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AGING OF CONCRETE CONTAINMENT STRUCTURES IN NUCLEAR POWER PLANTS*

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Abstract

Concrete structures play a vital role in the safe operation of all light-water reactor plants in the U. S. Pertinent concrete structures are described in terms of their importance, design considerations, and materials of construction. Degradation factors which can potentially impact the ability of these structures to meet their functional and performance requirements are identified. Current inservice inspection requirements for concrete containments are summarized. A review of the performance history of the concrete components in nuclear power plants is provided. A summary is presented of the Structural Aging (SAG) Program being conducted at the Oak Ridge National Laboratory for the U. S. Nuclear Regulatory Commission. The SAG Program is addressing the aging management of safety-related concrete structures in nuclear power plants for the purpose of providing improved bases for their continued service. The program consists of a management task and three technical tasks: materials property data base, structural component assessment/repair technologies, and quantitative methodology for continued service conditions. Objectives and a summary of accomplishments under each of these tasks are presented.

INTRODUCTION

History tells us that concrete is a durable material. However, a number of factors can compromise its performance: (1) faulty design, (2) use of unsuitable materials, (3) improper workmanship, (4) exposure to aggressive environments, (5) excessive structural loads, (6) accident conditions, and (7) a combination of the above. Furthermore, aging of nuclear power plant concrete containment structures occurs with the passage of time and has the potential, if its effects are not controlled, to increase the risk to public health and safety. Many factors complicate the contribution of aging effects to the residual life of the concrete containment structures in a plant. Uncertainties arise due to the following: (1) differences in design codes

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and standards for components of different vintage; (2) lack of past measurements and records; (3) limitations in the applicability of time-dependent models for quantifying the contribution of aging to overall structure, system, or component failure; and (4) inadequacy of detection, inspection, surveillance, and maintenance methods or programs.[1]

BACKGROUND

Within the nuclear power industry, factors which affect the performance of the concrete structures due to aging or environmental stressor effects have become the subject of significant research in the last few years.[2-4] This interest is prompted by the need to quantify these effects in terms of potential loss of component integrity or function and to support current or future condition assessments in association with requests to continue the service of nuclear power plants. Since certain concrete structures (Category I) play a vital role in the safe operation of nuclear power plants,[5-8] guidelines and criteria for use in evaluating the remaining structural margins (residual life) of each structure are needed. Standardized review guidelines for near-term evaluation of operating license renewal applications may be required as early as the mid-1990's, when utilities are planning to submit initial requests.

CATEGORY I CONCRETE STRUCTURES

Design Considerations

Category I structures are those essential to the function of the safety-class systems and components, or that house, support, or protect safety-class systems or components, and whose failure could lead to loss of function of safety-class systems and components housed, supported, or protected. In addition, these structures may serve as barriers to the release of radioactive material and/or as biological shields. The basic laws that regulate the design (and construction) of nuclear power plants are contained in Part 50 of Title 10 of the *Code of Federal Regulations* (10CFR50),[9] which is clarified by Regulatory Guides, Standard Review Plans, NUREG reports, etc. "General Design Criteria" of Appendix A to 10CFR50 requires that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. "General Design Criteria 2" requires that the structures important to safety be designed to withstand the effects of natural phenomena (e.g., earthquakes, tsunamis, hurricanes, floods, seiches, and tornados) without loss of capability to perform their safety function. "General Design Criteria 4" requires that structures important to safety be able to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including loss-of-cooling accidents. Furthermore, these structures must be appropriately protected against dynamic effects including the effects of missiles, pipe whip, and flooding that may result from equipment failures and from events and conditions outside the nuclear power facility. Section III, Division 2, Subsection CC, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), first published in 1975, contains current rules for the design of concrete containments. Prior to 1975, concrete containments were designed and constructed to codes and standards developed by the American Concrete Institute such as ACI 318 (Ref. 10). Code requirements for nuclear safety-related concrete structures other than containments and reactor pressure vessels are contained in ACI 349 (Ref. 11).

Materials of Construction

Category I concrete structures are composed of several constituents which, in concert, perform more than one function, i.e., load-carrying capacity, radiation shielding, and leak tightness. Primarily, they include the following material systems: concrete, mild steel reinforcement, prestressing steel, and steel liner plate.

The concrete typically used in nuclear safety-related structures consists of Type II portland cement, fine aggregates, water, various admixtures for improving properties or performance of the concrete, and either normal-weight or heavyweight coarse aggregate. Type II portland cement has been used because of its improved sulfate resistance and reduced heat of hydration relative to the general purpose or Type I portland cement. Coarse aggregate consists of gravel, crushed gravel, or crushed stone. For those concrete structures in nuclear power plants which provide primary (biological) radiation shielding, heavyweight or dense aggregate materials (e.g., barites, limonites, magnetites, ilmenites, etc.) may have been used to reduce the section thickness requirements needed for attenuation.* The hardened concrete typically provides the compressive load capacity for a structure. Design 28-day compressive strengths for the concrete materials utilized in nuclear power plant structures have typically ranged from 21 to 41 MPa depending on the application.

Most of the mild, or conventional, reinforcing steels used in nuclear power plants to provide primary tensile and shear load resistance/transfer consist of low-alloy carbon steels rolled or drawn into standard sizes and shapes. The surfaces of the reinforcing steel are either smooth (plain) or are provided with deformations (lugs or protrusions) to increase bond strength. The minimum yield strength of this material ranges from about 270 to 415 MPa, with the 415-MPa material being most common. Conventional reinforcing steel also encompasses welded wire fabric, deformed wire, bar and rod mats, and all accessory steel components used in positioning/placing the reinforcement, e.g., seats, ties, etc.

A post-tensioning system consists of prestressing tendons, which are installed and tensioned using jacks and other devices and then anchored to hardened concrete. A number of containment structures utilize steel prestressing tendons to provide primary resistance to tensile loadings. Three major categories of prestressing systems exist depending on the type of tendon utilized: wire, strand, or bar. These materials typically have minimum ultimate tensile strengths ranging from 1035 to 1860 MPa. The tendons are installed within preplaced ducts (conduits) in the containment structure and are post-tensioned from one or both ends after the concrete has achieved sufficient strength. After tensioning, the tendons are anchored by buttonheads, wedge anchors, or nuts, depending on the prestressing system utilized. Corrosion protection is provided by filling the ducts with corrosion-inhibiting grease (unbonded) or portland cement grout (bonded). With the exception of Robinson 2 (bar tendons) and Three Mile Island 2 (strand tendons), plants that have post-tensioned containments utilize unbonded tendons. A few plants have used bonded rock anchor tendons, e.g., Ginna and Bellefonte.

Leak tightness of reinforced and post-tensioned concrete containment structures is provided by a liner system. A typical liner system is composed of steel plate stock less than 13-mm thick, joined by welding, and anchored to the concrete by studs, structural steel shapes, or other steel products. The pressurized-water reactor (PWR) containments and the "dry well" portions of

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* Applications of heavyweight concretes primarily have been associated with research reactors.

boiling-water reactor (BWR) containments are typically lined with carbon steel plate. The liner of the "wet well" of boiling-water reactor containments, as well as that of the light-water reactor (LWR) fuel pool structures, typically consists of stainless steel plates. Certain LWR facilities have used carbon steel plates clad with stainless steel for liner members. Although the liner's primary function is to provide a leaktight barrier, it also acts as part of the formwork during concrete placement and is used for supporting internal piping and equipment.

Description of Category I Concrete Structures

A myriad of concrete structures are contained as a vital part of an LWR facility to provide support, foundation, shielding, and containment functions.. Table 1 provides a general listing of safety-related concrete structures in LWR plants.

Table 1 Representative LWR Safety-Related Concrete Structures[a]

Primary Containment/Basemat	Intake Structure
BWR Reactor Building	Cooling Tower
PWR Shield Building	Spray Ponds
Containment Internal Structures	Utility or Piping Tunnels
Auxiliary Building	Part of Turbine Building (Category I Components)
Control Room/Control Building	Auxiliary Feedwater Pump House
Diesel Generator Building	Switchgear Room
Fuel Storage Facility	Unit Vent Stack
Tanks and Tank Foundation	Radwaste Building

[a]Source: "Class I Structures License Renewal Industry Report," NUMARC 90-06, Nuclear Management and Resources Council, Washington, D.C., June 1990 (draft).

The names and configurations of these structures vary somewhat from plant to plant depending on the nuclear steam supply system vendor, architect-engineer firm and owner preference. Primary containment construction types utilized in the U.S. include: steel (PWR ice condensor, PWR large dry, BWR pre-MK, BWR MK I, BWR MKII, and BWR MK III), reinforced concrete (PWR ice condensor, PWR large dry, PWR subatmospheric, BWR MK I, BWR MKII, and BWR MK III), and post-tensioned concrete (PWR large dry and BWR MKII). As noted in Table 2, a reinforced or post-tensioned concrete material system has been selected as the primary construction type for over 60% of the LWR plant containments in the U. S. For the remainder of the plants which utilize steel primary containments, reinforced concrete is relied upon for fabrication of many of the safety-related structures (see Table 1). More detailed descriptions of the primary containments and other safety-related concrete structures are provided in Refs. [7, 8, 12, 13, and 14].

Potential Degradation Factors

The longevity, or long-term performance, of Category I concrete structures is primarily a function of the durability or propensity of these structures to withstand potential degradation effects. Over the life of a nuclear power plant, changes in the properties of the structure's constituent materials will in all likelihood occur as a result of aging and environmental stressor effects. These changes in properties, however, do not have to be detrimental to the point that the

Table 2. LWR Containment Distribution by Construction Type

Plant Type	Containment Designation	Construction: Steel	Construction: Reinforced Concrete	Construction: Post-Tensioned Concrete	Total
BWR	Pre - Mark	1	–	–	1
	Mark I	22	2	–	24
	Mark II	1	5	2	8
	Mark III	2	2	–	4
PWR	Large Dry	9	11	42	62
	Ice Condenser	8	2	–	10
	Sub-Atmospheric	–	8	–	8

structure has deteriorated and is unable to meet its functional and performance requirements. In fact, it has been noted that when specifications covering concrete's production are correct and are followed, concrete will not deteriorate.[15] Concrete in many structures, however, can suffer undesirable degrees of change with time because of improper specifications, a violation of specification, or environmental stressor or aging factor effects. Table 3 summarizes primary mechanisms (factors) which can produce premature deterioration of reinforced and post-tensioned concrete structures. Reference [16] presents a good summary of potential degradation of reinforced and post-tensioned concrete structures in nuclear power plants in terms of locations, mechanisms, indications, potential problem areas, failure modes and examination methods/remedies. More detailed discussions of the potential degradation mechanisms are provided in Refs. [8, 12, and 13].

INSERVICE INSPECTION REQUIREMENTS

Licensing and regulation of the nuclear industry in the U.S. is the responsibility of the USNRC under the provisions of the Atomic Energy Act of 1954, the Energy Reorganization Act of 1974, and the National Environmental Policy Act. Title 10, Part 50 of the U.S. Code of Federal Regulations (10CFR50) requires that safety-related nuclear systems and components, including containments, comply with specific editions and addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). General Design Criteria in Appendix A of 10CFR50 specifically require containments to be designed to permit: (1) appropriate periodic inspection of all important areas, such as penetrations; (2) an appropriate surveillance program; and (3) periodic testing at containment design pressure of the leak tightness of penetrations which have resilient seals and expansion bellows. Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, of 10CFR50, prescribes periodic leakage testing requirements for containments to assure that leakage does not exceed allowable leakage rate values as specified in the plant technical specifications, and that periodic surveillance of penetrations and isolation valves is performed. Section XI of the ASME Code provides rules for inservice inspection of nuclear power plant components. Before use is mandated in the U.S., specific editions and addenda of Section XI (also Section III) must be approved by the USNRC (paragraph 55a to 10CFR50).

Table 3 Degradation Factors That Can Impact Category I Concrete Structures

Material System	Degradation Factor	Primary Manifestation
Concrete	Chemical attack	
	Efflorescence and leaching	Increased porosity
	Salt crystallization	Cracking
	Alkali-aggregate reactions[a]	Volume change/cracking
	Sulfate attack	Volume change/cracking
	Bases and acids	Increased porosity/ erosion
	Physical attack	
	Freeze/thaw cycling	Cracking/spalling
	Thermal exposure/ thermal cycling	Cracking/spalling
	Irradiation	Volume change/cracking
	Abrasion/erosion/cavitation	Section loss
	Fatigue/vibration	Cracking
Mild Steel reinforcement	Corrosion	Concrete cracking/ spalling
	Elevated temperature	Decreased yield strength
	Irradiation	Reduced ductility
	Fatigue	Bond loss
Prestressing	Corrosion (including microbiological)	Reduced section
	Elevated temperature	Reduced strength
	Irradiation	Reduced ductility
	Stress relaxation	Prestress force loss
Liner/Structural steel	Corrosion	Section loss

[a]Includes reactions of cement aggregate and carbonate aggregate.

USNRC Regulatory Guides for Inservice Inspection

Regulatory Guides (RGs) are issued by the USNRC to provide guidance for practices acceptable to the USNRC for nuclear-related activities. Pertinent guides include RG 1.35, *Inservice Inspection of Ungrouted Tendons in Concrete Containments*, its companion RG 1.35.1, *Determining Prestressed Forces for Inspection of Prestressed Concrete Containments*, and RG 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*. ASME Section XI Code Cases acceptable to the USNRC are contained in RG 1.147, *Inservice Code Case Acceptability for ASME Section XI, Division 1*.

ASME Code Requirements for Inservice Inspection

Section XI, Subsections IWE and IWL of the ASME Code address requirements for preservice examination, inservice inspection, leakage testing, repair and replacement for metallic and concrete containments, respectively. At present, neither Subsection IWE nor Subsection IWL has been endorsed by the USNRC.

Subsection IWE, although it was developed primarily for metallic containments, applies to steel liners and steel portions not backed by concrete for concrete containments and their integral attachments. Additional rules for preservice examination and inservice inspection, leakage testing, repair and replacement of containments and for other portions of a nuclear plant are contained in Subsection IWA, *General Requirements*. Subsection IWE requires periodic leakage tests as specified in Appendix J of 10CFR50 during which a general visual examination of the entire containment boundary (e.g., liner of concrete containments) is required prior to each Type A containment integrated leak rate test (three tests required in each ten year inspection interval).

Subsection IWL includes rules for the preservice and inservice examination of concrete pressure-retaining shells and shell components and for unbonded post-tensioning tendon systems, tendons, and anchorages. Preservice examination requirements for concrete include a general visual examination of the entire exposed concrete surface including painted or coated areas. For unbonded post-tensioning systems, the preservice examination includes documentation of the construction records, e.g., tendon tensioning date, initial tendon seating force, location of any tendon system defects such as missing or cracked buttonheads, and the product designation for the corrosion inhibitor. A general visual inspection of all exposed portions of the concrete surface, including painted or coated areas, tendon anchorage assembly hardware, and bottom grease caps of all vertical tendons, is required at 1, 3, and 5 years following completion of the Initial Structural Integrity Test (ISIT) and every 5 years thereafter. With minor exceptions, rules for inservice inspection of unbonded post-tensioning systems in Subsection IWL closely parallel the requirements in RGs 1.35 and 1.35.1. For inspections at 1, 3, and 5 years after the ISIT, 4 % of the population of each group (vertical, hoop, dome, and inverted U) of tendons is selected randomly with a minimum of four tendons from each group (sample size from any group need not exceed ten). Thereafter, if no abnormal degradation of the post-tensioning system is indicated, the sample size may be reduced to 2 % of the population of each group, or five tendons, whichever is less, provided at least three tendons are inspected for each group. One tendon from each group should be kept unchanged after initial selection for use as a control (common) tendon during each inspection. For each tendon inspected, lift-off tests are performed to determine tendon force. One sample tendon from each group is detensioned, and a single wire or strand removed for examination and testing, e.g., corrosion, mechanical damage, tension tests, etc. In addition, samples of corrosion protection medium are analyzed for reserve alkalinity, water content, and concentrations of water soluble chlorides, nitrates, and sulfides.

Additional information on ASME Code rules for inservice inspection of both concrete and steel containments is provided in Ref. 17.

PERFORMANCE HISTORY OF CATEGORY I CONCRETE STRUCTURES

In general, the performance of concrete materials and structures in nuclear power plants has been good. This to a large degree can be attributed to the effectiveness of the quality control/quality assurance programs in detecting potential problems (and subsequent remedial measures) prior to plant operation.[18] However, there have been several instances in nuclear power plants where the capability of concrete structures to meet future functional/ performance requirements has been challenged due to problems arising from either improper material selection, construction/design deficiencies, or environmental effects. Examples of some of the potentially more serious instances include anchorhead failures, voids under vertical tendon bearing plates, dome delaminations, and corrosion of steel tendons and rebars. Other problems such as the presence of voids or honeycomb in concrete, contaminated concrete, cold joints, cadweld (steel reinforcement connector) deficiencies, concrete cracking, higher than code-allowable concrete temperatures, materials out of specification, misplaced rebar, lower than predicted prestressing forces, post-tensioning system buttonhead deficiencies, water contaminated corrosion inhibitors, water intrusion through basemat cracks, low tensile strength of post-tensioning tendon wire material, leaching of concrete in tendon galleries, and leakage of corrosion inhibitor from tendon sheaths also have been identified.[8,12,18,19] Although several of these documented problems were not due to environmental stressors or aging factors, if not discovered they could potentially compromise integrity of the structures during an extreme event or exhibit synergistic effects with any environmental stressors or aging factors present.

STRUCTURAL AGING PROGRAM

While the performance of safety-related concrete components in nuclear power plants has been reasonably good, there is a need for improved surveillance, inspection testing, and maintenance to enhance the technical bases for assurances of continued safe operation of nuclear power plants throughout any protracted continued service period. Results of a study [12] conducted under the NRC Nuclear Plant Aging Research (NPAR) Program[21] were utilized to help formulate the Structural Aging (SAG) Program which was initiated in 1988. The SAG Program has the overall objective of preparing a handbook or report which will provide the NRC license reviewers and licensees with the following: (1) identification and evaluation of the structural degradation processes; (2) issues to be addressed under nuclear power plant continued-service reviews, as well as criteria, and their bases, for resolution of these issues; (3) identification and evaluation of relevant inservice inspection or structural assessment programs in use, or needed; and (4) methodologies required to perform current assessments and reliability-based life-predictions of safety-related concrete structures. To accomplish this objective, the SAG Program is addressing the sources of uncertainty identified earlier with respect to determination of the residual life of safety-related components or structures. Structural Aging Program activities are conducted under a management task and three major technical task areas: (1) materials property data base, (2) structural component assessment/ repair technologies, and (3) quantitative methodology for continued service determinations.

Program Management

The overall objective of the program management task is to effectively manage the technical tasks undertaken to address priority structural safety issues related to nuclear power plant continued-service applications. Primary management activities include: (1) program planning and resource allocation, (2) program monitoring and control, and (3) documentation and technology transfer. Under the first of these activities, a five-year plan was prepared.[22] and 12 subcontracts related to meeting objectives of the technical task areas have been implemented. The program monitoring and control activity primarily addresses the preparation of management reports, annual technical progress reports [Ref. (23) presents the most recent information], and participation in NRC information meetings. The major emphasis under this task has been related to documentation and technology transfer actions. These actions have included: preparation of 42 technical reports and papers, 41 formal technical presentations, and participation in 11 national or international technical committees. Technology exchange at both the national and international levels has been very active with 89 domestic and 98 foreign organizations having been contacted.

Materials Property Data Base

The objective of the materials property data base task is to develop a reference source which contains data and information on the time variation of material properties under the influence of pertinent environmental stressors and aging factors. This source will be used to assist in the prediction of potential long-term deterioration of critical structural components in nuclear power plants and to establish limits on hostile environmental exposure for these structures, i.e. establish component service life or improve the probability of a component surviving an extreme event. Primary activities under this task include the development of the Structural Materials Information Center, assemblage of materials property data, and formulation of material behavior models.

Structural Materials Information Center (SMIC)

Utilizing results of a review and assessment of materials property data bases[24] and a plan which had been prepared for development of the Structural Materials Information Center (SMIC),[25] initial formatting of SMIC has been completed and results presented in a report.[26] Contained in the report are detailed descriptions of the *Structural Materials Handbook* and the *Structural Materials Electronic Data Base* which form the SMIC.

Structural Materials Handbook

The *Structural Materials Handbook* is an expandable, hard-copy reference document that contains complete sets of data and information for each material in the SMIC. The handbook consists of four volumes that are provided in loose-leaf binders for ease of revision and updating. Volume 1 contains performance and analysis information useful for structural assessments and safety margins evaluations, for example, performance values for mechanical, thermal, physical, and other properties presented as tables, graphs, and mathematical equations. Volume 2 provides test results and data used to develop the performance values in Volume 1. Volume 3 contains material data sheets which provide general information, as well as material composition and constituent material properties, for each material system contained in the handbook. Volume 4 contains appendices describing the handbook organization, as well as updating and revision procedures.

Volumes 1, 2, and 3 of the handbook presently contain four chapters of materials property data and information, with the chapters consistent between volumes. Each material in the data base is assigned a unique seven-character material code which is used in the handbook and the electronic data base to organize materials with common characteristics. This code consists of a chapter index, a group index, a class index, and an identifier. The chapter index is used to represent the various material systems in the data base. The group index is used to arrange materials in each chapter into subsets of materials having distinguishing qualities such as common compositional traits. The class index is used to organize groups of materials with common compositional traits into subsets having a similar compositional makeup or chemistry. The identifier is used to differentiate structural materials having the same chapter, group, and class indices according to a specific concrete mix, American Society for Testing and Materials standard specification for metallic reinforcement, etc.

A wide variety of information and materials property data is collected and assembled for each material system included in the data base, for example, general description, composition, mechanical property data, etc. In setting up the data base, each material property has been identified by a unique four-digit property code[26] selected from an established set of material property categories, e.g., general information, constituent material and plastic concrete properties, mechanical properties, etc.

Associated with each entry of data (numerical results of tests) or values (results of evaluation of data) into the data base is an assessment of the quality of the entries presented in the form of a letter grade. Although the criteria for assessing the quality of data and values are somewhat subjective, five quality levels have been developed. These levels, presented, in order of descending quality, include recommended, selected, typical, provisional, and interim. The 11 criteria utilized to evaluate the quality of data and values are provided in Ref. 26.

Structural Materials Electronic Data Base

The *Structural Materials Electronic Data Base* is an electronically accessible version of the *Structural Materials Handbook.* It has been developed on an IBM-compatible personal computer using a data base management system designed specifically for maintaining and displaying properties of engineering materials. To ensure that the handbook and electronic data base are compatible, each material included in the electronic data base is identified by the same common name and material code that has been used to represent the material in the handbook. Also, each electronic data base material record contains data and information taken directly from the handbook. Due to software limitations, the electronic data base is not as comprehensive as the handbook, but it does provide an efficient means for searching the various data base files to locate materials with similar characteristics or properties.

The electronic data base management system includes two software programs: Mat.DB[27] and EnPlot.[28] Mat.DB is a menu-driven software program that employs window overlays to access data searching and editing features. It is capable of maintaining, searching, and displaying textual, tabular, and graphical information and data contained in electronic data base files. EnPlot is a software program that incorporates pop-up menus for creating and editing engineering graphs. It includes curve-fitting and scale-conversion features for preparing engineering graphs and utility features for generating output files. The graphs generated with EnPlot can be entered directly into the Mat.DB data base files. These graphs are compatible with Microsoft Word, the word processing software used to prepare the handbook. Both Mat.DB and EnPlot operate on an IBM PC, PC/XT, PC/AT, or compatible computer. System requirements include 640 K of memory, hard disk, graphics card, monitor, and DOS 3.0 or later.

Each material record in the electronic data base could include up to nine major categories of data and information: designations, specifications, composition, notes, forms, graphs, properties, classes, and rankings.[26] The user may search an entire data base file to locate materials with similar material properties. During the search each material may be screened for selected tabular data and certain property values based on comparison operators (for instance, =, >, <, >=, <=, and <>). The user may elect to perform property searches using either the International System of Units (SI) or customary units.

Data Collection

In parallel with efforts to develop the SMIC, activities are being conducted to establish materials property data for input into the SMIC. To date, two approaches have been utilized: (1) pursuing technology exchange with U. S. and foreign research establishments and (2) obtaining and testing of prototypical concrete materials.

Technology Exchange

Domestic and foreign organizations have been contacted in an effort to obtain concrete properties information for input into the data base. To date, 46 concrete, 12 metallic reinforcement, 1 prestressing steel, 1 structural steel, and 1 rubber material data bases have been developed and are contained in the SMIC. A description of the these data bases is provided in Ref. 23.

Prototypical Sample Procurement

Several U.S. utilities and concrete research organizations, as well as a national laboratory, have been contacted to pursue the possibilities of removing and testing concrete core samples from prototypical structures. These contacts have resulted in procurement of samples from the Shippingport Power Station (Battelle Pacific Northwest Laboratories), EBR-II (Argonne National Laboratory-West), Palisades and Midland Power Stations (Consumers Power Co.), Dresden and LaSalle Power Stations (Commonwealth Edison Co.), and Vallecitos Nuclear Center (General Electric). Five concrete material property data bases have been developed for the EBR-II materials. Furthermore, subcontracts have been implemented with Sargent & Lundy Engineers, Construction Technology Laboratories, Inc., and Taywood Engineering Ltd. to provide data on prototypical concrete material properties.

The purpose of the subcontract with Sargent & Lundy Engineers (Chicago, IL) is to locate, review, evaluate and provide baseline material property information regarding the concrete materials utilized in the construction of Commonwealth Edison Company's (CECO) nuclear power stations at Byron, Braidwood, Dresden, LaSalle, Quad Cities and Zion. Sargent & Lundy Engineers also will assist in the obtaining and shipment of concrete core samples to ORNL for testing. The samples will be provided from CECO facilities when modifications are made that require the coring and removal of concrete materials. In addition to the core samples, information will be provided which identifies the plant from which the sample was obtained, the location in the plant, and any available background information (constituents, mix designs, environments, etc.). To date, background information has been provided pertaining to the construction specifications and reference concrete compressive strength data (90-day) for each of the six stations.

Two subcontracts have been implemented with Construction Technology Laboratories, Inc. (Skokie, Illinois), to provide concrete material property data. Under the first activity data have been provided for three of four series of tests from a study that has been ongoing since the early 1940s to investigate the long-term performance of cements in concrete. The overall program encompassed about 500 concrete mixes fabricated using a wide variety of cement and aggregate materials. Beam and cylindrical test specimens were either moist cured, air cured, air cured followed by soaking 48 hours in water before testing, cured outdoors at Skokie, Illinois, or cured outdoors at Dallas, Texas, until testing. Periodically, at ages from 1 day to 34-years, specimens were tested to determine values of compressive strength, modulus of elasticity, and modulus of rupture as a function of materials, environment, and time. Reference 29 presents summary descriptions of the overall program, material characteristics, mixture proportions, specimen geometries, curing conditions, test methods, and available data. Under the second activity which has just initiated, selected specimens remaining from one of the test series will be tested to provide compressive and flexural strength results. Also petrographic examinations will be conducted on several of the specimens. The results of these tests will extend the period for which results are available from 34 to 42 years.

A subcontract has been implemented with Taywood Engineering Ltd. (London, England) to provide data from archived test specimens that were cast in conjunction with fabrication of several of the United Kingdom nuclear power stations. Over 100 test specimens, generally 450-mm long by 150-mm diameter and having ages from 4 to 22 years, are available from the Wylfa, Heysham, Heysham II, Hartlepool, Torness and Sizewell "B" stations. The specimens have been stored in a sealed, stable moisture state at temperatures from 20° to 90° C with some having been under sustained loading. Available baseline data for the six concretes include compressive strength (up to 1 year), thermal expansion, thermal conductivity, and details on constituent materials and mixture proportions. Also, a limited number of cylinders for the Heysham and Heysham II concretes have been tested for elastic and creep recovery, and for compressive strength at ages to 4 years. Elastic modulus and creep results are available for the Wylfa concrete up to an age of 12.5 years. Twenty-nine specimens of the over 100 available have been selected for testing to provide information on the long-term performance of nuclear grade concretes. Variables being investigated include age of specimen, concrete mix design, loaded or unloaded while curing, and storage temperature. After being subjected to a series of nondestructive tests (density, ultrasonic pulse velocity, Schmidt hammer, surface hardness, and dynamic modulus of elasticity), the specimens are loaded to one-third their estimated compressive strength to determine the static modulus of elasticity, followed by determination of the compressive strength. These tests are being complemented by results from inspections of power station buildings to investigate the state of concrete and rebar in-situ. Also, the results from routine inspections of withdrawn tendons at power stations are being analyzed to determine the durability of nongrouted post-tensioning systems.

Material Behavior Modeling

The main activity under this subtask has been related to the work done at the National Institute of Standards and Technology (Gaithersburg, Maryland) to identify and evaluate models and accelerated aging techniques and methodologies which can be used in making predictions of the remaining service life of concrete in nuclear power plants. The program consisted of two major tasks. The first task involved an evaluation of models which possibly can be used to predict the remaining service life of concrete exposed to major environmental stressors and aging factors potentially encountered in nuclear power plant facilities. Degradation processes considered included corrosion of steel reinforcement, sulfate attack, alkali-aggregate reactions, frost attack, leaching, irradiation, salt crystallization and microbiological attack. Each of these processes was reviewed based on considerations of its mechanism, likelihood of occurrence, manifestations, and detectability. Models identified for each process were evaluated based on considerations of (1) their basis (for example, theoretical, empirical, or some combination), (2) correctness of assumptions used in their derivation, (3) availability of data to perform an evaluation, (4) their applicability to the problem, and (5) degree of quantitativeness of their predictions.

The second task involved a review and evaluation of accelerated aging techniques and tests which can either provide data for service life models or which by themselves can be used to predict the service life or performance of reinforced concrete materials. In comparison to predicting the life of new concretes, few studies were identified on predicting the remaining service life of in-service concretes. Most of the reported studies dealt with corrosion of steel reinforcement, reflecting the magnitude and seriousness of this problem. Methods which are often used for predicting the service lives of construction materials include: (1) estimates based on experience, (2) deductions from performance of similar materials, (3) accelerated testing, (4) applications of reliability and stochastic concepts, and (5) mathematical modeling based on the chemistry and physics of degradation processes. The most promising approach for predicting the remaining service life of concrete involves the application of mathematical models of the degradation processes. Theoretical models need to be developed, however, rather than relying solely on empirical models. Advantages of this approach include more reliable predictions, far less data are needed, and the theoretical models would have wider applications, e.g., applicable to a broad range of environmental conditions. Deterministic and stochastic models should be combined to give realistic predictions of the service life of an engineering material. Purely stochastic models will probably have limited application because of the lack of adequate data bases to determine the statistical parameters. Accelerated tests do not provide a direct method for making the life predictions, but can be useful in obtaining data required to support the use of analytical models. Results of this activity are presented in Ref. 30.

Structural Component Assessment/Repair Technology

The objectives of this task are to: (1) develop a systematic methodology which can be used to make quantitative assessments of the presence, magnitude, and significance of any environmental stressors or aging factors which adversely impact the durability of safety-related concrete structures in nuclear power plants; and (2) provide recommended inservice inspection or sampling procedures which can be utilized to develop the data required both for evaluating the current condition of concrete structures as well as trending the performance of these components. Primary activities under this task include development of a structural aging assessment methodology for concrete structures in nuclear power plants, review and evaluation of inservice inspection and structural integrity assessment methods for detection and quantification of potential deterioration phenomena in concrete structures, and evaluation of remedial/preventative measures considerations for concrete structures.

Structural Aging Assessment Methodology

Under a subcontract with Multiple Dynamics Corporation (Southfield, Mich.), a report has been published which identifies safety-related concrete structures in LWR plants as well as the degradation factors which can impact the performance of these structures.[19] Results of this study will assist in providing a logical basis for identifying the critical structural elements for evaluation. Pertinent sections of the report are summarized below, for instance, concrete component description/classification system, degradation factor significance classification, and structural aging assessment methodology.

Typical safety-related concrete structures at LWR facilities are identified, described, and their design and construction requirements and primary materials of construction designated. The relative importance of the structure's subelements, safety significance of each Category I structure, and influences of environmental exposure are presented in terms of numerical rating systems. The importance of a subelement to a specific structure is related to its impact on the ability of the structure to meet its functional and performance requirements. The rating system established for structural subelement importance is based on a 1 to 10 scale, with 10 being highest. The safety significance of the subelement is assessed based on the importance of the safety function (developed in compliance with 10CFR regulations) the subelement may be (or is) required to perform, as well as the number of safety functions it must meet. Each subelement is ranked on a scale of 0 to 10, with 10 being highest, using ranking criteria that have been established. Since environmental effects are highly influential on the service life of concrete structures, an environmental exposure classification procedure was also developed. A rating system was established to incorporate environmental exposure conditions and is based on (1) historical environmental data, (2) exposure conditions for all surfaces of the structure, (3) accessibility of the structure's exposed surfaces for inspection, and (4) quantity/severity of the specific environmental conditions to which it is exposed. Seven environmental exposure categories, ranging from most severe (subterranean) to mildest (controlled interior), were identified and a rating system developed by comparing each of the environments to one another and identifying their relative significance. The resulting environmental rating system is based on a 1 to 10 scale, with 10 being most aggressive.

Potential degradation or aging factors which could affect the performance of the Category I concrete structures during their lifetime were identified. The significance of a particular degradation factor is evaluated for a particular structure/subelement in terms of (1) its effect on overall structural integrity, (2) environmental conditions present, and (3) materials of construction. The effect of a degradation factor on structural integrity includes its rate of attack, inspectability/early identification, repairability, and ultimate impact on the structure. Because of the variability in likelihood of occurrence of degradation to concrete structures in U.S. LWR plants due to design differences, material utilization, geographical location, etc., the grading system for degradation factors is stated in terms of a possible range of values. Pertinent degradation factor grading values are selected from the ranges of possible values, based on site-specific characteristics. The resulting degradation factor grading values for the individually evaluated subelement (between 1 and 10) are then combined into a single degradation factor significance value by summing the degradation factor grading values and dividing by the number of degradation factors, for instance,

$$DFS = \left(\sum_{i=1}^{n} DFG_i\right) / n, \qquad (1)$$

where,

DFS = degradation factor significance value, rounded to nearest integer,
DFG = degradation factor grading value, and
n = number of degradation factors, up to a total of three.

The structural aging assessment methodology that has been developed is founded on several criteria: relation of subelements to overall importance of the parent safety-related structure, safety significance of the structure as a whole, influence of applied environment, and possibility of occurrence as well as end result of degradation. Determination of the relative ranks of the Category I structures and their subelements is based on the weighted contributions of the four criteria discussed earlier: (1) structural importance of subelements, (2) safety significance, (3) environmental exposure, and (4) degradation factor significance. A subelement rank within each Category I structure is determined as follows:

$$SR = w_1(I) + w_2(SS) + w_3(DEG), \tag{2}$$

where

SR	= subelement rank,
I	= subelement importance,
SS	= safety significance
DEG	= (EE + DFS)/2, rounded to nearest integer,
EE	= environmental exposure,
DFS	= degradation factor significance [Eq. (1)], and
w_1, w_2, w_3	= weighting factors.

Use of weighting factors (1 to 10, with 10 highest) permits certain components of Eq. (2) to be emphasized. Since the degradation factor significance (DFS) was considered to be heavily influenced by the environmental exposure (EE), these two criteria were combined, averaged, and rounded to the nearest whole integer. Based on a sensitivity analysis,[19] recommended values for w_1, w_2, and w_3 are 4, 9, and 7, respectively, resulting in a possible range of subelement ranks between 20 and 200. The cumulative rank for each Category I concrete structure is determined as follows:

$$CR = \sum_{i=1}^{N} SR_i / N, \tag{3}$$

where

CR = cumulative rank,
SR = subelement rank, and
N = number of subelements for the particular primary structure.

Application of Eq. (3) ensures that the cumulative rank of a Category I concrete structure is based on aging importance rather than total number of subelements. Data input into the structural aging assessment methodology has been organized around R:Base for DOS computer software. The methodology has been applied to a PWR with large dry metal containment, a BWR with reinforced concrete Mark II containment, and PWR with large dry prestressed concrete containment.[19] The highest ranking primary structure for each of these plants was found to be the shield building, containment vessel, and containment vessel, respectively.

NDE/Sampling Inspection Technology

Basic activities under this subtask have been related to an evaluation of nondestructive and sampling/analysis procedures which are available for performing inservice inspections of the critical concrete components in nuclear power plants. These activities have been conducted through subcontracts with Construction Technology Laboratories, Inc. (CTL) and the National Institute of Standards and Technology (NIST).

The overall objective of the subcontract with CTL was related to a review and assessment of inservice inspection techniques and methodologies for application to concrete structures in nuclear power plants. Both direct and indirect methods used to detect degradation of concrete materials have been reviewed. Direct techniques generally involve a visual inspection of the structure, removal/testing/analysis of material, or a combination. Periodic visual examinations of exposed concrete provides a rapid and effective means for identifying and defining areas of distress, for example, cracking, spalling, and volume change. In areas exhibiting extensive deterioration, or when quantitative results are desired, core samples can be removed for strength testing and petrographic examination. The indirect techniques measure some property of concrete from which an estimate of concrete strength, elastic behavior, or extent of degradation can be made through correlations that have been developed. Several potential nondestructive techniques for evaluating concrete materials and structures include: (1) audio, (2) electric, (3) impulse radar, (4) infrared thermography, (5) magnetic, (6) microscopic refraction, (7) modal analysis, (8) nuclear, (9) radiography, (10) rebound hammer, (11) ultrasonic, and (12) pulse echo. In addition to core sampling, potential destructive testing techniques that can be used to evaluate concrete materials include (1) air permeability, (2) break-off, (3) chemical, (4) probe penetration, and (5) pullout. Contained in the final report for this activity are: (1) a description of typical safety-related concrete components in a nuclear power plant (reactor containment buildings, containment base mats, biological shield walls and buildings, and auxiliary buildings); (2) a listing of potential environmental stressors, as well as their mechanisms and manifestations, which can impact the performance of the concrete, mild steel reinforcement, and prestressing steel materials; (3) reviews in the form of capabilities, accuracies, and limitations, of available nondestructive and destructive techniques that may be utilized in the assessment of concrete components; (4) current in-service inspection methodologies that have been utilized in the assessment of concrete components in civil engineering structures, e. g., routine and periodic inspections, condition surveys, and examples of applications of several of the nondestructive testing techniques; (5) recommended testing methods for use in the detection of the occurrence of the effects of several of the potential degradation factors; and (6) relatively new techniques that potentially have application in the detection of degradation of concrete, e.g., magnetic (leakage flux, nuclear magnetic resonance), electrical (capacitance, polarization resistance, half-cell potential using impulse radar), ultraviolet radiation and finite-element analysis methods.[31]

Under the subcontract with the NIST, correlation curves and other statistical data are being developed for selected nondestructive testing techniques. Monovariant linear regression analyses (Mandel's method[32]) are being performed on data obtained from publications on selected nondestructive testing techniques, i.e. break-off, pullout, rebound hammer, ultrasonic pulse velocity, and probe penetration. These methods were selected since they comprise an overwhelming majority of the nondestructive tests performed. For each of the nondestructive techniques investigated, the data identified were subdivided by coarse aggregate type and coarse aggregate content (by weight). This subdivision was based on results provided in the literature indicating that the techniques are influenced by aggregate characteristics, e.g. the pullout and break-off tests are dependent on the aggregate type and maximum aggregate size and the probe penetration and rebound hammer results are influenced by the aggregate hardness. Unfortunately insufficient data were available to further subdivide the data by maximum coarse aggregate size. Since all of the data used in the study were not the result of careful experimentation, a quality rating system was developed for application to each of the data sources. The rating system utilizes nine criteria, e.g., completeness of material description, type of input, completeness of data, completeness of resources, quality of resources, consistency of results, precision and scatter, uncertainty and bias, and statistical methods used. Each data source used was given a rating from A to D (A being highest), based on these criteria. Results developed under this activity will facilitate the evaluation of the in-situ concrete strength results based on the use of nondestructive techniques when only limited destructive information is available. Also, in the absence of destructive test results, correlations developed for the nondestructive test methods can be used as guidelines to estimate the in-situ strength of concrete based solely on nondestructive evaluation results.

Remedial/Preventative Measures Considerations

Activities under this subtask are related to an assessment of repair procedures for concrete material/structural systems and establishment of criteria for their utilization. Techniques which can be used to mitigate the effects of environmental stressors or aging factors are being identified. Current work under this subtask is through subcontracts with Wiss, Janney, Elstner (WJE) Associates (Northbrook, Illinois), Taywood Engineering Limited (TEL) (London, England), Howard University (Washington, D. C.), and CORRPRO Companies (Medina, Ohio).

The overall objective of the WJE subcontract is to identify, describe, and quantify the effectiveness of techniques and materials which are available for use in the repair of civil engineering concrete structures. A state-of-the-art manual on repair of deteriorated concrete structures is being prepared which identifies repair methodologies and materials for repair, and ranks repair procedures based on historical performance. Basic components of the manual will include discussions of: when a specific repair technique is applicable, e.g. specific crack sizes; how the techniques or materials are used, e.g. injection, routing; how to evaluate and test a repair; how to maintain the repair after it has been installed; the expected life of the repair technique; methods for determining when a repair has failed; and methods for re-repair. An important activity under this subcontract has been the preparation of a questionnaire which has been sent to the utilities. The questionnaire requests information on repairs that have been undertaken of concrete structures, research investigations on repair materials, and studies on the long-term effectiveness of repair procedures that have been utilized. Responses to the questionnaire are just beginning to be provided so no trending or quantitative information is available at present.

A companion research effort with the same general overall objective as the WJE subcontract is being conducted by TEL. The distinction between the two efforts is that TEL is addressing the assessment of repair procedures from a European perspective. In the U. S. the primary repair activities have addressed roads and bridge structures, whereas in Europe substantial activities have taken place with respect to buildings and other engineered structures. Damage occurring from carbonation and chloride presence are important sources of concrete distress in Europe and several remedial programs have been developed to address the resulting corrosion problems. For carbonation, the emphasis has been placed on anti-carbonation surface treatments, protective properties of patch materials, and the durability/compatibility of these materials. For chloride attack, efforts are underway to provide an improved understanding of the corrosion mechanisms, the mechanism of incipient anode development, and the use of cathodic protection to overcome the problem. Taywood Engineering Limited is currently involved in a collaborative European (BRITE/EURATOM) project addressing repair of reinforced concrete structures. As well as managing the project, TEL has a number of specific technical tasks, the principal activity involving the development of standardized performance tests and criteria to provide guidance for selection of repair materials that will be durable. Other tasks involve the preparation of a state-of-the-art report summarizing repair activities used by member states of the Commission of European Communities and investigation of the effect of repair techniques on structural performance. A second BRITE/EURATOM project, completed in 1991, investigated methods that can be used to extend the lifetime of structures, either during the construction stage, by the incorporation of chemical admixtures or alternative cementing materials, or post-construction use of surface coatings and treatments. Results of TEL's participation in these programs will be used to provide an assessment of European repair practices. Specific topics being addressed include identification of the repair procedures that have been utilized, establishment of criteria used in the selection of a particular repair procedure, and an assessment of the effectiveness of the various techniques as determined through in-situ evaluations (test methods) or performance history. Results of this activity will be available shortly.

Under a subcontract with Howard University, a systematic methodology for repair, restoration, and rehabilitation of concrete structures in nuclear power plants is being developed. Also to be provided is an expandable manual and user-friendly computer program for use in the inspection of concrete structures in nuclear power plants. A similar computerized rating system has been developed for evaluating general civil works concrete structures, that is, BRAIN (Building Rating Analysis and Investigation System). The first of two planned phases for this program is in progress and is: establishing the differences and similarities between existing repair prioritization systems that have been developed for bridge and building structures; determining the applicability of these systems to nuclear power plant concrete structures; and establishing an approach for development of a repair, restoration, and rehabilitation methodology for nuclear power plant concrete structures. Input developed under the first phase of the study will be used in the second phase to develop a manual and user-friendly computer program. When completed, results of this work will complement the aging assessment methodology developed by Multiple Dynamics Corporation (described previously) and provide a link between the materials studies, nondestructive evaluations, repair technology assessments, and quantitative methodology for current and reliability-based future condition assessment activities.

The overall objective of the subcontract with CORRPRO Companies is to provide a state-of-the-art report addressing corrosion of reinforced concrete structures, where reinforcement includes both mild steel reinforcement and post-tensioning tendons. The report will contain a brief overview relative to the three primary mechanisms leading to corrosion of steel embedded in concrete: (1) chloride ions, (2) carbonation, and (3) stray electrical currents. However, the major emphasis of the report will be to discuss: (a) the potential for stray electrical current corrosion in reinforced concrete structures contained as a part of a nuclear power plant, i.e., equipment foundations, basemat, containment building, balance-of-plant structures, etc.; and (b) cathodic protection systems for use with reinforced concrete structures in nuclear power plants. Specific areas being covered relative to stray electrical current corrosion include: potential sources, methods for detection, factors that affect the rate of corrosion and threshold limits below which stray electrical currents may not be a problem, impact on structural performance, mitigation techniques and their effectiveness, and any synergistic effects (e.g., it has been indicated in at least one reference that stray currents accelerate alkali-aggregate reactions in reinforced concrete structures). With respect to cathodic protection systems, pros and cons for their use to mitigate corrosion of reinforced concrete structures are being addressed, as well as criteria for their application, their overall effectiveness, and any limitations on their use such as the potential to accelerate or induce corrosion. As this work has just initiated, no results are presently available.

Quantitative Methodology for Continued-Service Determinations

The overall objective of this task is to develop a methodology which can be used for performing condition assessments and making reliability-based life predictions of critical safety-related concrete structures in nuclear power plants. The methodology will integrate information on degradation and damage accumulation, environmental factors, and load history into a decision tool that will enable a quantitative measure of structural reliability and performance under projected future service conditions based on an assessment of the existing structure. When completed, the methodology will take into account the stochastic nature of past and future loads due to operating conditions and the environment, randomness in those physical processes and environmental stressors that may lead to degradation in strength, and uncertainty in non-destructive evaluation techniques. Activities associated with this task include: (1) identification and appraisal of existing condition assessment methods and damage prediction models, (2) assembly of pertinent data for use in the predictive models, (3) development of reliability-based condition assessment methodologies for the analysis of current and future reliability, and (4) validation of condition assessment using laboratory or prototypical structures data. Results to date are discussed below and include development of the methodology for use in condition assessments and reliability-based life prediction of concrete structures in nuclear power plants. A more detailed discussion of the methodology is provided in Ref. [33].

Reliability-Based Condition Assessment

Concrete structures may be affected by aging, or changes in strength and stiffness beyond the baseline conditions assumed in initial structural design. The evaluation of safety-related concrete structures for continued service should provide quantitative evidence that their strength is sufficient to withstand future extreme events within the proposed service period with a level of reliability sufficient for public safety. Structural loads, engineering material properties, and strength degradation mechanisms are random in nature. Thus, time-dependent reliability analysis can provide a framework for performing condition assessments of existing structures and for determining whether inservice inspection/maintenance is required to maintain reliability and performance at the desired level.

The strength of structural members and components can be described statistically by data gathered in research over the past decade to develop improved bases for structural design of new reinforced concrete structures.[34,35] Time-dependent changes in concrete strength due to aging phenomena were not considered in developing these statistics, and they are not directly applicable to the evaluation of existing, possibly degraded, structures with a given service history.[36] Some of the environmental stressors that may affect the strength or deformations of reinforced concrete structures in nuclear power plants include sulfate or acid attack on the concrete, alkali-aggregate reactions within the concrete, freeze-thaw cycling, temperature and irradiation affects, corrosion of reinforcement, and detensioning of prestressing tendons due to relaxation, anchorage failure, or creep in the concrete.

The statistical descriptions of concrete structure strength must account for such aging effects. This can be done by modeling the structural resistance as a time-dependent function,

$$R(t) = R_0\, g(t) \qquad (4)$$

in which R_o is the initial resistance (comparable to Table 4) and g(t) is a time-dependent degradation function defining the fraction of initial strength remaining at time, t. Conceptually, a function g(t) can be associated with each environmental stressor,[37] and most significant degradation mechanisms have been identified, at least qualitatively.

Structural loads occur randomly in time and are random in their intensity. If the load intensity varies slowly during the load event, its effect on the structure is essentially static. Moreover, the duration of significant load events usually is short, and such events occupy only a small fraction of the total life of a structure. With these assumptions, structural loads can be modeled as a sequence of pulses, the occurrence of which is described by a Poisson process with mean rate of occurrence, λ, with random intensity S_j and duration τ. A typical sample function of such a load process is illustrated in Figure 1. Many of the loads for which nuclear power plant structures are designed can be modeled by such processes.[38] A summary of the parameters describing several load processes is given in Table 4. Some of these were determined through a consensus estimation survey.

Table 4. Load process parameters.

	Mean[m]	C.O.V.	pdf	λ (yr^{-1})	τ
Dead Load	$1.00D_n$	0.0	---	---	40 years
Live Load	$0.40\ L_n$	0.50	Type I	0.5	3 months
Acc. Pres. Load	$0.80P_a$	0.20	Type I	10^{-4}	30 min.
Earthquake Load	$0.08E_{ss}$	0.85	Type II	0.11	30 sec.

[m]D_n, L_n, P_n, and P_a are nominal loads and E_{ss} is safe shut-down earthquake load specified for design.

Time-Dependent Reliability Analysis

The reliability analysis of a structure can be visualized by the sample functions of time-dependent strength and loads illustrated in Figure 1. We assume that g(t) is independent of the load history, and arises from deterioration mechanisms such as corrosion and sulfate attack.
The reliability function, L(t), is defined as the probability that the structure survives during interval of time (0,t). If n events occur within time interval (0,t), the reliability function for a structural component can be represented as:

$$L(t)=P\left[R(t_1)\rangle S_1\cap R(t_2)\rangle S_2\cap\cdots\cap R(t_n)\rangle S_n\right] \tag{5}$$

Taking into account the randomness in the number of loads and the times at which they occur as well as in the initial strength, the reliability function becomes,[39]

$$L(t)=\int_0^\infty \exp\left[-\lambda\left[t-\int_0^t F_s(r\bullet g(t))dt\right]\right]f_{R_o}(r)dr \tag{6}$$

in which $f_{R_o}(r)$ = probability density function (pdf) of the initial strength R_o. The limit state

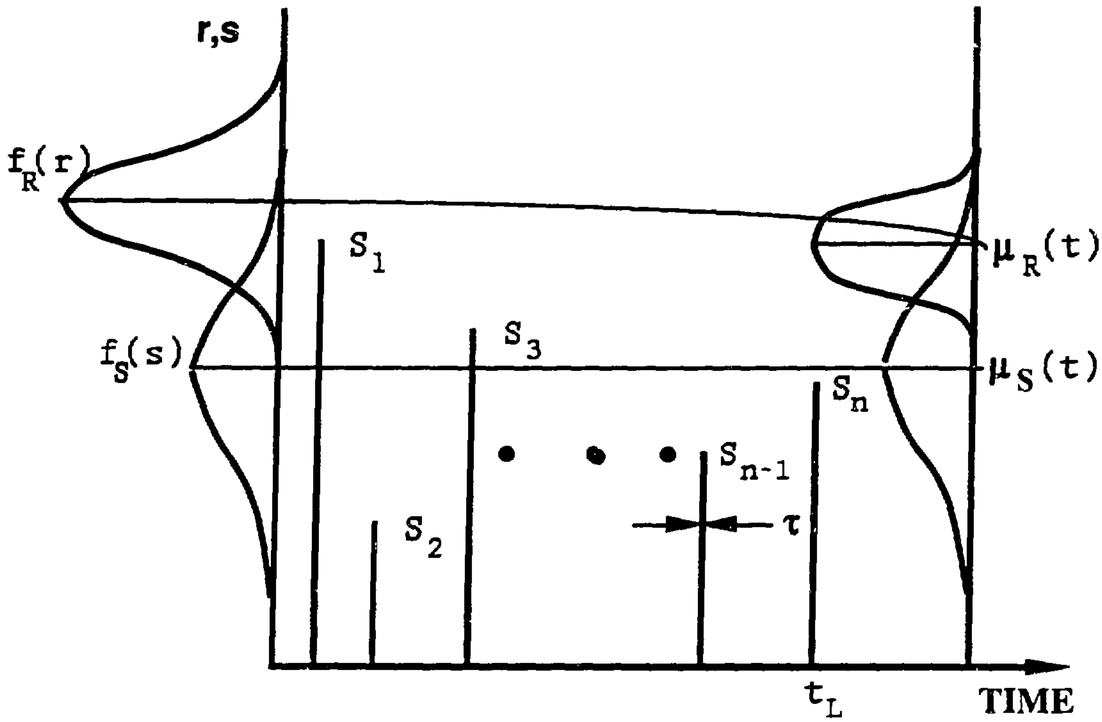

Figure 1. Schematic Representation of Load Process and Degradation of Resistance.

probability or probability of failure during (0,t) is,

$$F(t) = 1 - L(t) \tag{7}$$

The hazard function, h(t), is defined as the probability of failure within time interval (t,t+dt), given that the component has survived up to time t. This conditional probability can be expressed as,

$$h(t) = -\frac{d}{dt}\ln L(t) \tag{8}$$

The reliability function can be determined from h(t) as,

$$L(t)=\exp\left[-\int_0^t h(\xi)\,d\xi\right] \tag{9}$$

When structural failure occurs due to aging, h(t) increases with time. The common assumption in some time-dependent reliability studies that the failure rate is linear gives rise to a Rayleigh distribution for the limit state probability, F(t). As will be shown subsequently, this assumption may not be valid for concrete structures in nuclear plants.

The methods summarized above have been extended to structures subjected to combinations of structural load processes and to structural systems.[39] The reliability function has a similar appearance to that in Eq. 6, but the outer integral on resistance increases in dimension in accordance with the number of components in the system. The system reliability is evaluated by Monte Carlo simulation, using an adaptive importance sampling technique[40] to enhance the efficiency of the simulation.

Illustration of Time-Dependent Reliability

The effect of degradation in component strength on component reliability function is illustrated using several simple parametric representations of time-dependent strength summarized in Table 5. Additional data to define the time-dependent resistance are expected to become available later

Table 5. Degradation model.

Shape of the Degradation Function	Corresponding Degradation Mechanism
Linear: $g(t) = 1 - at$	Corrosion
Parabolic: $g(t) = 1 - at^2$	Sulfate Attack
Square Root: $g(t) = 1 - a\sqrt{t}$	Diffusion Controlled Degradation

in the Structural Aging Program. The sensitivity study herein identifies some of the more important parameters for condition assessment purposes. Each reliability analysis is carried out for a period of 60 years, the sum of the initial service period of 40 years and a tentative 20-year period of continued service. The degradation is defined with reference to the residual strength at 40 years; e.g., $g(40) = 0.8$ means that 80 percent of the initial strength remains at 40 years.

Components were designed using three design requirements for concrete structures in nuclear plants:[39]

$$0.9\, R_n = 1.4\, D_n + 1.7\, L_n \tag{10}$$

$$0.9\, R_n = -\,0.9\, D_n + 1.5\, P_a \tag{11}$$

$$0.9 R_n = D_n + L_n + E_{\xi\xi} \tag{12}$$

in which D_n, L_n, P_a, and $E_{\xi\xi}$ are nominal dead load, live load, accidental pressure load, and structural action due to the safe-shutdown earthquake, respectively. Each of these loads has different temporal characteristics, as summarized in Table 4.

The effect of the general characteristics of the degradation function on F(t) is presented in Figs. 2(a) and 2(b) for the load combination involving dead and live load. Up to 40 years, the failure probability associated with the square root model is the highest. However, after 40 years, the failure probability associated with the parabolic model increases rapidly when the degradation rate increases. Note from Fig. 2(b) the effect of neglecting strength degradation entirely in a time-dependent reliability assessment.

Hazard functions, h(t), associated with the reliability analyses presented in Fig. 2 are illustrated in Fig. 3. It may be observed that h(t) clearly is nonlinear for linear and parabolic degradation models, and its slope increases as g(40) decreases. Thus, the assumption of a linear failure rate may be unconservative for components whose strength is governed by such mechanisms.

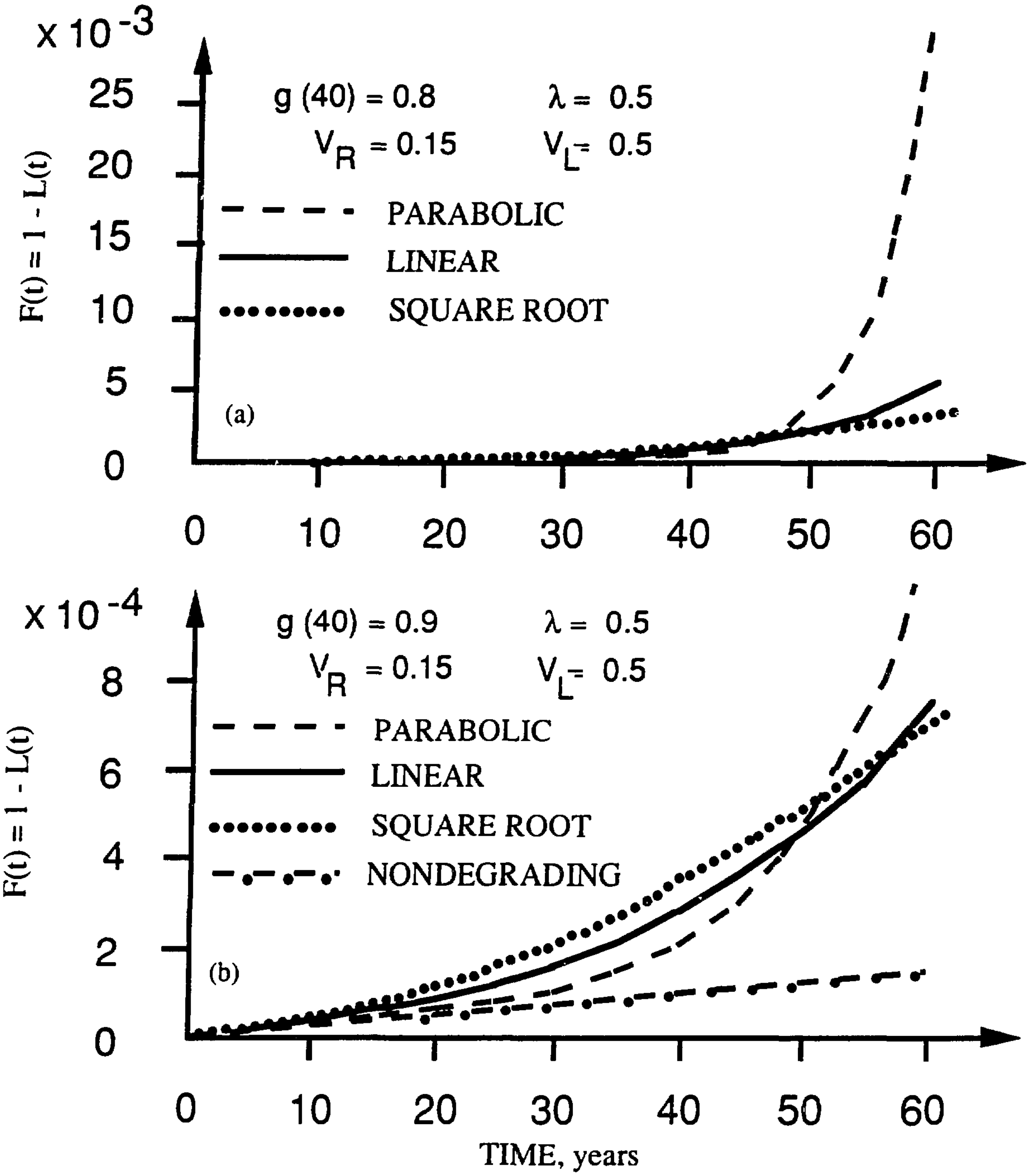

Figure 2. Dependence of Single Component Failure Probability on Degradation Model: D+ L.

Similar illustrations are presented in Fig. 4 for the load combinations involving dead load and accidental pressure and in Fig. 5 for the dead plus live plus earthquake effect combination, respectively. The degradation models affect the failure probability under either D + P or D + L + E in a manner similar to that under D + L. However, the failure probability under D + P

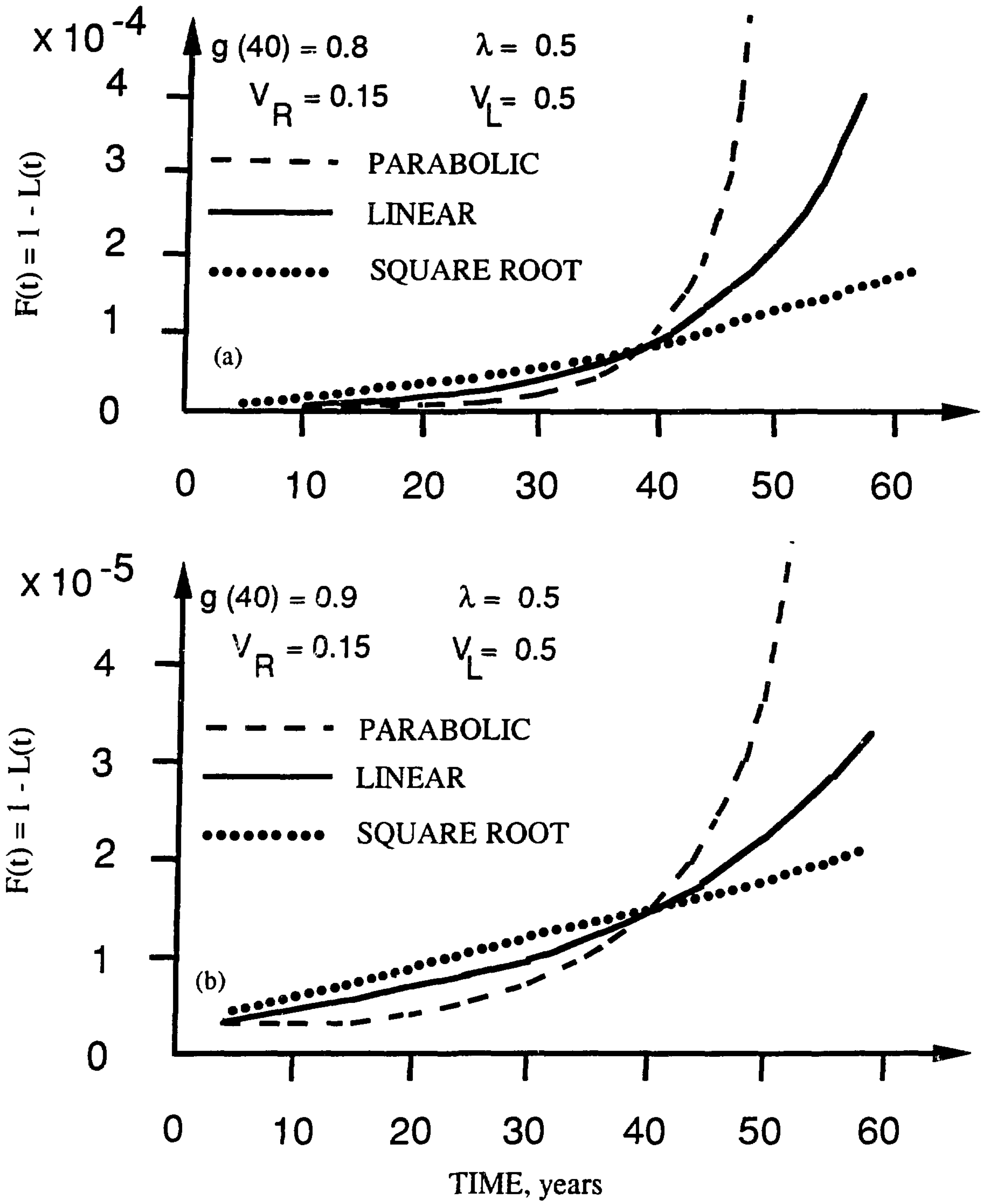

Figure 3. Hazard Function of Single Component.

is smaller by four orders of magnitude than that under D + L because of the small mean occurrence rate of P. In an absolute sense, then, the failure probabilities for these load combinations are less sensitive to degradation mechanism than under D + L, because of the small mean load occurrence rate and, in the case of E, its large variability. In other words, given that a

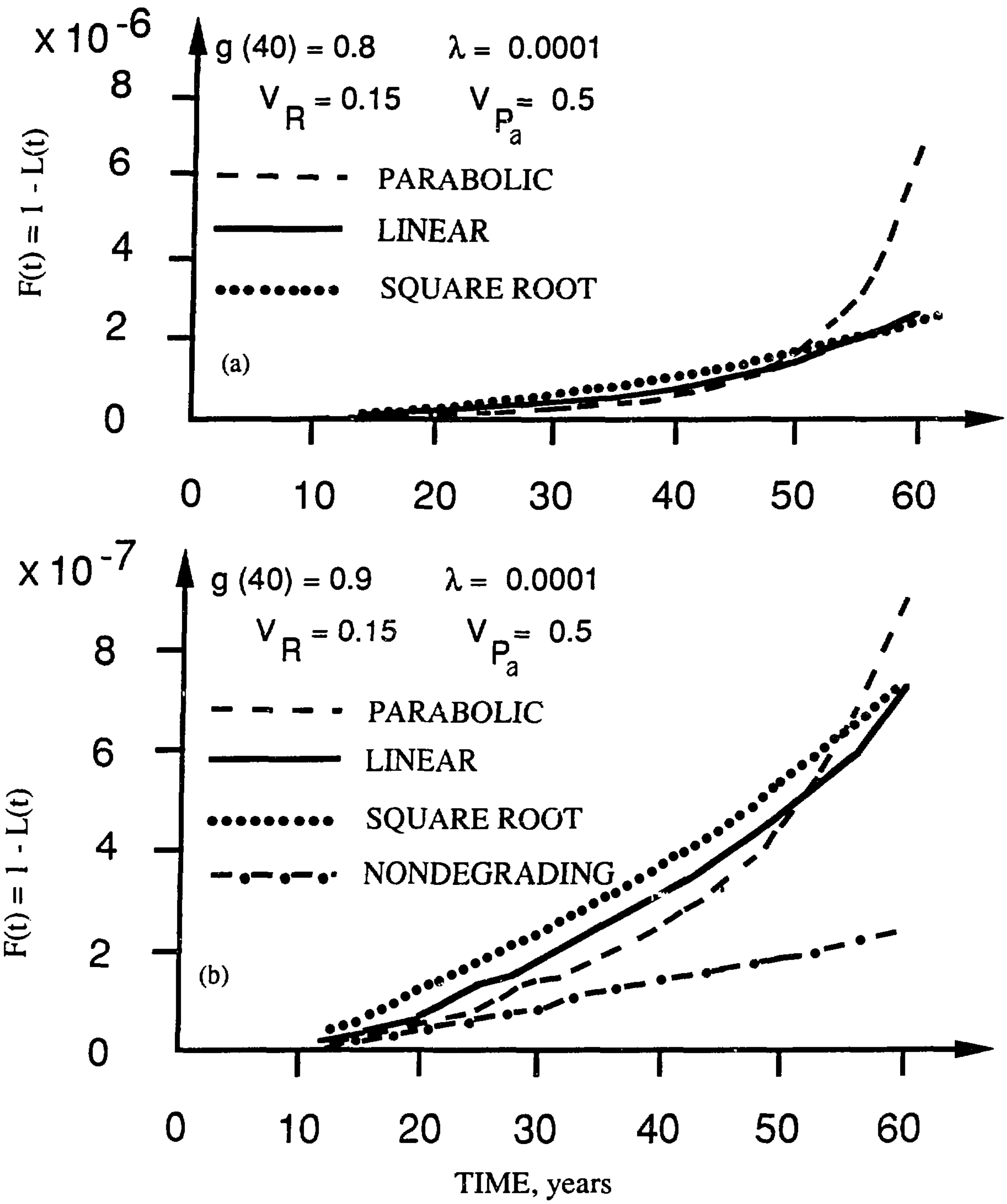

Figure 4. Dependence of Single Component Failure Probability on Degradation Model: D+P.

(rare) load event occurs, the intensity of the load may be large enough to cause failure of the component regardless of whether or not the component has degraded.

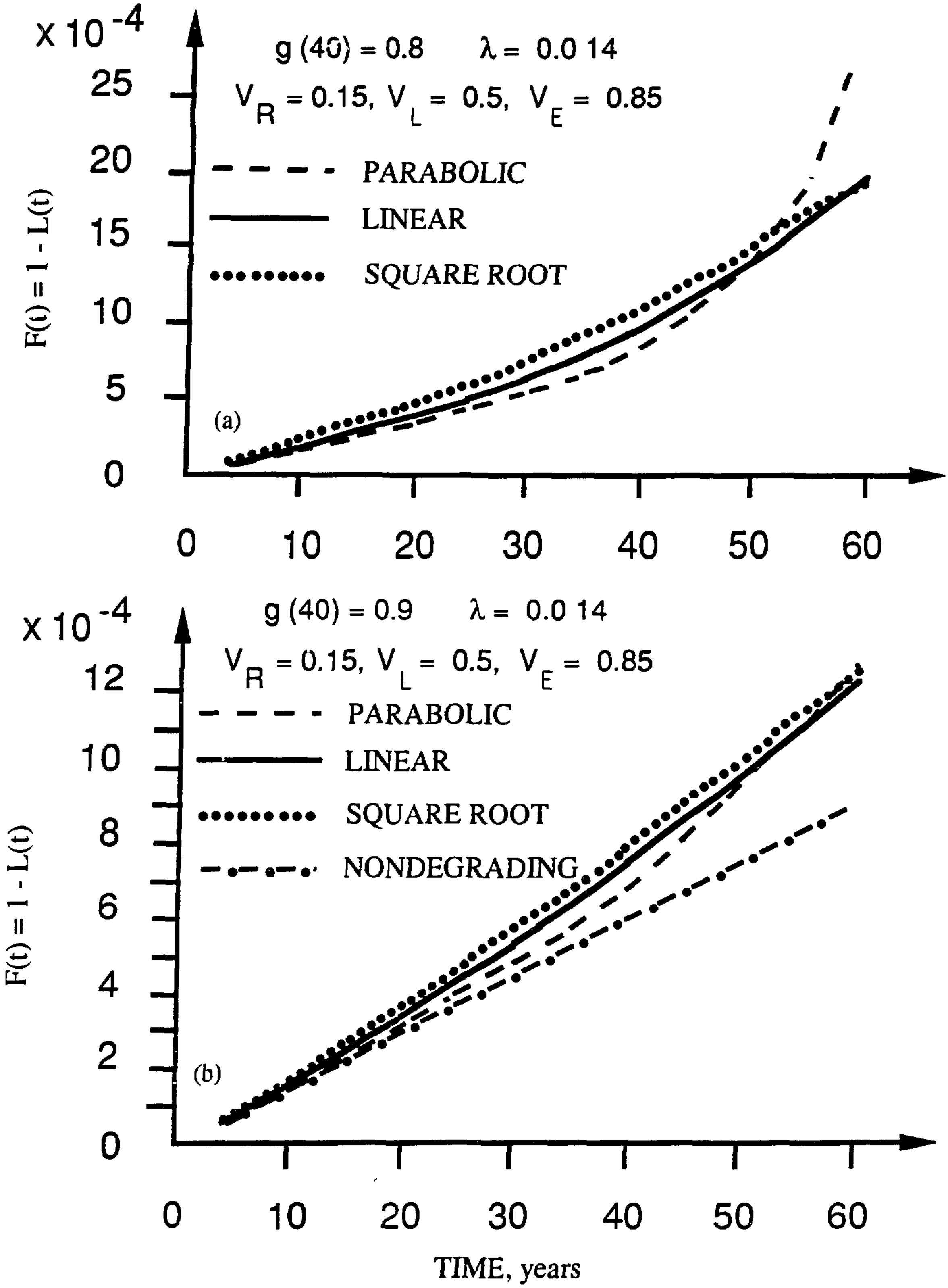

Figure 5. Dependence of Single Component Failure Probability on Degradation Model: D+L+E.

Inservice Inspection/Maintenance Strategies

Periodic in-service inspection followed by suitable maintenance may restore a degraded reinforced concrete structure to near-original condition. Such inspection/maintenance strategies should be designed so that the failure probability of the component is kept lower than an established target probability, P_f, during its service life. Since inspection and maintenance are costly, there are tradeoffs between the extent and accuracy of inspection, required reliability, and cost. An optimum inspection/maintenance program might be obtained from the following contrived optimization problem:

$$\text{Minimize } C_T \tag{13}$$
$$\text{Subject to } F(t) < P_f \tag{14}$$

in which C_T is the total cost of inspection/maintenance plus expected losses if the component fails in service.

Time-dependent reliability analysis can be used in performing this minimum cost analysis. To illustrate this with a very simple example, consider two alternative strategies: (1) infrequent but thorough inspection/maintenance performed at 20, 40, and 60 years, with restoration of full strength; and (2) frequent but limited inspection/maintenance performed at 10, 20, 30, 40, and 50 years, with restoration of 97% of full strength. Degradation is assumed to occur linearly, with $g(40) = 0.8$. The failure probabilities associated with these strategies (as well as with doing nothing) are compared in Fig. 6 for the D + L combination. At the time of inspection/maintenance, F(t) changes slope. If, e.g., $P_f = 0.00025$ in 40 years, strategy (1) would be unacceptable, while strategy (2) would be acceptable. Note in this case, at least, frequent cursory inspection seems preferable to infrequent thorough inspection.

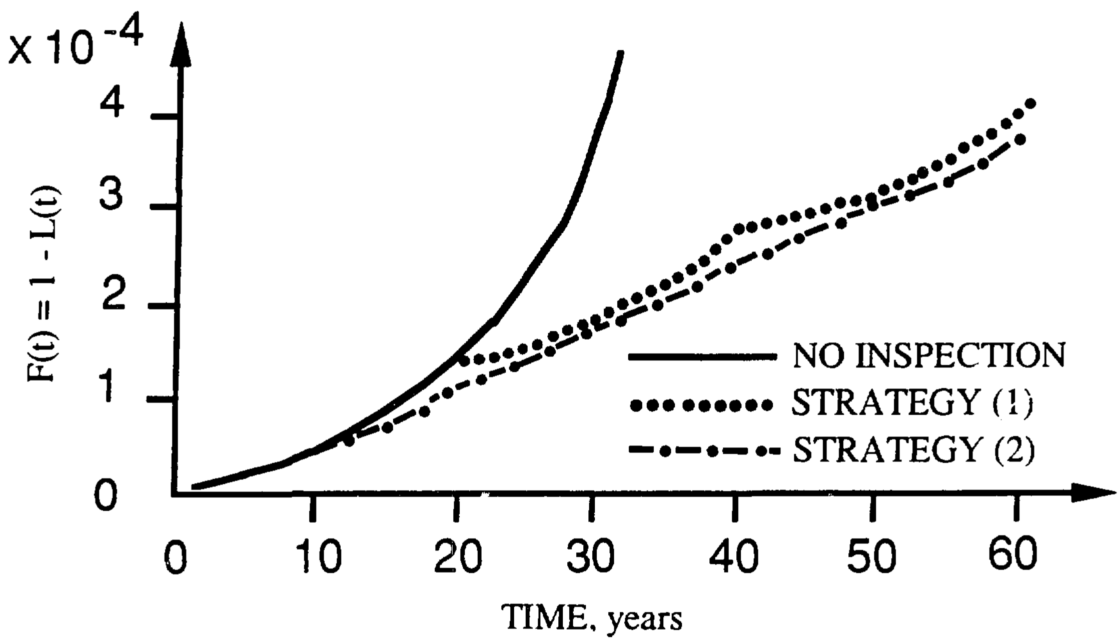

Figure 6. Failure Probability with Repair.

APPLICATION OF STRUCTURAL AGING PROGRAM RESULTS

When completed, the results of this program will provide an improved basis for the USNRC staff to permit continued operation of a nuclear power plant. More specifically, potential regulatory applications of this research include: (1) improved predictions of long-term material and structural performance and available safety margins at future times, (2) establishment of limits on exposure to environmental stressors, (3) reduction in total reliance by licensing on inspection and surveillance through development of a methodology which will enable the integrity of structures to be assessed (either pre- or post-accident), and (4) improvements in damage inspection methodology through potential incorporation of results into national standards which could be referenced by standard review plans.

References

1. "Regulatory Options for Nuclear Power Plant License Renewal," NUREG-1317, Division of Reactor and Plant Systems, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C., June 1988.

2. "LWR Plant Life Extension," EPRI NP-5002, Electric Power Research Institute, Palo Alto, CA, January 1987.

3. "*BWR Pilot Plant Life Extension Study at the Monticello Plant*, EPRI NP-5181M, Electric Power Research Institute, Palo Alto, CA, May 1987.

4. PWR Pilot Plant Life Extension Study at Surry Unit 1: Phase 1," EPRI NP-5289P, Electric Power Research Institute, Palo Alto, CA, July 1987.

5. D. J. Naus, "Appendix B, Concrete Material Systems in Nuclear Safety-Related Structures – A Review of Factors Related to their Durability, Degradation Detection and Evaluation, and Remedial Measures for Areas of Distress," in *The Longevity of Nuclear Power Plant Systems*, by I. Spiewak and R. S. Livingston, EPRI NP-4208, Electric Power Research Institute, Palo Alto, CA, August 1985.

6. "Pressurized Water Reactor Containment Structures License Renewal Industry Report," Nuclear Management and Resources Council (NUMARC), Washington, D.C., August 16, 1989 (draft).

7. C. J. Hookham, "Life Assessment Procedures for Major LWR Components – Concrete Containments," NUREG/CR-5314, Idaho National Engineering Research Laboratory, Idaho Falls, April 1990 (draft).

8. "Class I Structures License Renewal Industry Report," NUMARC 90-06, Nuclear Management and Resources Council, Washington, D.C., June 1990 (draft).

9. 10 CFR Part 50, *Domestic Licensing of Production and Utilization Facilities*.

10. ACI Committee 318, "Building Code Requirements for Reinforced Concrete, ACI Standard 318-77, American Concrete Institute, Detroit, MI, 1977.

11. ACI Committee 349, Code Requirements for Nuclear Safety Related Concrete Structures, ACI Standard 349-85, American Concrete Institute, Detroit, MI, 1985.

12. D. J. Naus, "Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants," NUREG/CR-4652 (ORNL/TM-10059), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, TN, September 1986.

13. D. J. Naus, C. B. Oland, and E. G. Arndt, "Management of the Aging of Critical Safety-Related Concrete Structures in Light-Water Reactor Plants," *Proc. United States Nuclear Regulatory Commission Eighteenth Water Reactor Safety Information Meeting*, Rockville, MD, October 22-24, 1990.

14. P. Lobner et al., "Overview and Comparison of U.S. Commercial Nuclear Power Plants," NUREG/CR-5640 (SAIC-89/1541), Science Applications International Corporation, San Diego, CA, September 1990.

15. B. Mather, "Concrete Need Not Deteriorate," J. Am. Concr. Inst. 1(9), 32-37, September 1979.

16. R. F. Sammataro, "Containment Inservice Inspection and Testing," Lecture Notes, Technical Seminars, Great Neck, New York, 1990.

17. R. F. Sammataro, "Updated ASME Code Rules for Inservice Inspection of Steel and Concrete Containments," *Proceedings - Fifth International Workshop on Containment Integrity,* Washington, D.C., May 12-14, 1992.

18. J. H. Willenbrock et al., "Final Summary Report: A Comparative Analysis of Structural Concrete Quality Assurance Practices of Nine Nuclear and Three Fossil Fuel Power Plant Construction Projects," COO/4120-3, Dept. Civil Eng., The Pennsylvania State Univ., University Park, December 1978.

19. C. J. Hookham, "Structural Aging Assessment Methodology for Concrete Structures in Nuclear Power Plants," ORNL/NRC/LTR-90/17 (Subcontract report from Multiple Dynamics Corp., Southfield, MI), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, TN, September 1990.

20. H. Ashar and D. Jeng, "Effectiveness of Inservice Inspection Requirements of Prestressed Concrete Containments - U.S. Experience," Ibid.

21. B. M. Morris and J. P. Vora, "Nuclear Plant Aging Research (NPAR) Program Plan," NUREG-1144 (Rev. 1), Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C., July 1985.

22. D. J. Naus, M. F. Marchbanks, C. B. Oland, and E. G. Arndt, "Structural Aging (SAG) Program Five-Year Plan: FY 1988-1992," ORNL/NRC/LTR-89/1, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, TN, March 1989.

23. D. J. Naus and C. B. Oland, "Structural Aging Program Technical Progress Report for Period January 1, 1991, to December 31, 1991," ORNL/NRC/LTR-92/3, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, TN, February 1992.

24. M. F. Marchbanks, "A Review and Assessment of Materials Property Databases with Particular Reference to Concrete Material Systems," ORNL/NRC/LTR-89/3, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, TN, March 1989.

25. C. B. Oland, and D. J. Naus, "Plan for Use in Development of the Structural Materials Information Center," ORNL/NRC/LTR-89/8, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, TN, September 1989.

26. C. B. Oland, and D. J. Naus, "Structural Materials Information Center for Presentation of Time Variation of Material Properties," ORNL/NRC/LTR-90/22, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, TN, September 1990.

27. *Mat.DB*, Version 1.0, ASM International, ASM/Center for Materials Data, Materials Park, OH, 1990.

28. *EnPlot*, Version 2.0, ASM International, ASM/Center for Materials Data, Materials Park, OH, 1989.

29. M. G. Van Geem and D. J. Naus, *Summary of Test Results for Portland Cement Association Study on Long-Term Properties of Concrete*,ORNL/NRC/LTR-91/26 (Subcontract Report 11X-SF710V from Construction Technology Laboratories, Inc., Skokie, Illinois), Martin Marietta Energy Systems, Inc., Oak Ridge Nat'l Lab., Oak Ridge, Tenn., December 1991 (draft).

30. J. R. Clifton, *Predicting the Remaining Life of In-Service Concrete*, NISTIR 4712, National Institute of Standards and Technology, U.S. Department of Commerce, Gaithersburg, Maryland, November 1991.

31. T. M. Refai and M. K. Lim, *Inservice Inspection and Structural Integrity Assessment Methods for Nuclear Power Plant Concrete Structures*, ORNL/NRC/LTR-90/29 (Subcontract Report 17X-SE611V from Construction Technology Laboratories, Inc., Skokie, Ill.), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., September 1991.

32. J. Mandel, "Fitting Straight Lines When Both Variables are Subject to Error," *J. of Quality Technology* **16**(1), 1- 14, January 1984.

33. B. Ellingwood and Y. Mori, "Condition Assessment and Reliability-Based Life Prediction of Concrete Structures in Nuclear Plants," ORNL/NRC/LTR-92/4 (Subcontract Report 19X-SD684V from The Johns Hopkins University, Baltimore, Maryland), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Oak Ridge, Tenn., January 1992.

34. J. G. MacGregor, S. A. Mirza, and B. Ellingwood, "Statistical Analysis of Resistance of Reinforced and Prestressed Concrete Members," *ACI Journal* 80(3): 167–176, 1983.

35. B. Ellingwood and H. Hwang, "Probabilistic Descriptions of Resistance of Safety-Related Structures in Nuclear Plants," *Nuclear Engineering and Design* 88(2): 169–178, 1985.

36. A. J. M. Siemes, A. Vrouwenvelder, and A. van den Beukel, "Durability of Buildings: A Reliability Analysis," *Heron* 30(3): 1–48, 1985.

37. J. R. Clifton and L. I. Knab, "Service Life of Concrete," National Bureau of Standards, NUREG/CR-5466, U.S. Nuclear Regulatory Commission, Washington, D.C., 1989.

38. H. Hwang, B. Ellingwood, M. Shinozuka, and M. Reich, "Probability-Based Design Criteria For Nuclear Plant Structures," *Journal of Structural Engineering, ASCE* 113(5): 925–942, 1987.

39. B. Ellingwood and Y. Mori, "Reliability-Based Condition Assessment of Concrete Structures in Nuclear Power Plants," NRC Aging Research Information Conference, Rockville, Md., March 1992.

40. R. G. Melchers , "Search-Based Importance Sampling," *Structural Safety* 9(2): 117–128, 1990.

AGING OF CONCRETE CONTAINMENT STRUCTURES IN NUCLEAR POWER PLANTS

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E. G. ARNDT
NUCLEAR REGULATORY COMMISSION

PRESENTATION TO:

FIFTH WORKSHOP ON CONTAINMENT INTEGRITY
WASHINGTON, D.C.
MAY 12-14, 1992

PRESENTATION WILL ADDRESS FOUR TOPICS

- INTRODUCTION
- BACKGROUND
- STRUCTURAL AGING PROGRAM DESCRIPTION AND STATUS
- SUMMARY

PRESENTATION WILL ADDRESS FOUR TOPICS

- INTRODUCTION
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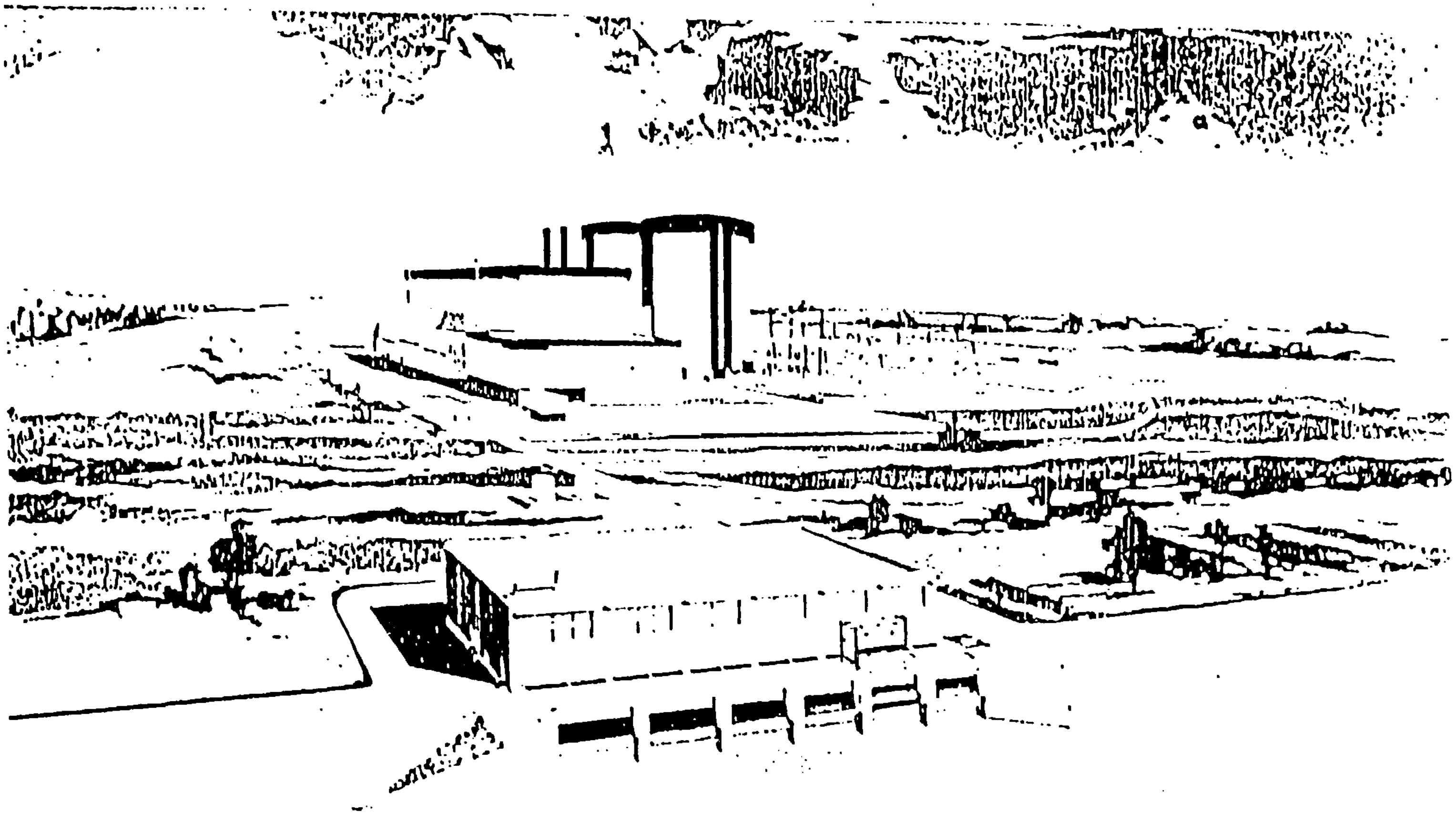
- SUMMARY

A MYRIAD OF CONCRETE–BASED STRUCTURES IS CONTAINED AS PART OF A LWR PLANT



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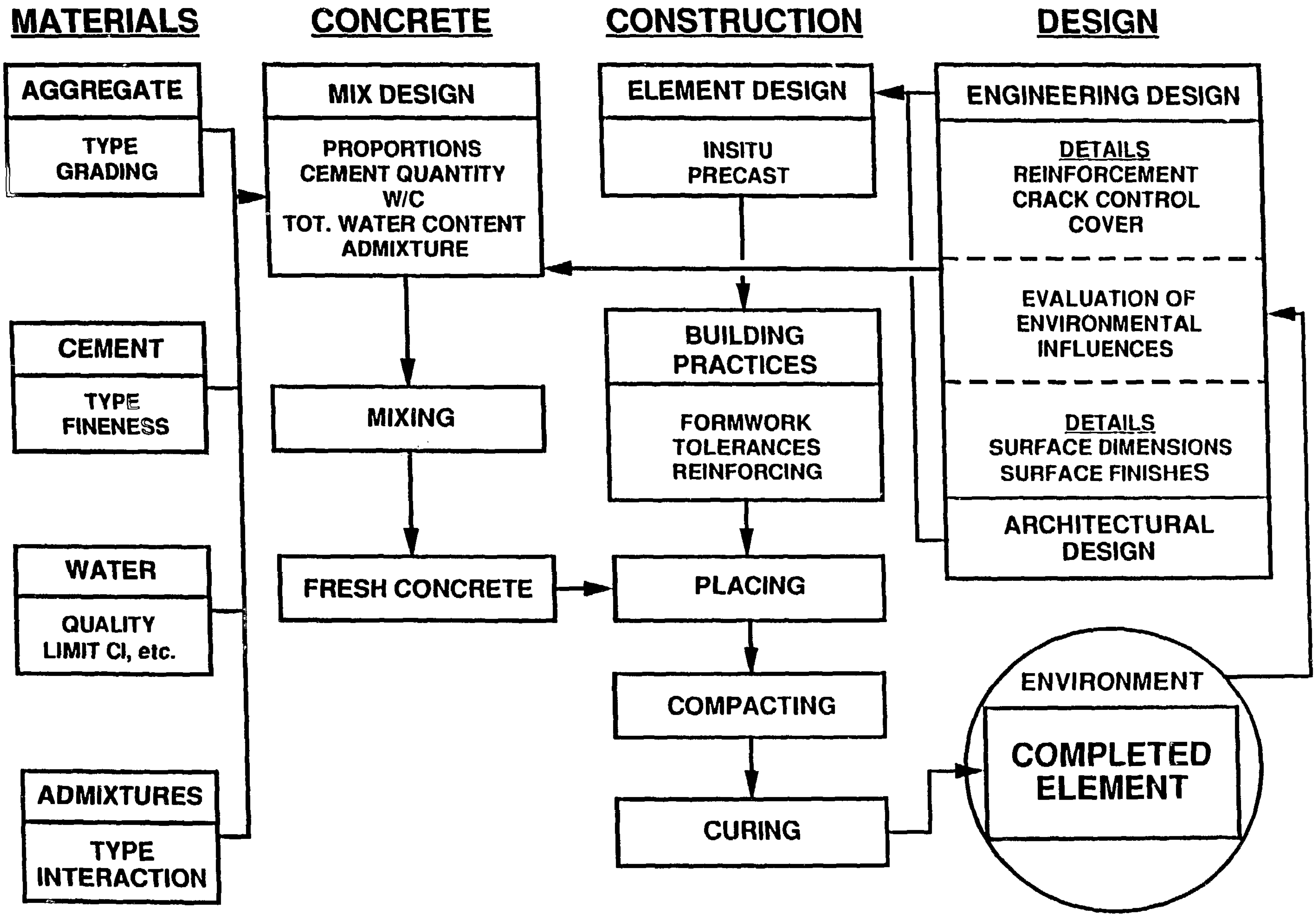
PRESENTATION WILL ADDRESS FOUR TOPICS

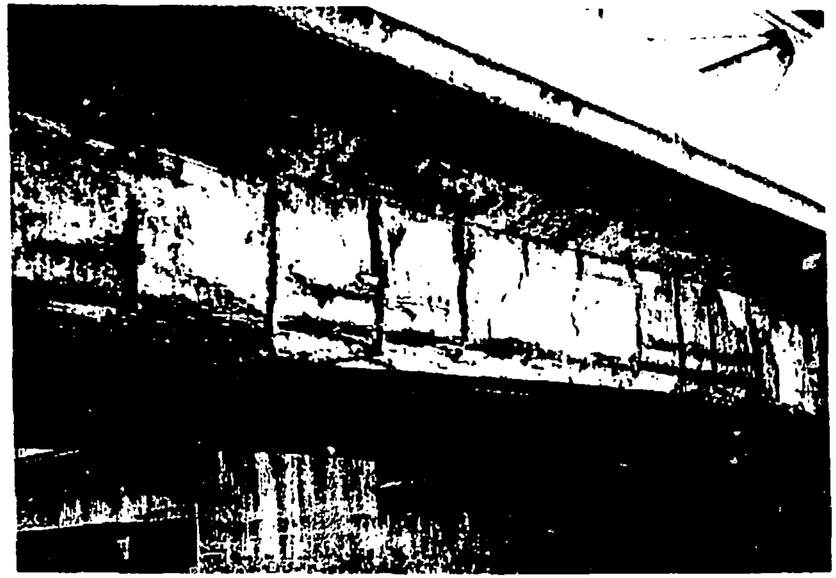
- INTRODUCTION
- BACKGROUND
- STRUCTURAL AGING PROGRAM DESCRIPTION AND STATUS
- SUMMARY

FACTORS WHICH AFFECT THE QUALITY OF CONCRETE STRUCTURES ARE MANY, VARIED AND INTERACTIVE



Longevity of Concrete Structures is Primarily a Function of Their Propensity to Withstand the Potential Effects of Environmental Stressors and Aging Factors

ALKALI-AGGREGATE REACTION



CORROSION OF STEEL REINFORCEMENT

V/91M-576C ETD

PRIMARY DEGRADATION FACTORS WHICH CAN IMPACT THE PERFORMANCE OF CONCRETE AND CONCRETE-RELATED MATERIALS IN NPPs HAVE BEEN IDENTIFIED

MATERIAL SYSTEM	DEGRADATION FACTOR
CONCRETE	CHEMICAL ATTACK EFFLORESCENCE AND LEACHING SALT CRYSTALLIZATION ALKALI - REACTIVE AGGREGATES SULFATE ATTACK ACIDS AND BASES PHYSICAL ATTACK FREEZE / THAW CYCLING THERMAL EXPOSURE / THERMAL CYCLING ABRASION / EROSION / CAVITATION
MILD STEEL REINFORCEMENT & PRESTRESSING STEEL	CORROSION ELEVATED TEMPERATURE IRRADIATION FATIGUE RELAXATION (PRESTRESSING STEEL)

CONTRIBUTION OF AGING TO RESIDUAL LIFE DETERMINATIONS OF SAFETY-RELATED STRUCTURES, SYSTEMS AND COMPONENTS IS COMPLICATED BY SEVERAL FACTORS

- DIFFERENCES IN DESIGN CODES AND STANDARDS FOR COMPONENTS OF DIFFERENT VINTAGE
- LACK OF PAST MEASUREMENTS AND RECORDS
- LIMITATIONS IN THE APPLICABILITY OF TIME-DEPENDENT MODELS FOR QUANTIFYING THE CONTRIBUTION OF AGING TO STRUCTURE, SYSTEM OR COMPONENT FAILURE
- INADEQUACY OF DETECTION, INSPECTION, SURVEIL-LANCE, AND MAINTENANCE METHODS OR PROGRAMS

PRESENTATION WILL ADDRESS FOUR TOPICS

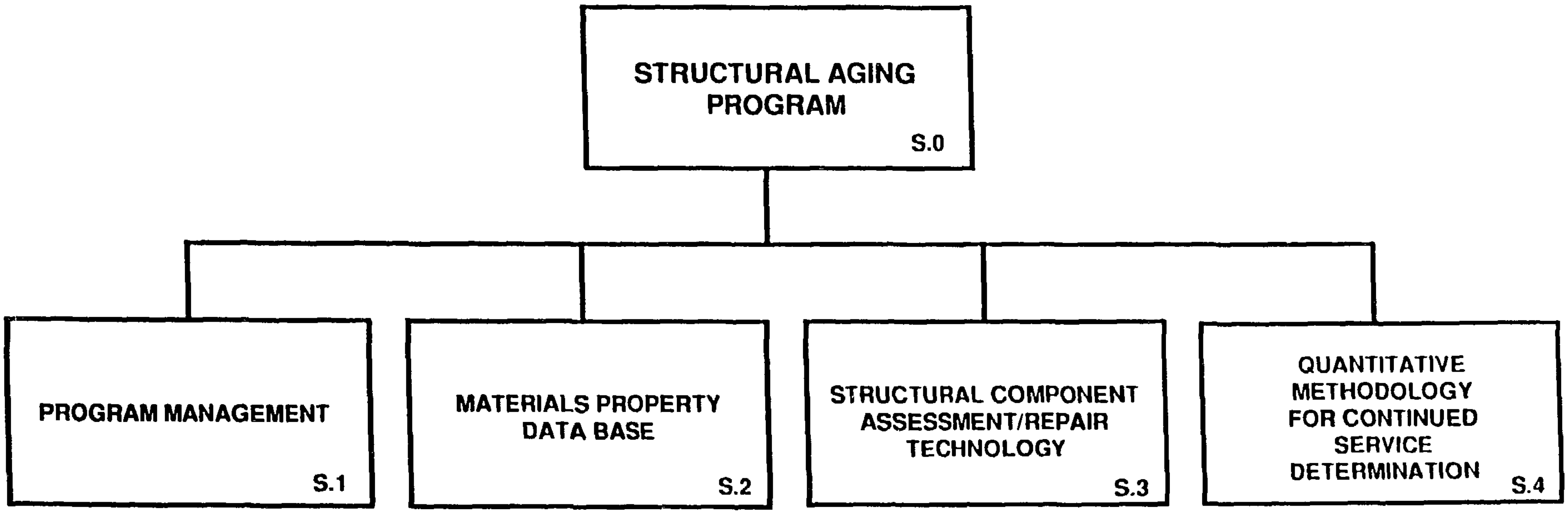
- INTRODUCTION
- BACKGROUND
- STRUCTURAL AGING PROGRAM DESCRIPTION AND STATUS
- SUMMARY

OVERALL OBJECTIVE OF THE PROGRAM IS TO PROVIDE NRC WITH STRUCTURAL SAFETY ISSUES AND ACCEPTANCE CRITERIA FOR USE IN NPP EVALUATIONS FOR CONTINUED SERVICE

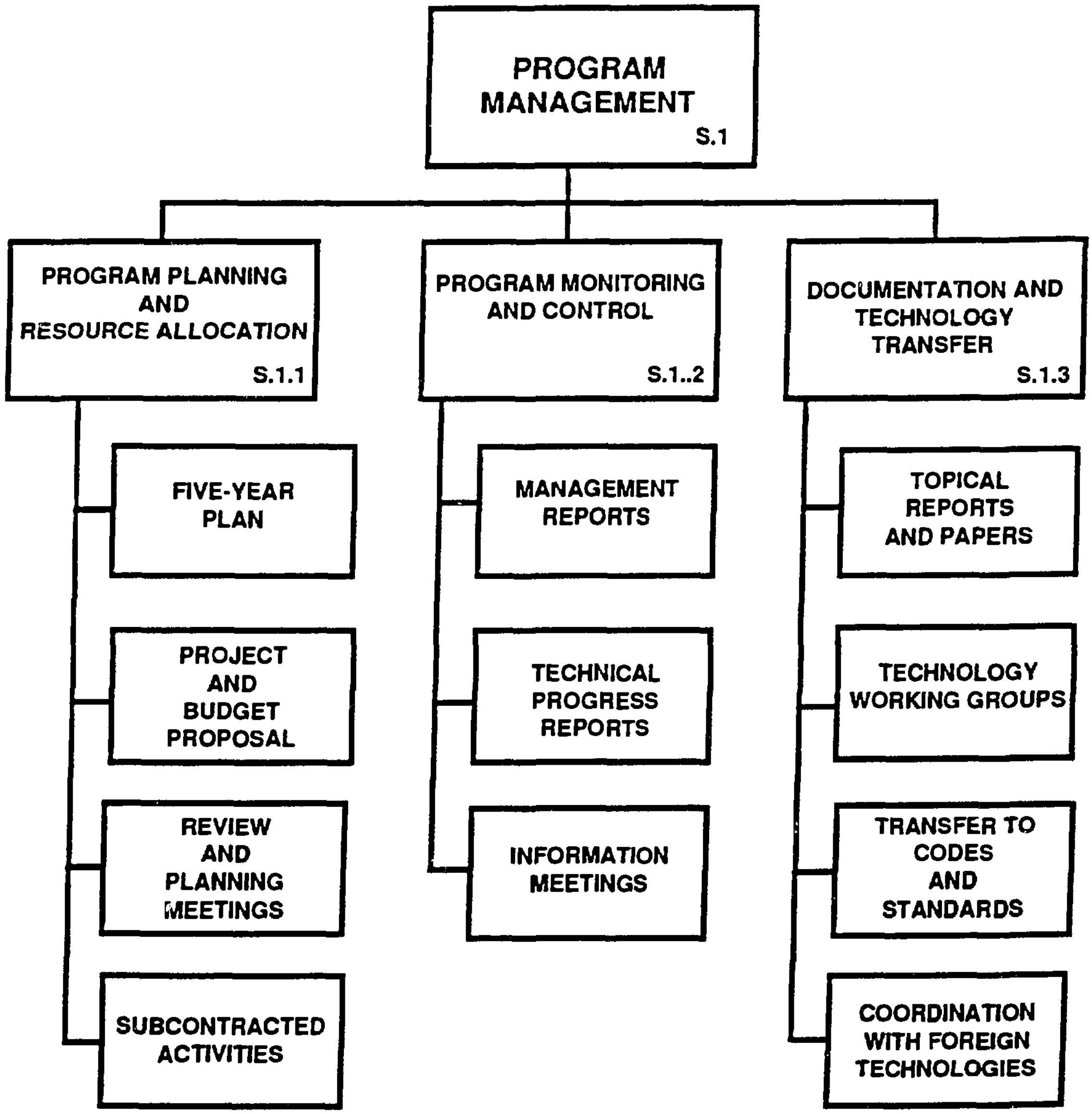
FINAL PRODUCT WILL BE AN EXPANDABLE HANDBOOK (OR REPORT) WHICH WILL PROVIDE NRC LICENSE REVIEWERS AND LICENSEES WITH:

- IDENTIFICATION AND EVALUATION OF THE DEGRADATION PROCESSES THAT AFFECT THE PERFORMANCE OF STRUCTURAL COMPONENTS
- ISSUES TO BE ADDRESSED UNDER NUCLEAR POWER PLANT CONTINUED SERVICE REVIEWS, AS WELL AS CRITERIA, AND THEIR BASES, FOR RESOLUTION OF THESE ISSUES
- IDENTIFICATION AND EVALUATION OF RELEVANT INSERVICE INSPECTION OR STRUCTURAL ASSESSMENT PROGRAMS IN USE, OR NEEDED
- METHODOLOGIES REQUIRED TO PERFORM CURRENT ASSESSMENTS AND RELIABILITY-BASED LIFE PREDICTIONS OF SAFETY-RELATED CONCRETE STRUCTURES

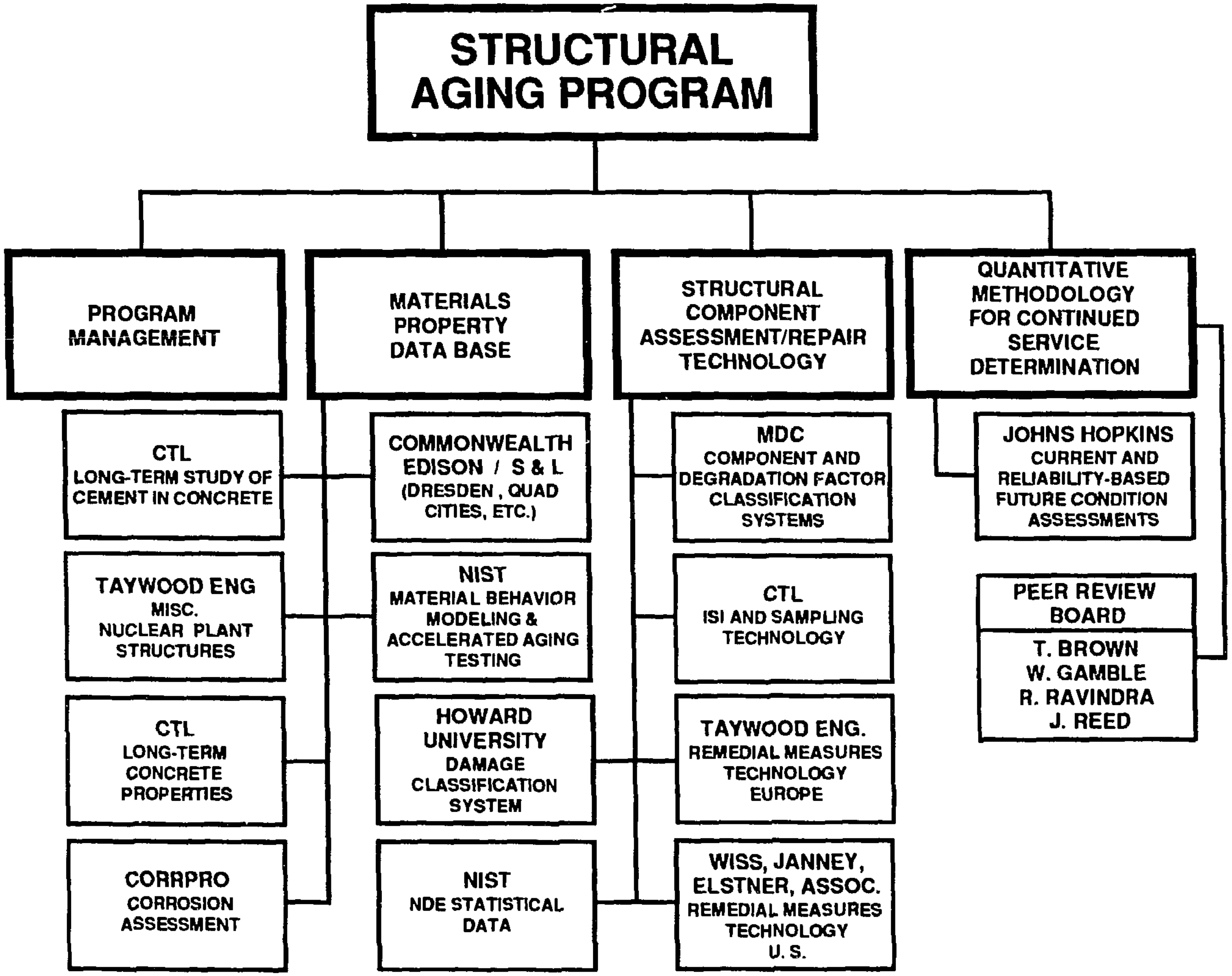
STRUCTURAL AGING PROGRAM CONSISTS OF THREE TECHNICAL TASKS AND A MANAGEMENT TASK



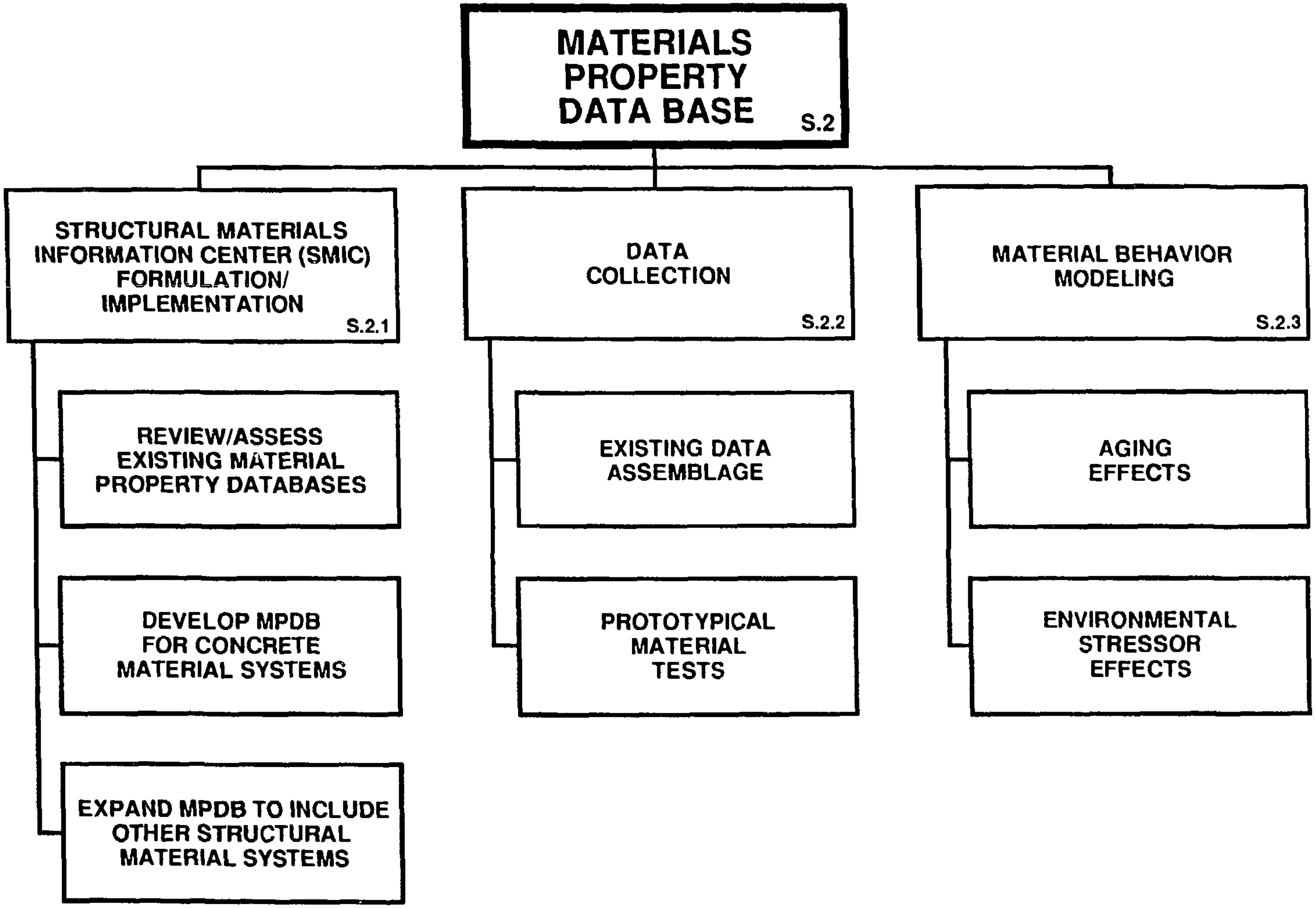
OBJECTIVE OF TASK S.1 IS TO EFFECTIVELY MANAGE THE TECHNICAL TASKS UNDERTAKEN TO ADDRESS PRIORITY STRUCTURAL SAFETY ISSUES RELATED TO NPP CONTINUED SERVICE APPLICATIONS



STRUCTURAL AGING PROGRAM UTILIZES SUBCONTRACTED ACTIVITIES TO TAKE ADVANTAGE OF EXPERTISE AVAILABLE AT OTHER LOCATIONS



OBJECTIVE OF TASK S.2 IS TO DEVELOP A COMPUTER-BASED STRUCTURAL MPDB WHICH WILL CONTAIN INFORMATION ON THE TIME VARIATION OF MATERIAL PROPERTIES UNDER THE INFLUENCE OF PERTINENT ENVIRONMENTAL STRESSORS AND AGING FACTORS



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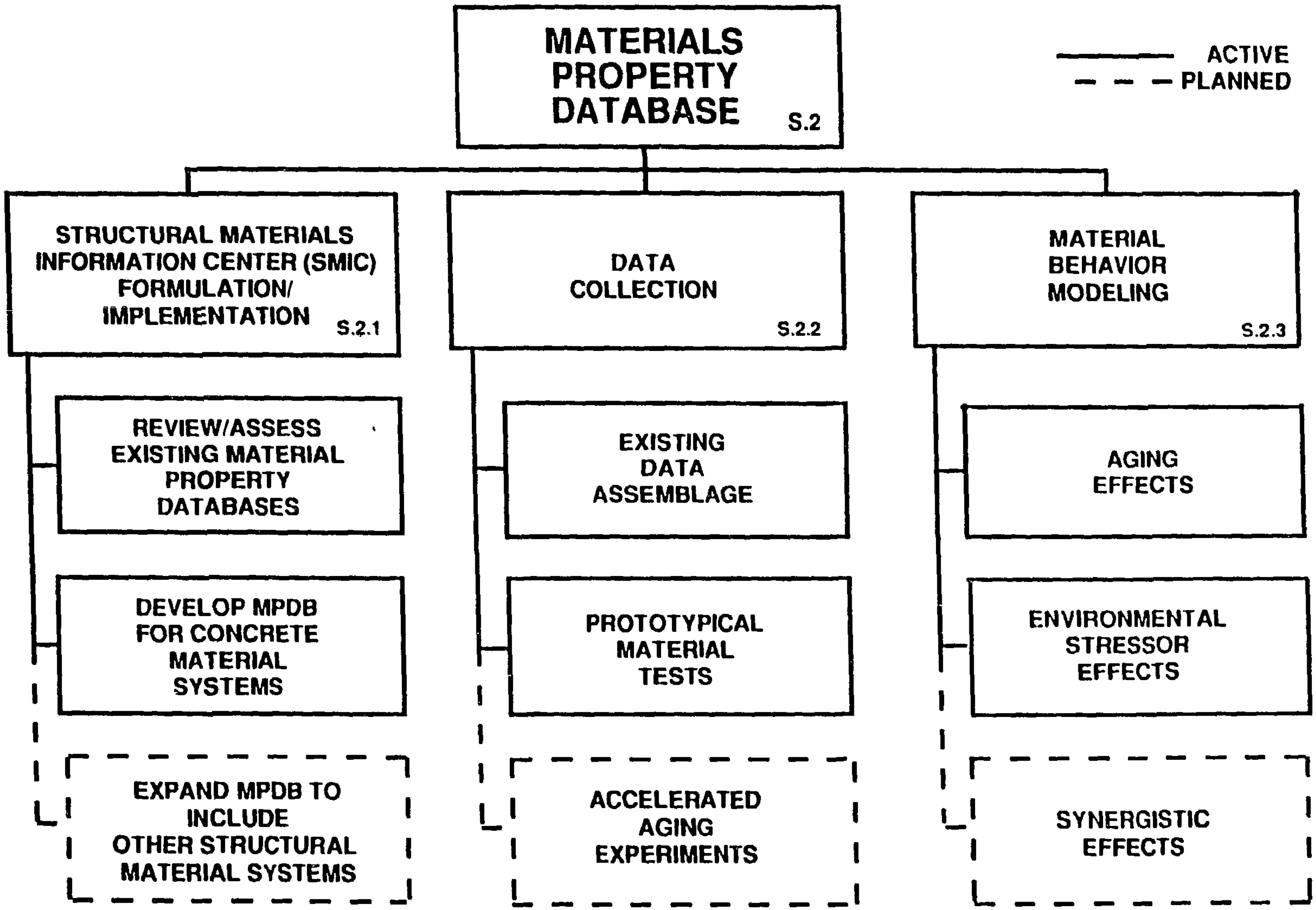


A MATERIALS PROPERTY DATA BASE HAS BEEN ESTABLISHED AT THE OAK RIDGE NATIONAL LABORATORY

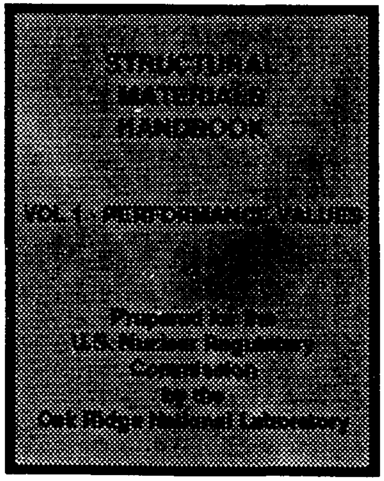
- THE DATA BASE WILL CONTAIN INFORMATION ON THE TIME VARIATION OF MATERIAL PROPERTIES UNDER THE INFLUENCE OF PERTINENT ENVIRONMENTAL STRESSORS AND AGING FACTORS

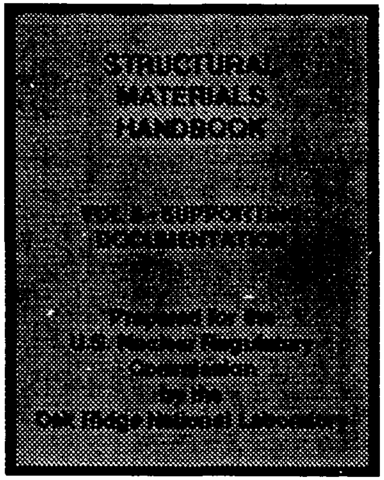
- INITIALLY THE DATA BASE WILL CONCENTRATE ON CONCRETE AND CONCRETE-RELATED STRUCTURAL MATERIALS
  - PORTLAND CEMENT CONCRETES
  - METALLIC REINFORCEMENTS
  - PRESTRESSING STEELS
  - STRUCTURAL STEELS

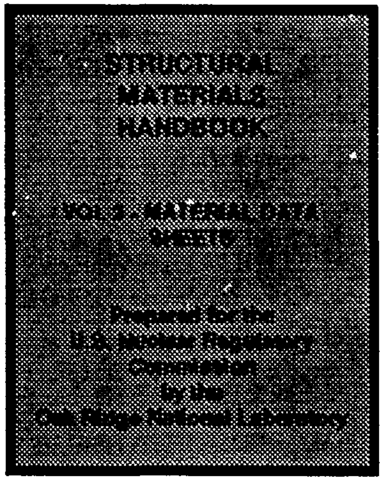
- THE DATA BASE WILL BE PRESENTED AND DISTRIBUTED IN TWO COMPLEMENTARY FORMATS
  - STRUCTURAL MATERIALS HANDBOOK
  - STRUCTURAL MATERIALS ELECTRONIC DATA BASE

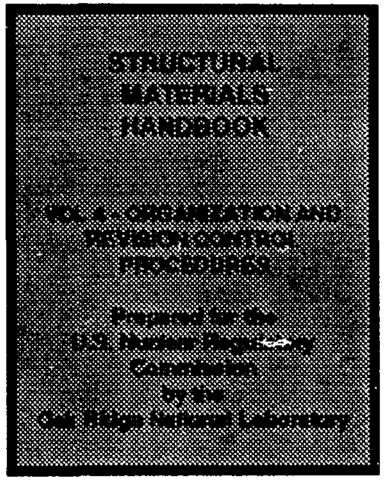
THE STRUCTURAL MATERIALS HANDBOOK IS AN EXPANDABLE, HARD-COPY REFERENCE DOCUMENT THAT IS PRESENTED IN FOUR VOLUMES

- **VOLUMES 1, 2 AND 3 ARE SUBDIVIDED INTO CHAPTERS WITH EACH CHAPTER CORRESPONDING TO A SPECIFIC TYPE OR CATEGORY OF MATERIAL**
- **VOLUME 4 CONTAINS ORGANIZATIONAL AND REVISION CONTROL PROCEDURES**
- **EACH VOLUME IS PUBLISHED IN A LOOSE-LEAF BINDER**



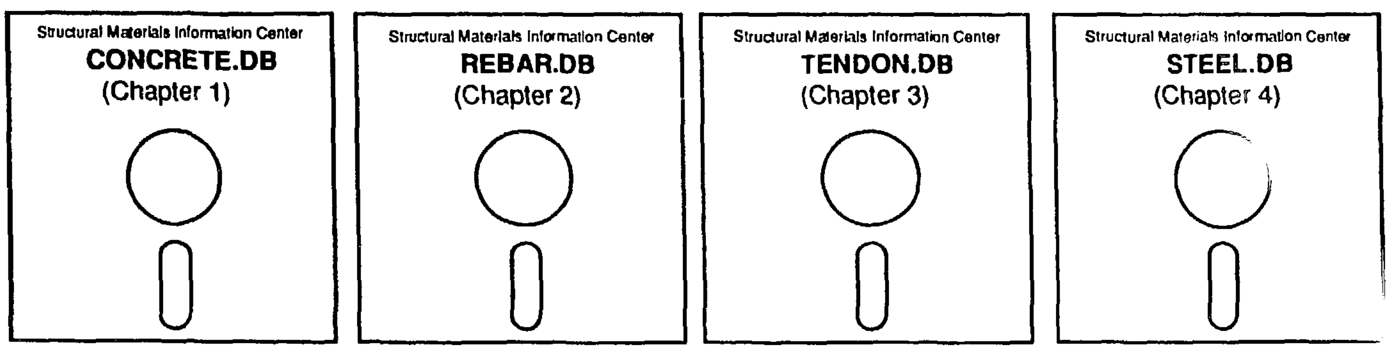




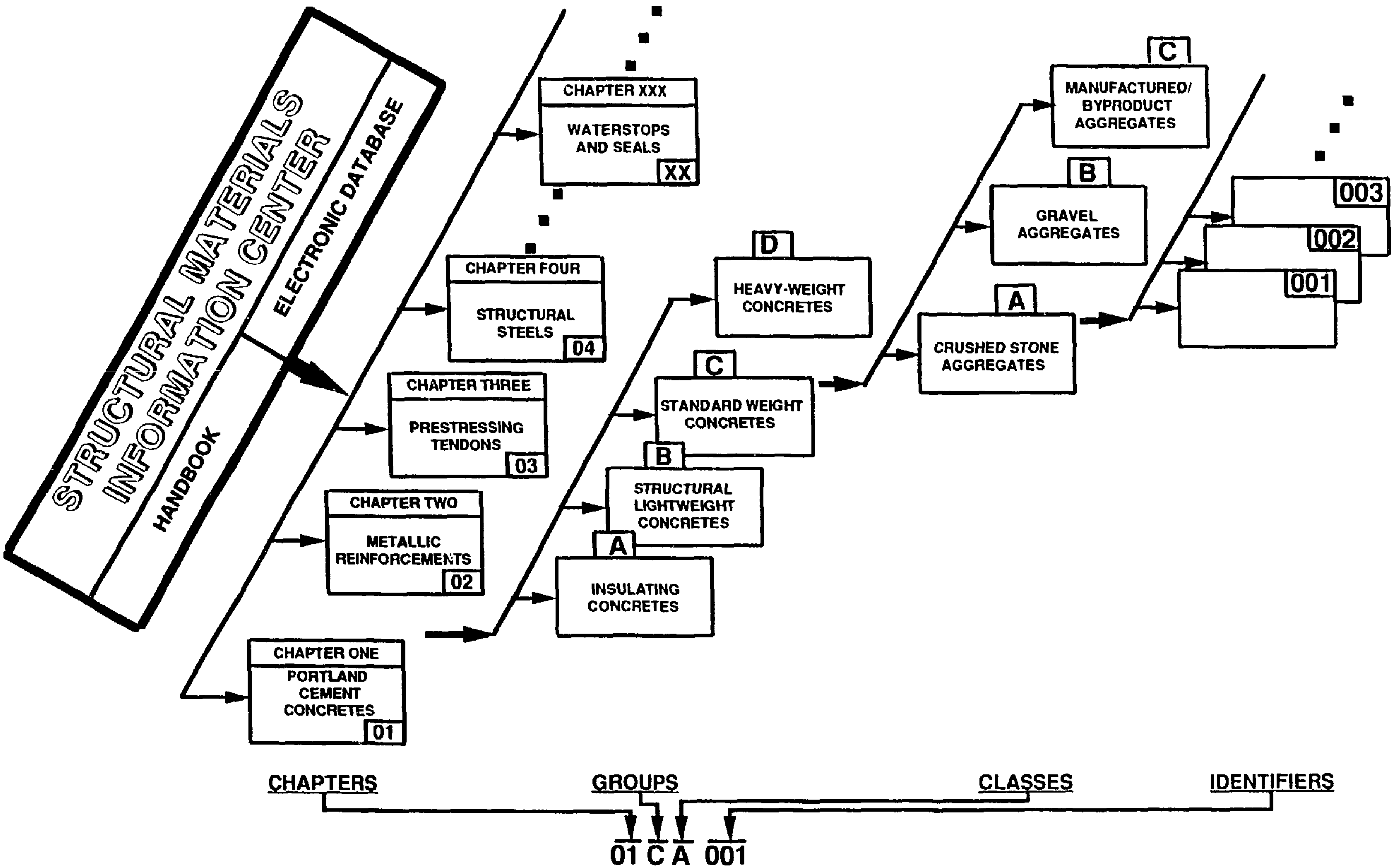


THE STRUCTURAL MATERIALS ELECTRONIC DATA BASE IS AN ELECTRONICALLY ACCESSIBLE VERSION OF THE HANDBOOK

- HARDWARE REQUIREMENTS - IBM-PC OR COMPATIBLE
  - GRAPHICS CARD AND MONITOR
  - HARD DISK DRIVE
  - FLOPPY DISK DRIVE
- SOFTWARE REQUIREMENTS
  - Mat.DB AND EnPlot
  - ASM INTERNATIONAL, MATERIALS PARK, OHIO
- EACH ELECTRONIC DATA BASE FILE CORRESPONDS TO A CHAPTER IN THE HANDBOOK.



MATERIAL PROPERTIES DATA BASE IS BEING FORMULATED SO THAT INFORMATION CAN BE SELECTED BASED ON SEVERAL PARAMETERS



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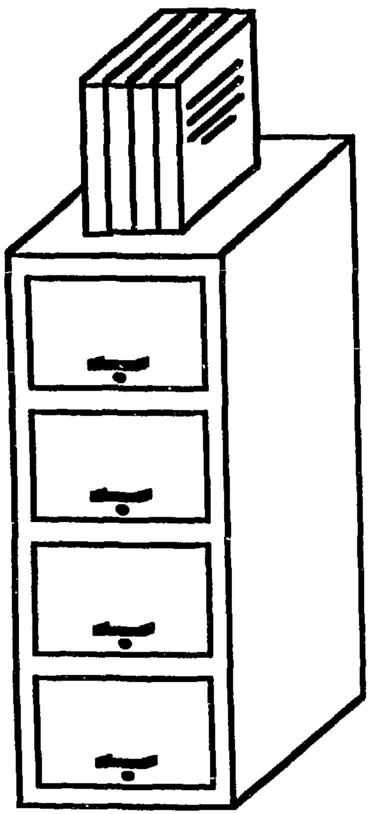
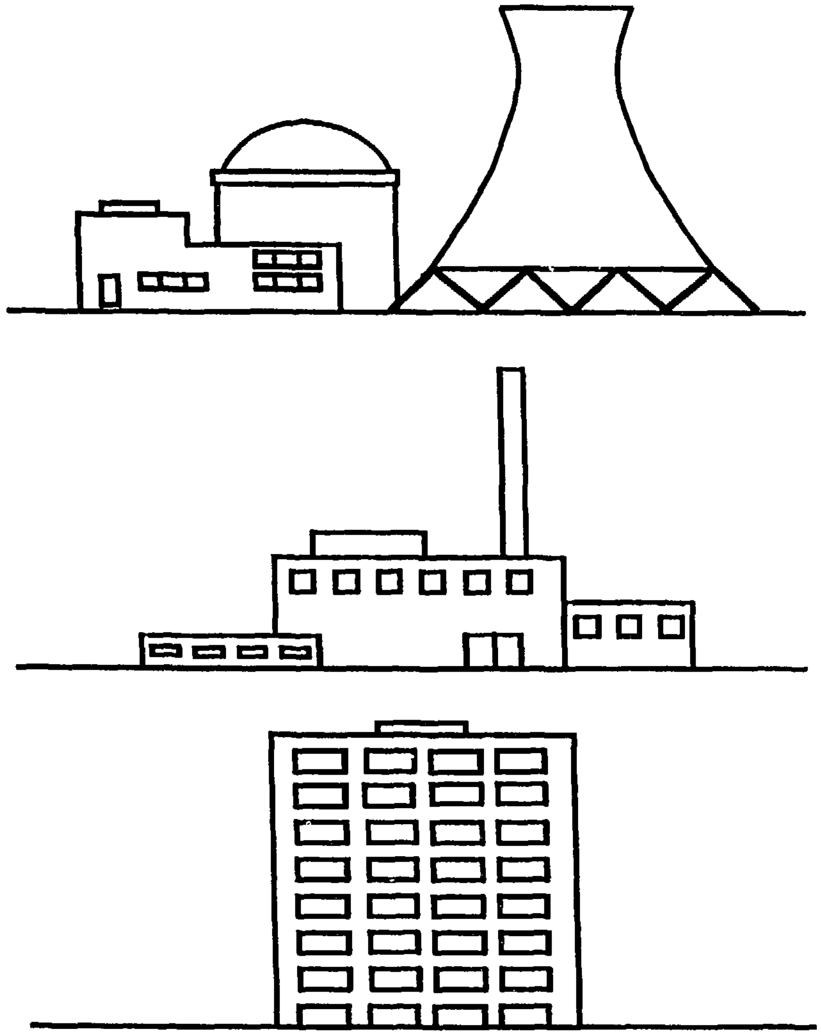
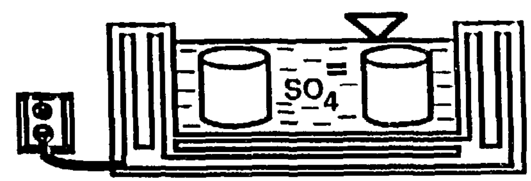


REFERENCES

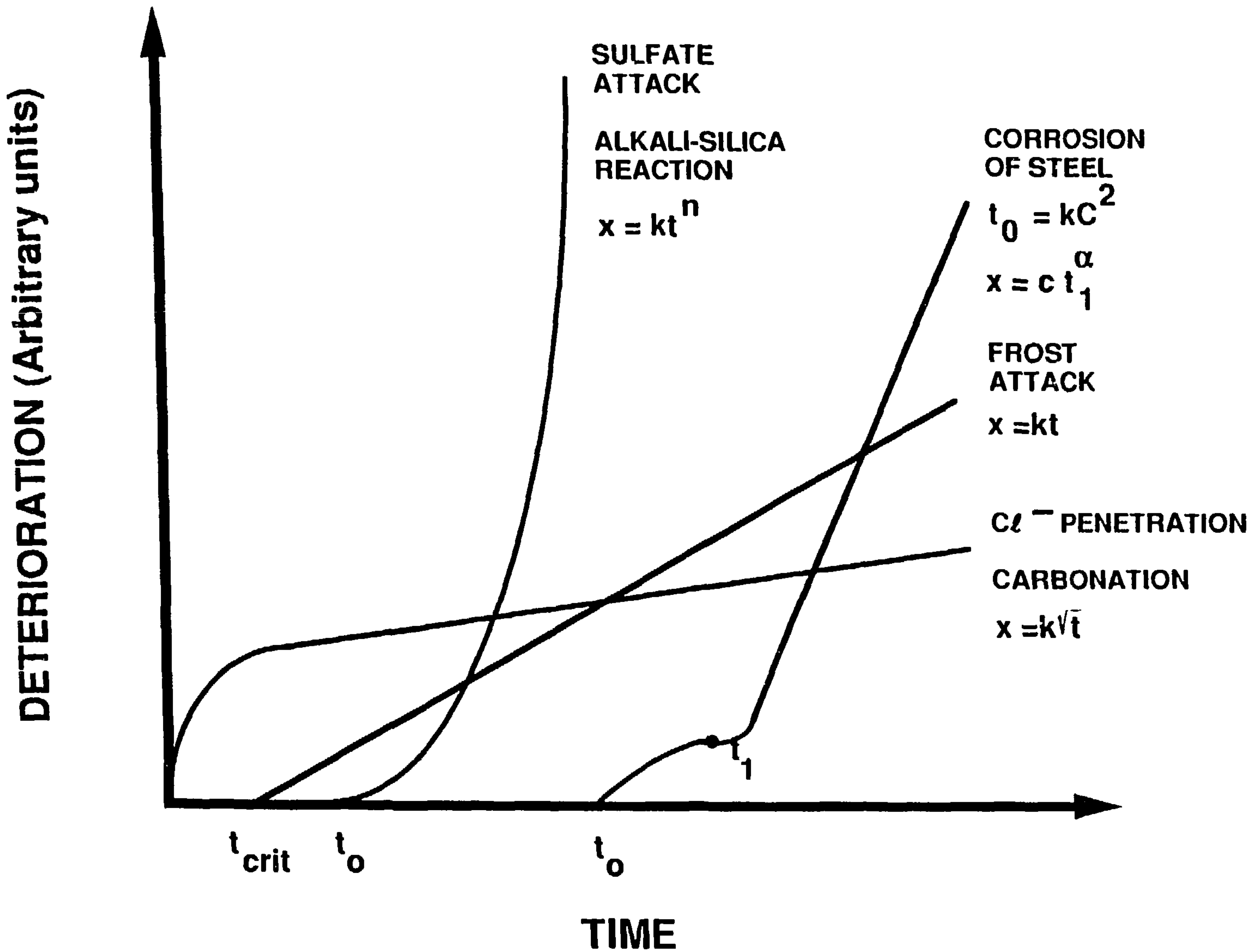


CONCRETE STRUCTURES



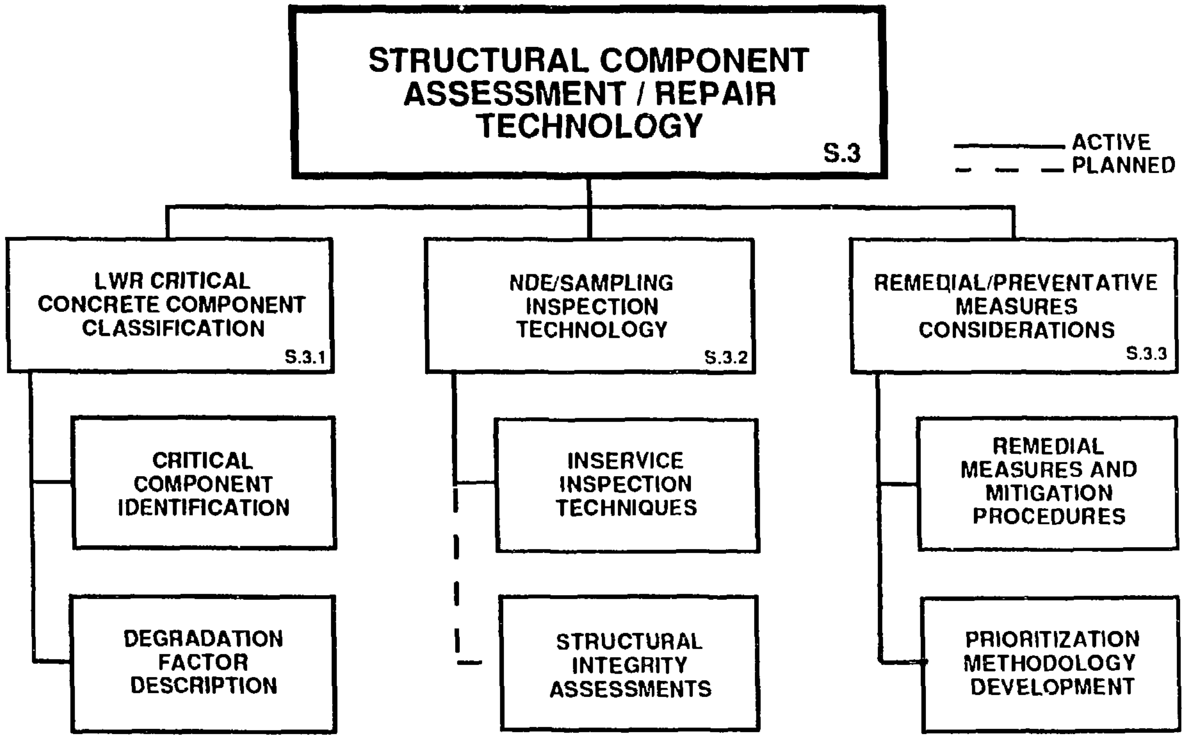
ACCELERATED AGING

TIME-DEPENDENT DEGRADATION FUNCTIONS ARE REQUIRED TO EVALUATE THE RELIABILITY AND SERVICE LIFE OF REINFORCED CONCRETE STRUCTURES



OBJECTIVES OF TASK S.3 ARE TO DEVELOP PROCEDURES TO:

- QUANTITATIVELY ASSESS PRESENCE, MAGNITUDE AND SIGNIFICANCE OF ANY DEGRADATION FACTORS THAT CAN IMPACT DURABILITY
- PROVIDE DATA (ISI OR SAMPLING TECHNIQUES) FOR USE IN CURRENT OR FUTURE STRUCTURAL CONDITION ASSESSMENTS



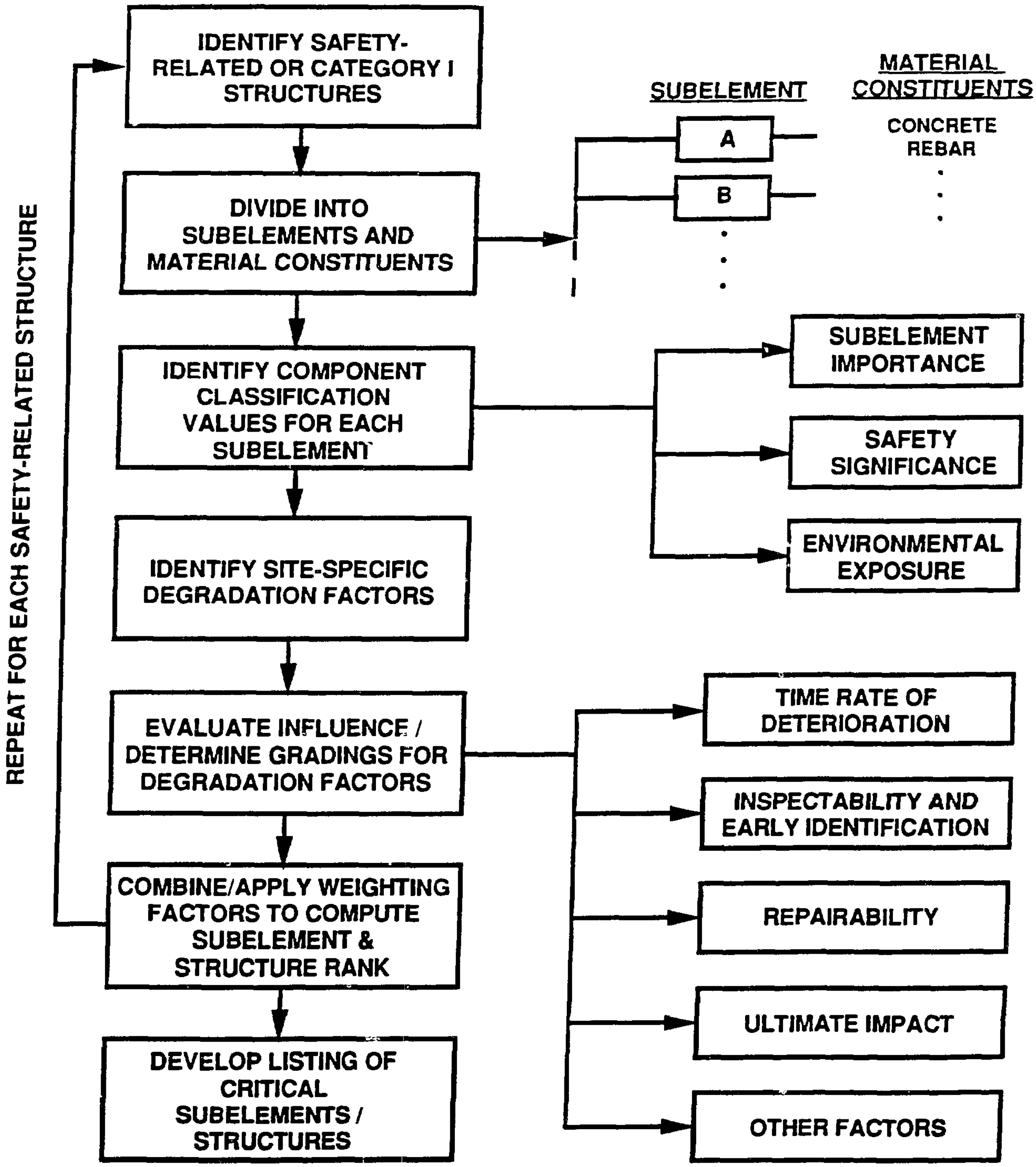
IN ADDITION, TECHNIQUES WILL BE ESTABLISHED FOR:

- MITIGATION OF ENVIRONMENTAL STRESSOR OR AGING FACTOR DETRIMENTAL EFFECTS
- REPAIR, REPLACEMENT OR RETROFITTING OF DEGRADED CONCRETE COMPONENTS

STRUCTURAL COMPONENT SIGNIFICANCE ASSESSMENT (RANKING) METHODOLOGY UTILIZES FOUR CRITERIA TO INCORPORATE INTERACTION OF AGING EFFECTS

- FUNCTIONAL IMPORTANCE
- SAFETY SIGNIFICANCE
- ENVIRONMENTAL EXPOSURE
- DEGRADATION FACTOR SIGNIFICANCE

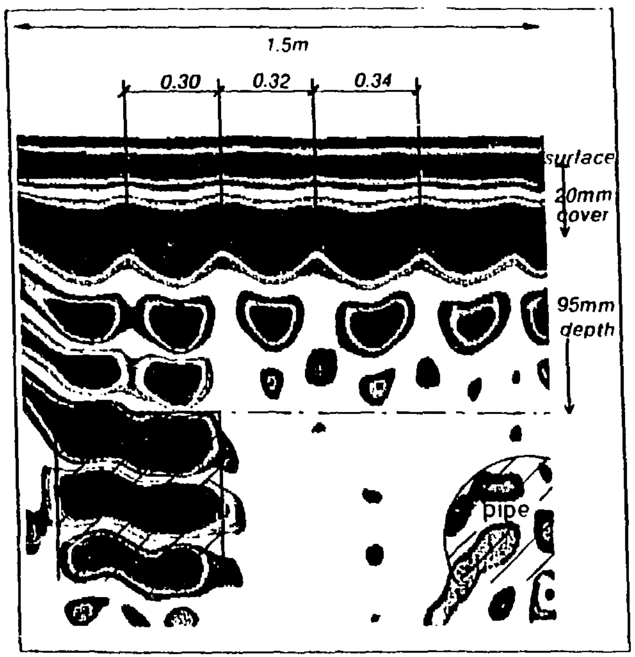
STRUCTURES AND SUBELEMENTS MOST IMPORTANT TO AGING ARE IDENTIFIED THROUGH CALCULATION OF SUBELEMENT RANK AND CUMULATIVE STRUCTURE RANK



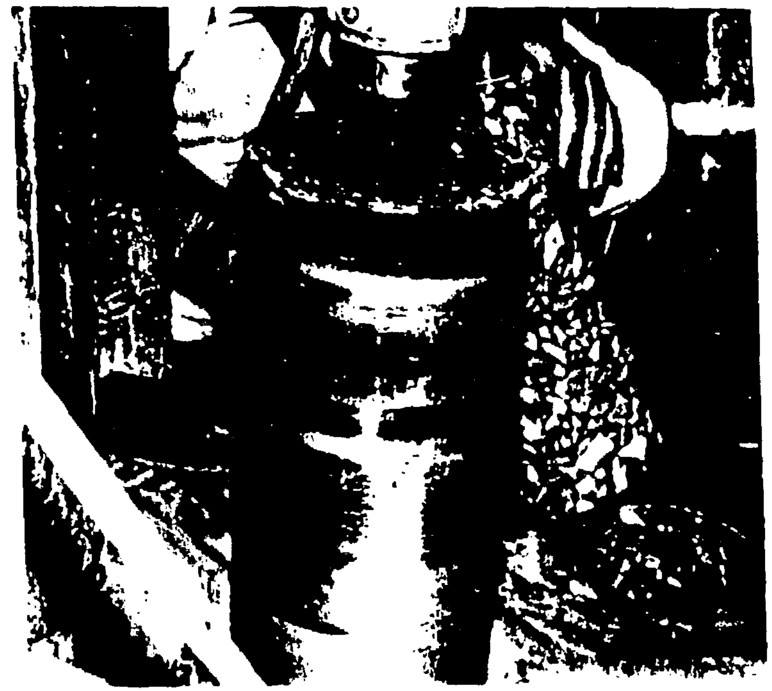
METHODOLOGY APPLIED TO PWR PLANT WITH LARGE-DRY PRESTRESSED CONCRETE CONTAINMENT LOCATED IN THE MIDWEST

PRIMARY STRUCTURE	SUBELEMENT	SUB-ELEMENT RANK	CUMULATIVE RANK
CONTAINMENT VESSEL			172
	DOME	178	
	VERTICAL WALLS (INC. BUTTRESSES)	182	
	MAT FOUNDATION	200	
	•		
	•		
CONTAINMENT - INTERNAL STRUCTURES			153
	BOTTOM SLAB (ABOVE LINER PLATE)	151	
	POLAR CRANE SUPPORT WALL	144	
	FLOOR SLABS	131	
	•		
	•		

Nondestructive and Destructive Testing Techniques are Utilized to Evaluate the Condition of Concrete and Steel Reinforcing Materials in Structures



IMPULSE RADAR

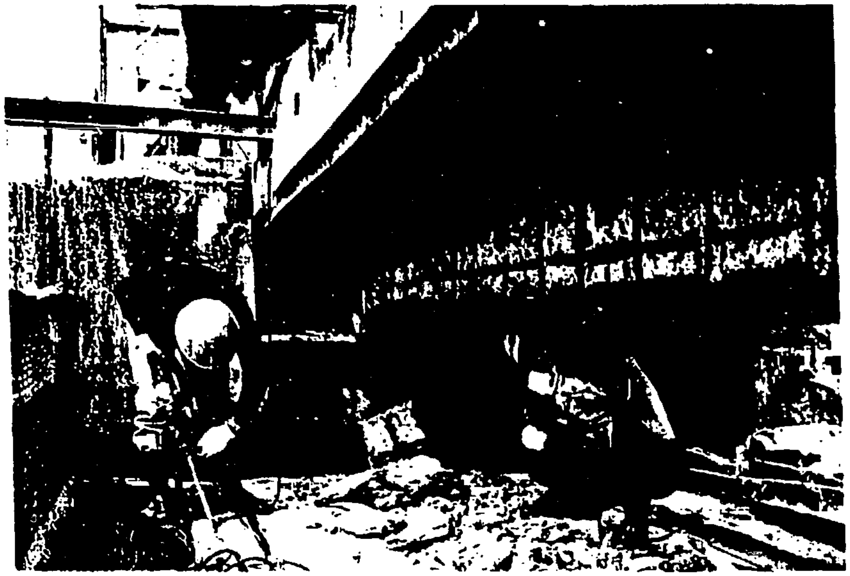


CORE TESTS

V/91-577C ETD

Objectives of Remedial Work Include:

- Restore component's structural integrity
- Arrest mechanism(s) producing distress
- Ensure (as far as possible) that cause(s) of distress will not reoccur

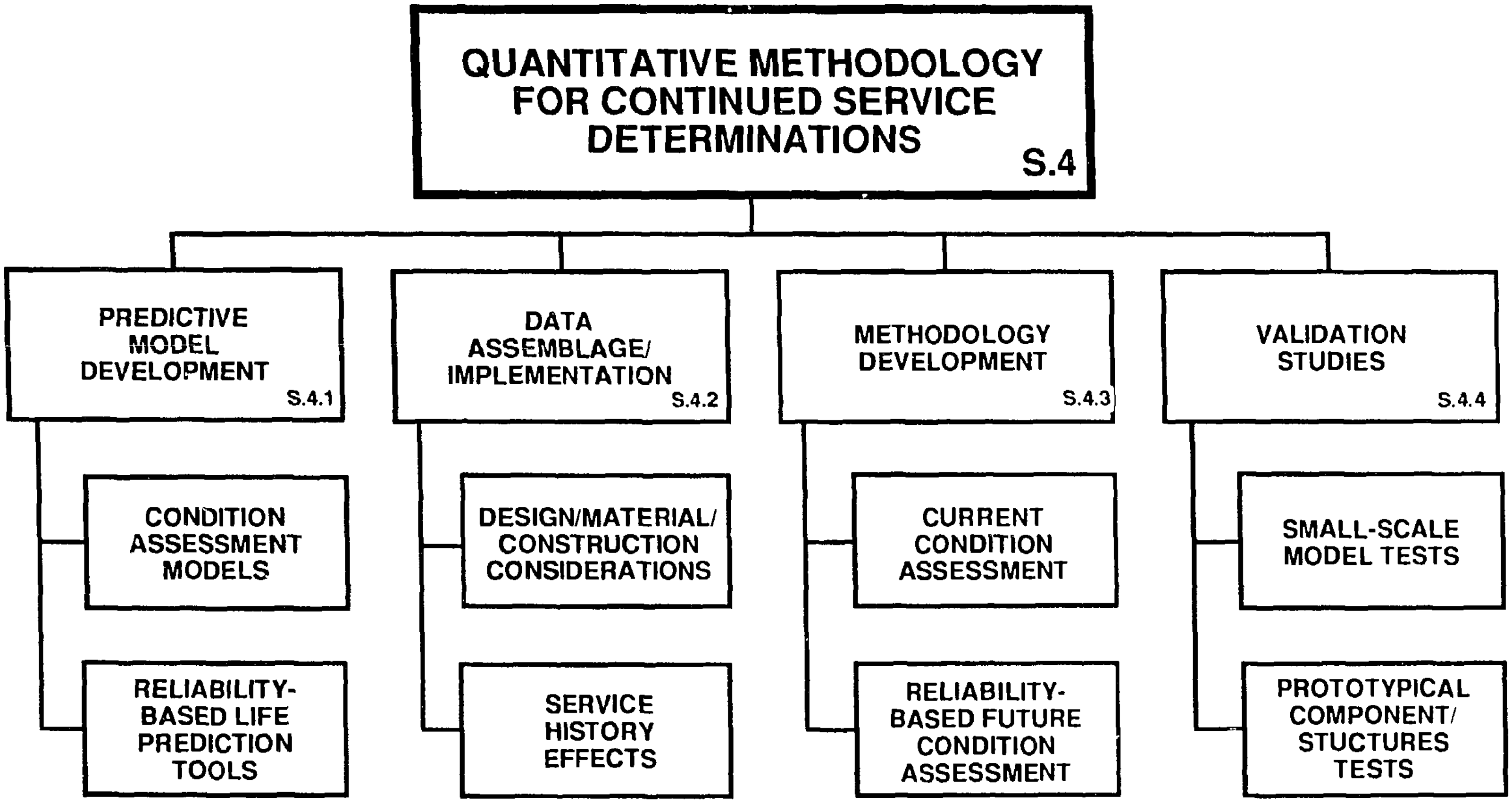


REMOVAL OF DETERIORATED CONCRETE

GROUTING OF POLYMER CEMENT MORTAR INTO PREPLACED AGGREGATE

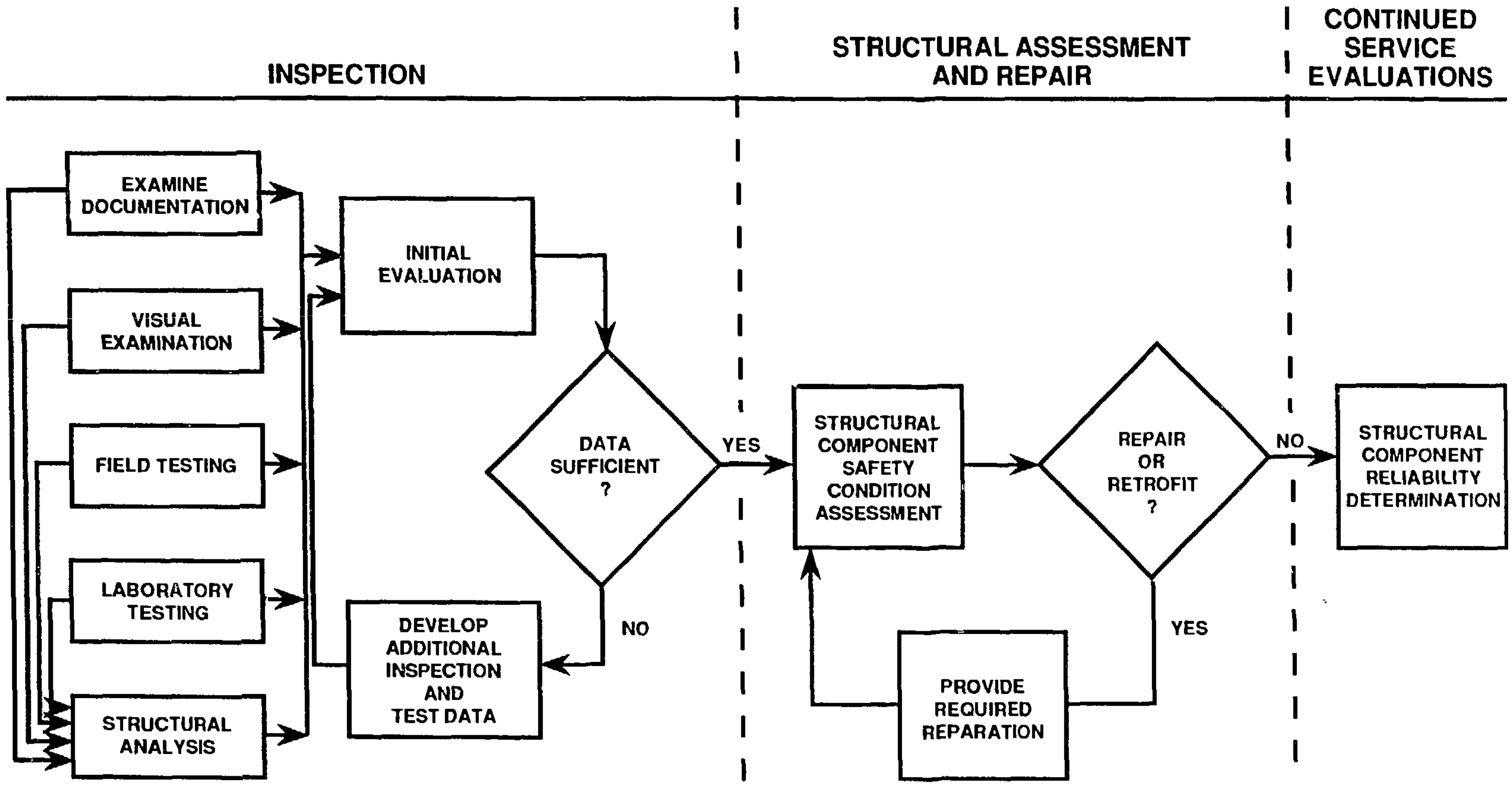
V/91-579C ETD

OBJECTIVE OF TASK S.4 IS TO DEVELOP A METHODOLOGY WHICH CAN BE USED FOR PERFORMING CONDITION ASSESSMENTS AND MAKING RELIABILITY-BASED LIFE PREDICTIONS OF CRITICAL CONCRETE STRUCTURES



EVIDENCE WILL BE PROVIDED WHETHER STRUCTURES IN THEIR CURRENT CONDITION WILL BE ABLE TO WITHSTAND POTENTIAL FUTURE DESIGN EVENTS WITH A LEVEL OF RELIABILITY ADEQUATE TO MEET REQUIREMENTS FOR PROTECTING PUBLIC HEALTH AND SAFETY

INFORMATION ON DEGRADATION AND DAMAGE ACCUMULATION, ENVIRONMENTAL FACTORS, AND LOAD HISTORY WILL BE INTEGRATED INTO A DECISION TOOL

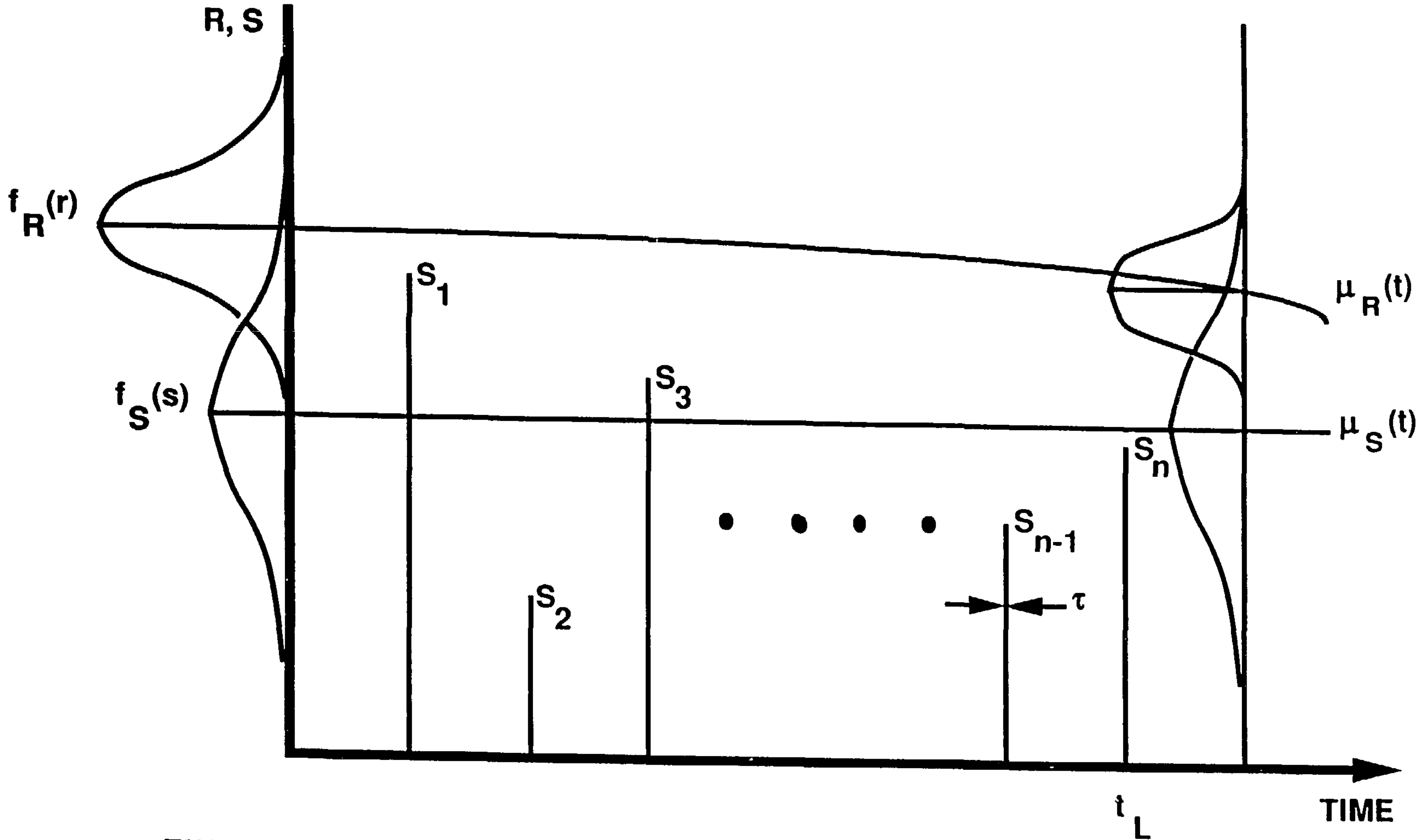


METHODOLOGY WILL PROVIDE A QUANTITATIVE MEASURE OF STRUCTURAL RELIABILITY AND PERFORMANCE UNDER PROJECTED FUTURE SERVICE REQUIREMENTS BASED ON THE CONDITION OF THE EXISTING STRUCTURE

RELIABILITY-BASED TECHNIQUES FOR CONDITION ASSESSMENT OF CONCRETE STRUCTURES (CATEGORY I OR GENERAL) NEED TO TAKE THE FOLLOWING INTO ACCOUNT:

- STOCHASTIC NATURE OF PAST AND FUTURE LOADS DUE TO OPERATING CONDITIONS AND ENVIRONMENT
- RANDOMNESS IN STRENGTH
- UNCERTAINITY IN NDE TECHNIQUES

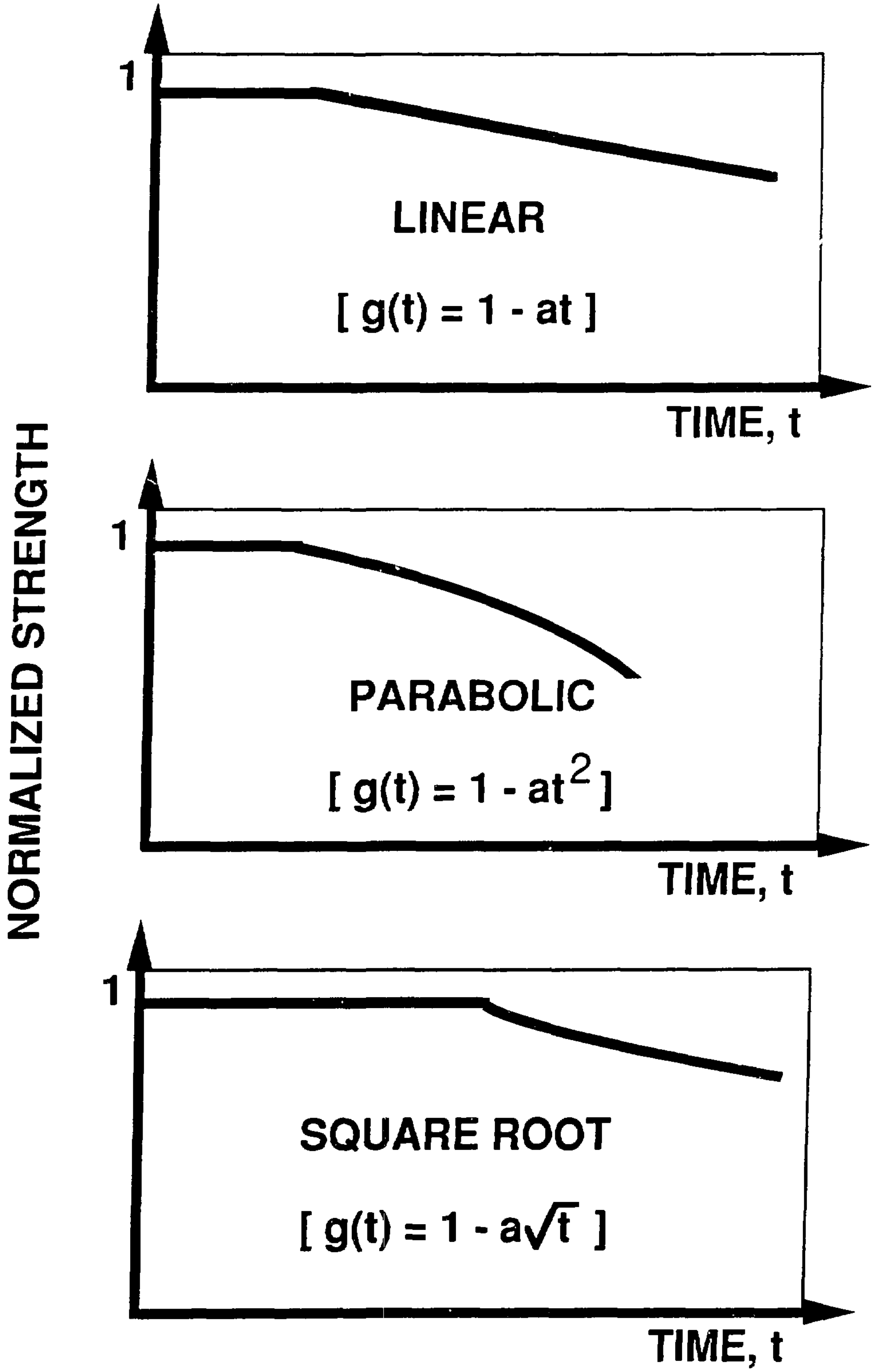
A SEQUENCE OF LOAD PULSES (Sj) IS USED TO STATISTICALLY DESCRIBE THE STOCHASTIC STRUCTURAL LOAD



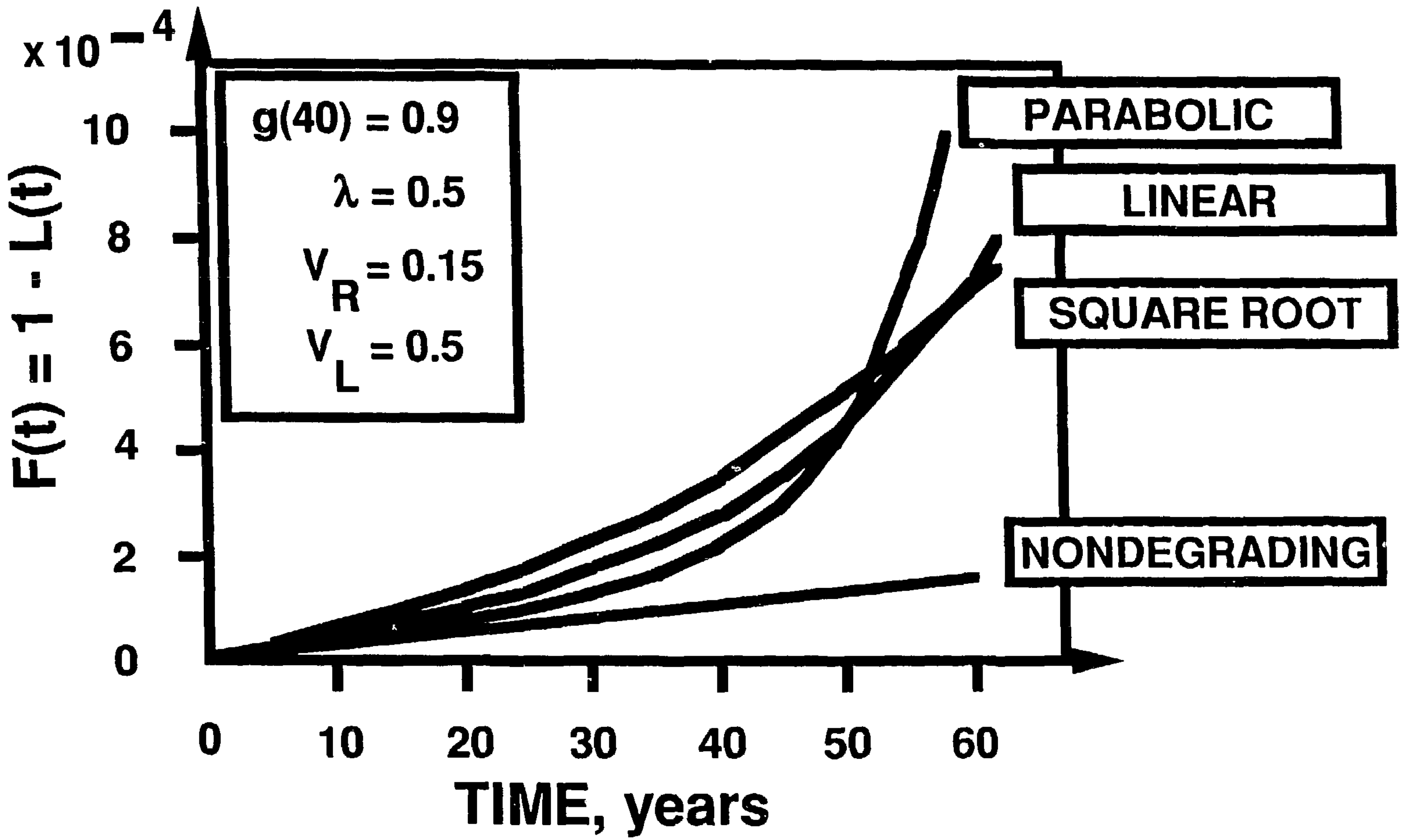
EXTREME LOADS CAN BE MODELED STOCHASTICALLY AS STATIONARY POISSON PULSE PROCESSES WITH THE LOAD PROCESS DESCRIBED BY:

- MEAN RATE OF OCCURRENCE, I
- PULSE DURATION
- PULSE INTENSITY

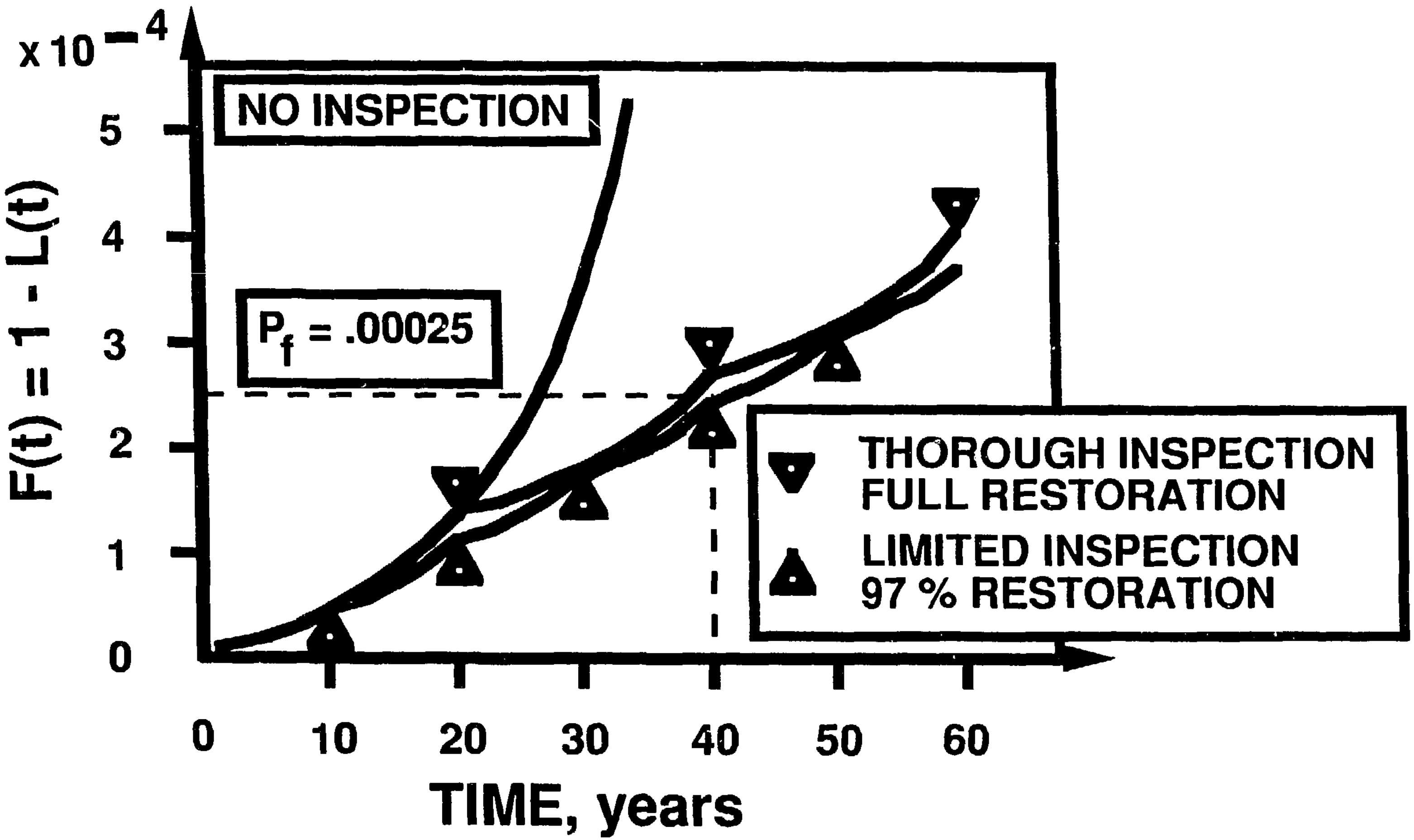
SIMPLE PARAMETRIC REPRESENTATIONS OF TIME-DEPENDENT LOAD AND STRENGTH HAVE BEEN UTILIZED TO REPRESENT EFFECT OF DEGRADATION IN COMPONENT STRENGTH ON THE COMPONENT RELIABILITY FUNCTION



Single Component Failure Probability Depends on Selection of Degradation Model



INSPECTION/MAINTENANCE STRATEGIES CAN BE DESIGNED SO THAT FAILURE PROBABILITY IS KEPT LOWER THAN TARGET PROBABILITY



PRESENTATION WILL ADDRESS FOUR TOPICS

- INTRODUCTION
- BACKGROUND
- STRUCTURAL AGING PROGRAM DESCRIPTION AND STATUS
- SUMMARY

STRUCTURAL AGING PROGRAM IS ADDRESSING BARRIERS THAT EXIST RELATIVE TO BEING ABLE TO PREDICT FUTURE PERFORMANCE OF CRITICAL CONCRETE STRUCTURES IN NUCLEAR POWER PLANTS

- A SYSTEMATIC APPROACH OR METHODOLOGY FOR TREATING THE PROBLEM
- AN EFFECTIVE MECHANISM FOR OBTAINING AND REPORTING DATA ON THE ACTUAL INSERVICE PERFORMANCE OF MATERIALS
- KNOWLEDGE OF THE MECHANISMS OF DEGRADATION
- KNOWLEDGE OF THE ENVIRONMENTAL FACTORS CAUSING DEGRADATION
- THE ABILITY TO SIMULATE OR ACCOUNT FOR THE SYNERGISM BETWEEN DEGRADATION FACTORS
- MATHEMATICAL MODELS DESCRIBING MATERIAL BEHAVIOR IN SPECIFIC ENVIRONMENTS OR APPLICATIONS

SEVERAL PRODUCTS WILL RESULT FROM THE STRUCTURAL AGING PROGRAM

- FORMULATION AND IMPLEMENTATION OF A "PC-BASED" STRUCTURAL MATERIALS INFORMATION CENTER

- DEVELOPMENT OF CRITERIA FOR ISI/STRUCTURAL ASSESSMENT PROCEDURE(S) TO PROVIDE DATA NEEDED TO EVALUATE THE CURRENT CONDITION AND TO TREND PERFORMANCE OF CRITICAL CONCRETE COMPONENTS

- DEVELOPMENT OF TECHNIQUES TO IDENTIFY, ASSESS, AND PREDICT RATE EFFECTS OF ENVIRONMENTAL STRESSORS AND AGING FACTORS THAT CAN LEAD TO CONCRETE DETERIORATION, AS WELL AS PROCEDURES TO MITIGATE THESE EFFECTS

SEVERAL PRODUCTS WILL RESULT FROM THE STRUCTURAL AGING PROGRAM (CONT.)

- DEVELOPMENT OF GUIDELINES, AS WELL AS CRITERIA FOR THEIR APPLICATION, FOR REPAIR, REPLACEMENT AND RETROFITTING OF DEGRADED CONCRETE COMPONENTS
- DEVELOPMENT OF A RELIABILITY-BASED METHODOLOGY TO ESTIMATE CURRENT AND FUTURE CONDITION OF AGED CONCRETE STRUCTURES