsite laboratory, weighed, and counted with the model 2200/44-10 system. The regression equation described above shall be solved for  $y$  (the radium-226 concentration in pCi/g) for each measured value of  $x$  (count rate per 1000 gram sample). If the calculated radium-226 concentration is less than 4 or 12 pCi/g (as appropriate), the area shall be considered clean (4 and 12 pCi/g + 25 percent are 5 and 15 pCi/g, respectively). If the value is between 4 and 5 pCi/g or between 12 and 15 pCi/g, an additional sample will be collected from the corresponding 100 m<sup>2</sup> area. If the result from the analysis of this sample is less than 5 or 15 pCi/g, the area will be considered clean. If the calculated concentration exceeds the applicable criteria, additional material shall be removed, and the above process repeated, or if time permits, the sample may be submitted to TVA's radioanalytical laboratory for a more precise determination of the radium-226 concentration.

## CONCLUSIONS

A correlation has been demonstrated between gamma survey measurements and the concentration of radium-226 in soil for the Edgemont Uranium Mill site. For areas in which the 15 pCi/g criterion is applicable, the delta measurement technique may be used by applying the cutoff values determined in this study. For areas in which this technique indicates a concentration of about 15 pCi/g, the analysis of a soil sample will be required to determine the radium-226 concentration. Analysis of soil samples will also be required in areas in which the 5 pCi/g criterion is applicable. A relationship between the count rate from the onsite analysis of soil samples and the measured radium-226 concentrations was established. This will allow the use of the onsite laboratory for rapidly determining the radium-226 concentration in a significant number of the samples.

ANALYSIS OF URANIUM AND THORIUM CONTAMINATED SOILS  
DURING A CLEANUP AND INTERMENT OPERATION

Leslie W. Cole  
Michael Bollenbacher  
Aerojet Heavy Metals Company  
Jonesborough, Tennessee

ABSTRACT

At the Aerojet Heavy Metals Company facility near Jonesborough, Tennessee, a remedial cleanup operation was completed during September, 1985. This operation involved burying approximately 550,000 ft<sup>3</sup> of soil contaminated with uranium and thorium. Approximately 120,000 ft<sup>2</sup> of area was cleared of the contaminated soil. To maintain a reasonable construction pace, it was necessary to complete the analysis very rapidly. Only one gamma spectroscopy system was available to perform the analysis.

This paper discussed the procedure followed during the remedial operation. Counting times were kept short by using the largest sample mass in the chosen sample container, having a relatively high efficiency detector, and standardizing on the low energy x-rays of Thorium-234 and Protactinium-234. Counting accuracy of + 20% at the 95% confidence level was achieved in counting times of 500 to 1,000 seconds for levels of 30 pCi/g of Uranium-238 and 5 pCi/g of Thorium-232. The analytical procedure was able to keep up with the construction effort.

**IDAHO NATIONAL ENGINEERING LABORATORY RELEASE CRITERIA  
FOR DECONTAMINATION AND DECOMMISSIONING**

Max R. Dolenc and Marilyn J. Case  
EG&G Idaho, Inc.  
P. O. Box 1625  
Idaho Falls, ID 83415

**ABSTRACT**

Criteria have been developed for release of Idaho National Engineering Laboratory (INEL) facilities and land areas following decontamination and decommissioning (D&D). Decommissioning release criteria in the form of dose guidelines were proposed by the U.S. Nuclear Regulatory Commission as early as 1980. These criteria were used on an interim basis for INEL D&D projects.

However, dose guidelines alone do not adequately cover the criteria necessary to release sites for unrestricted use. In actual practice, other parameters such as pathways analyses, sampling and instrumentation techniques, and implementation procedures are required to develop the basis for unrestricted release of a site. Thus, a rigorous approach for evaluating these other parameters is needed to develop acceptable D&D release criteria. Because of the complex and sensitive nature of the dose and pathways analyses work, a thorough review by experts in those respective fields was desired. Input and support in preparing or reviewing each part of the criteria development task was solicited from several DOE field offices. Experts were identified and contracted to assist in preparing portions of the release criteria, or to serve on a peer-review committee. Thus, the entire release criteria development task was thoroughly reviewed by recognized experts from each DOE field office, to validate technical content of the INEL site-specific document.

## DECONTAMINATION OF AN AMERICIUM-241 SITE

G. L. Murphy and E. J. Deming  
Oak Ridge Associated Universities  
P. O. Box 117  
Oak Ridge, Tennessee 37831-0117

R. J. Caniano  
U. S. Nuclear Regulatory Commission  
Region III  
799 Roosevelt Road  
Glen Ellyn, Illinois 60137

T. R. Emswiler and G. E. Kirsch  
Battelle Columbus Division  
Building JN-2  
505 King Avenue  
Columbus, Ohio 43201-2693

### ABSTRACT

A rural residence was modified for operation as a hot lab with glove box capabilities designed to handle up to 25 Ci of Americium-241 in the powder oxide form. The facility operated under an NRC license for over ten years, when the license was amended for storage of contaminated materials only. A 1983 site survey conducted by Oak Ridge Associated Universities Radiological Site Assessment Program (ORAU/RSAP) indicated radioactive contamination present in the environment (soils and groundwater) and in unrestricted areas of the house and unattached garage. The site was declared an imminent hazard by U.S. Nuclear Regulatory Commission (USNRC III), and the facility was closed pending decontamination under the Environmental Protection Agency (EPA) "superfund".

This paper will demonstrate that a cooperative effort between USNRC (III), EPA, Department of Energy (DOE), Battelle-Columbus Laboratory, and ORAU/RSAP resulted in an efficient and effective decontamination of the site. The co-operative effort enabled many problems to be overcome, and resulted in the clean-up being completed under budget and on schedule.

## INVENTORY OF SITES USED TO DEVELOP RESIDUAL RADIOACTIVITY CRITERIA

Melinda Ronca-Battista  
U.S. Environmental Protection Agency  
Washington, D.C. 20460  
and  
James C. Hardeman, Jr.  
Environmental Protection Division  
Georgia Department of Natural Resources

### ABSTRACT

The U.S. Environmental Protection Agency (EPA), in conjunction with the National Conference of Radiation Control Program Directors (CRCPD), has compiled an inventory of never licensed or otherwise poorly documented sites that may be contaminated with radioactive materials. This effort is in support of the EPA's development of radiation protection criteria for residual radioactivity at decommissioned sites. The inventory will help to establish the range of circumstances for which criteria are needed, as well as the suitability of candidate criteria for actual situations. The information will also be used to develop model sites and facilities for analyzing technical and economic feasibility of residual radioactivity criteria and to assess costs and benefits of alternate criteria. Relevant information about each site, such as radionuclides, waste forms, and quantities present will be included in the inventory when such information is available. The CRCPD has requested that each State radiation control agency furnish the information for the inventory. The inventory supplements the relatively extensive documentation of sites regulated by Federal or State agencies with information on old or unlicensed sites, such as old waste storage sites or radium ore processing facilities.

### INTRODUCTION

The EPA is developing criteria for cleanup of residual radioactivity on decommissioned lands and facilities. The purpose of these criteria is to assure the protection of public health and the environment after facilities and land are returned to public or private use, free from restrictions based on radioactivity. To support the development of these criteria, the EPA needs information on the full range of site types to which criteria would apply. These include sites owned, operated, or licensed by Federal or State governments as well as the private sector. For a complete description of the site types and the development of EPA residual radioactivity criteria, see "Development of Residual Radioactivity Criteria", by Stanley Lichtman and Melinda Ronca-Battista, elsewhere in these proceedings.

EPA will rely upon the extensive documentation of many Federal or State operated or licensed facilities as a primary resource during the development of the criteria. However, there are also an uncertain number of old, never licensed, or otherwise poorly documented sites for which cleanup criteria may be needed. To help EPA obtain information on the full range of contaminated sites, the EPA requested the National Conference of Radiation Control Program

Directors (CRCPD) to gather the available information on this particular class of sites. To limit the information to that which is specifically useful to this EPA project, we did not request information on sites contaminated with high volume mining wastes, such as phosphate and gypsum piles, and other high volume, low concentration wastes with specific activities less than 15 pCi/g such as coal ash. Management and disposal of these wastes are being studied under EPA's Resource Conservation and Recovery Act program, and will not be covered by these criteria.

The information for the inventory was gathered from February to September, 1985, by the CRCPD Committee on Decommissioning and Decontamination. The CRCPD sent a questionnaire to each State Radiation Control Program Director, and EPA compiled the results into the inventory. This information supplemented previous information gathered by the EPA Regional Radiation Representatives in 1983. The preliminary analysis of site categories presented here was developed from the final inventory which comprises information gathered through both efforts.

A total of 37 States responded to our questionnaire, and the inventory now contains brief descriptions of 90 sites. Since we have not received information from all States, we cannot make conclusions about the relative proportions of the different site categories or the absolute number of sites. Although we have tried to eliminate licensed sites from the inventory, some of the sites in the inventory are or have been licensed, and therefore conform to all the documentation requirements of their licensor. In general, however, they are relatively poorly documented.

The EPA has sorted 90 the sites into four general categories, based on site descriptions in the inventory. This paper gives preliminary results of the survey in terms of these categories.

The four categories are: a) small waste burial sites, b) radium processing facilities; c) sites contaminated as a result of military research or testing activities; and d) ore processing sites. Some sites could be classified into more than one category. The four site categories are described below, mainly in terms of characteristics of typical sites.

#### 1. Small Waste Burial Sites

This category includes land where relatively small amounts of low-level radioactive waste were buried, such as at university, research or other laboratory waste burial sites. These wastes include disposable trash, scintillation fluids and vials, radioactive solutions, animal carcasses, and some industrial processing wastes. Twenty-six such sites were reported in the inventory. Many similar small burial areas associated with larger well-documented facilities were not reported here because of our specific request for information on poorly-documented or unlicensed sites.

Typically, these small waste burial sites conform to the regulations for disposal of radioactive waste in soil by a licensee on its own property specified by the NRC in 10 CFR Part 20.304 before changes instituted in 1981. These regulations allowed specified amounts of radionuclides to be disposed of at specified annual rates, at a minimum depth of four feet and with successive burials separated by distances of at least six feet.

Most of the small low level waste burial sites are located near large urban areas. Typical radionuclide inventories reported include mCi amounts of C-14, H-3, Co-60, Cs-137, Tc-99, Ra-226, and Am-241. The sizes of sites reported ranged from three square meters to waste ponds covering hundreds of square meters. Reported amounts of radionuclides ranged from 1 mCi of C-14 to curie amounts of H-3 and Co-60. Many sites in this category contain fewer radioisotopes and some larger quantities of isotopes. Some burial areas resulted from accident, rather than design, when land became contaminated by materials leaked from storage containers, or buried without knowledge of its radioactivity.

## 2. Radium Processing Facilities

The inventory includes 20 sites and facilities reported where radium was processed or used. The most common such facilities in this class are buildings that housed radium dial painting activities, and land associated with or immediately surrounding such facilities. Others include places where, in the 1920's and 1930's, devices containing radium and radon were manufactured for cancer therapy or as general health aids.

A typical radium processing facility is a multistory commercial structure that housed radium watch dial painting activities during World War I. The walls, floors, and sewers of the structure are extensively contaminated with radium. In some areas of the structure, the direct gamma exposure levels are several times higher than background levels, but the major health risk is from inhalation of radon. Some facilities also include contaminated soil adjacent to building structures. Half of the reported sites were buildings that old company records suggest might have housed radium dial painting activities, but that have not yet been surveyed to determine whether they are contaminated.

This category of facilities is fairly homogenous. The major differences between sites are whether the sites have been investigated and surveyed and whether there are any controls on the use of the facility. Several facilities reported are being decontaminated, using Superfund or State monies.

## 3. Experimentation Sites

This category includes sites where radiation experiments were conducted, generally for the military. Most experimental military sites were not included in this inventory because they are well documented; these include atomic testing ranges, and other military sites that are under administrative controls. The four sites reported to us in this category are relatively small, and have not been used since before the 1970's.

A typical site has 500 acres of land located in a rural area. It was used in the 1960's for various weapons systems tests, and tests of the effects of radiation on military equipment. The top several meters of a small area within the site are contaminated with fission products. Tritium has leached from this area and contaminated the groundwater. There are two small buildings on the site contaminated with mCi amounts of Co-60 and Cs-137. The soil for several meters around these buildings is also contaminated with these radionuclides. The site is not presently used, is fenced off and posted, and groundwater is periodically monitored.

#### 4. Ore Processing, Storage, or Mining Sites

We placed 40 sites in the inventory in this category, after attempting to eliminate sites that are covered by the Uranium Mill Tailing Radiation Control Act (UMTRCA), or are currently under a large-scale investigation and characterization effort by a State or Federal agency. The remainder are seven abandoned uranium mills, 14 abandoned surface uranium mines, five uranium ore buying stations, four abandoned vanadium mills, four areas where slag has been used as fill or for paving and construction purposes, three sites where zirconium mining wastes were deposited, two inactive phosphogypsum piles, and one old thorium oxide production facility. We will investigate these sites further to verify whether they are covered by UMTRCA. These sites were included in the inventory for completeness, but the EPA does not intend to develop residual radioactivity criteria for large volume mining wastes.

The major problems reported for these sites were exposure to radon and its decay products in homes built on processed ore, and contamination of groundwater. There was a wide variety of site sizes reported, from small abandoned ore buying stations less than one acre in area, to large sites covering hundreds of acres. Most are located in rural areas, except for the areas where ore slag was used for construction purposes.

#### CONCLUSIONS

The inventory provides an overview of the types of sites that, in general, are less well documented than most other types of sites. Four categories adequately describe all 90 sites: a) 26 small waste burial sites; b) 20 radium processing facilities; c) 4 sites contaminated as a result of military research or testing activities; and d) 40 ore processing facilities. The EPA will expand the descriptions of these site categories to create generic sites, using existing information on typical well-characterized sites. Use of these generic sites, in combination with the extensive body of literature developed on decommissioning, will help to ensure that EPA's criteria for residual radioactivity following cleanup of lands and facilities will be appropriate for existing and future decommissioning needs.

## DATABASE CONSTRUCTION FOR ASSESSMENT OF CHRONIC RADIONUCLIDE RELEASES\*

L. M. Hively<sup>+</sup>, A. L. Sjöreen<sup>#</sup>, and J. L. Bledsoe<sup>#</sup>  
Oak Ridge National Laboratory  
P. O. Box X  
Oak Ridge, Tennessee 37831

### ABSTRACT

Oak Ridge National Laboratory (ORNL) has developed a comprehensive dose/risk assessment methodology, CRRIS (Computerized Radiological Risk Investigation System), for assessment of atmospheric radionuclide releases (e.g., during decontamination and decommissioning). Radiological effects are calculated for direct atmospheric and ground exposure and for consumption of contaminated agricultural products. Previously, population and agricultural data for CRRIS were tabulated on 1/2-degree square grid cells. This coarse resolution resulted in anomalies such as population in water bodies, agricultural production in city centers, and nonconservation of population over the assessment grid. An even finer mesh (2-minute square grid) is inadequate in densely populated cities where 20,000 or more people live in a square block. These difficulties are overcome by a higher resolution technique using U.S. Census Bureau population data at the most detailed [enumeration district (ED)] level available and by excluding population from unpopulated areas (e.g., water bodies). Agricultural data are available at only the county level and are apportioned to the ED grid according to farm population, conserving the original data. This paper discusses the database construction using Tennessee as an illustration.

### INTRODUCTION

ORNL has developed the CRRIS suite of models to calculate the atmospheric transport of radionuclides and daughters, doses, and risks (1-2). Within 100 km of the source, air concentrations and ground deposition rates are determined from a straight-line, Gaussian-plume atmospheric dispersion model, which includes considerations for plume reflections from the ground and the atmospheric mixing layer, along with particulate size distributions, meteorological data, and release geometry. Beyond 100 km, a trajectory atmospheric dispersion model coupled with upper-air wind data is used. In both cases, long-term (>1 year) exposures are obtained. Radionuclide buildup on the ground surface and in the root zone is used to calculate uptake into food and feed crops; transport to beef and cow's milk from ingested feed is then calculated from feed concentrations. Radiological doses and risks are obtained from population-averaged or individual intake and usage rates.

-----  
\*Research sponsored by the Office of Radiation Programs, U.S. Environmental Protection Agency (EPA), under Interagency Agreement DW89930857-01-1 under Martin Marietta Energy Systems, Inc., contract DE-AC05-84OR21400 with the U.S. Department of Energy. This is the text of a presentation and does not necessarily reflect EPA policy.

<sup>+</sup>Health and Safety Research Division.

<sup>#</sup>Computing and Telecommunications Division.

By acceptance of this article, the publisher or recipient acknowledges the U.S. Government's right to retain a nonexclusive, royalty-free license in and to any copyright covering the article.

Previously, population and agricultural data for CRRIS assessments were tabulated on 1/2-degree square grid cells (3). This coarse resolution resulted in anomalies such as population centers lying in water bodies, agricultural production occurring in city centers, and nonconservation of population over the assessment grid (i.e., the sum of population over the grid cells was not equal to the total original population). Even a much finer mesh (2-minute square grid cells) is inadequate within densely populated cities, where 20,000 or more live in one square block (about 0.3-minute square area). A higher resolution technique overcomes these problems by using U.S. Census Bureau data at the most detailed level, the enumeration districts (ED). Population is excluded from unpopulated areas such as lakes and rivers; such regions are called exclusion areas. Agricultural data are available only at the county level; for this database, such data are apportioned to the ED grid in according to farm population within that ED.

The remainder of this paper discusses the database construction and data interpolation for the CRRIS system (Fig. 1). The second section describes the data (population, agriculture, hydrology, and geography), the checks to ensure its consistency, and the construction of the combined population and agricultural database for the United States. The third section discusses the apportionment of this data from the ED grid, via Thiessen polygons, to the assessment grid, including county and water exclusion boundaries; results are illustrated for Tennessee. The fourth section is a description of CRRIS system assessments using the database. Our conclusions are presented in the last section.

## DATABASES

Five kinds of information were required for this database construction. Because all the data were not available from one source, considerable effort was required to locate all the data.

### Data Acquisition

The U.S. Census Bureau (4) is the source of most of the data, including:

1. Population data at the ED level [Master Area Reference File (MARF)] from the 1980 census,
2. Farm population data at the ED level (Summary Tape File 3A, Table 7) from the 1980 census,
3. County boundary data for the entire United States, consistent with the ED locations (County Boundary File--1980),
4. Agricultural data at the county level (Census of Agriculture, 1974, Final County File). These data include vegetable production (leafy vegetables; fruit and vegetables exposed to airborne radionuclides; fruits, nuts, and vegetables protected from airborne contamination; and grains), livestock feeds (hay, silage, grains, and pasture production), and nonvegetable food production (milk, beef, pork, lamb, poultry, eggs, goat milk, and honey). See Ref. 5 for a description of the agricultural database methodology.

Population data are collected by the U.S. Census Bureau every ten years at various levels of detail. The smallest unit is a "census block," which is defined only in urban areas as representing a single block; population data are not made available to the public at this level. The most detailed level of spatial resolution available from the census is the block group, corresponding to one city block in an urban area. In less densely populated locations (e.g., rural regions), the corresponding area is an ED, representing an area which can

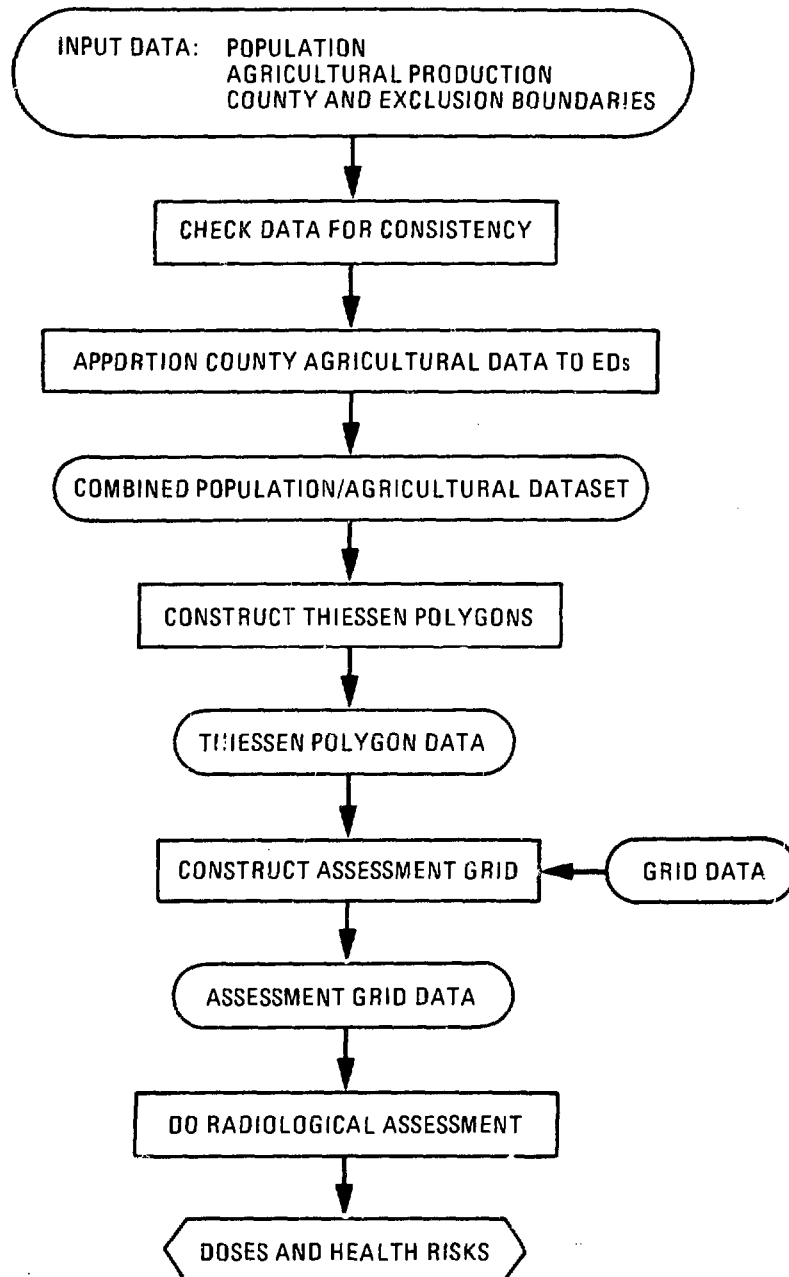


Figure 1. Flowchart for database construction and use.

be covered by one census enumerator. There are roughly 300,000 EDs and block groups in the entire United States, typically containing 500-1000 people. Block groups and EDs will be referenced collectively as EDs in the remaining discussion. The next largest area is a census tract, followed by a minor civil division (called a census civil division in rural areas); the subsequently higher levels are a county, a state, and then the entire country. While the most detailed population information is available at the ED level, the ED boundaries have not been digitized by the U.S. Census Bureau. Rather, the ED location is characterized by the population-weighted centroid in latitude/longitude form. The manipulation of this census information on the ED grid is discussed in the third section.

The agricultural data for CRRIS assessments are based on the 1974 Census of Agriculture (6). The data for exclusion boundaries were obtained from the U.S. National Cartographic Information Center (7), and consist of boundary data (e.g., federal parks and forests) and hydrology data (e.g., rivers, lakes, streams, and reservoirs) developed as a part of the National Digital Atlas Series by the U.S. Geological Survey. Location information in all these data sets is given in latitude and longitude. Before these data were combined into the database, they were checked for consistency. A description of this procedure follows.

#### Consistency Checks of the Data

Two consistency checks of the U.S. Census data were used here, at three levels of information: state, county, and ED. These checks consisted of verifying that:

1. The sum of the populations of the EDs within a county equals the county population.
2. The sum of the populations of the counties within a state equals the state population.

The 1980 U.S. Census data were consistent according to these checks, except that the sum of the county populations was unequal to the state populations for California (23,667,902 vs 23,667,904) and New York (17,558,072 vs 17,558,080); these differences appear to be only typographical. A summary of this population data for the entire United States is shown in Table 1. There are several interesting features of the data in Table 1:

- \* The number of counties in one state ranges from 1 (District of Columbia) to 254 (Texas).
- \* The number of EDs in a state ranges from 572 (Vermont) to 24,082 (California).
- \* The number of EDs in one county ranges from 1 (Montana) to 6,507 (California).
- \* The population in an ED ranges from 0 (all states have zero-population EDs) to 26,454 (New York).
- \* The number of zero-population EDs in a state ranges from 16 (Vermont) to 1,930 (Texas). These zero-population EDs are present to provide complete area coverage for the census, indicating areas where no one lives.
- \* The population in a county ranges from 91 (Texas) to 7,477,503 (California).

The other verifications that are made in this methodology are as follows:

- \* All ED centroids lie inside the correct county and state.
- \* Exclusion areas lie inside the proper county and state.
- \* No EDs lie within exclusion areas.

Table 1 Summary of ED Data for Entire United States (see text)

					Max EDs Per Co	Min EDs Per Co	Avg EDs Per Co	Max Pop Per ED	Num Zero Pop EDs	Avg Pop Per ED	Max Pop Per Co	Min Pop Per Co	Avg Pop Per Co
State	Tot Pop	#Co	#EDs		Co	Co	Co	ED	EDs	ED	Co	Co	Co
AL	1	3893888	67	5088	1075	11	76	7725	512	765	671324	10596	58118
AK	2	401851	23	1160	131	6	50	10573	255	346	174431	1094	17472
AZ	4	2718215	14	3760	1296	21	269	9475	466	723	1509052	11406	194158
AR	5	2286435	75	4508	387	16	60	7925	458	507	340613	6079	30486
CA	6	23667902	58	24082	6507	10	415	22394	1877	983	7477503	1097	408067
CO	8	2889964	63	4317	573	5	69	13219	417	669	492365	408	45872
CT	9	3107576	8	2985	762	94	373	10447	49	1041	807766	92312	388447
DE	10	594338	3	753	418	132	251	5181	38	789	398115	98004	198113
DC	11	638333	1	591	591	591	591	7040	29	1080	638333	638333	638333
FL	12	9746324	67	11429	1270	8	171	14222	1127	853	1625781	4035	145468
GA	13	5463105	159	6673	639	3	42	15074	398	819	589904	2032	34359
HI	15	964691	5	708	408	2	142	18421	54	1363	762565	144	192938
ID	16	943935	44	2284	213	9	52	6403	291	413	173036	798	21453
IL	17	11426518	102	15797	4901	14	155	24596	1740	723	5253655	4404	112025
IN	18	5490224	92	7269	732	8	79	8653	554	755	765233	5114	59676
IA	19	2913808	99	5630	482	18	57	5872	512	518	303170	5731	29432
KS	20	2363679	105	4617	492	6	44	14430	390	512	366531	1845	22511
KY	21	3660777	120	4224	893	3	35	19549	369	867	685004	2265	30506
LA	22	4205900	64	5326	648	11	83	12498	538	790	557515	8525	65717
ME	23	1124660	16	1855	244	29	116	3845	240	606	215789	17634	70291
MD	24	4216975	24	4529	884	26	189	14083	328	931	786775	16695	175707
MA	25	5737037	14	6084	1308	14	435	9491	130	943	1367034	5087	409788
MI	26	9262078	83	11211	2706	15	135	9938	474	826	2337891	1963	111591
MN	27	4075970	87	7354	1033	16	85	6731	1023	554	941411	3764	46850
MS	28	2520638	82	4446	285	11	54	6720	344	567	250998	2513	30739
MO	29	4916686	115	7569	1193	10	66	14276	567	650	973896	3008	42754
MT	30	786690	57	2090	234	1	37	6675	294	376	108035	275	13802
NE	31	1569825	93	3724	508	3	40	8787	299	422	397038	513	16880
NV	32	800493	17	1046	400	7	62	9350	176	765	463087	777	47088
NH	33	920610	10	1090	348	36	109	6812	40	845	276608	27931	92061
NJ	34	7364823	21	7220	809	94	344	12863	183	1020	851116	64676	350706
NM	35	1302894	32	2623	478	10	82	7245	449	497	419700	1090	40715
NY	36	17558072	62	18924	2127	29	305	26454	583	928	2230936	5034	283195
NC	37	5881766	100	8181	500	6	82	23325	819	719	404270	3975	58818
ND	38	652717	53	2594	160	4	49	8863	139	252	88247	1138	12315
OH	39	10797630	88	12918	1384	17	147	8996	618	836	1498400	11310	122700
OK	40	3025290	77	5644	948	10	73	15924	643	536	568933	3648	39289
OR	41	2633105	36	3601	640	7	100	7191	439	731	562640	1513	73142
PA	42	11863895	67	13712	1856	11	205	9529	358	865	1688210	5072	177073
RI	44	947154	5	939	583	41	188	4611	36	1009	571349	46942	189431
SC	45	3121820	46	5052	427	24	110	14849	449	618	287913	7797	67866
SD	46	690768	66	2676	172	9	41	5371	191	258	109435	1463	10466
TN	47	4591120	95	5789	745	7	61	12605	433	793	777113	4358	48328
TX	48	14229191	254	17927	2104	2	71	18748	1930	794	2409547	91	56020
UT	49	1461037	29	2097	602	7	72	11646	353	697	619066	769	50381
VT	50	511456	14	572	110	6	41	3791	16	894	115534	4613	36533
VA	51	5346818	136	5520	481	3	41	25245	212	969	596901	2937	39315
WA	53	4132156	39	5460	1565	7	140	23761	460	757	1269749	2468	105953
WV	54	1949644	55	2540	260	7	46	6012	223	768	231414	4922	35448
WI	55	4705767	72	7363	999	13	102	6461	746	639	964988	3373	65358
WY	56	469557	23	1148	143	16	50	8476	133	409	71856	2924	20416

- \* Exclusion boundaries do not overlap (e.g., a river does not overlap a lake).
- \* County boundaries correctly fit together (i.e., no overlap or missed areas).
- \* Agricultural data are consistent with population data. Clearly, there should be farm production where there is farm population, and there should be no farm production where there is only urban population. There can be farm production in rural areas where there is no farm population (e.g., unpopulated grazing areas).

#### Apportionment of Agricultural Data to EDs

A product of this effort is a combined database of population and farm production on a grid of ED centroids. The agricultural data are tabulated at the county level, and this methodology determines the farm production at the ED level in proportion to the farm population in each ED. Specifically, for a total county farm population ( $P$ ) and an ED farm population ( $p$ ), the agricultural production for that ED is  $(p/P)$  times the total county production for each farm produce category. The population and agricultural production for each ED are then combined for the entire United States, forming the population/agricultural database (see Fig. 1).

#### DATA APPORTIONMENT

Radiation dose assessments in the CRRIS codes require population and agricultural production over a grid which covers the area of study. It is therefore necessary to apportion the data from the ED centroids to the assessment grid. Using the method described in the following paragraph, the area of each ED can be approximated, allowing a direct estimate of the population and agricultural production. Then, these local values can be apportioned to the assessment grid.

The procedure for estimating areas associated with the ED centroids is based on "Thiessen polygons," first proposed by A. H. Thiessen in 1911 (8) for analyzing precipitation data from unevenly distributed weather stations. See Ref. 9 for a review of the literature on Thiessen Polygons (TPs) and a discussion of procedures to generate them. TPs are defined based on a set of data points in a plane (e.g., weather stations or EDs) so that each polygon is enclosed by a line midway between the point under consideration and the surrounding points. More precisely, the sides of the TPs are perpendicular bisectors between the nearest-neighbor points in a plane. Also, if the nearest-neighbor points are taken as vertices of triangles covering the plane, then the triangular network possesses good properties for efficient data interpolation. For this work, the technique of Durfee and Coleman (10) was extensively modified to adapt it from an IBM 370-3033 to a Cray X-MP. In addition, the algorithm was modified for the present purpose as follows.

- \* TPs are bounded by county borders because ED boundaries were originally defined by the U.S. Census Bureau as lying wholly inside one county. The agricultural data are also tabulated on a county basis. Previously, county boundaries were not included; see, for example, Fig. 17 of Ref. 10.
- \* Exclusion boundaries are used to eliminate regions where no one lives (e.g., rivers, lakes, and oceans). At a more detailed level, these exclusion areas might include areas such as cemeteries, rugged topography, swamps, marshes, and roads; however, such a precise level of detail is beyond the scope of this work. Previously, exclusion regions were not considered in estimating ED areas via TPs.

- \* Zero-population EDs are retained in the population/agricultural database because information about where people do not live is as important as information about where people do live. Previously, zero-population EDs were eliminated from the database, and TPs were constructed only for nonzero-population EDs.
- \* The internal digitizing grid, used to sort and construct the TPs, is allowed to vary so that each ED corresponds to a unique digitizing grid point. Previously, the internal digitizing grid was fixed, combining closely spaced EDs unto the same grid point, and thus reducing the spatial resolution.

Figure 2 shows a simple example (Pickett County, Tennessee) which contains 10 EDs (points) inside the county boundary (heavy solid line). The latitude/longitude coordinates have been transformed to spatial offsets in the northern and eastern directions (northing/easting) using a Lambert map projection algorithm from Ref. 10. The ED population is the integer number adjacent to the corresponding ED centroid. Note that there are three zero-population EDs. The model adapted from Ref. 10 first establishes an internal region for analysis (the dashed box in Fig. 2). A grid is defined over this internal region; each ED centroid is associated with the closest grid point. The EDs are then sorted over this digitized grid space to determine the nearest neighbors associated with every ED. Next, TPs are constructed, as a series of perpendicular bisectors between these nearest neighbors; the TP boundaries extend to the edge of the internal digitizing region. Then the county boundary is imposed over the unrestricted TP boundaries to obtain the final TP boundaries (light solid lines) which lie wholly inside the county border, while the unrestricted TP boundaries (dotted lines) extend to the internal grid boundary. The area of each ED is estimated as the area of the TP in which the ED centroid lies and is decremented by any exclusion areas occurring in that TP (not shown in Fig. 2).

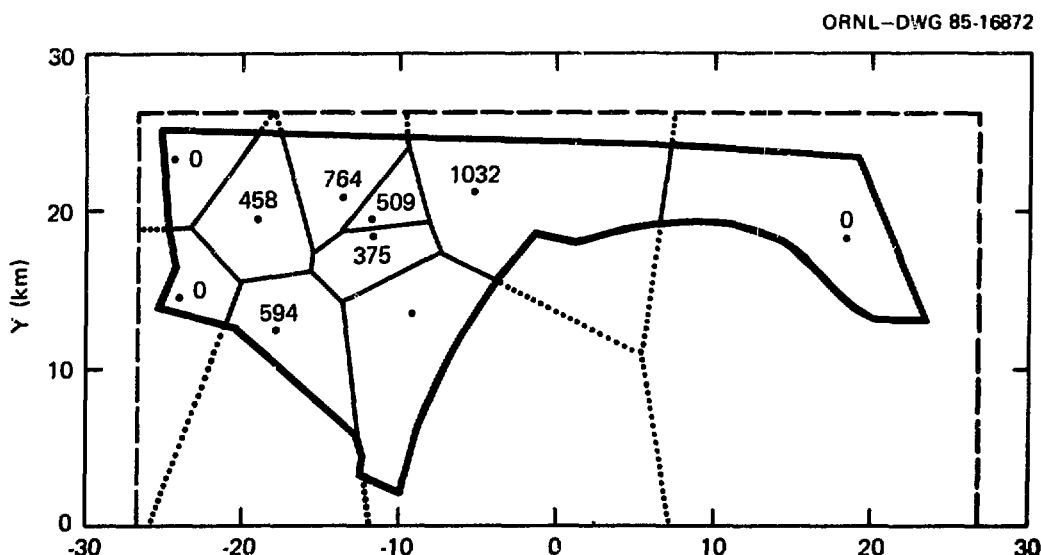


Figure 2. An example of Thiessen polygon construction.

The TP data, including TP boundaries and areas, form the second product of this database construction for the entire United States. These data are constructed on a county-by-county basis as described previously, and can be used for radiological assessments, which are described in the next section.

## DESCRIPTION OF RESULTS

The CRRIS codes allow assessments over three different grids: rectangular, circular, and individually specified points. For a rectangular grid, it is necessary to specify the grid boundaries; rectangular cells are constructed by CRRIS. The circular grid is specified by a center (e.g., release from a nuclear reactor during decontamination and decommissioning) and ring spacings; 16 sectors about the center are constructed by CRRIS. The population and agricultural data are interpolated to the assessment grid according to the following algorithm:

1. For a TP which lies completely inside a given grid cell, include all the population and agricultural production associated with that TP with the other EDs lying in that grid cell.
2. For a TP which lies partially inside a given grid cell, include a fraction of the population and agricultural production for that TP with the other EDs lying in that grid cell. The fraction attributed to a particular grid cell is  $(a/A)$ , where the total TP area is  $(A)$  and the area of the TP lying inside the given grid cell is  $(a)$ .
3. For a TP which lies completely outside a given grid cell, none of the population or agricultural production for that TP is included in the grid cell.

For purposes of area construction, the circular arcs associated with the annular grid cells are approximated by short straight-line segments. Then, the area of intersection between the TPs and grid cells (circular or rectangular) reduces to the polygon-polygon intersection method adapted from Ref. 10. From the total population and agricultural production for each grid cell, along with the area of that grid cell, it is easy to determine the population per unit area (e.g., people per square kilometer) and agricultural production per unit area (e.g., kilograms of corn per year per square meter). For a randomly specified grid point, the population and agricultural production are taken from the ED associated with the TP in which that point lies. These grid values are used for the radiological assessments which are described next.

As described in the Introduction, the CRRIS codes take a specified radionuclide release, estimate the airborne transport, and determine the average annual air concentration and ground deposition rate in each grid cell. Direct deposition on plant surfaces and root uptake of the deposited radionuclides is calculated, from which radioactivity concentrations in the food chain are determined using the agricultural database. The radiation dose is found from:

1. Ingestion of these contaminated foods,
2. Exposure to the contaminated ground surface,
3. Immersion in contaminated air,
4. Inhalation of contaminated air.

Health risks are estimated from the various organ exposures (e.g., excess probability of death due to thyroid cancer). Thus, this database construction is only a small part of the overall CRRIS.

## CONCLUSIONS

A methodology has been developed to overcome some difficulties of the previous population and agricultural database for the CRRIS codes. The use of exclusion boundaries eliminates population appearing in water bodies. Apportionment of farm data to the finest level of detail (EDs) avoids agricultural production in city centers. Retention of information at the ED level in the database ensures that sums over grid cells conserve the original population of the EDs. These improvements eliminate the inconsistencies in radiological assessments via CRRIS. The information in this database is sufficiently general to be useful in other computerized environmental assessments.

We add a note of caution about using the agricultural data apportioned to a finer degree (ED level) than they were collected (counties). While the higher resolution may appear better, errors inherent in the apportionment can give erroneous results for some applications (e.g., detailed accident assessments).

## REFERENCES

1. C. F. Baes III and C. W. Miller, "CRRIS: A Computerized System to Assess Atmospheric Radionuclide Releases," Nuc. Saf. 25, pp. 75-85, 1984.
2. C. F. Baes III, C. W. Miller, D. C. Kocher, A. L. Sjoreen, and B. D. Murphy, "An Introduction to CRRIS: A Computerized Radiological Risk Investigation System for Assessing Atmospheric Releases of Radionuclides," ORNL/TM-8573, Oak Ridge National Laboratory, Oak Ridge, TN, 1985.
3. C. F. Baes III, R. D. Sharp, A. L. Sjoreen, R. W. Shor, "A Review and Analysis of Parameters for Assessing Transport of Environmentally Released Radionuclides through Agriculture," ORNL-5786, Oak Ridge National Laboratory, Oak Ridge, TN, 1984.
4. Bureau of the Census, U.S. Department of Commerce, Washington, DC 20233.
5. R. W. Shor, C. F. Baes III, R. D. Sharp, "Agricultural Production in the United States by County: A compilation of the 1974 Census of Agriculture for Use in Terrestrial Food-Chain Transport Assessment Models," ORNL-5768, Oak Ridge National Laboratory, Oak Ridge, TN, 1982.
6. Bureau of the Census, U.S. Department of Commerce, Washington, DC 20233, "The 1974 Census of Agriculture, U.S. Summary and State Data," U.S. Government Printing Office, Washington, DC, 1977.
7. U.S. Geological Survey, National Cartographic Information Center, 536 National Center, Reston, VA 22092.
8. A. J. Thiessen and J. C. Alter, "Precipitation Averages for Large Areas," Monthly Weather Review 39, pp. 1082-1084, 1911.
9. K. E. Brassel and D. Reif, "A Procedure to Generate Thiessen Polygons," Geographical Analysis 11, pp. 289-302, 1979.
10. R. C. Durfee and P. R. Coleman, "Population Distribution Analysis for Nuclear Power Plant Siting," NUREG/CR-3056, prepared for the U.S. Nuclear Regulatory Commission, Washington, DC, 1983.

# **THE USE OF COMPUTERIZED, 3-DIMENSIONAL SOLIDS MODELING AND DATA BASE MANAGEMENT TO SUPPORT RADIATION MAPPING AND ALARA PLANNING**

Richard D. Schauss, Consultant  
Nuclear Information Sciences Company  
P.O. Box 68  
Middletown, Pa. 17057

Michael S. Slobodien, CHP  
GPU Nuclear Corporation  
P.O. Box 480  
Middletown, PA. 17057

## **ABSTRACT**

The Radiation Mapping and AlArA Planning System (RADMAPS) under development by GPU Nuclear Corporation in conjunction with Construction Systems Associates, Inc., integrates computerized solids modeling and data base management to provide an automated, integrated systems solution to the problems associated with acquiring, managing, and communicating plant radiological data at nuclear facilities. This presentation describes the status of the RADMAPS development and outlines plans for future expansion.

## **INTRODUCTION**

Although many factors contribute to the acquisition, management, and communication of radiological data at nuclear power facilities, experience gained in the Three Mile Island Unit-2 (TMI-2) Recovery Program has shown that the ability to effectively depict the exact location at which radiation data are to be collected can significantly improve the efficiency of the acquisition effort and the quality of data. If the precise location and type of measurements to be acquired are known prior to entering a radiation area, a technician is then free to concentrate on the actual measurement process. In addition to increasing the quality of the data acquired, this can also result in a significant reduction of jobhours to perform the survey. These savings then equate to reduction of occupational radiation exposure.

GPU Nuclear has been using various computer-aided design and drafting (CADD) techniques over the past three years to improve the quality of radiation survey maps used to characterize TMI-2 radiological conditions. The application of CADD technology has proven very successful and has evolved into a computer system application that integrates traditional data base management technology with the capabilities of computerized 3-dimensional solids modeling. This integrated systems approach is called the Radiation Mapping and AlArA Planning System, or RADMAPS. RADMAPS, which is in the early development phase, has the following major attributes:

- o The ability to create 3-dimensional survey maps (actually orthographic projections) to support field acquisition of radiological data,

- o A data base structure that supports the retention of radiological data and multiple data relations between survey point locations, plant geometry, and radiological data,

- o The ability to retrieve radiological data from the data base based on various selection criteria (e.g., type of survey, type of radiation, surveyor, plant zone, system/component),

- o The ability to print selected data or generate full-color graphic displays (CRT or hard copy) of selected data.

In March 1985, GPU Nuclear contracted with Construction Systems Associates, Inc. (CSA), of Marietta, Georgia, to develop the RADMAPS capability to support the TMI-2 Recovery Program. As an integral part of the RADMAPS application, CSA also developed a computerized, 3-dimensional solids model of the TMI-2 reactor building facilities and systems. CSA was chosen primarily for their existing computerized Space Modeling and Interference Detection System (SMIDS), used to provide plant design and construction-related services to the nuclear industry. (1) SMIDS is a comprehensive, 3-dimensional solids modeling system with a fully integrated data base management system. It is the basis of the RADMAPS application.

#### CURRENT STATUS OF RADMAPS IMPLEMENTATION

As of this writing, the reactor building model is complete and the RADMAPS system is installed on GPU Nuclear's DEC VAX 11/750 mini-computer system. The utility is in the process of operationally testing and evaluating the system. RADMAPS is not being used in an operational mode for all radiation surveys associated with the TMI-2 Recovery Program, but rather is currently being applied to specific characterization problems on a case-by-case basis. There is, however, much interest in the potential of RADMAPS to support production of radiological surveys on an operational basis at GPU Nuclear facilities, because the utility expects that this technology will set the standard in the years to come.

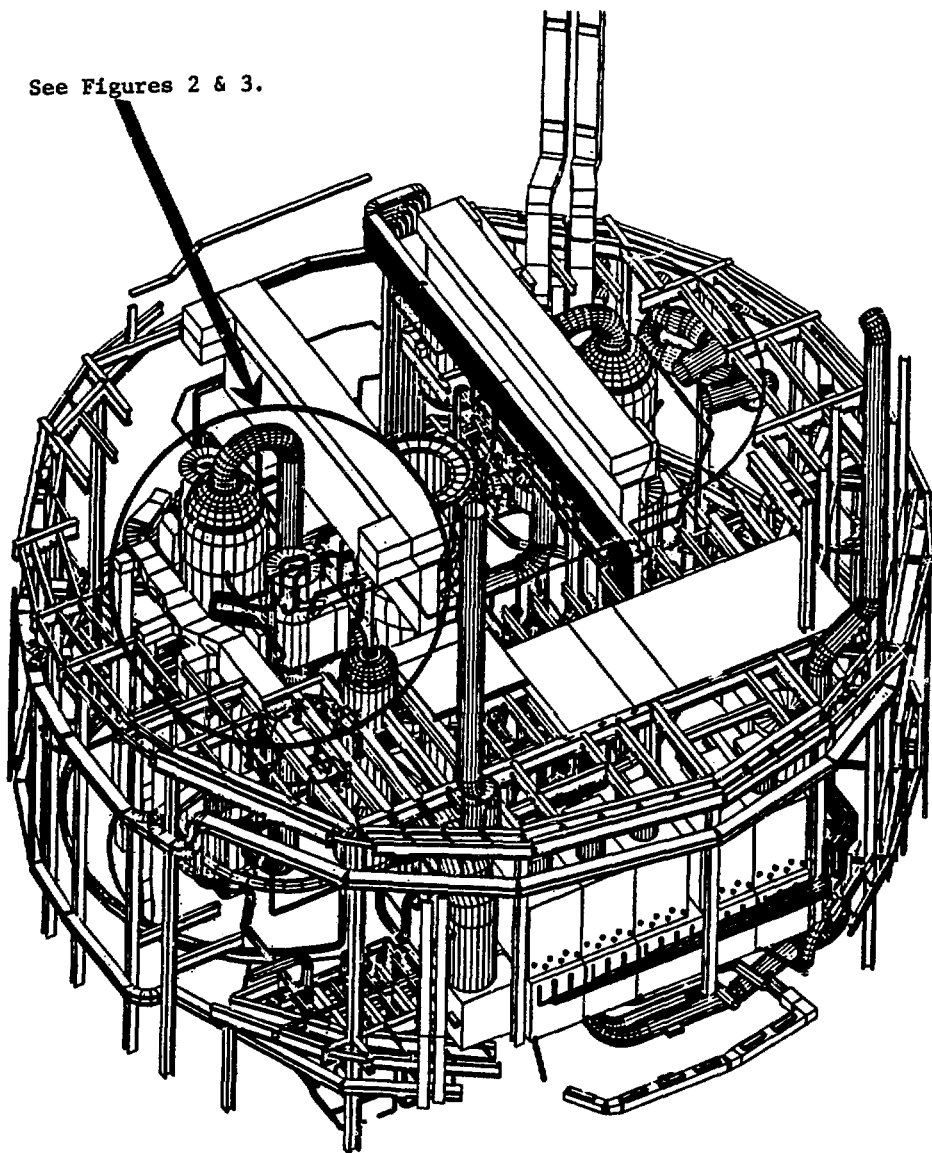
#### WHAT IS 3-D SOLIDS MODELING AND HOW DOES IT RELATE TO RADMAPS?

Understanding RADMAPS and how it performs does not require an in-depth understanding of 3-dimensional solids modeling; however, because this technology (specifically, the CSA, Inc., SMIDS system) is at the core of RADMAPS, a brief explanation of its major attributes follows:

Simply stated, 3-dimensional solids modeling, as applied to RADMAPS, provides a mechanism by which virtually any view of the TMI-2 systems and facilities may be generated. This is accomplished by developing a computerized 3-dimensional model based upon data from engineering design and as-built drawings. Information from these drawings -- including dimensions and orientation of systems, components, and structures -- is coded into the computer model. Once developed, the computer model represents the plant configuration graphically in much the same manner as a plastic model. In addition to the geometric attributes of the model, there is an associated data base structure that provides the ability to link other data (e.g., component specs and material lists) with any model component or any location within the model space. This latter attribute provides RADMAPS with the capability to link radiological data to the physical configuration of the plant.

To date, approximately 900 drawings have been used as input to develop the computer model of the TMI-2 reactor building systems and facilities. A representation of this model is shown in Figure 1.

See Figures 2 & 3.



**Figure 1**

**Three-dimensional model of the TMI-2 Reactor Building  
facilities and systems.**

As can be seen in Figure 1, significant detail is contained in the TMI-2 reactor building model. This is even more impressive when one considers that this representation excludes small-bore piping, (piping of less than 2-1/2 in. diameter), major pipe supports, whip restraints, and mirror insulation. Also not shown, but included in the actual model, is the concrete structure. Although impressive, this level of detail can be difficult to assimilate when taken as a whole.

#### REDUCING MODEL COMPLEXITY

The model (through the capability inherent in SMIDS) is logically divided and categorized when it is initially developed. Logical division of the model by categories such as zone, system type, system name, component type, and component name allows the geometry from the model to be selected and extracted based on these criteria. This reduces the complexity of the model on the user level.

An example of the simplification of the model, in systematic stages is shown in Figures 2 and 3. These examples show how the complexity can be reduced while simultaneously increasing the level of detail at an area of interest. In this case, the 'A' D-ring area, and ultimately the area around the top of the pressurizer, is the subject of investigation. Once the RADMAPS user has isolated the area of interest and is satisfied with the content and view, the survey map is created.

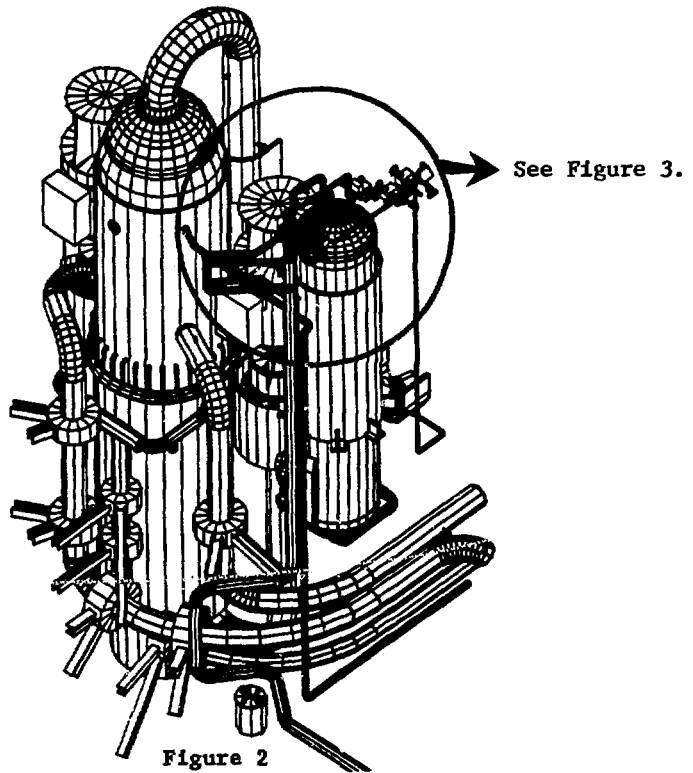
#### CREATING A RADIOLOGICAL SURVEY MAP

To create a survey map the user first creates a geometry map file, which is a 3-dimensional representation of the area to be surveyed. The geometry map file is then used as input to a computer process in which the user interactively identifies survey point locations in 3-dimensional Cartesian space. Since points are located in three dimensions, at least two views are needed to locate and display these points. Examples of two views used to identify survey points around the top of the pressurizer are shown in Figures 4 and 5.

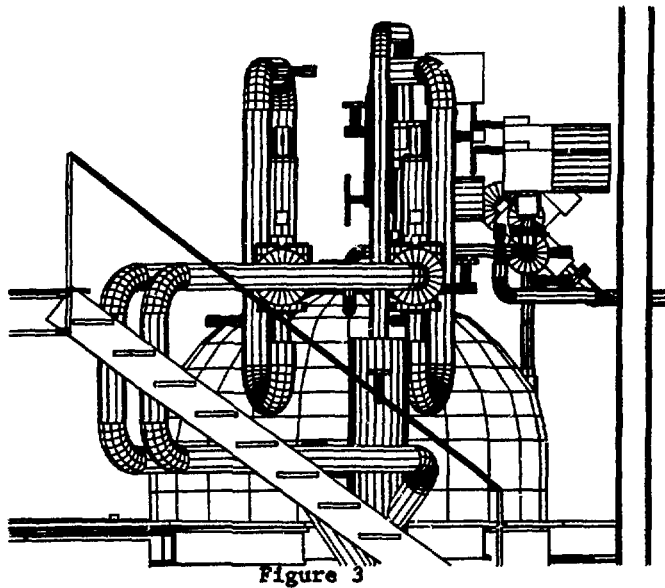
Once identified, the survey points are automatically attached to a logical survey master record. The survey master record contains the details of the survey. The outputs of this processing are: 1) plotted maps showing the area and locations to be surveyed (Figures 4 and 5), 2) a survey data-acquisition sheet (Figure 6), 3) data base records identifying the data associated with the survey points and the survey master, and 4) the data relationships between the survey points, the survey master, and plant geometry. The actual data base structure is beyond the scope of this paper; however, a simple block diagram of the relationship between these various data structures is shown in Figure 7.

Once the radiological survey has been performed and the survey data transcribed to the survey data-acquisition sheet by the surveyor, the data can then be input to RADMAPS. Because the survey data record structure and the survey data-acquisition sheet were created at the same time and match one-for-one in format, the data base can be updated by data-entry clerks. Various built-in supervisory checks are planned to ensure that proper review of survey data has been completed by qualified personnel prior to releasing data for operational or historical purposes.

An example of the completed survey for the top of the pressurizer area is shown in Figure 8. For purposes of reproduction, this survey map is shown in black and white. Unfortunately, this makes the exact locations of some of the survey points difficult to see. These types of problems are easily dealt with



Model extraction showing the TMI-2  
'A' D-ring area, looking northeast.



Model extraction showing the top of  
the pressurizer, looking east.

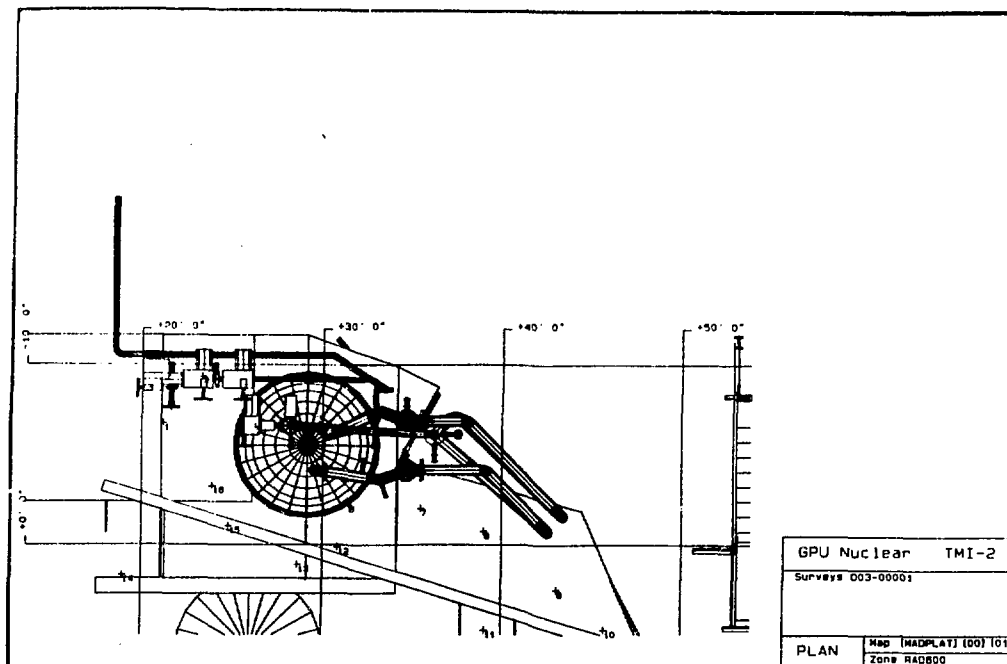


Figure 4

Plan view of a RADMAPS survey map showing the survey locations.

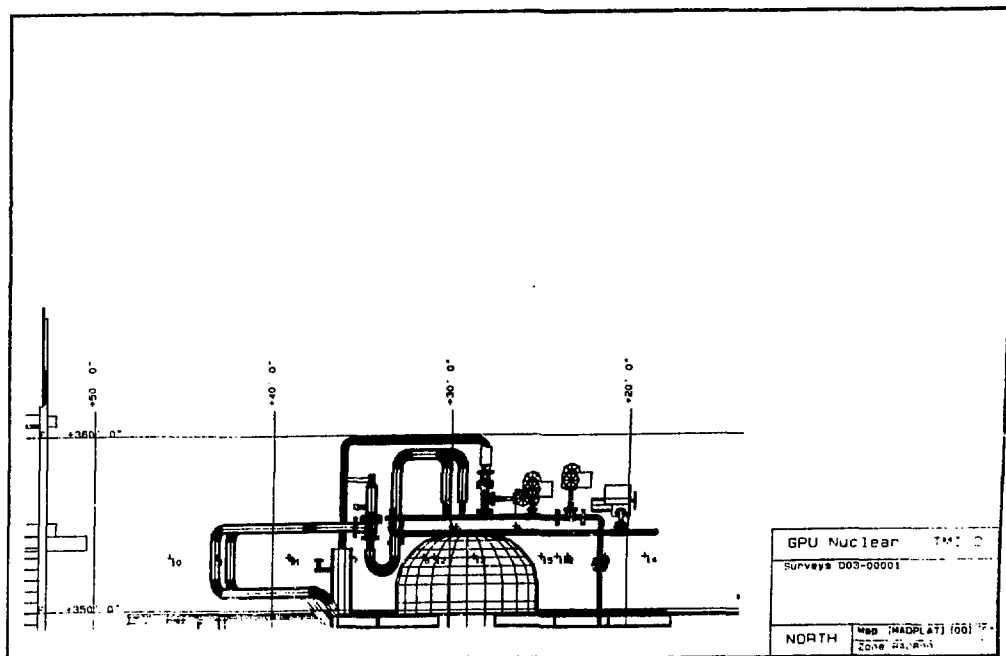


Figure 5

East view of the top of the pressurizer showing survey locations.

SPACE MODELING AND INTERFERENCE DETECTION SYSTEM  
RADMAPS 3-0  
MAP DATA ACQUISITION SHEET

SURVEY NAME [D03-00001] MAP ID [RADPLAT][00]-[01]

1. RAD CON FILE ID [XXXXX]  
2. ORIGINATOR [SCHADSS ] DATE [28-OCT-88]  
3. COMMENTS [A' D-RING ROOM 4590-84-0438 ]  
4. PLANNED DATE [01-NOV-88] TIME [ ]  
5. PERFORMED DATE [ ] TIME [ ]  
6. SURVEYOR NAME [ ] SS # [ ]  
7. SUPERVISOR NAME [ ] SS # [ ]  
8. CONDITIONS CODE [ ] -COMMENT [ ]  
9. ROW STATUS [AD] INSTRUCTION CODE [PL13]  
10. ROW [85-1234] [ ] [ ]

AS	POINT ID	SURVEYOR	INSTN TYPE	INSTN NUMBER	BCP	SUR RAD/ISO TYP TYPE	QUANTITY
1	[DNAD6001]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
2	[DNAD6002]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
3	[DNAD6003]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
4	[DNAD6004]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
5	[DNAD6005]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
6	[DNAD6006]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
7	[DNAD6007]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
8	[DNAD6008]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
9	[DNAD6009]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
10	[DNAD6010]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]
11	[DNAD6011]	[ ]	[ ]	[ ]	[ ]	[AD][GAMMA]	[ ]

Figure 6

A RADMAPS survey "Data Acquisition Sheet".

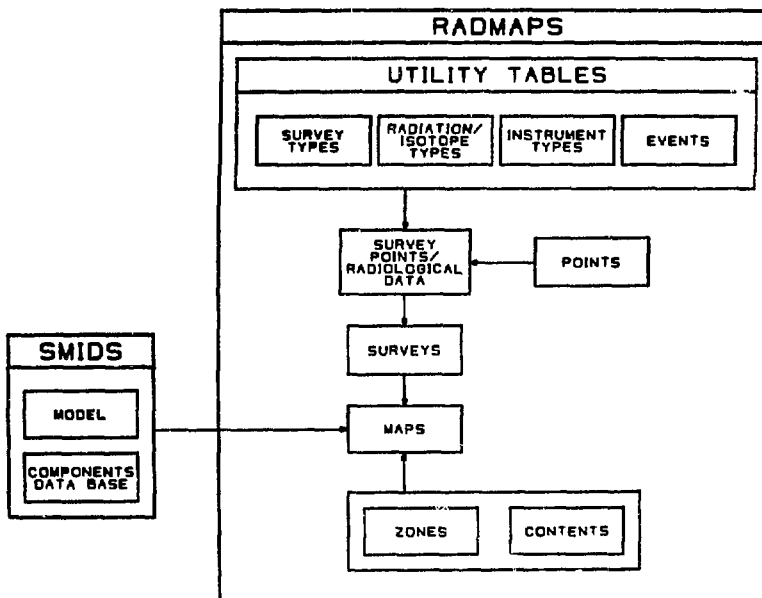


Figure 7

RADMAPS data structure block diagram.

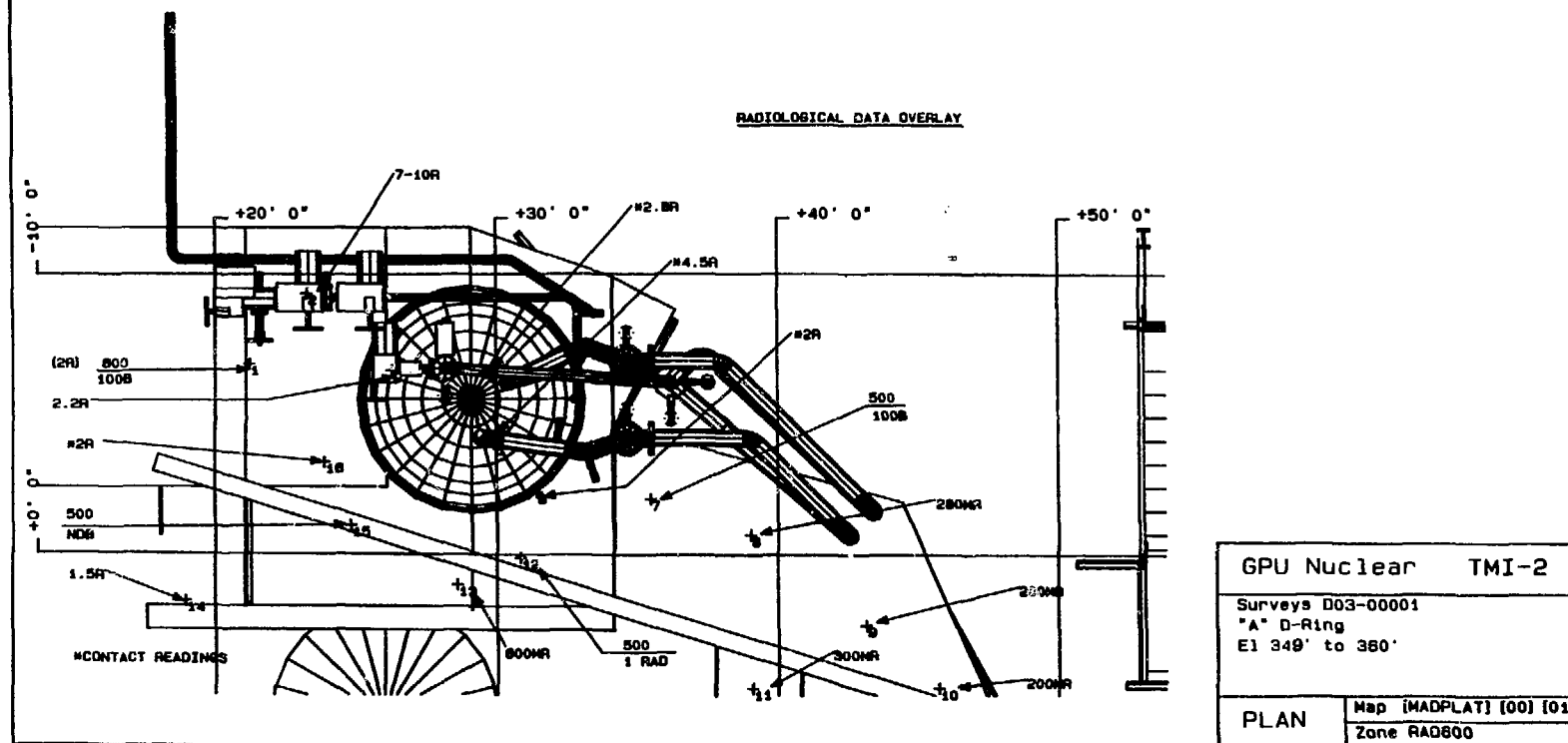


Figure 8

RADMAPS survey map showing survey locations and radiation data.

by using color to separate the survey points from the plant geometry. Also, color is used extensively when separation of system and component types is required. When reproduction costs inhibit the use of color, other techniques, such as multiple view plots showing a fewer systems or components in each plot, may be necessary to maintain clarity.

#### FUTURE EXPANSION

The present RADMAPS configuration provides the basis for expansion of its capabilities as a sophisticated data retrieval and analysis tool. Our concentration in the initial design and development phase was on providing a state-of-the-art mechanism for the acquisition and management of radiological data, with the primary objective of increasing the quality of acquired data.

Future expansion of RADMAPS will initially focus on the development of effective and efficient data retrieval and communication capabilities. In addition to traditional data retrieval and reporting, the potential of the built-in graphics capabilities will be further explored and additional graphics facilities added as users become familiar with the capabilities and limitations of the technology.

Plans include development of a computer file interface between the RADMAPS data base and packaged spreadsheet-type software (e.g., Lotus 1-2-3<sup>TM</sup>, VISICALC<sup>TM</sup>, and MEGACALC<sup>TM</sup>). This will allow technicians and engineers to automatically transfer actual radiological data into ALARA programs which analyze and predict job-related exposure. This would allow analysis and manipulation of data without jeopardizing the integrity of plant records.

Interest has also been expressed (both internal and external to GPU Nuclear) to integrate a radiation shielding/source-term analysis capability with RADMAPS. The authors feel that this type of development is possible, and, although beyond scope of this paper, the development of this capability could be easily justified on a technical and economic basis.

#### References:

1. A. M. Burger, and T. K. Duncan, "Three Dimensional Design Modeling of Plants," Mechanical Engineering pp. 40-44, April 1985.

HEALTH PHYSICS AND OPERATIONAL EXPERIENCE GAINED FROM SLURRY  
TRANSFER OF WASTES CONTAINING 780 CURIES OF RADIUM-226

William C. Borden and Roger R. Land\*  
Bechtel National, Inc., P.O. Box 350, Oak Ridge, TN 37830

ABSTRACT

During remedial action at the DOE Niagara Falls Storage Site 3.5 million kilogram (3,891 ton) of uranium ore residues (code named K-65 during the Manhattan Project) were slurry transferred about one mile from a concrete water tower to an engineered waste containment facility. This operation presented a number of unusual health physics problems, mandating special handling, measurement, and exposure control methodologies. Principles among these were: (1) a specific activity of 220,000 pCi/gm; (2) open air concentrations of radon-222 of up to 60,000 pCi/l; (3) radon concentrations inside the tower of 1,700,000 pCi/l; (4) exposure rates of up to 350 mR/hr; (5) widely varying climatic conditions; and (6) working atop a 165 foot tall, 40 foot diameter, structure in these conditions. This paper will describe the transfer operation, its problems, problem solutions, and successes. Detailed attention is given to personnel and environmental monitoring, as well as contamination and exposure control methodologies. Operations concluded with the K-65 residues stored in an environmentally stable condition, less than 10 man-rem of worker exposure, and average off-site radon-222 concentrations of less than 0.3 pCi/l above background.

**ASSURANCE PROGRAM FOR REMEDIAL ACTION (APRA)  
MICROCOMPUTER-OPERATED BIBLIOGRAPHY MANAGEMENT SYSTEM<sup>(a)</sup>**

R. D. Stenner, D. K. Washburn and D. H. Denham  
Pacific Northwest Laboratory  
P.O. Box 999  
Richland, Washington 99352

**ABSTRACT**

Pacific Northwest Laboratory (PNL) provided technical assistance to the Office of Operational Safety (OOS) in developing their Assurance Program for Remedial Action (APRA). The APRA Bibliography Management System (BMS), a microcomputer-operated system designed to file, locate and retrieve project-specific bibliographic data, was developed to manage the documentation associated with APRA. The BMS uses APRABASE, a PNL-developed computer program written in dBASE II<sup>(b)</sup> language, which is designed to operate using the commercially available dBASE II database software. This paper describes the APRABASE computer program, its associated subprograms, and the dBASE II APRA file. Although the BMS was designed to manage APRA-associated documents, it could be easily adapted for use in handling bibliographic data associated with any project.

**INTRODUCTION**

Pacific Northwest Laboratory (PNL) provided technical assistance to the Department of Energy's Office of Operational Safety in developing their Assurance Program for Remedial Action (APRA). One of PNL's tasks in the APRA project was to define and locate various published documents that were associated with the DOE remedial action programs. As remedial action documents were discovered and located, it became apparent that some form of tracking system was needed to organize and manage all the bibliographic information associated with the documents. It was felt that such a system should be computerized, preferably on a microcomputer for flexibility and ease of access, to allow for rapid tracking and retrieval of the information. It was also thought desirable to use commercially available database software, because this would greatly reduce the cost of developing the system.

The APRA Bibliography Management System was established using the dBASE II<sup>(b)</sup> database software. The system's bibliography data are entered and edited using standard dBASE II procedures. Data organization, searches, and report writing are accomplished using the APRABASE computer program, which was written by PNL in the dBASE II programming language. The PNL APRA Bibliography Management System was developed to operate on an IBM<sup>(c)</sup> personal

---

(a) Work supported by the U.S. Department of Energy, Office of Operational Safety, under Contract DE-AC06-76RLO 1830.

(b) dBASE II is a trademark of Ashton-Tate, Culver City, California.

(c) IBM is a trademark of the International Business Machines Corporation.

computer that was connected to a 20-megabyte Tallgrass Technologies Corporation hard disk. The system can be run using only the IBM-system floppy disks for storage; however, this greatly limits the size of the bibliography file. The size of the present APRA file requires use of the hard disk.

As with most data management systems, consistency of format of the data entered into the database file (APRA file) is critical. Data must be entered in the same format each time or it will be difficult to conduct a search of the data. The system is also sensitive to spacing between words and characters in the data entries. For example, if a blank space is left after a word or character in the data set, it will also have to be entered as such in the search query or the system will not find it during a search of the data set. The dBASE II system is also sensitive to upper- and lower-case letters. Thus, during search routines, it is necessary to query using the same upper- and lower-case letters as those used during data entry into the APRA file.

#### APRA DATABASE FILE

The APRA Bibliography Management System uses standard dBASE II procedures to set up and maintain the APRA bibliography file. The dBASE II operational procedures will not be discussed in this paper, because they are readily available in the dBASE II User's Manual that is supplied with the dBASE II software.

The dBASE II system requires that database files be set up using user-specified fields to identify the parameters being used for search purposes. The fields established for the APRA file are as follows: LEADAUTHOR, AUTHORS, DAY, MONTH, YEAR, INITITLE, TITLE, VOLUME, DOCNUM1, DOCNUM2, PAGES, ORIGINATOR, PUBLISHER, CITY, STATE, COUNTRY, APRAPROG, DOCTYPE, LOCATION, KEYWORD1 through KEYWORD10 AND DUP. The APRA file structure is shown in Table 1.

To help perform a search on the main author of a document, the author parameter is broken down into two fields: LEADAUTHOR and AUTHORS. The LEADAUTHOR field accepts up to 25 characters. If a name longer than 25 characters is encountered, the program will truncate the remaining characters; however, this is not expected to be a problem since the first 25 characters should be sufficient to specifically identify the particular author. All author names should be entered as "Last name," "First initial," and "Middle initial."

The DAY, MONTH, and YEAR fields are used to enter the date of the publication. They are entered as separate fields, so the system can easily search for documents published during a particular year, a particular time frame during a year, a several-year period, or on a particular date. The DAY is recorded as the numerical calendar date for the particular day. For consistency and ease of reporting, the system is designed to store these numbers as if they were characters. Storing them as characters allows the entry of a range of days (e.g., 12-15). The convention used for entering the MONTH is to spell out the entire name of the month. The YEAR field provides for up to 9 characters, which also allows for entering a date range (e.g., 1981-1985).

Table 1. APRA Structure

<u>Field Number</u>	<u>Field Name</u>	<u>Type (Character or Numeric)</u>	<u>Width (Number of Characters)</u>
001	LEDAUTHOR	C	25
002	AUTHORS	C	75
003	DAY	C	05
004	MONTH	C	09
005	YEAR	C	09
006	INITITLE	C	10
007	TITLE	C	200
008	VOLUME	C	10
009	DOCNUM1	C	20
010	DOCNUM2	C	20
011	PAGES	C	08
012	ORIGINATOR	C	80
013	PUBLISHER	C	80
014	CITY	C	20
015	STATE	C	15
016	COUNTRY	C	25
017	APRAPROG	C	07
018	DOCTYPE	C	05
019	LOCATION	C	05
020	KEYWORD1	C	30
021	KEYWORD2	C	30
022	KEYWORD3	C	30
023	KEYWORD4	C	30
024	KEYWORD5	C	30
025	KEYWORD6	C	30
026	KEYWORD7	C	30
027	KEYWORD8	C	30
028	KEYWORD9	C	30
028	KEYWORD10	C	30
030	DUP	C	01

The document title is entered using the fields INITITLE and TITLE. Because most document titles are too long to search efficiently, the INITITLE field was set up to allow for a rapid title search. The first 10 characters of a document's title should be entered in the INITITLE field. The TITLE field allows up to 200 characters.

The VOLUME field is provided for recording the volume number of a several-volume set of documents. In many cases this may be the only field that distinguishes between the different documents of a several-volume set because their document numbers, titles, authors, etc. may be the same. This field may be left blank if the document being recorded is a single volume.

The document numbers are entered using the fields DOCNUM1 and DOCNUM2. Two fields are provided since some documents have two different numbers assigned to them. For example, some NUREG documents are assigned both a Nuclear Regulatory Commission number and a generating-laboratory number. Both document number fields allow up to 20 characters or numbers.

The PAGES field is provided for recording the total number of pages in a document. This information will tell you whether you are looking for a

large or small document when physically looking for the document on a shelf. It also indicates the depth to which a subject is covered in the document.

The name of the organization(s) producing the document is covered in the fields ORIGINATOR and PUBLISHER. In many cases the originator and publisher are the same. The ORIGINATOR field is set up to record the name of the company or laboratory that wrote the document. The PUBLISHER field is set up to record either the sponsoring organization's name or the name of the organization that made the document available to the outside community. Both the PUBLISHER and the ORIGINATOR fields allow up to 80 characters.

The geographical area of the organization responsible for the document is recorded in the CITY, STATE and COUNTRY fields. The CITY field allows up to 20 characters, the STATE field provides for a maximum of 15 characters, and the COUNTRY field permits up to 25 characters.

The APRAPROG field allows the user to record the name of the DOE remedial action program that is associated with the document. The APRA Bibliography Management System includes the following five categories of APRAPROG choices:

- UMTRAP (Uranium Mill Tailings Remedial Action Program)
- GJRAP (Grand Junction Remedial Action Program)
- FUSRAP (Formerly Utilized Sites Remedial Action Program)
- SFMP (Surplus Facilities Management Program)
- GENERAL (Documents associated with other programs)

A special screening classification system was set up for the APRA Bibliography Management System that allows the user to generally classify by code each document entered into the system. DOCTYPE is a five-character field used to record this classification code. The document classification codes established for the APRA Bibliography Management System are presented in Table 2.

Table 2. APRA Bibliography Management System Document-Classification Code

Code	Description
CP	Document is located in a conference proceedings (This can stand alone or it can be added to the end of any of the other categories to indicate that the document can be found in a conference proceedings, e.g., DCP.)
CT	Document contains mainly remedial action project control technology information
D	Document contains mainly dose criteria, standards, guidelines or limits
E	Document contains mainly environmental pathway analysis information
DE	Document contains both D and E
DR	Document contains both D and R
DSD	Document contains both D and SD
DSP	Document contains both D and SP
EA	Document is a Remedial Action Engineering Assessment
ER	Document contains both E and R
F	Document is a final decommissioning report

Table 2. Continued

Code	Description
GP	Document is of general program interest but does not address any area specifically
Q	Document contains mainly quality assurance information
QD	Document contains both Q and D
QSP	Document contains both Q and SP
R	Document contains mainly risk assessment information
SD	Document contains mainly survey data
SDSP	Document contains both SD and SP
SP	Document contains mainly survey protocol

The LOCATION field is used to record the physical location of a document at PNL. A five-character location-code system was developed to describe and record information pertaining to the physical location of a document. This five-character location-code system is described in Figure 1.

The five-character location code format is:

XXXYZ

where X - represents space for the initials of the person whose office contains the document

Y - represents space for one of these information-type code letters

- H - Hardcopy
- M - Microfiche
- B - Bibliography Information Only
- L - Located at One of the Main Libraries (some information at office indicated)

Z - represents space for one of these codes which will describe location with the office or further specify the document

If Y = H:

- S - Bookshelf
- F - File Cabinet

If Y = L:

- B - Battelle Technical Library
- R - Battelle Research Technical Laboratory (RTL) Library

If Y = B:

- T - Title Only
- A - Abstract Only

Figure 1. Five-Character Document-Location Code for the APRA Bibliography Management System.

A series of ten keywords may be recorded in the fields KEYWORD1 through KEYWORD10. These keywords can be descriptive words or phrases containing no more than 30 characters each. The system will accept any keywords or phrases entered. However, for ease in searching on key descriptive words, a set of keywords and phrases was developed for the APRA Bibliography Management System.

To illustrate the type of words/phrase to be recorded on the keyword fields, a partial list of the keywords developed for the APRA Bibliography Management system is presented in Table 3. The actual list of keywords is much larger and is expected to expand over time.

Table 3. Partial Keyword List for APRA Bibliography Management System Keyword Search Option

<u>KEYWORD</u>	<u>KEYWORD</u>	<u>KEYWORD</u>
DECOMMISSIONING	STANDARDS	NATURAL PRODUCTS
DECONTAMINATION	REGULATIONS	FISSION PRODUCTS
R&D FACILITIES	STATUTES	ACTIVATION PRODUCTS
IRRADIATION FACILITIES	GUIDES	FUSION PRODUCTS
REACTORS	CRITERIA	PLUTONIUM
ACCELERATOR	HANDBOOK	UNRESTRICTED USE
FUEL FABRICATION	DATABASE	SAMPLING

The DUP field is provided to indicate that more than one physical copy of the document is available at the location indicated in the LOCATION field. The DUP field is designed to accept a "Y" when more than one copy is available. The DUP field should either be left blank or have an "N" entered in it if only a single copy is available.

#### APRABASE PROGRAM

The APRABASE program and its subprograms MIREPT3, MKREPT3A, MKREPT3B, MKREPT3C, MSREPT3, MSFORM3, and MTITLE were written to arrange and report the bibliographical data in the APRA file by selected fields. These programs were also designed to allow searching of the file on specified parameters to locate documents and prepare user-selected/tailored reports of the bibliographic information found.

APRABASE is a menu-driven interactive program that is written in the dBASE II language. The user operates the program from the main menu, which provides for selection of the following options:

- Option I Index the file on a choice of specified parameters, and receive a written report organized according to index parameters.
- Option K Search the file for selected keywords, and receive a written report describing the documents containing the selected keywords.
- Option S Search the file for a document that contains a specified known parameter (e.g., document number), and, if found, receive a written report describing the document.

Option D Organize the file by document number, and print out a listing that can be used to easily locate and remove duplicate entries.

Option Z Exit the APRABASE program and place control back at the main dBASE II program level.

Option I allows the user to index the APRA file by selecting parameters from the displayed parameter menu. Upon entry of the parameter, the user will receive a written report that is organized as specified by the selected parameters. The following are index parameter choices:

LEADAUTHOR(25)	INITITLE(10)	DOCNUM1(20)	CITY(20)
LOCATION(5)	DOCTYPE(5)	DOCNUM2(20)	STATE(15)
APRAPROG(7)	YEAR(9)	PAGES(8)	COUNTRY(25)
PUBLISHER(80)	ORIGINATOR(80)		

The numbers in parentheses indicate the number of characters located in the respective fields. APRABASE can index on one, two, or three index parameters at one time. The dBASE II system will permit a maximum combined length of 100 characters in the field for indexing. Thus, any combination of up to three parameters can be chosen as long as the total of the numbers in parentheses does not exceed 100.

Option K is designed so the user can search the APRA file on selected keywords and receive a written bibliographic report describing the documents that contain such keywords. A list of suggested keywords used when conducting a search is provided in Table 3.

Option S allows the user to search the APRA file for a document that contains a specified known parameter (e.g., document number, author(s), originator, etc.) and, if found in the file, receive a written report describing the document. This option involves entering the appropriate parameter field name, relational operator and known parameter. A list of the possible parameter field names and relational operators available for use in this option is provided in Table 4. These entries are delimited using the dBASE II format. Delimiting is accomplished by enclosing the entry in quotation marks. The known parameter has to be delimited using double quotation marks. An example of how each entry should be delimited is provided on the screen when APRABASE queries such an entry. Because it is easy to forget to delimit the entries, upon receipt of the entries the program will immediately ask if they were delimited. If they were not delimited, it will loop back and allow reentry of the parameters.

Table 4. APRA Bibliography Management System Option S Search Parameter and Relational Operator Choices

<u>Parameter</u>		<u>Relational Operator</u>
AUTHORS	PUBLISHER	<> (Not Equal To)
YEAR	CITY	>= (Greater Than or Equal To)
TITLE	STATE	<+ (Less Than or Equal To)
DOCNUM1	COUNTRY	> (Greater Than)
DOCNUM2	APRAPROG	< (Less Than)
DOCTYPE	KEYWORDS	
LOCATION		

Option D automatically organizes the file by document number and prints out a listing that can be used to locate and remove duplicate entries in the APRA file. The printout from this option includes the APRA file individual record numbers, which make it easy to use the standard dBASE II editing routine to locate a record and either eliminate a duplicate or correct a field.

Option Z allows the user to exit APRABASE. It returns the user to the main dBASE II program. It is then necessary to enter QUIT to completely leave the dBASE II system. The user can quit a run at any time by pressing ESC, which will automatically exit to the computer's operating system. However, if the program is exited this way, it may be necessary to manually turn off the printer (press Ctrl PrtSc). This is necessary only when the program is exited during operation within a loop containing an internal command to print.

Menu-driven options I, K and S allow the user to choose report formats for printing the results of the index or search operations. When the user enters the number indicated, the report format choices are as follows:

- 1 - Informal report in tabular form that includes the first document number, title, lead author and document location.
- 2 - Informal report in tabular form that includes the first document number, second document number, volume, year, document type, APRA program code, country of publication, and number of pages.
- 3 - Data listing that includes lead author, all authors, title, volume, year, document numbers, originator, publisher, country of publication, APRA program code and number of pages. This report format produces a separate page of output for each document.
- 4 - Informal report in tabular form which user designs (using the menu-driven REPORT FORM routine of dBASE II; any of the field names contained in the APRA file structure can be used when designing the informal report).
- 5 - Informal report in tabular form which user previously designed and renamed. Do not select this report unless you have previously selected informal report format choice 4 and saved the report format.

The MIREPT3, MKREPT3A, MKREPT3B, MKREPT3C, MSREPT3, MSFORM3 and MTITLE subprograms are automatically called by APRABASE. MIREPT3 is used to print the report when report format option 3 is selected under the main program option I. MKREPT3A, MKREPT3B and MKREPT3C are used to print the report for report format option 3 under the main program option K. MSREPT3 and MSFORM3 are used to print the report for report format option 3 under the main program option S. MTITLE is used by all of the programs to label the reports.

## SUMMARY

The BMS is a user-friendly microcomputer-operated interactive system designed to assist in tracking documents associated with a specific project. It is not intended to replace any of the larger mainframe type general bibliographic systems. It is designed to be used in conjunction with the larger systems and, in fact, we have found that the BMS is a good tool for organizing and tracking blocks of data obtained from some of the general bibliographic systems.

Although the BMS was designed to manage APRA-associated documents, it can be easily adapted for use in handling bibliographic data associated with any project. It could also be adapted to run under the new Ashton Tete dBASE III software.

#### ACKNOWLEDGMENTS

This work was performed for the Department of Energy, Office of Operational Safety, Carl Welty, Jr. sponsor. The authors wish to acknowledge Rebecca Peloquin for her assistance in reviewing and advising on the APRABASE code development; Kathi Hanson for her efforts in editing the text; and Norma Van Houten for her typing and paper preparation efforts.

## DECOMMISSIONING THE WORLD'S FIRST COMMERCIAL NUCLEAR FUEL REPROCESSING FACILITY

Stephen Marchetti and Dean Hoffman  
West Valley Nuclear Services Company, Inc.  
P. O. Box 191  
West Valley, New York 14171-0191

### ABSTRACT

An overview of the West Valley Demonstration Project is presented. Nuclear waste forms managed at the project premises include high level liquid and solid waste, spent fuel, transuranic and low level wastes. The key purpose of this paper is to outline the decontamination and decommissioning program and to provide a pictorial description of associated techniques developed here. The principal goal of the Project is to vitrify more than 2 million liters of liquid high level waste. This is addressed and a pictorial presentation of vitrification equipment is included.

### INTRODUCTION

Westinghouse engineers and technicians are completing the decontamination and decommissioning of the world's first commercial nuclear fuel reprocessing facility. This is very significant in that it marks the first time that facility decontamination of this magnitude has been attempted. In addition to demonstrating the capability to manage complex and innovative nuclear service projects, this program will help to better gauge the potential cost required to decommission the nation's nuclear facilities.

The former reprocessing plant, located in western New York about 30 miles south of Buffalo, was built by the W. R. Grace Co. in 1966 and was operated by the Nuclear Fuel Services Co. (NFS) until 1972. Escalating regulatory requirements and declining economic prospects in the nuclear fuel reprocessing business lead to its closure. During the plant operation about 600 metric tons of spent nuclear fuel were reprocessed. This recycling produced about two million liters of highly radioactive liquid waste and left the majority of the 33,000 m<sup>2</sup> plant contaminated with the radioactive byproducts of reprocessing.

Though production at the facility terminated in 1972, Nuclear Fuel Services which became a subsidiary of Getty Oil Co. continued to maintain the facility in accordance with its licensing agreements. In 1980 the federal government authorized the U. S. Department of Energy to conduct a high level waste solidification demonstration at the facility. Under this agreement, facility decontamination is to support the high level waste solidification program. Enactment of this legislation and subsequent legal settlements between New York State and the former operator cleared the way for NFS to leave the site in February 1981. The West Valley Nuclear Services Co., a wholly owned subsidiary of the Westinghouse Electric Corporation, was selected by competitive bid to design and operate the solidification process.

## HIGH LEVEL WASTE SOLIDIFICATION

The West Valley facility is unique in that it contains virtually all types of nuclear waste including high level liquids, spent fuel, transuranic waste, and both low level solid and liquid radioactive waste. The high level radioactive wastes are stored at the site in two tanks: one of the tanks contains 2 million liters of neutralized (Purex) high level waste. A clay-like sludge has deposited at the tank bottom leaving a clear highly radioactive liquid above. The other smaller tank contains 30,000 liters of acidic waste produced from Thorex reprocessing.

Engineers at the West Valley Demonstration Project are developing the world's largest nuclear waste solidification system to convert the liquid high level waste to borosilicate glass for easy handling. Nonradioactive test runs are being conducted this year. Initial tests have confirmed the design, construction, and operation of the major equipment.

The waste processing system features a 52 ton ceramic lined melter and a 13 foot tall turntable. Beginning in 1988 high level nuclear wastes, sludges, and molten glass will be combined in the melter at 2000°F. This high temperature allows the waste to bond into the glass and the mixture is poured into stainless steel canisters to harden. The canisters will be transported to a federal repository for permanent disposal.

In addition to the high level waste solidification effort, the plant will be decontaminated and decommissioned as agreed to by federal and state governments and permitted by the U. S. Nuclear Regulatory Commission. The waste solidification portion of the Project will be completed by 1990 and is expected to cost about \$440 million. That is two years ahead of schedule and almost \$30 million below the original cost estimate for the solidification project. Final decommissioning activities are expected to continue into the late 1990s.

## THE DECOMMISSIONING CHALLENGE

The Purex process used at West Valley required three types of processing areas based on work area radiation and/or contamination levels. Those areas include primary cells containing the fuel chopping, dissolution, and high level waste handling equipment. Dose rates in these areas range from 50 R/hr to 1800 R/hr and were not designed for personnel entry.

Process areas used for solvent extraction of nuclear fuel and relatively low-dose rate processing equipment were contained in secondary cell areas. Moderately high dose rates (< 100 mr/hr) and very high levels of surface contamination in these areas permitted short-term maintenance access but precluded routine access during plant operations. These areas also included crane and remote equipment maintenance facilities which served the primary cells.

The third type of plant area includes routinely occupied support facilities and operating aisles which are generally free of surface contamination with low dose rates (generally less than 10 mr/hr). Prior to its shutdown in 1972 the plant equipment was flushed to allow processing system modifications to be made. These flushes were effective in removing residual fuel and fission product activity from the plant piping but had little impact on the general radiation level in areas where manned entry is required. This is principally because of small pockets of residual radioactivity and extremely high levels of surface contamination.

Characterization of the facility during the early stages of the project indicated that decontamination work could be best performed using both "hands on" and remote methods depending on the conditions encountered. While innovation and some limited development work in tool and environment design are encouraged, every effort is made to use commercially available products to complete the facility cleanup in a safe and effective manner.

The extent of decontamination work required is massive. In terms of material quantity alone more than 500 tons of contaminated stainless steel must be extracted from the plant. The vast majority of this material must be removed from plant areas not designed for decommissioning. In fact the entire extraction system was put in place before the roof was built.

#### WORKING IN A HIGH RADIATION ENVIRONMENT

The primary cells which contain the process from fuel chopping through the first cycle of solvent extraction were designed for totally remote operation. The largest such area, the Chemical Process Cell, which is 93 by 25 by 45 feet high was designed for remote replacement of equipment but not complete decommissioning. Cleanup of this cell was further complicated by the fact that the remotely operated cranes and power assisted manipulator required complete rebuilding. Work using this remote equipment is in progress. To date about half of the remote connectors and processing vessels have been removed from the cell.

Specially designed modifications have been made to commercially available saws and cutting equipment to size reduce the highly contaminated equipment for packaging. Vessels such as fuel dissolvers weighing almost 14 tons will be inspected, dried, cleaned, and packaged in sealed overpacks using careful radiological control practices. This precludes cross contamination of work areas during removal of the equipment.

After the overpacks are filled and sealed, they are carefully transported to a specially designed temporary storage area using vinyl coated nylon stretched between extracted aluminum beams. This building provides weather protection for the containers stored there. Shielding is provided by a wall composed of 45 concrete modules each 6 feet in diameter by 9 feet high. To improve the shielding effectiveness of the 10 ton concrete modules, they are filled with up to 20,000 pounds of cemented low level waste providing a shield wall capable of reducing 2.5 R/hr contact dose rates at the box surfaces to less than 2.5 mr/hr outside the shield wall. The 40 foot by 140 foot Sprung Structure is designed to open and permit mobile crane access for placement of heavy loads within the building.

The decontamination of the Chemical Process Cell will allow it to provide temporary storage for about 300 canisters of borosilicate high level waste glass before their transport to a federal repository. Work will also continue on the remainder of the primary cells. The Process Mechanical Cell, the fuel chopping area, and the General Purpose Cell, used to support the movement of chopped fuel for dissolution are the next challenges. Decontamination of these areas will be performed using specially designed track-mounted robots capable of performing functions not permitted by remotely operated cranes.

## CONTACT DECONTAMINATION

In certain plant areas, the dose rates are low enough to permit "hands on" or contact decontamination. These areas generally include the operating aisles and galleries as well as the second and third solvent extraction cycle through uranium loadout. Major areas have been decontaminated and are now ready to be refurbished for use in the low level waste water treatment system.

These cells are 60 feet high with a 400 square foot floor area. There are no cranes, manipulators, or windows in the cells further complicating operations there. Access to these cells is limited to a 6 foot square hatch in the ceiling and a personnel door at the bottom. A large containment tent was installed at the top of cells to permit them to be opened for decontamination without spreading the radioactive contaminants to adjacent clean plant work areas.

One cell contained the solvent recovery system and another housed the product purification system. Nineteen large vessels weighing as much as 7,000 pounds and 3 miles of piping have been removed and disposed of.

The solvent recovery system contained in Extraction Cell 3 (XC-3) was decontaminated using manned entry and manual dismantling of equipment. Personnel were transported into the cell using an electrically operated scaffold system called a Spider. Following equipment removal the walls and floor of the cell were scrubbed using an alkaline foam followed by high pressure water washes. Every effort was made to limit the use of water during final decontamination. In the Product Purification Cell, an area similar in size to XC-3, dramatic improvements in productivity were made possible by using a hydraulic manipulator designed to reach deep into the cell and cut piping systems. Not only did this tool prove effective in improving crew productivity but reduced the number of people assigned to the work crew, since manned entry into the work area was not required.

## CLEANUP PROGRESS

In addition to the record of accomplishment in the process cell areas, significant progress has been made in the plant support areas. The analytical laboratories have been decontaminated and refurbished to provide facilities necessary to support high level waste operations.

The entire cleanup project is moving at a fast pace. When Westinghouse accepted operational control of the site in February 1982, only 6 percent of the plant area could be classified as free of radioactive contamination. Only three years later, 43 percent of the 33,000 m<sup>2</sup> facility can be classified as clean. Westinghouse is not only demonstrating that nuclear facility decommissioning is technically possible, but is also developing the methods and tools to do it safely.

## THE FINAL RESTING PLACE OF A CESIUM 137 GAUGE

James L. McNees, Kirksey E. Whatley, and Cornelius Maryland  
Bureau of Radiological Health  
State Office Building  
Montgomery, Alabama 36130

### ABSTRACT

In April of 1985 an industry reported the loss of four industrial gauges that contained cesium 137. The State of Alabama surveyed eight scrap iron yards where the devices were believed to have been taken and found one of the four. The State then requested aerial monitoring assistance from the U.S. Department of Energy. In July 1985 the DOE helicopter located an area of cesium 137 contamination in a waste slag dump yard. Examination of the waste slag dump showed maximum exposure levels of approximately 100 microR per hour and maximum soil contamination of  $4.9 \times 10^{-4}$  microcuries per gram. The area of contamination extended over 13,000 square feet. Erosion of cesium-containing waste slag was identified as the greatest potential health hazard associated with the contamination. The cost-benefit of spending \$500,000+ to remove the material to a commercial burial site is at least questionable. Consideration is being given to a request by the owners of the property for burial on site.

# AUTHOR INDEX

Page	Page
Allen, Jere K. ....	403
Baumann, B. L. ....	459
Beck, William L. ....	505
Bledsoe, J. L. ....	543
Boerner, A. J. ....	301
Bollenbacher, Michael ....	533
Bopp, C. D. ....	389
Borak, Thomas B. ....	53, 339, 359
Borden, William C. ....	197, 563
Bramlett, E. ....	251, 321
Branagan, Edward F., Jr. ....	437
Brosey, B. ....	369
Brown, Jeffrey A. ....	295, 299
Caniano, R. J. ....	537
Carpenter, Dena ....	53
Case, Marilyn J. ....	535
Chattin, J. C. ....	95
Chepulis, P. ....	513
Christian, K. L. ....	211
Chu, L. ....	513
Cole, Leslie W. ....	81, 533
Collins, E. D. ....	389
Congel, Frank J. ....	439
Copenhaver, E. D. ....	407
Cotter, S. J. ....	125
Deming, E. J. ....	301, 337
Dendahl, Katherine H. ....	295
Denham, D. H. ....	565
DeVore, J. R. ....	241
Distenfeld, C. H. ....	135, 369
Dolenc, Max R. ....	535
Dooley, M. A. ....	251, 321
Dunstan, G. ....	431
Dziuk, Timothy W. ....	181
Ed, D. ....	513
Emswiler, T. R. ....	537
Eng, Jeanette ....	51, 303, 497
Etter, Stephen D. ....	181
Farmer, A. J. ....	389
Feldman, Joyce ....	161, 417
Ferguson, R. R. ....	423
Fields, D. E. ....	115
Frank, G. W. ....	311
Glenn, R. Dan ....	211, 297
Golden, M. P. ....	471
Gonsalves, Laura L. ....	53
Grant, T. J. ....	461
Gross, D. L. ....	143
Hale, V. Q. ....	105
Hamel, M. ....	513
Hardeman, James C., Jr. ....	539
Haywood, Fred F. ....	227
Heine, W. F. ....	487
Hermetz, Rudy E. ....	235
Hickey, C. R. ....	211
Hinds, S. Scott ....	229
Hively, L. M. ....	543
Hoffman, Dean ....	575
Hoffman, L. G. ....	199
Holladay, D. W. ....	389
Hopper, R. ....	303
Hoyt, Kenneth R. ....	229
Huang, S. ....	143
Igarashi, H. ....	369
Irving, J. T. ....	261
Jacobs, D. G. ....	423
Jensen, Larry ....	161
Johnson, J. K. ....	389
Johnson, Nels R. ....	297
Jones, E. D. ....	271
Justus, A. L. ....	347
Kark, J. B. ....	359
Kaye, Mark E. ....	197
Kearney, P. D. ....	53, 339, 359
Kennedy, W. E. Jr., ....	19, 37
Kharkar, D. P. ....	431
Kimbrough, C. W. ....	423
Kirsch, G. E. ....	537
Kniazewycz, B. G. ....	261
Kocher, David ....	9
Land, Roger R. ....	563
Lazo, Edward N. ....	505
Leventhal, L. ....	431
Lichtman, Stanley ....	1
Long, D. C. ....	261
Lynt, Richard ....	505
Maan, M. A. ....	69
Marchetti, S. ....	331, 575
Maryland, Cornelius ....	579
McDougall, Michael R. ....	505
McNees, James L. ....	579
McVay, C. W. ....	331
Merchant, Dale J. ....	253
Merry-Libby, P. ....	217
Meyer, Bruce A. ....	293
Meyer, H. R. ....	171, 199
Milakovic, J. ....	311
Miller, C. E. ....	95, 459
Miller, C. H. ....	389
Miller, C. W. ....	125
Moe, H. J. ....	39
Mohaupt, T. H. ....	251, 321
Molello, Stephen A. ....	191
Moore, Robert E. ....	151
Murphy, G. L. ....	537
Nair, R. P. ....	69

	Page
Napier, B. A. ....	19, 37
Osborne, T. W. ....	443
Ostrow, S. L. ....	461
Owen, Park T. ....	89
Oztunali, O. I. ....	461
Parzyck, D. C. ....	407
Pavelek, M. D., II ....	229
Penfold, L. ....	431
Phillips, E. C. ....	471
Polehn, J. L. ....	63, 523
Potter, R. F. ....	95
Powers, B. A. ....	389
Prewett, Stephen V. ....	81
Provencher, Richard ....	339
Ragland, C. ....	401
Rao, C. D. ....	181
Rasmussen, O. R. ....	379
Reed, R. P. ....	63, 523
Robertson, Richard C. ....	299
Robinet, M. J. ....	39, 347
Rodgers, D. ....	431
Ronca-Battista, Melinda ....	1, 539
Rood, A. S. ....	359
Schafer, S. ....	69
Schauss, Richard D. ....	553
Sjoreen, A. L. ....	125, 543
Skinner, D. J. ....	171, 199
Slobodien, M. J. ....	311, 553
Smith, Gregory ....	339
Solday, J. K. ....	37
Speer, D. R. ....	105, 487
Steener, R. D. ....	565
Stimmel, J. R. ....	331
Tarpinian, James E. ....	253
Terpilak, M. S. ....	443
Till, John E. ....	151
Tupin, E. A. ....	443
Vallem, R. J. ....	135
Veluri, V. R. ....	39
Vesper, Bruce E. ....	251, 321
Voilleque, Paul C. ....	27
Wagoner, H. ....	261
Wallace, R. G. ....	63, 523
Wang, J. ....	217
Washburn, D. K. ....	565
Welty, T. ....	401
West, L. ....	401
Whatley, Kirksey E. ....	579
White, A. Duncan ....	303, 497
Whittaker, Mark S. ....	151
Wilcox, A. D. ....	143
Wilhelm, G. ....	323
Williams, Peter A. ....	79
Wilson, G. T. ....	63, 523
Wynveen, R. A. ....	39, 347
Youd, J. ....	323