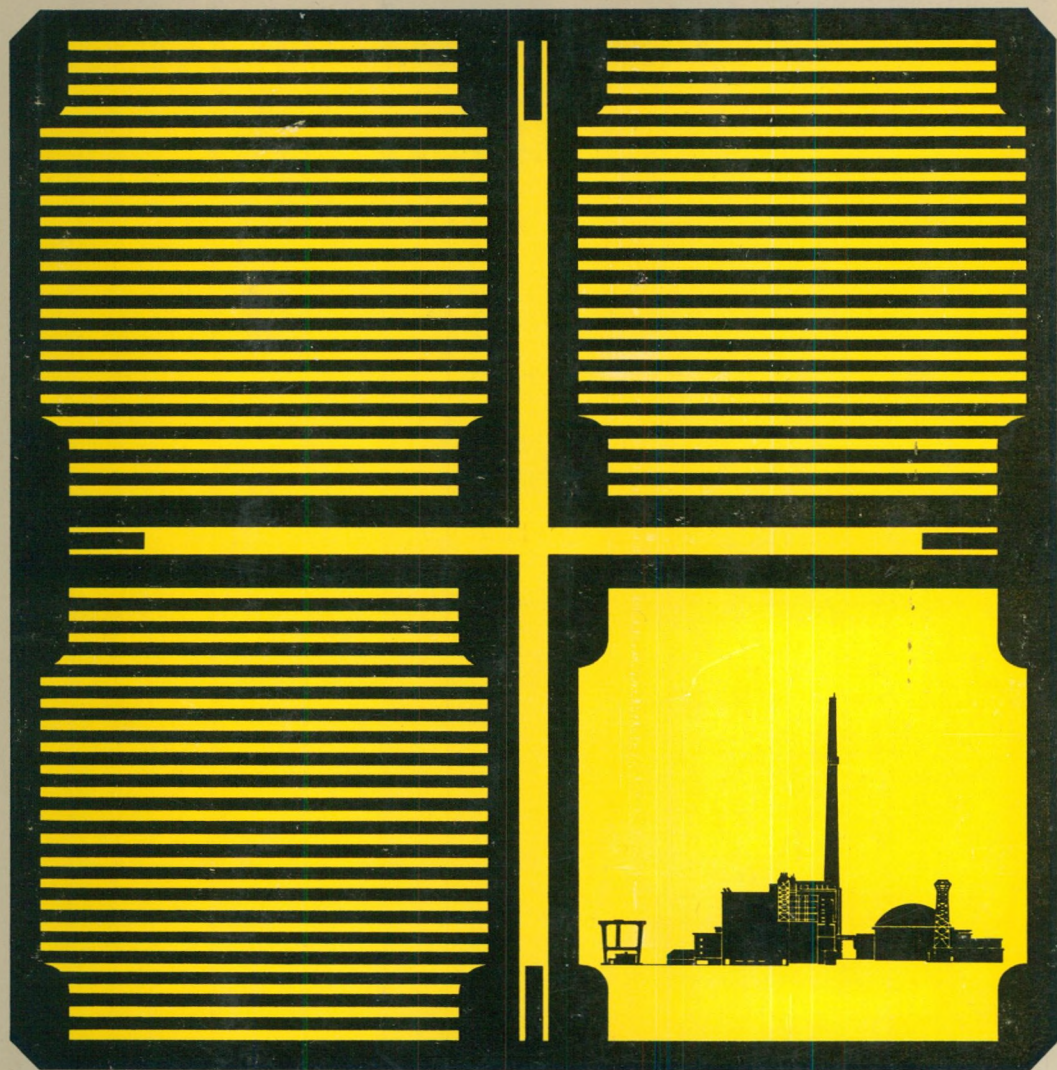




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Conference Proceedings, Vienna, 4-8 June 1963

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INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1963

OPERATING EXPERIENCE
WITH POWER REACTORS

VOL. I

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

PROCEEDINGS SERIES

OPERATING EXPERIENCE WITH POWER REACTORS

PROCEEDINGS OF THE CONFERENCE ON OPERATING
EXPERIENCE WITH POWER REACTORS
HELD BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AT VIENNA, 4-8 JUNE 1963

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In two volumes

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VIENNA, 1963

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FOREWORD

At the beginning of 1963 nuclear power plants produced some 3 500 000 kW of electrical power to different distribution grids around the world. Much significant operating experience has been gained with these power reactors, but this experience is often not collected in such a way as to make it easily available.

The International Atomic Energy Agency convened a Conference on Operating Experience with Power Reactors in Vienna from 4-8 June 1963 which was attended by 240 participants representing 27 of the Agency's Member States and six international organizations. At the Conference, 42 papers giving detailed experience with more than 20 nuclear power stations were discussed. Although similar meetings on a national or regional scale have been held earlier in various countries, this is the first arranged by the Agency on a world-wide basis. Some of the detailed material may have been given earlier but for the most part it represents new and recently acquired experience, and for the first time it has been possible to compile in one place such extensive material on the operating experience with power reactors.

The Conference discussed the experience gained both generally in the context of national and international nuclear power development programmes, and more specifically in the detailed operating experience with different power reactor stations. In addition, various plant components, fuel cycles, staffing of nuclear plants and licensing of such staff were treated.

It is hoped that these Proceedings will be of interest not only to nuclear plant designers and operators who daily encounter problems similar to those discussed by the Conference, but also to those guiding the planning and implementation of power development programmes.

EDITORIAL NOTE

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For the sake of speed of publication the present Proceedings have been printed by composition typing and photo-offset lithography. Within the limitations imposed by this method, every effort has been made to maintain a high editorial standard; in particular, the units and symbols employed are to the fullest practicable extent those standardized or recommended by the competent international scientific bodies.

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I

GENERAL REVIEWS OF EXPERIENCE WITH NUCLEAR PLANTS IN THE CONTEXT OF NATIONAL PROGRAMMES

THE EVOLUTION OF THE UNITED STATES CIVILIAN POWER PROGRAMME

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UNITED STATES ATOMIC ENERGY COMMISSION, WASHINGTON, D.C.

Abstract — Résumé — Аннотация — Resumen

THE EVOLUTION OF THE UNITED STATES CIVILIAN POWER PROGRAMME. The elements of major importance in our civilian power programme are economics and efficient utilization of nuclear fuel resources. Extensive development work will be required by each of these factors, but development work alone in the absence of successful nuclear power plant operating experience will not suffice. An important start towards this end has been made with water reactors and additional operating experience on these systems will accumulate at an accelerated pace over the next few years. These efforts are producing results inasmuch as utility companies are beginning to undertake the construction of large water reactors on the basis of firm economics. Operating experience with other reactor concepts is now beginning to be accumulated in the United States. To the extent that these reactor concepts satisfy the requirement of economics and efficient utilization of nuclear fuel, additional large nuclear power plants employing these reactor concepts will be undertaken by utility companies.

ORIENTATION DU PROGRAMME DE PRODUCTION D'ÉNERGIE A DES FINS NON MILITAIRES AUX ETATS-UNIS. Les considérations qui influent le plus sur l'orientation du programme de production d'énergie à des fins non militaires aux Etats-Unis, sont la rentabilité et l'utilisation efficace des ressources en combustible nucléaire. L'un et l'autre de ces facteurs exigent non seulement des recherches approfondies, mais aussi une bonne expérience pratique de l'exploitation de centrales nucléaires. On a déjà acquis d'importants éléments d'information sur le fonctionnement des réacteurs à eau et on en recueillera de plus en plus au cours des prochaines années. Ces efforts sont fructueux puisque les compagnies de distribution d'électricité commencent à faire construire de grands réacteurs à eau dont la rentabilité est pratiquement assurée. On commence maintenant aux Etats-Unis à rassembler des données sur le fonctionnement d'autres types de réacteurs. Dans la mesure où ces réacteurs satisferont aux critères de rentabilité et d'utilisation efficace du combustible nucléaire, les compagnies de distribution d'électricité les utiliseront pour la construction de grandes centrales nucléaires nouvelles.

ОЦЕНКА ГРАЖДАНСКОЙ ЭНЕРГЕТИЧЕСКОЙ ПРОГРАММЫ СОЕДИНЕННЫХ ШТАТОВ. Самыми важными элементами нашей гражданской энергетической программы являются экономические аспекты и эффективное использование ресурсов ядерного топлива. Каждый из этих элементов потребует проведения обширных опытно-конструкторских работ, однако одних опытно-конструкторских работ без опыта успешной эксплуатации ядерной электростанции будет недостаточно. Важный первый шаг в этом направлении был сделан с водяными реакторами, а в последующие несколько лет накопление дополнительного эксплуатационного опыта в отношении этих систем будет происходить ускоренными темпами. Эти усилия дают свои результаты, так как коммунальные компании начинают строить крупные водяные реакторы на основе устойчивой рентабельности. Эксплуатационный опыт в отношении других видов реакторов должен быть накоплен сейчас в Соединенных Штатах в таком объеме, который позволит бы сделать эти виды реакторов экономичными, а коммунальные компании осуществят эффективное использование ядерного топлива и строительство новых крупных ядерных электростанций с этими видами реакторов.

EVOLUCIÓN DEL PROGRAMA CIVIL DE ENERGÍA ELÉCTRICA DE LOS ESTADOS UNIDOS. Los elementos de mayor importancia en el programa civil de energía eléctrica de los Estados Unidos son los aspectos económicos y el aprovechamiento eficaz de los combustibles nucleares. El conocimiento cabal de estos factores requerirá una extensa labor de desarrollo que, sin embargo, no será suficiente si se carece de experiencia práctica en materia de explotación normal de centrales nucleoelectricas. Los reactores de agua constituyen un primer paso muy importante hacia la consecución de ese objetivo y en los próximos años se seguirá acumulando experiencia acerca de dichos sistemas. Estos esfuerzos comienzan a dar resultados positivos por cuanto

las compañías de electricidad están iniciando la construcción de grandes reactores de agua basados en parámetros económicos bien establecidos. En los Estados Unidos se está empezando a adquirir experiencia práctica con otros conceptos de reactor; en la medida en que estos conceptos de reactor satisfagan los requerimientos económicos y aseguren un aprovechamiento eficaz del combustible nuclear, las compañías de electricidad irán construyendo grandes centrales nucleoelectricas basadas en estos conceptos de reactor.

1. INTRODUCTION

I am grateful for this opportunity to discuss with you the evolution of the USAEC civilian power reactor programme, as it exists today as well as the future programme based on the objectives established in the 1962 Civilian Nuclear Power Report to the President of the United States of America.

2. HISTORY

In its early phases the programme was largely one of developing technology. It relied to a great extent on knowledge gained from other reactor programmes, such as the plutonium-producing production reactors, naval propulsion reactors and research and test reactors used for scientific purposes. In 1953, the Commission embarked upon a five year experimental programme to develop reactors holding promise for civilian power application. Construction was started on several experimental power producing reactors on Commission-owned sites, and one prototype reactor, Shippingport, on a utility grid. In 1954-1955 increased emphasis was given to the programme through the addition of a "Power Demonstration Program" under which the Commission and industry have co-operated in building and operating a number of nuclear power-plants on utility grids. One phase of this programme consisted of the Commission building and owning "prototype" reactors which are operated by utilities that buy steam; the other phase being where utilities are given research and development assistance in designing and constructing their own reactors and for a few years no charge is made for the lease of Government-owned nuclear fuel. This phase was recently modified to the extent that, in addition to the above, the Commission furnished Title I and II design assistance for the complete plant. In 1958, as the five year experimental programme ended, the Commission conducted a series of detailed studies and evaluations of all reactor concepts to assess their economic and technical potential. On the basis of these studies, the Commission published a series of reports, known as the "Ten Year Program", which established short-range economic targets together with long-range goals in economics, resources conservation and international leadership. This programme has served as a general guide to the Commission until 1962, at which time the President requested that the Commission take a "new and hard look at the role of nuclear power in our economy". This request resulted in a very detailed review of the programme and a report has since been prepared and submitted to the President. I will discuss the conclusions of this report later in these remarks.

3. PRESENT STATUS

The prime objective of the "Ten Year Program" was the achievement of competitive nuclear power in high-fuel-cost areas by 1968. The achievement of this objective now appears attainable with water-cooled and moderated reactors producing saturated steam. Probably no astounding technical "break throughs", in the classical sense of the term, can be identified, except perhaps the development of oxide fuels and the demonstration that neither local nor bulk boiling of the coolant-moderator in these cores produced deleterious consequences. Nevertheless, extremely importance progress has been made, beginning with BORAX 1-4, EBWR and VBWR, and later supplemented by the only commercial size nuclear power-plants which have thus far operated extensively in the United States of America - SHIPPINGPORT, DRESDEN and YANKEE and supplemented recently by the Consolidated Edison, Big Rock Point and Humboldt Bay plants. The former three of these plants have, as of the end of April 1963, generated approximately 1 516 675 500, 2 254 698 000 and 2 137 270 800 gross kWh, respectively.

Each has contributed, in part, to the state of knowledge being employed in the *19 additional light-water civilian-power reactors now committed and in various stages of negotiation, design, construction, start-up and operation in the United States, Europe and the Far East.

Light-water reactors of 400 MW(e) and greater, producing saturated steam, are now available for purchase on the basis of firm, warranted prices applied to both capital and fuel costs with estimated resultant total power costs of approximately 6.0-7.0 mill/kWh on a national basis. If certain assumptions are correct, these reactors have thus reached a state of economic competitiveness in high-fuel-cost areas of the United States where system loads permit the use of large units.

Perhaps the best over-all measure of accomplishments since the time the Dresden and Yankee plants were committed is to compare their estimated capital costs (approximately \$ 350 to \$ 400/kW) and initial fuel costs (approx-

* BODEGA BAY	(USA)	313 MW(e)
BONUS	(USA)	16.3 MW(e)
BORAX V	(USA)	3 MW(e)
BR-3	(Belgium)	11.5 MW(e)
City of Los Angeles	(USA)	490 MW(e)
Connecticut Yankee	(USA)	490 MW(e)
JAERI	(Japan)	12 MW(e)
Jersey Central	(USA)	500 MW(e)
KAHL	(Fed. Rep. Germany)	15 MW(e)
KRB	(Fed. Rep. Germany)	237 MW(e)
LACBWR	(USA)	50 MW(e)
PATHFINDER	(USA)	62 MW(e)
RAVENSWOOD	(USA)	1000 MW(e)
RCPA	(USA)	22 MW(e)
SAXTON	(USA)	3.2 MW(e)
SELNI	(Italy)	165 MW(e)
SENA	(Italy)	242 MW(e)
SENN	(Italy)	150 MW(e)
S. Calif. Edison	(USA)	375 MW(e)

mately 4 mill/kWh) with the warranted capital and fuel costs now being offered on plants of 400 MW(e) or larger. These are respectively, \$165 to \$190/kW and roughly 1.8 to 2.25 mill/kWh corresponding to reductions of 50% in both capital and fuel costs.

The capital cost reductions cited are due in part to increase in plant size, but this in itself is a major accomplishment inasmuch as the 100-200 MW(e) plants undertaken in the mid-fifties were about as large as could be justified on the basis of then existing technology. I believe this trend of increasing plant size is an important factor and in the formulation of our development programmes we have taken into account its expected continuation.

The fuel cost reductions are due to improvements in fabrication technology and the higher fuel exposures which are being offered—of the order of 15 000–22 000 MWd/t. Admittedly, the exposure of large reactor cores to these levels is not yet an accomplished fact. The YANKEE Core I, for example, containing 20.9 metric tons of uranium, achieved an average exposure of 8350 MWd/t, with peak local exposures of roughly twice that value. Thus the attainment of the high average fuel exposures, on which present fuel-cost estimates are predicated, necessarily presume continued success in fuel development programmes.

It is believed that capital and fuel cost reductions of 20–25% beyond present levels will be realized if programmes now in progress are successful.

4. SATURATED LIGHT-WATER-REACTOR IMPROVEMENTS

The major areas of improvement which can now be identified are:

- (1) More accurate knowledge of design limits;
- (2) Scale-up of reactors to 1000 MW(e);
- (3) Chemical shim and spectral-shift control; and
- (4) Recycling of plutonium.

4.1. Design limits

Virtually every water reactor, including experimental reactors, has demonstrated the ability to operate at power levels substantially in excess of the design power level. For example, the DRESDEN reactor had its authorized thermal power increase from 629 to 700 MW(t); the YANKEE first core, designed for 392 MW(t), operated for a large part of its lifetime at 485 MW(t); YANKEE Core II, identical in all respects to Core I (with the exception of the retention of two Core I assemblies in the reactor for extended burn-up) operated at 540 MW(t). While these power increases, exceeding the design power levels, have been gratifying, they also indicate the lack of precision which exists in our knowledge of design methods and performance limits. Similarly, the ability to build plants within initial cost estimates and schedules has not, in every instance, been demonstrated.

In August 1962 the Commission issued an invitation to utilities and industrial firms requesting proposals for 400 MW(e) or larger water-reactor plants. Under this invitation, the Commission, in addition to the previous forms of research and development and waiver of use charge assistance

available under the third round of our power demonstration reactor programme, extended Title I and II design assistance for the entire plant.

Two proposals were received as a result of this invitation: one from the City of Los Angeles Department of Water and Power and the other from Connecticut-Yankee. Both plants have a rating of 490 MW(e) with an excess capacity being designed into the secondary system to allow for an anticipated increase in reactor thermal ratings.

With respect to core-design methods and performance limits, uncertainties exist in the area of heat transfer and hydraulics, materials behaviour and physics (particularly lifetime and reactivity coefficient predictions at high fuel exposures). A programme has been undertaken to evaluate the performance of the YANKEE Core I, involving destructive and non-destructive examination of a large number of the discharged fuel assemblies. A Core I control rod will similarly be examined. All mechanical, chemical, physical and metallurgical properties of the fuel assemblies and control rod will be determined, compared with design predictions and pre-irradiated values. Design methods employed will then be revised where necessary, to provide agreement prediction and experience.

It is planned that similar programmes on other reactor cores embodying unique features will be undertaken in the near future.

4.2. Scale-up of reactors to 1000 MW(e)

An important factor currently receiving attention in the drive to reduce power-generation costs is the trend toward increased plant size. It is characteristically true for both fossil fuel and nuclear plants that unit costs decrease with increasing plant size. Considering the projected pattern of utility load growth and system interconnections in the United States, and the somewhat higher ratio of nuclear-plant capital costs to fuel costs as opposed to fossil-fuelled plants. It appears that optimum nuclear plants must have the ability of being built in single-unit sizes of up to 1000 MW(e).

We have initiated design studies aimed at determining the technical and economic feasibility of a 1000 MW(e) single unit indirect and direct cycle saturated-steam water reactors. It is probable that problems will be encountered in component design, fabrication and shipment, turbine design, steam-water separation, and perhaps, core design.

4.3. Chemical-shim and spectral-shift control

Development work on both chemical shim and spectral-shift control, applied to indirect-cycle water reactors, has been in progress in the United States under sponsorship by both government and industry. Both of these concepts have the potential of extending core reactivity lifetime, and achieving higher average power density and specific power through improved power distributions.

Supplementary control via chemical means (boric acid) has been successfully employed for cold shut-down of the Yankee reactor. Continuous chemical-shim control will be tested in the SAXTON reactor and is also provisionally included in the SELNI, SENA, Southern California Edison, City of Los Angeles and Connecticut-Yankee reactors. The problem requiring

resolution is to control, in an optimum manner, the reactor power output and fuel exposure.

The physics of the spectral-shift reactor has been the subject of an USAEC sponsored programme of physics experiments and analysis such that this reactor concept is now available from industry on a firm-price basis. The novel features of this concept centre on the use of varying D_2O - H_2O mixtures to control reactivity during the core lifetime.

This concept's potential advantages are realized by absorption of neutrons in fertile materials, thereby achieving high conversion ratios (up to 0.95 in equilibrium U^{233} , thorium fuel). This is accomplished by hardening the spectrum through the use of a high ratio of D_2O - H_2O at the beginning of life, thereby increasing resonance absorption in fertile material. As fuel burn-up progresses, the spectrum is gradually softened by decreasing the D_2O - H_2O ratio to maintain core reactivity. This control concept also results in minimizing the number of control rods in the core during operation, thus permitting improved power distribution, higher specific power and power density and higher average fuel exposures. Its principal uncertainties are D_2O loss rates and possible variations in reactivity coefficients at high fuel exposures.

4.4. *Plutonium recycle*

Because of the potential extensions of our nuclear fuel resources through the use of by-product plutonium as a reactor fuel and the effect of the assigned plutonium value on nuclear fuel cost calculations, it is important that the technology and economics of plutonium utilization in converter reactors, and particularly in light-water reactors, be established as soon as possible. We are now entering the stage in our over-all plutonium-recycle programme where significant numbers of practical plutonium-bearing fuel elements will be irradiated in typical light-water reactor environments. The first such irradiation is expected to occur in the EBWR in early 1964 and will consist of a partial core loading of plutonium-bearing fuel assemblies. This experiment will reduce the present uncertainties in physics and extended burn-up characteristics of plutonium-bearing fuels in a typical boiling-water-reactor environment. Other experimental light-water reactors may be used in a manner similar to EBWR. These experiments are expected to establish the minimum fuel value of plutonium, i. e. its value when used in a thermal converter reactor.

I have devoted a great deal of time to discussing the saturated-water reactor programme as well as its technical potential inasmuch as this concept, through technology developed over the past years and favourable operating experience, has resulted in acceptance by utility companies as a reliable and economic source of power.

In addition to the light-water reactors mentioned earlier the United States has an additional seven commercial-power reactors in various stages of design, construction and operation. These include the Hallam Nuclear Power Facility, Carolinas-Virginia Tube Reactor, Enrico Fermi Atomic Power Plant, Piqua Organic Moderated Reactor, Experimental Gas Cooled Reactor, Peach Bottom Atomic Power Station and the EBR II. Widespread acceptance by the utility industry of these concepts is expected to occur when successful

operating experience of statistically significant proportions has been accumulated.

5. FUTURE

As I stated earlier, the President of the United States requested the Commission to take a "new and hard look at the rôle of nuclear power in our economy". This review was completed taking into account the need for nuclear power, the responsibilities of the USAEC, the state of nuclear power technology, its future possibilities and the existence and potentialities of the nuclear industry. Based on these items, the USAEC arrived at the following statement of objectives: The over-all objective of the Commission's nuclear power programme should be to foster and support the growing use of nuclear energy and, importantly, to guide the programme in such directions as to make possible the exploitation of the vast energy resources latent in the fertile materials, uranium-238 and thorium.

More specific objectives may be summarized as follows:

(1) The demonstration of economic nuclear power by assuring the construction of plants incorporating the presently most competitive reactor types.

(2) The early establishment of a self-sufficient and growing nuclear-power industry that will assume an increasing share of the development costs.

(3) The development of improved converter and, later, breeder reactors to convert the fertile isotopes to fissionable ones, thus making available the full potential of the nuclear fuels.

(4) The maintenance of United States technological leadership in the world by means of a vigorous domestic nuclear power programme and appropriate co-operation with, and assistance to, our friends abroad.

The future programme for civilian nuclear power has been divided into two phases, that is a programme for the immediate future and a long-range programme. In the immediate future, emphasis will be given to other converter type reactors, i. e. light-water saturated systems, but only if they promise early marked improvement in unit costs for power, are markedly higher ratio converters, have direct, important technical bearing on breeder systems, or offer potential for other applications such as process heat. Several reactor systems give promise of meeting the above criteria and will be continually evaluated to assess if these concepts possess significant advantages to warrant prototype or full scale construction.

Major programme emphasis will be placed on development of thermal and fast breeder reactors. Prime emphasis is being placed on this concept to realize achievement of one of the major objectives of Commission's programme, that is, to make maximum utilization of energy resources. Experimental reactors will be built in the near future to assess control and safety problems. It is hoped that in the late 1960's or early in the following decade the stage of operating prototypes will be reached. In the absence of unforeseen difficulties and continued vigorous developmental support, practical and economic full-scale breeder reactors are expected to be achieved by late 1970's or early 1980's.

6. SUMMARY

In summary I should like to reiterate the elements of major importance in our civilian power programme – economics and efficient utilization of nuclear fuel resources. Extensive development work will be required by each of these factors, but development work alone, in the absence of successful nuclear power-plant operating experience will not suffice. An important start towards this end has been made with water reactors and additional operating experience on these systems will accumulate at an accelerated pace over the next few years. These efforts are producing results inasmuch as utility companies are beginning to undertake the construction of large water reactors on the basis of firm economics.

Operating experience with other reactor concepts is now beginning to be accumulated in the United States to the extent that these reactor concepts satisfy the requirement of economics, and efficient utilization of nuclear fuel, additional large nuclear power-plants employing these reactor concepts will be undertaken by utility companies.

DISCUSSION

S. YIFTAH: You mentioned average figures of 6.7 mill/kWh and \$165-\$190 per kW installed. You also said that there is a tendency to favour larger plants, ranging from 500 to 1000 MW(e), and I wonder whether the cost figures apply to these large plants. If not, to what sizes do they apply?

C.A. PURSEL (on behalf of A. Giambusso and W.R. Voigt): The figures apply to light-water-moderated plants in the 400-500 MW(e) range. They are based on quotations recently made to the Department of Water and Power of the City of Los Angeles and to the Connecticut Yankee Power Company for plants in this range.

U. ZELBSTSTEIN: Could you give us any idea of the increment that one would have to pay in order to operate such reactors in Europe or the Far East? In other words, what would be the real cost of operation, taking into account transport and fuel reprocessing costs, if such reactors were installed in countries other than their country of origin?

C.A. PURSEL: I understand that fuel-cycle costs are not appreciably influenced by the cost of overseas shipment (United States to Europe and return), and I presume that they have been estimated quite carefully in connection with, for example, the SENN and SELNI plants.

P. EDDY: Perhaps I can contribute a partial answer. The General Electric Co. has devoted considerable study to the possibilities of boiling-water plants in overseas areas. It has been found, in general, that the prices which Mr. Pursel quoted for the United States would remain valid in most countries. Local write-offs, taxes, the manner in which the utility keeps its financial records and staffs its plant – all these things would probably have a greater effect on operating costs than the mere fact of being located outside the United States. Most light-water reactors use very slightly enriched fuel which does not bring an extreme penalty. If anyone is interested in a specific application for a particular area, we should be pleased to co-operate in providing more precise information.

F.R. BELOT: You mentioned a burn-up of 8350 MWd/t for the first YANKEE core. I should like to know what you consider the end of the core's life to be. Is it assumed that the core has attained its maximum burn-up when xenon poisoning makes it impossible to start up again promptly after shut-down, or is it considered advantageous to exhaust the last reserves of reactivity even at the risk of less reliable operation?

C.A. PURSEL: The YANKEE core was removed, as you probably know, in order to obtain a higher price for the plutonium, which was guaranteed, I believe, up to June 1963. Needless to say, it had not run out of reactivity by the time it was removed. The question when to remove a core because of fission-product poisoning is a continuing problem. I have long believed that reactor cores are too expensive to be removed from service simply because the combination of loss of enrichment and build-up of fission products no longer permits full xenon override after shut-down. As a practical matter, I believe that plants will be derated, if necessary, near the end of core life to obtain as much thermal energy as possible.

B. SAITCEVSKY: What is the price per kilowatt installed in a conventional thermal plant as compared with the \$165-\$200 per kW quoted for light-water nuclear plants?

C.A. PURSEL: I cannot give you precise figures, although I know that the cost of a conventional fossil-fuelled plant is considerably less than that of a nuclear plant. Costs vary, of course, depending on the type and location of the plant and the time when the estimate is made. I have heard of quotations varying from \$100 to \$125 per kW installed. Possibly someone else here is better informed.

L. MINNICK: Perhaps I can shed a bit of light on the subject. Costs vary throughout the United States depending on the proximity of the plant to conventional fuel deposits.

In New England the cost of a conventional plant of 150 MW, built in 1960 and similar to the Yankee plant, would have been about 8.0 mill/kWh. YANKEE is now operating at a cost of approximately 10 mill/kWh. The Connecticut Yankee plant - to go into operation in 1967 - will produce power at approximately 7.0 mill/kWh, a cost comparable to that of a conventional plant similar in size and built at the same time. Its initial capacity will be 490 MW(e), but this is expected to rise to something like 600 MW(e) once the core has been thoroughly tested. I believe the total cost of the plant is to be \$70 million.

G.B. SCURICINI: In further reply to Mr. Saitcevsky, it might be recalled that United States sources last year, in a statement before the Joint Congressional Committee, gave a figure of about \$50 as the difference between the price per kW installed in conventional and nuclear plants. The costs were approximately \$120 for conventional plants and \$175-\$180 for nuclear plants. Perhaps a representative of General Electric would care to comment on these figures, since they are now a year out of date.

P. EDDY: They are just about right, although the cost of nuclear plants is gradually getting lower. In one specific case I know of in the United States, where the construction of a 600 MW(e) coal-fired plant is being considered in the neighbourhood of a coal mine, the capital cost will be something over \$100 per kW, and the transmission cost to the load centre will represent an additional \$20 per kW. We have to bear in mind that there are many

extraneous factors which can affect the capital cost of a plant. One has to consider whether a completely new plant is to be built, or whether additional capacity is to be added to an already existing site; whether the site preparations have already been paid for or must be reckoned into the capital cost; the location of the plant and the climate prevalent there; whether the turbine can be installed outside or an insulated building must be provided, and so forth. For these reasons it is difficult to apply perfectly uniform standards in calculating costs.

LE DÉVELOPPEMENT EN FRANCE DE LA FILIÈRE URANIUM NATUREL - GRAPHITE - GAZ CARBONIQUE. EXPÉRIENCE ACQUISE ET PERSPECTIVES D'AVENIR

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Abstract — Résumé — Аннотация — Resumen

THE DEVELOPMENT OF NATURAL URANIUM-GRAPHITE-CO₂ REACTORS IN FRANCE: EXPERIENCE GAINED AND FUTURE PROSPECTS. Natural uranium-graphite-CO₂ reactors are the foundation of the French nuclear power station programme. The Chinon reactors EDF1, EDF2 and EDF3 follow on from the Marcoule plutonium producers G2 and G3, and this group will account for most of the 850 MW(e) to be installed under the third equipment plan by 1965.

G2 and G3 have been supplying the French national grid since 1959 and 1960 respectively. EDF1 underwent preliminary tests at the beginning of 1963 preparatory to coming onto power operation; EDF2 is expected to go into service at the end of 1964 and EDF3 during 1965.

The purpose of the paper is to summarize the development of this reactor type in the light of the experience already acquired therefrom. Points considered are:

- (a) Progress in the laboratory on various technical aspects, including neutron, thermal, mechanical and metallurgical problems, and on the development prospects which this progress opens up;
- (b) Experience gained from the construction of different reactors, and the technical and economic conclusions to be drawn therefrom; and
- (c) Experience gained from the operation of the Marcoule reactors and from the first tests of EDF1, together with the implications of this experience for future reactor projects.

Finally, an attempt is made to evaluate the future prospects of this reactor type within the French programme. In this assessment, a distinction is made between technical feasibility - which can be considered as already established - and economic viability which, it is hoped, will be demonstrated by the reactors now under construction.

LE DÉVELOPPEMENT EN FRANCE DE LA FILIÈRE A URANIUM NATUREL GRAPHITE - GAZ CARBONIQUE. EXPÉRIENCE ACQUISE ET PERSPECTIVES D'AVENIR. Les réacteurs à uranium, graphite et gaz carbonique constituent la base du programme français de centrales électronucléaires. Les réacteurs de Chinon, EDF1, EDF2 et EDF3 font suite aux piles plutonigènes de Marcoule G2 et G3, et cet ensemble représente la majeure partie des 850 MWe dont le troisième plan d'équipement prévoyait l'installation pour 1965.

Les réacteurs G2 et G3 alimentent le réseau électrique français, respectivement depuis 1959 et 1960. EDF1 est soumis au début de 1963 aux essais préliminaires à la montée en puissance; la mise en service d'EDF2 est prévue pour la fin de 1964 et celle d'EDF3 pour 1965.

Le présent rapport se propose de faire le point du développement de la filière à la lumière des enseignements apportés par l'expérience déjà acquise sur ce type de réacteur. On passera successivement en revue:

- a) les progrès réalisés dans les laboratoires sur les différents aspects techniques: neutronique, thermique, mécanique, métallurgique et les perspectives de développement qui en résultent;
- b) les enseignements apportés par la construction des différents réacteurs et les conclusions techniques et économiques que l'on peut en tirer;
- c) l'expérience du fonctionnement des réacteurs de Marcoule et des premiers essais du réacteur EDF1 et ses conséquences sur les projets des réacteurs à venir.

En conclusion, on s'efforcera de préciser les perspectives d'avenir de ce type de réacteur dans le programme français, en distinguant la praticabilité technique qui peut être considérée comme acquise et la rentabilité économique dont on espère qu'elle sera démontrée par l'expérience des réacteurs en construction.

РЕЗУЛЬТАТЫ И ПЕРСПЕКТИВЫ РАЗРАБОТКИ ВО ФРАНЦИИ УРАН-ГРАФИТОВЫХ РЕАКТОРОВ С ТЕПЛОНОСИТЕЛЕМ CO_2 . Уран-графитовые реакторы (на природном уране) с углекислым газом в качестве теплоносителя представляют собой основу французской программы атомных электростанций. Реакторы в Шиноне (EDF-1, EDF-2 и EDF-3) следуют за реакторами для производства плутония в Маркуле (G-2 и G-3), и эта группа реакторов будет вырабатывать большую часть электроэнергии из общего количества 850 мвт (эл.), намеченных к производству в 1965 году по третьему плану.

G-2 и G-3 питают французскую электросеть соответственно с 1959 и 1960 годов. В начале 1963 года прошел предварительное испытание по подъему мощности EDF-1; ввод в эксплуатацию EDF-2 предусмотрен на конец 1964 года, а EDF-3 - в 1965 году.

Излагается вопрос о разработке семейства реакторов в свете данных, полученных в процессе испытаний этого типа реактора. Дается обзор:

- а) достижений лабораторий в различных технических областях: нейтронной, тепловой, механической, металлургической, и вытекающих из этого последствий и перспектив;
- б) сведений, полученных в процессе строительства различных реакторов, а также технических и экономических выводов, которые могут быть из этого извлечены;
- в) опыта работы реакторов в Маркуле и первых испытаниях реактора EDF-1 вместе с его влиянием на последующие проекты реакторов.

В заключение уточняются перспективы этого типа реактора во французской программе. При этом проводится различие между технической практичностью, которая может считаться достигнутой, и экономичностью, по поводу которой полагают, что она будет выявлена при испытаниях строящихся реакторов.

DESARROLLO EN FRANCIA DE LA SERIE DE REACTORES DE URANIO NATURAL - GRAFITO - CO_2 . EXPERIENCIA ADQUIRIDA - PERSPECTIVAS FUTURAS. El programa francés de centrales nucleoelectricas está basado en los reactores de uranio-grafito-anhídrido carbónico. Los reactores de Chinon, EDF1, EDF2 y EDF3, derivan de los reactores plutonígenos G2 y G3 de Marcoule, y la producción de este conjunto de reactores representa la mayor parte de los 850 MW(e) de potencia instalada previstos para 1965 en el tercer plan de desarrollo de la electrificación.

Los reactores G2 y G3 alimentan la red francesa de electricidad desde 1959 y 1960 respectivamente. El EDF1 está sometido desde principios de 1963 a los ensayos preliminares a la puesta en marcha; el EDF2 comenzará a funcionar a fines de 1964 y el EDF3 en 1965.

En la memoria se expone el estado actual de desarrollo de esta serie de reactores. Se examinan sucesivamente:

- a) los progresos realizados en los laboratorios en lo que se refiere a los diversos aspectos técnicos - neutrónico, térmico, mecánico y metalúrgico - y las perspectivas de desarrollo que dichos progresos ofrecen;
- b) las enseñanzas aportadas por la construcción de los diversos reactores y las conclusiones técnicas y económicas que pueden sacarse de ellas;
- c) la experiencia adquirida con el funcionamiento de los reactores de Marcoule y con los primeros ensayos del reactor EDF1 y sus consecuencias para los futuros proyectos de reactores.

Por último, los autores de la memoria tratan de precisar las perspectivas de este tipo de reactores dentro del programa francés, estableciendo una distinción entre su viabilidad técnica - problema que ya puede considerarse resuelto - y su rentabilidad económica que, según se espera, quedará demostrada cuando entren en servicio los reactores actualmente en construcción.

I. INTRODUCTION

Les réacteurs à uranium-naturel-graphite-gaz carbonique constituent la base du programme français de centrales électronucléaires. Les réacteurs en construction à Chinon par l'Electricité de France (EDF) font suite aux piles plutonigènes G2 et G3, construites et exploitées à Marcoule par le Commissariat à l'énergie atomique, et cet ensemble représente la majeure partie des 850 MWe dont le troisième plan d'équipement prévoyait l'installation pour 1965.

TABLEAU I

**PUISSANCE UNITAIRE ET DATE DE MISE EN SERVICE
DES RÉACTEURS G2, G3, EDF1, EDF2 et EDF3**

	G2	G3	EDF1	EDF2	EDF3
Puissance thermique (MW)	250	250	300	800	1560**
Puissance électrique nette (MW)	37	37	68	198	480**
Date de mise en service	avril 59	avril 60	mi 63*	fin 64*	début 66*

* Prévission. Actuellement (avril 1963) EDF₁ est soumis depuis le début de l'année aux essais préliminaires à la montée en puissance.

** Les caractéristiques seront atteintes avec le deuxième jeu d'éléments combustibles

Le tableau I donne pour chaque réacteur la puissance unitaire et la date de mise en service (couplage sur le réseau électrique).

En accord avec les prescriptions du quatrième plan d'équipement couvrant la période 1962-1965 qui prévoit l'engagement de 200 MWe en moyenne par an, EDF vient de décider d'engager la réalisation d'une nouvelle centrale, EDF4, dont la construction débutera en 1964 sur le site de Saint-Laurent des Eaux, d'une puissance égale à celle d'EDF3, 480 MWe net, et dont la mise en service doit avoir lieu au cours de l'année 1968.

Nous nous proposons dans cette note de faire le point sur l'expérience pratique acquise à cette date sur les réacteurs de ce programme.

II. PROBLÈMES TECHNIQUES

De G2 à EDF3 des progrès considérables ont été effectués dans tous les domaines comme on peut le constater sur les tableaux de caractéristiques reproduits en annexe. Parmi les accroissements des performances du cœur, on peut noter en particulier l'augmentation simultanée de la puissance spécifique maximum dans l'uranium, qui passe de 3,5 MWth par tonne d'uranium dans G2 et G3 à 6,2 MWth/t dans EDF3, et de la puissance extraite du canal le plus chargé, 260 kW dans G2 comparé à 660 kW dans EDF3.

Ces progrès n'ont été possibles qu'à la suite de modifications importantes dans la conception de l'élément combustible: utilisation d'un tube à la place d'un barreau plein, alliage uranium-molybdène, gaines en chevrons, etc. Les éléments combustibles font l'objet d'une autre communication présentée à cette conférence [1] et nous n'examinerons pas ici dans le détail les modifications intervenues de G2 à EDF3. Mais celles-ci réagissent également sur le bilan de réactivité dans le réacteur qu'il est nécessaire d'étudier avec un soin particulier.

C'est ainsi qu'avant la mise en service des réacteurs G2 et G3, on a procédé, pendant les essais, à une série d'expériences neutroniques [2, 3], qui ont permis de mesurer les caractéristiques nucléaires de réseaux dans lesquels on avait fait varier le type d'éléments combustibles. Des essais

nucléaires ont également eu lieu au démarrage du réacteur EDF1, et font l'objet d'une autre communication [4]. La connaissance du bilan de réactivité est loin d'être le seul but de ces essais; on étudie également les déformations de la distribution du flux dans le cœur en fonction de la position des barres de contrôle et des absorbeurs fixes. L'expérience acquise dans ce domaine au cours de l'exploitation des réacteurs G2 et G3 a été appliquée aux piles EDF: utilisation de barres grises pour le pilotage et la compensation, disposition d'absorbeurs ayant une capture voisine de celle de l'uranium dans des canaux combustibles pour améliorer l'aplatissement radial du flux neutronique obtenu normalement par sous-modération dans la zone centrale.

Il est un autre aspect neutronique qui est particulièrement important: c'est l'évolution de la réactivité et des coefficients de température en fonction de l'irradiation du combustible. Dans ce domaine, un programme expérimental a été mis sur pied, combinant les résultats directs de l'exploitation des réacteurs G2 et G3, avec des mesures spéciales sur réacteurs (oscillations et perturbations) et des mesures indirectes sur combustibles irradiés (analyses et oscillations). Ce programme est exposé dans une autre communication [5]. Il semble qu'on puisse conclure des résultats déjà obtenus que dans ce type de réacteur, du point de vue neutronique, l'irradiation moyenne des éléments déchargés doit pouvoir atteindre 3500 à 4000 MWj/t, et que ces valeurs peuvent être largement dépassées si l'on accepte de compliquer les schémas de déchargement et de procéder à des réarrangements de combustibles.

Mentionnons enfin un dernier aspect neutronique: la cinétique et particulièrement les instabilités spatiales qui se produisent dans les réacteurs de grandes dimensions lorsque le coefficient de température est positif. Ici l'expérience de G2 et G3 est insuffisante car ces piles sont de trop petites dimensions; des essais ont pu cependant être effectués pour vérifier la validité des modèles cinétiques utilisés sur machine à calculer électronique.

L'aspect neutronique n'est évidemment pas le seul qui intervient dans l'augmentation des performances du cœur des réacteurs. Pour la thermodynamique, ce sont les essais hors-pile qui jouent le rôle essentiel; c'est ainsi qu'ont été mis au point les profils de gaine dits «en chevrons» qui, par rapport aux ailettes longitudinales utilisées dans G2, G3 et EDF1, ont permis une augmentation considérable des coefficients d'échange mise à profit dans EDF2 et EDF3. Mais l'expérience du fonctionnement en pile est également très importante, pour la vérification des résultats des essais hors-pile et pour la connaissance expérimentale des facteurs points chauds, facteurs qui fixent les performances des réacteurs en fonction des limites de température admissibles sur l'élément combustible. Les réacteurs G2 et G3 se prêtent particulièrement bien à une connaissance étendue des variations de température dans le cœur en raison de la présence de mesures de température de sortie de gaz individuelles par canal, solution reprise sur EDF1. Pour faciliter la surveillance poussée de ces températures, chaque réacteur a été équipé d'un calculateur permettant une analyse rapide des 1200 mesures [6]. Par ailleurs, les réacteurs vont être prochainement équipés de cartouches de mesure de température de gaine pouvant être chargées en marche, cartouches munies de contacts pour transmission du signal électrique, qui viendront s'ajouter aux cartouches de mesure à fil déjà uti-

lisées. De telles cartouches, déjà en place dans EDF1, sont également prévues sans EDF2 et EDF3.

L'augmentation de la pression du gaz carbonique de refroidissement qui est passée de 15 kg/cm² dans G2 et G3 à 25 kg/cm² dans EDF1, et qui est restée à cette valeur pour EDF2 et EDF3 malgré l'augmentation très importante de la puissance unitaire, donc de la taille, nous conduit à parler des enceintes sous pression. Une communication spéciale à cette conférence est consacrée aux enceintes en béton précontraint [7]. Indiquons seulement que les résultats très satisfaisants obtenus après 4 ans de fonctionnement des réacteurs de Marcoule, ainsi que ceux obtenus sur maquettes, permettent d'accorder une entière confiance à cette solution, susceptible de développements beaucoup plus importants que celle des enceintes en acier utilisées à EDF1 et EDF2. Elle permet en effet d'envisager une augmentation du volume du cœur et de la pression du gaz; elle permet également d'incorporer le circuit primaire de gaz carbonique dans le caisson, ce qui sera réalisé dans EDF4, conduisant à un ensemble cœur-circuit très compact, ne comportant plus les organes délicats que sont les vannes et les soufflets de dilatation, et présentant ainsi une sécurité intrinsèque plus grande. Il est évident que le béton précontraint est la voie à suivre pour les réacteurs futurs de ce type. Des études se poursuivent actuellement dans le but d'aboutir à des simplifications constructives, en ce qui concerne en particulier le refroidissement de la face interne du caisson. On peut signaler à ce sujet que l'expérience du fonctionnement de G2 a montré que le béton peut supporter sans dommage des températures relativement élevées [6].

Il n'est pas possible de passer ici en revue tous les points techniques importants et d'examiner toutes les solutions qui ont été apportées depuis G2 jusqu'à EDF3. Nous mentionnerons seulement le dispositif de chargement-déchargement et les automatismes. On sait que le renouvellement du combustible dans G2 et G3 se fait réacteur en marche. Après des difficultés de mise au point pendant la période suivant la mise en service [6] les installations donnent actuellement toute satisfaction, et ont confirmé tout l'intérêt du déchargement en marche: accroissement du facteur de disponibilité, possibilité d'optimiser les programmes de renouvellement et d'ajuster à tout instant la distribution de puissance par mouvement d'absorbeur. G2 et G3 étant des piles horizontales, les solutions adoptées pour EDF2 et EDF3 sont différentes et n'ont fait l'objet jusqu'à présent que d'essais hors-pile. Des études sont également en cours pour simplifier les mécanismes de chargement-déchargement en profitant des possibilités offertes par le caisson en béton: les dispositifs sont incorporés dans le caisson, ce qui donne un dessin très compact pour l'ensemble du réacteur et de ses annexes. Une maquette de cette solution est actuellement en cours d'essais.

En allant de G2 à EDF3, on a cherché à obtenir l'automatisation la plus complète possible de la centrale. Déjà dans G2 et G3, les mesures de température de sortie donnent lieu comme nous l'avons vue à une analyse rapide par calculateur; de même le système de chargement a permis la mise au point d'un ensemble important d'automatismes séquentiels. Mais c'est dans les réacteurs de Chinon que l'automatisation a été spécialement poussée: démarrage et arrêt automatique, chaîne de régulation transistorisée, utilisation des techniques récentes de traitement des informations. Cette

tendance résulte d'une part du nombre important des mesures et des appareils à contrôler et d'autre part d'un souci d'améliorer la sécurité de l'installation. On espère obtenir ainsi un fonctionnement permanent de la centrale très sûr, avec une exploitation aisée par le personnel de quart.

Pour terminer cette revue des questions techniques, il nous faut mentionner les problèmes de sécurité [8, 9]. Rappelons que dans les réacteurs à uranium naturel-graphite-gaz carbonique ce sont des critères de sécurité, liés aux excursions de température dans le cas de l'accident maximum prévisible, qui limitent les performances du réacteur tout autant que les limites métallurgiques en fonctionnement normal. L'accident maximum envisagé, de G2 à EDF3, c'est la rupture du circuit de gaz carboniqué. Le concept de circuit intégré, tel qu'il sera appliqué à EDF4, permet de repousser les limites de sécurité, et d'envisager pour l'avenir une augmentation notable des performances intrinsèques.

III. PROBLÈMES ÉCONOMIQUES

La France ne dispose actuellement d'une expérience de fonctionnement que sur les piles plutonigènes de Marcoule, G2 et G3, qui, d'un point de vue économique, ne sont pas représentatives des réacteurs électrogènes de la filière, d'une part en raison de leur puissance unitaire trop faible (37 MW par réacteur) et d'autre part en raison de leur mode de fonctionnement plutonigène qui conduit à limiter l'irradiation moyenne de la majorité des éléments combustibles à un taux très inférieur à leur limite technologique ou neutronique.

Par ailleurs, le premier des réacteurs électrogènes de Chinon (EDF1) est pratiquement achevé, mais il a un caractère de prototype (puissance 68 MWe). Les deux autres réacteurs, EDF2 et EDF3, sont en cours de construction. On peut certes se faire une idée assez précise des coûts de réalisation de ces piles, mais il est encore trop tôt pour avancer des chiffres définitifs.

Les considérations précédentes doivent donc nous rendre très prudents en matière de prévisions économiques; il nous paraît cependant possible et utile de donner quelques éléments à la lumière de l'expérience déjà acquise.

Nous examinerons d'abord le coût du cycle de combustible. En ce qui concerne le coût de fabrication des éléments combustibles, la France dispose d'une expérience industrielle importante, tant sur la fabrication des recharges G2 et G3 que sur celle des premières charges EDF1 et EDF2. Pour ce qui est du coût de l'uranium lui-même, le prix du concentré sur le marché mondial a beaucoup baissé dans les dernières années. Dans ces conditions, on peut estimer que pour un prix de concentré de 80 F par kg d'uranium contenu ($6 \text{ } \$/\text{lb } \text{U}_3\text{O}_8$) le coût d'un combustible du type EDF3 et EDF4 (tube $\phi 43 \times 23 \text{ mm}$, avec chemise de graphite) ne devrait pas dépasser 175 F/kg d'uranium gainé, pour une série importante, de l'ordre de quelques centaines de tonnes par an. Il faut noter que la complexité relative de cet élément par rapport au barreau G2 entraîne une augmentation du coût de fabrication par cartouche, mais que cette augmentation est plus que compensée par l'augmentation de la section d'uranium par canal, et par conséquent du poids d'uranium par cartouche.

L'élément essentiel du coût du cycle de combustible est évidemment le taux d'irradiation moyen des éléments combustibles déchargés du réacteur. Nous avons parlé plus haut de l'aspect neutronique, évolution de la réactivité avec l'irradiation, et l'aspect technologique est présenté ailleurs [1]. Il est certain que la France ne dispose pas encore d'une expérience statistique sur le comportement en pile d'une charge complète du réacteur EDF3; elle dispose par contre de nombreux résultats sur la tenue sous irradiation des alliages uranium-molybdène et magnésium zirconium, ainsi que de l'expérience du comportement en pile de plusieurs centaines de milliers de cartouches dans G2 et G3; sur ce dernier point, on peut noter que l'on constate actuellement dans G2/G3 un taux de ruptures de gaines inférieur à 1/20 000, ce qui, même en tenant compte du taux d'irradiation limité auquel sont déchargés les éléments combustibles, est un résultat très satisfaisant. C'est cette expérience qui nous permet d'affirmer que 3500 MWj/t constitue une hypothèse raisonnable pour l'irradiation moyenne dans EDF3, valeur qui a même toutes les chances d'être dépassée dans la pratique.

A partir des données précédentes, il est aisé de calculer le coût du cycle de combustible: en prenant 31% comme valeur du rendement net de la centrale, qui est la valeur obtenue à EDF avec un cycle de récupération d'énergie simple, une seule pression de vapeur avec resurchauffe, et en attribuant une valeur nulle au combustible irradié qui contient pourtant plus de 1 kg de plutonium par tonne, on trouve que le coût moyen du renouvellement du combustible est de 0,67 c/kWh. L'augmentation du taux d'irradiation conduirait à un coût plus réduit, ou compenserait une éventuelle remontée du prix du concentré.

Pour obtenir la valeur du poste combustible dans le prix du kWh, au coût du renouvellement il faut ajouter les charges financières sur le combustible en pile et en réserve sur le site, tenir compte du paiement anticipé du combustible avant son chargement, d'une mauvaise utilisation de la première charge, etc. Avec des hypothèses raisonnables, la répercussion sur le prix du kWh est de l'ordre de 0,23 c/kWh, ce qui conduit à une valeur totale du poste combustible de 0,90 c/kWh. Il est intéressant de signaler à ce sujet que l'investissement correspondant à la charge en pile est de 150 F/kWe.

En ce qui concerne le coût des investissements, quelques points importants méritent d'être soulignés, et en premier lieu l'intérêt présenté par l'augmentation des puissances unitaires des réacteurs pour la diminution du coût par kilowatt électrique installé. Dans ce domaine EDF3, avec une puissance électrique brute de 500 MW répartie en 2 groupes de 250 MWe, représente un progrès important par rapport à EDF2 (200 MWe net pour 2 groupes de 125) et constitue un palier qui a été conservé pour EDF4. Ultérieurement, l'étape 1000 MWe pourra probablement être abordée, avec un gain correspondant supplémentaire sur le coût du kW. Dans le sens opposé, il apparaît qu'en dessous de 250 à 300 MWe le coût du kW augmente rapidement lorsque la puissance diminue.

Bien que nettement moins important, il est un autre facteur qui permet de réduire le montant des investissements: c'est la duplication, ne serait-ce que parce qu'elle entraîne une diminution de la part des études dans le prix total; les études sont particulièrement importantes sur certains postes tels

l'appareil de chargement et le caisson (pour EDF3 on a construit 3 maquettes de caisson sur le site du Chinon). Ce point pourra être vérifié partiellement pour EDF4, dont le cœur et toute la partie supérieure du réacteur sont identiques à ceux d'EDF3. Pour ce qui est des conséquences de l'intégration du circuit primaire dans le caisson, les simplifications constructives importantes qu'elle entraîne devrait conduire à une réduction des coûts, mais il est évidemment beaucoup trop tôt pour pouvoir l'affirmer avec certitude.

Ces diverses raisons, jointes à l'expérience acquise par le CEA, l'EDF, et les constructeurs, ont permis de diminuer le coût des investissements par un facteur 1,6 entre EDF1 et EDF2, et voisin de 3 entre EDF1 et EDF3. Il ne semble plus utopique maintenant d'espérer atteindre dans quelques années un objectif de 1000 F/kWe.

A côté du cycle de combustible et des investissements, la durée de vie de la centrale est un paramètre important du point de vue économique. C'est également celui sur lequel il ne sera pas possible avant longtemps d'avoir des certitudes absolues. L'expérience acquise permet cependant d'apporter des éléments de réponse en ce qui concerne les points les plus délicats, parce que inaccessibles après la mise en service: l'empilement de graphite et le caisson.

Les données expérimentales obtenues sur l'accumulation d'énergie Wigner dans le graphite aux températures rencontrées dans ces réacteurs montrent que la vitesse d'accumulation de cette énergie se sature lorsque l'irradiation croît et qu'il n'y a pas de risque d'atteindre un niveau d'énergie emmagasinée dangereux et incompatible avec la poursuite d'un fonctionnement sûr du réacteur [10]. Quant aux problèmes posés par la déformation du graphite sous rayonnement et ses conséquences possibles sur la géométrie de l'empilement et l'alignement des canaux, la structure entièrement clavetée adoptée sur EDF2 et EDF3 doit permettre à l'empilement de s'accommoder des déformations prévisibles pendant toute la vie du réacteur. Il peut y avoir un risque de fissuration des briques de graphite due à des contraintes excessives; ce risque est très limité dans EDF2 et EDF3 où la variation de température et de flux dans une brique est faible en raison de la présence des chemises, et de toutes façons, d'éventuelles fissures ne devraient pas avoir des conséquences graves sur la tenue de l'empilement en raison des facteurs de sécurité élevés qui sont adoptés. La réaction d'oxydation du graphite par le CO_2 ne devrait pas quant à elle être d'une importance significative pour les températures, pression et intensités de rayonnement mis en jeu dans ces réacteurs.

L'expérience déjà acquise sur les caissons en béton précontraint donne une grande confiance dans leur capacité à soutenir une longue durée de vie. A G2 et G3, après des ennuis mineurs résultant d'une surveillance insuffisante des câbles de précontrainte, un système simple et sûr a été mis en place permettant le contrôle permanent de la tension des câbles. Si une anomalie est constatée, il est toujours possible de procéder au remplacement des câbles. En ce qui concerne le comportement du béton lui-même, il ne peut poser de problème que dans le cas où une détérioration du calorifuge interne ou un incident sur le système de refroidissement de la peau d'étanchéité conduirait à des échauffages anormaux. Des essais récents semblent montrer que même dans ce cas le béton pourrait supporter

aisément des températures plus élevées que celles actuellement prévues en fonctionnement normal

IV. CONCLUSION

La mise au point de centrales électro-nucléaires industrielles pouvant être construites et fonctionner sans aléas est une entreprise qui, même si on se limite à un seul type de réacteur, exige une somme considérable d'efforts et beaucoup de temps. L'expérience française dans le domaine des réacteurs uranium naturel-graphite-gaz carbonique a été acquise à partir du fonctionnement des piles de Marcoule, de la construction des réacteurs de Chinon et des travaux effectués dans les bureaux d'études et les laboratoires du CEA, de l'EDF et de l'industrie. La technique est pratiquement acquise, la rentabilité économique ne l'est pas encore, mais les prévisions que l'on est en mesure de faire à ce sujet sont des plus encourageantes, et il n'est pas téméraire d'espérer que la compétitivité avec l'énergie classique peut être atteinte dès EDF4 qui doit entrer en service au milieu de 1968, et ceci sans préjuger d'une utilisation future du plutonium contenu dans le combustible irradié.

Par ailleurs, et bien que ce ne soit pas le sujet de cette conférence, nous ne saurions passer sous silence les importantes perspectives de progrès technique et économique que l'on aperçoit dès maintenant. L'utilisation d'éléments combustibles annulaires, refroidis extérieurement et intérieurement, l'intégration plus poussée du réacteur et de ses annexes dans le caisson en béton précontraint, sont actuellement à l'étude. Les premiers résultats sont encourageants, et devraient permettre à bref délai la réalisation de centrales de performances spécifiques plus élevées et de puissance unitaire supérieure, conduisant à une réduction supplémentaire des coûts d'investissements et du prix de revient de l'énergie.

ANNEXE

Dans cet annexe on trouvera les caractéristiques d'ensemble des réacteurs (tableau II), les caractéristiques du cœur (tableau III), du combustible (tableau IV), du circuit de CO₂ (tableau V), des circuits de vapeur (tableau VI) et des groupes turbo-alternateurs (tableau VII).

TABLEAU II

CARACTÉRISTIQUES D'ENSEMBLE

	G2/G3 *	EDF1	EDF2	EDF3 **
Puissance électrique nette (MW)	37	68	198	480
Puissance thermique (MW)	250	300	800	1560
Rendement net (%)	15	23	25	31

* Les caractéristiques données pour G2/G3 sont les caractéristiques actuelles, sensiblement supérieures à celles du projet.

** Les caractéristiques du cœur EDF4 sont identiques à celles d'EDF3.

TABLEAU III
CARACTÉRISTIQUES DU CŒUR

	G2/G3	EDF1	EDF2	EDF3
Tonnage d'uranium (t)	135	140	240	410
Volume du cœur (m^3)	410	490	840	1 380
Nombre de canaux	1 200	1 148	1 980	2 700
Nombre d'éléments combustibles	33 600	17 200	23 700	41 000
Puissance spécifique maximum (MW/t)	3,5	4,4	5,8	6,2
Puissance volumique moyenne (MW/ m^3)	0,6	0,6	1,0	1,1
Puissance électrique spécifique moyenne (MWe/t)	0,3	0,5	0,8	1,2
Puissance canal central (kW)	260	380	500	660

TABLEAU IV
CARACTÉRISTIQUES DU COMBUSTIBLE

	G2/G3	EDF1	EDF2	EDF3
Nature du combustible	U métal traité	U métal 0,5% Mo	U métal 1% Mo	U métal 1% Mo
Barreau	plein	tube	tube	tube
Diamètre (mm)	31	14 × 35	18 × 40	23 × 43
Longueur d'un élément (cm)	30	60	60	60
Orientation des canaux	horiz.	vert.	vert.	vert.
Supportage	couché	empilé	chemise graphite	chemise graphite
Ailettes	longit.	longit.	chevrons	chevrons
Température max. gaine (°C)	455	410	425	445
Température max. uranium (°C)	590	520	570	590

TABLEAU V
CARACTÉRISTIQUES DU CIRCUIT CO₂

	G2/G3	EDF1	EDF2	EDF3
Pression (bar)	15	25	25	25
Caisson	béton précont.	acier	acier	béton précont.
Dimensions (m)	14-15,7	10-23	18,3	19-21
Température entrée CO ₂ (°C)	140	142	198	240
Température sortie CO ₂ (°C)	340	345	365	410
Soufflantes {	nombre	2	1	4
	puissance (MW)	3,7	9,2	6
	entraînement	vapeur condens.	électr.	électr.
Echangeurs de chaleur	4 tours	120 el.	96 el.	vapeur contre-press. 192 el.

TABLEAU VI
CARACTÉRISTIQUES DES CIRCUITS VAPEUR

	G2/G3	EDF1	EDF2	EDF3
Pression vapeur HP (bar)	10,3	21,3	32,6	52,5
Débit (kg/s)	47,8	79,7	2 × 95,5	2 × 287
Température (°C)	334	342	340	400
Pression vapeur BP (bar)	2,2	4,3	8,7	1 seul étage de pression
Débit (kg/s)	6,6	29,6	2 × 52,2	
Température (°C)	171	215	340	

TABLEAU VII

CARACTÉRISTIQUES DES GROUPES TURBO-ALTERNATEURS

	G2/G3	EDF1	EDF2	EDF3
Nombre	1	1	2	2
Puissance nominale (MW)	40	82	115/125	250
Tension aux bornes (kV)	10,5	14,5	15,5	20
Vitesse (t/min)	3000	3000	3000	3000

R É F É R E N C E S

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DISCUSSION

P. BALLIGAND: I should like to ask Mr. Tanguy a question concerning fuel-element tests carried out in the first prototype power reactors. I know that a research reactor specially designed for fuel-element irradiation tests has just been completed in France. What is the difference between tests carried out in the power reactors themselves and those that require a special-purpose reactor?

P. TANGUY (on behalf of J. Horowitz and J.P. Roux): The Commissariat à l'Energie Atomique has built a fuel-element testing reactor called PÉGASE which reached criticality at Cadarache two or three months ago and rose to its rated power of 30 MW last week. In independent loops surrounding the core it will be possible to test simultaneously eight gas-cooled fuel elements. These tests are designed to reveal the limits of performance

for fuel elements for stations planned or under construction, by subjecting them to conditions of temperature and power considerably in excess of what they will actually undergo in the power reactors.

The tests conducted in the power reactors themselves involve large numbers of fuel elements, but on the other hand the safety requirements imposed on these reactors prevent the elements from being subjected to the very stringent trials that can be carried out in PÉGASE.

J. TERPSTRA: Could Mr. Tanguy or someone from Electricité de France describe the thermal insulation of the EDF3 pre-stressed concrete vessel?

G. LAMIRAL: The thermal insulation will consist of an assembly of special concrete bricks with a pumice-stone base; this will be lined on the inside with an expandable leak-proof metal casing.

M. R. SRINIVASAN: What size of plant does the cost figure of F 1000 refer to, and when will such a station be commissioned?

P. TANGUY: The reference was to a 500 MW(e) station of the EDF4 type, which could go into service about 1968 or 1969.

A. K. HANNERZ: France is building CO₂-cooled power reactors of two types - graphite-moderated plants and the EL4 type moderated by heavy water. Could you compare the economic potentialities of these two types?

P. TANGUY: EL4, which is a prototype reactor of 70 MW(e), is due to reach criticality in 1966. The first loading will consist of slightly enriched uranium oxide clad in stainless steel. However, the intention is that the heavy-water reactors now being studied in France, of which EL4 is the first, should use natural uranium and it is therefore essential to perfect a low-absorption cladding. Studies are at present under way on beryllium and other possible clads, such as iron-aluminium alloys and zirconium alloys. These few remarks should suffice to show that the two types of reactor are far from being in the same stage of development. It is impossible, at present, to make a really useful economic comparison, although we do feel that heavy-water reactors have great economic possibilities.

OPERATING EXPERIENCE IN NUCLEAR POWER-PLANTS WITH BOILING-WATER REACTORS

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Abstract — Résumé — Аннотация — Resumen

OPERATING EXPERIENCE IN NUCLEAR POWER-PLANTS WITH BOILING-WATER REACTORS. A significant amount of operating experience has now been accumulated by boiling-water-reactor power plants.

By the end of 1962, over 2200 million kWh of electricity have been generated by three plants operating on utility system - Dresden Nuclear Power Station, Commonwealth Edison Company, Morris, Illinois; Vallecitos Atomic Power Plant, Pacific Gas and Electric Company and General Electric Company, Pleasanton, California; and Kahl Nuclear Power Station, Rheinisch Westfälisches Elektrizitätswerk and Bayernwerk, Kahl-am-Main, West Germany.

Boiling-water-reactor power-plant performance, under routine electric-utility operating conditions, has been excellent. Reactor and plant availability and capacity factor provide a sound basis for anticipation of continuing reliable performance from boiling-water-reactor power stations.

During 1963, four additional boiling-water-reactor plants will begin power operation - Big Rock Point Nuclear Plant, Consumers Power Company, Charlevoix, Michigan; Humboldt Bay Plant Nuclear Unit, Pacific Gas and Electric Company, Eureka, California; Garigliano Nuclear Power Station, Società Elettronucleare Nazionale, Scauri, Italy; and Japan Power Demonstration Reactor, Japan Atomic Energy Research Institute, Tokai Mura, Japan. The start-up and initial operation of these plants confirms the expectation of reliable performance established by Dresden, Kahl, and Vallecitos.

Performance records of Dresden, Kahl and Vallecitos have clearly proved the stability and safety of boiling-water reactors. Additionally, radiation levels within the plant and in the environs have been significantly below limits established by operating licences.

Simplicity and ease of operation of boiling-water reactors has been confirmed. Load following characteristics of the Dresden dual-cycle boiling-water reactor have been excellent.

Major and minor maintenance and repair work can be accomplished by ordinary craft unions, and without undue hardship or time limits caused by radioactivity exposure considerations. Recent full-scale inspection and overhaul of the Dresden turbine provided no maintenance problems, after over 12 000 h of operation on direct-cycle steam and after operation with known failed fuel elements in the reactor.

EXPÉRIENCE ACQUISE DANS L'EXPLOITATION DES RÉACTEURS A EAU BOUILLANTE. On a maintenant acquis une expérience appréciable dans l'exploitation des centrales équipées de réacteurs à eau bouillante.

Vers la fin de 1962, on avait produit plus de $2,2 \cdot 10^9$ kWh dans trois centrales nucléaires rattachées à des réseaux de distribution: la centrale de Dresden (Commonwealth Edison Company, Morris, Illinois), la centrale de Vallecitos (Pacific Gas and Electric Company and General Electric Company, Pleasanton, Californie) et la centrale de Kahl (Rheinisch-Westfälisches Elektrizitätswerk et Bayernwerk, à Kahl-sur-le-Main, République fédérale d'Allemagne).

Le rendement de ces réacteurs à eau bouillante, exploités dans les conditions normales de production d'électricité, est excellent. On peut donc s'attendre que les centrales à eau bouillante continueront d'être sûres, étant donné le facteur de disponibilité et le facteur de puissance des réacteurs et des installations de ce type.

Au cours de 1963, quatre nouvelles centrales équipées de réacteurs à eau bouillante entreront en service: la centrale de Big Rock Point (Consumers Power Company, Charlevoix, Michigan), la centrale de Humboldt Bay (Pacific Gas and Electric Company, Eureka, Californie), la centrale de Garigliano (Società Elettronucleare Nazionale, Scauri, Italie) et la centrale de démonstration japonaise (Institut de recherches nucléaires du Japon, Tokai Mura, Japon).

Les résultats obtenus lors du démarrage et pendant le fonctionnement initial de ces installations confirment les espoirs suscités par les centrales de Dresden, Kahl et Vallecitos.

Les journaux de marche des centrales de Dresden, Kahl et Vallecitos mettent en évidence la stabilité et la sécurité des réacteurs à eau bouillante. De plus, les niveaux de rayonnements dans la centrale et aux alentours restent nettement au-dessous des limites fixées par les permis d'exploitation.

La simplicité et la facilité d'exploitation des réacteurs à eau bouillante se sont confirmées. La rapidité de réponse du réacteur de Dresden aux variations de la demande est excellente.

Aucun des travaux d'entretien et de réparation, quelle que soit leur importance, n'exige de qualification spéciale ni ne comporte de risques de radioexposition excessifs. L'inspection générale et la révision de la turbine de la centrale de Dresden n'ont révélé aucun problème d'entretien après 12 000 h de fonctionnement avec la vapeur en cycle direct. Or, on avait continué l'exploitation après avoir constaté que des cartouches de combustible étaient détériorées.

ОПЫТ ЭКСПЛУАТАЦИИ КИПЯЩАГО РЕАКТОРА. К настоящему времени приобретен значительный опыт в эксплуатации электростанций с кипящим реактором.

К концу 1962 года произведено свыше 2200 млн. квт.ч электроэнергии на трех электростанциях энергосистемы коммунального обслуживания: на Дрезденской атомной электростанции, "Коммонвелс Эдисон компани", Моррис, Иллинойс, на атомной электростанции Валлеситос, "Пасифик газ энд электрик компани", Плезантон, Калифорния, и на Кальской атомной электростанции, "Рейниш вестфалишес электрицитетсверк унд байернверк", Каль-на-Майне, Западная Германия.

Рабочая характеристика кипящего реактора атомной электростанции при обычном режиме работы в коммунальной энергосистеме очень хорошая. Коэффициент использования и мощности реактора и электростанции дает твердое основание полагать, что электростанции с кипящими реакторами являются надежными с точки зрения их рабочей характеристики.

В течение 1963 года будут введены в строй четыре дополнительные электростанции с кипящими реакторами: атомная электростанция в Биг Рок Пойнт, "Консьюмерс пауэр компани", Шарльвуа, Мичиган, атомная энергетическая установка в Хамболдт Бей, "Пасифик газ энд электрик компани", Юрика, Калифорния, атомная электростанция в Гаригьяно, Национальное общество по атомной энергии, Скаури, Италия, и Японский демонстрационный энергетический реактор, Японский научно-исследовательский институт по атомной энергии, Токаи-Мура, Япония. Пуск и первоначальная эксплуатация этих электростанций подтверждают предположение о надежности их работы, что уже продемонстрировано атомными электростанциями в Дрездене, Кале и Валлеситосе.

Рабочая характеристика атомных электростанций в Дрездене, Кале и Валлеситосе является наглядным доказательством стабильности и безопасности кипящих реакторов. Кроме того, уровни радиации на самой электростанции и в окружающей среде значительно ниже пределов, установленных лицензиями на эксплуатацию.

Подтвердились простота и легкость эксплуатации кипящих реакторов. Характеристика контроля за нагрузкой у кипящего реактора с двойным циклом Дрезденской электростанции оказалась очень хорошей.

Крупные и небольшие работы по уходу и ремонту могут осуществляться обычными ремонтными группами без вредных последствий или без лимита времени, связанными с соображениями радиоактивного облучения. В результате проведенных недавно полной инспекции и капитального ремонта турбины Дрезденской электростанции не обнаружено никаких ремонтных проблем как после более 12 000-часовой эксплуатации с прямым паровым циклом, так и после эксплуатации с заведомо неисправными топливными элементами в реакторе.

EXPERIENCIA ADQUIRIDA CON LA EXPLOTACIÓN DE REACTORES DE AGUA HIRVIENTE. El autor señala que la experiencia adquirida con la explotación de centrales nucleoeleónicas que utilizan reactores de agua hirviente es ya bastante considerable.

En efecto, al finalizar el año 1962 se habían generado más de $2,2 \cdot 10^9$ kWh en tres centrales nucleoeleétricas integradas en redes de distribución: la central nucleoeleétrica de Dresden (Commonwealth Edison Company, Morris, Illinois), la central nucleoeleétrica de Vallecitos (Pacific Gas and Electric Company, Pleasanton, California) y la central nucleoeleétrica de Kahl (Reinisch-Westfälisches Elektrizitätswerk y Bayernwerk A. G., Kahl-am-Main, República Federal de Alemania).

El rendimiento de estas centrales explotadas en condiciones normales de producción de electricidad es excelente. Los factores de disponibilidad y de capacidad de los reactores y de las plantas constituyen una base firme para prever que las centrales nucleoeleétricas dotadas de reactores de agua hirviente funcionarán a régimen continuo en condiciones de seguridad.

En 1963 entrarán en servicio cuatro nuevas centrales equipadas con reactores de agua hirviendo: la central nucleoelectrica de Big Rock Point (Consumers Power Company, Charlevoix, Michigan), la central nucleoelectrica de Humboldt Bay (Pacific Gas and Electric Company, Eureka, California), la central nucleoelectrica de Garigliano (Società Elettro-nucleare Nazionale, Scauri, Italia), y el reactor de potencia japonés para fines de demostración (Instituto de Investigaciones Nucleares, Tokai Mura, Japón). La puesta en marcha y el funcionamiento inicial de estas centrales confirma las esperanzas despertadas por las centrales de Dresden, Kahl y Vallecitos.

Los registros de rendimiento de las centrales de Dresden, Kahl y Vallecitos han puesto claramente de manifiesto la estabilidad y seguridad de los reactores de agua hirviendo. Además, los niveles de radiación dentro de la central y en sus alrededores se han mantenido muy por debajo de los límites fijados en los permisos de explotación.

Ha quedado confirmada la sencillez y facilidad de explotación de los reactores de agua hirviendo. La rapidez de la respuesta del reactor con doble ciclo de Dresden a las fluctuaciones de la demanda es excelente.

Ninguno de los trabajos de conservación y reparación, sea cual fuere su importancia, exige calificaciones especiales ni entraña riesgos de irradiación excesivos. La inspección general y la revisión de la turbina de la central de Dresden puso de manifiesto la inexistencia de problemas de conservación después de más de 12 000 h de funcionamiento con el vapor en ciclo directo, a pesar de que la explotación prosiguió después de haberse comprobado algunas fallas en los elementos combustibles.

1. INTRODUCTION

Over the past few years, sufficient operating experience has been accumulated with boiling-water-reactor power plants to support a number of significant and highly favourable conclusions regarding this reactor type as applied in central stations in utility service. Perhaps of equal significance, this extensive operating experience also has furnished a firm base for complete elimination of areas of concern and of some unknowns that existed before the operation of these plants.

Through the end of 1962, over two thousand, two hundred million gross kilowatt-hours were generated by three boiling-water-reactor plants operating on utility systems - Dresden Nuclear Power Station, owned and operated by the Commonwealth Edison Company at Morris, Illinois, Kahl Nuclear Power Station, owned and operated by Rheinisch Westfälisches Elektrizitätswerk and Bayernwerk, at Kahl-am-Main, West Germany, and Vallecitos Atomic Power Plant. The nuclear portion of this plant is primarily devoted to testing of advanced fuel and is owned and operated by General Electric. Pacific Gas and Electric Company owns and operates the remainder of the plant. Earlier this year, two more boiling-water-reactor plants were placed in service: the Big Rock Point Nuclear Plant, owned and operated by the Consumers Power Company at Charlevoix, Michigan, and the Humboldt Bay Power Plant Nuclear Unit, owned and operated by the Pacific Gas and Electric Company, located at Eureka, California.

The over-all performance of the boiling-water reactors in these nuclear power-plants under standard electrical-utility operating conditions has been uniformly excellent. Their safety, reliability, ease of maintenance by contact methods and high degree of availability have been clearly demonstrated. To fully appreciate the meaning and significance of this general statement, it is useful to consider performance of these reactors in more detail.

2. RELIABILITY OF BOILING-WATER REACTORS IN NUCLEAR POWER-PLANT SERVICE

Among the many important criteria for evaluating the performance of a particular reactor type when applied to electrical generation service, one of the most significant is reliability - the capability to operate over long periods of time with a minimum outage time requirement. Operating statistics from the Dresden, Kahl, Big Rock Point, and Humboldt Bay plants clearly illustrate the high reliability of boiling-water reactors. Operating experience at Dresden, for example, shows the boiling-water-reactor plant has a reliability as good as the best coal-fired plant on the Commonwealth Edison system. There are three principal reasons for this: the relative simplicity of the boiling-water-reactor steam supply system, the inherently stable operating characteristics of the boiling-water reactor itself, and the dual safety circuitry with its high-quality components and its demonstrated capability to effectively eliminate spurious scrams.

Another crucially important facet of "reliability" is the demonstrated capability of the reactor safety devices to operate without fail to shut down the reactor whenever such a shut-down is required. There has never been an instance of failure of a reactor safety device in any of these boiling-water reactors. Specifically, there never has been an instance of failure of a control rod to insert on scram signal and reduce reactivity at the specified rate. This is a unique advantage of General Electric-designed power reactors which few others can match.

In the early operation of Dresden, some mechanical and material problems with the locking-piston control-rod drives developed. These problems were readily corrected and valuable information regarding various alloys was disseminated broadly to the benefit of most other reactor manufacturers and operators. Control-rod drives of similar design are in service at the Big Rock Point and Humboldt Bay plants. They have had their share of minor difficulties during the debugging and start-up phases, but never have they failed to perform their safety-oriented function; never have they failed to scram.

3. CONTACT MAINTENANCE AND RADIATION EXPOSURE

The significant capital and construction cost advantages of the direct-cycle boiling-water reactors and to a lesser extent the dual-cycle system, are obvious and generally accepted. Perhaps less well known is the fact that experience at Vallecitos and Dresden clearly demonstrates that contact maintenance has been routinely carried out on turbines and all the other equipment in such systems without significant radiation exposure to personnel. This is particularly significant in the Dresden experience, where turbine overhaul was accomplished in approximately the same time that would have been required on a conventional turbine and without significant radiation exposure to personnel, despite the fact that over many months of operation before the overhaul the reactor was operating with several fuel-cladding failures, and the steam passing through the turbine contained fission

products. These, of course, were subsequently removed either through the demineralizers or the non-condensable gas-removal system.

Another important and interesting outcome of the overhaul of the Dresden turbine, with its detailed inspections, was the verification that there was essentially no corrosion or erosion of blades and diaphragms. Based on these recent findings, Commonwealth Edison Company has elected to place the Dresden turbine-generator on the same preventive maintenance schedule that is used for their conventional units. In addition, the General Electric Turbine Division is now designing and is ready to build turbine-generators for these systems in sizes of at least 600 MW.

Further, Dresden experience has indicated that large boiling-water reactors can operate at full power and with a significant number of leaking fuel elements without any significant radiological effects on their environs. For example, the Dresden stack emission of radiogases for the whole of 1962 averaged only about 1/300th of the permissible emission rate established in the USAEC licence. The calculated maximum off-plant dose for all of 1962 was about 0.5 mrem, compared to the natural background radiation present of about 150 mrem/yr.

The Dresden air release was indistinguishable in the ion chamber and air filter measurements at the 18 environmental monitoring stations, which are situated up to 15 miles from the plant. Milk samples from three farms and vegetation samples showed radioactivity consistent with the National Surveillance Network samples, indicating no detectable contributions attributable to the nuclear-power station.

Similarly, the Dresden liquid-waste experience was most favourable. Samples taken upstream and downstream of the plant of water, silt, plankton and algal slime indicated no increase in radioactivity due to plant operation. Of course, most of the plant water is re-used.

4. POTENTIAL FOR UPRATING

Another significant advantage arising from boiling-water-reactor experience is the high potential for uprating at relatively low cost. Illustrative examples of this point are the following:

- (1) With the addition of recirculating pumps and change in baffling within the reactor vessel, the thermal output of the Vallecitos Boiling Water Reactor was increased from about 25 MW(t) to about 40 MW(t).
- (2) By making minor changes in the Dresden turbine, output was increased from 626 MW(t) and 180 MW(e) to 700 MW(t) and 215 MW(e). The reactor system has been demonstrated capable of operation to at least 270 MW(e).
- (3) By adding more fuel and conducting additional thermal performance tests on the core, it is expected that the Big Rock Point Plant can be uprated from 50 MW(e) to about 75 MW(e) at 1500 lb/in².
- (4) By conducting additional tests to evaluate steam carry-under, it is expected that Humboldt Bay Nuclear Unit can be uprated from its already demonstrated 53 MW(e) to at least 60 MW(e) and probably 70 MW(e).

5. EASE OF OPERATION AND MAINTENANCE OF BOILING-WATER REACTORS

Experience has shown that ease of operation and maintenance is an important characteristic of the boiling-water reactor. Several illustrative examples follow.

Early experience in the operation of some large reactors indicated that flux cycling due to xenon transients is a significant operating problem, imposing a heavy burden on the operator and sometimes increasing the operating manpower requirements. In contrast, operations at Dresden demonstrate that with the large boiling-water reactor the flux distribution remains essentially constant once the basic control-rod patterns are established. In practice, the movement of a small increment of one control rod once each day is all that is required. As an aid to establishing appropriate control-rod patterns, the in-core ion chambers have proved to be an extremely useful operating tool. These miniature ion chambers are interspersed through the core and provide an accurate measure of radial and axial flux profiles within the core on a continuous basis. They also provide an important mechanism for achieving the lowest possible fuel-cycle costs.

The ease of operation of the boiling-water reactor also has been demonstrated in "abnormal" operating conditions. A pertinent illustration has been recorded in Dresden operating experience. A tornado passed near the Station, tore the roofing on the turbine building and intake structure and knocked out a transformer which fed two of the four reactor recirculating pumps. Reactor power decreased automatically to half of rated power and the turbine-generator did likewise. Upon restoring auxiliary power to the pumps, the operator placed them in service. In turn, the reactor and turbine-generator recovered to full power. The only controls manipulated during this entire process were those required to return the pumps to service. This experience also demonstrated the excellent load-following characteristics of a dual-cycle boiling-water-reactor system.

The Big Rock Point and Humboldt Bay plants provide more examples of the ease with which a boiling-water reactor can be operated. In each case, the controls were designed for one-man operation, and actual practice has proven the validity of the design. Further, the basic-shift operating crew consists of only three operators and one technician.

A significant amount of maintenance experience on BWR's has been accumulated during the past few years. Also important are the improvements that are continually being made to simplify maintenance procedures further. Typical examples of these are:

- (1) Control drives can now be removed from the bottom of the reactor vessel of our most recently designed plants such as Bodega, KRB and Tarapur, without removing the vessel head. Thus, the time required to remove a drive for maintenance and replace it with a spare has been reduced to about four hours (not including cool-down time).
- (2) Because of recent improvements in hydraulic stud tensioners, reactor-vessel heads designed to take advantage of this bolting method can be installed in 20 h.

(3) In-core ion-chamber assemblies are now fabricated so that their signal cables penetrate the vessel at the bottom. This means that these irradiated assemblies do not have to be removed when the reactor-vessel head is removed.

6. CONCLUSION

In conclusion, operating experience with boiling-water reactors in nuclear power-plants clearly demonstrates that they are highly reliable, have a remarkable availability record, an impeccable record of safety performance and a significant history demonstrating ease of operation and maintenance. Their adaptability to continuing upgrading and new technology such as superheat is unmatched. On the over-all picture, the boiling-water reactor has no peer in today's nuclear power-plant application.

DISCUSSION

U. ZELBSTEIN: Were the only fuel-element failures those which affected the experimental elements? I should also like to know whether any experiments have been carried out on plutonium-enriched elements; if so, whether these have raised any special problems; and what the consequences of such problems were from the operational point of view.

P. EDDY (on behalf of R.J. Ascherl): The failures involved both experimental and developmental elements. The zirconium-clad UO_2 production fuel elements gave only 0.007% failures. The developmental elements that failed were of stainless-steel clad UO_2 . None of these failures resulted in operating problems or hazards, and the release of radioactive material to the environment was about 0.02% of the limit provided for in the licence.

No plutonium elements were loaded in the DRESDEN reactor and I have no knowledge personally of plutonium fuel. There have been some failures in the Plutonium Recycle Test Reactor (PRTR) at Hanford, I believe, and you may wish to discuss them with the USAEC.

C. A. PURSEL: In fact, the only failure of a plutonium-bearing fuel assembly that I know of was the one which occurred in the PRTR some time during the last few months, and all I know about that is that it took some time to clean up the system.

G. B. SCURICINI: Will the system for removing control-rod drives at KRB, Bodega Bay and Tarapura be the same as in the SENN reactor?

P. EDDY: I am not sure how the control-rod drives are removed in the SENN reactor, although I understand that the system represents a major advance over the DRESDEN reactor. At SENN the drives can be removed for inspection, maintenance and so on in a fraction of the time required at DRESDEN.

B. SAITCEVSKY: I should like to know two figures for DRESDEN: first, the availability, and secondly, the burn-up obtained with the first loading.

P. EDDY: I can best answer the first part of your question by quoting a statement of the Commonwealth Edison Company. "For the year 1962. DRESDEN's gross electrical output was 1250 million kWh with a capacity

factor of 73% and availability of 79.4%. For the eleven-month period prior to refuelling the capacity factor was 85.6% and availability was 93.1%. The rather high capacity factor reflects our desire to obtain high fuel burn-up for fuel development research, to gain experience in continuous high-level power operation of the reactor and to take advantage of AEC high plutonium buy-back price. The six largest coal-fired units (rated at 315 to 340 MW net) on our system had an average capacity factor of 67% and an average availability of 82.4% for 1962. However, two of these units were put into service during the year and were having the usual problems attendant with initial operation. This adversely affected the capacity factor and availability of the group. The coal-fired unit with the best performance of 1962 had a capacity factor of 75.5% and availability of 87.9%. These figures indicate that Dresden Station has demonstrated reliable operating performance and in this respect can hold its own with coal-fired units."

The average burn-up for the 40% of fuel discharged was 6100 MWD/t.

A.K. HANNERZ: I have heard that after the last fuel charge in DRESDEN there was considerable delay in going to full power because of crud in the inlet water and ion-exchange beds. Could you suggest how this might be avoided in future?

P. EDDY: There was some delay in performing tests of full-power demonstration (which is greater than the current rated turbine power) because of a pressure drop in the condensate demineralizers. The plant operator wanted to continue using the resin beds, which became quite dirty during the initial operating period, after refuelling. This limitation could have been eliminated at any time by putting in fresh resin. There were no abnormal amounts of crud in the system.

In any event the rise to full power after the refuelling outage was planned to take several weeks. It is customary to hold half-power for a while so as to calibrate in-core instrumentation. In addition, a comprehensive series of physics tests necessitated a progressive rise to full power.

K. EFFAT: What has been your experience with radioactive steam in the turbine and other parts of the steam circuit? How does it affect the operation of the plant and maintenance of the various components of the system?

P. EDDY: There has been no difficulty due to radioactive steam either in the turbine or in other parts of the steam system. The turbine was overhauled by the regular turbine maintenance crew working a 56-h week, and the average exposure per shift was of the order of 2-3 mrem. The turbine had of course become contaminated with radioactive corrosion products and special precautions, such as covering the parts with plastic sheets, were taken to prevent the spread of contamination. Craftsmen engaged in welding or grinding wore respiratory masks. However, these precautions did not add to the time or cost of the job.

The DRESDEN turbine is provided with a special decontamination system. This system was not used, however, and it is believed now that it will not be required.

A. SCHNEIDER: I have two questions. First, do the boiling-water reactors have a system for continuous detection of failed fuel elements or is the monitoring done exclusively by radiochemical analyses? Secondly,

has radiolysis of the water (steam) been a serious problem, and if so, has it been remedied satisfactorily by oxygen-hydrogen recombiners?

P. EDDY: The main method of detecting fuel-element failure is continuous monitoring of the off-gas to the stack. Any increase in fission gases can be detected very rapidly, and the off-gas line is closed when the content of fission gases rises to a certain level (well below the limit provided for in the licence). The primary water is routinely sampled and analysed for radioactive isotopes of iodine. By manipulation of the control blades the flux can be "titled" to establish the general location of a failure.

It is in fact known that the DRESDEN reactor operated with a small number of failed fuel elements for some time before shut-down.

So far as I know, radiolysis of water or steam has never presented a problem; but I might ask Dr. Weckesser to comment on this part of your question.

A. WECKESSER: The activity of the gases released is also indicated by a check on the water itself. Normally the activity in the off-gas is fairly low at the Kahl plant.

P. ERKES: What programme is being conducted with the 40% of fuel unloaded from the DRESDEN core, with a view to comparing theory and operational results on plutonium production?

P. EDDY: Some work of analysis and evaluation has been done, and a programme similar to the one sponsored by the United States Atomic Energy Commission for evaluation of the YANKEE fuel is being considered. It will apply not only to DRESDEN but to all boiling-water reactors. Unfortunately I do not know how far advanced it is.

REVIEW OF DEVELOPMENT STATUS OF NUCLEAR SUPERHEAT

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Abstract — Résumé — Аннотация — Resumen

REVIEW OF THE DEVELOPMENT STATUS OF NUCLEAR SUPERHEAT. The General Electric Company has been actively engaged in development work on nuclear superheat from light-water-moderated reactors since 1959, at which time the Company-financed Superheat Advance Demonstration Experiment (SADE) produced the first nuclear superheated steam in the United States. The current status of nuclear superheat is divided into two major categories. The first is a description of the three major superheat fuel irradiation facilities used by the General Electric Company, and the second is a description of the two major development programme activities with an up-to-date review of the significant superheat development results.

1. Major development facilities

(a) A brief description is given of the Superheat Advance Demonstration Experiment (SADE) utilized in the Vallecitos Boiling Water Reactor (VBWR), with tables of operating conditions, fuel elements irradiated during the period between May 1959, and June 1962, and a discussion of significant experimental results.

(b) A brief description is given of the Expanded Superheat Advance Demonstration Experiment (E-SADE) in operation in the Vallecitos Boiling Water Reactor, with tables of operating conditions, fuel elements irradiated in the E-SADE facility and a discussion of the significant development results.

(c) A brief description is given of the Empire States Atomic Development Associates-Vallecitos Experimental Superheat Reactor (ESADA-VESR), list of expected operated conditions including design conditions of the initial superheat core loading, and a report on the current status of construction.

2. Major superheat development programme activities

(a) A brief description is given of the United States Atomic Energy Commission (USAEC) sponsored Nuclear Superheat Project which has been in progress at the San José site of the General Electric Company since July 1959. A brief description of the individual tasks is given with tables giving significant development results in the areas of superheat fuel irradiation performance, in-pile and out-of-pile uniform and localized corrosion evaluations, results from thermal superheat critical experiments, results from experimental heat-transfer testing and a brief appraisal of the economic incentives of the separate superheat reactor, integral superheat reactor, and mixed-spectrum superheat reactor design studies.

(b) A brief description is given of the USAEC-Sponsored ESADA-VESR nuclear superheat fuel development programme. The development tasks, the initial core superheat fuel-element design, the range of experimental variables, and the expected results for the three-year fuel development programme are discussed.

EXPOSÉ SUR L'ÉTAT ACTUEL DES TRAVAUX CONCERNANT LA SURCHAUFFE NUCLÉAIRE. Depuis 1959, la Société General Electric s'occupe activement de mettre au point la surchauffe nucléaire dans les réacteurs ralentis à l'eau ordinaire. A cette époque, le «Superheat Advance Demonstration Experiment» (SADE), financé par la société, a permis d'obtenir, pour la première fois aux Etats-Unis, de la vapeur surchauffée par un dispositif nucléaire. Les auteurs du mémoire font le point de la situation actuelle concernant la surchauffe nucléaire. Ils décrivent, dans une première partie, les trois principales installations de surchauffe nucléaire utilisées par la General Electric et, dans la seconde, les deux principaux programmes de recherches sur la surchauffe ainsi que les résultats les plus importants obtenus jusqu'ici dans ce domaine.

1. Principales installations pour les expériences de surchauffe

(a) Brève description du SADE utilisé dans le réacteur à eau bouillante de Vallecitos (VBWR), avec tableaux des conditions dans lesquelles se déroulent les expériences, et des éléments combustibles irradiés pendant la période comprise entre mai 1959 et juin 1962; puis examen critique des résultats les plus importants ainsi obtenus.

(b) Brève description de l'«Expanded Superheat Advance Demonstration Experiment» (E-SADE) installé dans le réacteur de Vallecitos, avec tableaux des conditions dans lesquelles se déroulent les expériences, et des éléments combustibles irradiés dans l'installation; puis, examen critique des résultats les plus importants ainsi obtenus.

c) Brève description de Empire States Atomic Development Associates - Vallecitos Experimental Superheat Reactor (ESADA-VESR), liste des conditions dans lesquelles on compte effectuer les expériences, notamment conception de la charge initiale du cœur du surchauffeur; puis, description de l'état actuel de la construction du dispositif.

2. Principaux programmes de recherches sur la surchauffe

a) Brève description du projet de surchauffe nucléaire patronné par la Commission de l'énergie atomique et en cours de réalisation, depuis juillet 1959, au centre de San José, créé par la General Electric. Aperçu des diverses tâches, avec tableaux des principaux résultats obtenus en ce qui concerne le rendement du combustible nucléaire utilisé pour la surchauffe, évaluations de la corrosion uniforme et localisée observée dans la pile et hors de la pile, et tableaux des résultats obtenus lors d'expériences critiques de surchauffe thermique et lors des essais expérimentaux de transfert de chaleur; enfin, bref examen des avantages économiques que présentent, d'après les études, les réacteurs à surchauffe séparée, les réacteurs à surchauffe intégrée et les réacteurs à surchauffe à spectre mixte.

b) Brève description du programme de mise au point des combustibles pour la surchauffe nucléaire, exécuté sous l'égide de la Commission de l'énergie atomique par ESADA-VESR. Examen des travaux de recherche, de l'étude des éléments combustibles du cœur initial, de la gamme des variables expérimentales et des résultats que l'on compte obtenir de l'exécution de ce programme de trois ans.

ОБЗОР РАЗРАБОТКИ ВОПРОСА О ЯДЕРНЫХ ПЕРЕГРЕВАТЕЛЯХ. Начиная с 1959 года компания "Дженерал электрик" активно занималась проведением работы в области ядерного перегрева на реакторах с замедлителем из обычной воды. За этот период в США впервые был произведен ядерный перегретый пар в ходе проведения усовершенствованного демонстрационного эксперимента с перегревом на установке SADE. Этот проект финансировался компанией. Нынешнее состояние с ядерным перегревом подразделяется на две главные категории. Первая является описанием трех основных установок по облучению перегретого топлива, используемых компанией "Дженерал электрик", и вторая - описанием двух главных разработанных программ деятельности, вместе с обзором по состоянию на сегодняшний день значительных результатов развития в области перегрева.

1. Главные усовершенствованные установки:

a) Дается краткое описание эксперимента (SADE) на Валлеситосском реакторе с кипящей водой (VEWR), приводятся таблицы эксплуатационных условий, данные о топливных элементах, облученных в период между маем 1959 года и июнем 1962 года, и итоги обсуждения значительных экспериментальных результатов.

b) Дается краткое описание расширенного усовершенствованного демонстрационного эксперимента с перегревом (E-SADE) при работе Валлеситосского реактора с кипящей водой, приводятся таблицы эксплуатационных условий, данные о топливных элементах, облученных на установке E-SADE, и итоги обсуждения значительных результатов проделанной работы.

в) Дается краткое описание реактора Эмпайр Стейтс Атомик Девелопмент Ассошиейтс-Валлеситосского экспериментального реактора с перегревом (ESAD-VESR), приводится список ожидаемых рабочих условий, включая конструкционные условия первоначальной загрузки активной зоны с перегревом и отчет о текущем состоянии строительных работ.

2. Главные виды деятельности по программе развития реактора с перегревом:

a) Дается краткое описание ядерного проекта с перегревом, осуществляемого при поддержке Комиссии по атомной энергии, который находится в стадии осуществления с июля 1959 года на установке "Дженерал Электрик Компани" в Сан-Хосе. Приводятся краткое описание отдельных задач вместе с таблицами, содержащими значительные результаты проделанной работы в области характеристики облучения топлива при перегреве, оценки однородной и локализованной коррозии внутри и вне реактора, результаты критических экспериментов с тепловым перегревом, результаты экспериментальной передачи тепла и краткая оценка экономических выводов, полученных на отдельном реакторе с перегревом, интегральном реакторе с перегревом и изучения конструкции реактора с перегревом при смешанном спектре.

b) Дается краткое описание программы развития работ с ядерным перегревом топлива, проводимых при поддержке Комиссии по атомной энергии на реакторах ESADA-VESR. Обсуждаются дальнейшие задачи по развитию работ, вопросы первоначальной конструкции топливных элементов с перегревом в активной зоне, размера экспериментальных отклонений и ожидаемых результатов трехлетней программы развития топлива.

ESTUDIO DE LOS PROGRESOS REALIZADOS EN MATERIA DE SOBRECALENTAMIENTO NUCLEAR. Desde 1959, la General Electric Company se ocupa activamente del desarrollo del sobrecalentamiento nuclear en

reactores moderados con agua ligera. En aquel año, el «Superheat Advance Demonstration Experiment» (SADE), financiado por la citada compañía, permitió obtener por vez primera en los Estados Unidos vapor sobrecalentado por medios nucleares. Los autores examinan la situación actual en materia de sobrecalentamiento nuclear. Describen, en la primera parte de la memoria, las tres instalaciones para irradiación de combustible con sobrecalentador empleadas por la General Electric Company, y en la segunda los dos principales programas de investigaciones acerca del sobrecalentamiento, así como los resultados más importantes alcanzados hasta ahora en esa esfera.

1. Principales instalaciones

- a) Breve descripción del SADE empleado en el reactor de agua hirviendo de Vallecitos (VBWR), con cuadros en que figuran las condiciones de funcionamiento, los elementos combustibles irradiados entre mayo de 1959 y junio de 1962, y un examen crítico de los resultados experimentales de mayor importancia.
- b) Breve descripción del «Expanded Superheat Advance Demonstration Experiment» (E-SADE) que funciona en el reactor de agua hirviendo de Vallecitos, con cuadros en que figuran las condiciones de funcionamiento, los elementos combustibles irradiados en la instalación E-SADE, y una discusión de los resultados más importantes.
- c) Breve descripción del «Empire States Atomic Development Associates-Vallecitos Experimental Superheat Reactor» (ESADA-VESR), con una lista de las condiciones de funcionamiento previstas, inclusive el diseño de la primera carga del sobrecalentador, y examen del estado actual de la construcción del dispositivo.

2. Principales programas de investigaciones acerca del sobrecalentamiento nuclear

- a) Breve descripción del proyecto de sobrecalentamiento nuclear patrocinado por la Comisión de Energía Atómica, cuya ejecución comenzó en julio de 1959 en las instalaciones de San José de la General Electric Company. Se resumen los diversos problemas estudiados, con cuadros en que figuran los principales resultados obtenidos en lo que respecta al rendimiento del combustible nuclear utilizado en el sobrecalentador, evaluaciones de la corrosión uniforme y localizada, tanto en el interior como en el exterior del reactor, resultados de experimentos críticos de sobrecalentamiento térmico, y de las comprobaciones experimentales de transmisión de calor; por último, examen sucinto de las ventajas económicas que, según los estudios, presentan los reactores de sobrecalentador separado, los reactores de sobrecalentador integrado y los reactores con sobrecalentamiento de espectro mixto.
- b) Breve descripción del programa ESADA-VESR de desarrollo de combustibles para sobrecalentamiento nuclear, patrocinado por la Comisión de Energía Atómica. Examen de los trabajos de investigación, del diseño de los elementos combustibles de la primera carga del sobrecalentador, de la gama de variables experimentales y de los resultados previstos para el programa trienal de desarrollo del combustible.

1. INTRODUCTION

By the beginning of 1963 more than 3.5 million kilowatts of electrical power will be generated by nuclear power stations around the world. It is expected that the cost of power from the larger more advanced nuclear power plants now under construction will be less than 6.0 mil/kWh [1].

Economic studies [2] have indicated that with the successful development of nuclear superheat, there is a potential cost advantage of 0.3 to 0.5 mil/kWh as compared to, for example, the best boiling-water-reactor power plant to be constructed during the 1968-1970 period. The predicted cost incentive results from about 0.1 to 0.2 mil/kWh fuel-cost reduction and about 0.2 to 0.3 mil/kWh capital-cost reduction. These projected power cost reductions depend upon the ability of the nuclear superheat concept to use modern steam turbines with high thermal efficiencies. Thermodynamic considerations show that a 20% improvement in plant heat rate is a realistic objective when a comparison is made between 1000 lb/in² saturated steam and 1000 lb/in², 1000°F superheated steam at the turbine inlet. This reduced steam rate at the turbine has the effect of significantly reducing the size and cost of conventional power-plant equipment outside the reactor. The higher thermal efficiency provides the potential for reduced fuel costs.

A look at the history of central station power development indicates that the improvement in turbine conditions was based on evolutionary advances over a relatively long period of time. In the case of nuclear power, there may be a need to make a step improvement in technology since in general, nuclear power plants are characterized by higher capital cost and lower fuel cost as compared to fossil-fuelled power generation. There are two significant aspects to the demonstration of an economic incentive for nuclear superheat. The first is that reductions in conventional power costs seem to be keeping pace with improvements in nuclear-power technology. The second is that nuclear superheat should be utilized only if it provides a reduction in the cost of power from nuclear plants utilizing saturated-steam conditions.

At the present time there is a very active programme for the development of nuclear superheat under the direction of the United States Atomic Energy Commission (USAEC) [3]. This includes the construction of the Northern States Power Plant [4], BONUS reactor [5], and BORAX V [6, 7]. In addition to these construction projects, the USAEC [8-14] is sponsoring development work on the superheat concept at the Combustion Engineering Corporation [15], Westinghouse [16] and the General Electric Company [20, 34].

In order to effect power-cost reductions and achieve the full potential of higher thermal efficiencies resulting from nuclear superheat, it is necessary to find solutions to all of the new problems introduced with superheat technology [19]. Recognizing this need, the USAEC initiated the Nuclear Superheat Project at the General Electric Company to establish the experimental basis for nuclear superheat technology. As a result of almost four years of development activity on the Nuclear Superheat Project, there has been a significant amount of information obtained [20, 33] which is of value in providing specific direction and emphasis for the continued development of the nuclear superheat concept and for establishing economic incentives and timing for power producing applications.

It is the purpose of this paper to review the current technical status of nuclear superheat development at the General Electric Company. This review is composed of two parts, consisting of brief descriptions of the three major superheat fuel-irradiation facilities, and secondly, a brief description of the two major superheat development programme activities with up-to-date reviews of results.

2. SUPERHEAT ADVANCE DEMONSTRATION EXPERIMENT (SADE)

The SADE produced the first nuclear superheated steam in the United States on 4 May 1959 [17, 20]. This loop provided an in-pile superheated fuel-element test facility which replaced one of the fuel elements in the active core of the Vallecitos Boiling Water Reactor (VBWR). Saturated steam from VBWR is introduced at the top of the hollow annular fuel element, makes a down-pass between the fuel element outside diameter and the process tube, and an up-pass between the fuel element inside diameter and a velocity booster tube. The superheat steam exits from the VBWR reactor and goes to the independent SADE heat sink. The initial fuel-element configuration was 1 1/2-in outside diam. $\times$ 3/4-in inside diam. $\times$ 36-in long. The

stainless-steel SADE process tube, which is provided to separate the VBWR reactor water from the SADE saturated steam was uninsulated to improve radiant heat transfer in the event of loss of steam coolant [37, 38]. For normal start-up and shut-down, the SADE loop is provided with atmospheric air coolant at 100 lb/in². The nominal design conditions for the SADE loop are 825°F superheated steam exit at a thermal power of 50 kW. An external loop and heat sink are provided to accommodate either air or steam cooling of the superheat fuel (Figs. 1 and 2).

The external loop also provides for flow control, a radiation monitoring and hold-up tank system for off-gas, and other required temperature, pressure instrumentation. The SADE loop was utilized from May 1959 to July 1962 to irradiate eight superheat fuel elements. Table I shows the complete list of fuel elements irradiated in the SADE loop with pertinent design characteristics such as cladding material, length of exposure, heat fluxes and performance evaluation. Three of these fuel elements exhibited a loss of cladding integrity [53]. These failures were attributed to stress-corrosion cracking. The stress-corrosion cracking resulted from the combination of strain cycling and chloride deposits on the steam side of the fuel element.

3. EXPANDED SUPERHEAT ADVANCE DEMONSTRATION EXPERIMENT (E-SADE)

The E-SADE loop installation was completed during the Spring of 1962 with the first superheated steam produced on 18 July 1962 [25, 31, 32]. The replacement of SADE by E-SADE in the VBWR reactor was intended to improve on the two major limitations of the SADE loop as follows:

- (a) To increase the irradiation capacity of the superheat fuel development facility from a single superheat element occupying one 3 in × 3 in VBWR fuel channel, to nine superheat fuel elements replacing four VBWR fuel channels.
- (b) To permit the accumulation of extended fuel burn-up by incorporation of removable connections to permit reinsertion of new fuel elements in bundles previously irradiated or insertion of previously-irradiated fuel elements in new fuel bundles.

The major differences in the E-SADE and SADE systems are increased power level and heat sink capability, utilization of VBWR reactor water for decay heat removal and start-up mode of operation, provision for the introduction of VBWR steam to the E-SADE facility from within the VBWR vessel, permitting the utilization of process tubes that are not designed for full VBWR pressure differential, and in-pile facility mechanical arrangement to incorporate both increased number of superheat fuel elements and the removable fuel feature (Fig. 3).

During the nine-month period between initiation of the E-SADE operation in July 1962 and 1 April 1963, the total integrated exposure of experimental superheat fuel elements was increased threefold [32, 33] over that accumulated in three years of SADE operation. Tables II and III indicate the composition of the first two fuel bundles irradiated in the E-SADE facility. Based

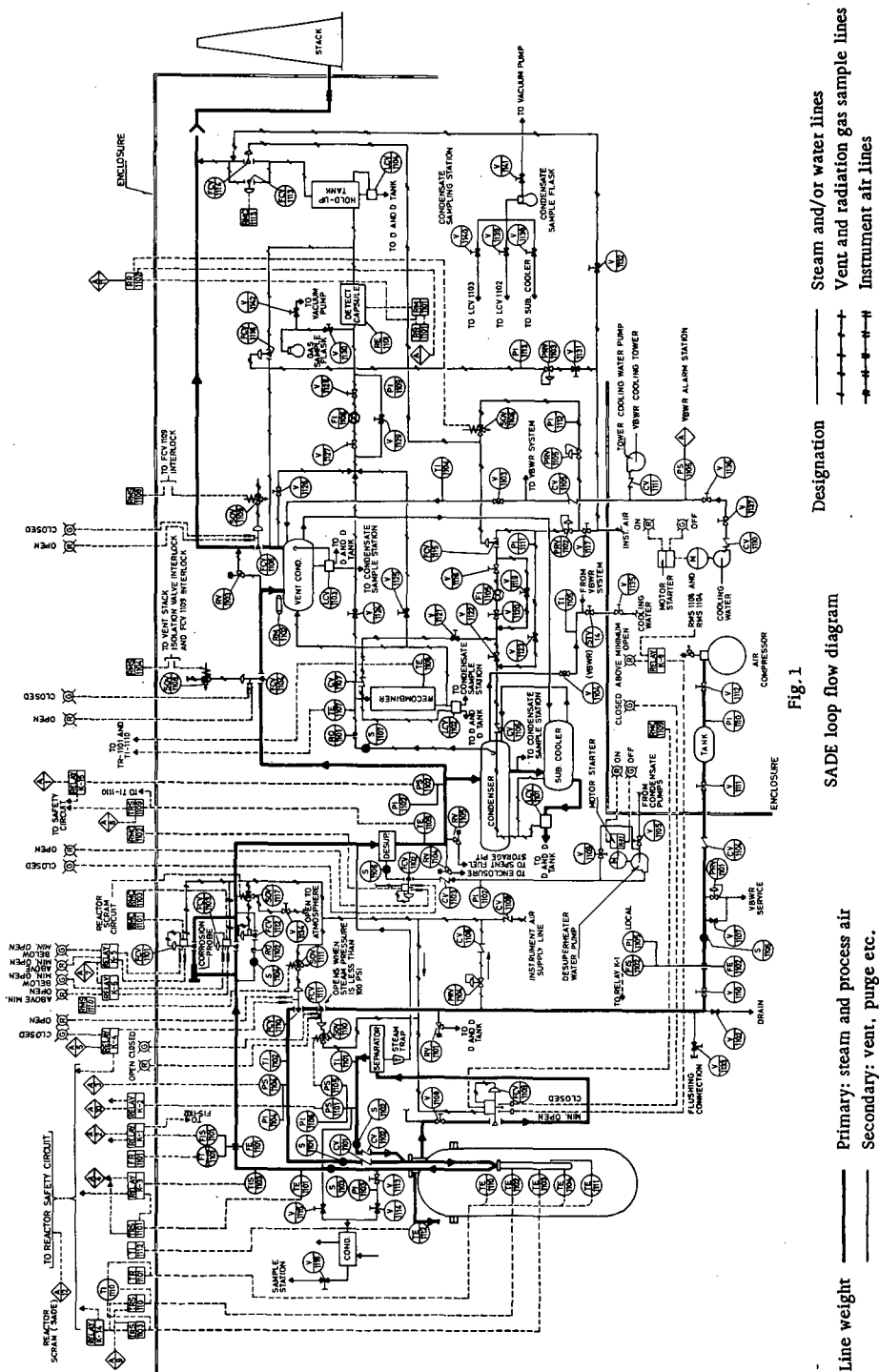


Fig. 1

SADE loop flow diagram

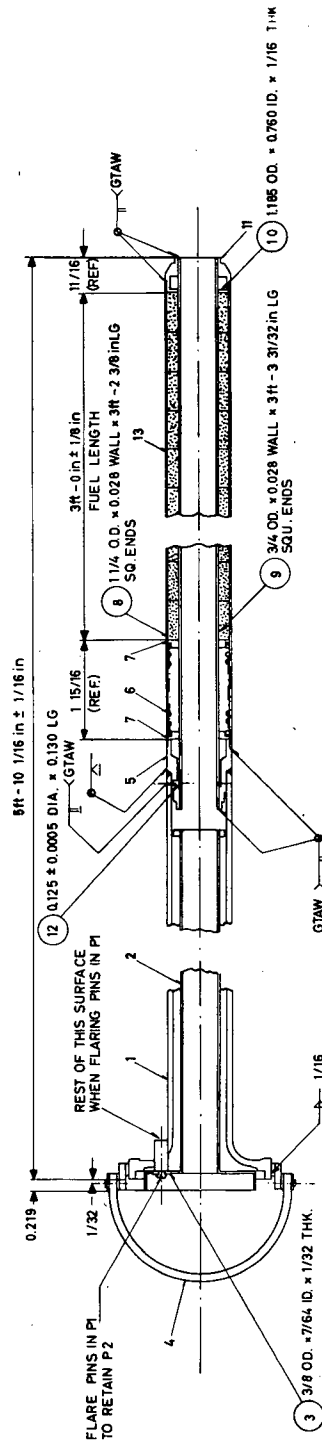


Fig. 2

Typical SADE fuel element

TABLE I
SUPERHEAT FUEL IRRADIATION EXPERIENCE IN SADE IN VBWR

	SH-1	SH-2	SH-4B	SH-4	SH-4C	SH-5A	NUSU	SH-4D
Process tube	304	304	304	304	304	304	None	304
Outer diam. (in)	1.660	1.660	1.660	1.660	1.660	1.660	--	2.000
Wall thickness (in)	0.109	0.109	0.109	0.109	0.109	0.109	--	0.120
Liner - outer diam. (in)	--	--	--	--	--	--	--	1.45
Wall thickness (in)	--	--	--	--	--	--	--	0.025
Material	--	--	--	--	--	--	--	304
Steam gap, first pass (cold) (in)	0.096	0.096	0.096	0.096	0.096	0.096	--	0.075
Outer clad	304	304	304	304	304	304	347	Inconel
Outer diam. (in)	1.250	1.250	1.250	1.250	1.250	1.250	1.500	1.250
Wall thickness (in)	0.049	0.049	0.028	0.028	0.028	0.016	0.028	0.028
VO_2 (% enrichment)	2.3	2.3	4.5	3.5	3.5	4.0	3.5	3.9
Outer diam.	1.151 $\pm$ 0.001	1.151 $\pm$ 0.001	1.194 $\pm$ 0.0005	1.190 $\pm$ 0.005 - 0.000	1.190 $\pm$ 0.004 - 0.000	Not Completed	--	--
Inner diam.	0.750 $\pm$ 0.005 - 0.000	0.750 $\pm$ 0.005 - 0.000	0.755 $\pm$ 0.009 - 0.000	0.753 $\pm$ 0.011 - 0.000	0.757 $\pm$ 0.008 - 0.000	Not Completed	--	--
Weighted inside diametral fuel-clad gap (cold) (mil)	Not deter- mined	Not deter- mined	7.5	5.4	7.0	Not deter- mined	--	--
Inner clad	347	347	304	304	304	304	Inconel X	Inconel
Outer diam. (in)	0.750	0.750	0.750	0.750	0.750	0.750	1.072	0.750
Wall thickness (in)	0.035	0.035	0.028	0.028	0.028	0.016	0.025	0.028
Steam gap, second pass (cold) (in)	0.153	0.153	0.097	0.097	0.097	0.109	0.104	0.159

TABLE I (cont'd)

Instrument tube outer diam. (in)	0.375	0.375	0.500	0.500	0.500	0.500	0.500	0.375	0.375	0.500	0.375	0.375
Active fuel length (in)	36	36	36	36	36	36	36	36	36.5	36	36	36
Power (kW)	50	50	100	100	73	78	122	65	122	65	65	65
Power density (kW/kg)	16.7	16.7	26.6	26.6	18.3	18.2	28.7	15.2	28.7	15.2	15.2	15.2
Fuel section inlet temp. (°F)	545	545	545	545	545	575	545	545	545	545	545	545
Max. clad temp. (STEADY) (°F)	1070	1070	1200	1200	1200	1200	1200	1200	1200	1200	1200	1200
Max. clad temp. (TRANSIENT) (°F)	--	--	1292	1292	1300	1300	1300	--	1300	--	--	--
Max. superheat temp., fuel exit (STEADY) (°F)	825	825	925	925	915	857	747	900	747	900	900	900
Max. superheat temp., heater exit (STEADY) (°F)	None	None	None	None	None	803	None	None	None	None	None	None
Max. superheat temp., fuel exit (TRANSIENT) (°F)	--	--	980	980	980	910	794	--	794	--	--	--
Max. heat flux (STEADY), (BTU ft ⁻² h ⁻¹) (Steam)	200 000	200 000	332 000	332 000	276 000	289 000	248 000	260 000	248 000	260 000	260 000	260 000
Max. heat flux (TRANSIENT), (BTU ft ⁻² h ⁻¹) (Steam)	--	--	332 000	332 000	276 000	289 000	242 000	--	242 000	--	--	--
Min. steam flow rate (STEADY) (lb/h)	730	730	1032	1032	795	1180	740	865	740	865	865	865
Min. steam flow rate (TRANSIENT) (lb/h)	--	--	915	915	690	1030	615	--	615	--	--	--

TABLE I (cont'd)

	SH-1	SH-2	SH-4B	SH-4	SH-4C	SH-6A	NUSU	SH-4D
Max. fuel temp. (STEADY) (°F)	2080	2080	3030	3030	2700	2770	3330	2550
Max. fuel temp. (TRANSIENT) (°F)	--	--	3120	3120	2800	2860	3380	--
Design drawing	141F607	141F607	141F807	141F785	762D438	762D437	612D850	762D771
Outer clad spacer	Fin (4)	Fin (4)	Fin (4)	Fin (4)	S. wire (3)	Pin (4) Attached to process tube	None	Dimpled process tube
Instrument tube or velocity tube spacer	Fin (4)	Fin (4)	Fin (4)	Fin (4)	Fin (4)	Fin (4)	Fins,	Fins,
Inert ZrO ₂ spacers	Yes	Yes	Yes	Yes	No	No	At ends	No
Bellows	None	Yes	None	None	None	None	None	None
Gum-drop (UO ₂ expansion monitor)	None	None	Yes	Yes	Yes	Yes	None	None
Outer clad F = free standing NF = non-free standing	F	F	Marginal	Marginal	Marginal	NF	NF	Marginal
Total irradi. time, (h)	Not compiled	Not compiled	617	864	492	1100	820	428
Approx. irradi. period (START)	7-31-59	5-1-59	1-15 & 4-24-61	6-7-61	8-28-61	11-13-61	2-2-62	4-4-62
Approx. irradi. period (END)	10-15-59	6-30-59	1-29 & 5-25-61	8-20-61	9-23-61	1-12-62	3-30-62	4-29-62
Burn-up, Mwd/t *** (APPROX)	620	620	370	610	170	550	635	124
Estimated max. clad temp. (°F)	≈ 950	≈ 1050	1350 to 1500**	1350 to 1500**	1200 to 1350**	1150 to 1300	1200 to 1300	--
Max. superheat exit temp. (°F)	775	825	893	900	840	845	780	850
Estimated max. heat flux, BTU ft ⁻² h ⁻¹	216 000	176 000	356 000	240 000	240 000	265 000	260 000	225 000

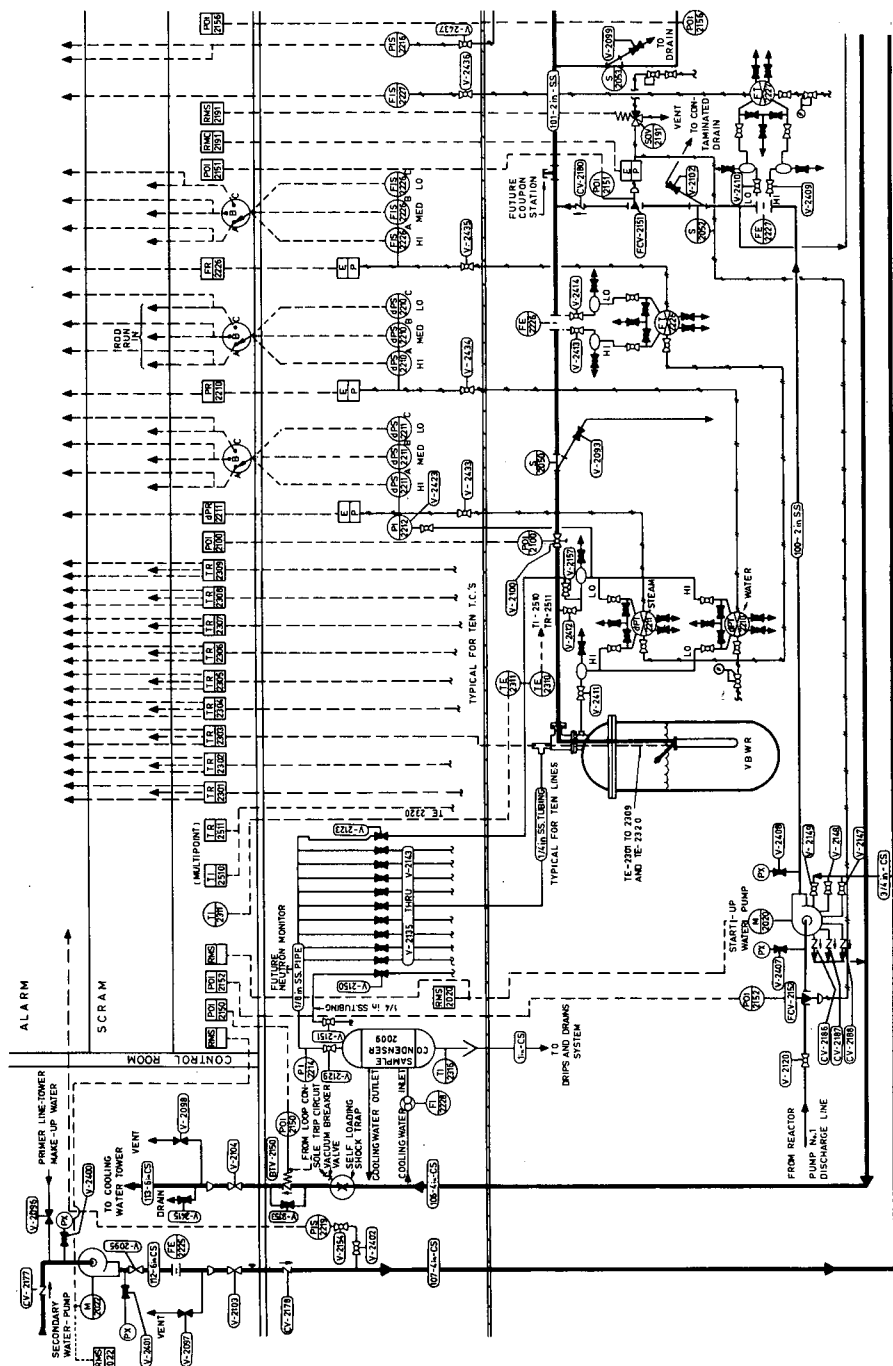
TABLE I (cont'd)

	SH-1	SH-2	SH-4B	"H-4	SH-4C	SH-5A	NUSU	SH-4D
Estimated max. power (kW)	54.5	44	94	63	63	75	122	56
Reason for termination of test	VBWR modification	Failure	Failure	Scheduled	Failure	Scheduled	Scheduled	Scheduled
Chloride corrosion detected	Not measured	Not measured	Yes	C1 detected in corrosion film	Yes	Yes	--	--
Position and cause of failure		Crack in bellows - probably caused by inadequate or defective bellows	Pinhole defects adjacent to middle spacer - longitudinal split defect about 2 in below middle spacer - corrosion attack - predominantly intergranular.	--	Circumferential crack near the top of inside clad brittle failure.	Small perforations lower outside cladding	--	--
Fabrication, irradiation and evaluation reports	GEAP-3211 GEAP-3387	GEAP-3211 GEAP-3387	GEAP-3796	--	--	--	--	--

* Boiling water side.

** Values take into account the lower fuel-to-clad contact coefficient of $150 \text{ BTU ft}^{-2} \text{ }^{\circ}\text{F}^{-1} \text{ h}^{-1}$, a flux skewing factor of approximately 1.3, and a film temperature correction on the steam film coefficient. The temperature exists over a length of about 4 in and a circumferential arc of at least 30° .

*** Per ton uranium metal.



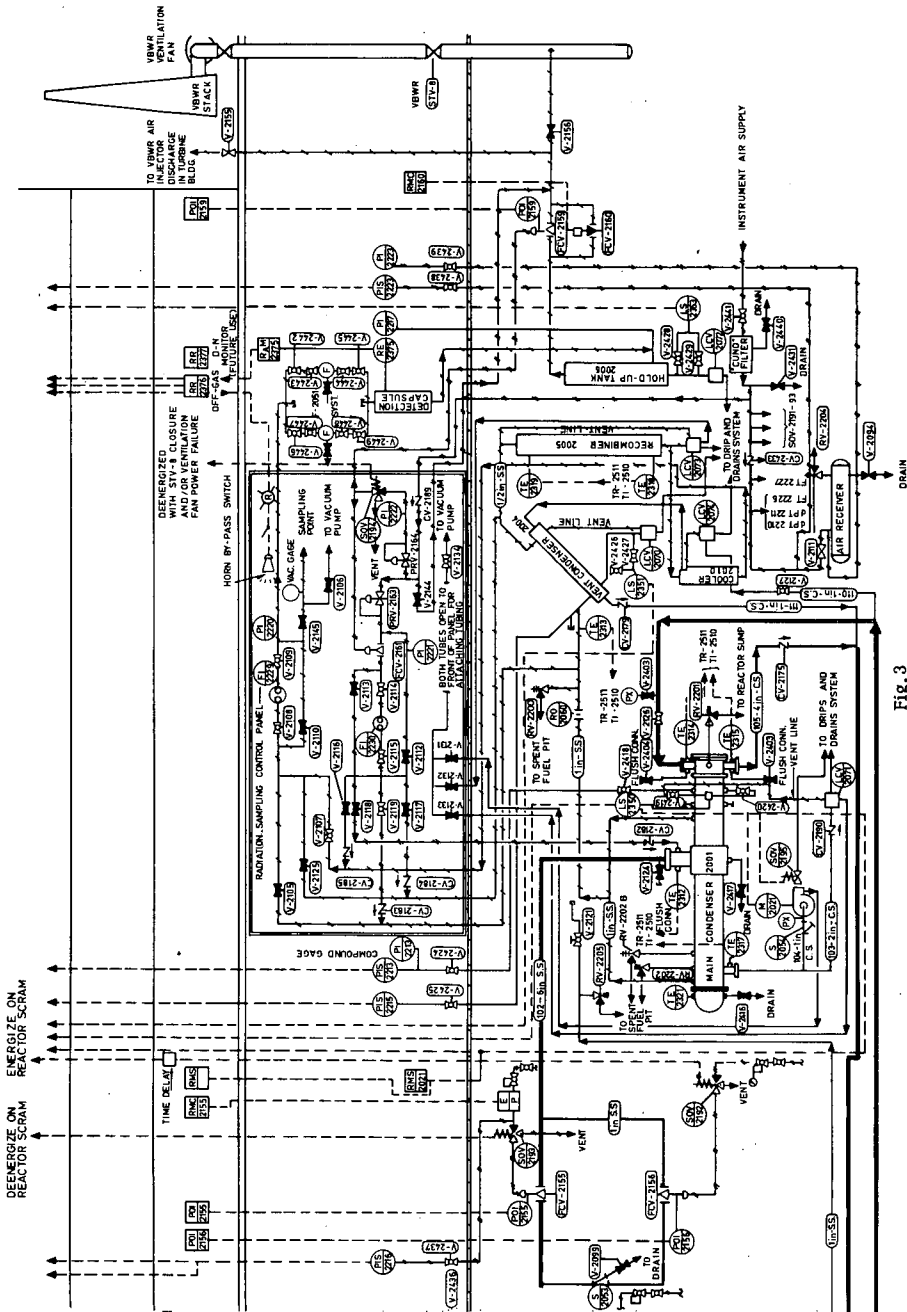


Fig. 3
E-SADE loop flow diagram

TABLE II
DESIGN DATA FOR ESH 1.0 IN E-SADE IN VBWR

	Elements	A	B	C	D	E	F	G	H	I
Independent Power	Fuel enrichment (% U^{235})	8	10	10	8	6	8	6	8	8
	Peak to avg. power scallop	1.24	1.24	1.24	1.22	1.18	1.21	1.19	1.19	1.23
	Linear radial flux depression factor	1.37	1.62	1.62	1.37	1.27	1.37	1.27	1.37	1.37
	Clad material	Inconel	Incoloy	Incoloy	Inconel	Inconel	304 SS	Inconel	Incoloy	Inconel
At 38 MW	Process tube thickness (mil)	49	49	49	49	25	49	49	49	49
	Power (kW)	76.8	73.2	75.5	75.4	54.2	67.3	82.7	86.0	84.5
	Fuel element, pressure drop wt. % orifice (lb/in ²)	27.6	27.1	26.4	26.4	11.7	20.2	31.4	33.8	34.2
	Orifice pressure drop (lb/in ²)	17.4	17.9	18.6	18.6	33.3	24.8	13.6	11.2	10.8
At 42 MW	Max. heat flux (BTU ft ⁻² h ⁻¹)	305 000	291 000	299 000	299 000	213 000	267 000	329 000	342 000	336 000
	Min. steam flow rate (lb/h)	1142	1135	1111	1110	726	950	1221	1272	1282
	Max. fuel temperature (°F)	2700	2620	2680	2680	2300	2500	2830	2950	2860
	Power (kW)	84.9	80.9	83.4	83.3	59.9	74.4	91.5	95.0	93.4
At 42 MW	Fuel element pressure drop wt. % orifice (lb/in ²)	35.0	34.4	33.5	33.5	14.8	25.6	39.9	42.9	43.4
	Orifice pressure drop (lb/in ²)	22.1	22.7	23.6	23.6	42.3	31.5	17.2	14.2	13.7
	Max. heat flux (BTU ft ⁻² h ⁻¹)	337 000	322 000	330 000	330 000	235 000	285 000	364 000	378 000	372 000
	Min. steam flow rate (lb/hr)	1287	1279	1252	1251	818	1070	1376	1433	1444
At 42 MW	Max. fuel temperature (°F)	2870	2810	2850	2850	2380	2660	3020	3130	3080

TABLE II (cont'd)

Steam pressure	1000 lb/in ² abs.
Active fuel length	36 in
Outer clad, outside diameter	1.25 in
Inner clad, outside diameter	0.75 in
Outer and inner clad thickness	0.028 in
Process tube, inside diameter	1.402 in
Gap between pellet and outer clad	0.006 in
Velocity boost tube, outside diameter	0.500 in
Method of plenum support	Spring
Weight of UO ₂ /element (calculated)	4.5 kg
Fuel (UO ₂) % theoretical density	95%
Axial peaking factor	1.58
Type of fuel fabrication	Pellet
Inside clad-to-fuel contact coefficient	150 BTU ft ⁻² °F ⁻¹ h ⁻¹
Outside clad-to-fuel contact coefficient	1000 BTU ft ⁻² °F ⁻¹ h ⁻¹
Inlet steam temperature	560° F
Maximum clad temperature*	1250° F
UO ₂ thermal conductivity	1.6 BTU ft ⁻² (°F/ft) ⁻¹ h ⁻¹
Indicated start-up cooling water flow	81 gal/min

* Accounts for power scalloping assuming complete circumferential steam mixing.

TABLE III
DESIGN DATA AND DESIGN CONDITIONS - ESH-1.1
(For 40 MW reactor power)

Channel position	A	B	C	D	E	F	G	H	I
Type of fuel element	3-rod assy.	BONUS rod	2-pass annular	Rod	2-pass annular	Rod	2-pass annular	2-pass annular	2-pass annular
Steam pressure (lb/in ² abs.)	1000	1000	1000	1000	1000	1000	1000	1000	1000
Fuel length (in)	36	35.80	36	36	36	36	36	36	36
Outer clad, outer diam. (in)	0.438	0.539	1.250	0.550	1.250	1.550	1.250	1.250	1.250
Outer clad, thickness (in)	0.016	0.018	0.028	0.024	0.028	0.024	0.028	0.028	0.028
Inner clad, outer diam. (in)	None	None	0.750	None	0.750	None	0.750	0.750	0.750
Inner clad, thickness (in)	None	None	0.028	None	0.028	None	0.028	0.028	0.028
Cladding material	2 Incoloy	348 S S	Incoloy	304 vac. melt	Inconel	310 vac. melt	Inconel	Incoloy	Inconel
Process tube wall thickness (in)	1347 SS	0.049	0.049	0.049	0.025	0.049	0.049	0.049	0.049
Process tube, inner diam. (in)	0.049	1.402	1.402	1.402	1.402	1.402	1.402	1.402	1.402
Velocity boost tube, outer diam. (in)	(0.585)	(0.733)	0.500	(0.750)	0.500	(0.750)	0.500	0.500	0.500
Velocity boost tube, inner diam. (in)	0.555	0.7093	---	0.666	--	0.666	Specimens	Specimens	--
Velocity tube wall thickness (in)	0.015	0.012	---	0.042	--	0.042	--	--	--
Outer (fuel) flow annulus clearance (in)	0.059	0.085	0.076	0.058	0.076	0.058	0.076	0.076	0.076
Inner (fuel) flow annulus clearance (in)	None	None	0.097	None	0.097	None	0.097	0.097	0.097
Weight of UO ₂ (lb)	5.05	2.63	0.12	2.60	8.88	2.63	8.73	8.95	8.90
Theoretical fuel density (%)	95	95	95	95	95	95	95	95	95
Fuel enrichment, U ²³⁵ (%)	8	3.41	10	2.6	6	3.9	6	8	8
Axial peaking factor	1.58	1.58	1.58	1.58	1.58	1.58	1.58	1.58	1.58
Peak to average power scallop (Linear) radial flux depression factor	1.08	1.07	1.17	1.06	1.09	1.05	1.23	1.16	1.19
	--	--	1.62	--	1.27	--	1.27	1.37	1.37

TABLE III (cont'd)

Channel Position	A	B	C	D	E	F	G	H	I
Type of fuel fabrication	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet	UO ₂ pellet
Inside clad-to-fuel contact coefficient (BTU ft ⁻² °F ⁻¹ h ⁻¹)	--	--	150	--	150	--	150	150	150
Outside clad-to-fuel contact coefficient (BTU ft ⁻² °F ⁻¹ h ⁻¹)	1000	1000	1000	1000	1000	1000	1000	1000	1000
Inlet steam temperature	570	555	555	555	555	555	555	555	555
Maximum clad temperature*	1190	1145***	1110	1070	1250	1090	1220	1050	1000
UO ₂ Thermal conductivity (BTU ft ⁻² °F/ft ³ h ⁻¹)	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6
Fuel spacing method	2 spiral	Finned collars	Process tube pins	2 spiral wires	Process tube pins	2 spiral wires	Process tube pins	Process tube pins	Process tube pins
Method of plenum support	Wires	In end plug	Spring	Spring	Spring	Spring	Spring	Spring	Spring
Power (kW)	71.1 Total	16.5	78.0	16.2	67.4	16.7	95.2	81.4	74.4
Fuel element pressure drop (less orifice) (lb/in ²)	22.25	5.5	39.0	6.96	18.4	6.96	47.0	50.07	50.0
Orifice pressure drop (lb/in ²)	44.75	61.5	28.0	60.04	48.6	60.05	20.0	16.93	17.0
Max. heat flux (BTU ft ⁻² h ⁻¹)	371 000	208 000	311 000	204 000	268 000	211 000	379 000	324 000	296 000
Design steam flow rate (lb/h)**	1237 Total	459	1440	273.5	900	274	1550	1680	1760
Design superheat exit temp. (°F)**	856	700	752	826	845	835	790	728	699
Maximum fuel temperature (°F)	3590	2720	2610	2715	2570	2790	3090	2660	2450

* Accounting for power scalloping and assuming complete circumferential mixing.

** Not including fuel end seal leakage which bypasses thermocouple section.

*** Using the same factors for flow channel eccentricities, spacer effects, steam film coefficient, etc. as used for ESH-2 design conditions.

on the results from fuel elements irradiated in both the SADE and E-SADE loop, the following general conclusions can be drawn.

- (a) Commercial 18-8 stainless steel may be subject to rapid failure due to stress-corrosion cracking when placed in a nuclear superheat environment. Both the magnitude of strain cycling and the presence of chlorides are important factors in determining the length of time to failure.
- (b) Based on in-pile and out-of-pile evaluations, alloy materials with increasing amounts of nickel inhibit stress-corrosion cracking and have operated satisfactorily in the E-SADE at exposures up to 1100 MWd/t.

4. ESADA VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR (EVESR)

The EVESR reactor is being constructed through the combined efforts of the Empire States Atomic Development Associates, (ESADA) Incorporated, an organization of New York State investor owned electric-utility corporations, and the General Electric Company [18, 34]. The EVESR Nuclear Superheat Development Project sponsored by the USAEC is described in section 6.

The EVESR reactor has been designed as a flexible fuel test-bed with the capability of testing a large number of superheat fuel elements in an operating superheat reactor environment. The EVESR reactor utilizes externally generated steam as the primary reactor coolant. This independence of coolant source provides one of the key features of EVESR in terms of superheat fuel-element irradiation flexibility, in that steam can be supplied by either the fossil-fuelled saturated-steam boiler or the VBWR reactor, or both. In order to perform experiments on superheated-fuel performance limits, the reactor and associated systems are designed to operate with either purposely defected or in-service failed superheat fuel elements. Experimental flexibility is additionally supplied by utilization of twelve coolant-manifold systems with suitable bias control valves. The instrumentation system provides for the determination of inlet and exit steam temperature, steam pressure, steam flow and selected coolant-sampling stations. The initial power level of the EVESR reactor is 12.5 MW(t) with provision for eventual operation at 23 MW(t). The EVESR biological shielding is adequate to operate at 30 MW(t). The 7-ft inner diam. vessel is provided with blanked connections for a forced circulation system for possible modification to permit operation as a combination boiling-superheat reactor (Figs. 4 and 5). For integral boiling-superheat operation the power level would be about 70 MW(t). Although the primary purpose of the EVESR reactor is to provide a flexible engineering tool for the development of superheat fuel elements, it is also expected that a significant technical first will be obtained with the operation of VBWR and EVESR in series. In addition to reactor dynamics and fuel-development information, it is expected that a significant amount of operating experience will be obtained on direct-cycle nuclear superheated systems including the distribution and level of radioactive contamination resulting from operation with purposely defected and in-service failed superheat fuel elements.

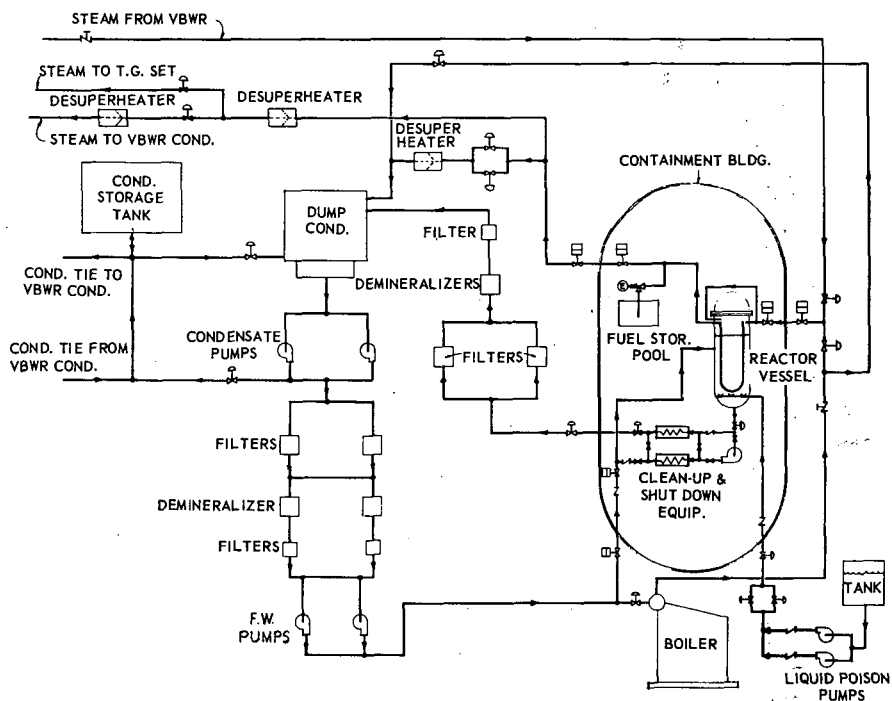


Fig. 4

EVESR flow diagram

The initial core fuel-cladding material selection for the 32 fuel bundles of 9 elements is shown in Table IV [35].

Sufficient experimental information is available to show that commercial stainless steels are subject to gross failure from stress-corrosion cracking. This has been shown through boiling MgCl_2 tests, failures in the SADE loop with and without in-leakage of VBWR reactor water and in out-of-pile testing of Type 304, sensitized Type 304 and Type 348 stainless steels. A search of available literature [54] indicated two different approaches to the solution of the stress-corrosion cracking problem for cladding materials. These are the use of high-purity stainless-steel alloy and use of alloy materials with increased nickel content.

Recent tests performed at General Electric [31-33, 59, 62, 64] have confirmed that low-impurity stainless steels exhibit a significant improvement in resistance to chloride stress cracking in boiling MgCl_2 tests. These tests indicate that high-purity Type 304 stainless steel will not fail in less than 6 h, as compared to consistent failures of commercial stainless steels in less than 30 min. Out-of-pile corrosion tests on heat-transfer sheath specimens have also shown an improvement of materials with increasing nickel-alloy content.

Based on satisfactory performance in boiling MgCl_2 tests and uniform corrosion and cycling-corrosion tests performed out-of-pile at General

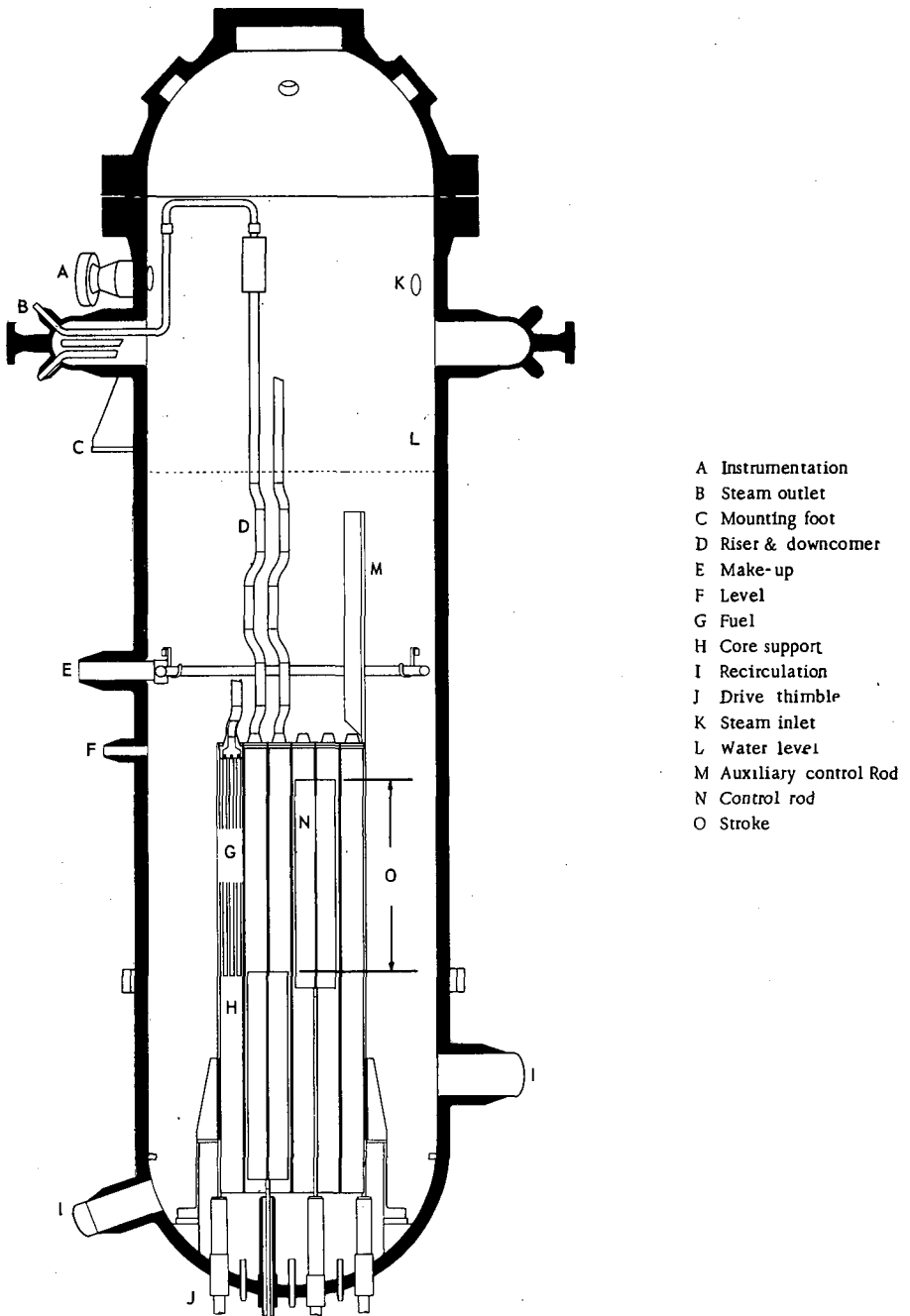


Fig. 5

Cross-section of the EVESR pressure vessel

TABLE IV

PROPOSED TEST CONDITIONS FOR INITIAL EVESR CORE LOADING

Number of bundles	Zone number	Material	Power local bundle to core average	Power hot element to core average	Calculated maximum surface temperature (°F)
1	1	304 SS Comm'l.	1.216	1.25	1250
3	1	304 VM	1.216	1.25	1250
4	3	Incoloy	1.162	1.41	1250
4	3	Incoloy	1.162	1.41	1000
4	5	Incoloy	0.790	1.11	1250
4	5	Incoloy	0.790	1.11	1000
4	4	Inconel	0.870	1.22	1250
4	4	Inconel	0.870	1.22	1000
4	2	310 VM	1.129	1.50	1250

Electric, and in recognition of the relatively improved high-temperature strength properties and relatively small increase in thermal cross-section. Incoloy would seem to be a promising superheat cladding material at present. For this reason one-half of the initial fuel load for the EVESR will be fabricated from fuel clad with this material.

The EVESR reactor is scheduled to go critical during July 1963. The construction of the reactor facility will be completed during May 1963. Fuel fabrication was initiated in February with the delivery of the initial core loading now scheduled to be made at the reactor site by 15 June 1963. The Final Hazards Report was submitted to the USAEC in October 1962. Meetings were held with the USAEC staff during Spring 1963 and the final meeting with the Auxiliary Committee on Reactor Safety was held on 11 April 1963. The operating licence for EVESR will be effective on 24 June 1963.

5. USAEC BASE DEVELOPMENT, NUCLEAR SUPERHEAT PROJECT

The USAEC-sponsored Nuclear Superheat Project was initiated in July 1959 and is still in progress. The objective of the Nuclear Superheat Project [20] is to provide a fundamental basis for the extension of superheat technology into the realm of practical competitive application for commercial production of power. The following is a brief summary of significant result areas over the period of the programme.

Superheat reactor conceptual design

The objective of these design evaluations has been to provide development-programme direction and to establish realistic test conditions for the experimental programme. This work has been implemented through the conceptual design studies of three different reactor types. The first concept studied was the separate-superheat reactor [41, 61] concept where saturated steam from a conventional boiling-water reactor is used to provide the source of primary coolant for the superheat reactor, which is physically isolated from the boiling-water reactor in its own pressure vessel. The separate-superheat reactor concept has the advantage of individual design consideration of boiling-water and steam-cooled reactor fuel such that each system may be optimized within its own performance limits. Since there is no nuclear coupling between regions of different nuclear characteristics, complete power level independence between the boiling and superheat region leads to a number of major simplifications in reactor power-level control and exit steam-temperature control (Fig. 6).

The second reactor concept studied is the once-through superheat reactor [44, 46] where regions of boiling, transition boiling and superheating of the steam all occur within a single reactor vessel. The conceptual design studies indicated that this should be done in three different fuelled regions in order to minimize the reduction of heat-transfer performance between the various heat-transfer regions. The source of the cost incentive in this concept is the potential for elimination of the boiling-water recirculation system; however, the once-through superheat reactor has several major areas of technical uncertainty which are:

- (a) Development of flow-control system to match local fuel-bundle flow rate with local power as necessary to limit excessive superheat temperature rise. This problem is due to the low ratio of enthalpy rise in superheat to the total enthalpy rise.
- (b) Local power perturbations and reactor power-level control difficulties due to power interdependence between fuel regions of variable nuclear properties due to geometric or other variances resulting from specific heat-transfer limitations.
- (c) Possibility of increased radioactivity levels in primary coolant system due to primary coolant system impurities which could be deposited on the fuel surface in the high neutron-flux region. For a once-through system, the amount of deposited material can be substantial even for systems of extremely high purity.

The third reactor concept studied was the mixed-spectrum superheat (MSSR) [42, 47, 60] concept which consists of a fast-energy fission spectrum, superheated-steam cooled region surrounded by a conventional, light-water moderated, thermal spectrum, boiling-water reactor. The boiling water and superheat regions are separated by two buffer regions to reduce the power perturbations between the two regions. This concept has a number of significant advantages as compared to thermal, integral superheaters. In general these advantages are obtained at the cost of a significantly more complex neutron physics system (Fig. 7).

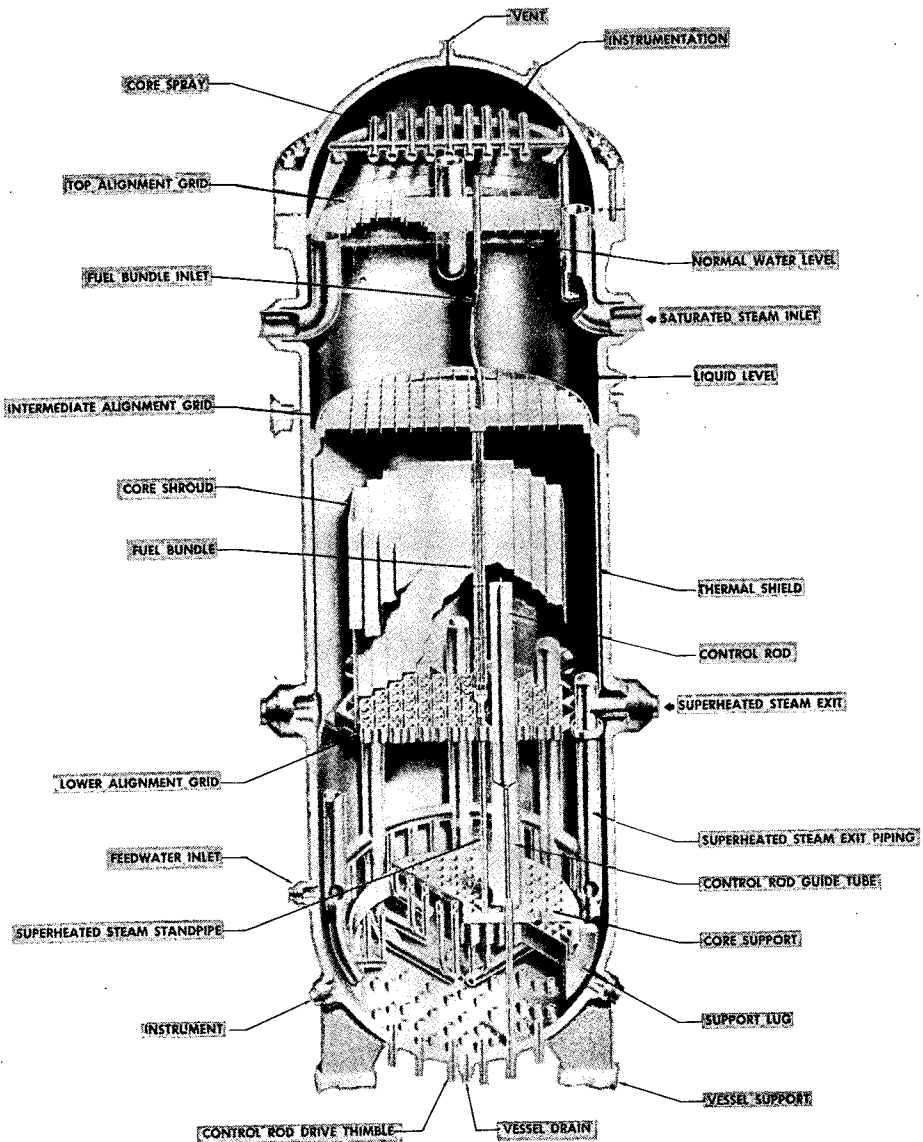


Fig.6

Separate superheat reactor for 600 MW(e)

- (a) Mechanical simplification is obtained by physical separation of the high-temperature, superheated-steam cooled region from the relatively lower temperature boiling-water region. This is possible since no moderator is required in the fast-flux region.
- (b) Low fuel-cycle costs due to the possibility of superheat fuel burn-ups as high as 50 000 MWd/t. Very high burn-up is possible since there

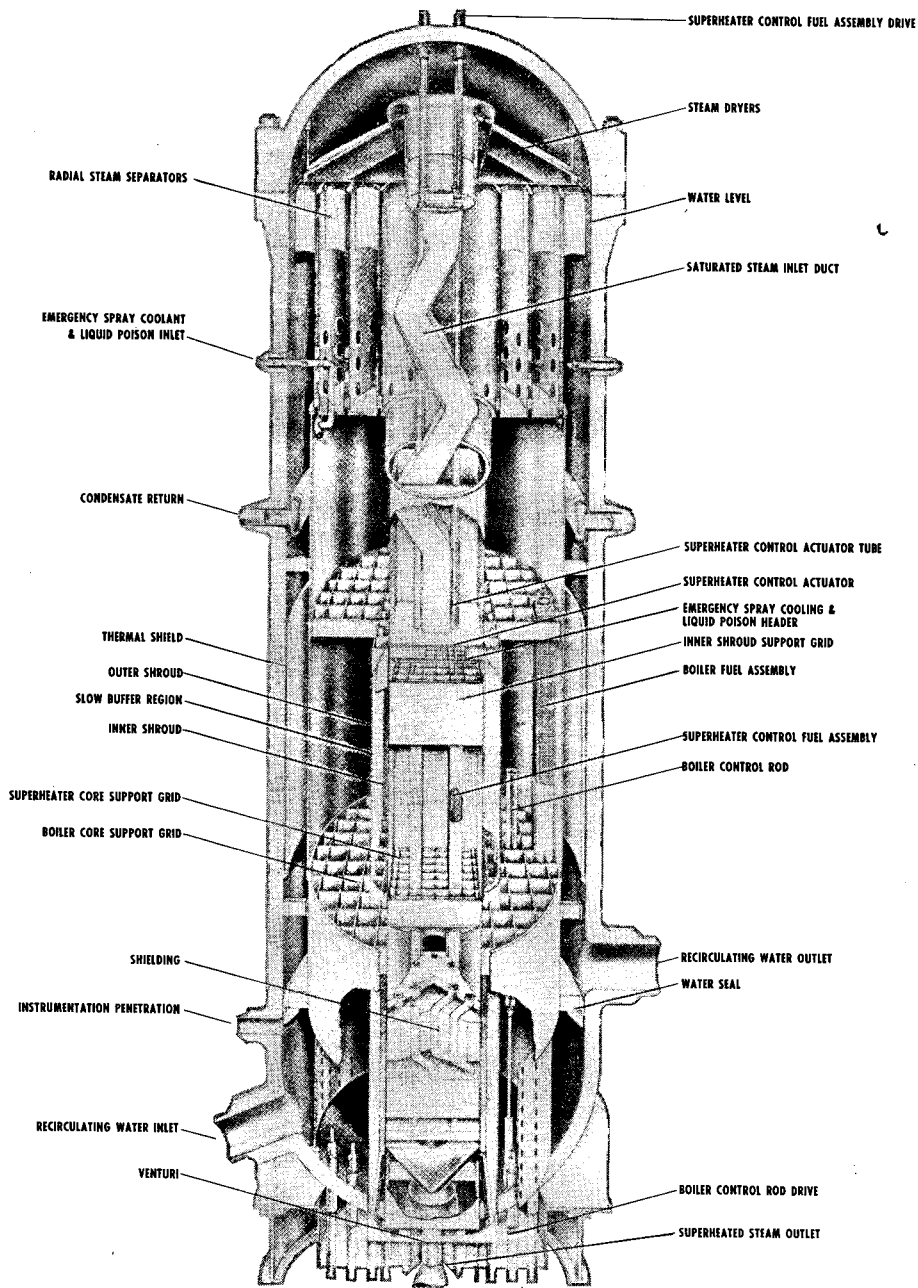


Fig. 7

Mixed-spectrum superheater reactor

will be very little loss of reactivity in the fast region because of the high initial enrichment and high conversion ratio.

- (c) Efficient utilization of plutonium as a fuel because of the fast-energy spectrum.
- (d) Low capital costs due to the potential for significant increases in the volumetric heat release from a single pressure vessel.

Fuel technology

The fuel technology programme has had the objective of fabrication, irradiation and evaluation of various superheat fuel concepts to establish an experimental basis for determining performance limits. This task has resulted in the irradiation of eight SADE fuel elements as listed in Table I and two E-SADE superheat irradiation experiments as listed in Tables II and III [38, 53]. In addition to the fuel irradiations, determinations have been made of materials that would provide improved performance in nuclear superheat environment leading to recommendations for the selection of cladding materials and fuel elements to be irradiated in EVESR [43, 45, 48, 51, 56, 63]. In addition, work has been done to develop non-destructive techniques for fuel-element inspection [64], and post-irradiation evaluations have been performed to determine the cause of failure of fuel elements irradiated.

Two fuel elements were irradiated in 1959 for one month each, but clad thickness was higher (0.049 in, 1-1/2 in outer diam.), annular design 304-clad tubes containing 94% minimum density UO_2 pellets) and power densities lower (50 kW) than those considered economically attractive [38]. In experiments conducted in 1961, the fuel-element power was increased by a factor of two to approximately 90 kW and the 304-clad thickness was simultaneously decreased to 0.028 in. [53]. Both of these experimental variables were in line with plans to improve fuel-cycle economics associated with the thermal-spectrum, superheat reactor concept.

In April 1961, fuel prototype SH-4B was irradiated in the VBWR long enough [53] at relatively high heat fluxes so that the results were significant. Fuel element designated as SH-4B failed during reactor operation mainly through intergranular corrosion attack of the outer cladding. There was ample evidence of high moisture carryover in the steam and slightly higher than normal chloride concentration in the water during SH-4B irradiation. Even though these extreme conditions of high chloride and moisture in steam were considered severe when compared to normal design conditions, the test pointed out a limiting case in environmental conditions for superheat fuel elements clad in 300-series stainless (austenitic) steels.

Type 304 stainless steel, both in annealed and slightly cold-worked conditions, was used in SH-4B. Intergranular attack occurred in both metallurgical conditions, with some transgranular attack in areas where the sensitization of the base metal was not so prominent (clad surfaces located over the ZrO_2 pellet where heat generation was restricted to low ratings).

In another irradiation test in September 1961, SH-4C fuel element similar to SH-4B failed through circumstances similar to those observed in SH-4B. Severe intergranular attack of the new clad occurred, followed by a through-the-tube wall failure within 10 d of reactor exposure. Radioactive tracer techniques were used in both runs to determine water leakage

into the fuel compartment from a flange which separates the incoming steam from the reactor water.

Still another experiment, SH-5A, where the water leakage into the fuel compartment was eliminated by welding a previously bolted flange, pointed out that 304-type clad was subject to rapid, highly-localized corrosion attack.

Upon post-irradiation examination of the failed elements, it was found that large quantities of chloride impurities deposited on the fuel-clad surface. This evidence plus the morphology of the crack under careful metallographic examination led to the conclusion that the jacket failures were due to stress-corrosion attack. Further comparison of present knowledge regarding stress-corrosion attack in austenitic alloys, with the conditions prevalent in SADE environment, led to the conclusions that there were three avenues of approach for the solution of the superheat fuel cladding problem:

- (a) The stress level in fuel elements must be minimized through improved design or less severe performance requirements;
- (b) Chloride deposits on fuel must be reduced either by reducing moisture carryover or better control of chlorides in the reactor water; or
- (c) Alternate materials must be selected which are less susceptible to stress-corrosion attack.

Materials development

The purpose of the materials development evaluations is to investigate mechanical properties of potential fuel-cladding materials as these materials are affected by long term exposure to superheat environments of high-temperature, corrosive atmosphere of hydrogen and oxygen, and neutron flux. These evaluations have consisted of the experimental determination of strain-cycle limits, the aging effect of alloys and tensile property measurements for irradiated samples at room temperature and elevated temperatures. In addition, an extensive literature survey [54] was conducted to find potential superheated fuel-element cladding materials. This work provided a basis for the selection of alternate cladding materials and also provided a better understanding of the behaviour of materials in reactor environments.

Information regarding strain-cycle fatigue of 304, Incoloy, Hastelloy X and Inconel has been obtained at General Electric. The results are summarized in Fig. 8. Irradiation of 304-type stainless steel at 1250-1300°F reduced the time to failure at a given strain range by approximately a factor of 2/3. These tests differ from those previously reported in that they were performed with tubular specimens and stresses were applied biaxially. Short lengths of tubing (0.016-in and 0.028-in thick, 1.25-in diam.) were alternately expanded and contracted pneumatically between rigid fixed mandrels. Frequency of cycling was 30 min for a complete cycle. The gases used were either argon or nitrogen and the strain was determined by the dimensions of the mandrel. Failures occurred at the convex side of the specimen. For strains greater than 0.01 the actual strain ranges cannot be accurately determined because of the formation of wrinkles, which in some cases was arrested by the size of the mandrel. The microstructural characteristics of the fractures for Inconel, Hastelloy X and 304 are essentially identical.

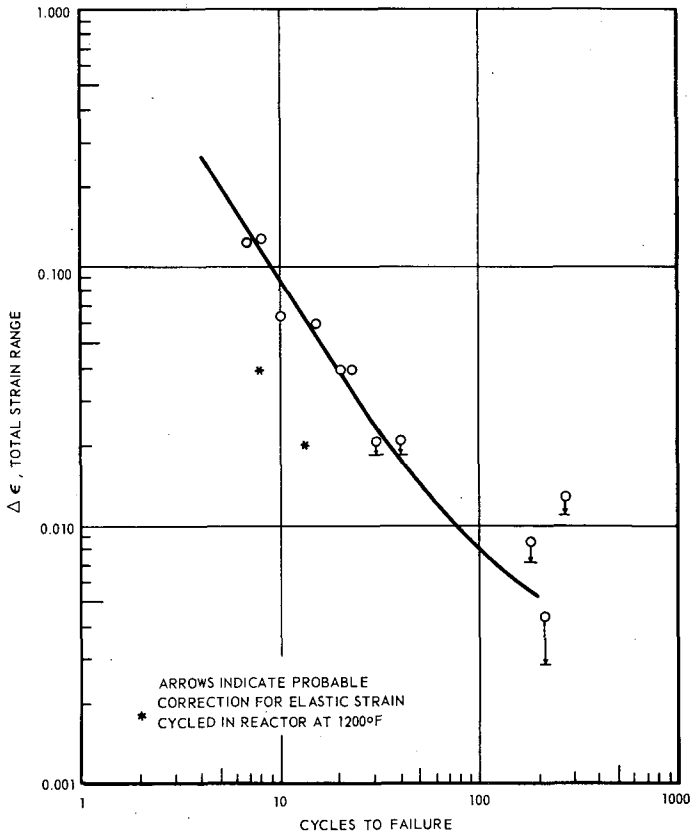


Fig. 8

Strain-cycle tests, Type-304 stainless steel at 1300°F

The grains remain equiaxed up to the fracture surface indicating little or no plastic deformation before failure. There was evidence of considerable deformation in Incoloy and fracture was transgranular. Incoloy exhibited more ductility at fracture due to strain cycling than any of the materials tested to date.

Experimental physics measurements

This work was intended to provide a basis for predicting the nuclear physics characteristics of superheat reactors utilizing slightly-enriched UO_2 annular fuel geometry [43] where the moderator to fuel ratio is subject to step changes due to flooding and unflooding of steam passages adjacent to fuel [55]. The experimental measurements included were:

- (a) Critical size
- (b) $(\partial \rho / \partial H)$ versus H (ρ = reactivity, H = water height)

- (c) Void coefficient
- (d) Temperature coefficient
- (e) Flux distributions
- (f) Thermal utilization
- (g) Conversion ratio

All the measurements were carried out utilizing uniformly-spaced arrays of fuel (no controls or water gaps). The experimental results were compared to the prediction of engineering design methods as well as more detailed calculations. A reliable engineering design model based on these comparisons was developed for this lattice type.

Coolant chemistry

This task consists of two major work areas which are the in-pile and out-of-pile coolant chemistry evaluations. The initial out-of-pile evaluations were intended to provide experimental information on uniform corrosion in a dynamic system simulating all superheat conditions except neutron flux [49, 50]. The corrosion facility consists of a 1000 lb/in² loop producing 1050°F steam containing approximately 20 ppm O₂ (Fig. 9). The heater sheath temperatures ranged from 850 to 1300°F. These facilities were modified (in July 1961) to provide experimental evaluations of local corrosion effects such as corrosion-stress cracking on stainless-steel materials. The loop was modified by introduction of uniaxial stress on the heater sheaths and the introduction of 1.5 ppm of chloride ion to the circulating water of the loop [59]. The results of these evaluations are shown on Tables VI and VII.

Conclusions based on these experimental results indicate that the uniform corrosion level of Type 18-8 stainless steel is satisfactory for fuel-cladding application in a nuclear superheat environment, but that corrosion-stress cracking will occur on Type 18-8 stainless steels if the proper conditions are applied. Direct comparisons of the higher nickel alloys indicate a significant improvement in resistance to corrosion-stress cracking. This out-of-pile evaluation has been confirmed by similar results from fuel elements irradiated in the SADE and E-SADE loops in VBWR.

Parallel with the irradiation of superheat fuel elements in the SADE and E-SADE loops in VBWR, extensive radiochemistry measurements were taken of influent and effluent samples of superheated steam from the loops. These measurements indicate that the level of release of radioactive fission products from superheat fuel elements is of the same general order of magnitude as from boiling-water fuel elements. In the case of SADE fuel failure, the release rate was measured at about 45 mc/s.

Heat transfer

Two experimental heat-transfer evaluations were conducted on the Nuclear Superheat Project. The first set of experiments provided experimental determination of heat-transfer coefficients in the region of steam qualities higher than of general interest to boiling-water reactors (above 50%) [46]. Fig. 10 shows the results of this work. These heat-transfer evalu-

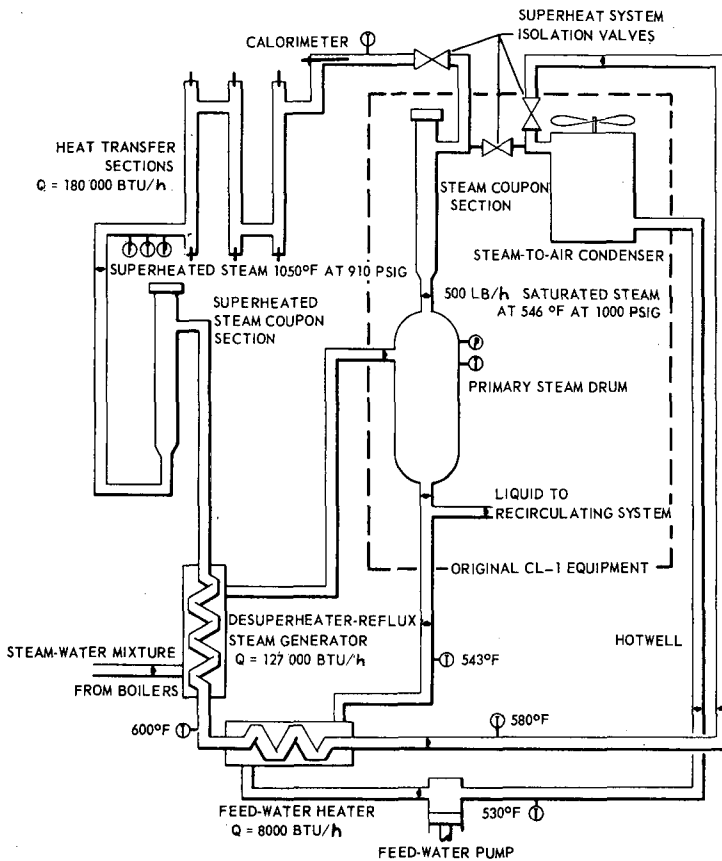


Fig. 9

Schematic diagram of the superheat corrosion test facility

(F) Flow (T) Temperature (P) Pressure

ations were the technical basis for the once-through reactor study referred to in the conceptual design studies mentioned previously.

The second phase of the experimental work consisted of the experimental determination of heat transfer coefficients in the regions of Reynolds numbers above 300 000. In these tests both circular and annular geometries were utilized. The results of these experiments and heat-transfer equation providing the best correlation of experimental results is shown in Fig. 11.

Mechanical development

The purpose of the mechanical development activity was to provide a basis of design information for specialized mechanical equipments which are necessary for the operation of superheated-steam-cooled reactor systems. This work area has involved primarily the development of the radial steam separator and mechanical seals [39, 40, 52].

TABLE V

SUMMARY OF COOLANT CHEMISTRY SURVEILLANCE

Sample points	Parameters to be measured
Superheat steam, individual exit lines (32)	Gross activity
Superheat steam, 8-in header*	Chloride
Superheat steam, 4-in divert header*	Conductivity
Saturated steam, EVESR inlet*	Dissolved oxygen
Saturated steam, EVESR individual fuel bundle inlets (4)	pH
Saturated steam, gas-fired boiler	Gas content
Water, reactor IX inlet	Dissolved impurities
Water, reactor IX outlet	Insoluble impurities
Water, EVESR hotwell	Dissolved radioisotopes
Water, EVESR condensate demineralizer effluent	Insoluble radioisotopes
Water, boiler feed (near boiler)	
Water, gas-fired boiler	
Water, make-up to EVESR hotwell	
Water, make-up to condensate storage tank	
Liquid poison	

* Special entrance tap to insure representative sample.

The work consisted of trying out several types of centrifugal separators. The most promising of these appeared to be the radial vane separator. Separation is accomplished by directing the 2-phase steam-water mixture from the reactor core onto a curved vane. The centrifugal field which results from flow on a concave surface causes the separation of the steam from the water.

Two air-water and one steam-water loops were used for this work. The larger loop is capable of water flows up to 5×10^6 lb/h and air flows to 15 000 lb/h. A 1000 lb/in² steam-water test loop is located at the Pacific Gas and Electric Company Moss Landing plant on Monterey Bay. This loop is capable of up to 600 000 lb/h of water and 60 000 lb/h steam at 1000 lb/in² abs.

Steam-water tests of a radial vane separator with a 130° vane arc, 4 in vane radius and 1/4 in $\times$ 18 in nozzle cross-section, showed the vane capacity to be 9500 ft³/h with an inlet steam quality of 6 to 12% by weight. The vane capacity is defined as the total volumetric flow rate per nozzle of per pair of vanes.

A full-circle radial vane separator having the vane and nozzle design of the two-vane separator mentioned above has also been tested. This unit has 36 vanes with only eighteen 18-in nozzles. Capacity limitations of the equipment limit the operation to eighteen 6-in nozzles or six 18-in nozzles.

TABLE VI
SUPERHEAT EXPOSURES IN CL-1
(Type 304 stainless steel)

Run No.	Dates	Exposure time (h)	Heater sheath location			Special run conditions
			Entrance 800-900°F*	Middle 900-1100°F	Exit 1100-1300°F*	
40	21/7 - 30/9/60	1000	E	9	10	No control of O ₂ and H ₂ gas in steam
41-42	24/10 - 20/11/60	950	P-3	P-6	P-5	Normal operation
43-45	22/11 - 30/3/61	2465	P-7	P-9	P-2	Normal operation
45C-46A	27/4 - 25/5/61	1000	P-11	P-8	P-12	Normal operation
46B	29/5 - 20/6/61	500	P-11	P-10	P-23	Normal operation
47	23/6 - 30/6/61	200	P-17	P-14	P-24	Special gas recombination study
48	1/7 - 25/8/61	1000	X9**	X3**	X33**	Start of alternate material study
49	1/9 - 22/9/61	495	Q-1	Q-4	Q-3	Start of stress chloride study - 1.5 ppm Cl ⁻ added to recirculating water - longitudinal stress placed on sheaths
49A	22/9 - 16/10/61	447	Q-2	Q-4	Q-3	Same
49B	23/10 - 15/11/61	433	Y-3	Q-1 (Reversed)	Q-2 (Reversed)	Same
50	22/11 - 2/12/61	210	Y-4	Y-5	Y-3 (Reversed)	Same
50B	2/12 - 14/12/61	238	Y-8	Y-5	Y-3 (Reversed)	Same
50D	18/12 - 31/12/61	242	Y-2	Y-5	Y-3 (Reversed)	Same

TABLE VI (cont'd)

Run No.	Dates	Exposure time (h)	Heater sheath location			Special run conditions
			Entrance 800-900° F*	Middle 900-1100° F*	Exit 1100-1300° F*	
51	4/1 - 29/1/62	534	Y-2	Y-5	Y-3 (Reversed)	Same plus cycle run ***
53	21/3 - 4/4/62	324	Y-10 (Sensitized)	Y-11 (Sensitized)	Y-9	Same plus cycle run
28(CL-4)	9/6 - 14/6/62	115		Y-12 (Sensitized)		Same plus cycle run Inconel in entrance and exit positions

* Calculated metal temperature

** Type 347 stainless steel.

*** Cycle run consists of part time low pressure (~ 75 lb/in²) low superheat ($\sim 10^\circ$ F), part time high pressure (~ 1000 lb/in²) low superheat and part time high pressure (1000 lb/in²) high superheat ($\sim 500^\circ$ F).

TABLE VII

SUPERHEAT EXPOSURES
Alternate materials

Facility	Run No.	Dates	Material	Exposure time (h)	Type of run	Remarks
CL-1	52	5/2 - 19/3/62	Inconel	1000	Normal*	
CL-1	53	21/3 - 4/4/62	Type 304 SS	324	Cycle**	Entrance and middle sheaths presensitized
CL-1	54	12/4 - 27/4/62	Inconel	345	Cycle	Entrance and middle sheaths presensitized
CL-1	55	5/5 - 19/5/62	Type 347 SS	327	Cycle	
CL-1	56	26/5 - 6/7/62	Incoloy	965	Normal	Entrance and middle sheaths presensitized
CL-1	57	16/7 - 27/8/62	Hastelloy X	988	Normal	
CL-4	26	9/6 - 14/6/62	Inconel + Type 304 SS	115	Cycle	Sensitized Type 304 SS for middle sheath failed 1st step of cycle; replaced by Inconel for 26A
	26A	14/6 - 26/6/62	Inconel	248		
CL-4	27	12/7 - 6/8/62	Incoloy	361	Cycle	
CL-4	28	14/8 - 2/9/62	Hastelloy X	361	Cycle	

* Normal- Longitudinal stress on sheath - no chlorides added - continuous operation.

** Cycle - Steam Temperature (°F)

Inlet saturated	Outlet superheat	Approximate (d)
350	360	4
546	550	3
546	1060	2
350	360	3
546	550	2

Steam-water tests using eighteen 6-in nozzles showed a vane capacity of 1900 ft³/h. Steam-water tests of the same unit using six 18-in nozzles showed a vane capacity of 5200 ft³/h. However, this was accompanied by poor carryunder performance (0.5 - 0.9% by weight steam) and an unstable water level.

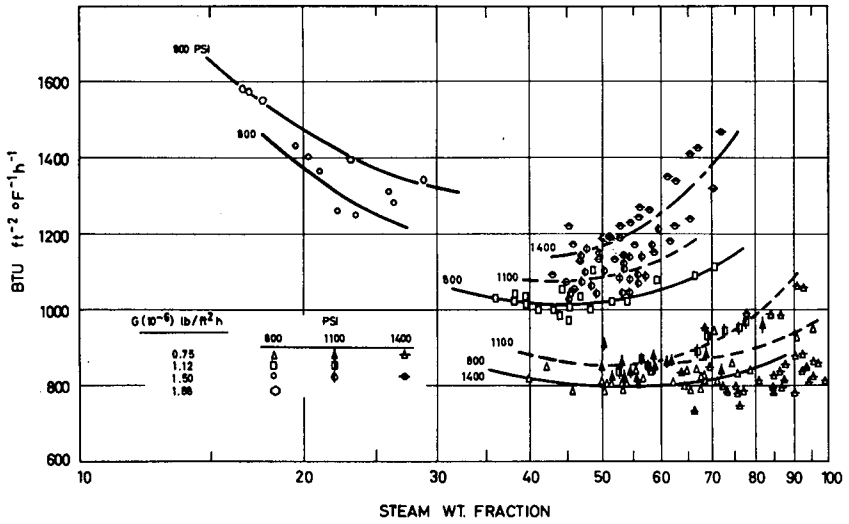


Fig.10

Film boiling heat transfer coefficients plotted against steam quality (unrestricted heater flow areas) for 0.120-in annulus

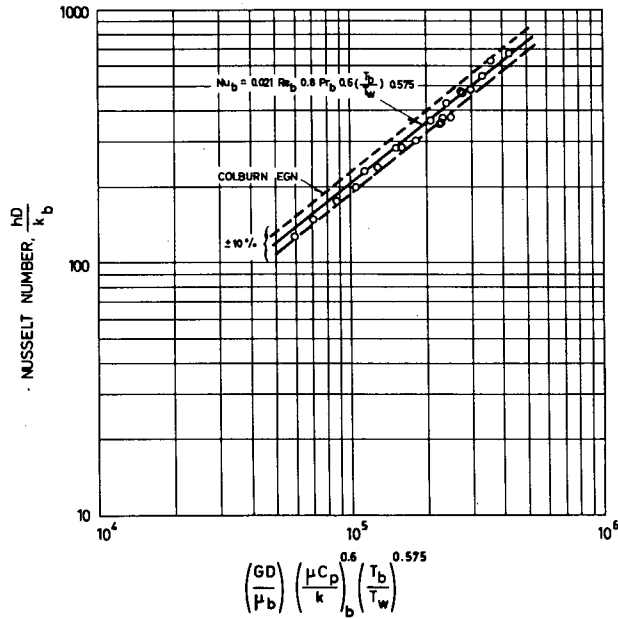


Fig.11

Heat transfer to superheated steam at 1000 lb/in²

The air-water tests of the separator using six 18-in nozzles showed good carryunder ($< 0.2\%$ by weight) performance and smooth separation. These tests showed a vane capacity at vane pressure and temperature of $7200 \text{ ft}^3/\text{h}$ with water level variation of 18 in. At $5200 \text{ ft}^3/\text{h}$ vane capacity, the water level could be varied 26 in.

Mixed-spectrum critical assembly (MSCA)

The major area of uncertainty in the mixed-spectrum superheat concept is in the complex nuclear physics of neutron flux dependence between the fast and thermal neutron spectrum regions. In particular, the safety of the system is dependent upon a negative Doppler coefficient. In addition, an important aspect of the economic incentive is the ability to improve the volumetric heat release of the MSSR reactor system as compared to boiling-water reactor systems. This will depend upon utilization of efficient buffer regions between the two regions of fast and thermal neutron fluxes which minimize local power peaking. The construction of the MSCA was started in April 1963 and will be completed during May 1963 [47]. The experimental work is expected to last about one year. It is expected that the MSCA will provide the first measurements on Doppler coefficient so that the programme results will be of value to not only the mixed spectrum reactor concept but also to other fast reactor studies.

The MSCA will be located in the present Critical Experiment Facility at the Vallecitos Atomic Laboratory. It will be a low-power (0-400 W) mock-up of the 75 MW(e) MSSR prototype. The core will consist of four distinct fuel regions as shown in Fig. 12. From the centre of the core outward, these regions are: (a) central unmoderated fast core; (b) annular unmoderated fast buffer; (c) annular moderated thermal buffer; and (d) annular moderated thermal core. An elevation of the MSCA is shown in Fig. 13.

The outer cylindrical wall in the region between the fast and thermal cores is the inner tank wall for the thermal core. The combination of this wall and the fast-core shroud separated by the $3/8$ -in gap will prevent water from entering the fast core if a leak should occur in the tank wall.

The location of the control elements in the MSCA can be seen in Fig. 13. In the thermal core, the eight boral safety sheets can be dropped into the core to scram the reactor. These eight sheets are used as safety sheets only, and thermal core control is provided by changing the moderator level in the thermal core tank. Interlocks are provided to prevent raising the safety sheets unless the thermal core is dry. The four boral coupling control sheets are used primarily to modify the power split between the fast and thermal cores, but they can also be scrambled to provide additional shut-down capability.

In the fast core, six of the fuel assemblies are mounted on control drives to provide fast-core control. Four of the assemblies are provided with B_4C followers and act as safety scram controls. The remaining two assemblies have Inconel followers and are used as shim controls.

There are 16 fused fuel assemblies. The purpose of the fuse is to shut down the reactor during an excursion if the excursion is not stopped by instrument scram or by other means. The fuse consists essentially of a

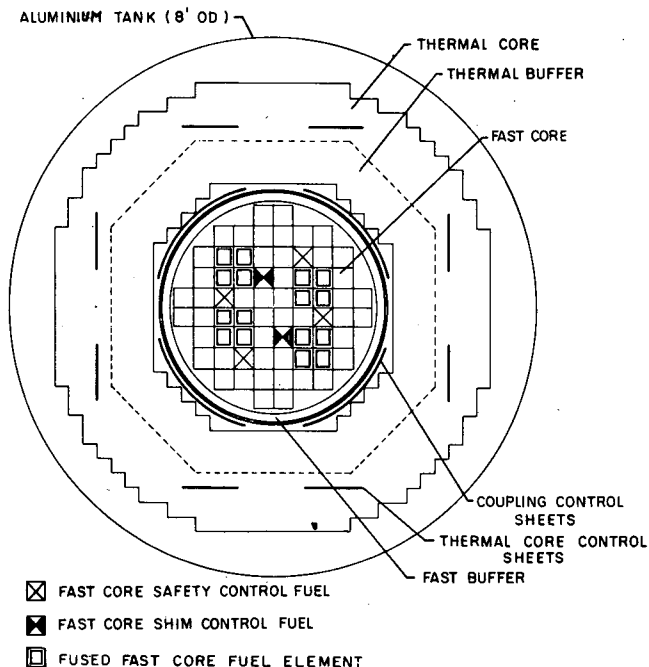


Fig.12

Core cross-section of the MSSR critical experiment facility

highly enriched metallic uranium pin used to support the full weight of a fuel assembly.

The fast buffer region consists of three rings of closely spaced fuel rods placed around the periphery of the fast core. These fuel rods are 1.3% enriched UO_2 pellets loaded into 9/16-in outer diam., 0.028-in wall, 304 stainless-steel tubing. Stainless-steel end plugs are welded to the cladding to form the 9-in thick reflector regions comparable to the fast core reflectors. The fuelled length (30 in) of the buffer is the same as and is located in line with the fast-core fuel region.

The fast-core fuel enrichment necessary to give a desirable power split has been calculated to be about 21% in U^{235} . To provide for uncertainties in the calculation, the standard fuel rods are provided in two enrichments, 26.5% and natural. The fast core will be loaded with a uniform mixture of these fuel rods to give an effective enrichment of 21%. Sufficient quantities of fuel rods are provided so that the effective fast-core loading can be varied above and below the predicted loading.

The following is a brief description of the planned experimental programme. The thermal core will first be loaded to criticality with only the fast-buffer fuel in the fast section. The fast-core fuel will then be loaded gradually, proceeding in steps from a thermal reactor to the mixed-spectrum reactor design loading. The neutron lifetime will be measured at intermediate stages to assure that the neutron lifetime is not less than $5\mu\text{s}$.

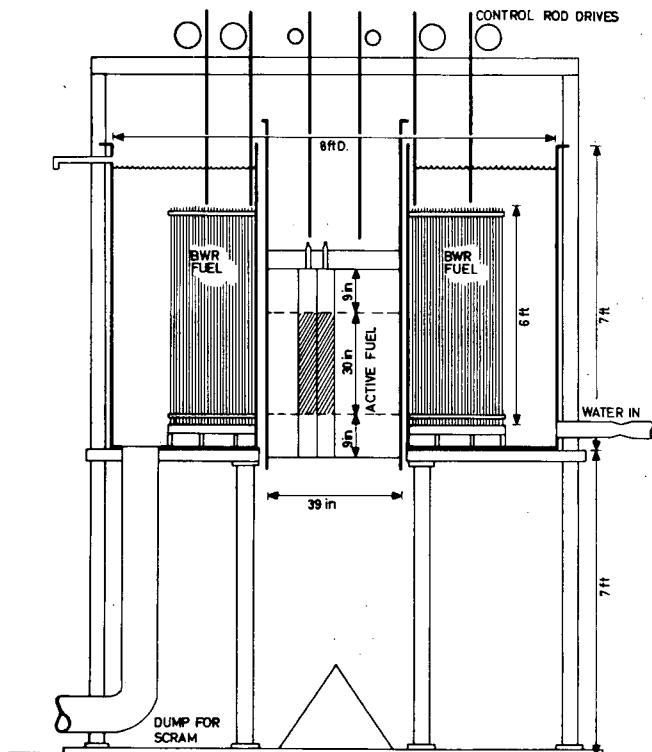


Fig.13

Sketch showing elevation of the MSSR critical experiment facility

The principal experimental measurements aimed at proving the feasibility of the MSSR concept include power distribution, Doppler effect, flooding effects, distribution of reactivity, control-rod worths and the effect of the control system on the power distribution.

6. AEC-EVESR NUCLEAR SUPERHEAT FUEL DEVELOPMENT PROJECT

The AEC-EVESR Nuclear Superheat Fuel Development Project [34, 35, 36] is a 4-yr development programme sponsored by the USAEC which is specifically aimed at the development of nuclear superheat fuel which, when applied to large power reactors, would produce electric power which is competitive with power produced in conventional fossil-fuelled plants. The programme, which was initiated in May 1962, consists of associated research and development including the design, fabrication, irradiation and post-irradiation examinations of the 32 fuel assemblies which constitute the initial load of the EVESR reactor, and approximately 20 advanced fuel assemblies which will be irradiated during the later phases of the programme. The specific objective of the programme is to establish performance limits

of superheat fuel elements. Since the EVESR reactor has not yet operated, there are no superheat fuel performance results available. The work completed as of this time consists primarily of the design and fabrication of the initial fuel loading, design and installation of the special fuel-performance evaluation instrumentation and coolant-chemistry instrumentation systems, and completion of the digital codes for fuel-performance analysis, reactor operating data handling and data retrieval.

Fuel-performance instrument system and evaluation methods

The two prime variables to be studied in the EVESR Fuel Development Program are (1) variations in fuel cladding material, with initial selection composed of superheat fuel clad with Inconel, Incoloy, Type 310 vacuum melt, Type 304 vacuum melt, and commercial Type 304 stainless steel; and (2) variable stress levels introduced by the combination of operating heat flux and clad surface temperature. Heat flux may be varied by the combination of core position and enrichment at a given reactor power level. Surface temperature at a given heat flux may be varied by adjusting the steam-flow rate. Steam flow will be adjusted within a bundle by fixed orifices. Total bundle flow may be adjusted by individual bundle or by multiple bundles, depending upon the location in terms of individual or multiple-bundle manifold system.

Fig. 14 shows the range of test conditions which are proposed to be maintained on the initial EVESR core loading and the implementation for the two major test variables.

There are thirty-two fuel bundles in the EVESR core, each of which contains nine individual fuel elements for a total of 288 elements. The development programme objectives require that certain stress levels and peak clad temperatures be maintained for each bundle and that these two parameters be varied throughout the core for the various fuel cladding materials in order to provide a maximum of fuel development information. The stress level in the cladding depends on the as-built clearances between the fuel and the cladding plus the existing combination of heat flux and cladding temperature in the individual fuel element.

The individual element heat fluxes will depend on the bundle power which in turn is subject to whatever power level exists in the core as well as the relative distribution of the power between bundles. Although it is desirable from a development test standpoint to maintain a constant total reactor power and a fixed relative power distribution axially and radially in the core, the individual bundles and fuel elements in the bundle will be subject to start-up and shut-down cycles and to shifts in power distribution resulting from rod movements and fuel burn-up. The planned test procedure will consist of having the EVESR reactor operator maintain as nearly as practical a constant rated total reactor power level while the test operator regulates the flow rates to the individual fuel bundles in order to produce the desired maximum cladding temperatures.

Fig. 15 shows a section through the reactor pressure vessel wall and part of the core consisting of one bundle and the associated piping and development test instrumentation. Note that the temperature, pressure, and flow is measured in the bundle piping extending outside the reactor vessel. The

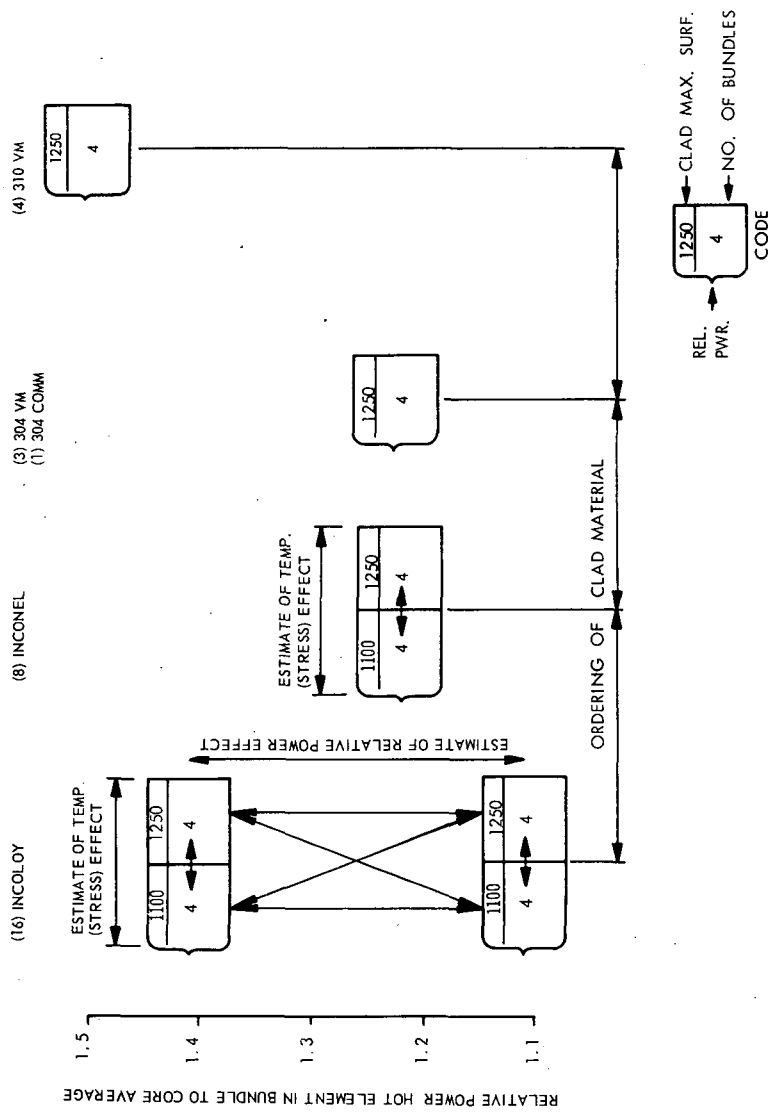


Fig. 14
Theoretical flow diagram of possible performance comparisons

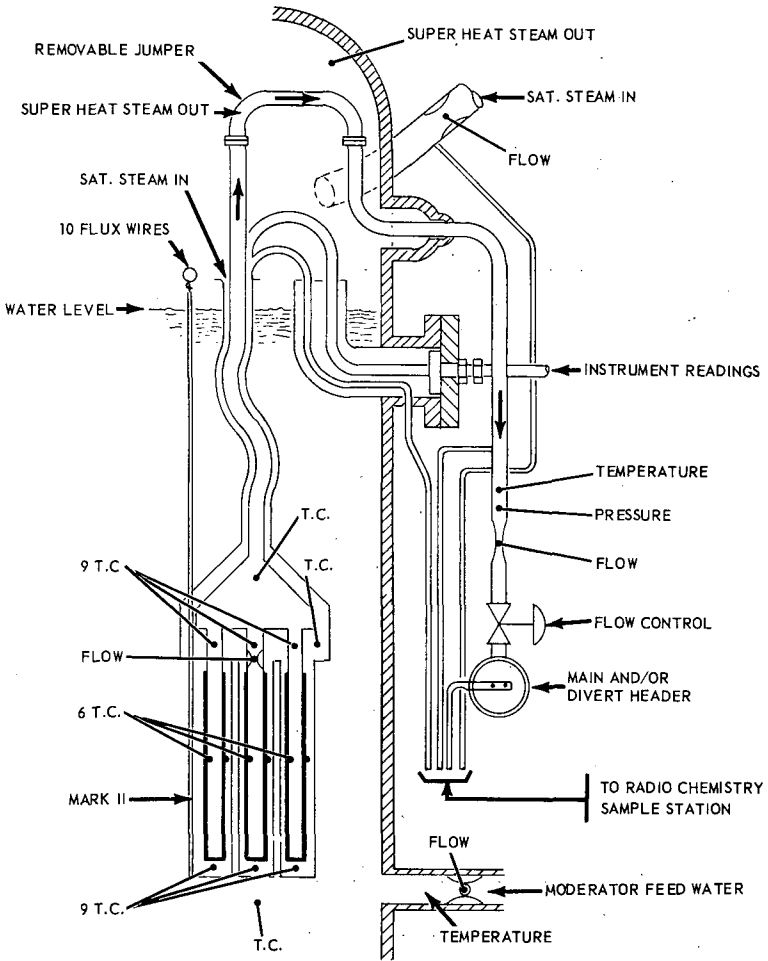


Fig. 15

Schematic of EVESR R and D instrumentation

steam entering the bundles comes from the steam dome in the reactor pressure vessel.

With the bundle flow rate, the exit enthalpy, and the inlet enthalpy measured it is possible to establish the overall bundle power except for heat losses to the moderator and the pressure and heat losses from the bundle inlet and exit piping. An analysis has been developed for this heat transfer system. The control-room recorded readings of flows, pressures, and temperatures are corrected analytically for pressure and heat losses back to the bundle outlet. The local heat generation in the nine fuel elements in each bundle will be obtained from nuclear calculations compensated to agree with the data from the critical test facility and the ten flux wire monitors in the five typical bundle locations in the EVESR core. With the distribution of

the heat generation set, the moderator heating established from appropriate feedwater and moderator measurements, and the flow and bundle exit enthalpy established, a complete solution of the cladding and fuel temperatures is possible for the fuel assembly.

This programme and the analysis procedure for the Philco 2000 computer will be used initially to construct operating charts which will guide the test operator in setting the proper flows in the bundles and will be used later in the test programme to monitor day-to-day performance variations.

Except for the flux wire monitors the procedure for testing EVESR fuel as outlined above involved no in-core instrumentation. Since the computer analysis is subject to analysis discrepancies and accumulated errors, one-quarter of the core has been highly instrumented to provide detailed test checks against the machine calculations.

Fig. 16 shows how the flow-control valves for all of the thirty-two bundles are distributed for the core. It will be noted that in some instances as many as four bundles are controlled by a single bias flow control valve. If the test programme requires different flow rates in the four bundles controlled by a single bias control valve, it will be necessary to orifice them differentially to produce the desired flows.

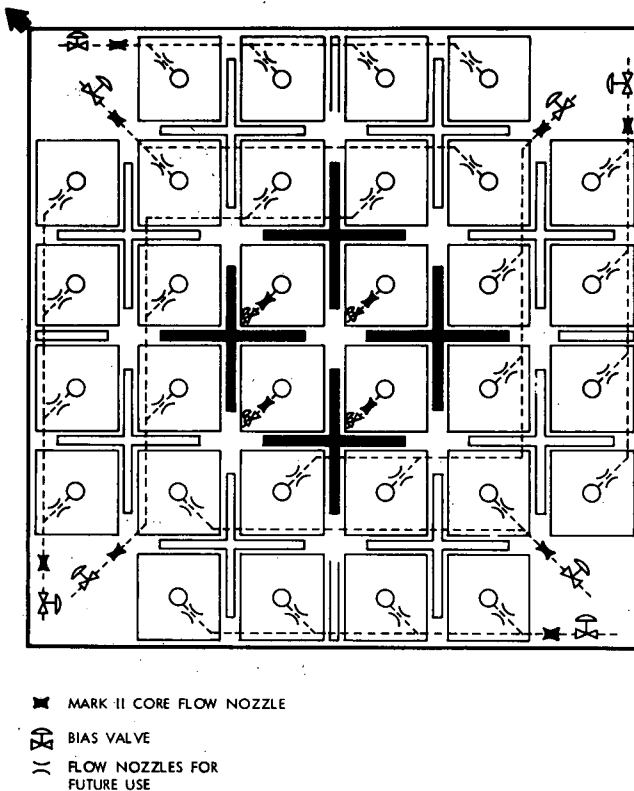


Fig. 16

EVESR exit steam manifold arrangement

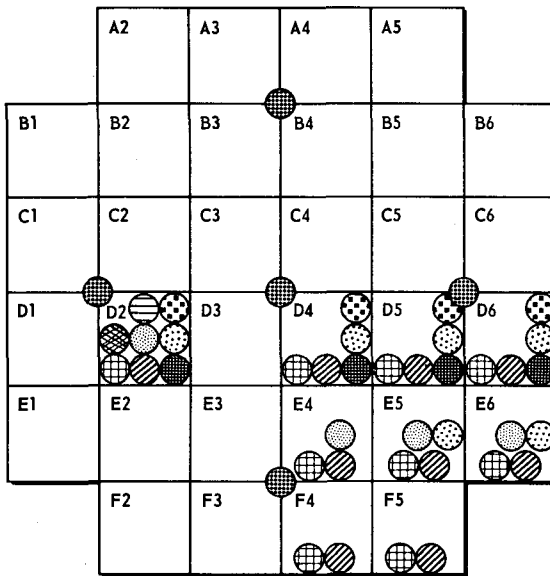
The thirty-two fuel assembly flows, pressures, and temperatures are measured in the individual piping extending outside the reactor pressure vessel as shown in Fig. 15. These control room recorded readings are corrected analytically for pressure and heat losses back to the bundle outlet. The local heat generation in the nine fuel elements in each of the fuel assemblies will be obtained from nuclear calculations compensated to agree with the data from the critical test facility and the ten flux wire monitors in the five typical bundle locations in the EVESR core. With the distribution of the heat generation set, the moderator heating established from appropriate feedwater and moderator measurements, and the flow and exit enthalpy established from the readings in the external piping, a complete solution of the cladding and fuel temperatures is possible for the fuel assembly. The flow-control valves operated from the control room can be used to set up individual test conditions in the various fuel assemblies.

The above data reduction procedure would permit a complete evaluation of bundle performance with essentially no in-core instrumentation. To be of value the accompanying analytical techniques must be checked out thoroughly and corrected where required to improve accuracy. To enable such a check to be made, extensive in-core instrumentation is planned for the EVESR core. There is one fully instrumented assembly in the core. Included in the fully instrumented fuel assembly are:

- (a) Ten flux wire monitors for determining the axial and radial distribution of the neutron flux within the assembly.
- (b) Nine mid-pass and nine exit steam temperature readings to permit a closer study of the power distribution between elements in the fuel assembly and to check the heat split between the inner and outer fuel claddings on all nine elements.
- (c) The temperature in the inlet plenum is checked to determine the degree of superheating obtained in the downcomer.
- (d) The temperature in the exit steam plenum will be used to check the analytical prediction based on corrected readings from the external piping.
- (e) Six fuel-cladding temperatures will be measured in this one assembly. This will require direct measurement of the cladding temperature in-pile.
- (f) Three of the typical element positions in the fuel assembly will be checked for flow split between elements.
- (g) A thermocouple will be attached to the bottom of the fuel assembly to measure the moderator-inlet temperature to the bundle.

There are eight other bundles with lesser amounts of instrumentation of greater reliability which will be arranged as shown in Fig. 17. The codes key the instruments in the fully instrumented fuel assembly to each of the eight assemblies. These eight fuel assemblies are located strategically in the quarter core symmetry. Some of the duplications in instrumentation are intentional to provide cross-checks on readings and to allow for spares in case of fuel failure.

As shown in Fig. 15, there are flow meters installed in the moderator fuel water and the reactor inlet steam. Not shown are flow meters and thermocouples in the main and divert headers. All of this instrumentation



-  FUEL ELEMENT STEAM EXIT TEMPERATURE, 9
-  FUEL ELEMENT STEAM TEMP. BETWEEN PASSES, 9
-  FLUX WIRES, 6
-  WATER TEMPERATURE, 4
-  GROSS STEAM INLET TEMPERATURE, 4
-  GROSS STEAM OUTLET TEMPERATURE, 4
-  MIDPOINT ON THE PASSES, 1
-  ELEMENT FLOW RATE, 1
-  RADIO CHEMISTRY, 32
-  IN-CORE FLUX MONITORS, 5

Fig. 17

Schematic of EVESR instrumentation in plan view

permits gross heat and mass balances to be made which, in turn, can be checked against the data from the thirty-two fuel assemblies.

In Fig. 15, the principal radiochemistry sample points around the reactor are shown. These are as follows:

- (a) Four fuel-assembly inlet samples to determine if centrally-located fuel assemblies suffer from more or less moderator water carryover than fringe-located fuel assemblies.

- (b) Exit steam samples from each of the thirty-two fuel assemblies will be drawn from the individual external steam pipes.
- (c) Steam samples are also drawn from the incoming saturated steam and the mixed outlet steam in the main and divert headers.

Reactor water chemistry control and surveillance

The water chemical control and surveillance programmes are divided into two categories: (1) routine process control, which comprises chemical measurements of water environments essential to the safe operation of the reactor and boiler and to the interpretation of the experimental studies performed in the reactor, and (2) research and development programmes, which are designed to answer specific questions of major importance to the development of successful superheat reactors. Specifically, the objectives of these programmes are as follows:

- (a) To monitor various chemical parameters throughout the plant in order to provide protection against undue corrosion, to assess the operation of plant equipment, to assure compliance with established safety criteria, and to provide background information essential for the interpretation of research and development studies.
- (b) To detect and locate defective fuel bundles and study the characteristics of fission product release from defective superheat fuel.
- (c) To develop an effective simplified system for the location of defective fuel in superheat reactors.
- (d) To determine the extent and character of radioactive contamination in the steam and condensate systems, and to study mechanisms of contamination build-up in superheat reactor systems as a function of fuel failure mode, burn-up and operating history.

The equipment required to accomplish the above objectives includes sample lines and analysis equipment for a variety of plant parameters as shown in Table IV. Not all of the parameters would be measured on each sample line; however, the sample system would be designed for a fair degree of flexibility so that many sample lines can be analysed for the important process parameters. This is considered to be essential since the controlling factors which will limit superheat fuel performance have not yet been identified.

Initial loading physics test programme

It is planned to load the core to criticality with fuel having unflooded superheat passages. After making a temperature coefficient measurement, the loading will continue to the full core size (32 bundles). With the full core loading at room temperature, control-rod calibrations and scram rod worth, stainless-steel fuel-channel worth and temperature coefficient measurements will be made.

Similar measurements will then be performed for the flooded core, beginning with a minimum critical loading. After completing the flooded, head-off measurements, the pressure vessel head will be secured in place. By the use of boiler steam the core water will be heated in steps from 68 to

545°F. The control-rod position for a just critical full core will be determined as a function of moderator temperature. Also, the minimum control-rod shut-down margin will be demonstrated in the flooded condition where the reactivity is shown to be the greatest. Control-rod calibrations will be made at two elevated temperatures (~400 and 545°F) after which the superheat passages will be unflooded by the use of boiler steam. Control-rod calibrations will be repeated for the unflooded core at two elevated temperatures. Maintaining steam flow, critical controls positions will be determined as the core temperature is reduced from 545°F to a minimum of about 300°F.

Upon completion of the zero-power testing, low-power testing will be used initially to calibrate power-range instruments against thermal data. Then, an attempt will be made to measure void and fuel Doppler reactivity for several power and steam flow combinations. Finally, power distribution mapping and measurements of xenon transients will be made.

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DISCUSSION

E. PARKER: Do you think that the future of boiling-water reactors is dependent on the success of nuclear superheat?

R.T. PENNINGTON: Not really. Superheat plants, after all, merely complement boiling-water systems and are not essential to their economic utilization. The development of superheat can clearly be of direct benefit to boiling-water reactors if it results in an improved technology, as it certainly will, but it is generally agreed that nuclear superheat should not be pursued unless it can be shown that it will eventually produce power at lower cost.

E. PARKER: Thank you. I have, however, another question. A tremendous effort is being made throughout the world to develop fast, plutonium-fuelled reactors, and these will have to be supplied with fuel by good converter reactors - a fact which was stressed in the report addressed by the United States Atomic Energy Commission (USAEC) to President Kennedy. Good converters are reactors which not only produce electricity cheaply but also produce a large amount of plutonium per annum per MW(e) installed. Taking into account the characteristics required of the fuel and the anticipated yield, it would seem that superheat reactors could not be good converters. In fact, they appear to be less valuable as converters than present-day boiling-water reactors, especially those having Zircaloy cladding. A simple calculation is

enough to show that a boiling-water reactor such as KRB in Grundremmingen requires as long as 15 to 20 yr - depending on the assumptions one makes - to produce the plutonium needed by a fast reactor of the same power. A superheat reactor would require something of the order of 25 or 30 yr, depending on the type of fuel and the yield obtained. Thus it is hard to see how superheat reactors can have any place in a co-ordinated programme for the development of plutonium breeders. It would be interesting to hear your comments on this "programmatic" aspect of the superheat problem.

R. T. PENNINGTON: It is perfectly true that at the present stage of development superheat reactors have higher cladding/fuel and water/fuel ratios than boiling-water systems with zirconium-clad fuel; the conversion ratio is therefore poorer and they are bound to produce less plutonium. It should be remembered, however, that nuclear superheat is in an early stage of development, and one should not assume that the poor conversion ratio is an inherent defect that cannot be overcome. Future designs will tend to emphasize those features - including a high conversion ratio and high efficiency - which will make superheat reactors economically more attractive.

C. A. PURSEL: Mr. Parker has raised a very interesting question: the problem of striking a balance between immediate commercial advantage and long-term utilization of natural resources. The superheat reactor admittedly has a poor conversion ratio; on the other hand the price of enriched uranium and plutonium has decreased over the past few years, and this fact automatically makes the high conversion ratio less vital economically - less essential, at least, than it appeared to be a few years ago. There is no incentive to save neutrons that do not cost much, or to produce plutonium that is not worth much. One of our immediate goals is to develop a commercially self-sufficient nuclear industry, based on existing fuel-cycle economics, such as will be able to bear a large share of the development costs for future reactors. In the long run, however, the price of U^{235} and plutonium is bound to increase as they become more rare, and the breeder will then clearly come into its own. The long-term programme must therefore aim at bringing a commercially acceptable breeder onto the market before that time comes.

U. ZELBSTEIN: Let me point out first that any comparison of the cost of a nuclear installation with that of fossil-fuelled plants should take into account possible cost developments in the latter. That is merely a general comment. Then I should like to know whether any experiments with plutonium fuel elements are anticipated within the framework of the existing superheat programme.

R. T. PENNINGTON: No work with plutonium is planned as part of the USAEC superheat programme as plutonium technology is being dealt with under other programmes - at Hanford and the Argonne National Laboratory.

If I may return for a moment to the previous topic of discussion, economic studies have shown that the superheat reactor, if successfully developed, will be able to produce power more cheaply than nuclear saturated-steam systems. It is really impossible, however, to make broad generalizations regarding the cost of different systems. Each new plant must be evaluated and costed separately, account being taken of its size, location and the way in which it is to be financed.

P. EDDY: Perhaps I could amplify some of the comments that have been made. It has already been pointed out that the boiling-water programme is not at all dependent on the superheat programme, though it certainly stands to profit by superheat studies. Under the superheat programme the fuel will be subjected to very stringent tests; we will learn a great deal about materials, and all this is bound to assist the development of fuel for boiling-water reactors.

As to the fast reactor programme, you may know that a fast ceramic reactor is to be built for testing purposes in Arkansas. It will be backed by a group of utilities in the southwestern United States and by the USAEC, and will also receive some support from Karlsruhe and the General Electric Company. Its purpose is to demonstrate that a fast ceramic reactor using plutonium fuel is feasible, and that it will work - or so we hope - as a partner with the boiling-water or light-water reactors in establishing a realistic price for plutonium. Once the fast ceramic reactor has been demonstrated, we believe that it will in fact establish a price for plutonium somewhat above the USAEC's current support price, and that by the 1970's this partnership will result in better fuel-cycle economics for both the boiling-water reactor and the fast ceramic reactor.

II

EXPERIENCE WITH SPECIFIC NUCLEAR POWER PLANTS

POST-CONSTRUCTION TESTING OF THE ELK RIVER, HALLAM AND PIQUA POWER REACTOR PLANTS

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Abstract — Résumé — Аннотация — Resumen

POST-CONSTRUCTION TESTING OF THE ELK RIVER, HALLAM, AND PIQUA POWER REACTOR PLANTS. Actual experience gained during the post-construction testing of three nuclear power plants, under the USAEC Power Reactor Demonstration Program, may permit some generalizations concerning this phase of plant construction and operation.

The three plants, Elk River Reactor (ERR), Hallam Nuclear Power Facility (HNPF), and the Piqua Nuclear Power Facility (PNPF), represent three different reactor concepts: natural-circulation boiling water, sodium-graphite, and organic cooled and moderated, respectively.

The post-construction testing period included the time between the end of construction (erection of structures and installation of equipment) and the beginning of power operation (generation of significant net electrical power). The tests were intended to: (a) verify the performance characteristics of the as-installed equipment; (b) obtain initial criticality and reactivity coefficient measurements; and (c) determine reactor physics and plant performance characteristics at a sequence of increasing power levels.

The experience gained can be reported in six separate but interrelated categories: (1) schedule; (2) costs; (3) staffing requirements; (4) procedures; (5) equipment performance (including malfunctions); and (6) actual, as compared to predicted, system performance characteristics.

The average project staffing, including craftsmen, operators, supervisors, technical support and trainees, was approximately 50 for ERR, 115 for HNPF, and 60 for PNPF.

Detailed written Pre-operational Test Procedures were prepared for each major component and system. To the maximum possible extent, all tests were performed before fuel loading and operation of the integrated plant.

Authorization procedures (duplicates of the licensing procedures for non-USAEC-owned plants) were in progress during almost all of the post-construction testing periods.

The time required for post-construction testing of each of these plants significantly exceeded the original estimates. The tests disclosed numerous, observed or suspected, deficiencies or malfunctions of components which led to additional testing and analyses. In some instances, repair or modification of components was necessary to correct fabrication or engineering errors.

Major problem areas are discussed:

Elk River Reactor. Discovery of cracks in portions of the reactor vessel surface cladding led to extensive investigations and analyses and required some repairs and vessel modifications. Insufficient steam separation capacity required replacement and modification of some reactor vessel internal hardware.

Hallam Nuclear Power Facility. Entrainment of the helium cover gas led to modifications of the secondary sodium loops. Failure of a tube in the intermediate (sodium to sodium) heat exchanger led to analyses to determine the cause of failure followed by removal and repair of the heat exchanger.

Piqua Nuclear Power Facility. Chemical cleaning of the piping system damaged several valves which required mere repair or replacement. Leaks in the organic coolant and steam tracing systems caused repeated delays.

After completion of the necessary repairs and modifications, the actual performance characteristics of each of the three reactors closely matched design predictions.

ESSAIS APRÈS CONSTRUCTION DES CENTRALES NUCLÉAIRES D'ELK RIVER, DE HALLAM ET DE PIQUA. Les résultats des essais après construction de trois centrales nucléaires, dans le cadre du programme de démonstration des centrales nucléaires de la Commission de l'énergie atomique des Etats-Unis (CEA-EU), permettront peut-être de faire certaines généralisations concernant cette phase de la construction et de l'exploitation des centrales.

Ces trois centrales, le réacteur de puissance d'Elk River (ERR), la centrale nucléaire de Hallam (HNPF), et la centrale nucléaire de Piqua (PNPF), appartiennent à trois filières différentes: réacteur à eau bouillante à circulation naturelle, réacteur à graphite et à sodium et réacteur ralenti et refroidi par un fluide organique.

La période des essais après construction a commencé à la fin de la construction (érection des bâtiments et installation du matériel) et s'est terminée au début du fonctionnement en puissance (production nette de courant électrique appréciable). Les essais avaient pour but: a) de vérifier les performances du matériel ainsi installé; b) de mesurer les valeurs initiales des coefficients d'état critique et de réactivité; c) de déterminer les paramètres de physique des réacteurs et les performances de l'installation à différents niveaux de puissance progressivement croissants.

L'expérience acquise peut être décrite sous six rubriques distinctes mais apparentées: 1. calendrier; 2. coûts; 3. besoins en personnel; 4. méthodes; 5. performances du matériel (y compris le mauvais fonctionnement); 6. performances réelles de la filière par rapport aux prévisions.

Les effectifs moyens pour chaque installation, y compris les ouvriers, les opérateurs, les cadres supérieurs, le personnel technique d'appui et les stagiaires sont d'environ 50 personnes pour l'ERR, 115 pour l'HNPF, et 60 pour le PNPF.

Des listes détaillées d'essais à effectuer avant fonctionnement ont été préparées pour chaque pièce et partie constitutive importante. Autant que possible on a procédé à tous les essais avant de charger le réacteur et de mettre en route l'ensemble de l'installation.

Les demandes d'autorisations (correspondant aux demandes de permis pour les installations qui n'appartiennent pas à la CEA-EU) ont été instruites pendant presque toute la période des essais après construction.

Le temps nécessaire pour les essais après construction de chacune de ces centrales a été nettement plus long que prévu. Les essais ont mis en évidence de nombreux défauts ou vices de construction des pièces, constatés ou supposés, qui ont nécessité des essais et des analyses supplémentaires. Dans certains cas, il a fallu réparer ou modifier des pièces pour porter remède à des défauts de fabrication ou des erreurs techniques.

Les principaux défauts relevés ont été les suivants:

Réacteur de puissance d'Elk River. On a découvert des craquelures dans certaines parties du revêtement du caisson du réacteur. Il a fallu faire des recherches et des analyses poussées ainsi que des réparations et apporter des modifications à la cuve. La capacité de séparation de la vapeur était insuffisante; il a fallu remplacer et modifier certaines parties du circuit à l'intérieur du caisson.

Centrale nucléaire de Hallam. On a constaté que de l'hélium était entrainé dans le sodium, et il a fallu modifier le circuit secondaire du sodium. On a découvert une défaillance d'un tube dans l'échangeur de chaleur intermédiaire (sodium-sodium); il a fallu procéder à des analyses pour en rechercher la cause, et démonter l'échangeur pour le réparer.

Centrale nucléaire de Piqua. En nettoyant les canalisations avec des produits chimiques, on a endommagé plusieurs vannes et il a fallu soit les réparer soit les remplacer. Des fuites dans le circuit du fluide de refroidissement organique et dans le circuit de réchauffage par circulation de vapeur ont provoqué des retards répétés.

Une fois terminées les réparations et apportées les modifications nécessaires, les performances de chacun des trois réacteurs concordaient étroitement avec les prédictions des plans.

ПРЕДЭКСПЛУАТАЦИОННОЕ ИСПЫТАНИЕ ЭЛК-РИВЕРСКОЙ, ХЭЛЛЕМСКОЙ И ПИКУАСКОЙ ЭНЕРГЕТИЧЕСКИХ РЕАКТОРНЫХ УСТАНОВОК. Фактический опыт, накопленный во время предэксплуатационных испытаний трех ядерных энергетических установок, построенных по демонстрационной программе энергетических реакторов Комиссии по атомной энергии Соединенных Штатов, позволяет сделать некоторые обобщения в отношении этой фазы строительства и эксплуатации установок.

Три установки, а именно Элк-риверский реактор (ERR), Хэллемская ядерная энергетическая установка (HNPF) и Пикюаская ядерная энергетическая установка (PNPF), представляют три различных типа реакторов: реактор с кипящей водой с естественной циркуляцией, натриево-графитовый реактор и реактор с органическим теплоносителем и замедлителем соответственно.

Период предэксплуатационных испытаний охватывает время между окончанием строительства (сооружение зданий и установка оборудования) и началом эксплуатации на мощности (производство значительного количества электрической энергии нетто). Испытания преследовали следующие цели: 1) проверить рабочие характеристики установленного оборудования; 2) определить начальную критичность и коэффициент реактивности и 3) определить физические характеристики реактора и рабочие характеристики установки по мере увеличения мощности.

Накопленный опыт может быть изложен в шести отдельных, но взаимосвязанных разделах: 1) программа; 2) стоимость; 3) потребности в кадрах; 4) технология проведения испытаний; 5) работа

оборудования (включая неполадки) и 6) фактические (по сравнению с расчетными) характеристики работы системы.

Средняя проектная численность персонала, включая рабочих, операторов, дозиметристов, вспомогательный технический персонал и стажеров, составляет: для ERR - 50, для HNPF - 115 и для PNPF - 60 человек.

Подробная технология предэксплуатационных испытаний в письменном виде была подготовлена для каждого важного компонента и каждой системы установки. Насколько это было возможно, все испытания были проведены до загрузки топлива и начала эксплуатации всей установки.

Оформление разрешений на эксплуатацию установок (дубликаты лицензий на непринадлежащие КАЗ США установки) заняло почти все время проведения предэксплуатационных испытаний.

Время, потребовавшееся для проведения предэксплуатационных испытаний каждой из этих установок, значительно превысило первоначально намеченное время. Испытания вскрыли ряд явных и предполагаемых неисправностей компонентов, в результате чего потребовалось провести дополнительные испытания и анализы. В ряде случаев для исправления ошибок, допущенных при изготовлении или конструировании оборудования, компоненты пришлось ремонтировать или видоизменять.

Основными неполадками были:

Элк-риверский реактор. В некоторых частях бака реактора обнаружены трещины, что потребовало обширных исследований и анализов, а также некоторого ремонта и модификации бака. Недостаточная производительность парогенератора потребовала замены и модификации некоторых металлических деталей бака реактора.

Халлемская ядерная энергетическая установка. Увлечение защитного газа гелия привело к необходимости модифицировать вторичные натриевые контуры. Повреждение трубопровода в промежуточном (натриево-натриевом) теплообменнике потребовало проведения анализов для определения причин повреждения, в след за чем последовало извлечение и ремонт теплообменника.

Пикуасская ядерная энергетическая установка. При химической очистке системы трубопроводов повреждено несколько вентилях, что потребовало их ремонта или замены. Утечки в системе органического теплоносителя и в системе индикации пара привели к повторным задержкам.

После окончания необходимого ремонта и внесения изменений фактические рабочие характеристики каждого из трех реакторов приблизились вплотную к проектным расчетам.

ENSAYOS POSTERIORES A LA CONSTRUCCIÓN DE LAS CENTRALES NUCLEOÉLECTRICAS DE ELK RIVER, HALLAM Y PIQUA. La experiencia adquirida directamente en los ensayos efectuados después de terminada la construcción de las tres centrales nucleoelectricas del programa de demostración de reactores de potencia de la Comisión de Energía Atómica de los Estados Unidos permite deducir ciertos principios generales acerca de esta fase de la construcción y explotación de las centrales.

Las tres instalaciones, a saber, el reactor de Elk River (ERR), la central nucleoelectrica de Hallam (HNPF) y la central nucleoelectrica de Piqua (PNPF), representan tres conceptos diferentes en materia de reactores: el de agua hirviendo con circulación natural, el de sodio-grafito y el de moderador y refrigerante orgánicos, respectivamente.

El período de ensayos comenzó al concluirse la obra (construcción de las estructuras e instalación de los equipos) y se terminó con la iniciación del desarrollo de potencia (producción neta de una cantidad apreciable de energía eléctrica). Los ensayos tenían por finalidad: a) verificar las características de funcionamiento del equipo instalado, b) determinar las condiciones de criticidad y el coeficiente de reactividad y c) determinar los parámetros físicos del reactor y las características de rendimiento de la central para una serie creciente de valores de la potencia.

La experiencia adquirida puede agruparse en seis categorías de datos distintos pero relacionados entre sí: 1) calendario, 2) costos, 3) necesidades de personal, 4) métodos, 5) comportamiento del equipo (incluso defectos) y 6) características reales de rendimiento del sistema y comparación con los valores calculados.

En promedio, la plantilla de cada instalación, incluyendo operarios, operadores, jefes, técnicos auxiliares y aprendices, comprende unas 50 personas para el ERR, 115 para el HNPF y 60 para el PNPF.

Para cada sección y circuito importante se prepararon normas detalladas de ensayos previos. Dentro de lo posible, todas las pruebas se realizaron antes de cargar el combustible y de iniciar la explotación de la central incorporada a la red.

Los trámites de autorización (idénticos a los de concesión de permisos para las centrales no dependientes de la USAEC) siguieron su curso durante casi todo el período de ensayos posteriores a la construcción.

El tiempo exigido por los ensayos mencionados en cada una de las centrales excedió apreciablemente del

calculado en un principio. La pruebas pusieron en evidencia numerosos defectos, ya comprobados o supuestos, de construcción o de funcionamiento de ciertos elementos, que obligaron a realizar nuevos ensayos y estudios. En algunos casos, hubo que reparar o modificar partes de la instalación para subsanar defectos de fabricación o errores de cálculo.

He aquí los principales defectos hallados:

Reactor de Elk River. Se descubrieron grietas en parte del revestimiento superficial del recipiente del reactor; ello obligó a efectuar una serie de investigaciones y análisis, así como ciertas reparaciones y modificaciones del recipiente. La insuficiente capacidad de separación de vapor obligó a sustituir y modificar algunas piezas metálicas en el interior del recipiente del reactor.

Central nucleoelectrónica de Hallam. Debido al arrastre de helio, hubo que modificar los circuitos secundarios de sodio. La falla de un tubo del intercambiador de calor intermedio (sodio-sodio) obligó a llevar a cabo una serie de análisis para descubrir su causa y extraer y reparar el intercambiador.

Central nucleoelectrónica de Piqua. Durante la limpieza de las tuberías con agentes químicos, se dañaron varias válvulas que fue preciso reparar o sustituir. Las fugas en el circuito del refrigerante orgánico y del vapor secundario provocaron demoras repetidas.

Una vez concluidas las reparaciones e introducidas las modificaciones necesarias, se comprobó que las características de rendimiento reales de cada uno de los tres reactores se ajustaban estrictamente a las previstas en el proyecto.

I. INTRODUCTION

One of the most important phases in the construction and operation of a nuclear power plant is the period between the completion of physical construction and the first attainment of full power. In the early years of planning nuclear power plants, the importance of this period, together with the time, staffing and funding required, was frequently, but not always, significantly underestimated.

Three small nuclear power projects, the Elk River Reactor (ERR), the Hallam Nuclear Power Facility (HNPF), and the Piqua Nuclear Power Facility (PNPF), all of which are receiving substantial support under the USAEC Power Reactor Demonstration Program, are each nearing the completion of this post-construction testing period. These nuclear plants are all in the small power range and represent three different reactor concepts: ERR-22 MW(e) net, natural-circulation boiling-water reactor with fossil-fired superheat; HNPF-75 MW(e) net, sodium-cooled and graphite moderated; and PNPF-11.4 MW(e) net, organic cooled and organic moderated.

The experience obtained from these three nuclear power projects may permit some generalizations which could be used in improving cost or schedule estimates for future small nuclear power plants. Since all three of these plants are small, demonstration, "first-of-a-kind" plants, the experience may not be particularly applicable to large, "commercial", second-generation nuclear power projects.

II. BACKGROUND

The ERR and the PNPF, but not the HNPF, were authorized under "Second Round" legislation which provides for the construction and operation of nuclear power plants under an arrangement between the USAEC and a co-operatively or municipally-owned, as contrasted to an investor-owned, utility company. The utility agrees: (1) To furnish the site and the generator plant; (2) to purchase the steam generated by the reactor plant; (3) to purchase

the reactor plant after a period (usually five years) of USAEC operation; and (4) to supply a reactor operating staff during the period of USAEC ownership. The USAEC agrees: (1) to design and construct the reactor plant, and (2) to operate the reactor plant for the period of USAEC ownership, utilizing the utility company's operating staff. The co-operating utility company for the ERR is the Rural Co-operative Power Association (RCPA) and for the PNPf is the City of Piqua (CP).

The HNPf was authorized under an arrangement similar to that for the "Second Round" plants. The essential differences in the arrangement are that: (1) the co-operatively owned utility company, Consumers Public Power District (CPPD), makes a financial contribution toward the construction of the reactor plant, and (2) assumes responsibility for operating the reactor plant, as well as the generating plant, as soon as the reactor achieves full power. The USAEC furnishes some financial assistance for the initial five-year period of operation.

Information concerning design power output, ownership and major contractors is listed in Table I. Detailed descriptions of these projects can be found in the Bibliography.

III. SCHEDULES

The highlight schedules in existence at the start of the project, of construction, of post-construction testing, and as of 1 April 1963, are shown in Table II.

The date given for the start of the project is the effective date of the plant construction contract. The date given for the completion of construction is less clearly defined and was arbitrarily chosen as that at which the first post-construction test was begun. In each case, some construction work remained to be completed at that time. Substantial amounts of rework and plant modification were also required. This work, which usually involved construction workers, proceeded concurrently with the test programmes unless the extent of the work was too great to permit continuation of the tests.

It is apparent that, without exception, the time required for plant testing, completion of construction, and plant and equipment modifications during the post-construction testing period was underestimated in the early project stages. There is no easy explanation for this. Certainly a lack of experience in nuclear reactor construction for civilian use was a major factor. One of the goals of the Power Reactor Demonstration Program is to gain this experience.

Another contributing factor was that these plants, with the exception of the ERR, were first-generation plants. The major delays which were easily identifiable were the result of faulty equipment or poor equipment performance and can be traced to one or more of the following factors: poor design, incomplete or ambiguous specifications, inadequate component proof testing, sub-standard fabrication, inadequate inspection, or wrong installation.

Each of the three reactor plants is made up of hundreds of structural and equipment components. Almost all of these components have proven to be soundly engineered, fabricated and installed. However, the experience gained from these reactor plants demonstrates that, at least at present, it would be prudent to anticipate some problems with equipment and to make

TABLE I
PROJECT DATA*

	ERR	HNPF	PNPF
Type	Boiling water	Sodium graphite	Organic
Design power (MW(t))	58.2	240	45.5
Fossil fired superheater (MW(t))	14.8	-	-
Electrical generation (MW(e)) gross	23	82	12.5
net	22	75	11.4
Steam conditions, turbine inlet, °F	825	825	550
lb/in ²	600	800	445
Owner, reactor plant	USAEC	USAEC & CPPD	USAEC
Owner, generating plant	RCPA	CPPD	CP
Prime contractor	AC	USAEC	AI
Major subcontractors:			
Architect engineer	S&L	Bechtel	Holmes & Narver
Construction	Maxon	Kiewit	Messer
Reactor system	AC	AI	AI
Reactor core	Martin	AI	AI

* The abbreviations and shortened titles used in Table I refer to the following:

USAEC	United States Atomic Energy Commission, Germantown, Maryland
CPPD	Consumers Public Power District, Columbus, Nebraska
RCPA	Rural Cooperative Power Association, Elk River, Minnesota
CP	City of Piqua, Piqua, Ohio
A-C	Allis-Chalmers Manufacturing Co., Washington, D.C.
AI	Atomics International, a Division of North American Aviation, Inc. Canoga Park, California
Bechtel	Bechtel Corporation, San Francisco, California
Kiewit	Peter Kiewit Sons, Inc., Omaha, Nebraska
Messer	Frank Messer & Sons, Inc., Cincinnati, Ohio
S&L	Sargent & Lundy, Chicago, Illinois
Maxon	Maxon Construction Company, Dayton, Ohio
Martin	The Martin Company, Baltimore, Maryland
Holmes & Narver	Holmes & Narver, Inc., Los Angeles, California
ERR	Elk River Reactor
HNPF	Hallam Nuclear Power Facility
PNPF	Piqua Nuclear Power Facility

TABLE II
SCHEDULE HIGHLIGHTS
(Underlining indicates completion dates)

	Start of project	Start of construction	Start of testing	As of April 1963
<u>ELK RIVER REACTOR</u>				
Start of project	<u>June 1958</u>			
Complete Title I design	(1)			
Start construction	Aug. 1958	<u>Aug. 1958</u>		
Complete Title II design	Aug. 1960	Aug. 1960	<u>Aug. 1960</u>	
Complete construction	Oct. 1960	Oct. 1960	<u>April 1961</u>	
Begin fuel loading	Nov. 1960	Nov. 1960	June 1961	Nov. 1962
End of testing	June 1961	June 1961	Jan. 1962	Nov. 1963 (4)
<u>HALLAM NUCLEAR POWER FACILITY</u>				
Start of project	<u>Nov. 1957</u>			
Complete Title I design	Aug. 1958	Sept. 1958		
Start construction	March 1960	<u>April 1959</u>		
Complete Title II design	May 1960	May 1960	<u>April 1961 (3)</u>	
Complete construction	Jan. 1962	July 1961	<u>Aug. 1961</u>	
Begin fuel loading		Jan. 1962	<u>April 1962 (4)</u>	<u>Aug. 1962 (3)</u>
End of testing	Dec. 1962	Oct. 1962	Nov. 1962	Aug. 1963 (5)
<u>PIQUA NUCLEAR POWER FACILITY</u>				
Start of project	<u>March 1958</u>			
Complete Title I design	Nov. 1958	<u>March 1959</u>		
Start construction	Aug. 1959	<u>July 1959</u>		
Complete Title II design	March 1960	<u>March 1960</u>	<u>March 1960</u>	
Complete construction	April 1961	June 1961	<u>Nov. 1961</u>	
Begin fuel loading	May 1961	July 1961	<u>March 1962</u>	May 1963
End of testing	July 1961	Sept. 1961	May 1962	Nov. 1963 (6)

(1) Design was largely completed before formal start of project.

(2) Includes 28-d warrantee operating period and 60-d operating period for utility operator training before turnover to operating utility.

(3) Delayed due to addition of Radioactive Waste Facility.

(4) Wet fuel loading. Dry nuclear tests were performed in Jan. -Feb. 1962.

(5) Includes 30-d load following operation period before turnover to the operating utility.

(6) Includes two weeks base-load operation and two weeks load-following operation before turnover to the operating utility.

adequate provisions for the time, staffing and funds required for equipment repair or replacement during the post-construction testing phase of a nuclear power reactor project.

It is interesting to speculate as to whether or not, had the time and effort been taken to provide a higher degree of engineering, more life tests of prototype components, and 100% inspections of all equipment, the net

result would have been an increase or a decrease in the total time and manpower required to reach full power and to demonstrate plant reliability.

A less easily identifiable source of delays was the nature and extent of the procedures required in the authorization (licensing) process. Since these procedures were, in the main, all instituted after the start of the projects, the time and effort required could not have been foreseen. It is recognized that both the nature and the extent of licensing or authorization procedures will not all necessarily parallel those of the USAEC nor will they remain constant within the USAEC. Nevertheless, the experience gained from the ERR, HNPF, and PNPF is discussed in this report in the belief that it may assist in the development of estimates of the time, funding and manpower requirements that should be anticipated for these actions during the post-construction testing phase of future nuclear power projects.

It should be noted that there were few, if any, delays which could be attributed to operator error. It should also be noted that, although the measured values of the nuclear parameters in some instances did not fully agree with calculated predictions, there were no delays due to the nuclear performance of any of the three reactors.

IV. STAFFING AND COSTS

Staffing

The average staffing requirements, during the post-construction period, were: 50 for the ERR, 90 for the HNPF, and 58 for the PNPF. The distribution of this manpower is shown in Table III.

The differences in the staffing requirements can be explained, at least in part, by recalling the differences in the plants which would affect the manpower requirements.

TABLE III

POST-CONSTRUCTION TESTING - AVERAGE STAFFING

	ERR	HNPF	PNPF
Testing organization ⁽¹⁾			
Management	2	5	2
Testing & evaluation	10	19	7
Operation & maintenance	30	45	40
Clerical	2	14	3
Consults & specialists	2	2	2
Staff	4	5	4
	<u>50</u>	<u>90</u>	<u>58</u>

(1) Includes utility personnel who were part of the testing organization

The ERR is a natural-circulation boiling-water reactor with the fewest pieces of equipment external to the reactor vessel. It is a modest extrapolation of existing operating boiling-water reactor plants. It is to be integrated into a well-established power-generation system. Because of the long post-construction testing period there was less need for a high rate of work accomplishment at the site.

The HNPF is a "first of a kind" sodium-graphite reactor. It has many equipment items external to the reactor vessel. It is being integrated into a relatively new power-generation system. It is the largest of the three reactors. The rate of accomplishment of work at the site determined the length of post-construction testing period.

The PNPF is a "first of a kind" reactor. It has many equipment items external to the reactor vessel. It is being integrated into an established power-generation system. The rate of accomplishment work at the site determined the length of the post-construction testing period. It is the smallest of the three reactors. The estimated staffing required during the period of routine plant operation is shown in Table IV.

TABLE IV

ESTIMATED STAFFING FOR ROUTINE OPERATION

	ERR	HNPF	PNPF
Plant manager	1	1	1
Assistant plant manager	1	0	1
Shift supervisors	5	5	4
Reactor & plant operators	14	27	10
Health physics	2	2	2
Mechanical & process equipment	2	10	13
Electrical & instrumentation	2	10	5
Statistician/accountant	1	0	1
Utility man	1	2	1
Contract manager	1	1	0
Clerical	1	3	1
	<u>31</u>	<u>61</u>	<u>39</u>
Technical consultants	3	4	2
	<u>34</u>	<u>65</u>	<u>41</u>

Costs

In keeping a current estimate of the total cost to complete a project, it is necessary to include the costs of unanticipated schedule delays as they

occur. This cost estimate is made up of two components: (1) The cost of any necessary new or replacement equipment, plant modification, or off-site work, and (2) salaries, overhead costs, and the maintenance of essential operating equipment. The first component is directly associated with the cause of the delay. Experience gained on these three reactor projects has shown that the costs of the second component are very nearly proportional to the staffing requirements.

V. AUTHORIZATION PROCEDURES

The USAEC has established the principle that, for USAEC-owned and operated nuclear power reactors, a set of authorization procedures must be followed which are parallel to the licensing procedures required for investor-owned plants.

Organizationally, the USAEC has two large sub-divisions – one represented by the General Manager and one represented by the Director of Regulation. In effect, the "parallel procedures" concept requires that, insofar as licensing procedures are concerned, the major sub-division of the USAEC, represented by the General Manager, must fill a role parallel to that of the owner of an investor-owned nuclear power plant. In practice, the General Manager, in turn, delegates this role of applicant to the prime contractor. The prime contractor applies directly to the Director of Regulation for a construction or an operating authorization.

The prime contractor must, therefore, receive two authorizations to proceed: one (parallel to a licence) from the Director of Regulation, and one from the Contract Administrator, who represents the General Manager as the contractor's official point of contact with the USAEC. For each project, there are two required authorizations – a construction authorization and an operating authorization. However, in many instances it has been found desirable to issue operating authorizations in a series of steps, each at a higher power level. Therefore, most of the authorization procedures occur in connection with the operating authorization during the post-construction testing period.

Typical procedures in acquiring an authorization at each step are as follows: (1) The contractor prepares a Reactor Safeguards Report which describes the plant and analyses the course of events which would follow several credible postulated incidents involving combinations of equipment malfunctions and operator errors. One of these incidents is normally selected as the "maximum credible" accident. (2) The Safeguards Report is submitted to both the Contract Administrator and the Division of Licensing and Regulation. (3) The report is studied by USAEC personnel and, after some correspondence with the contractor to clarify portions of the report, meetings are held with the contractor and as a result certain revisions are made to the Safeguards Report. (4) The Safeguards Report and the application for an authorization are then referred to the Advisory Committee on Reactor Safeguards (ACRS) for review. This is followed by one or more meetings between the ACRS and the contractor. (5) A formal Public Hearing is held concerning the pending application. Although the character of the Public Hearings has changed in recent months, the Public Hearings held for the

three reactors under discussion were quasi-legal in nature and were conducted by a Hearing Examiner.

As new information is developed, supplements to the Safeguards Reports may be prepared. Not all of the above steps may be required for amendments to the authorization. At the present time, a Public Hearing may be held for any portion of the authorization processes but is only obligatory for the application for the initial construction authorization. At present, the minimum possible elapsed time from the first submittal of a formal request for an authorization to the receipt of an authorization is approximately:

60 days - Division of Licensing & Regulation (DL&R) action only

90 days - DL&R and Advisory Committee for Reactor Safeguards (ACRS) action only

120 days - DL&R, ACRS and Public Hearing

An indication of the time, funds and the technical, legal and administrative manpower required for the authorization processes can be assessed from Table V.

VI. POST-CONSTRUCTION TESTING PROCEDURES

The primary purpose of the post-construction tests was to demonstrate the functional adequacy of the individual items of equipment and sub-systems before integration of the plant into the utility company's power distribution network. A related purpose was to permit the reactor plant operators to become thoroughly familiar with the plant through actual operation of the equipment.

The post-construction tests were rather arbitrarily divided into two categories:

(a) Pre-operational tests

All systems and components were tested before nuclear operation of the plant under conditions which duplicated actual operating conditions as closely as possible. Testing of the reactor coolant loops with hot coolant was one of the most important tests for the HNPF and PNPf. Recirculation of the ERR primary coolant takes place within the reactor vessel.

(b) Operational tests

These tests began with reactor-fuel loading and were designed to evaluate plant performance characteristics during the approach to full power operation. Important tests in this series included the determination of nuclear parameters.

Each test was conducted in accordance with an approved written procedure which included the following information and instructions:

- (a) Purpose or objective of test;
- (b) Description of system to be tested;
- (c) Summary of test method;
- (d) Prerequisites to test performance;
- (e) Precautions;

TABLE V
AUTHORIZATION "PARALLEL" PROCEDURES

	ERR	HNPF	PNPF
Construction authorization			
Date of safeguards report	March 1959	Jan. 1958 Feb. 1959	Oct. 1958
Date of public hearings	None	April 1960	Oct. 1959
Date of authorization	Dec. 1959	July 1960	Feb. 1960
Supplementary actions ⁽¹⁾			
Safeguards reports supplements	0	5	5
Unofficial meetings			
Contractor & USAEC	4	2	2
ACRS meetings	3	3	7
Operating authorization			
Date of safeguards report	June 1960	April 1961	Feb. 1961
Date of public hearings	March 1961 Aug. 1962 Sept. 1962	Dry Critical: Nov. 1961 Wet Critical & Operation to 15% of Full Power: May 1962 Continuation of 15% Power Hearing: July 1962	Feb. 1961 June 1962
Date of authorization	Nov. 1962	Dry Critical: Jan. 1962 15% of Full Power Authorization: Aug. 1962	Aug. 1962
Supplementary actions ⁽¹⁾			
Safeguards reports supplements	0	5	1
Unofficial meetings			
Contractor & USAEC	13	10	3
ACRS meetings	8	4 ⁽²⁾	4

(1) Does not include routine correspondence.

(2) ACRS Meetings on full power operation were held April 1963. As a result of a change in the regulatory procedures, it is not expected that a Public Hearing will be required for full power authorization.

- (f) Initial plant conditions required;
- (g) Procedure for test performance;
- (h) Data sheets and illustrations, if required.

Each procedure, when used in conjunction with the plant-operating manual, was self-sufficient.

The test procedures were written by the prime contractor and reviewed by USAEC personnel. Performance of each individual test required the approval of the contractors' field organization manager and a USAEC Site Representative. Their approvals were given only if all initial plant conditions and prerequisites to the test performance had been met, and the necessary personnel were available and understood the test procedures and precautions. Any necessary minor revisions to the test procedures required the approval of the Site Representative. Substantial revisions required review by the USAEC staff. The Site Representative witnessed all tests, as necessary, to assure that they were performed safely and in accordance with the written procedures.

After the performance of each test, a test report and evaluation was prepared by the contractor and submitted to the USAEC staff. The test reports summarized the test and the results, including modifications and repairs required for satisfactory performance, and an evaluation of both the system operability and the significance of test results relative to further testing. The completed master test procedure and data sheets were retained at the site for reference.

The form and the manner of preparation of the test procedures and the test reports varied somewhat among the various projects. Most of the HNPF and PNPF test procedures were written well in advance of plant completion, and in many cases extensive revisions were required to conform to "as-built" conditions or to improve test performance. In form, the procedures were set up to test components collectively in "systems" tests. The ERR test programme was based primarily on tests of individual components. The written procedures were much less detailed than those of HNPF and PNPF, and consisted of functional instructions for test performance and data sheets which were included as a part of the written procedures. Topical reports which analysed and correlated the test results of major subdivisions of the test programme were subsequently prepared.

A listing of the titles of the test procedures for each of the three reactors is shown in Tables VI, VII, VIII and IX.

VII. EQUIPMENT FAILURE AND MALFUNCTIONS

Each of the reactor plants has hundreds of items of equipment. Almost all of these have demonstrated satisfactory performance. Only relatively few have failed to function properly, but these have caused both delays in plant completion and increases in project costs. The more significant failures and malfunctions are discussed below.

Elk River Reactor

The Elk River Reactor project was relatively free from the relatively minor problems of improper installations, wiring, and pipe fitting. Almost

TABLE VI
ELK RIVER REACTOR

Pre-operational tests	
Item	Number of individual tests
1. Containment-building spray system	1
2. Filling primary system	3
3. Primary system hydrostatic tests	2
4. Containment-building leak rate	1
5. Primary purification system check list	1
6. Control-rod drive system checkout	6
7. Plant steam-supply system checkout	1
8. Startup heater	1
9. Decay-heat cooling system	1
10. Shield cooling system	1
11. Fuel-storage-pool system	1
12. Waste disposal	1
13. Service-water system	1
14. Emergency and test condenser	2
15. Boric acid injection system	1
16. Core emergency spray system	1
17. Fuel-transfer system	2
18. Filling secondary system	1
19. Secondary hydrostatic test	1
20. Building air-conditioning system	1
21. Recombiner system	1
22. Control air system	1
23. Core structure installation	1
24. Reactor vessel-head bolt tensioners	1
25. Electrical circuit breakers	6
26. Pulverized coal-fired superheater mechanical components	22
27. Primary and secondary system pumps	16
28. Primary and secondary system valve controls	9
29. Air compressors	2
30. Emergency power supply	3

TABLE VI (continued)

Item	Number of individual tests
31. Primary and secondary system pressure, level, flow, temperature, and other recorders	16
32. Primary and secondary system pressure, level, temperature, and flow indicators or gauges	21
33. Level control systems	3
34. Nuclear instrumentation	9
35. Reactor scram circuits	1
36. Reactor start and withdraw permit	1
37. Control-rod insert and withdraw circuits	1
38. Radiation monitors	16
39. Level alarms	1
40. Pressure alarms	1
41. Temperature alarms	1
42. Breaker and auxiliary relay contacts	1
43. Recorder and indicator contacts	1
44. Valve limit switches	1
45. Annunciators	1
46. Miscellaneous devices	1
TOTAL PRE-OPERATIONAL TESTS	168
Retests required	
Item	Number of individual tests
Process and mechanical	
(a) Filling of primary system	3
(b) Control-rod-drive system (cold and hot)	4
(c) Reactor-vessel heat installation and removal	2
(d) Primary system hydrostatic test	1
(e) Purification system	1
(f) Startup heater (165°F to 425°F)	1
(g) Emergency and decay-heat cooling (425°F to 160°F)	2
(h) Emergency test condenser flow	1

TABLE VI (continued)

Item	Number of individual tests
Process and mechanical	
(i) Fuel-transfer system	1
(j) Portable purification system	1
(k) Boric acid injection system	1
(l) Containment-building exhaust system	1
(m) Containment-building spray system	1
(n) Containment-building leak system	1
Electrical and instrumentation	
(a) Nuclear instrumentation	9
(b) Primary system flow temperature, pressure and level instruments	6
(c) Radiation monitoring systems	3
(d) Secondary system flow, temperature, pressure and level instruments	12
TOTAL INDIVIDUAL RETESTS	51

all of the schedule delays during the post-construction testing period were associated with deficiencies in the design, fabrication, and installation of the reactor vessel and its internals. This probably could have been anticipated since the ERR, a natural-circulation boiling-water reactor, has relatively less equipment external to the reactor vessel and the reactor vessel is relatively more complex when compared to either the HNPF or PNPF. The major equipment deficiencies and the associated schedule delays are discussed below:

Reactor vessel fabrication - one month

The completion of fabrication and the delivery of the reactor vessel was delayed due to a nationwide strike of steel workers.

16-in nozzle repairs - two months

The reactor vessel has four 16-in nozzles which were included in the design so that the plant could be converted to a forced circulation reactor at a later date. The nozzles are low-alloy steel, clad internally with stainless steel. The outer end of each nozzle is welded to a stainless-steel pipe stub.

TABLE VII
ELK RIVER REACTOR

Operational tests	
Test	Title
101	Fuel loading to critical mass
102	Critical mass determination
103	Temperature-coefficient measurement
104	Control-rod calibration
201	Fuel loading to twice critical mass
202	Control-rod calibration
203	Temperature-coefficient measurement
301	Fuel loading to full core
302	Temperature-coefficient measurement
303	Control-rod calibration
304	Shut-down safety margin
305	Core excess reactivity measurement
306	Reactivity-coefficient measurement
307	Fuel load optimization
308	Neutron-flux distribution measurement
401	Power calibration
402	Temperature-coefficient measurement
501	Temperature-coefficient measurement
502	Shielding survey
601	Power determination
602	Void reactivity <u>versus</u> power
603	Reactivity in equilibrium xenon <u>versus</u> power
604	Water-level check
605	Shielding survey
606	Subcooling measurements
607	Stability margin
701	Response to load changes
702	Pressure-coefficient measurement
703	Response to reactivity changes
704	Response to 4-rod scram
705	Response to load demand - cold leg
801	Initial power run
802	Plant response to load changes
803	Plant response to 4-rod scrams 28-d full-power warranty run 60-d RCPA training run

TABLE VIII
HALLAM NUCLEAR POWER FACILITY

Pre-operational tests	
Helium system	Control-rod operability
Nitrogen system	Cooling-water system
Preheat system	Portable purge unit
Na pressure instrumentation	Loading-face shield cooling
Na level instrumentation	Dry criticality
Emergency power	Reactor heaters
Na flow instrumentation	Dry excess loading
Thermocouples	Purge of secondary Na system
Na service system	Purge of primary Na system
Instrument air system	Auxiliary steam
Na leak detection system	Fill of secondary Na system
Radiation detection and monitoring system	Fill of primary Na system
Protective system	Hot sodium circulation
Lighting, power and communications	Main heat-transfer system
Control-rod structure	Decontamination room
Radioactive vent system	Wash cells
Nuclear instrumentation	Maintenance cell
Neutron chambers	Radioactive liquid waste
Control-rod drives and actuators	Wet criticality
Plant-control system	
Post critical testing	
Wet excess fuel loading	Plant-control system
Control-rod calibration	Emergency feedwater system
Axial power distribution	Scram and loss of load
Zero power coefficients of reactivity	Radiation shielding
Plant protective system	Power coefficients
Hydraulic test — main heat-transfer system	Radial power distribution
Heat-transfer test	Rise in power
Steam-generator — steam purity	Xenon
Cooling systems	

TABLE IX
PIQUA NUCLEAR POWER FACILITY

Pre-operational tests	
Test	Title
I-1	New coolant storage
I-2	Coolant purification and waste gas systems
I-3	Main heat transfer, degasification and pressurization and decay-heat removal systems
I-4	Organic waste-disposal system
I-5	Industrial aqueous waste disposal
I-6	Fuel-handling system
I-7	Nuclear instrumentation
I-8	Radiation detection and monitoring and failed-element location system
I-9	Plant protective system
I-10	Auxiliary service system
Initial nuclear tests	
II-1	Critical loading
II-2	Loading for excess reactivity
II-3	Measurement of isothermal temperature coefficient of reactivity
II-4	Control-rod worth measurements
Post-critical-operational tests	
III-1	Approach to full power
III-2	Power-coefficient measurement
III-3	Xenon effects
III-4	Sustained full-power operation
III-5	Load-following power operation

TABLE X
NUCLEAR CHARACTERISTICS

	HNPF		ERR	
	Predicted	Measured	Predicted	Measured
Critical loading (No. of fuel assemblies)	$32 \pm 2^{(1)}$ $50 \pm 5^{(2)}$	$30.6^{(1)}$ $52.4^{(2)}$	30 ± 6	40.4
Full core loading	137	140	148	$148^{(5)}$
Full core reactivity (% Δk)	$13^{(1)}$ $8^{(2)}$	$10.6^{(1)}$ $6.5^{(2)}$	$11.3^{(6)}$	$9.3^{(6)}$ $10.5^{(7)}$
Control-rod worth (% Δk)				
Total	14	12.9	18.3	18
Max. single rod	1.5-2.0	1 ⁽³⁾	3.5	~ 4.5
Reactivity insertion rate-max. (% $\Delta k/s$)	<0.03	$0.012^{(3)}$	0.06	$0.12^{(4)}$
Flux peaking factors				
Axial	<1.65	$1.5-1.6^{(3)}$	$2.4^{(8)}$	
Radial	<1.50	$1.3-1.5^{(3)}$	$1.6^{(8)}$	
Local			$1.25^{(8)}$	
Total			$4.8^{(8)}$	
Reactivity coefficients				
Isothermal temperature (% Δk)	$+0.3 \times 10^{-3}$	$+0.7 \times 10^{-3}$	$-2.8 \times 10^{-3}^{(9)}$	$-4 \times 10^{-3}^{(10)}$
Fuel temperature (% Δk)	-1.6×10^{-3}			
Moderator temp. (% Δk)	$+1.3 \times 10^{-3}$			
Coolant temp. (% Δk)	$+0.6 \times 10^{-3}$			
Void (% $\Delta k/\%$ void)			$-0.12^{(11)}$	
Zero power flow (% $\Delta k/\%$ flow)	~ 0	~ 0		
Power (% Δk)				
Prompt	$-3.5 \times 10^{-3}^{(12)}$	$-2 \times 10^{-3}^{(12)}$ -3×10^{-3} to	$-5.1 \times 10^{-3}^{(13)}$	
Delayed	$+4.1 \times 10^{-3}^{(12)}$	$+3 \times 10^{-3}^{(12)}$ $+6 \times 10^{-3}$ to		
Steady state	$+0.6 \times 10^{-3}^{(12)}$	0 to (12) $+4 \times 10^{-3}$	-72×10^{-3}	

(1) Dry (sodium-free) core

(2) Wet (sodium-free) core

(3) In-normal operating configuration

(4) With abnormal control-rod configuration

(5) Includes 20 spiked elements

(6) Full core without spikes

(7) Full core with spikes

(8) Revised prediction based on air-water hydrodynamic tests and analyses in February-March 1963.

(9) Average 68-536°F

(10) Value at 160°F

(11) At 0 void

(12) At 20% full power

(13) Doppler only

A crack was discovered in the external surface of one of these nozzles at a point near the dissimilar (carbon steel to stainless steel) weld. The crack was of the weld shrinkage type. All four nozzle weld joints were repaired by fabricating and welding in new stainless-steel clad, carbon-steel transition sections.

Grout removal - two months

A space is provided below the reactor vessel to provide for thermal expansion of the reactor vessel and attached control-rod nozzles. The vessel is supported at the vessel flange. Cement grout was used to fill the cracks between concrete shielding blocks in the reactor biological shielding piping tunnels. Some of the grout unexpectedly ran down into the expansion space. It was removed by chipping and high-pressure water jets.

Fuel-rod spacer straps - two months

The rods in each fuel assembly are positioned by spacer straps. Originally, spacer straps were held in position by two side plates welded to the spacer straps. In making the welds, some of the fuel-cladding tubes were penetrated accidentally. The penetrated tubes were replaced and a modified design was adopted which eliminated the side straps and, therefore, the welding operations. The tubing of selected fuel rods was dimpled above and below the spacer straps to prevent them from slipping either up or down from their original position.

Reactor-vessel cladding cracks and resulting test programme - 12 months

The ERR vessel is constructed of low-alloy - ASTM A302B - steel, clad with stainless steel for corrosion resistance. In April 1961, following the initial non-nuclear heating and hydrostatic pressure tests of the primary system, cracks were discovered in the stainless-steel cladding.

The cracks were confined to areas which were clad by the weld overlay process. None were found in the areas clad by roll bondings. The defective clad areas in the vessel and vessel head were repaired by removing the cladding and replacing it. Instead of 308 SS weld material, 309 SS was used for the first pass, and welding techniques were improved.

This discovery of cladding cracks led to an extensive analytical and experimental programme to define the adequacy of the vessel for its intended service. The principal consideration throughout the test programme was the possibility that cladding cracks existed which had not been removed and repaired, that these could spread as a result of pressure and temperature cycling of the reactor vessel and that, in time, result in a leak or rupture of the reactor vessel. The analytical programme included a detailed stress analysis of critical areas under both steady state and transient conditions, a complete review of all of the original vessel radiographs, and a complete review of all vessel fabrication and inspection records.

The experimental programme included: (1) ultrasonic inspection of the vessel and vessel-head nozzle strength welds, (2) vessel-head nozzle re-

radiography, (3) chemical analysis of approximately 160 cladding samples taken from defective cladding areas of both vessel and head, (4) a determination of the material properties of the head and vessel flange steel by cutting plugs from the reactor vessel flange, and (5) a determination of the material properties of the reactor vessel base material by use of material which had been prepared in the same manner as the base reactor vessel material.

Reactor vessel modifications - two months

As a result of the analyses and experiments, the interior nozzle corners of eight reactor vessel nozzle-to-vessel interior joints were rounded (radius) and then reclad, thermal sleeves were inserted in the two feedwater nozzles, and provisions were made for an annual inspection of the eight surfaces adjacent to the rounded interior nozzle-to-vessel joints.

This work was completed in April 1962, and it was decided that the vessel was adequate for its intended service. As a result of regulatory review, the vessel service life was limited to five full-power years or 250 full pressure cycles, whichever occurs first. In April 1963, an experimental programme was initiated at Southwest Research Institute to define the useful life of the Elk River vessel, in terms of full pressure cycles, by tests to destruction of geometries and materials similar to those used in the ERR vessel.

Pre-operational retests - two months

As a result of the delay of more than one year in startup, several of the system-oriented post-construction tests were repeated (Table VI) in order to prepare the plant systems components, and instruments for the Nuclear Startup Program (Table VII).

Steam separation analysis and resulting modifications - four months

In January 1963, as a result of the data obtained from EBWR high power operation, the hydraulic performance of the ERR was re-evaluated. These data are reported in "Performance characteristics of the Experimental Boiling Water Reactor, (0 to 100 MWt)", Reactor Engineering Division, Argonne National Laboratory, April 1963. It was concluded that steam carry-under into the downcomer, reactivity in voids, operating water level, and water carry-over to the evaporators all might be significantly higher than originally predicted. These adverse conclusions were then confirmed by detailed analyses and concurrent air-water tests.

A core riser (chimney) with a new configuration, a steam collection baffle, and in-core temperature and pressure instrumentation to measure the magnitude of core inlet subcooling were made and installed. Core and steam separation performance is now expected to closely approach design predictions.

Reactor startup heating system

The reactor startup heating system used an external heat exchanger (steam heated) and a piping loop connected to the reactor vessel to heat the primary water by natural circulation. Pre-operational checkout of this system indicated that, because the lower piping loop connection to the reactor vessel was above the bottom of the vessel, as the bulk temperature of the water in the vessel increased, cool water tended to stagnate in the lower reactor vessel head. This situation remained until the bulk stagnant water temperature difference was approximately 125°F. At this point, the cool water in the bottom of the reactor vessel rose quickly to the top of the vessel rapidly cooling the upper reactor vessel and vessel-head surfaces.

This condition was corrected by connecting the recirculation loop piping to a line which enters the bottom head of the reactor vessel. Piping and valving changes were also made which now permit the use of the decay-heat cooling system pump during startup heating operations.

Control rods

As a result of Dresden reactor experience with the use of 17-4 pH steel for certain control-rod drive parts, the ERR control-rod drives were removed in 1961 to age harden the 17-4 pH parts at 1100°F instead of at 875 to 900°F. This increased the ductility but decreased the hardness of the material. Wear resistance tests using a spare ERR control-rod drive demonstrated that the slight decrease in wear resistance was acceptable.

As a result of EBWR control-rod operating experience, the type 304 SS rivets used to join the boron stainless-steel poison sections to the Zircaloy-2 follower sections were replaced with Inconel rivets to minimize the possibility of stress-corrosion cracking. Scram tests were performed to verify the mechanical strength of this design change.

Containment vessel

Following Commission Safety and Regulatory Group review of plant design features in May 1962, it was recommended that, in order to avoid the possibility of brittle fracture when the containment vessel is pressurized in cold weather, all pressure containing parts of the reactor containment vessel should be maintained at a temperature 30°F above the nil ductility transition temperature of the containment vessel materials.

The nil ductility transition temperature of the containment vessel material is -20°F. Since the containment building is insulated on the outside and heated on the inside, the only modifications required were to penetrations such as the fresh-air inlet duct. Modification consisted of installing electrical cable heaters and additional insulation. During the past winter, all portions of the containment vessel were maintained above 10°F with an outside air temperature as low as -35°F.

Hallam Nuclear Power Facility

In addition to numerous minor failures; such as poor electrical connections, incorrectly wired instrumentation, and wrong pipe connections,

none of which were serious individually but in the aggregate were the cause of significant delays in attempting to perform the post-construction tests on schedule; there were several malfunctions of major equipment items. These are discussed below.

Intermediate heat-exchanger tube failure

The HNPF has three separate cooling loops, each containing two intermediate heat exchangers separating the radioactive sodium in the primary loops from the non-radioactive sodium in the secondary loops. In November 1962, during zero power testing, a leak developed in one of the intermediate heat exchangers. This was detected by an increase in sodium level in the reactor and a decrease in the sodium level in the expansion tank in the secondary loop.

The leaking unit was removed and cleaned by washing with anhydrous ammonia. The leak was located by use of an ultrasonic leak detector. The failure had occurred in one tube in the periphery of the tube bundle and was located at the first baffle above the tube sheet near the shell side inlet.

It has been determined that the failure was caused by metal fatigue. Both analyses and vibration tests of the remaining units showed that a vibration, near the calculated natural frequency of the heat exchanger tubes, could occur at flow velocities equal to those present in a localized region near the shell side inlet. Metal strip spacers have been installed at right angles to the tubes between the tube sheet and first baffle on the shell side inlet. Tests of the heat exchangers in actual service have demonstrated that these strip spacers eliminate the tube vibration.

Secondary system helium entrainment

The secondary loops contain expansion tanks. The cover gas is helium. Originally, the full flow of the sodium in the loop passed through the expansion tanks. During pre-operational testing of the main heat-transfer systems, it was discovered that at high flow rates helium was being entrained causing audible gurgling noises throughout the system and inducing pipe motion in long flexible piping runs. The entrainment was eliminated by installing a line which bypassed approximately 96% of the total mass flow around the expansion tank.

Reactor cavity moisture

The reactor vessel and core are contained in a welded and sealed reactor cavity. During initial heating of the sodium in the primary system, it was observed that some of the pipe hangers, which support the piping in the reactor cavity, did not move as predicted. An inspection of the reactor cavity revealed that the condensation of the moisture, which was being driven from the reactor thermal insulation, had caused an accumulation of corrosion products on the pipe hangers and that the pipe hangers had been installed without the specified corrosion protection.

The pipe hangers were cleaned and coated with a radiation resistant grease. Temporary ventilation was provided until most of the moisture was driven from the insulation.

Sodium system valves

Throttle valves are used to control convection flow to prevent thermal transients in the primary loops after a scram. During post-construction testing, the throttle valve in primary loop No. 3 stuck in the open position. Inspection showed that the operating shaft (valve stem) was bent and the shaft guide bushing was cracked. After repair, the valve satisfactorily passed a reliability test.

During low power testing, there were several failures of the throttle valve operating mechanisms. The source of trouble was in the gear boxes which operate the jacking mechanism which positions the valve stem. The jacking mechanisms have been strengthened and the gear boxes modified.

During hydraulic tests, the throttle valve which controls the flow of sodium around the graphite moderator logs was discovered to have been installed in reverse. The valve was cut from the system and installed properly.

Loading-face shield cooling system

The loading-face shield is cooled by nitrogen pressurized to 235 lb/in² gauge. The cooling system compressors are of the conventional reciprocating type. Pre-operational testing showed that the piping vibrated excessively, and the compressors did not meet the required performance specifications. The units were modified by the supplier and now meet the performance specifications. Piping vibration has been eliminated by installing additional piping supports. The leakage of nitrogen around the compressor shaft was then found to be excessive. The compressors were modified to provide a nitrogen recovery system which collects the nitrogen and returns it to service by the use of a small diaphragm type compressor.

Fuel assembly handling

Minor difficulties were encountered in removing some of the fuel assemblies from the reactor after dry critical tests. The problem was traced to the disconnect sleeve between the process tube and the hanger rod which apparently sometimes engaged the lower end of the loading-face shield sleeve. The probability of this occurrence is believed to have been reduced by increasing the length of a chamfer to eliminate a small flat surface above the original chamfer. Theoretically, at least, binding could still occur since there is enough clearance to allow the sleeve to shift off centre and present a flat surface to the loading-face shield sleeve. If this should occur, the problem of freeing the fuel assembly is more in the nature of an annoyance than a major malperformance. During loading for the wet critical test, about one-third of the fuel assemblies did not seat properly in the lower support plate on the first attempt at insertion. The difficulty appeared to be with the piston rings on the lower part of the fuel assembly which are intended to fit in the grid plate nozzles and prevent bypass flow of sodium. There has been no proven explanation for this seating difficulty. It is believed to be associated with high levels of sodium oxide in sodium since there were

no further seating difficulties after the sodium oxide content was reduced to operating levels.

Loading-face shield

A large rotating stepped plug (loading-face shield) of reinforced dense concrete, encased in ferritic stainless steel, provides shielding above the reactor. A frozen metal seal, between the periphery of the loading-face shield and the reactor cavity liner, prevents the escape of reactor cover gas. Small vertical openings are provided in the loading-face shield for the insertion and removal of core fuel assemblies. Three larger plugs are provided, which by rotation of the shield, can be positioned over any of the moderator or reflector elements.

During the first attempt to rotate the loading face shield, it began to bind after it was rotated 110°. The rotation was reversed and after rotating 75°, it could not be moved. The shield was lifted from the reactor vessel and it was determined that the cavity liner had a slight eccentricity. This was corrected by machining to provide greater clearance.

Piqua Nuclear Power Facility

The Piqua Nuclear Power Facility, although not particularly complex in principle, does contain a multitude of individual parts. Almost all of these are quite conventional in nature. For example, steam tracing lines are provided to maintain the primary loop piping temperatures above the melting point of the organic coolant and the associated steam traps. In the aggregate, the installation, maintenance and repair of these rather conventional components have contributed significantly to delays in the project schedule.

Malfunctions of equipment are discussed below.

Cleaning of the reactor systems

Four attempts were made to clean the reactor system piping. Each cleaning attempt left a film of ferric oxide. Successful cleaning was eventually accomplished after modifying the cleaning process as follows: (1) The citric acid cleaning solution concentration was reduced from 5% to 3%; (2) the acid solution was buffered from a pH of 3.5 to 6; (3) the temperature of the cleaning solution was reduced from 190°F to 160°F; and (4) the rinse water pH was maintained above 9.

Small gear pumps

Gear pumps, used in the fuel storage system and organic fluid purification systems, developed leaks due to worn impeller bearings after a short period of operation. The metal bearings on these pumps have been replaced with teflon bearings and the pump speed has been reduced. The bearings have an acceptable life expectancy.

Steam tracing

Steam tracing is required around organic lines because of the high melting point of the organic coolant. The steam tracing lines are made of carbon-steel tubing. Sections of this tubing are connected by swage-lock fittings. Steam leaks at these fittings, and through holes in the tubing due to corrosion, resulted in many delays. It was found that, by removing the insulation from the fittings, they could be more easily maintained and further, if an exposed fitting leaked, moisture did not get under the insulation and travel along the tubing causing corrosion of the tubing.

With steam tracing throughout the plant, numerous steam traps are required for collecting condensate. Bucket-type traps were installed initially. These were replaced with vertically mounted impulse traps. The impulse traps require significantly less piping and better access to equipment. The number of trap failures is still high but may be reduced with improvements to the pressure relief system.

Inadequate steam tracing at several points in the system permitted freezing of the organic fluid. This occurred most often in relief and other small process lines as well as in areas which permitted a concentration of the para-terphenyl which has a relatively high melting point. The coolant is a mixture of ortha, meta and para-terphenyl. Additional steam tracing has been added.

Heating and ventilating system

Heat losses from the steam-tracing system and pump housings require a large amount of cooling and ventilation to maintain the ambient temperatures in the plant at permissible levels. Exposure of steam-tracing fittings also added to the heat load. As a result, the cooling capacity of the heating and ventilating system was doubled. Peak ambient temperatures were maintained below 110°F during the past summer.

Pump seal temperatures

After a short operating period, the temperature of one of the pressurization pump seals exceeded its design limit. It was determined that scale from the cooling water was collecting in the pump cooling cavity. The scale formation was traced to the use of tap water. The shield cooling system (a demineralized system) was expanded to provide water for cooling the pressurization pumps and the main heat-transfer pumps.

Further testing of the pumps, following the modification of the cooling water system, demonstrated that the pressurization pump seals were still running too hot. It was then determined that holes in the hold-down plate at the seal were permitting flow of hot coolant to the seal. After plugging of these holes, operation has been satisfactory.

Organic leaks

The organic piping systems of the PNPf use flanged connectors. These have required a very high level of maintenance to reduce flange leakage to an acceptable level.

Many of the valves in the organic system were improperly assembled. Reassembly and rework has reduced leakage. Overheating and failure of valve seals has caused leaks in the large butterfly valves. Water-cooled packing glands have been installed to reduce packing temperature and thus reduce leakage.

Proper sealing of the reactor vessel head has been difficult and time consuming. As the temperature of the system increased, leaks appeared. This required retightening of the head bolts. The torque wrench, which was provided by the contractor, was difficult to use and the care required to prevent galling of the fine ground threads on the flange bolts added to the problem of repeated tightening.

Rugged stud bolts, to replace the existing flange bolts, and stud tensioners, to accurately and uniformly apply bolt stresses, are now on order. Their use is expected to solve this leakage problem.

Control-rod operation

Checkout of the control-rod circuitry revealed operational difficulties which required repairs and modifications. A short in the power lead to the indicator coil (inside the reactor) required repair. Subsequent tests led to the replacement of the circuit breakers and rectifiers with equipment with a higher rating.

Rectifiers in the driver coil circuits failed due to high reverse voltages. Suppressors have been installed. Additional shorts in the grip coil power leads necessitated rework of the insulation around the coil case leads of all 13 control rods. The sleeves on the movable armatures have been reworked to obtain satisfactory tolerances.

VIII. RESULTS OF NUCLEAR TESTS

Nuclear tests were prepared and followed for each of the three reactors. Initial nuclear parameters were measured with the major emphasis on critical loading, full core reactivity, control-rod worth and shut-down margin, and pertinent coefficients of reactivity. As of 1 April 1963, core loading and zero power, low-temperature tests had been performed at the ERR; wet and dry zero power tests and power tests up to 15% of full power (38 MW(t)) had been completed at HNPF; and fuel loading was anticipated within the month at the PNPF. Major nuclear parameters determined by these tests are compared to values predicted before the tests in Table X.

Fuel loading at each reactor was monitored by special in-reactor nuclear instruments which were incorporated into the plant protective system. Plots of inverse multiplication or count rate versus the number of fuel assemblies loaded were developed, and incremental fuel additions were limited to one-half (one-third for PNPF) of the extrapolated critical fuel element loading (rounded to the next higher integer). After the critical loading was reached, incremental additions were limited to less than the prescribed shut-down margin limits. The most significant aspects of the nuclear test programmes are discussed in the following sections.

Elk River Reactor

Fuel loading and zero power, low-temperature nuclear tests with a full core have been performed at the ERR.

During fuel loading, the Pu-Be source produced 10^6 neutrons/s. Flux levels were monitored by two proportional counters and one compensated ion chamber in the reactor. These were relocated from channel to channel as the loading progressed.

In general, the measured results were in reasonable agreement with predictions. The predictions were made without benefit of a critical experiment using thoria-urania fuel.

The critical loading was 40.4 fuel assemblies, compared to a predicted 30 ± 6 . Part of the difference is attributable to a failure to include the stainless steel in the dummy element containing the source in the calculated prediction.

Isothermal temperature coefficient data were taken at fuel loadings of 45, 69, and 148 fuel assemblies. In the temperature range from 70°F to 160°F , the temperature coefficient was positive for the small core, becoming negative at the 69-fuel assembly loading. This effect is largely due to the increasing amount of control-rod material in the core and the corresponding negative temperature coefficient of a thermally black absorber.

With all control rods, except the centre rod, banked near the core centre-line and the centre rod out, a slight reactivity increase was observed upon insertion of the centre rod. This was due to the expulsion of water from the lower region of the core by the Zircaloy control-rod follower which is 12 in shorter than the core height.

After loading 148 regular fuel assemblies (enrichment 4.2%), 20 were removed and replaced with spike fuel assemblies (enrichment 5.3%). This increased the core reactivity to approximately the maximum permissible with a required reactivity shut-down margin of 2%.

Hallam Nuclear Power Facility

Nuclear characteristics of the dry (sodium-free) HNPF core were measured to provide data to establish the dry shut-down margin and the sodium reactivity effect. The dry critical loading was 30.6 fuel assemblies, compared to an original prediction of approximately 23. A check of the prediction showed that the reactivity worth of the Sb-Be source in a central fuel-element position had not been taken into account. After accounting for the source and the absence of the fuel assembly which it replaced, the calculated critical loading was approximately 32 fuel assemblies.

During the testing, three in-reactor fission chambers were used to monitor neutron flux levels. The useful range of these instruments was restricted by procedural specifications to the range, 3×10^2 to 9×10^5 cpm. It was necessary to reposition the chambers after the 26th and again after the 30th fuel assemblies were loaded to maintain the count rate in this range.

After criticality was reached, the range of the in-reactor instrumentation was too limited for further period measurements. Withdrawal of the chambers was not satisfactory for two reasons:

(a) Sensitivity remained too great until the chamber was withdrawn into

a borated sleeve surrounding the detector thimble, at which point virtually all sensitivity was lost abruptly.

(b) Nuclear heating of the fuel (2°F to 3°F) was observed at flux levels too low for accurate period measurements in the presence of the strong source used in the test (1 to 2×10^9 neutrons/s).

The source reduction was accomplished by mechanically withdrawing the Sb capsule from the Be sleeve until the desired reduction was obtained. This procedure, which reduced the source strength by a factor of 100 to 1000, with the reactor slightly subcritical and a flux level appropriately high in the nuclear instrument range, was used repeatedly throughout the dry and wet fuel-loading tests.

Dry fuel loading was continued until a total of 42 fuel assemblies were loaded. At this point, the reactor was critical with all control rods one-half withdrawn. Core reactivity was $3.8\% \Delta k$ and control-rod worth for this loading was approximately $12\% \Delta k$.

The dry isothermal temperature coefficient of reactivity was measured at the 42 fuel-assembly loading; the average value was $+0.36\text{c}/^{\circ}\text{F}$ between 70°F and 313°F and decreased with increasing temperature.

The reactor was unloaded following these tests for the performance of the hot sodium circulation tests, after which wet nuclear testing was begun. The wet critical loading was 52.4 fuel assemblies, compared to a predicted 50 ± 5 (after accounting for the source). Following the critical loading, fuel was added in increments of 3, 6, 6, 6, 9, 9, 9, 9, 15, and 15 to a total of 140 fuel assemblies. The loading limits were: a maximum incremental addition of 1% reactivity and a minimum calculated 2% reactivity shut-down margin. The major reactivity parameters are as follows:

Full core reactivity (140 fuel assemblies)	6.5%
Total control-rod worth	12.9%
Sodium worth	4.1%
Dry core shut-down margin	2.3%

Because of perturbation effects, inherent in measuring large reactivities by rod calibration techniques, the absolute accuracy of these data remains uncertain; however, the sodium draining tests established, with certainty, the existence of a dry shut-down margin.

After the reactivity is reduced by fuel burn-up, additional fuel may be added, up to a maximum of 182 fuel assemblies. All remaining core positions are now filled with dummy fuel assemblies. One hundred thirty-seven fuel assemblies are required to provide full power output on the basis of fuel temperature limits alone.

Piqua Nuclear Power Facility

Upto 1 April 1963, nuclear testing had not started at the PNPF. The tests will be similar to those conducted at the HNPF and the ERR. It is anticipated that the initial core loading will be limited to 60 fuel assemblies. Fuel assemblies will be added to complete the full core loading of 85 fuel assemblies after reactivity is reduced by fuel burn-up.

IX. RESULTS OF SYSTEMS TESTS

Elk River Reactor

By 1 April 1963, after completion of the repairs and modifications discussed in section VII of this report, the plant was believed to be in a state of readiness for initiating the power tests.

The fuel-loading process was demonstrated during the critical tests. It was accomplished quickly and efficiently. The time required for a fuel-assembly loading cycle was usually less than 10 min per assembly. This accomplishment was possible because of the inherent simplicity of the manually operated fuel-loading equipment and the skill of the fuel-loading crew.

Hallam Nuclear Power Facility

Pre-operational testing, zero-power nuclear testing, and testing to 15% of full power have shown that the plant can be operated safely and perform essentially as designed. Discrepancies and deficiencies found during testing have been, or are in the process of being, corrected.

Low power operation

The plant has been operated up to 15% power in carrying out the low-power portions of the test programme. Before initiating the tests, the amplifier gains on the protective-system nuclear instrument channels were advanced so that a flux-level scram trip would occur at 125% of 15% power.

The reactor was operated at 2.5 MW(t), 10 MW(t), 20 MW(t) and 38 MW(t) to permit testing at these power levels. Before each power-level increase, data reduction was completed to the extent necessary to determine that the plant could be expected to operate satisfactorily and safely at the next power level. This procedure will be followed during the remaining power testing. Total integrated power to date is shown in Fig. 1.

At each step, power levels were calculated by two methods: (1) Total reactor mass flow as indicated by flow meters and temperature rise across the core, and (2) steam flow and enthalpy loss across the heat sink. Power level calculations were as follows:

Nominal (MW (t))	Reactor flow and temperature rise (MW(t))	Steam flow and enthalpy rise (MW(t))
2.5	2.3	-
10	8.7	10.3
20	18.0	17.05
38	38	47.3

It was determined that the flow-meter calibration was in error. The calibration curve has been corrected for flows up to 50% of full flow.

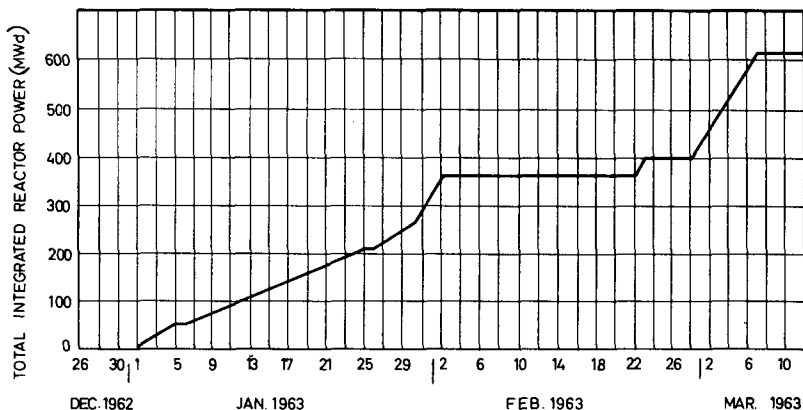


Fig. 1

Total power generated (HNPF)

Unplanned scrams and other programme interruptions

Eleven scrams occurred during this phase of the testing programme. Of these eleven, three were originally planned in the testing programme and two were added to obtain additional information. Two unplanned scrams, from the 10-MW(t) test level were initiated by the flow-ratio computers during preparation for the primary system cold trapping.

During recovery after the planned 10-MW(t) scram, two flow-ratio computer scrams were initiated at low power levels. It was determined at this time that the computer action was taking place well below the 20% flow level where protective action is supposed to become effective. The computer chasis was re-aligned to prevent action in the low flow range.

During performance of a small negative power ramp for the power coefficient test at the 20-MW(t) test level, the rate-of-change trip point of the fuel channel outlet temperature computer was exceeded, initiating a scram. A review of after-scram conditions led to the substitution of this scram for the one planned from 20 MW(t).

One scram was originally planned from 38 MW(t). However, it was decided that two were desirable, one at less than full temperature gradient and one at full gradient. It was planned to make a hot startup (restart without collapsing reactor temperature gradient) from the first of these two scrams; however, a throttle valve malfunction prevented successful completion of the hot startup. Therefore, a second scram at less than full temperature gradient was performed. From this scram, a hot startup was again attempted. A low-power scram was initiated during manual resetting of the rate of temperature change computers. Following recovery from this scram, a completely successful hot startup was performed.

Following scrams three secondary throttle-valve operator failures occurred. These deficiencies have been corrected.

Core alignment and control-rod operability

Subsequent to completion of testing at 38 MW(t), a reactor core alignment test and a control-rod operability test were performed. The core alignment test consisted of removing and reinserting three fuel elements with the reactor at an isothermal temperature of about 400°F. No excess binding or loading was indicated by the fuel-handling machine load cell.

The control-rod operability test consisted of operating the rods one at a time from the control-rod test chart and recording the following data: withdrawal and insertion times, latch release time, and drop times for the first nine feet, and for full insertion. All of the above times were found to be within the limits specified in the technical specifications.

Piping displacements

During the initial rise-in-power to 38 MW(t), the primary loop No. 1 reactor outlet did not move in the expected manner during the temperature increase. Inadequate clearances were determined to exist between main heat-transfer system and service system lines. Clearances were adequately increased by trimming insulation. A general inspection of the primary pipeway revealed no further areas of interference.

During the development of full operating temperatures, the reactor outlets of primary loops 1 and 3 moved in a manner that indicated an axial restraint existed. Small bits of debris were found between the rubbing faces of the restraint saddles.

Auxiliary cooling systems

The reactor cavity, concrete shielding surrounding the reactor, pipe tunnel, equipment cells, nitrogen coolers, loading-face shield, and heating and ventilating equipment are cooled by nitrogen or water-cooling systems. The systems have all operated satisfactorily up to 15% power. All equipment and areas have been adequately cooled with the exception of a small area in the reactor biological shielding between the upper section of the reactor vessel and the first set of cooling coils. It is not anticipated that the localized high temperatures will exceed full-power design requirements.

Hydraulic performance characteristics

The hydraulic head loss characteristics of the main heat-transfer system are within less than $\pm 15\%$ of predictions.

Control system, protective system, and instrumentation performance characteristics

The control and protective systems were calibrated during pre-operational testing. The control limits, set points, control or protective action, response time and accuracy have been satisfactory and within the specified limits.

Control-rod operation

The control-rod operation was checked during pre-operational testing, at room temperature and at 540°F and after operation at 15% of power. There were no significant differences in operating characteristics. Latch release time is between 50 and 60 ms, and the rod drop time is about 960 ms.

Fuel-handling machine

The fuel-handling machine has performed very satisfactorily. The original estimate was that, during fuel loading, the machine would handle one fuel element per hour. The actual fuel-loading rate was about one fuel element for each 40 min.

Cold traps (sodium oxide)

The primary-system cold-trapping duty has been greater than originally predicted. To date, five cold traps have been removed from the primary system and one from the secondary. The traps were designed to remove 250 lb of sodium oxide. Design flow-rate through the traps is 30 gal/min. The calculated removal is about 200 to 250 lb of oxide per trap.

The flow of sodium through the cold traps is dependent on the main pumps differential pressure. At low flow rates in the main heat-transfer loops, adequate flow rates through the traps have been difficult to obtain at times.

Hot trap (carbon)

After the repair of the electric heater which burned out during the first startup of the hot trap, the trap has operated satisfactorily. The heater is able to maintain the hot-trap temperature at the desired 1200°F operating point.

Preheating system

Sodium pipe and equipment preheating is accomplished using automatically controlled electrical resistance heaters. The heaters are attached directly to the equipment to be heated. A sheet metal cover is placed over the heaters and thermal insulation applied over this cover. The heaters are controlled by thermocouples in the gap between the equipment and the cover.

Some of these thermocouples were found to read as much as 50 to 100°F below actual equipment temperatures. It was necessary to relocate some thermocouples closer to the equipment or attach them directly to get proper control. The system now operates without difficulty and is capable of preheating and maintaining equipment at desired temperatures.

Piqua Nuclear Power Facility

New coolant storage system

Organic coolant in the flaked form was added to the fuel storage tanks. The bags were carried to the hopper of the storage tank with a conveyor. The original hopper did not function properly because the coolant formed into large masses which would not pass through the hopper throat. A new hopper was fabricated with a larger throat. Modifications of the steam tracing lines, steam trapping and Viking gear pumps were also required. The system now operates satisfactorily.

Coolant purification

It was demonstrated that the purification systems will process 1000 lb/h of new coolant in accordance with design requirements. Automation control of the process was established and demonstrated to be satisfactory.

Tests were run at four lesser feed rates, 350, 500, 700 and 825 lb/h. A total of more than 35 000 lb of coolant was processed. The column condenser exceeded its design rating of 700 lb/h by condensing 825 lb/h of coolant.

After the test, when the system was run at 1000 lb/h, the column condenser condensed the 950 lb/h overhead flow without difficulty. Steam consumption was 260 lb/h of 175 lb/in² gauge steam. The predicted requirement was 140 lb/h. The condensers and the aqueous waste holdup tank were demonstrated to be adequate for the higher than anticipated steam load.

The purification column performed satisfactorily. The overhead output was a clean, light-amber coloured product having a melting-point range of 140 to 304°F. An analysis of the bottoms showed a concentration of 2% high boilers, beige in colour with a melting-point range of 250 to 370°F. The feed to the purification system contained around 0.5% high boilers, light beige in colour with a melting range of 140 to 310°F. The column showed no signs of flooding at the maximum feed rate of 825 lb/h during the formal test or at the feed rate of 1000 lb/h after the test.

Main heat-transfer system

The pressure drop in the system (including allowances for fuel elements) was lower than calculated in the original design. The maximum measured core pressure drop is 35 lb/in² compared to the predicted 44.5 lb/in². Flow rates in the single-pump mode of operation are also higher than predicted. The flow rate is now estimated to be 9000 gal/min as compared to the 7200 gal/min design value.

A portion of this excess hydraulic head is available for the recently installed in-core filters. However, current indications are that the flow rates will exceed the design goals. This additional hydraulic head is available for use in obtaining the benefits of greater linear velocities and lower fuel temperatures.

Pressurization and degasification systems

The pressurization and degasification systems operated satisfactorily. No problems were encountered in maintaining a pressure of 5 lb/in² abs. in the degasifier tank. The pressurizing pump tests proved that these pumps, with one or both in operation, can delivery 200 gal/min coolant flow against the 105-lb/in² gauge system pressure. This test also demonstrated that a scram signal, which automatically causes flow to bypass the degasifier, will initiate automatic pressurizing-pump speed reduction to a pre-set condition.

Decay-heat removal

The decay-heat removal system exceeds performance requirements. At 575°F, the system removes 2.6×10^6 BTU/h. The design requirements are 2.3×10^6 BTU/h.

Control rods

Rod drop measurements were made at 575°F and 450°F. Total drop time on each of the 12 rods tested (one rod was inoperable at the time) measured less than 700 ms. These data were well below the required total drop time of one second and compared favourably with test data which were developed during acceptance tests at the component development loop.

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DISCUSSION

N. N. ARISTARKHOV: Could you give us further information about the cause of tube failures in the heat-exchanger of the Hallam reactor? Was it because the tubes were not fastened securely enough; i. e. was it due in the first instance to vibration of the tubes?

C. A. PURSEL: Yes. The heat-exchanger in question is between the primary and secondary loops. The pressure on the secondary-loop sodium is higher than on the primary-loop sodium, so that when the leak occurred sodium flowed from the secondary loop back into the primary loop; in fact, that is how the leak was detected. One component of the inlet flow is at right angles to the ducts, and this set up harmonic vibrations in a few tubes near the inlet to the exchanger. They are now more firmly fastened and do not vibrate.

N. N. ARISTARKHOV: Is there any device in the primary loop of the Hallam reactor for cleansing the circuit of radioactive contamination due to fission products?

C. A. PURSEL: No. There are, of course, the normal traps designed to remove sodium oxide and carbon, but nothing specifically intended to remove fission products.

N. N. ARISTARKHOV: What practical steps have been taken to prevent the passage of gas into the primary loop of the HALLAM reactor? Have you considered the possibility of gas bubbles collecting in the core and affecting the reactivity? How significant might the change in reactivity be?

C. A. PURSEL: I have heard about the gas bubbles collecting on the jackets of the fuel assemblies in the BR-5 reactor. In the HALLAM reactor no provision has been made for the removal of such bubbles, should they occur, and I rather think that the problem would not be so serious as it might be in a fast breeder like EBR-2. The HALLAM reactor operates on a thermal spectrum, and the physics characteristics should not be so sensitive to changes in the gas content of the core. Since the problem has not arisen I cannot tell you whether the nuclear characteristics would be noticeably affected.

U. ZELBSTEIN: Would you agree that it might be possible to divide plant deficiencies into two broad categories: those resulting from failures of the specifically nuclear components and affecting tightness, nuclear properties, etc; and those relating to problems of engineering in general? The latter category might be further divided into problems involving new techniques and new materials, and those that merely require the application of conventional engineering principles.

C. A. PURSEL: I dealt only with the nuclear components of these plants and not with the power generating equipment. As regards the fuel there have been few, if any, difficulties apart from those I discussed in connection with the fabrication of the fuel for the ELK RIVER reactor.

Of course, these being nuclear plants, some new materials are bound to be used, and new uses made of more conventional materials; by and large, however, the problems that have arisen could have been dealt with merely by applying classical engineering principles. I believe it is M. Conte's paper* that suggests that engineers are devoting so much attention to the strictly nuclear components of their plants that the more conventional portions are perhaps being neglected. In any case, with the benefit of hindsight, I believe that most of the difficulties encountered in these plants could have been avoided by sound classical engineering.

* CONTE, F. et al., these Proceedings.

COMMISSIONING EXPERIENCE FROM THE ÅGESTA NUCLEAR POWER PLANT

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Abstract — Résumé — Аннотация — Resumen

COMMISSIONING EXPERIENCE FROM THE ÅGESTA NUCLEAR POWER PLANT. The Ågesta Nuclear Power Plant is a pressurized heavy water reactor of the pressure vessel type, fuelled with natural uranium. It was commissioned with light water from December 1962 to May 1963.

Observations of a more general interest were made during this commissioning essentially on the following topics: (a) cleanliness of primary circuit (b) valve operation (c) pressurization of the primary circuit (d) water leakage (e) refuelling machinery (f) containment testing.

EXPÉRIENCE ACQUISE LORS DES ESSAIS DE MISE EN SERVICE DE LA CENTRALE NUCLÉAIRE D'ÅGESTA. Il s'agit d'un réacteur à uranium naturel et à eau lourde pressurisée, du type à caisson sous pression. Les essais de mise en service ont été faits avec de l'eau ordinaire, de décembre 1962 à mai 1963.

La mise en service a permis de faire des observations d'intérêt général sur les sujets suivants: a) non-contamination du circuit primaire; b) fonctionnement des vannes; c) maintien sous pression du circuit primaire; d) fuites d'eau; e) appareils de chargement du combustible; f) essais d'isolement.

ОПЫТ ПО ВВЕДЕНИЮ В ЭКСПЛУАТАЦИЮ ЯДЕРНОЙ ЭНЕРГЕТИЧЕСКОЙ УСТАНОВКИ ÅGESTA. Ядерная энергетическая установка Ågesta представляет собой тяжеловодный реактор под давлением, использующий природный уран в качестве топлива. Реактор был введен в эксплуатацию на обычной воде в период с декабря 1962 года по май 1963 года.

Замечания более общего характера были сделаны во время эксплуатации в основном по следующим темам: а) чистота первичного контура; б) работа клапанов; в) повышение давления в первичном контуре; г) утечка воды; д) установка по замене топлива; е) испытание на герметичность.

EXPERIENCIA ADQUIRIDA CON LA PUESTA EN MARCHA DE LA CENTRAL NUCLEOELECTRICA DE ÅGESTA. La central nucleoelectrica de Ågesta posee un reactor de agua pesada del tipo de recipiente de presión, con combustible de uranio natural. Se mantuvo en funcionamiento con agua ligera entre diciembre de 1962 y mayo de 1963.

Durante esta prueba, se efectuaron observaciones de interés más general, relacionadas esencialmente con las siguientes cuestiones: a) limpieza del circuito primario; b) funcionamiento de las válvulas; c) presión del circuito primario; d) pérdidas de agua; e) disposiciones de reposición del combustible; f) ensayos de confinamiento.

EXPERIENCE OF COMMISSIONING

The Ågesta Nuclear Power Plant is a heavy-water-moderated and cooled pressurized water reactor of the pressure-vessel type with natural uranium oxide as fuel.

It is designed for the combined production of electrical power (10 MW) and heating water for household use (55 MW). A highly simplified flow diagram of its major circuits is given in Fig. 1 and the site layout in Fig. 2

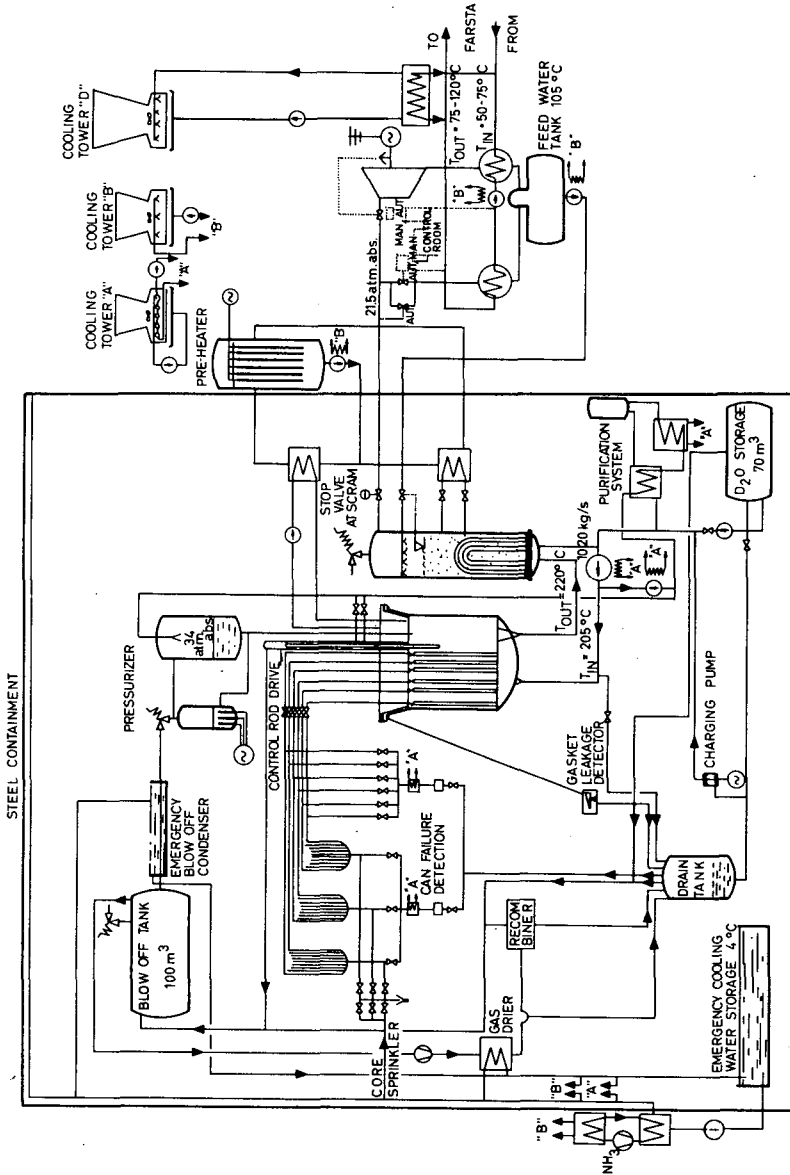


Fig. 1
Ågesta Nuclear Power Station
Simplified circuit diagram.

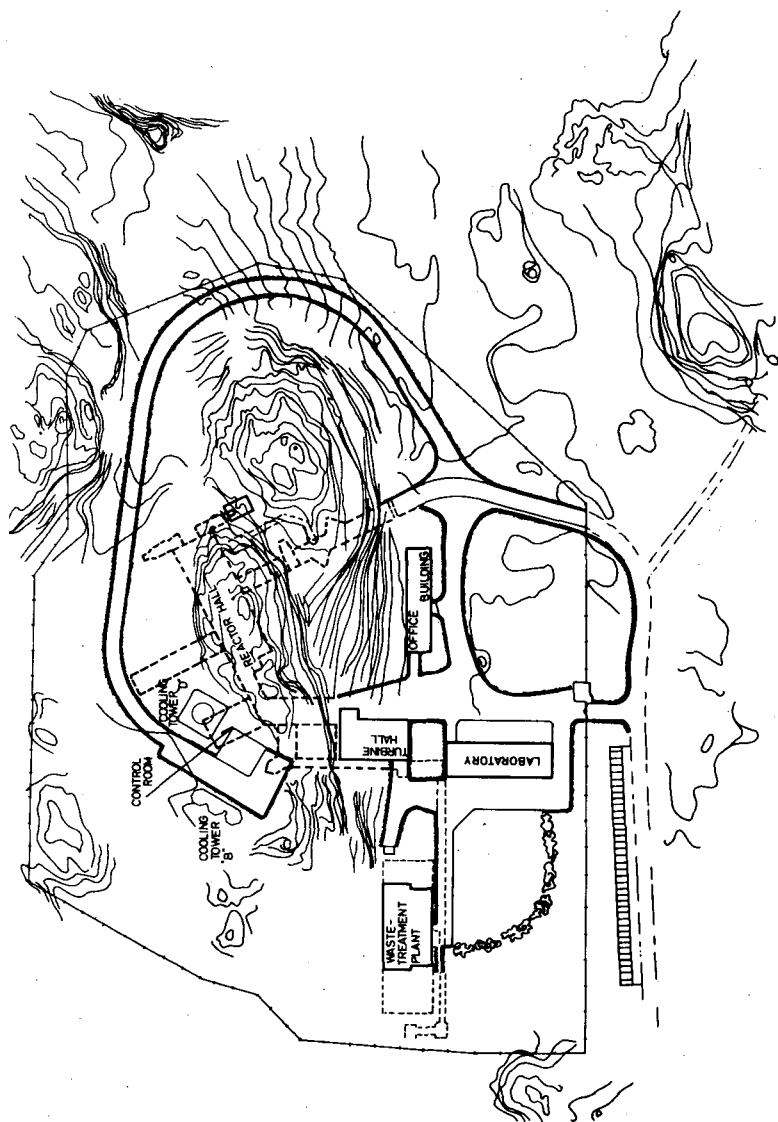


Fig. 2
Ågesta Nuclear Power Station
Plant site.

COMMISSIONING PROGRAMME

The process of putting the plant into commission, using light water, began in December 1962. Commissioning tests have been run parallel with the final stages of the installation work in the plant. The commissioning programme contained the following major phases for the primary circuits and, with appropriate omissions, also for the auxiliary circuits:

- (a) Exchange of atmosphere in the circuits by means of evacuation and subsequent filling with nitrogen;
- (b) Rinsing the circuits with water;
- (c) Fuel loading;
- (d) Water filling and hydrostatic testing;
- (e) Component and system tests with water at room temperature;
- (f) Heating the plant to operating temperature;
- (g) Component and system tests at operating temperature.

Instrumentation and power supply tests were run parallel to the main programme as a natural off-shoot. In this connection those tests where the external power supply was disconnected were of special interest. The containment leakage test falls outside this circuit commissioning programme, but is an important part of plant commissioning. The progress of the commissioning can be seen from Table I.

SPECIAL POINTS OF INTEREST

1. Cleanliness of the primary circuit

The water rinsing, although of little scientific or technological interest, is an important prerequisite for the subsequent tests of mechanical equipment.

The primary circuits have been installed under clean conditions. These require that the work area is sealed off from the outside, that the ventilating air is filtered, that all people entering wear coveralls and change shoes, that all equipment brought in has been previously cleaned and is kept clean while it is in the clean area and that all loose material is kept under control. It is, however, extremely difficult to fulfil these requirements stringently in actual practice, without completely stopping all activities in the clean area.

Clothes are worn out by use, wooden boards wear when walked on and welding produces fumes which penetrate everywhere. It is also difficult to maintain cleanliness discipline in the work force over a period of more than a year.

The clean area was maintained at Ägesta with reasonable rigour, but without drastic measures. It was therefore far from clinically clean. The rinsing of the primary circuits proved, however, that the cleanliness measures had been well balanced. The rinsing water was passed through filters. As little as a few grammes of dry material was collected there, and this proves that very little dirt existed in these systems, even if the walls of the reactor vessel and some other components could not be very effectively flushed by the method used. Also further experience with, and inspections of, these systems indicated no presence of dirt from installation work.

TABLE I
COMMISSIONING OF
THE ÅGESTA NUCLEAR POWER PLANT

	Nov.	Dec.	Jan.	Feb.	Mar.	Apr.	May
Station water supply	—/						
Auxiliary cooling circuits	/—	—/	/				
<u>Primary circuit</u>							
Atmosphere exchange	/—	/					
Water rinsing		/—					
Fuel loading		/—					
Water filling		/—					
Cold tests with pumps and control rods		/—	/				
Hydrostatic pressure test			/—				
Test of approach to critical procedure			/—				
Station heating test 20°C - 100°C - 20°C				/—			
Station heating test 20°C - 150°C - 20°C				/—			
Test programme halt due to failure of preheater circuit				/—			
Station heating test 20°C - 180°C - 20°C					/—		
Test programme halt due to primary pump failure					/—		
Station heating test 20°C - 215°C - 20°C						/—	
High-temperature commissioning, pump operation, control-rod tests						/—	/—
Halt for reinstallation of pump						/—	
Ventilation tests						/—	/—

There is no doubt that the extra costs for the clean area work have paid very well already during the commissioning, since no equipment malfunction has been caused by foreign bodies in the system.

2. Primary circuit pumps

The primary circuits contain 15 pumps of which 13 are centrifugal and 2 are piston pumps. The four main circulators are canned motor pumps, whereas the remaining nine are of the canned rotor type. Unsatisfactory manufacturing techniques had been used for the stator windings in one of the main circulators. This caused a break in one of the phases in the stator. This pump had to be disconnected from the circuit, dismantled and the stator body sent to the manufacturer for rewinding. This caused a delay in the commissioning and is not likely to recur, but it has been the only case of pump malfunction during the commissioning.

3. Valve operation

All valves with connections up to 50 mm are of bellows type, all bigger valves (25) have double stuffing boxes with drainage. In general the valves have performed well. Only three of them show a leakage of any significance across the seat when the valve is closed. About a third of the valves are manually operated from a remote position through mechanical links. Our experience with this type of transmission is poor and many of them had to be redesigned as a result of tests in cold condition. Some of those which must function both with hot and cold circuits have given almost insoluble problems, whether due to the links or to the valves or to a combination of the two. It is obvious that such links must be thoroughly tested for all possible angles between incoming and outgoing shafts and for possible movements of the valve shaft with pipe expansion and that the experience from such tests must be followed when the transmissions are installed in the station. This may, however, be very difficult in actual practice, when the links are installed after the jungle of pipes is installed. At that moment an unforeseen pipe or ventilation duct or other obstacle may necessitate a change of the linkage. Our experience is rather that it is both cheaper and safer to make all valves, which must be remotely operated, either pneumatic or motor driven, and avoid mechanical linkage entirely.

4. Control rod drives

The control rods are moved in steps by a hydraulic system with programme-operated solenoid three-way valves supplying the pressure to the three hydraulic connections for holding, gripping and lifting. An analysis of the function of the system shows many possible sources for faults, e.g. a leakage in the two mechanical joints in the mechanism and past the sealing rings in the clutches, clogging of the mechanism due to crud deposition, or sticking due to the long narrow clearances etc. Our experience is, however, so far very good, the steps are regular, the fall times equal, the necessary pressures to hold and lift the rod are smaller than expected and

the leakage within tolerable limits. Inspection of eight rod mechanisms after four months in the system showed that they were perfectly clean and free from crud depositions. This is probably due to the settling effect on the crud in the long hydraulic pipes through which only a minute flow passes. It is probable that increasing leakage due to wear of the graphite seal rings in the clutches will ultimately set the life-time between maintenance periods. Efforts are therefore made to replace the seal rings with bellows.

5. Pressurization of the primary circuit

The primary loop is pressurized at operating temperature by steam in a pressurizer vessel. The steam is produced in an electrically heated boiler, with the power supply, 336 kW, regulated from pressure signals. The pressure is maintained at 34 atm, i.e. 11 atm above the saturation pressure of the heavy water when it leaves the reactor. This effectively suppresses boiling in the pumps.

The situation is, however, different at lower temperatures, since the heavy-water mass is constant, and the volume thus decreases with temperature. The heavy water will actually only just cover the core at room temperature and will not reach the top lid until at about 130°C or higher, depending on the initial water level.

The high overpressure maintained at operating temperature cannot be applied when the water level is low, since the steam condenses on the large cold-water surface and on the cold lid walls, and thus cannot contribute to any overpressure relative to the saturation pressure of the water. The overpressure must thus be produced by a partial gas pressure. This prevents the water from boiling but saturates the water with gas instead. Gas is released in all regions where the pressure is lower than at the water surface, and particularly at the suction side of the pump. The worst situation occurs during the heating of the primary circuit. The water level is low, the pressure margin against boiling is minute and the primary water is heated in the heat exchangers before it enters the pumps. The circulation may then easily become unstable. A slight drop in flow, e.g. due to gas release in the pump, gives a higher temperature rise in the heat exchanger and causes boiling in the pump, which further reduces the flow etc. There would not be any problems with the pressurizing if, as in light-water reactors, the primary circuits could be filled up to the operating level at room temperature, and the temperature expansion could be compensated for by the drainage of the surplus water. But this would require a considerable extra heavy-water investment. Experience from the commissioning shows, however, that a moderate increase in the originally specified room-temperature level, is enough to bring the water level up in the lid below 130°C or so, where the temperature margins start to be dangerously small with the rate of heating used. While the pressurizing problems for routine operation can be thus handled, it will be difficult to carry out the reactor physics programme, which calls for prolonged operation with low water levels at elevated temperatures. It is not yet clear by which method boiling or gas release could be prevented in this situation, but it may be that a possible solution is to do all the physical measurements at high temperature with the water slowly cooled at a constant rate rather than heated.

6. Water leakage

The amount of water leakage is of particular importance in heavy-water reactors both because of the heavy-water price and of the tritium hazard. A very high degree of leak-tightness has been aimed at in Ägesta. All piping is welded, and most of the valves are of the bellows type. Where gaskets or stuffing boxes have been unavoidable they are always doubled with drainage between the seals. All welds have been radiographed and the majority leak-tested with helium. The water leakage from the primary system is, however, one of the most difficult plant data to establish. It can only be obtained by measurements of the level of the accessible free water surfaces, and only when the effective mean temperature of the water is known. This temperature is in practice only known with sufficient accuracy at room temperature. It is thus only by a comparison of levels before and after a period of high-temperature operation that a change in water content can be established.

But even then there are large sources of error. It is almost impossible to keep a complicated piping system completely filled with water, particularly when, for pressure reasons, it is saturated with gas at the higher temperatures. Gas may be trapped in the water space and water may be left in the gas space in the top of the primary system and particularly in the blanket gas space in the low-pressure parts of the primary system.

An effort was made to establish the net water losses during two high-temperature periods during the commissioning. The first was made at the first run at full temperature, but before the valve gaskets had been drawn tight. The apparent losses were 4.6 l/h, but a separate measurement showed that most of this "loss" was due to gas volumes which had been driven out from the system and replaced with water. Only 0.75 l/h could not be traced in this way and most of this had been dripping out through loosely drawn gaskets.

During the second hot run under more appropriate conditions for leakage observations and after the gaskets had been drawn more tightly, the loss was reduced to 0.25 l/h. The leakage through stuffing boxes and gaskets will be returned to the heavy-water drainage tank during reactor operation. This return system was, however, deliberately left open during the commissioning to permit better localization of leaking gaskets and an unknown fraction of the 0.25 l/h will be returned to the system when these leakage paths are closed. Already a loss of 0.25 l/h would be economically acceptable, but there is every reason to predict considerably lower effective losses when the system is properly closed and all water circuits are well filled.

Internal leaks from the high-pressure parts through, for example, safety valves and drain valves, to low-pressure tanks may also be of concern since the leakage must be cooled and the cooling capacity will limit the amount of leakage that can be tolerated. The safety valves are presumed to be the weakest type of component in this respect, but after the safety valves had been set to correct relief pressures at operating temperatures, no leakage of any concern could be detected. The development of leakage with time for these components will, however, be of considerable interest in the future.

7. Pressure balance between heavy-water and light-water circuits

All components where heavy and light water are separated by a single wall provide possible sources for direct contact between the two, in the event of a rupture occurring or a crack developing. A leakage out of heavy water is then of far smaller economic consequence than a leakage in of light water. The light-water circuits of this type are therefore kept at lower pressure than the adjacent heavy-water circuits. This is difficult when the plant is cold since large parts of the light-water circuits must then be kept at pressures below atmospheric. The consequences of this were not considered carefully enough in the design stage and conventional components without special leak-tightness qualities were used. Considerable difficulties have been met with these circuits, due to leakage in of air, which is trapped at high-level positions or in pockets where the pressure is low, e.g. at the suction side of the pumps. It collects there until the circulation becomes unsteady. A methodical sealing of all components concerned, such as drain valves, safety valves and pump glands, is under way.

8. Refuelling machinery

Refuelling of the Ågesta reactor is only carried out at atmospheric pressure with the reactor shut down. The vital parts to be tested on this machinery are the fuel-handling mechanism, the decay-heat cooling system, and the electrical interlock system.

The entire refuelling procedure is programmed by an interlock system which prevents the operator from deviating in any respect from the prescribed order of refuelling manoeuvres. Each step is, however, initiated by the operator. The interlocking system thus senses the position of the machine, e.g. whether it is above the reactor or above the storage pits, and locks the fuel grab until it reaches the level of the fuel hook corresponding to the position of the machine etc. The interlocking system is built up of simple mechanical devices and electromechanical relays. This conventional design approach has been very appropriate and resulted in a quite reliable product which, from the limited experience obtained so far, works very well.

The obvious drawback with an interlocked sequence of operations is, of course, that a fuel-loading situation, which the machine is not prepared for, is very difficult to handle. Such situations appear especially during the commissioning and we have had our share of these artificial difficulties, which have taught us that it is wiser, at least with this machine, to arrange the fuelling situation to fit the equipment rather than arrange the equipment to fit the situation.

The decay heat is carried away from the fuel by a forced flow of nitrogen and delivered to a water loop which is ultimately cooled by the reactor hall atmosphere. The nitrogen is circulated by fans, which are of the high-speed type in order to keep down their dimensions and thus also the dimensions of the radiation shield. These fans were originally ordered with gas bearings, but the technological problems with these bearings could not be solved. The design was therefore changed to ball bearings which at these speeds required oil lubrication. No oil leakage to the nitrogen flow could be tolerated because

of the cleanliness requirements on the fuel cans. A barrier gas flow through a labyrinth seal was used as a seal for the oil. It proved, however, to be difficult to prevent leakage effectively, and only after a thorough measurement had been made of the pressure distribution in and around the fan and the loop could the barrier gas in the labyrinth seal be made to function correctly.

9. Containment testing

The primary circuit and fuel storages are located in a cavern blasted in granite rock with about 20 m rock coverage. The proximity of the plant to Stockholm about 17 km from the city centre combined with the poor quality of the rock led to the construction of a steel lining inside the rock. The maximum credible accident has been given a very conservative definition as a total melt-down of the core combined with the release of steam from a break of the primary system. This puts the maximum permissible leak rate at the low value of $2 \times 10^{-4} \text{ h}^{-1}$ at 3 atm abs. pressure. The containment leakage test took 23 d and consisted of measurements of the leak rate at 3, 2.5, 2 and 1.3 atm abs.

This series of measurements was carried through in order to establish the leak-rate versus pressure, so that future containment tests can be carried out at lower pressures.

The test was an immediate success since already the first series of measurements gave acceptable results. Figs. 3, 4 and 5 illustrate the test procedure and results. The measured leak rate at 3 atm abs. was $1.51 \times 10^{-4} \pm 0.03/\text{h}$ and thus under the specified limit.

The chief reason why the containment proved to be tight enough already on the first test, is certainly that all segments of the steel shell had been individually soap-bubble tested, and that all air locks and pipe penetrations

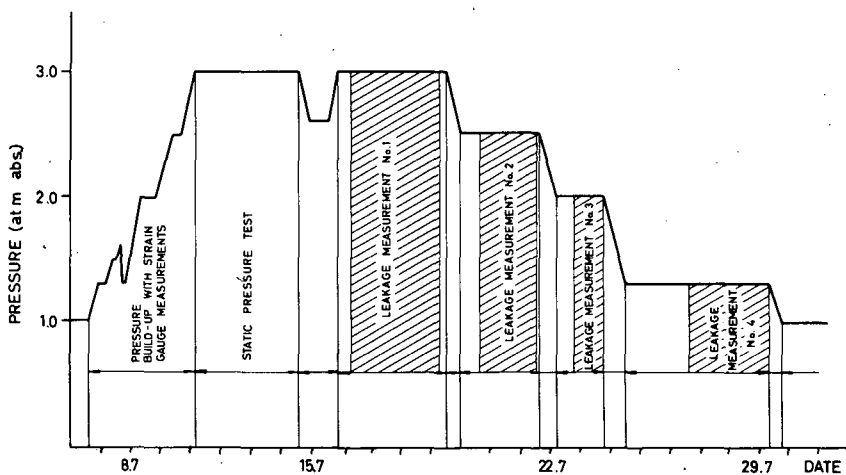


Fig. 3

Ägesta Nuclear Power Station.
Containment test.
Test proceeding.

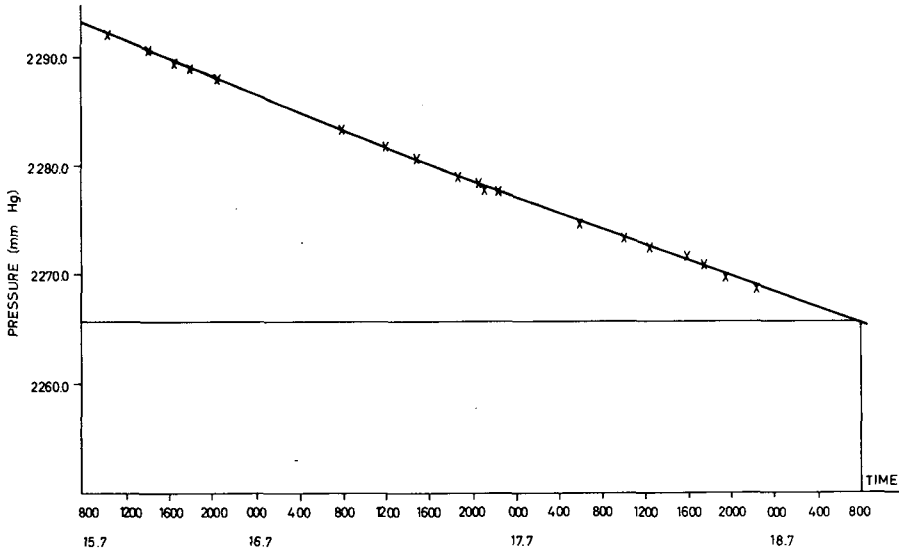


Fig. 4

Ågesta Nuclear Power Station
Containment test.

Pressure decrease with time during run No. 1

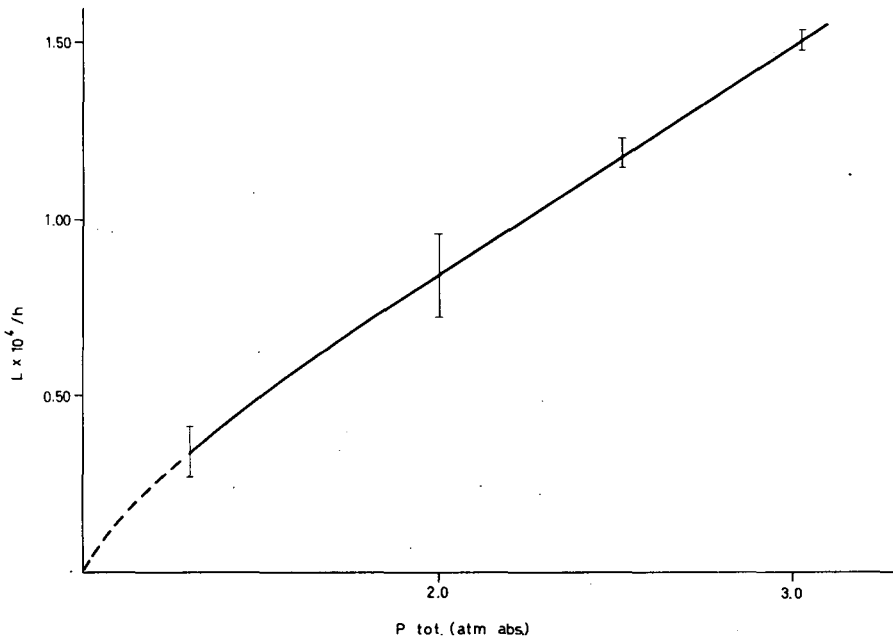


Fig. 5

Ågesta Nuclear Power Station.
Containment test.

Leakage factor as a function of pressure.

with their shut-off valves, and a statistical sample of the cable penetrations had been tested before and improved where necessary.

CONCLUSION

The commissioning period has been a very demanding time, since it has been run in parallel with the final stages of the plant construction. A rather small number of more serious equipment failures or breakdowns and a great many occasions of minor malfunctioning in both cases on conventional equipment in light-water circuits have constantly acted as brakes on the progress of commissioning and on three occasions caused interruptions in the work. The test programme has, however, on the whole succeeded very well and has demonstrated that the requirements that heavy - water operation put on circuits and components have been sufficiently well met. The plant has also worked well as an integrated unit during the hot commissioning stages.

BIBLIOGRAPHY

A rather detailed description of the plant is given in Nucl. Power, 8 No. 83 (1963) 40 - 52.

INITIAL OPERATING EXPERIENCE WITH THE "NPD" REACTOR

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Abstract — Résumé — Аннотация — Resumen

INITIAL OPERATING EXPERIENCE WITH THE NPD REACTOR. Canada's first nuclear power station, the Nuclear Power Demonstration station (NPD), is intended to serve as a means of proof-testing the performance of the Canadian type of station using natural uranium as fuel and heavy water as moderator and coolant. It reached full power on 28 June 1962. Although designed for base-load operation it will, during the early stages, be operated part of the time on high-capacity runs and part of the time on improvement periods. Progress has been favourable so far; the first high-capacity run of six weeks' duration yielded a capacity factor of 70%.

Improvements already made have increased safety, improved performance and demonstrated potential methods of capital-cost reduction for future stations. For example, shaft seals on primary coolant pumps have been modified for better performance, freezer-type vapour recovery equipment has been replaced in favour of absorption columns to reduce heavy-water vapour loss, and flow limiters are being installed in sample lines to reduce losses of heavy water in the event of joint failures.

During December 1962 two simultaneous leaks from the on-power refuelling machine led to an unusual sequence of events in which a considerable amount of hot high-pressure heavy water was spilled into the reactor vault where it suffered a slight downgrading in isotopic purity. It was upgraded and the reactor returned to operation by the end of the month. All safety devices operated correctly during the incident as did the provisions for containment of heavy water.

EXPÉRIENCE RECUEILLIE PENDANT LES PREMIERS MOIS DE FONCTIONNEMENT DU RÉACTEUR NPD. La première centrale nucléaire du Canada, NPD, est une centrale de démonstration, qui doit servir à vérifier les performances des réacteurs fonctionnant à l'uranium naturel et utilisant de l'eau lourde comme ralentisseur et comme fluide de refroidissement. Elle a atteint sa pleine puissance le 28 juin 1962 bien que conçue pour être exploitée comme centrale de base, elle fonctionnera au début comme centrale d'appoint, ce qui permettra d'y apporter des perfectionnements pendant les périodes d'arrêt complet. Les résultats obtenus jusqu'à présent ont été positifs; la première expérience de fonctionnement, qui a duré six semaines, a permis d'obtenir un facteur de puissance de 70%.

Les perfectionnements déjà apportés ont permis d'augmenter la sécurité, d'améliorer les performances et ont montré en même temps qu'il était possible de réduire les dépenses d'investissement pour les centrales futures. On a, par exemple, modifié les joints d'arbre des pompes du circuit primaire de refroidissement pour obtenir de meilleures performances; les appareils de récupération de la vapeur, à congélation, ont été remplacés par des colonnes d'absorption de façon à réduire les pertes de vapeur d'eau lourde; des régulateurs de débit sont installés en certains points pour réduire les pertes d'eau lourde pour le cas où des joints céderaient.

En décembre 1962, deux fuites simultanées dans l'appareil de rechargement en puissance ont entraîné une série inhabituelle d'incidents; une quantité importante d'eau lourde à haute pression et haute température a été projetée dans l'enceinte du réacteur où sa pureté isotopique a été légèrement altérée. On a redonné à l'eau lourde la pureté voulue et le réacteur a pu recommencer à fonctionner à la fin du mois. Tous les dispositifs de sécurité, notamment ceux destinés à parer aux fuites d'eau lourde, ont fonctionné correctement pendant l'accident.

ПЕРВЫЙ ОПЫТ ПО ЭКСПЛУАТАЦИИ РЕАКТОРА NPD. Рассматривается первая ядерная электростанция Канады, демонстрационная электростанция, предназначенная для испытания эксплуатационных качеств станций канадского типа, использующих в качестве топлива природный уран, а в качестве замедлителя и теплоносителя - тяжелую воду. 28 июня 1962 года эта станция достигла полной мощности. Несмотря на то, что она проектировалась для работы на базовой нагрузке, в течение первых стадий

она будет часть времени работать на высоких уровнях мощности и часть времени - на пониженных мощностях для внесения усовершенствований. До сих пор наблюдался прогресс; первый период работы на высокой мощности в течение шести недель дал коэффициент использования на мощность, равный 70%.

Улучшения, которые были достигнуты, увеличили безопасность, повысили эксплуатационные характеристики и продемонстрировали потенциальные методы снижения капитальных затрат для будущих станций. Так, например, в целях улучшения эксплуатационных качеств были видоизменены уплотнения валов на насосах охладителя первого контура; оборудование типа холодильников, использовавшееся для регенерации пара, было заменено поглощающими колонками с целью уменьшения потери пара в тяжелой воде. Устанавливаются также ограничители потока воды в линиях для взятия проб с целью уменьшения потери тяжелой воды в случаях неисправностей в соединениях.

В декабре 1962 года две одновременные утечки из машины, производящей загрузку топлива, привели к необычному обстоятельству, при котором значительное количество горячей тяжелой воды под высоким давлением попало в камеру реактора, где произошло незначительное уменьшение ее изотопной чистоты, которая затем была превышена и реактор был вновь пущен в конце месяца. Во время аварии все устройства по безопасности работали точно и обеспечили удержание тяжелой воды.

EXPERIENCIA INICIAL DE FUNCIONAMIENTO DEL REACTOR NPD. La primera central nucleoelectrica del Canadá, NPD, constituye una instalación de demostración, destinada a comprobar el funcionamiento de los reactores alimentados con uranio natural y moderados y refrigerados por agua pesada. Alcanzó su régimen normal de potencia el 28 de junio de 1962. Aunque ha sido diseñada como central para la carga básica, en las primeras fases funcionará en parte como central para la carga de cresta, lo que permitirá introducir mejoras durante los períodos de paro. Los resultados obtenidos hasta el presente fueron satisfactorios; el primer ensayo de explotación, que duró seis semanas, permitió alcanzar un factor de potencia del 70%.

Los perfeccionamientos introducidos han permitido mejorar el grado de seguridad y el rendimiento, y han revelado la posibilidad de reducir los gastos de inversión en centrales futuras. Por ejemplo, se han modificado las empaquetaduras de las bombas del circuito primario de refrigeración para alcanzar mayor rendimiento; el equipo de recuperación de vapor por congelación ha sido sustituido por columnas de absorción a fin de reducir las pérdidas de vapor de agua pesada; se están instalando limitadores de flujo en algunas tuberías para disminuir las pérdidas de agua pesada en el caso de averías en las juntas.

En diciembre de 1962, dos escapes simultáneos que se produjeron en el aparato de realimentación del reactor en marcha dieron origen a una serie de incidentes poco comunes; una cantidad considerable de agua pesada a alta presión y temperatura se derramó en el recinto del reactor, alterándose ligeramente su pureza isotópica. Luego de purificarse el agua, el reactor pudo reiniciar su funcionamiento a fines del mes. Todos los dispositivos de seguridad y en especial los destinados a evitar los escapes de agua pesada funcionaron correctamente durante el incidente.

Canada's first nuclear-electric station, the "Nuclear Power Demonstration" station, is intended to demonstrate the Canadian type of station which utilizes natural uranium as fuel and heavy water as both moderator and coolant. In particular, the station is intended to demonstrate high base-load capacity, continuity of service and safety. In addition, it is providing information applicable to the design, construction and operation of similar but larger nuclear-electric stations including information on fuel-element performance, fluid loss, power regulation, response to load demand, response to system disturbances, reliability of components, operating techniques, maintenance methods and last but not least "cost data" all of which will add to the 20 yr of Canada experience with this type of reactor.

First electricity was generated on 4 June 1962 followed by a smooth increase to full power on 28 June 1962. Next, the station was subjected to performance trials with associated modifications and the station was declared "In-Service" on 25 September 1962.

In this presentation, a brief description of the station is given first for those persons not familiar with the design. The following subject matter deals primarily with the commissioning and early operation.

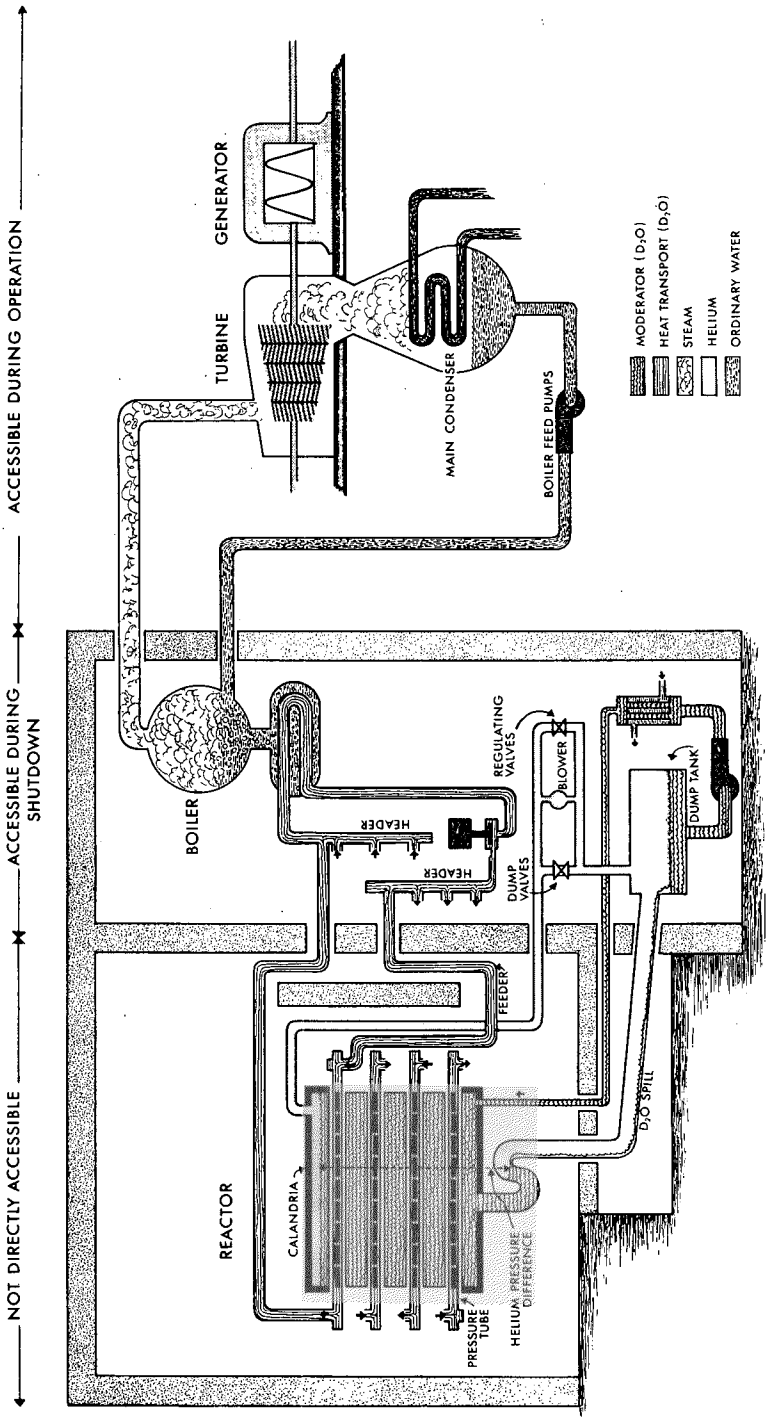


Fig. 1
Description of the NPD REACTOR

DESCRIPTION OF STATION (see Fig. 1)

The heat generated in the reactor is transported to the boiler using heavy water as the heat transport agent. The boiler is essentially a heat exchanger and steam drum in which steam is generated from ordinary water to drive the turbine-generator.

The reactor utilizes an aluminium tank called the calandria. This tank, approximately 4.6 m (15 ft) long and 5.2 m (17 ft) in diam, holds the heavy-water moderator and is arranged with its longitudinal axis in the horizontal. Through this tank run 132 horizontal, aluminium tubes through which are inserted zirconium-alloy pressure tubes. Each pressure tube is loaded with 9 fuel bundles with each bundle containing about 15 kg (33 lb) of uranium dioxide. The total fuel charge is approximately 17.8 t (metric) of uranium dioxide.

Heavy water, pressurized to 28 kg/cm² (1050 lb/in² gauge), is increased in temperature from 252°C (485°F) to 275°C (530°F) as it passes through the pressure tubes. Steam is generated from ordinary water at a pressure of 28.8 kg/cm² (410 lb/in² gauge) and a temperature of 232°C (450°F).

The dry and saturated steam is used to generate approximately 20 000 kW net electricity.

The control system does not require any moving parts in the reactor. The heavy-water moderator is supported in the calandria by a helium gas differential pressure provided by blowers. Power regulation is achieved by varying the differential gas pressure which in turn determines the heavy-water level. The reactor power is thus regulated through control of neutron leakage.

To shut down the reactor, the helium pressure differential is reduced to zero by opening helium dump valves, thereby dumping the moderator into the dump tank. The automatic protective system, which initiates such a dump, includes 22 trip (scram) functions and utilizes the "2 out of 3" method. The dump valves, regulating valves and blowers are readily accessible with the reactor shut-down.

Two fuelling machines are utilized to change fuel as illustrated in Fig. 2. One fuelling machine pushes new fuel bundles into one end of a pressure tube and spent bundles are at the same time pushed into a fuelling machine coupled on to the other end of the same tube. These fuelling machines are operated from the control room and are intended to change fuel under on-power conditions.

COMMISSIONING

The commissioning programme was classified into four time phases:

Phase A - Pre-critical

Phase B - Low-power reactor measurements

Phase C - Power run-up

Phase D - Full power tests

The Phase A work started before construction was complete. During the early part of this period the service systems such as water, compressed air, electricity and ventilation were placed in operation.

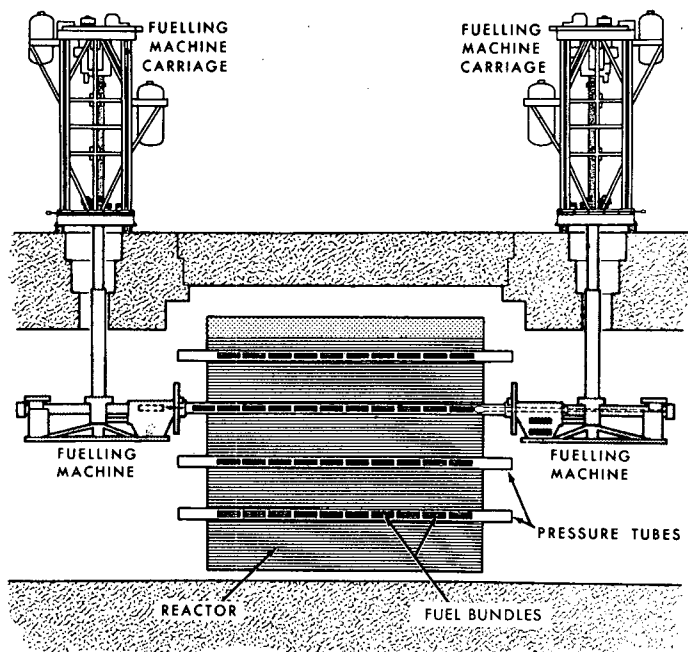


Fig. 2

Description of the two fuelling machines.

As construction neared completion, the process systems were thoroughly tested as far as practical before heavy-water addition. The heavy-water and helium systems were leak tested with helium before heavy water addition. The hydrostatic and performance tests were performed directly with heavy-water (no light-water tests were performed). This method proved to be very successful and modifications necessary were performed with little difficulty. Construction was essentially completed by 1 January, 1962.

Following the tests on the moderator system, reactor regulating system, protective systems and fuelling machines the heavy water was secured in the dump tank. The initial fuel, consisting of natural uranium and depleted uranium, was then charged into the reactor. The operation was performed remotely from the control room, between 28 February, 1962 and 21 March, 1962, and very little difficulty was encountered. The start-up apparatus was checked out with a polonium-beryllium neutron source. This phase A ended with the approach to critical 11 April, 1962 approximately $3\frac{1}{2}$ months after completion of construction.

Phase B included reactivity measurements at low reactor power. To calibrate the moderator level versus reactivity, cadmium sulphate was injected into the heat-transport system while the reactor was operated under automatic control at a constant low-power level. With one exception, all measurements essentially confirmed the design data. A discrepancy in initial critical height was caused by an underestimation of the effect of depleted uranium. The depleted uranium had been added to permit early

operation at full power and the reasons for this discrepancy have now been fully resolved. This phase ended on 8 May 1962 taking approximately 1 month.

During Phase C, the reactor was operated at power levels of 6%, 10%, 25%, 50%, 75% and 100% during which tests were performed to establish dynamic performance of the reactor-boiler and turbine-generator. Also measurements were conducted to establish the adequacy of cooling, shielding, recombination, etc. Generally, the power run-up went quite smoothly and no major difficulties were encountered. First electricity was generated 4 June and full power was reached 28 June 1962.

During Phase D (power tests), the station was modified to eliminate a number of early problems which caused poor performance. The station was declared In-Service on 25 September 1962. At this time, the fuelling machines had only been used for off-power fuelling and development work for on-power fuelling was still in progress.

Although the commissioning programme went smoothly, a number of problems were encountered as is usual with a new station. The following are a few items arbitrarily selected for discussion.

One of the problems encountered was achieving adequate leak-tightness of the vault ventilation system. Numerous leaks were encountered through and around wall penetrations. Several techniques were used to correct such leaks but, without doubt, the use of epoxy and fibreglass was most successful in achieving an acceptable leak rate of 40 l/min.

Gas locking of the moderator pumps was caused by helium entrainment when the dump tank was operated at low level. This problem was corrected by spilling the heavy water into the end of the dump tank permitting the gas to evolve before reaching the pump suction at the centre of the tank.

Heavy water in the heat-transport system is cooled and depressurized before passing through ion-exchange columns for purification. Gas locking was experienced in the charging pumps which return the purified heavy water back into the pressurized system. This problem was caused by the reduction in solubility of helium with reduced temperature and pressure conditions. This was corrected by providing degassing facilities.

The high-pressure heavy-water systems are generally fabricated using carbon steel piping and instrument tubing. This method has proved good with one exception, where they were cast into concrete and exposed to water permeating into cracks surrounding the tubes. Such tubes were replaced with stainless steel.

Ice plugs have been extensively used to isolate heavy-water lines where no valves have been provided. Such plugs can be made quickly, effectively and inexpensively using liquid nitrogen, solid CO₂, plastic sheeting and paper tape.

One major weakness identified during commissioning was poor quality workmanship in the mechanical joints of heavy-water instrument tubing. This problem can be avoided by adopting proper quality control methods for mechanical joints as stringent as those required for welded joints.

No measureable reduction in the isotopic content of the heavy water took place in either the moderator or heat-transport system due to down-grading during commissioning. The total inventory of heavy water required

was 74 400 kg (164 000 lb) as contrasted with the original estimate of 72 600 kg (160 000 lb). The total loss during the commissioning period was 5.4% which is considered economically acceptable and less serious than that predicted. The major loss during commissioning was caused by chronic vapour loss from numerous small leaks rather than acute spills. The heavy-water loss averaged 16 kg/d (35 lb/d) in the early stages of operation.

OPERATION

The station is designed as a base-load unit with an 80%-capacity factor objective. Therefore, it is necessary to operate the station for continued periods to obtain meaningful statistical data. Alternately, we consider it vital to deliberately shut down the station to incorporate modifications aimed at improved performance, safety etc. and to obtain further test data. Because these two requirements are not naturally compatible, we alternate between "capacity runs" and "improvement/test periods" to satisfy both objectives.

A capacity factor of 70% was achieved during the first 6 weeks capacity run from 1 October 1962 to 12 November 1962. Approximately half of the lost production was due to the reactor-boiler and half was due to a problem with the emergency stop valves on the turbine generator.

On 3 December 1962 a major problem was encountered during on-power proof testing of the refuelling machines. Two simultaneous leaks developed on one of the machines, one at the point where the machine connects to the reactor pressure tube and the other in one of the hydraulic control lines of the machine itself. Because of the first leak, hot high-pressure heavy-water coolant was released into the reactor vault, and because of the second the machine could not be moved to rectify the first. Gratifyingly, the automatic spray of cold heavy water into the reactor vault was actuated in just the fashion intended to reduce vault pressure in such emergencies, and indeed all safety devices and provisions for containment of heavy water performed successfully. Although the spilled heavy water suffered a small downgrading in isotopic purity, operation was resumed at the end of December. Further on-power trials are scheduled for mid-1963.

A capacity factor of 100% was achieved during the second 6 weeks capacity run from 12 February 1963 to 25 March 1963. Further operation is required to establish statistically significant conclusions. However, we have no reason to believe that we will not meet or exceed our capacity factor target of 80%.

At the time of preparing this paper, we were in the second improvement/test period which started on 25 March 1963. Some of the problems requiring solution are discussed later. The following is a general discussion of the behaviour of the station.

The average burn-up of the fuel is too low to demonstrate ultimate performance. To date, with a maximum burn-up of 1842 MWd/t, no fuel has failed.

Exclusive of the incident associated with the development of on-power fuelling mentioned earlier, the loss of heavy water has been favourable. Continued improvements have reduced the chronic leakage from 16 kg/d (35 lb/d)

to approximately 5.5 kg/d (12 lb/d) which we find economically acceptable. Further improvement is expected, the current target being 2.2 kg/d (5 lb/d) which amounts to 1% of the total inventory per annum.

It is beyond the scope of this paper to discuss the design and detailed operation of the power regulating equipment. However, to convey the general kinetic behaviour, a start-up is described following a shut-down of 2 to 7 d in duration. Further, let us assume that the moderator is down in the dump tank and the heat transport system is cold and at a temperature of 38°C (100°F).

First, the heavy-water level is automatically raised to a preset level of 183 cm (72 in) which is well below critical for any combination of circumstances. At this point, the ion-chamber currents are checked and typically are found to have increased to 10^{-6} full power. The neutron current, at this stage, is primarily due to the subcritical multiplication of the photoneutron source from the $D^2(\gamma, n)H^1$ reaction. This stage is completed at approximately 7 min after start-up.

The design does not provide for automatic start-up below 10^{-5} full power. Therefore, the second step in the start-up is to raise manually the heavy-water level until criticality is reached and power is 10^{-5} full power. This operation can be performed very smoothly because of the combined effects of good heavy-water level regulation, the photoneutron source and the effect of delayed neutrons. During this period of time, the power change is limited to an e folding rate of 100 s (trip 10 s). This corresponds to an excess reactivity of 1 mk ($k_{eff} = 1.001$). This operation is performed very carefully, although the rate of rise is inhibited by delayed neutrons for small reactivity excursions and by the relatively long neutron lifetime (10^{-3} s) in a heavy-water moderator and the negative power coefficient of 5 mk between zero and full power. Next the power is raised to 10^{-3} of full power, the auto computer is checked for response and the control is transferred to automatic low-power regulation. This stage is typically completed about 15 min after start-up.

The third step in starting the reactor is to place the reactor on "automatic high power" regulation. The power is initially raised to a preset value of 6% full power as this corresponds to that required to heat the equipment and piping in the heat transport system at the limiting rate of 110°C/h (200°F/h). This condition is typically reached 20 min after start-up. At this time the heat transport system will be at 52°C (125°F) due to previous heating by the pumps.

As the system is heated, the pressure in the boiler is gradually increased. To minimize the total start-up time, the turbine generator is prepared for operation in parallel with the raising of pressure in the boiler. The nitrogen blanket in the boiler is purged to atmosphere and the steam mains are heated through bleed valves. All operations are performed from the control room.

The pressure in the boiler is approximately 19.3 kg/cm² (275 lb/in² gauge) 1 h 20 min after start-up and steam is admitted to roll the turbine and raise vacuum. The reactor power automatically increases to sustain the maximum heating rate of 110°C/h (200°F/h). Half an hour later the overspeed tests are performed. Two hours after start-up the heat-transport system is at 235°C (455°F) and the boiler pressure is 28.8 kg/cm² (410 lb/in² gauge).

The turbine generator is then synchronized with an initial load of 2 MW(e). The power is then raised at 1 MW(e)/min thereby reaching a maximum permissible power of 12 MW(e) at time 2 h.12 min. Full power of 22 MW(e) (gross) cannot be reached until the xenon fission product reaches near-equilibrium at approximately 30 h after start-up. The lost production during start-up after a long shut-down is approximately 0.4 full power days.

The station can be restarted to full power in approximately 25 min after a trip has taken place and the fault corrected. Under emergency conditions, the heat-transport system can be cooled and depressurized in approximately 30 min.

Under normal conditions, the electrical output, system pressures, system temperatures, reactor flux and moderator heavy-water level are very stable. For example, the boiler pressure varies by less than $\pm 0.1 \text{ kg/cm}^2$ (1.0 lb/in^2 gauge) and the electrical output can easily be maintained to $\pm 0.05 \text{ MW}$.

The electrical output can be changed at the design rate of 20% full power/min without difficulty. The station can over-ride most disturbances such as change-over of pumps, changing temperature controller set points, line voltage changes, line frequency changes, on-power regulating channel tests, on-power emergency stop valve tests, on-power dump valve tests etc. However, further improvements are required to withstand the following three major disturbances:

- (a) Turbine overspeed trip – the turbine can handle a loss of line but trips out on full load rejection of generator.
- (b) Reactor overpressure trip – the reactor trips out on both generator rejection and loss of line due to a small transient overpressure in the surge tank.
- (c) Reactor undervoltage trips – on major transmission line faults where the 13.8 kV voltage drops to 10 kV, the reactor trips out on undervoltage due to fail-safe protective relays.

Such large disturbances occur infrequently. Nevertheless, we anticipate no major difficulty in correcting these problems and it is our intention to demonstrate the ability of the station to stay on line under such circumstances.

The following are a few technical highlights associated with operation since the In-Service date.

In the very early operation, the reactor tripped frequently because the reactor regulating system was not adequate to cope with minor disturbances. The major cause of this trouble was traced to a non-linear characteristic in the ion-chamber amplifiers. Today the regulating system is giving top performance and provides excellent response and stability. There were no interruptions of power during the second six weeks capacity run.

The moderator pumps and helium blowers employ multiple shaft seals. The outboard seals are in contact with oil and the inboard seal is of course in contact with heavy water. A labyrinth is provided between the inboard and outboard seals to separate the heavy water and oil-seal leakage. The interspace does not effectively separate oil and heavy-water leakage as intended and, as a result, oil was being added to the moderator system. Because of this oil converted to acid under irradiation, the ion-exchange columns in the moderator system were being consumed at the rate of one per day rather than at the design rate of one per 4 months. This problem has been temporarily solved by removing the oil before pumping the collected

heavy water back into the system. An improvement is planned to eliminate the oil mixing in the first place.

In the early operation, the mechanical shaft seals of the primary pumps typically lasted 200 h, which was entirely unacceptable. Improved degassing and venting techniques have already increased the life to more than 1500 h and we expect to demonstrate seal life in excess of 7000 h within the next two years.

The original poor workmanship on mechanical tubing joints still continues to worry us in NPD even though we know how to avoid this problem in a new installation. Further changes are planned to correct this situation, in view of the possibility of potential acute loss, not because of chronic vapour loss.

Although the proving of on-power fuelling has been delayed, the station performs acceptably with off-power fuelling and we are confident that on-power fuelling will be satisfactorily demonstrated this year for this type of station.

A chemical deposit was found on the spindle of the turbine emergency stop valves. This of course is not acceptable as it can result in seizure of the valves. The problem was not caused by the chemical quality of the steam but was due to the accumulation of trace quantities being deposited from the steam leaking off along the valve spindle. As the nearly dry steam reduced in pressure, it became superheated and deposited the trace chemicals on the spindle. It was corrected by utilizing a backseat which eliminated the steam leak-off.

Initially, a freezer-dryer was provided to prevent loss of heavy water from the vault ventilation system to the stack. A chemical sorber (electrically reactivated) which is less expensive than the freezer-dryer, is now being used and gives trouble-free operation.

Some problems still exist, further improvements are anticipated, and we await further performance information and cost data. However, chronic heavy-water losses have been gratifyingly low, costs of operation and maintenance have been close to the estimate and because the early performance has been most encouraging our programme is proceeding with even greater confidence.

DISCUSSION

S. YIFTAH: There has been some mention in the literature of an "up-rated NPD", i. e. a plant which, based on the NPD design, could produce about 80 MW(e) with only minor changes. Could you give us any further information?

L.G. McCONNELL: The Canadian General Electric Company has performed a design study with the object of uprating NPD to 80 MW(e). I cannot give you precise data, but I am sure the company could provide information. Generally speaking, the fuel rating and the number of fuel channels have been increased, the flux is flattened, and more 19-rod fuel elements are to be installed.

S. YIFTAH: You mentioned that NPD has been in service since 25 September 1962, i. e. about eight months. Since it was intended to be a prototype for the 200-MW(e) CANDU, could you give us some details about the

latter reactor? I should particularly like to know what stage of construction has been reached, and whether any changes have seemed advisable in the light of experience with NPD.

L. G. McCONNELL: The process equipment is now being installed in the CANDU plant. Commissioning will get under way next year and the station should reach full power in mid-1965. Manufacturing, design, construction and commissioning experience gained at NPD has been turned to good account at CANDU, although the construction of the larger plant is already so far advanced that some of our experience—especially in operation—may come too late to influence its design.

The latest estimate of nuclear fuel cost for the CANDU reactor—considered to be quite conservative—is in the range of 1.01 mill/kWh, based on a burn-up of approximately 9000 MWd/t of uranium.

S. YIFTAH: The economic viability of heavy-water reactors is presumably very dependent on the price of heavy water. Some mention has been made of Canadian D₂O production at \$22/lb, which is about \$2 less than the United States price. Could you tell us what the situation is now?

L. G. McCONNELL: Atomic Energy of Canada Ltd. has invited tenders for the construction of a heavy-water plant in Canada, and it has been stipulated that the price must not exceed \$22/lb; on the other hand, the company has agreed to give certain minimum purchase guarantees. The price of heavy water is important, certainly, but I doubt whether it would "make or break" a D₂O reactor.

H. J. BRÜCHNER: Am I right in thinking that no heavy-water production plant is yet operating in Canada? If so, how can you specify the price?

L. G. McCONNELL: Yes, you are quite right. Tenders have been solicited, as I said before, and it is up to the concerns submitting tenders to calculate the capital investment and quote a price at which they would be prepared to deliver D₂O for a guaranteed period of time,

M. R. SRINIVASAN: Could you tell us something more about the on-load fuelling tests? If I understood you properly, the difficulties that cropped up in December 1962 during on-load testing of the refuelling machines were fairly serious. Do you expect to solve them satisfactorily in the near future?

L. G. McCONNELL: We still look forward to on-load fuelling with considerable optimism, and the difficulty that arose in December 1962, though serious, was the only one. In fact, we hope that on-load fuelling will be demonstrated later this year. Even so, off-load fuelling in the NPD reactor has worked very well.

K. EFFAT: Could the figures on heavy-water losses in NPD be extrapolated to the CANDU reactor, and have you estimated probable heavy-water losses in the CANDU plant?

L. G. McCONNELL: I can do no more than venture an opinion. NPD is a 20-MW(e) plant, and the information we now have on heavy-water losses is accurate to within a factor of 3. Certain methods of reducing these losses have been discovered but not yet applied to NPD. Needless to say, all the knowledge we have gleaned from NPD will be applied, insofar as that is possible, to the 200-MW(e) CANDU plant. You can imagine, however, that there will be considerable scope for differences of opinion when it comes to extrapolating NPD results to CANDU. Estimates of heavy-water loss

from CANDU now vary from 10 to 48 lb per day, including chronic as well as average acute loss, and the average equivalent cost of upgrading resulting from periodic spills. A loss of 10 lb per day corresponds to a unit energy cost of approximately 0.008 cents (Canadian) per kWh.

THE ENRICO FERMI ATOMIC POWER PLANT

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Abstract — Résumé — Аннотация — Resumen

THE ENRICO FERMI ATOMIC POWER PLANT. Construction of the Enrico Fermi Atomic Power Plant, which utilizes a 100 MW(e) fast breeder reactor, was essentially completed in December 1961. During the past eighteen-month period, systems and components have been extensively tested. This pre-operational test programme has proved most valuable in verifying the design and in indicating needed modifications.

All the problems encountered have proved manageable. The more important modifications are summarized. Graphite shielding. In December 1960, the primary system was filled with sodium, and extensive testing began. When the primary shield tank was reopened after a 1000°F primary system test, it was found that much of the graphite block shielding which had been installed around the reactor had deteriorated. These high-temperature blocks which were impregnated with boron had increased in volume and lost strength. Extensive analysis indicated that the graphite binder had failed.

It was decided to replace all of the graphite, to use boron carbide as the boronating agent, to install the block with mechanical fasteners, and to keep moisture to a minimum.

Modifications within the reactor vessel. Repairs and design modifications were made to correct the cause of sub-assembly sticking, the damage which resulted, and to prevent further malfunctions of the offset handling mechanism.

In order to make repairs and alterations, the offset handling mechanism was removed, the reactor vessel was drained of sodium, and trained personnel wearing specially designed protective suits entered the reactor vessel. Entrance to the work area was through a special air lock since an argon atmosphere was maintained inside the vessel.

Steam generator modifications. During hydrostatic testing of the No. 2 steam generator, several leaking tubes were discovered. Tube failure was traced to stress corrosion cracking. The No. 2 bundle was retubed, all units were stress relieved, and a hydrogen detector system was installed.

In December 1962, a sodium-water reaction took place in the No. 1 steam generator. The rupture disc, which was installed for such an eventuality, operated correctly. The cause and effect of this failure is currently being investigated.

Completion of pre-operational testing. Anticipating the issuance of a low power licence, the final assembly and test programme is being completed in March.

LA CENTRALE NUCLÉAIRE ENRICO FERMI. La construction de la centrale nucléaire Enrico Fermi, équipée d'un réacteur surgénérateur à neutrons rapides de 100 MWe, a été virtuellement terminée en décembre 1961. Au cours des derniers 18 mois, on a minutieusement vérifié les circuits et les parties constitutives. Ce programme de vérifications avant la mise en service a été très utile pour vérifier les plans et déterminer les modifications à y apporter.

Tous les problèmes ont pu être résolus. Voici la liste des principales modifications:

Bouclier de graphite. En décembre 1960, on a rempli le circuit primaire de sodium liquide et on a procédé à des essais très poussés. Lorsqu'on a rouvert l'enceinte de protection primaire, après les essais à 540°C, on a constaté que la plus grande partie des blocs de graphite entourant le réacteur s'était détériorée. Ces blocs, destinés à supporter de hautes températures, étaient imprégnés de bore; ils s'étaient dilatés et avaient perdu leur résistance. Une analyse approfondie a montré que le liant du graphite avait cédé.

On a décidé de remplacer tout le graphite, d'utiliser du carbure de bore à la place du bore, d'assujettir les blocs par des dispositifs mécaniques et de maintenir l'humidité à la valeur minimum.

Modifications à l'intérieur du caisson du réacteur. On a procédé à des réparations et modifications pour supprimer la cause du collage des barreaux et les dommages qui en résultaient, et pour prévenir d'autres défaillances du mécanisme de manutention des cartouches.

Pour ce faire, on a dû retirer le mécanisme de manutention et vider le caisson contenant le sodium. Des spécialistes, portant des vêtements de protection spéciaux, ont pénétré dans le caisson. On entrait dans la zone de travail par un sas spécial, car on maintenait une atmosphère d'argon à l'intérieur du caisson.

Modifications aux générateurs de vapeur. Au cours des essais hydrostatiques du générateur de vapeur N° 2, on a constaté que plusieurs tubes fuyaient. La défaillance des tubes résultait d'un fendillage provoqué par la corrosion sous contrainte. On a donc remplacé les tubes du générateur N° 2, on a fait disparaître les contraintes auxquelles étaient soumises les pièces et on a installé un dispositif détecteur d'hydrogène.

En décembre 1962, il s'est produit une réaction sodium-eau dans le générateur de vapeur N° 1. Le disque de sécurité installé en prévision d'une telle éventualité a bien fonctionné. On cherche actuellement la cause et les effets de cette défaillance.

Fin des essais avant la mise en service. Dans l'attente d'un permis d'exploitation à basse puissance, on termine le montage et le programme d'essais (mars 1963).

АТОМНАЯ ЭЛЕКТРОСТАНЦИЯ им. ЭНРИКО ФЕРМИ. Строительство атомной электростанции им. Энрико Ферми с реактором-размножителем на быстрых нейтронах мощностью 100 мвт (эл.) было в основном закончено в декабре 1961 года. В течение последних 18 месяцев проводились широкие испытания систем и компонентов. Эта предпусковая программа испытания оказалась очень полезной для проверки конструкции и для определения необходимых изменений.

Все возникшие проблемы оказались разрешимыми. В докладе кратко освещаются наиболее важные изменения.

Графитовая защита. В декабре 1960 года первый контур был заполнен натрием и были начаты широкие испытания. Когда был снова открыт защитный бак первого контура после испытания первого контура при температуре 1000°F, было обнаружено, что большая часть графитового блока защиты, установленного вокруг реактора, повреждена. Високотемпературные блоки, насыщенные бором, увеличились в объеме и потеряли прочность. Проведенный тщательный анализ показал, что графитовая связь повреждена.

Было решено заменить весь графит, использовать карбид бора в качестве соединения, содержащего бор, установить блок с помощью механического крепления и довести влажность до минимума. Изменения в корпусе реактора. Был проведен ремонт и внесены изменения в конструкцию для устранения причины заедания сборок, для ликвидации повреждения, которое явилось результатом этого, и для устранения в дальнейшем неисправности в перемещающем механизме управления.

Перед ремонтом был удален перемещающий механизм управления, корпус реактора был осушен от натрия. После этого подготовленный персонал в специальных защитных костюмах был допущен внутрь корпуса реактора. Вход в рабочую зону обеспечивался с помощью специального воздушного шлюза, поскольку в корпусе реактора имелся аргон.

Изменения парогенератора. Во время гидростатических испытаний парогенератора № 2 были обнаружены течи в нескольких трубках. Причиной неисправности трубок было их растрескивание в результате коррозии от напряжения металла.

Узел № 2 был снабжен новыми трубками, все секции были освобождены от напряжения, и была установлена водородная детекторная система.

В декабре 1962 года в парогенераторе № 1 произошла реакция натрий - вода. Диск разрыва, который был установлен на такой случай, работал правильно. В настоящее время исследуются причины и влияние этого повреждения.

Завершение предпусковых испытаний. Планируется закончить в марте программу окончательной сборки и испытаний в ожидании выдачи лицензии на эксплуатацию реактора на малой мощности.

LA CENTRAL NUCLEOELÉCTRICA ENRICO FERMI. La construcción de la central nucleoelectrónica Enrico Fermi, que está equipada con un reactor productor de neutrones rápidos de 100 MW(e), quedó prácticamente terminada en diciembre de 1961. En el curso de los últimos 18 meses se han verificado minuciosamente los circuitos y partes constitutivas. Este programa de ensayos preliminares ha resultado en extremo útil para comprobar la idoneidad del diseño y obtener indicaciones sobre las modificaciones necesarias.

Todos los problemas con que se tropezó pudieron solucionarse. A continuación figura la lista de las principales modificaciones introducidas en la construcción de la central.

Blindaje de grafito. En diciembre de 1960 se llenó de sodio el sistema primario, y se dio comienzo a una amplia serie de ensayos. Cuando se volvió a abrir el recinto primario de protección después de someter este sistema a una temperatura de 540°C, se comprobó que gran parte de los bloques de grafito que rodean al

reactor se había deteriorado. Esos bloques para altas temperaturas estaban impregnados con boro; se habían dilatado y perdido resistencia. Un análisis detenido puso de manifiesto que el agente aglomerante del grafito había cedido.

Se decidió sustituir todo el grafito, utilizar carburo de boro en vez de boro, sujetar los bloques con dispositivos mecánicos y reducir al mínimo la humedad.

Modificaciones en el interior del recipiente del reactor. Se realizaron reparaciones y se introdujeron modificaciones en el diseño a fin de evitar la adherencia de los subconjuntos y el daño resultante, así como para prevenir otras fallas del mecanismo de manejo de los elementos combustibles.

Para ello, se desmontó el mencionado mecanismo y se extrajo el sodio del recipiente del reactor. Personal especializado que vestía trajes protectores especiales penetró en el interior del recipiente a través de una esclusa neumática especial, ya que en el interior del recipiente se mantenía una atmósfera de argón. Modificaciones en los generadores de vapor. Cuando se realizaron los ensayos hidrostáticos del generador de vapor N°2, se descubrieron fugas en varias tuberías. Las investigaciones subsiguientes pusieron de manifiesto que esa falla de las tuberías obedecía a un agrietamiento debido a la corrosión bajo tensión. Se cambiaron las tuberías del generador N°2, se eliminaron las tensiones de todas las unidades y se instaló un dispositivo detector de hidrógeno.

En diciembre de 1962 el sodio reaccionó con el agua en el generador de vapor N°1. El disco de seguridad instalado en previsión de tal contingencia funcionó correctamente. En la actualidad se están investigando las causas y efectos de esa falla.

Terminación de los ensayos preliminares. A la espera de un permiso de explotación a baja potencia, se está terminando el montaje y el programa de ensayos (marzo de 1963).

Pre-operational testing of the Enrico Fermi Atomic Power Plant has been successfully completed. Construction of the plant which utilizes a 200 MW(t) fast-breeder reactor was essentially completed in December 1961. During the past eighteen-month period, systems and components have been extensively tested.

A far more extensive pre-operational test programme was initially planned for this project than has been customary with more conventional reactors. This decision proved to be a wise one. Major portions of the plant have in effect been operated (without nuclear fuel) for almost two years. This operation has served the dual purpose of verifying the adequacy of the basic design and its major components and of exposing difficulties requiring design modification or other correction. The information derived from this extensive programme should be invaluable to later generations of fast-breeder reactors. Accordingly, a major portion of this review is devoted to this topic. The attention given to analysis of the difficulties exposed and of the way in which they have been corrected, however, should not obscure the basic fact that the fundamental design has been found satisfactory and that all troubles have proved capable of solution. An operating licence has been issued by the Atomic Energy Commission (AEC). Final inspection by the AEC has now been completed, and nuclear testing is about to start.

During the test period, all of the plant systems have been in operation, some for a few weeks and others, such as the primary sodium system, for longer periods of time — approaching two years. Delays were encountered as the pre-operation tests were being run, which involved the installation of systems and, in some cases, the installation of equipment. Considerable interest has been exhibited in the test results, the modifications made, and the method used in making the modifications.

PROJECT ORGANIZATION

PRDC undertook construction of the nuclear portion of the plant which it will own and operate; whereas The Detroit Edison Company designed, constructed, and will operate the conventional facilities. The reactor plant was designed by Atomic Power Development Associates, Inc. (APDA) with Commonwealth Associates, Inc. serving as architect-engineer. United Engineers & Constructors Inc. had responsibility of construction engineer for the entire plant.

A large portion of the research and development for the reactor plant amounting to more than \$25 million was financed and directed by APDA and PRDC. The AEC under their Civilian Power Demonstration Reactor Program provided important research and development assistance costing approximately \$3 million.

PLANT DESCRIPTION

The Fermi Plant, shown in Figs. 1 and 2, is built on a 915-acre site located 30 miles southwest of Detroit on the westerly shore of Lake Erie.

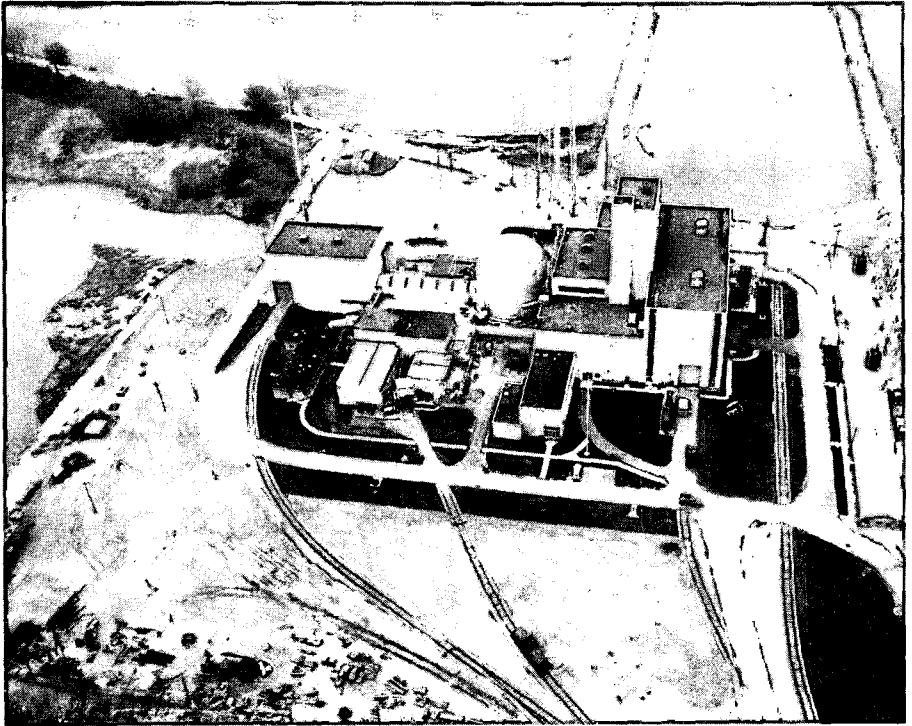


Fig. 1

Aerial photograph of Fermi Plant - west side

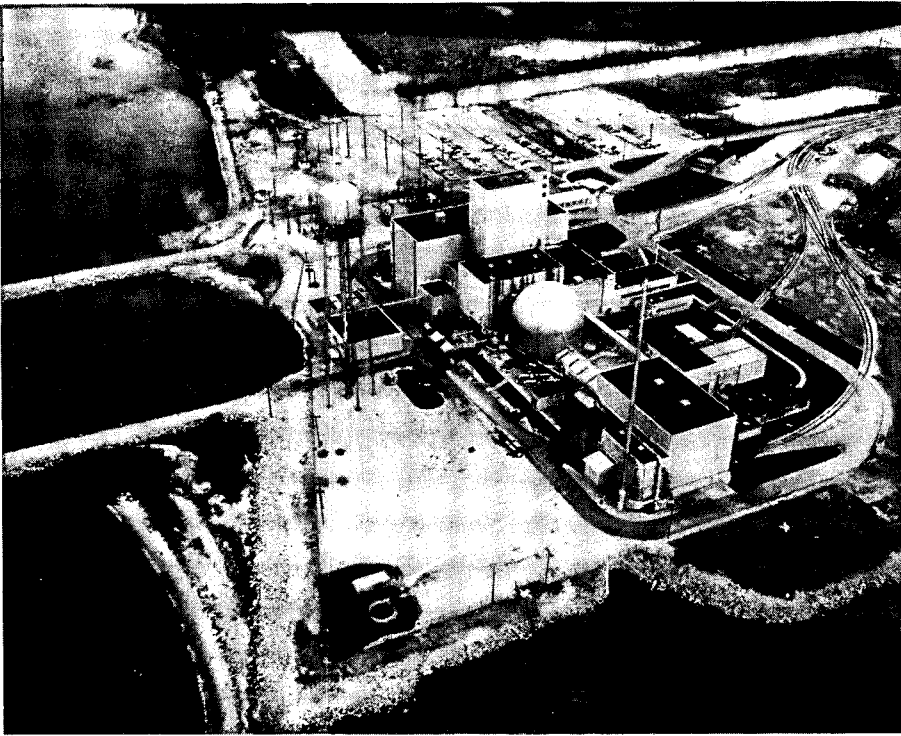


Fig. 2

Aerial photograph of Fermi Plant - east side

The construction area measures 15 acres, and the exclusion area on the land side has a radius of 3200 ft.

Schematic diagram

Liquid sodium is used as both primary and secondary coolant. Design power for the first type of fuel loading to be used is 200 MW(t) (65.9 MW(e)), although the equipment installed is capable of removing the full design power of the reactor which is 430 MW(t) (156 MW(e)).

The flow diagram, Fig. 3, shows schematically the arrangement of the reactor and primary sodium systems within the reactor building, the secondary sodium system which transports heat from heat exchangers to the steam generators, and the turbine-generator facility. Table I lists liquid metal and steam plant conditions for initial operation with the uranium 10 wt. % molybdenum alloy Core A.

Cross-section of reactor building

The relative position of major reactor components, piping, and shielding is shown in the cross-section of the reactor containment building, Fig. 4.

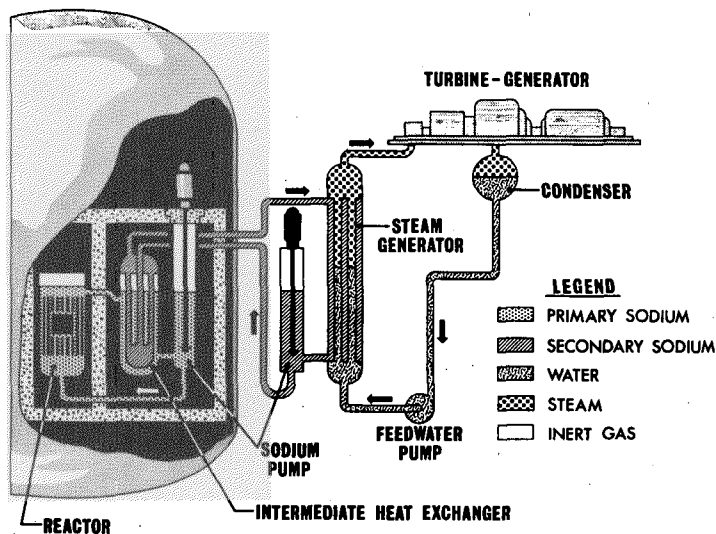


Fig. 3

Flow diagram - Fermi Plant

TABLE I

LIQUID METAL AND STEAM PLANT CONDITIONS

Net thermal efficiency	30.5
Primary sodium temperatures	
Leaving reactor	800°F
Entering reactor	550°F
Sodium flow	8 570 000 lb/h
Secondary sodium temperatures	
Entering steam generator	767°F
Leaving steam generator	517°F
Steam pressure	600 lb/in ² abs.
Steam temperature	764°F

Stainless steel is used throughout the primary system, and complete carbon steel secondary containment is provided around this system. Neutron and gamma shielding is designed so that access for operating and maintenance personnel to the operating floor is normal, that is a full eight-hour shift. Building dimensions are 120 ft in height, 51 ft of which are below grade, and 72 ft in diam.

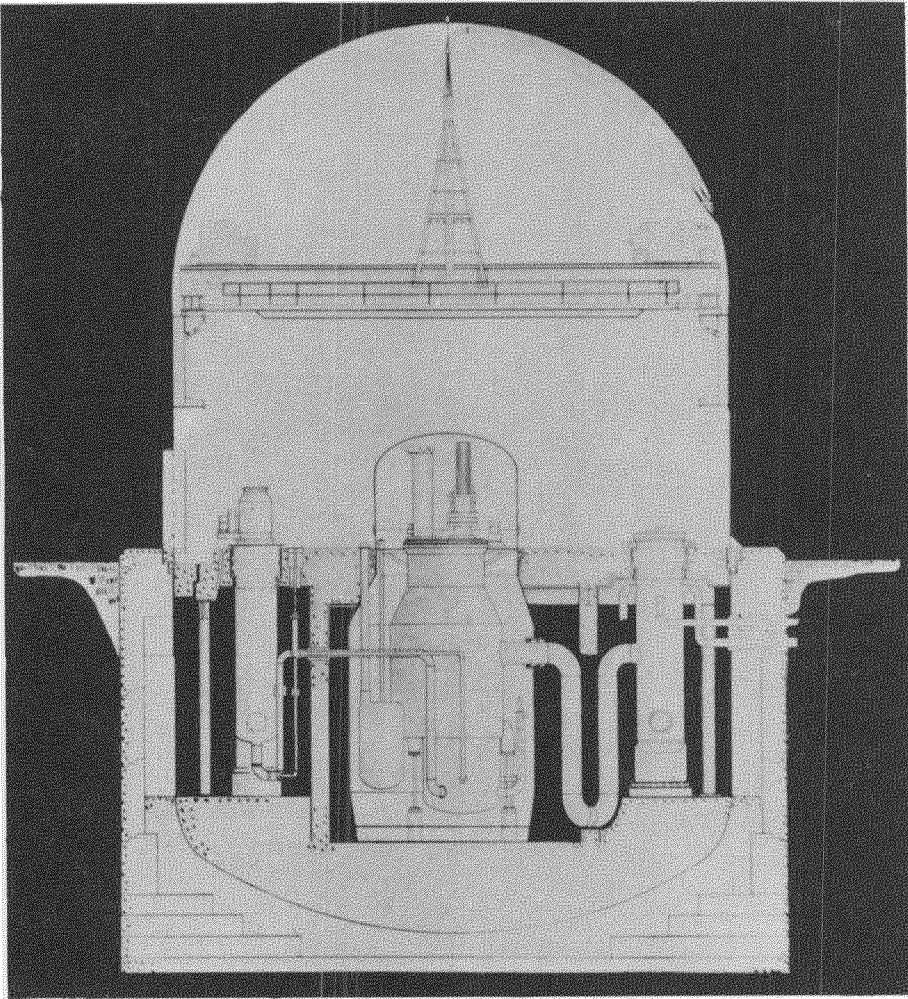


Fig. 4

Elevation of the reactor plant

Perspective view of the reactor

Fig. 5 serves to identify the fuel handling mechanisms and locates the core, blanket, and shielding within the primary shield tank. Core and blanket subassemblies are brought to the reactor in the cask car (upper left). They are loaded and unloaded by the offset handling mechanism mounted on the rotating shield plug. The hold-down equipment holds core subassemblies in place against the upward forces caused by the flow of sodium through the reactor. Both plain and borated graphite are used as shielded in the rotating plug and in the space between the reactor vessel and the primary shield tank.

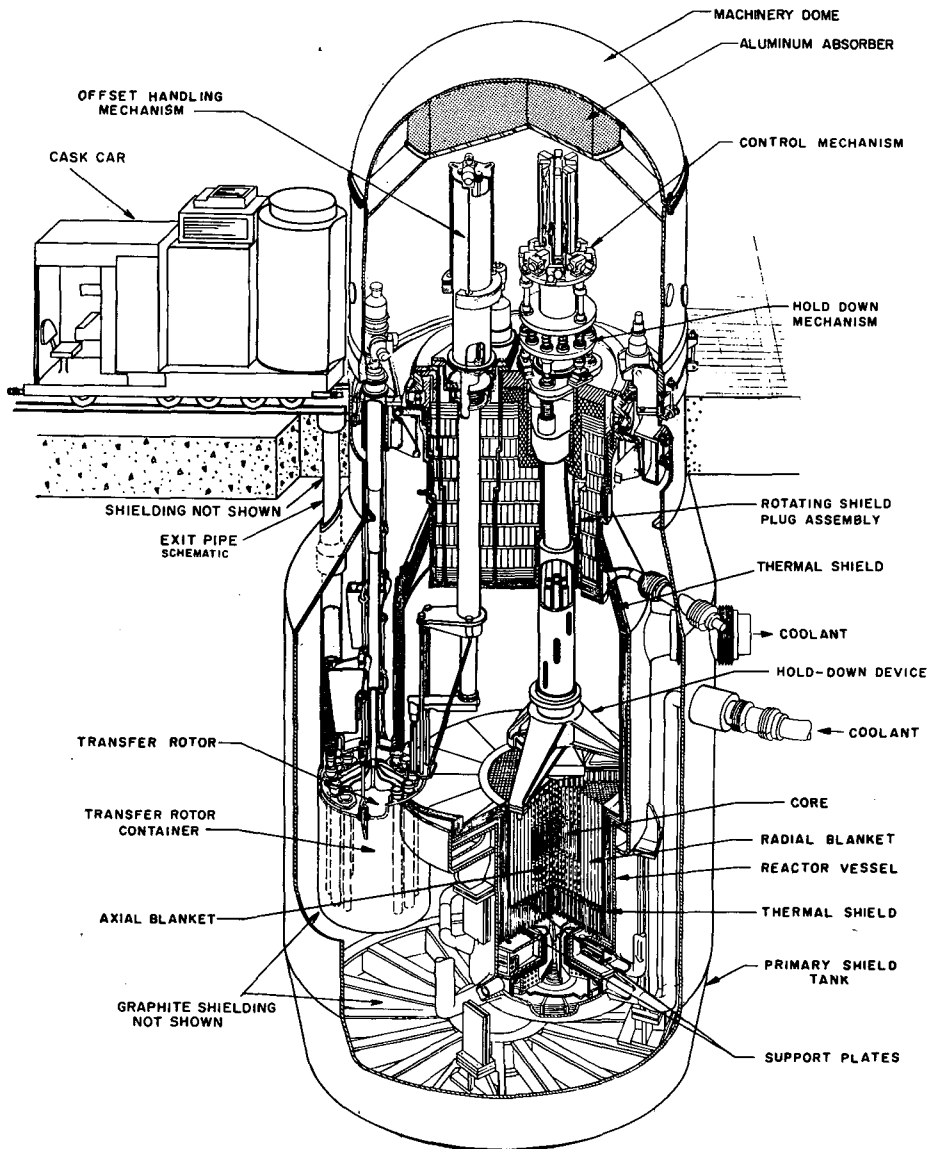
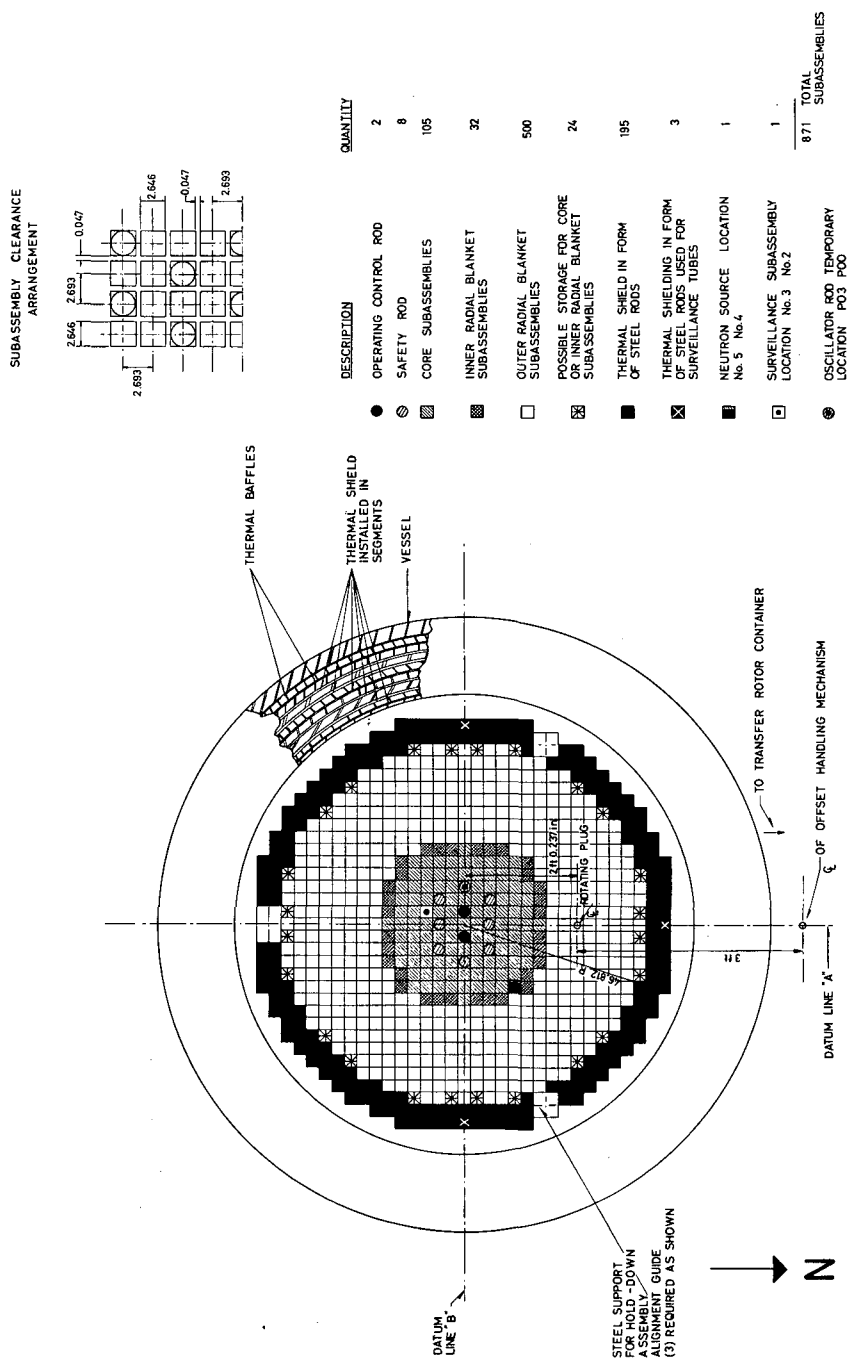


Fig. 5

Perspective view of reactor

FUEL AND BLANKET DESIGN

The arrangement of core and blanket subassemblies within the reactor vessel is shown in Fig. 6, a cross-section of the reactor. This diagram shows the relative position of 870 removable units and the location of the



eight safety and two operating control rods. It is to be noted that the core region, in the central area of the reactor, is supplied by a high-pressure plenum whereas the blanket region in the adjacent area is supplied from a lower pressure plenum.

Core subassembly

There are 105 core subassemblies, one of which is shown in Fig. 7. Each assembly is composed of three sections – the core or fuel section in the centre – and axial blanket sections at each end. All three sections are mechanically incorporated into a single unit.

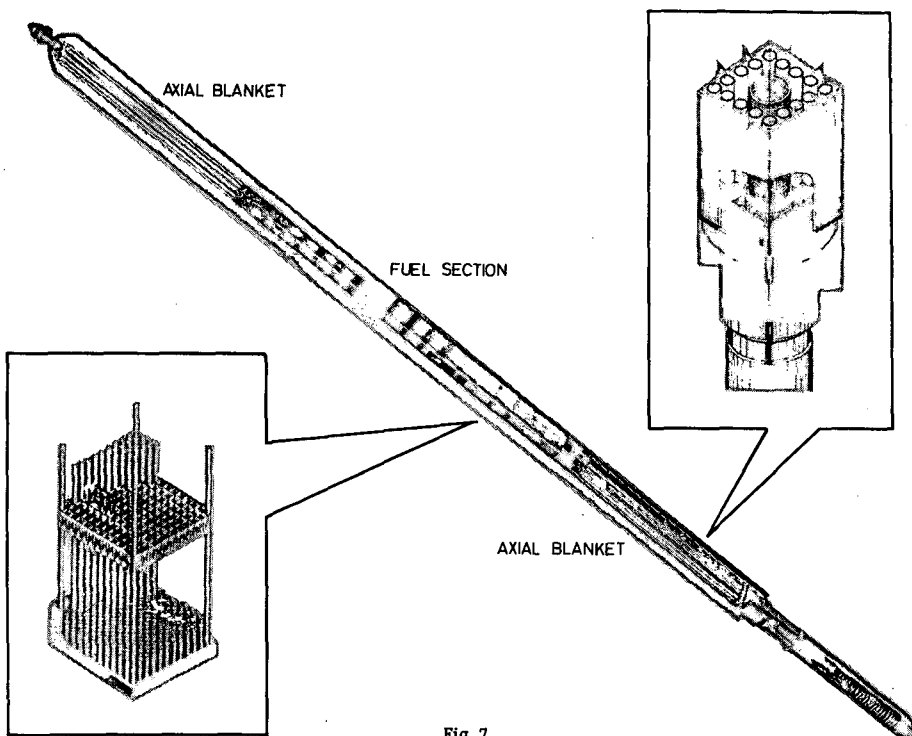


Fig. 7

Core subassembly

Each of the two axial blanket sections (upper right enlargement) contains 16 uranium 3 wt.% molybdenum alloy rods, 0.396-in diam., containing uranium depleted to 0.35% in U^{235} . These rods are enclosed and sodium-bonded to 0.010-in thick stainless-steel tubes having an outside diam. of 0.443 in.

The fuel section is made up of 140 uranium 10 wt.% molybdenum alloy pins containing enriched uranium to 25.6% in U^{235} . Each pin is clad with 0.005 in of zirconium that is metallurgically bonded to the fuel alloy enclosed at the top and bottom with zirconium end caps. The pins are $32\frac{1}{16}$ -in over-all length and measure 0.158 in in diam. They are maintained on a

square pitch of 0.200 in in a support structure, referred to as a "birdcage," made of stainless-steel plates and "egg-crate" grids (lower left).

1000°F ISOTHERMAL TEST AND ITS RESULTS

In December 1960, the primary system was filled with sodium and extensive testing of that system and of principal reactor components began. This test programme, conducted by APDA, was lengthy and thorough, and in addition to exposing certain difficulties which have now been corrected, the results confirmed many of the basic design and engineering concepts followed in the construction of the reactor plant and its principal components.

An important part of the APDA test consisted of maintaining the reactor vessel and one loop of the primary sodium system at an isothermal temperature of 1000°F for about four days, the necessary heat being supplied by induction heating in conjunction with a temporary oil-fuelled sodium heater. This isothermal test temperature imposed conditions on the primary system which were in many respects considerably more severe than will be encountered in operation.

Graphite shield description

The Enrico Fermi shield system was designed to provide biological protection for personnel, to maintain a non-radioactive secondary sodium system, and to prevent damage to concrete and other materials outside the shield. It is the primary or graphite shield shown in Fig. 8 that has required the major shielding effort during the last seven years. It was also the complete replacement of this graphite shield that constituted the major construction effort during 1962.

The graphite shield is contained in the gas-tight primary shield tank surrounding the reactor vessel and in the rotating shield plug in the top of the reactor vessel. It is divided into two general zones: an inner row next to the reactor vessel backed up by thermal insulation and lagging to cause heat to flow back to the sodium coolant and an outer or field layer between the thermal insulation and the primary shield tank wall.

Graphite shield blocks were installed in the "pan", the dished head at the bottom of the lower reactor vessel; in the "tub", the bottom of the primary shield tank directly under the lower reactor vessel; in the rotating shield plug at the top of the reactor vessel; and adjacent to the vertical walls of the reactor vessel inside the primary shield tank. This material consisted of non-borated and 5% borated graphite. Boron carbide was used for boring.

A lower grade graphite block was used as the bulk shielding or "field" graphite in the primary shield tank. It consisted of plain and 5 and 7% borated graphite. Boronation was accomplished with sodium tetraborate.

Clearance difficulties

During testing at sodium temperatures of 500°F, it was observed that clearances around a number of pipes and parts within the primary shield tank

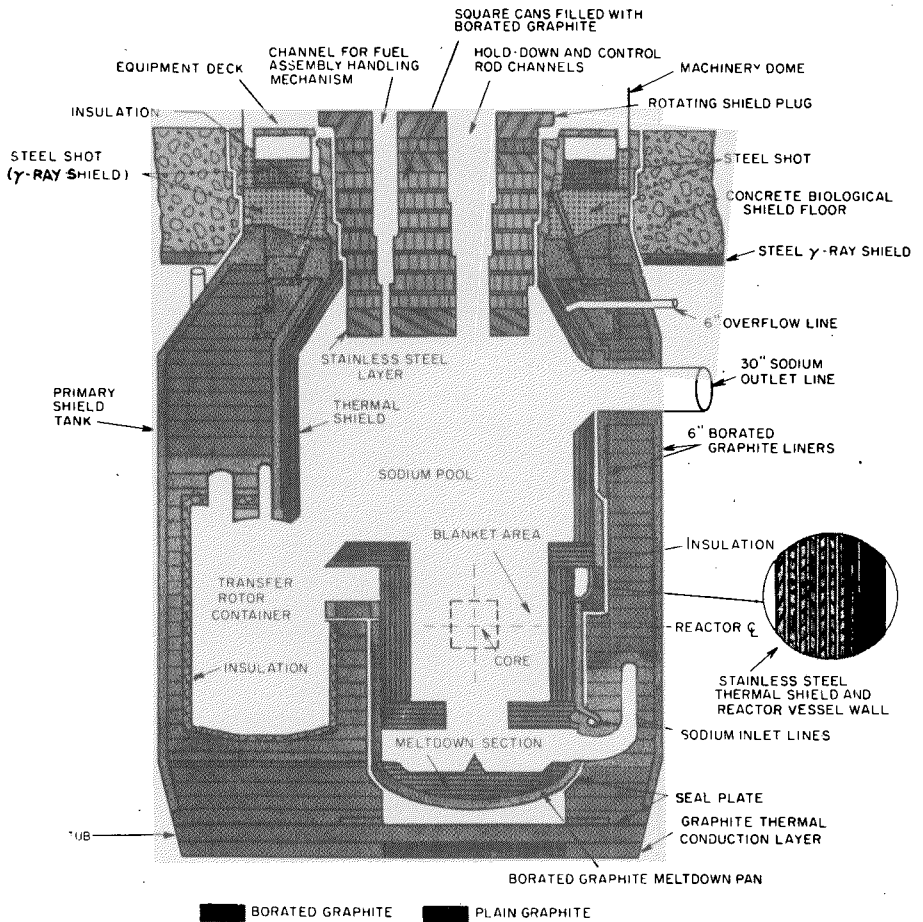


Fig. 8

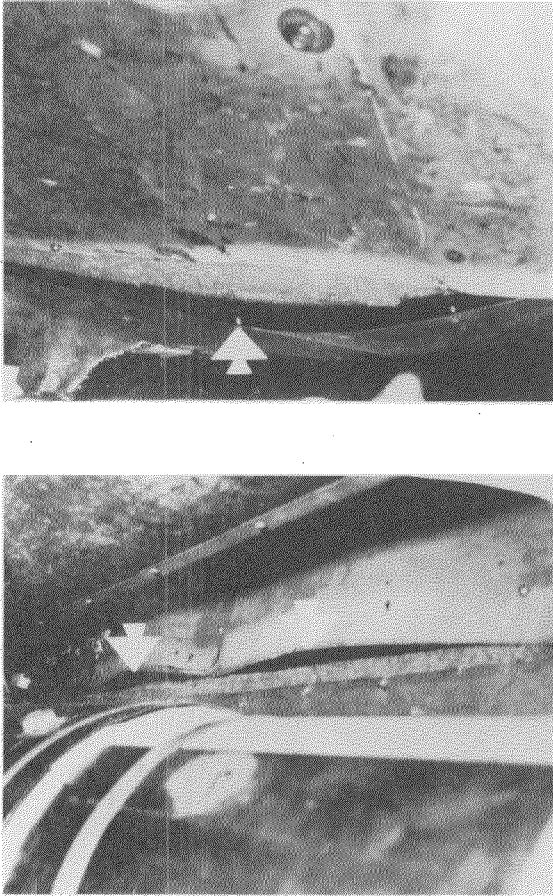
Primary shield tank

were not sufficient for expansion conditions at elevated temperatures. By checking the movement of the vertical exit port it was determined that portions of the field graphite shown in Fig. 8 had to be repositioned. About 65% of the field graphite was removed for inspection and repositioning. As the graphite was replaced, a carbon powder-furfural-alcohol resin was used to cement the blocks into place.

When the primary shield tank was about 50% filled, the 1000°F isothermal test was run. The system was held at 1000°F for four days. Severe outgassing of the graphite in the rotating shield plug occurred during the temperature elevation to 1000°F. This gas release contaminated the argon cover gas, and then a heavy layer of impurities developed on the surface of the liquid sodium. Chemical analysis showed the solids to be primarily sodium oxide, sodium carbonate, and carbon. These impurities have been eliminated by cold trapping and filtering.

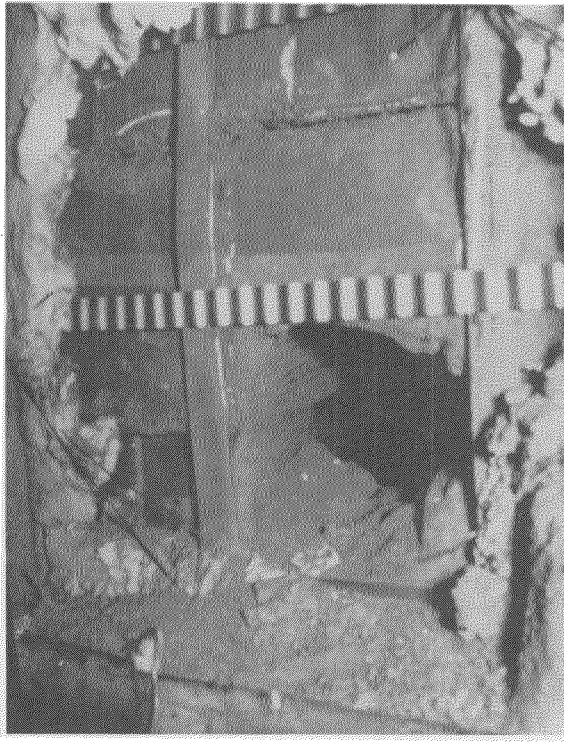
Inspection following 1000°F test

The temperature was reduced and the primary shield tank opened to permit continuation of field graphite installation. When the tank was opened, it was found that the insulation and lagging holding the thermal shield (inner row graphite) on to the reactor vessel had bulged out from the vessel at several points. This condition is shown in Fig. 9. An inspection of the gra-

**Fig. 9****Failure of inner row graphite shield**

phite adjacent to the reactor vessel revealed signs of spalling on the face of the blocks. While the blocks did not appear to be crushed or deformed, they were soft and susceptible to cutting or scraping as is shown in Fig. 10. No damage to the heaters or the leak detector sheath was observed.

The inner row graphite blocks were made from electrode graphite turnings and an organic binder in the ratio of three to one. This mixture was fed to heated rollers and the resultant pasty mass was scraped from the

**Fig. 10**

Soft inner row graphite block

rollers. Blocks were molded at 265°F. The blocks were kiln dried over a period of 10 h to 1000–1100°F, held at that temperature for 4 h, and then cooled in a CO-CO₂ atmosphere to ambient temperature over a 14-h period. The completed blocks consisted of graphite, carbonized, and uncarbonized binder, and 0.5% moisture.

In addition to the moisture in the inner row graphite, there was an estimated 50 lb of moisture in the pan due to cement, 300 lb of water in the tub due to the aqueous sodium silicate used to improve thermal conductivity, 700 lb of water in the field graphite that was in place, and 30 to 50 lb of water in the cement used to secure the field graphite.

The inner row graphite, that had been subjected to severe temperature conditions, failed because it was made of low-temperature block that had not been completely graphitized and because it was used under conditions conducive to rapid oxidation. Oxidation took place preferentially in the ungraphitized binder causing a great reduction in strength from a small amount of oxidation. The moisture released from the graphite, the cements, and thermal insulation in the primary shield tank enhanced the graphite oxidation and attacked the boron carbide borating agent.

Graphite replacement

After extensive study and consultation it was decided to replace both the inner row graphite and field graphite with high-temperature, high-density graphite. Boron carbide was chosen as the borating agent. Replacing the graphite in the tub and pan would have been almost impossible. Therefore, these areas were sealed off from the rest of the primary shield tank and vented outside to the waste-gas disposal building. These areas are maintained at a slightly lower pressure than the rest of the primary shield tank so that if there is a transfer of atmospheres it will be from the primary shield tank to the pan and tub.

In procuring the replacement graphite, all precautions were taken by both the vendor and PRDC to assure that the material received would be of the highest quality. The graphite was made from petroleum coke and coal tar pitch. Boron carbide was added as the borating agent. The mix was extruded into logs and baked at 1500°F for a six-week period. The material was then pitch impregnated to give it a high density and then graphitized at about 5500°F for a four-week period. For the inner row boronated material, a special high-purity boron carbide was specified.

An extensive inspection programme was set up which required one per cent sampling of all the blocks produced. Tests on these blocks were run by both the vendor and an outside laboratory. The tests determined: density, thermal conductivity, compressive strength (before and after oxidation) dry air oxidation, wet nitrogen oxidation, ash, sulphur, chloride, and boron content, thermal expansion, gas evolution, and moisture content.

Actual installation of the new graphite began on 6 May 1962, and was completed on 15 December 1962. Fig. 11 is a photograph of the inner row graphite installation. These blocks were precut wedge shaped. Figs. 12 and 13 show the field graphite installation. Since no cement was used in the final installation, mechanical fasteners, including rings and pins, were used to hold the blocks in position. These rings may be seen in Fig. 13.

Primary shield tank purging

An estimated 200 lb of water were present in the primary shield tank after completion of the graphite shield. Calculations indicated that approximately 150 lb of this water could be removed at temperatures up to 300°F. The graphite was heated to an average temperature of 300°F by means of heaters attached externally to the primary shield tank and by heat transfer from the molten sodium inside the reactor vessel. The graphite shield was purged with dry air and then cycled between 16.7 and 12.7 lb/in² abs. These methods were continued until the 150 lb of available moisture were removed as determined by monitoring the discharge purge air.

Now that the primary shield tank is in operation at high temperature, it has been charged with dry nitrogen. Careful specifications have been set on the maximum inner-row graphite temperature and maximum dew point. As the dew point reaches the maximum, the tank is purged with dry nitrogen.

The laboratory and other studies which were undertaken in connection with this problem and which were used to verify the adequacy of the replace-

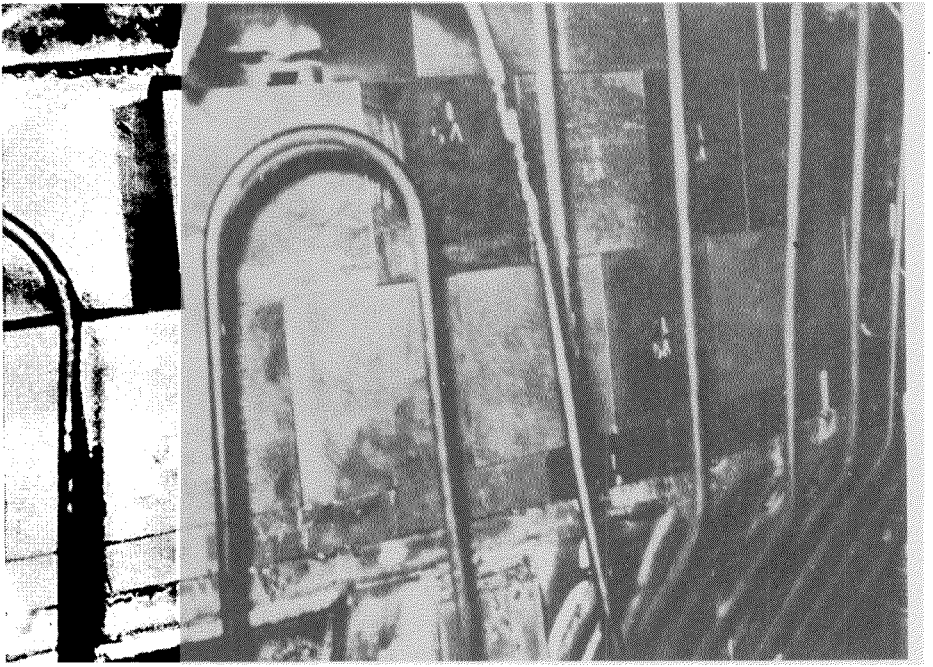


Fig. 11

Inner row graphite installation

ment graphite finally chosen have added considerably to existing knowledge of graphite technology and should be of considerable use to other reactor designers in the future. These studies indicated the importance of minimizing the exposure of such graphite at elevated temperature to both moisture and oxygen.

MODIFICATIONS WITHIN THE REACTOR VESSEL

In the summer of 1961, PRDC went forward with the pre-operational test programme including operation of the fuel-handling equipment and the core-positioning device or hold-down mechanism. Fig. 14 shows the location of these large components in the reactor vessel.

In the course of the test programme, several of the dummy subassemblies, which had been manufactured to less rigid specifications than the actual fuel and blanket subassemblies, were found to be sticking in the lower support plate. While removing these dummy subassemblies in order to determine the cause of this condition, the offset handling mechanism was bent when an inadvertent attempt was made to move it laterally before it was fully detached from a partially raised dummy subassembly. An interlock, which would have prevented such movement during nuclear operation, had been disconnected for the purpose of carrying out the test then in progress.

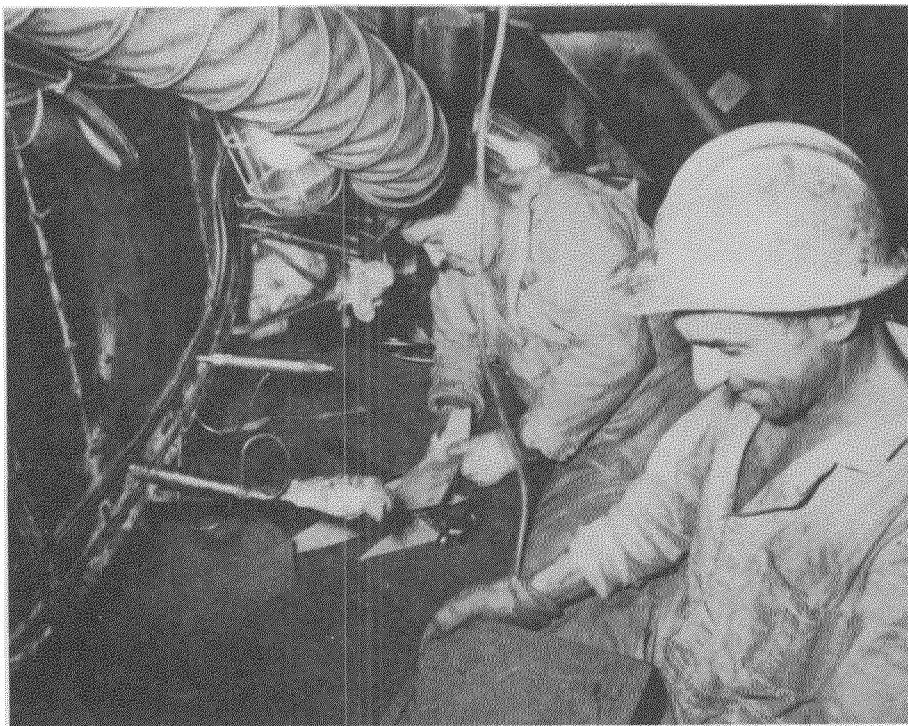


Fig. 12

Field graphite installation

After removal of the offset handling mechanism following reduction of the sodium level, observation of the tops of the dummy core and blanket revealed that a number of these subassemblies were displaced from their proper positions. Based on these observations and on remote viewing of the hold-down fingers using mirrors, it was concluded that some of the hold-down fingers had been bent.

Subsequent investigation indicated that this had been caused by the subassembly heads sticking to the fingers when the hold-down plate was raised; see Fig. 15, a model of the hold-down mechanism. Since no sweep arm was included in the design at this time, this sticking was not at once detected, and the plug was rotated with the heads engaged to the fingers, causing bending of the hold-down fingers and the subassemblies. When the hold-down was later re-engaged to the core, a complete misengagement of some of the fingers took place. The resulting non-seating of certain dummy subassemblies to the support plate in turn gave rise to flow erosion of support plate holes.

Hold-down mechanism modifications

A number of design changes to the hold-down mechanism and associated parts were made, both in order to correct the causes of the difficulties pre-

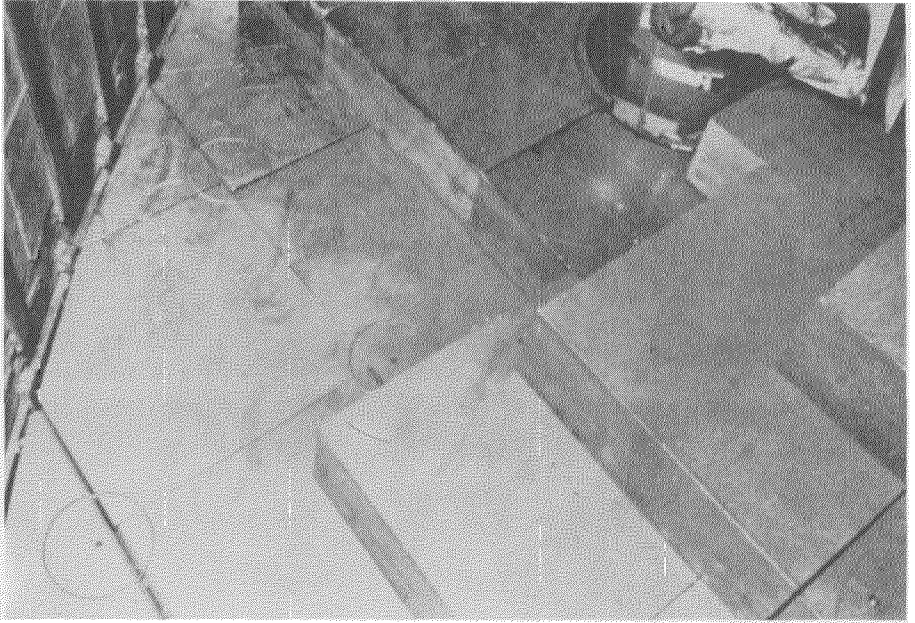


Fig. 13

Field graphite installed dry - metal ring fasteners used

viously encountered and to improve the design in other areas. The hold-down fingers have been strengthened so that if a similar malfunction occurs, subassembly handling heads will be bent without bending the fingers. The possibility of subassembly handling heads again sticking in the hold-down finger sockets has been eliminated by redesigning the finger socket to apply hold-down load on the end face of the subassembly handling head rather than on the conical surface. This end-bearing finger socket design is the outcome of a series of tests in high-temperature sodium which showed that sticking could occur with application of hold-down load on the conical surface but not with load on the end face. Fig. 16 shows the position of the finger-plate assembly, and Fig. 17 is a photograph of the replacement assembly before installation.

Offset handling mechanism modifications

Several design changes were made to ensure operation of the offset handling mechanism under conditions of abnormal stress. The gripper linkage was strengthened so that, even for severe sticking in the mechanism, enough strength exists to permit application of impact loads heavy enough to force the gripper open. The stabilizer lower plate, or foot, was redesigned to assure that the offset handling mechanism will not again be locked in position over a partially raised subassembly. This was the situation at the time the damage occurred. The stabilizer assembly and its attachment to

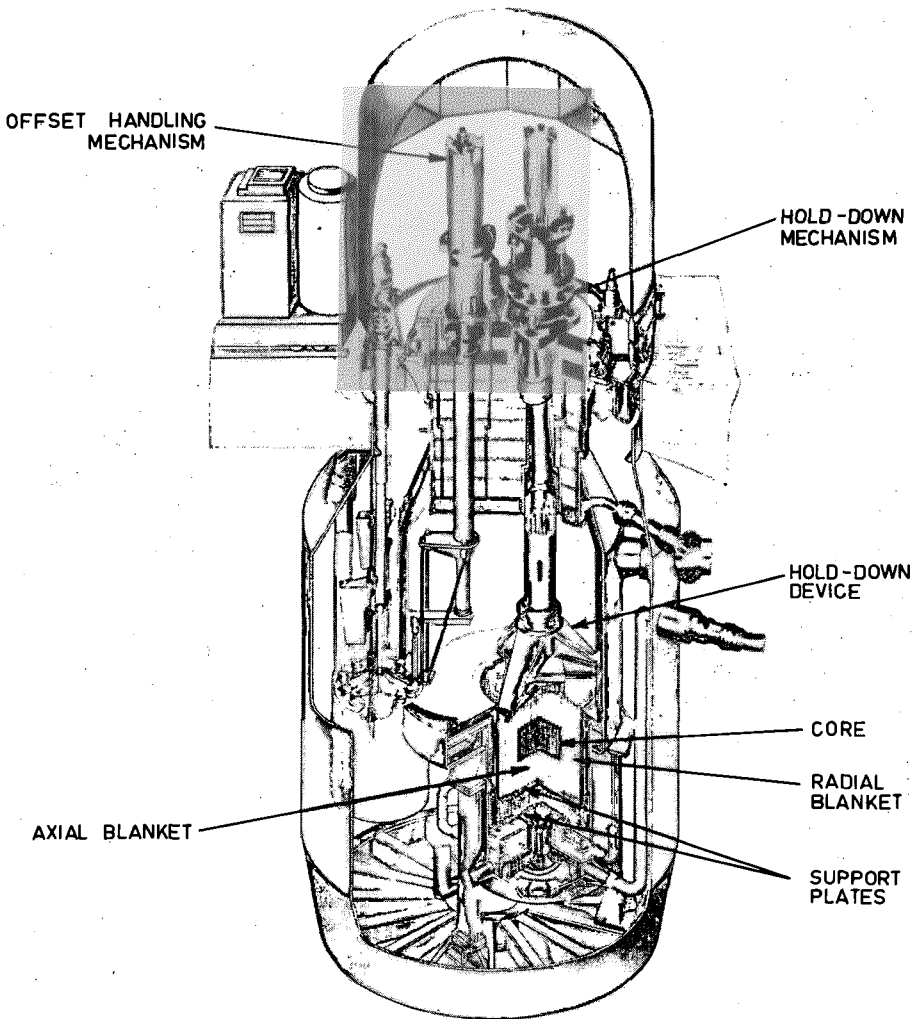


Fig. 14

Perspective view of reactor

the azimuth pipe has been strengthened to withstand higher lateral loads against the lower end before the onset of permanent deformation in the structure occurs. A completely separate digital position readout has been provided in the control room to show the actual azimuth position of the plug and offset handling mechanism at all times.

Addition of sweep device

A sweep arm operated from the top of the rotating shield plug was installed to determine if any subassemblies are raised before plug rotation.

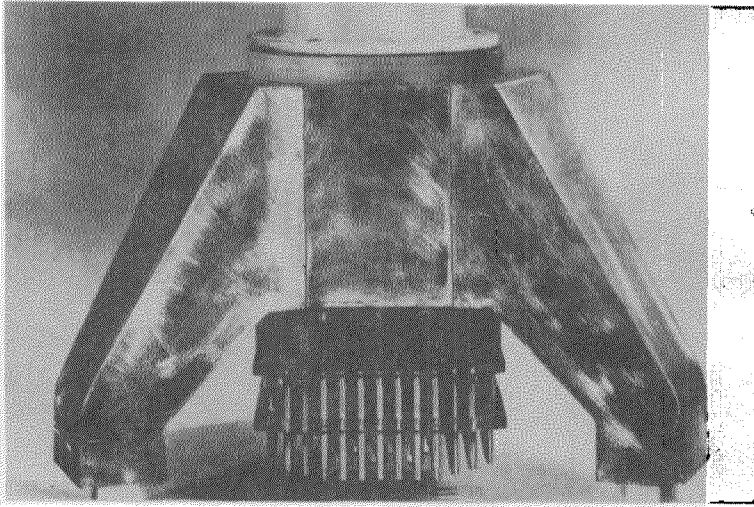


Fig. 15

Model of hold-down

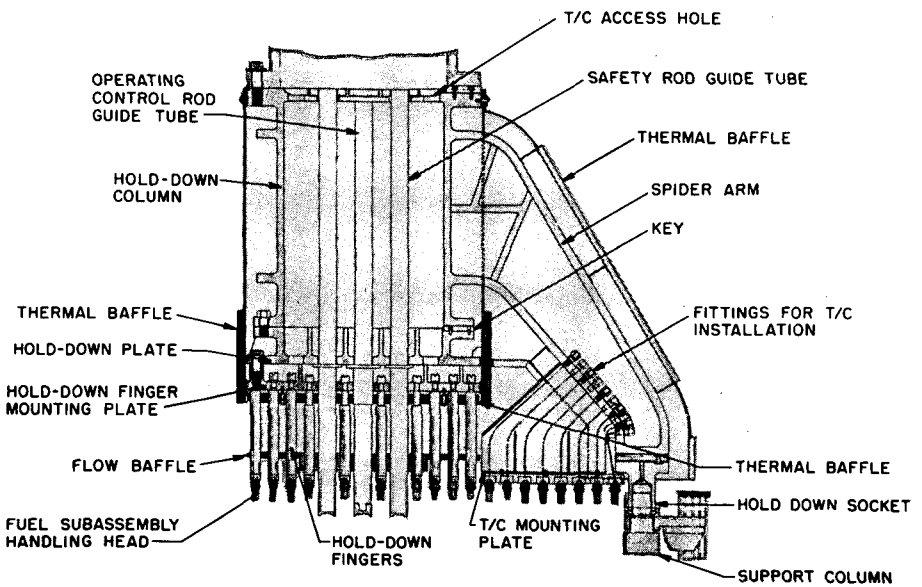


Fig. 16

Hold-down (sectional elevation)

This device will traverse the region above the handling heads of the core subassemblies.

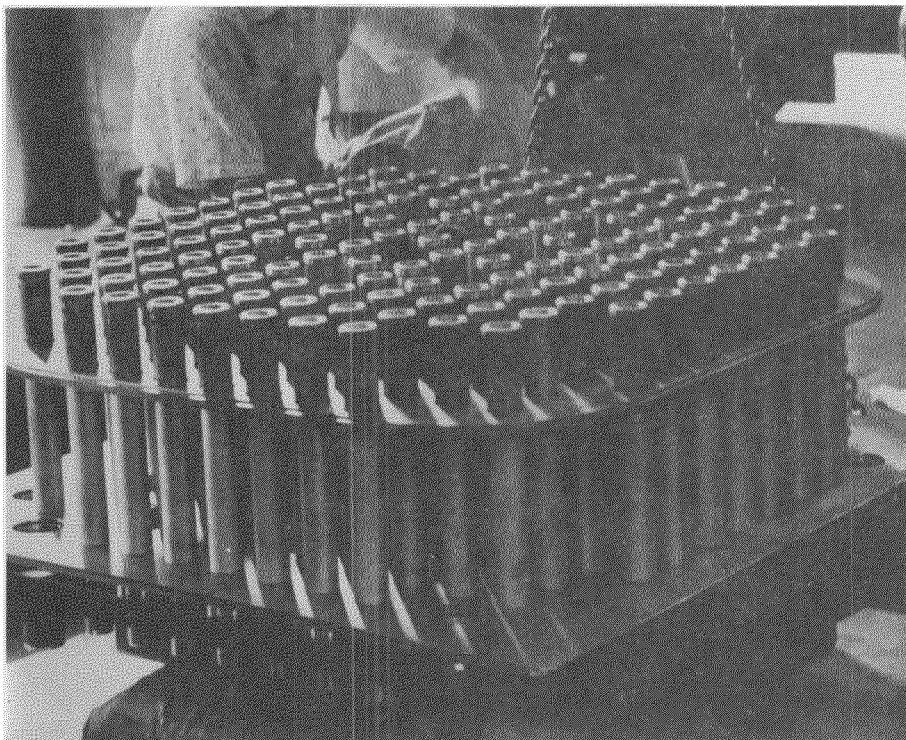


Fig. 17

Replacement finger plate assembly

Support plate rebushed

The support-plate assembly was also removed for inspection and as a result of evidence of flow damage, all of the support-plate holes were rebushed with Stellite insert rings to prevent such damage in the future. Fig. 18 shows the condition of the nitrided surface around the lower support-plate hole before the reworking.

Maintenance procedure used

At the time it was determined repairs had to be made to various components within the reactor vessel, the primary system was filled with 500°F sodium. After a careful study it was decided to use the following maintenance procedure:

- (a) Build an air lock which would fit into the offset handling-mechanism opening thereby providing personnel access to the inner reactor vessel.
- (b) Purchase special gas-tight suits and train maintenance personnel to work in these suits under restricted conditions.

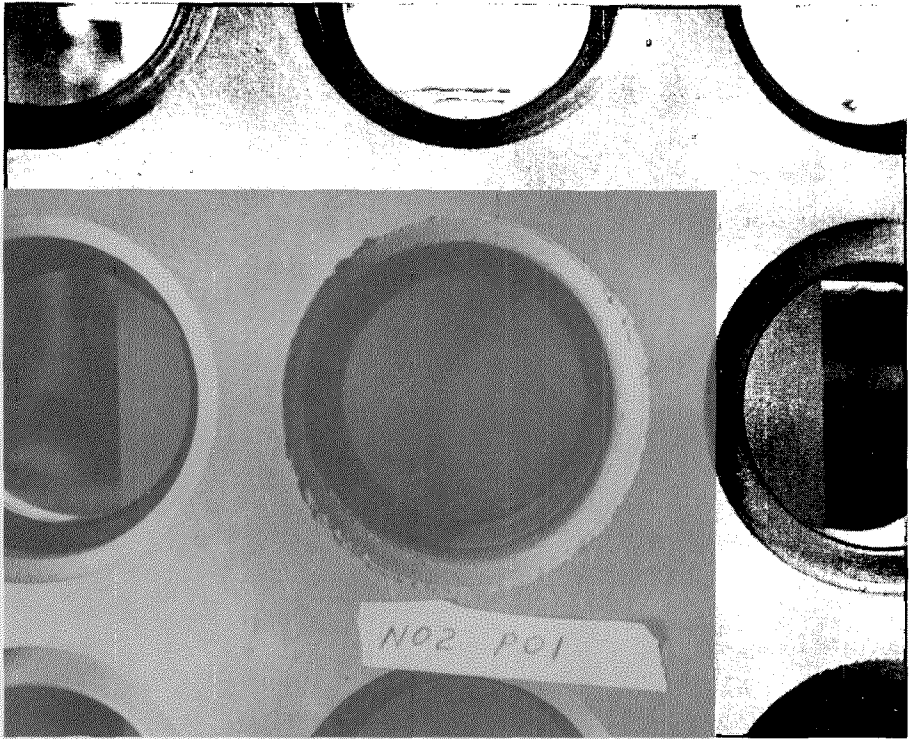


Fig. 18

Support plate flow damage

- (c) Remove the offset handling mechanism to provide access for piping, boroscopes, and other equipment.
- (d) Syphon the sodium out of the reactor vessel and the 30-in lines.
- (e) Allow the reactor vessel to cool, maintaining an argon atmosphere at all times.
- (f) Install the air lock and enter the vessel, two men at a time. Special care was taken to keep the moisture addition to the inner reactor atmosphere to a minimum to avoid damage to the stainless-steel reactor internals.

Personnel suits

The special gas-tight suits protected the men from the sodium on the surfaces of the reactor internals and likewise protected the reactor from the water and air given off by the men. These suits shown schematically in Fig. 19 were made of polyvinyl chloride. Initially a 10-oz nylon reinforced plastic was used, but testing disclosed unacceptable leaks occurred at the joints around the nylon threads. It was, therefore, decided to use a 20-oz non-reinforced material which was air tight but was more susceptible to tearing.

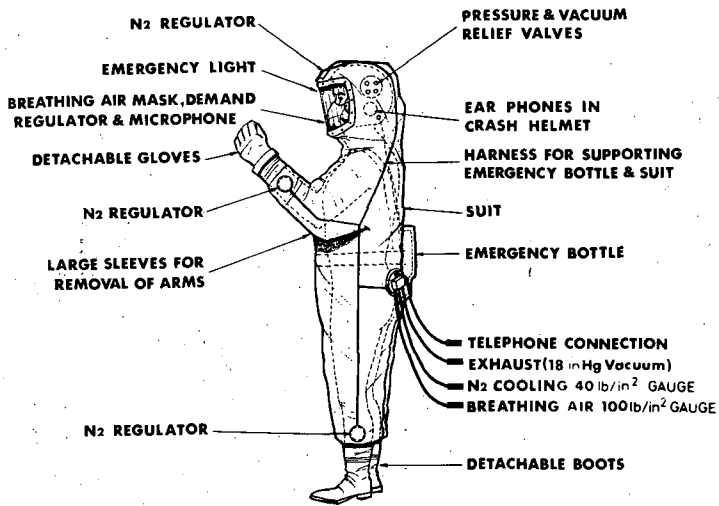


Fig. 19

Schematic of gas-tight suit

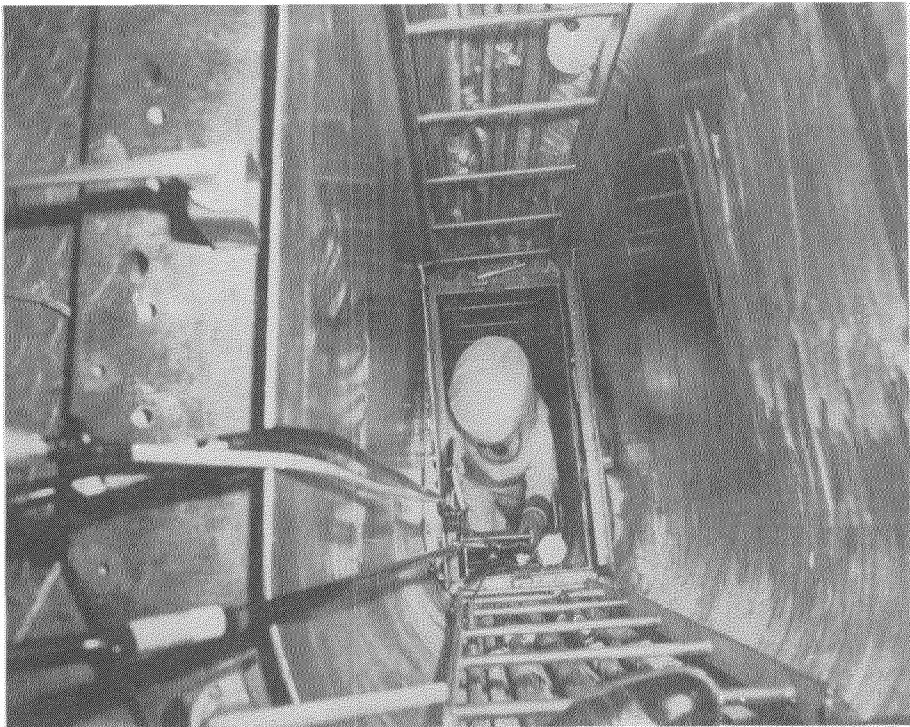


Fig. 20

Air lock

Air lock

The air lock which was used is shown in Fig. 20. It is 27 ft long, has two compartments and a gas-tight door on the top, one between the compartments, and one in the side of the bottom compartment. The lock is equipped with an inner communication system, breathing air system, nitrogen cooling system, and vacuum lines with connect, disconnect couplings and hoses. Two men connect their suits in either the upper or lower compartment or in the reactor vessel. The procedure followed in entering the reactor vessel required two men to descend into the lower compartment of the air lock and close the inner door. The lower compartment would then be purged with argon until the oxygen content was reduced to 100 ppm. The men would then open the lower door and enter the reactor vessel, connecting to the permanent hose manifold in the reactor vessel. Installation of the air lock in the reactor vessel is shown schematically in Fig. 21.

Numerous difficulties were encountered during the 1950 man-hours of work in the reactor vessel. These difficulties involved the equipment and the conditions under which the men were working. Rigid operating procedures required the men to leave the vessel as soon as there was any indication of a tear in the suit. Such a tear would be detected by the continuous

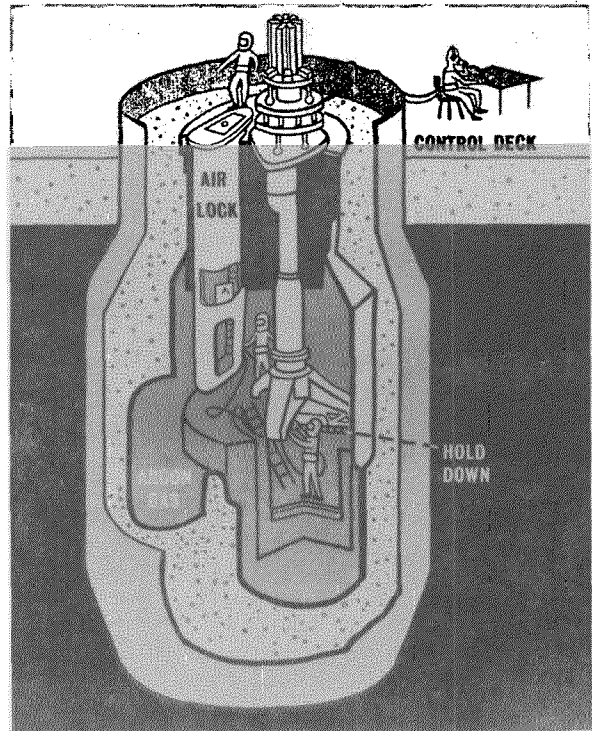


Fig. 21

Air-lock installation in reactor vessel

gas analysis system which would show the presence of an undue nitrogen condition. The amount of oxygen in the air fed to the breathing mask was increased to 30-35% when it was found that the mask did not completely discharge the exhaled air from the men's lungs. This step reduced severe fatigue which plagued the job during the first few weeks. It was necessary to set up a suit repair machine at the site to mend small tears in the poly-vinyl chloride. Seven suits in all were used, and twelve men in addition to two supervisors were trained to do the work.

Fig. 22 shows two men being dressed before entering the air lock. This operation required 30-45 min. A man is shown climbing to the top of the air lock in Fig. 23. Note the moisture on the mask. Photographs were taken inside the reactor. Fig. 24 is taken looking down on the support plate where careful inspections were made before installation of the rebushed plate.

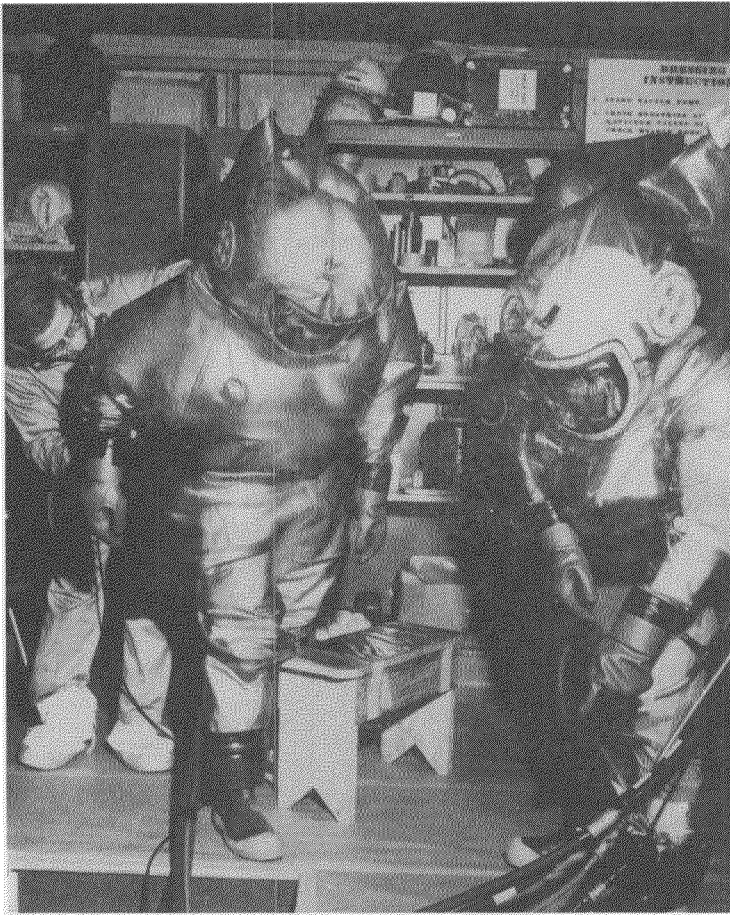


Fig. 22

Dressing for work in the reactor

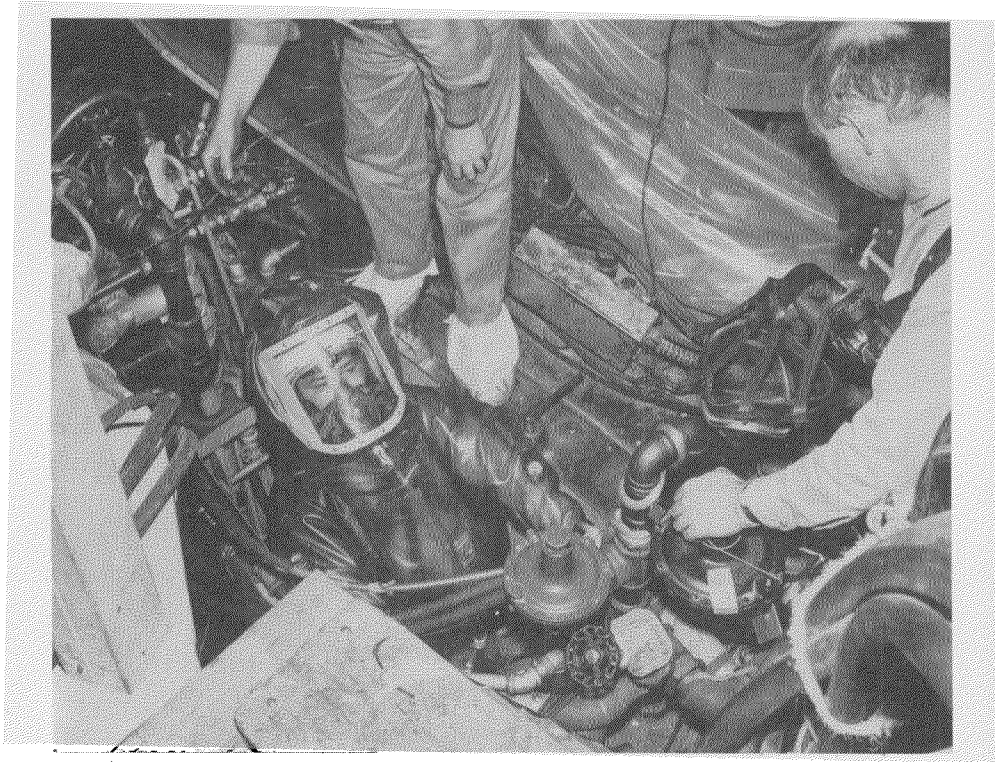


Fig. 23

Climbing out of the air lock

STEAM GENERATOR MODIFICATIONS

A schematic diagram of the steam generator design is shown in Fig. 25. Three identical liquid sodium-heated steam generators have been installed. These are vertical shell and tube, counter flow, once-through type units with water and steam in single wall tubes and sodium on the shell side. Steam will leave these units at $600 \text{ lb/in}^2\text{-}764^\circ\text{F}$.

Fig. 26, a photograph of a scale model of one of the steam generators, shows the unique tube arrangement and the liberal gas volume in the central section of the unit. The blanket of inert gas protects the tube sheet from sudden thermal transients in hot sodium and acts as a surge and expansion tank for the entire secondary sodium system. This gas blanket also serves to maintain a relatively constant temperature across the tube sheet.

Stress corrosion in No. 2 steam generator

Installation of the steam generators was completed in December 1961, when Fig. 27 was taken. One unit is installed in each of the three separate secondary sodium loops. The secondary sodium piping within the reactor

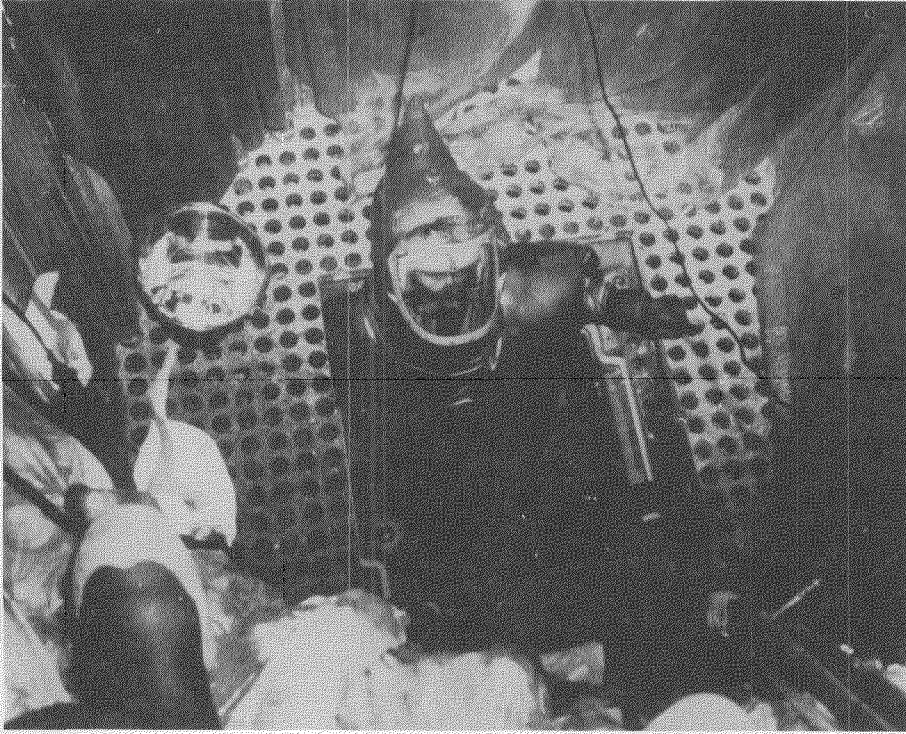


Fig. 24

Working in the support plate region

building is stainless steel. External to the reactor building the material is $2\frac{1}{4}\%$ chromium- 1% molybdenum steel. The secondary sodium pumps, the steam generators, and the 1200 tubes in each steam generator bundle are also fabricated of this same type of steel. Hydrostatic tests of the No. 2 unit during June 1962, disclosed tube failures which were caused by stress-corrosion cracking in the upper bend regions. Three conditions were found to exist, each of which contributes to and is a factor in such a failure – an active electrolyte, stressed tube material, and an elevated electrolyte temperature. The corrodants which are known to cause stress-cracking are nitrates and hydroxides, both of which were present in the cleaning solution or were generated from chemicals which were used in the solution. Nitrox ($1/3$ sodium hydroxide – NaOH and $2/3$ sodium nitrite – NaNO_2) was used by the fabricator in the cleaning of the tubing during assembly. It is speculated that the solution was not totally drained from the tubes and that blowing with compressed air did not clear the tubes of this cleaning solution.

With the determination that there was extensive stress-corrosion cracking within the No. 2 bundle, it was sent back to the fabricator for complete retubing. Numbers 1 and 3 steam generators were tested at 500°F and 1700 lb/in^2 conditions and no leaks were detected. A sampling of tubes was removed from the No. 1 and No. 3 units for metallurgical examination, even

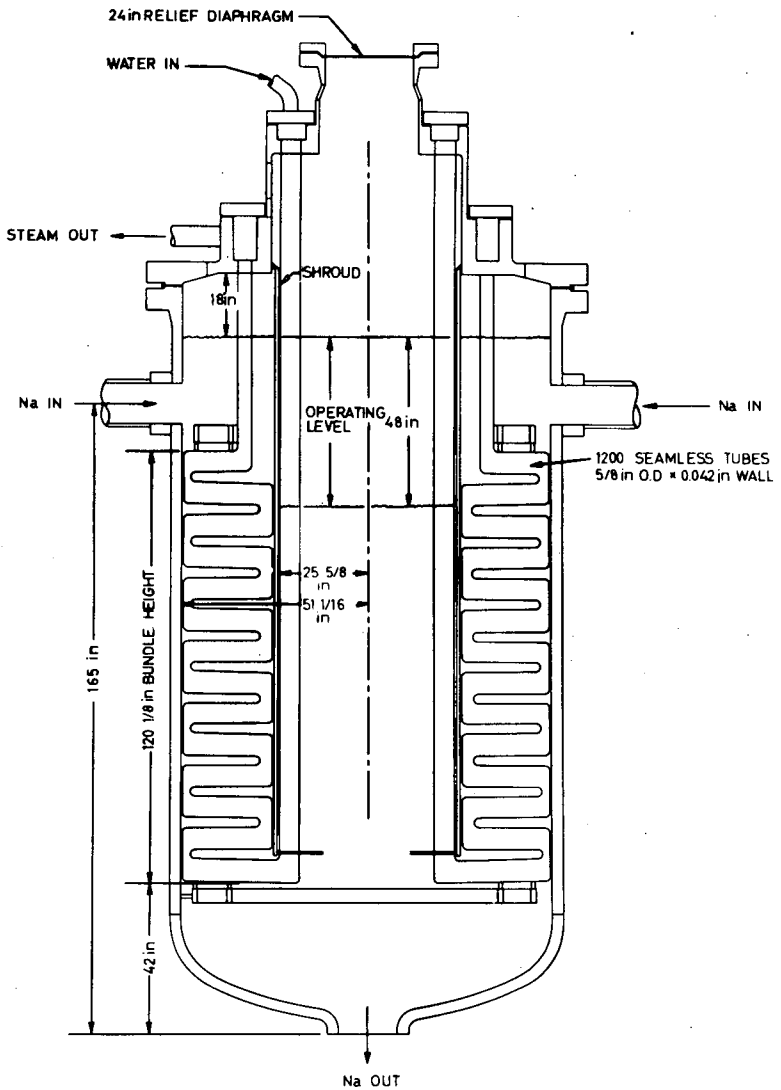


Fig. 25

Steam-generator design

though these units had not been exposed to the same adverse conditions as the No. 2 unit. No such defects were discovered. To assure their integrity, however, tube surfaces were acid cleaned, and all three units were stress relieved in place in the field. Hydrogen detection equipment was installed to monitor the argon cover gas above the sodium in each steam generator. All bundles were back in their shells ready for service in November 1962.

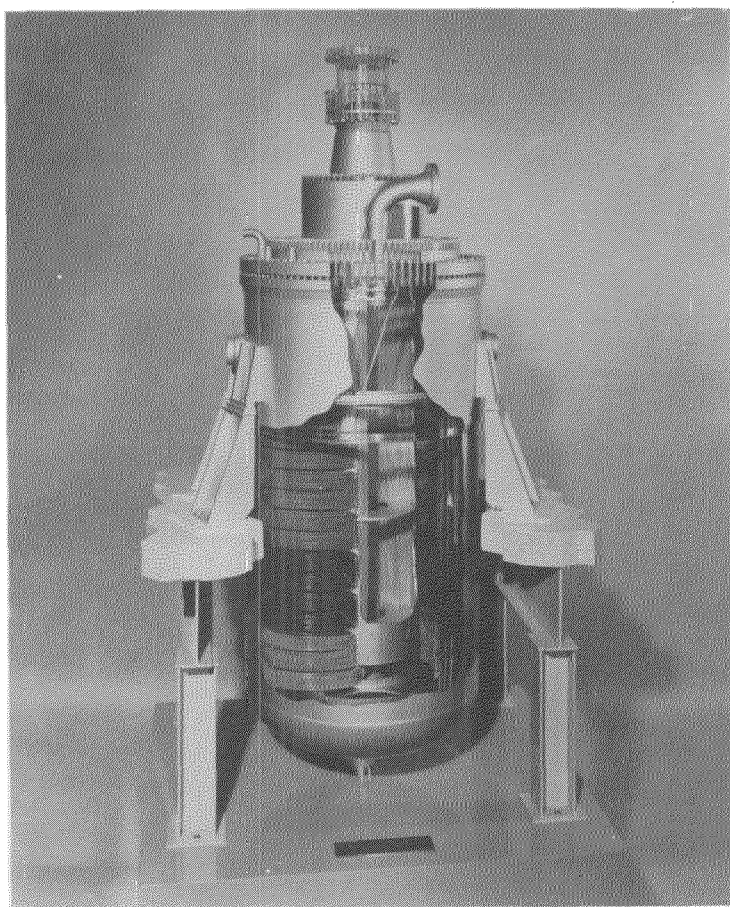


Fig. 26

Model of the steam generator

Sodium testing

No. 1 unit, the first to be tested, was first filled with water and then was filled with sodium on 28 November 1962. Cold trapping for purposes of cleaning up the sodium system was started promptly. To effectively cold trap and to test the secondary sodium pump, sodium was circulated continuously.

Sodium-water reaction

On the afternoon of 12 December, a sodium-water reaction took place in this unit. The rupture disc operated properly, and the pressure resulting from the reaction was relieved as planned through the relief piping and centrifuge. Only a small amount of sodium oxide was discharged from the centrifuge, and this was dissipated into the atmosphere without damage. There were no injuries.

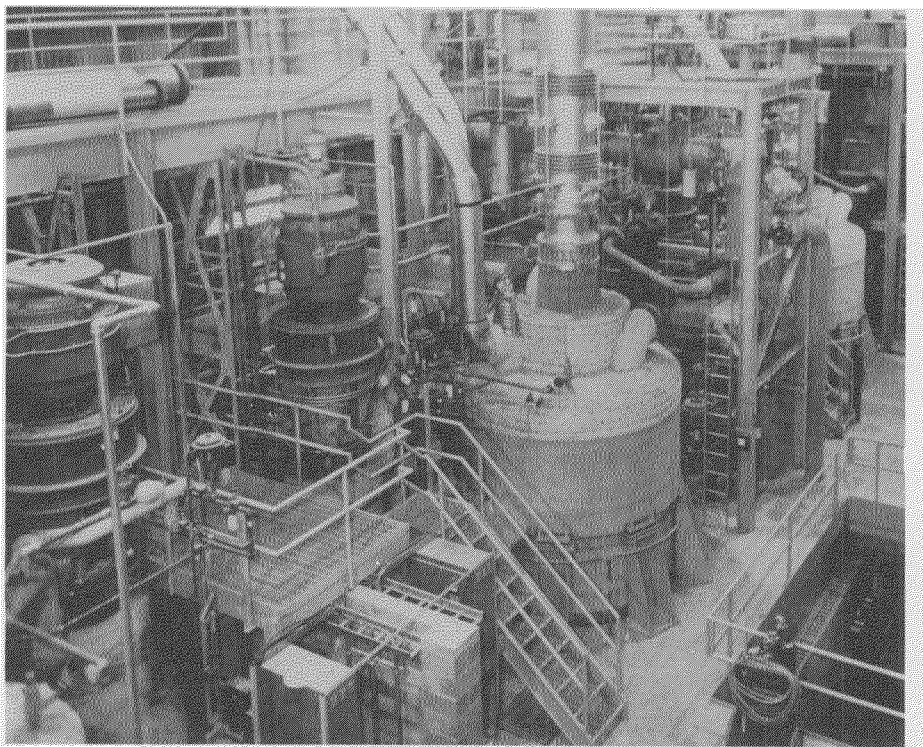


Fig. 27

Interior of steam - generator building

A short time before this incident plant operators had noted a rise in the hydrogen content and in the pressure in the No. 1 steam-generator cover gas. As soon as this pressure rise was detected, the operators immediately initiated isolation of this steam generator by manually operating the valves which dump water from the unit. These valves acted to drop the pressure from 500 lb/in² gauge to essentially atmospheric. About five seconds after opening of these valves was initiated, the rupture disc operated.

The steam and water manifolds were removed to determine the extent of the failure. Six leaking tubes were found, all located in the same general area in front of the east sodium inlet.

On 2 February, cold trapping of the sodium, which had been dumped into the storage tank after the reaction, was completed, and the No. 1 secondary sodium loop was refilled with this sodium. With a storage system temperature of 325°F, the saturation temperature of the sodium was 220°F. This indicates that this sodium, which had contained approximately 5000 ppm sodium oxide, was reasonably clean before filling. Six hours after the system was filled, the system temperature was 400°F and the saturation temperature was 380°F. Cold trapping of the system was started immediately. The system contained considerable contamination which went into solution in the clean sodium.

Vibration tests

Vibration tests were run shortly after filling with sodium at varying secondary pump speeds, 100 rpm to the maximum of 800 rpm. These tests indicated vibrations of only 0.0005- 0.0015-in amplitude and at frequencies varying from 140-3000 Hz. Subsequent inspection of the bundle disclosed that sodium-water reaction products formed around the tubes reduced the vibration effect.

Subsequent to the vibration tests, more tube leakage was noted. Nitrogen, which was maintained on the tube side, was first noticed in the argon cover gas over the sodium. The rate of leakage continued to increase during the five weeks of cold trapping and before draining the sodium in preparation for removal of the bundle. The system plugging temperature had been lowered below 300°F before draining.

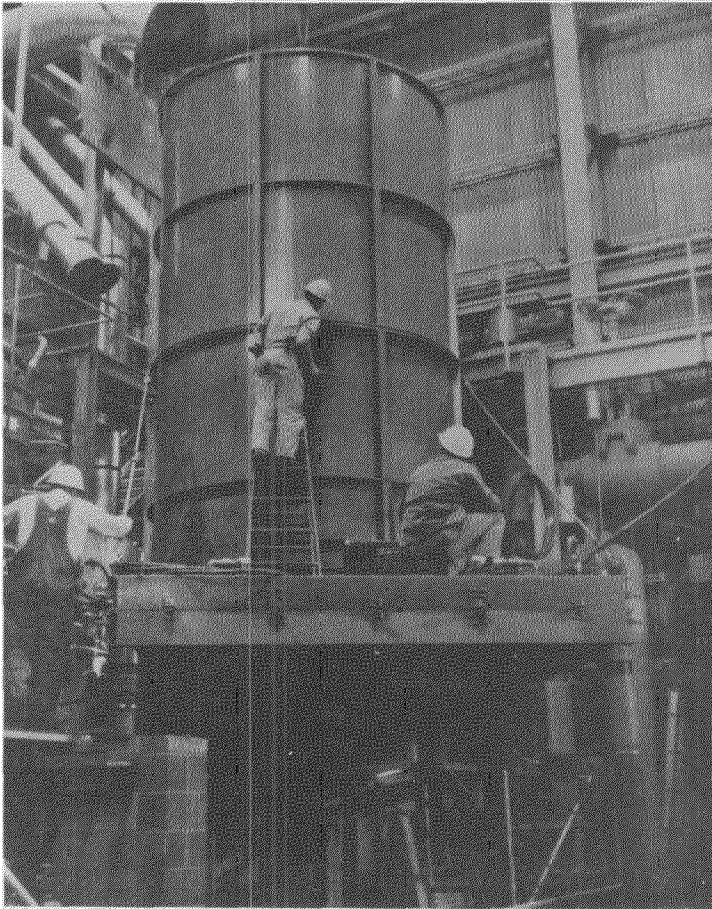


Fig. 28

Steam-generator gas-tight removal equipment

Additional leaking tubes

By removing the manifold covers and pressurizing the shell side with argon gas, 39 additional leaking tubes were located. Thirty-six of the leaking tubes were located in front of the west sodium inlet, while three additional leaking tubes were located in the region of the sodium reaction in front of the east sodium inlet.

Removal and cleaning

The bundle was removed utilizing the special equipment shown in Fig. 28. By inspecting through viewing ports, it was observed that the area above the location where the reaction occurred was covered with a heavy oxide deposit.

Cleaning of the bundle was completed after approximately eight hours exposure to ethyl alcohol. Alcohol was admitted slowly into the bottom of the cleaning vessel, and nitrogen was bubbled up through the alcohol for agitation; see Fig. 29. An inert atmosphere was maintained in the vessel at all times, and extreme care was exercised to prevent combustion of the alcohol or of the hydrogen which was evolved from the reaction. The gas pressure in the cleaning vessel was controlled easily with a small valve

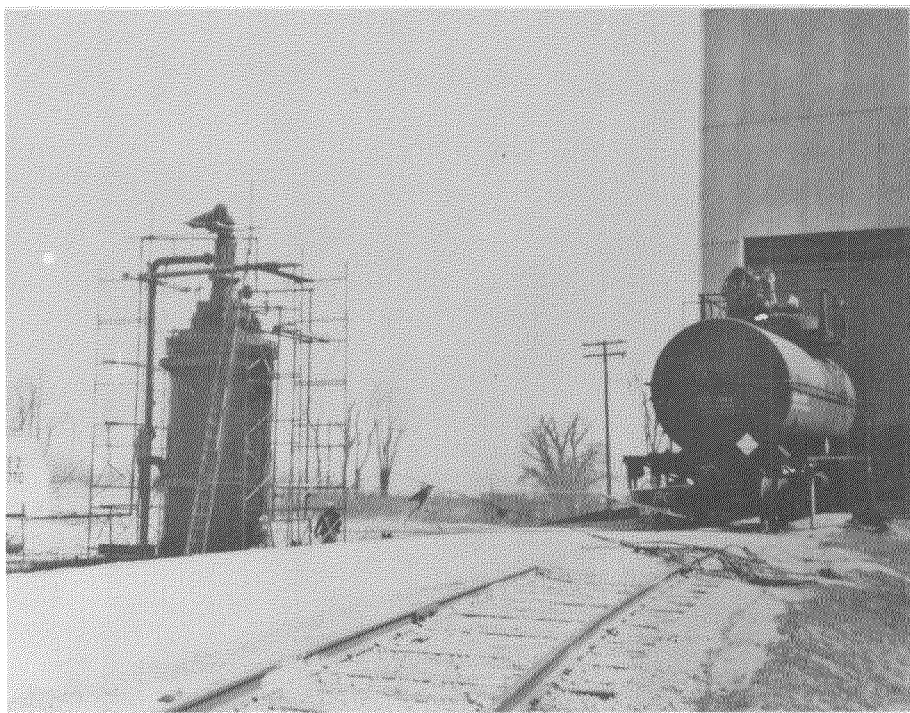


Fig. 29

Alcohol cleaning of steam-generator bundle

and the maximum pressure reached at any time was 4 lb/in² gauge. The maximum hydrogen content of the cover gas was 60%. Reaction between the alcohol and the residual sodium was quite mild. A preliminary analysis of the alcohol after cleaning showed that a total of 100 to 200 lb of sodium had been cleaned from the unit.

Vibration damage

A detailed visual and metallographic examination of the failures in the west inlet area reveals that the tubes failed due to vibration in the support bars. Figs. 30 and 31 show the enlarged support bar holes and evidence of tube rubbing. The damage, which was quite severe, was all located about four feet below the steam manifold where the tube makes the first horizontal pass through the inner support bar. The support bar clearances have increased considerably due to the hammering and sawing of the tubes in the support bars. Due to this action, the tubes finally wore thin and either partially or completely broke off. Several tubes were severed and these tubes caused considerable rubbing damage to adjacent tubes. The original specified support bar clearance was one-sixteenth inch. This clearance now appears to be excessive and was great enough to allow sufficient movement

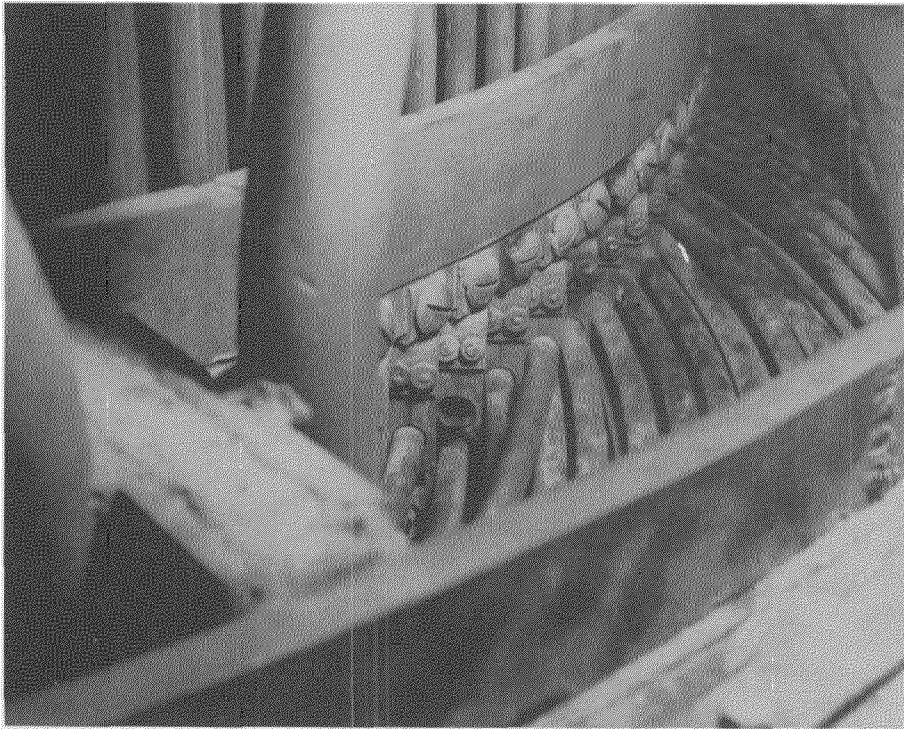


Fig. 30

Enlarged support bar holes

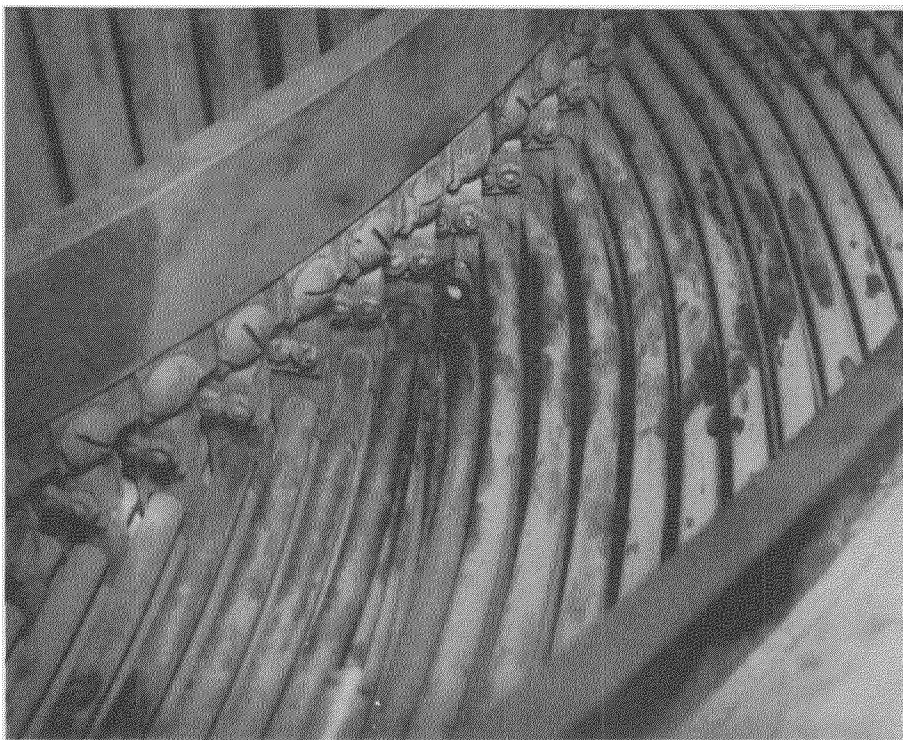


Fig. 31

Tube rubbing in support bar

to start the tube and support bar wear. As the hole size increased and the tube diameter reduced, the tube movement and damage rate increased.

Reaction area damage

Although the analysis of the six tubes which had failed originally opposite the east inlet nozzle has been somewhat masked by damage caused subsequent to the reaction, it is now possible to develop an hypothesis of the events before and subsequent to the reaction.

The failures in this area are all located at or near the inner support bar in the same locations as the vibration failures which occurred on the west side of the unit. Four failures occurred at the face of the support bar and were due to vibration damage. Four other failures (two tubes had two types of failures) were pressure ruptures. It appears that the tube walls of some tubes have thinned significantly due to the chemical action of the reactants with the tube metal. Our metallographic examination reveals that the tubes in the area of the reaction reached 1500°F, thus the reaction temperature must have been higher. When the tube metal had thinned, the internal pressure of 450 lb/in² gauge was sufficient to cause blow holes in the structurally and temperature weakened tube. One of the holes is a one-

inch long split which looks like a typical pressure failure. The other pressure failures, however, are one eighth to one quarter-inch circular shaped holes. It appears that the original failure in the reaction area was caused by vibration damage. The other failures occurred subsequent to the original failure and were due to a combination of chemical attack, temperature, and pressure.

No stress-corrosion cracking

All of the damaged tubes have been removed from the bundle and many have been metallurgically examined. Randomly selected banks of tubes 90° from the sodium inlet nozzles were removed and examined. There is no evidence of stress-corrosion cracking.

Full-scale vibration test

Inlet and outlet nozzles will be welded into one of the shipping containers so that full scale hydraulic flow testing can be conducted on the No. 3 steam generator bundle. This bundle, which was installed in its shell ready for service, will be removed and installed in the test loop. With full flow through the unit and proper vibration instrumentation, it will be possible to design suitable baffles and to study the effect of tube lacing. Baffles and possibly lacing will be added to all three bundles before operation.

CONCLUSION

The problems described are those that have been most troublesome and have had the greatest effect on the time schedule for the Fermi Plant. Numerous other problems of a technical nature have also been encountered during the three years the reactor plant has been subject to test. By the same token, we have been well satisfied with the operation of many components. Much has been learned, all of which emphasizes the value of a complete pre-operational testing programme in the building of a first-generation prototype reactor. While the emphasis in this review has been placed on particular trouble areas, it is significant that techniques have been developed for sampling and analysing sodium during the pre-operational test period. The primary sodium pumps have operated in a very satisfactory manner and the gas systems are performing well. The lessons learned from the test programme are providing valuable information in nuclear technology which should permit further advances in the development of power reactors in general and sodium-cooled fast reactors in particular.

ACKNOWLEDGEMENTS

This description of certain technical problems encountered in the construction and testing of the plant and the steps taken to solve these problems reflects the efforts of many dedicated engineers. It is appropriate to express

particular appreciation to the following for their assistance in the preparation of this report:

- (a) Hazen Van Vliet, Graphite Project Engineer for PRDC.
- (b) William M. Everett, Project Engineer assigned to the reactor repair and remote maintenance work.
- (c) Charles M. Heidel, project Engineer coordinating steam generator repair.

DISCUSSION

L. ZWINGENBERGER: Do you know how much water entered into the reaction with the sodium?

R. HARTWELL: No, although it was clearly a fairly large quantity. Cleaning the 10 000-gal system required 35 days of cold trapping, and significant amounts of "crud" were found above the reaction area when the bundle was removed. Metallurgical examination showed that the tubing had reached a temperature of 1500°F. A hydrogen detector was in the unit at the time of failure. Our procedure for operation prescribed that the operator would shut off the unit and that the pressure would rise. The pressure did not rise significantly but the hydrogen content of the cover gas did. In fact, hydrogen detection may well prove useful in steam generators, and we are working on the required instrumentation. When this sodium-water reaction occurred, the operators actually dumped the unit a few seconds before the diaphragm blew.

L. ZWINGENBERGER: Was there any device behind the rupture disc for separating water and sodium?

R. HARTWELL: Yes, a simple centrifuge separator is installed about 35 ft from the disc, its purpose being to catch large burning particles. Inspection after the reaction showed little, if any, "crud" in the separator, so it would appear that the reaction products stayed in the vessel.

As I said before, it took us quite a long time to clean up the sodium system, and during that time we tried to investigate the vibration effects by installing vibration detectors in and around the tubing that had failed. We determined that six tubes had failed in the initial accident. The vibration detectors revealed no undue vibration, but when we removed the bundle and found all the effects of the vibration we could see why: a lot of oxide had been packed in around the tubing, and we had unknowingly tried to detect vibration in tubes that were now held firmly by the reaction products.

L. ZWINGENBERGER: Was there an oxygen-hydrogen explosion?

R. HARTWELL: No. The hydrogen detector indicated a build-up in the hydrogen content of the gas, but no pressure increase was noted until about 15 s before the reaction. The relief disc blew at about 60 lb/in² (normal pressure is 1-3 lb/in²).

S. YIFTAH: It is generally accepted that future fast power reactors will be fuelled with plutonium. Is it therefore definitely intended to operate the Enrico Fermi reactor at a later stage with plutonium fuel? If so, what type of fuel will be used, and what is the present schedule for loading?

R. HARTWELL: We agree that future fast reactors will use plutonium, but there is no definite plan to fuel the Enrico Fermi reactor with it. Our

present supply of uranium-molybdenum fuel will serve to complete low-power testing and to operate at power for a period of 18-24 months. This does not, of course, allow enough time to procure proven plutonium fuel designs. We are therefore studying interim fuels that might be used after this period, including plutonium ceramets and mixed plutonium-uranium oxides.

N.N. ARISTARKHOV: In your recent published articles on the Enrico Fermi reactor there is a reference to the discovery of fuel-element instability due to coolant flow. Could you say what the causes of this instability or deformation are, how it manifests itself, and what steps you have taken to overcome it? Is the reduction of reactor power to 60 MW linked with this phenomenon?

R. HARTWELL: Fig. 7 shows a cut-away fuel element with grid-type spacers along the 30-in pin length. It was found that the uranium-molybdenum alloy did not provide the rigidity needed for the original, more flexible pin-support system. The addition of these 17 grid supports, one every 2 in has increased the pressure-drop through the fuel element. To avoid fuel-can distortion, the flow was limited to the equivalent of 200 MW(t). This is a limitation imposed by the particular core, however, and it will not be found in future cores. The plant has a nominal rating of 300 MW(t) and should be suitable for significant overload.

ОПЫТ ЭКСПЛУАТАЦИИ РЕАКТОРА БР-5

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СССР

Abstract — Résumé — Аннотация — Resumen

OPERATING EXPERIENCE WITH THE BR-5 REACTOR. The paper discusses the carrying-out of repair and maintenance work on the radioactive liquid-metal circuit of the BR-5 fast neutron reactor. Attention is also given to problems of reactor operation after achievement of the planned 2% fuel burn-up with some disturbance of leak-tightness in individual fuel elements. An account is given of experience in discharging the active section, examining the condition and leak-tightness of the fuel elements, and decontaminating the equipment and piping of the first radioactive circuit after reaching 5% fuel burn-up.

EXPÉRIENCE ACQUISE AUPRÈS DU RÉACTEUR BR-5. Dans ce mémoire les auteurs décrivent l'exécution des réparations et des travaux d'entretien dans le circuit radioactif liquide-métal du réacteur à neutrons rapides BR-5. Ils étudient également les problèmes liés au fonctionnement du réacteur au taux de combustion de 2% prévu avec quelques défauts d'étanchéité dans des éléments combustibles particuliers. Ils décrivent le déchargement en zone active et examinent les conditions d'étanchéité des éléments combustibles. Ainsi que la décontamination de l'appareillage et des tuyauteries du premier circuit radioactif après avoir atteint un taux de combustion de 5%.

ОПЫТ ЭКСПЛУАТАЦИИ РЕАКТОРА БР-5. В докладе рассматриваются вопросы производства ремонтных работ на радиоактивном жидкометаллическом контуре реактора на быстрых нейтронах БР-5. Освещаются вопросы эксплуатации реактора после достижения проектного 2%-ного выгорания топлива при наличии некоторого нарушения плотности отдельных тепловыделяющих элементов. Проводится опыт по разгрузке активной зоны, исследованию состояния и герметичности тепловыделяющих элементов, дезактивации оборудования и трубопроводов I радиоактивного контура после достижения 5% выгорания топлива.

EXPERIENCIA PRÁCTICA CON EL REACTOR BR-5. En la memoria se examinan los problemas planteados por el mantenimiento del circuito radiactivo de metal líquido del reactor de neutrones rápidos BR-5. Se tratan cuestiones relacionadas con la explotación del reactor una vez alcanzado el grado de combustión de 2%, previsto en el proyecto y luego de producirse ciertas alteraciones de la densidad de determinados elementos combustibles. Se describen la experiencia adquirida durante la descarga del cuerpo del reactor, las investigaciones del estado general y de la hermeticidad de los elementos combustibles y las operaciones de descontaminación de la instalación y de las tuberías del circuito radiactivo primario después de alcanzado un grado de combustión de 5%.

ВВЕДЕНИЕ

Для решения основных технических проблем, стоящих на пути промышленного развития реакторов на быстрых нейтронах, в Советском Союзе в 1958 и 1959 годах был сооружен исследовательский инженерный реактор БР-5, с натриевым охлаждением, мощностью 5000 квт. По своим основным параметрам этот реактор близок к параметрам экономически выгодных промышленных систем. Поток нейтронов в центре активной зоны реактора составляет $\sim 10^{15}$ н/см².сек, энергонапряженность ~ 500 квт/л и температура теплоносителя на выходе из реактора $\sim 500^\circ\text{C}$.

Краткое описание реакторной установки и опыт эксплуатации реактора в первоначальный период были даны в работах [1, 2].

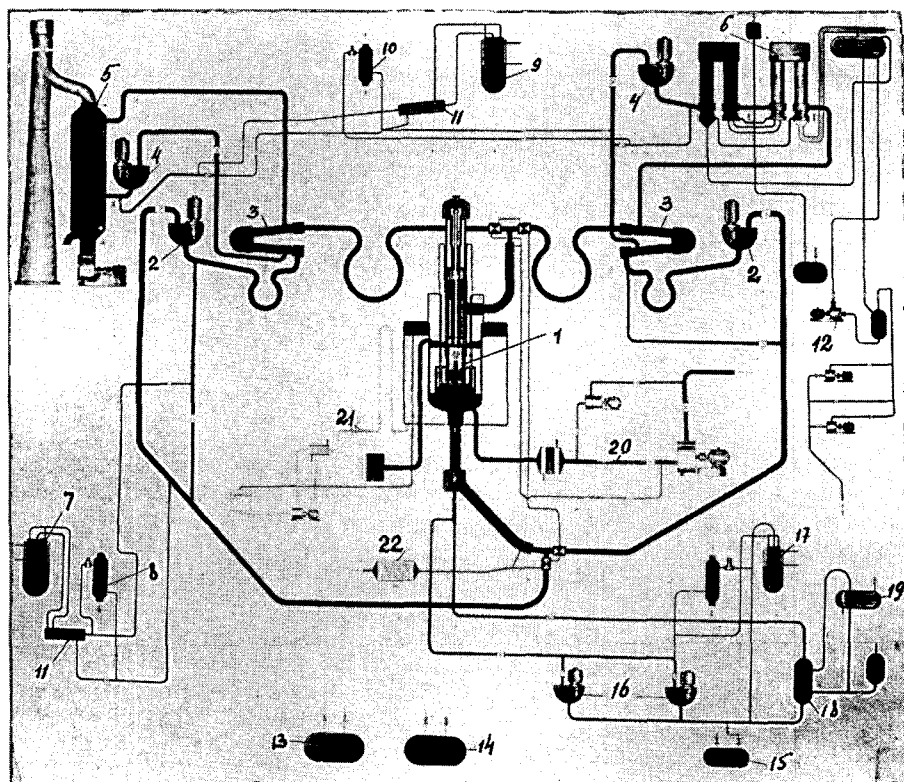


Рис.1

Технологическая схема реактора БР-5:

1 - активная зона; 2 - циркуляционный насос I контура; 3 - теплообменник Na - NaK; 4 - циркуляционный насос II контура; 5 - теплообменник NaK - воздух; 6 - парогенератор; 7 - фильтр-ловушка окислов натрия; 8 - индикатор окислов натрия; 9 - фильтр-ловушка окислов NaK; 10 - индикатор окислов NaK; 11 - рекуператор; 12 - циркуляционный насос дистиллята; 13 - сливной бак NaK; 14 - сливной бак Na; 15 - сливной бак петли; 16 - циркуляционный насос экспериментальной петли; 17 - фильтр-ловушка окислов петли; 18 - промежуточный теплообменник петли; 19 - конденсатор; 20 - тракт воздушного охлаждения экрана; 21 - тракт водяного охлаждения бака аппарата; 22 - электрокалорифер системы газового обогрева.

Начиная с января 1959 года, когда был осуществлен физический пуск реактора с натриевым теплоносителем, реактор эксплуатируется в различных режимах, в зависимости от требований экспериментов.

Успешная длительная эксплуатация установки позволила получить большой опыт работы с радиоактивным натриевым теплоносителем; провести комплексное испытание оборудования; провести испытание оксидных тепловыделяющих элементов на достижимую глубину выгорания в условиях, близких к рабочим.

Ниже приводятся некоторые результаты выполненных работ и исследований.

ПРОВЕДЕНИЕ РЕМОНТНЫХ РАБОТ НА ЖИДКОМЕТАЛЛИЧЕСКИХ КОНТУРАХ

В процессе эксплуатации реактора БР-5 с января 1959 года неоднократно возникала необходимость ремонта и замены оборудования основных жидкометаллических контуров, как-то: циркуляционных насосов, холодных ловушек окислов и т.п.

Насосы, осуществляющие циркуляцию натрия по I контуру и сплава натрий - калий по II контуру, представляют собой центробежные насосы погружного типа с вертикальным расположением вала. Чертеж насоса приведен на рис.2. Проточная часть насоса (рабочее колесо с улиткой) расположена в баке насоса объемом $\sim 0,6 \text{ м}^3$ под уровнем теплоносителя. Статор электродвигателя насоса герметизируется от внешней среды стальным колпаком-кожухом. Газовая полость кожуха электродвигателя через зазоры уплотнения вала насоса и по специальному газовому трубопроводу сообщается с газовой полостью бака насоса. Основными деталями насоса, определяющими продолжительность его работы без замены либо ремонта, являются подшипники качения, срок службы которых ограничивается несколькими тысячами часов. После введения дополнительного водяного охлаждения корпуса верхнего подшипника насоса срок работы насосов на нитках кожухов установки в среднем составил 8 тыс. часов, время работы отдельных насосов без их замены достигало 14 тыс. часов.

Замена или ремонт насосов I контура производились во время остановки реактора, и указанные работы требовали приблизительно недельной выдержки контура для спада активности натрия-24.

Нитка контура, насос которой выводился в ремонт, отсекалась от остальной части контура вентилями.

В первое время эксплуатации замена насосов I контура производилась при полностью сдренированном натрии из выведенной в ремонт нитки. В дальнейшем производилось лишь частичное дренирование натрия из ремонтируемой нитки до уровня на $\sim 1,5 \text{ м}$ ниже дна бака насоса. Оставшийся в нитке натрий замораживался для предотвращения утечки натрия в сдренированную нитку из остальной части контура через основные вентили. Частичное дренирование и замораживание натрия позволяло существенно снижать загрязнение контура окислами. После охлаждения выведенного в ремонт оборудования и трубопроводов до температуры $30 + 40^\circ\text{C}$ перед демонтажем насоса производилось сбрасывание инертного газа из газовой

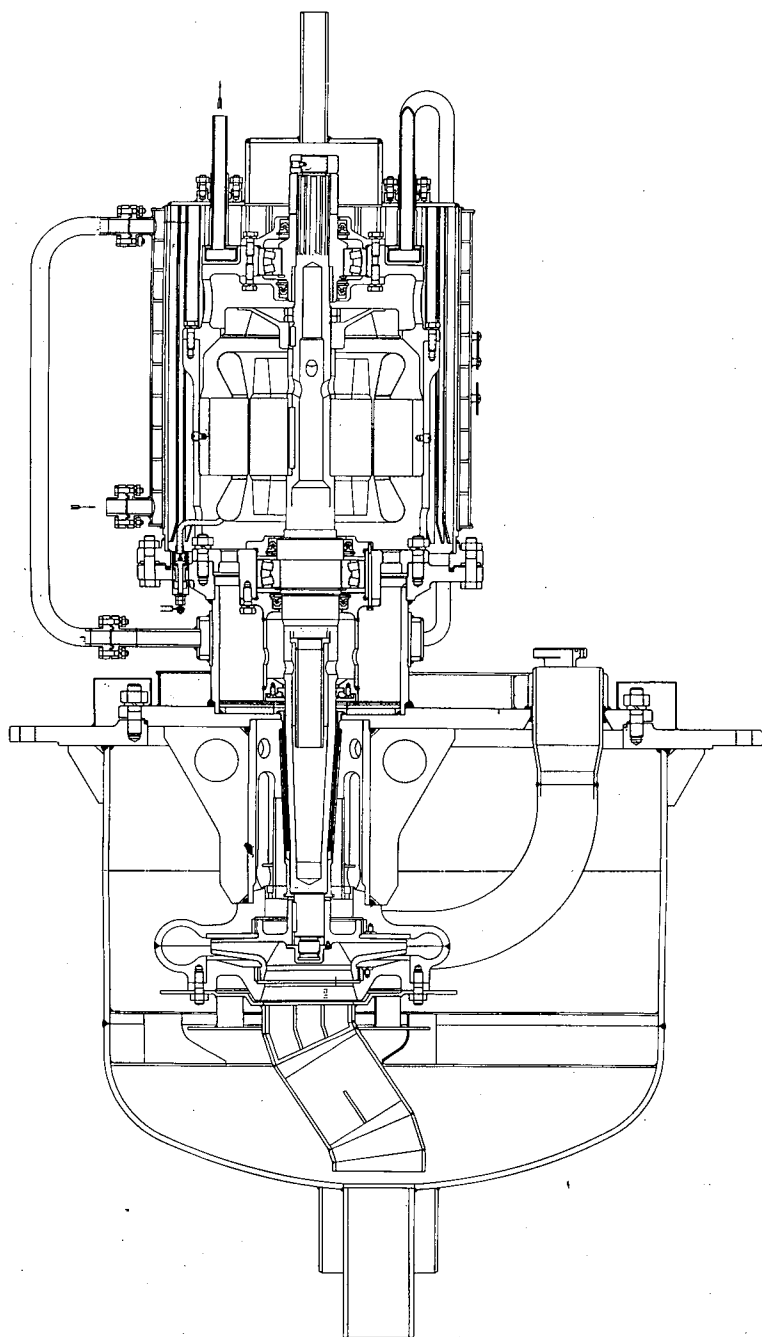


Рис. 2

Чертеж главного жидкометаллического насоса.

полости насоса и прилегающих трубопроводов до давления 1 ата. Газовые трубопроводы демонтируемого насоса разрезались механическим способом и глушились с обоих концов разжимными резиновыми пробками. Далее разрезался трубопровод теплоносителя на напоре насоса, и последний извлекался из бака. В течение всего времени демонтажа насоса (резка трубопроводов и извлечение насоса из бака) в ремонтируемой нитке создавалась инертная среда путем небольшой протечки аргона, подаваемого в бак насоса.

После съема насоса с обоих концов разрезанного напорного трубопровода и во всасывающий патрубок насосного бака устанавливались разжимные резиновые заглушки. Внутренняя полость насосного бака очищалась от остатков теплоносителя механическими скребками и увлажненным тампоном из асбеста. Окончательная очистка бака производилась тампонами, смоченными в спирте. Все работы по демонтажу и очистке оборудования от остатков теплоносителя производились в защитных скафандрах с принудительной подачей воздуха из-за наличия в боксах при производстве ремонтных работ аэрозольной активности.

После установки в бак нового насоса и сварки трубопроводов все сварные швы по контуру металла и газа подвергались рентгенокопии и гелиевым испытаниям под вакуумом. Гелиевым испытаниям под вакуумом подвергались также все фланцевые соединения насоса.

По окончании замены насоса оборудование и трубопроводы разогревались до температуры $\sim 150^{\circ}\text{C}$ включением электрического обогрева, причем при частично сдренированной нитке электрообогрев трубопроводов включался в такой последовательности, чтобы обеспечивалось расширение расплавленного металла в сторону свободного объема. Разогретая нитка заполнялась под вакуумом.

Замена насосов ниток II контура производилась всегда при полностью слитом сплаве натрий - калий из одной нитки и остановленной циркуляции сплава по другой.

Необходимость замены фильтров (холодных ловушек окислов натрия в процессе эксплуатации) возникала несколько раз.

В связи с тем, что фильтр I контура располагался в боксе 1-й нитки I контура, при его замене приходилось останавливать реактор и выждать спада γ -активности перед производством работ.

Первая замена фильтра-ловушки окислов I контура производилась при полностью сдренированной 1-й нитке и трубопроводов системы фильтрации. Конструкция фильтра не позволяет производить его дренирование. Поэтому натрий, оставшийся в фильтре после дренирования, объемом ~ 200 л замораживался. Для демонтажа фильтра трубопроводы системы фильтрации разрезались механическим способом при одновременной непрерывной подаче в нитку аргона для предотвращения окисления остатков натрия на трубах.

Перед приваркой трубопроводов к новому фильтру последний вакуумировался и заполнялся аргоном. Заполнение фильтра натрием производилось вместе с 1-й ниткой I контура.

Вторая замена фильтра производилась без дренирования нитки. Натрий в фильтре и подводящих трубопроводах замораживался. Тру-

бопроводы с натрием разрезались механическим способом, и концы труб на глубину ~100 мм очищались от натрия для надежной электро-сварки.

После предварительной продувки аргоном подготовленный для замены фильтр вваривался в контур и вакуумировался через специальный вентиль, приваренный к входному коллектору фильтра. После вакуумирования вентиль перекрывался и входной патрубок вентиля заваривался. Заполнение фильтра натрием из контура производилось после разогрева трубопроводов и оборудования системы фильтрации до температуры $t = 150^{\circ}\text{C}$.

Указанные замены фильтров производились, когда активность контура и фильтров определялась в основном активностью натрия-24.

Опыт работ по замене фильтров-ловушек окислов показал, что расположение фильтра в боксе совместно с основным оборудованием и трубопроводами одной из ниток I контура является неудобным, так как замена фильтра требовала длительной остановки реактора для спада γ -активности в боксе I контура до уровня, позволяющего производить ремонтные работы. Поэтому фильтр I контура был вынесен в специальный защитный бокс и предусмотрена возможность его дистанционного демонтажа. Замена фильтра в этом боксе производилась уже без прекращения циркуляции теплоносителя по контуру и остановки реактора. Демонтируемый фильтр отсекался от контура запорными вентилями, натрий в фильтре и его трубопроводах замораживался. После спада γ -активности в боксе фильтра последний демонтировался, а на его место устанавливался новый.

За период эксплуатации установки с января 1959 года по март 1963 года была произведена замена фильтра окислов: по I контуру натрия - 4 раза, по II контуру сплава - 1 раз.

Необходимо отметить, что содержание окислов в теплоносителе за время проведения ремонтных работ не поднималось выше $0,007\% \text{ O}_2$. Снижение содержания окислов в контуре до нормального уровня ($0,002 \div 0,003\% \text{ O}_2$) достигалось работой холодной ловушки в течение одних суток.

Опыт замены насосов и фильтров I и II контуров показал, что замена и ремонт оборудования жидкометаллических контуров при соответствующей подготовке и принятии мер, не представляют особых трудностей. При этом удается не допускать как значительного загрязнения контуров окислами щелочных металлов, так и переоблучения ремонтного персонала при ремонтных работах на радиоактивных контурах.

Следует указать на определенные технологические преимущества натрия по сравнению со сплавом натрий - калий.

Возможность замораживания натрия в трубопроводах и оборудования контура при ремонтных работах, отсутствие возгорания остатков натрия и продуктов его окислов в застывшем состоянии, удобства в сварочных работах на трубопроводах с остатками теплоносителя и ряд других факторов дают основание считать, что при выборе теплоносителя как для первого, так и для промежуточных контуров ядерной установки следует отдавать предпочтение натрию.

ЭКСПЛУАТАЦИЯ РЕАКТОРА ПОСЛЕ ДОСТИЖЕНИЯ ПРОЕКТНОГО ВЫГОРАНИЯ

В октябре 1960 года было достигнуто проектное 2%-ное выгорание плутония в тепловыделяющих элементах активной зоны реактора. Проведенные анализы радиоактивности теплоносителя и инертного газа первого контура показали отсутствие в теплоносителе α -активности и осколков деления. В инертном газе было обнаружено присутствие радиоактивного ксенона. Это свидетельствовало о некотором нарушении плотности тепловыделяющих элементов, не приводящему к размыву делящегося материала топливных элементов. В связи с тем, что одной из основных задач реактора БР-5 являлось массовое испытание оксидных тепловыделяющих элементов при глубоком выгорании, было принято решение о продолжении эксплуатации реактора без замены тепловыделяющих элементов до появления заметных признаков разрушения тепловыделяющих элементов. Контроль за целостностью тепловыделяющих элементов активной зоны в процессе эксплуатации реактора в 1959 - 1961 годах проводился несколькими методами:

- а) по изменению реактивности реактора в процессе выгорания плутония;
- б) по изменению остаточной активности теплоносителя первого контура (после распада основного изотопа Na^{24});
- в) по результатам радиохимического и спектрометрического анализа отбравшихся проб теплоносителя и инертного газа первого контура.

В течение всей эксплуатации реактора изменение реактивности реактора определялось выгоранием плутония, что указывало на отсутствие заметного вымывания горючего из тепловыделяющих элементов.

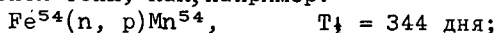
В течение 1959 - 1961 годов происходило изменение характера спада активности теплоносителя после остановки реактора. Если в начальный период работы реактора, когда выгорание горючего было меньше 2%, активность теплоносителя спадала в 100 тыс. раз с периодом полураспада 15 часов, то начиная с октября 1960 года, когда выгорание плутония достигло $\sim 2,4\%$, появилась остаточная активность. Остаточная активность теплоносителя по мере увеличения выгорания топлива увеличивалась (рис.3).

Остаточная активность натриевого теплоносителя связана с:

- а) изотопом Na^{22} ($T_{1/2} = 2,6$ г.), являющимся продуктом реакции $\text{Na}^{23}(n, 2n)\text{Na}^{22}$.

Активность Na^{22} в контуре пропорциональна интегральной мощности реактора;

- б) продуктами активации примесей теплоносителя. Активность примесей зависит от чистоты теплоносителя и внутренних поверхностей контура при монтаже;
- в) продуктами реакций (n, p) , протекающих на конструкционных материалах активной зоны, как, например:



- г) продуктами деления топливного материала активной зоны, выходящими из негерметичных тепловыделяющих элементов.

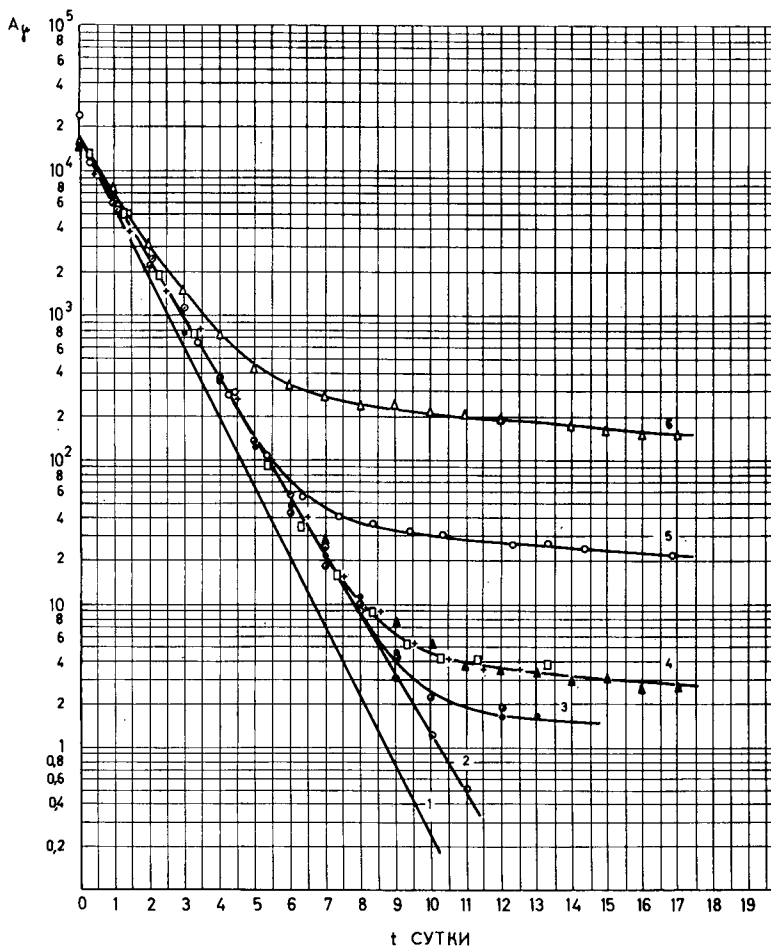


Рис.3

Кривые спада γ -активности в боксах I контура при заглушенном реакторе:

- 1 - спад γ -активности в боксах, обусловленной Na-24;
- 2 - спад γ -активности при остановках реактора в 1959 и 1960 годах;
- 3 - спад γ -активности в боксах при остановке реактора в октябре 1960 года;
- 4 - остановка реактора в мае 1961 года;
- 5 - остановка реактора в июле 1961 года;
- 6 - остановка реактора в сентябре 1961 года.

Указанные радиоактивные примеси могут не только растворяться или взвешиваться в теплоносителе, но и высаживаться на внутренних стенках трубопроводов и оборудования первого контура вследствие процесса переноса масс.

Активность проб теплоносителя, отобранных в феврале 1961 года (выгорание - 3,2%), кроме Na^{22} , в основном определялась изотопом Cs^{137} , образующимся в результате распада ксенона-137. Кроме того, был обнаружен изотоп Cs^{136} , который образуется при делении непосредственно.

Абсолютное значение радиоактивной составляющей цезия относительно радиоактивности изотопа Na^{22} составляло ~20%. Гамма-спектрометрический анализ внутренней поверхности образцов трубопровода, сделанный спустя 24 дня после остановки реактора показал на наличие значительной радиоактивности с энергией γ -лучей ~0,82 Мэв и 0,51 Мэв, что может быть приписано спектру изотопов Co^{58} и Mn^{54} . Следует отметить, что обе эти активности не удавалось удалить механическим способом (протиркой) и обработкой водой.

После остановки реактора 29 апреля 1961 года, когда выгорание топлива достигло 3,9%, общая остаточная активность отобранной пробы теплоносителя после трехмесячной выдержки составляла $2,3 \cdot 10^4$ расп/сек на 1 г натрия и распределялась между компонентами следующим образом:

Cs ~ $1,0 \cdot 10^4$ расп/сек на 1 г натрия;

Na^{22} ~ $1,3 \cdot 10^4$ расп/сек на 1 г натрия.

Таким образом, за время с 22 февраля 1961 года по 29 апреля 1961 года содержание цезия относительно Na^{22} выросло вдвое. Остаточная активность теплоносителя после остановки 17 июля 1961 года (выгорание 4,55%) выросла до $\sim 2,5 \cdot 10^5$ расп/сек на 1 г Na, т.е. в 10 раз по сравнению с апрелем.

Активность цезия составляла ~70% общей активности теплоносителя 1 сентября 1961 года, когда выгорание топлива достигло 5%; реактор был остановлен в связи с выходом из строя одного из циркуляционных насосов первого контура. После полного распада изотопа Na^{24} , через 14 дней после остановки γ -фон в помещениях первого контура составлял несколько тысяч мкр/сек. Дренаживание теплоносителя из отдельных ниток контура не снизило активности оборудования трубопроводов первого контура, что свидетельствовало о высаждении значительной части активности на внутренних стенках контура. Характер γ -спектра проб теплоносителя не изменился по сравнению с пробами, отобранными в июле. Однако остаточная активность примесей теплоносителя выросла в десять раз, главным образом за счет цезия. Значительное увеличение радиоактивности теплоносителя первого контура, обусловленной осколками деления, а также появление α -активности, связанной плутонием, свидетельствовали о разрушении оболочек некоторых тепловыделяющих элементов.

В связи с необходимостью исследования герметичности оболочек тепловыделяющих элементов и проведения дезактивации оборудования и трубопроводов первого контура в октябре 1961 года была проведена полная разгрузка активной зоны реактора.

РАЗГРУЗКА АКТИВНОЙ ЗОНЫ И ОТБРАКОВКА ПАКЕТОВ ТЕПЛОВЫДЕЛЯЮЩИХ ЭЛЕМЕНТОВ

В связи со значительной активностью рабочих пакетов, а также радиоактивной загрязненностью натрия первого контура и негерметичностью оболочек тепловыделяющих элементов части пакетов была проведена тщательная подготовка к разгрузке реактора. Особое внимание было обращено на обеспечение радиационной техники безопасности.

Порядок разгрузки активной зоны реактора был принят от центра к периферии. Выгрузка пакетов производилась с помощью разгрузочного контейнера в ячейки хранилища пакетов. Принудительное охлаждение пакета в контейнере при транспортировке не производилось в связи с незначительным остаточным тепловыделением в пакете. К моменту разгрузки активной зоны (через месяц после останковки реактора), остаточное тепловыделение в пакетах за счет осколков деления снизилось до 40 - 60 Вт на пакет. Проведенные расчеты показали, что при указанной величине тепловыделения, максимальная температура в центре тепловыделяющих элементов пакета, установленного в ячейку хранилища и охлаждаемого только посредством естественной конвекции воздуха снаружи пакета, не может превышать 500°C.

В связи с этим было принято решение ограничиться охлаждением пакетов в хранилище за счет естественной конвекции воздуха, усиленной вытяжной вентиляцией хранилища. Полная разгрузка реактора заняла 8 дней. В хранилище были выгружены все 120 рабочих и экранных пакетов.

Одиннадцать пакетов, выгруженных из активной зоны, были отмыты в специальной системе паровой обдувки. Пакет, предназначенный для отмывки, с помощью разгрузочного контейнера переносился в предварительно прогретый паром специальный промывочный колодец. Колодец герметично закрывался крышкой, и через него давался проток пара давлением ~2 атм с температурой ~130 + 140°C. После прохода через промывочный колодец пар конденсировался в конденсаторе. Конденсат охлаждался в доохладителе и сливался в сборники промывочных вод. Проток пара через пакет продолжался в среднем ~1 час с расходом ~250 пара/кг час. В процессе отмывки контролировалась γ -активность воды, сконденсированной из пара, проходящего через пакет. При отмывке четырех из одиннадцати пакетов было замечено значительное повышение активности конденсата, что указывало на наличие разрушения оболочек тепловыделяющих элементов у этих четырех пакетов. Анализ обмывочной воды показал наличие в ней плутония и осколков деления. Значительная активность обмывочной воды, присутствие в ней α -активности (плутония) и осколков деления подтвердили предположение о разрушении оболочек некоторых тепловыделяющих элементов.

В связи с тем, что паровая отмывка приводила к вымыву плутония из тепловыделяющих элементов, имеющих дефекты в оболочках, было принято решение паровую отмывку пакетов прекратить и произвести исследование герметичности оболочек ТВЭЛ всех пакетов.

Для определения целостности оболочек тепловыделяющих элементов была разработана методика, основанная на измерении величины радиоактивности газа, отсасываемого из пакета тепловыделяющих элементов. Проверка герметичности проводилась непосредственно в ячейках хранилища, в которых находились пакеты. Схема установки приведена на рис.4. Измерения проводились следующим образом: сверху на пакет, находящийся в ячейке хранилища, устанавливался специально изготовленный чехол, в верхней части которого был установлен фильтродержатель с двумя фильтрами, предназначенными для улавливания радиоактивных аэрозолей. Для измерения активности газа применялась ионизационная камера ДЗ-70 объемом 70 л.

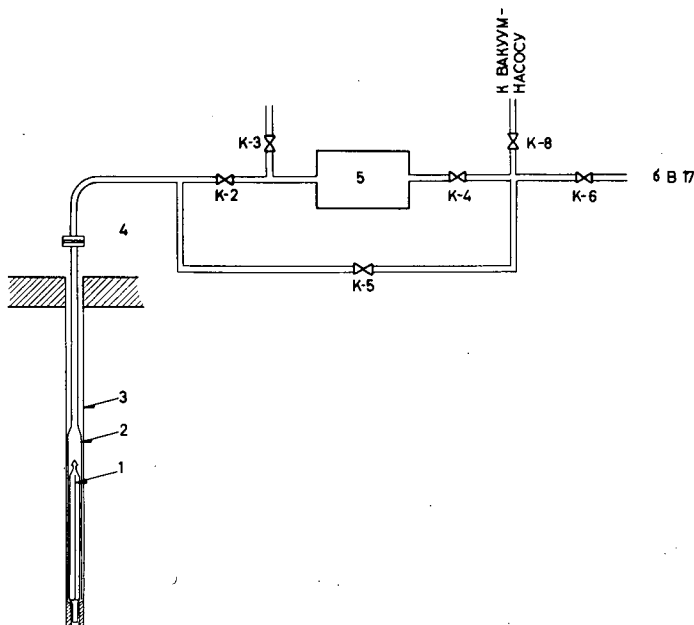


Рис. 4

Схема установки для исследования герметичности тепловыделяющих элементов:

- 1 - пакет; 2 - специальный чехол; 3 - чехол хранилища;
4 - фильтр; 5 - ионизационная камера.

Ионизационный ток камеры измерялся с помощью усилителя постоянного тока типа "Кактус". Ионизационная камера вакуумировалась до разряжения - 500 мм рт.ст., после чего открывался вентиль, соединяющий чехол, установленный на пакете, с камерой. Газ из пакета засасывался в камеру. По достижении в камере давления 1 ата производилось измерение ионизационного тока.

По этой методике были проведены измерения активности газа всех рабочих пакетов. Кроме того, испытывались для сравнения пакеты с заведомо целыми тепловыделяющими элементами.

Из 81 плутониевого пакета, подвергавшегося испытаниям, 8 пакетов были предварительно отмыты паром, 73 пакета отмывке не подвергались.

Неотмытые пакеты по результатам испытаний разделяются на две категории. К первой категории относятся 63 плутониевых пакета, относительная активность газа которых составляет $2 + 7$ мкр/сек. Следует отметить, что активность газа заведомо целых неотмытых пакетов также равнялась $2 + 7$ мкр/сек; следовательно, можно сделать вывод, что активность газа этих пакетов обусловлена радиоактивными примесями натрия, оставшегося внутри и снаружи неотмытых пакетов.

10 неотмытых плутониевых пакетов дали активность газа в тысячу раз превышающую активность газа первой группы пакетов ($1000 + 9000$ мкр/сек). Многократные повторные измерения подтвердили аномально высокую активность газа в этих пакетах. Длительное (в течение 5 суток) измерение величины активности газа, отобранного в ионизационную камеру из одного из этих пакетов, показало примерное постоянство величины активности со временем. Измерение активности газа из восьми плутониевых пакетов, подвергавшихся паровой отмывке, показали, что величина активности газа из этих пакетов лежит в интервале от 10 до 90 мкр/сек, причем активность газа, отобранного в ионизационную камеру, спадает со временем.

По результатам исследования герметичности оболочек тепловыделяющих элементов всех рабочих пакетов были отобраны два пакета для проведения тщательных исследований состояния ТВЭЛ в горячей лаборатории: один из отмытых паром пакетов, давший наибольшую α -, β -, γ -активность газа при исследовании герметичности, и в то же время этот пакет был одним из четырех пакетов, давших значительное повышение γ -активности в системе паровой отмывки при паровой обработке; и один из пакетов, не подвергавшихся паровой очистке и давший наибольшую β -, γ -активность газа при исследовании герметичности.

Максимальное выгорание плутония в обоих пакетах составляло 4,9%; интегральная доза облучения пакетов составляла $2,1 \cdot 10^{22}$ н/см².

В результате исследования этих пакетов в горячей лаборатории было установлено, что часть элементов в обоих пакетах оказалась с разрушенной оболочкой. На их поверхности просматривались продольные трещины. Характер трещин указывает на то, что они связаны с набуханием топливного материала при глубоком выгорании. Трещины появились в небольшой части элементов; по-видимому, где были минимальные зазоры между брикетами PuO_2 и оболочкой.

В пакете, прошедшем предварительную обработку паром, нарушения более выражены, чем в пакете без такой обработки.

Также было проведено исследование одного из центральных пакетов, показавшего незначительную активность газа при исследовании. Выгорание в центре пакета достигло 5,0%, интегральная доза облучения $2,2 \cdot 10^{22}$ н/см².

В результате исследования тепловыделяющих элементов в горячей лаборатории установлено, что все элементы пакета герме-

тичны, оболочка элементов имеет хорошее состояние. Поверхность элементов светлая, блестящая; отсутствуют какие-либо видимые дефекты или следы взаимодействия материала оболочки с теплоносителем.

Проведенные исследования в горячей лаборатории трех топливных пакетов с элементами, имеющими максимальное выгорание, и исследование герметичности оболочек рабочих пакетов тепловыделяющих элементов, проведенное на установке БР-5, позволили сделать заключение об удовлетворительном состоянии основной части рабочих пакетов.

Достигнутое высокое выгорание в тепловыделяющих элементах установки БР-5 при напряженной трехлетней эксплуатации установок с параметрами, близкими к параметрам энергетических установок на быстрых нейтронах, свидетельствует о хорошей стойкости окисных элементов, целесообразности применения их в подобных аппаратах и реальной возможности достижения в PuO_2 5% выгорания.

ДЕЗАКТИВАЦИЯ I КОНТУРА

После разгрузки активной зоны для проверки целостности тепловыделяющих элементов натрия из I контура был сдренирован в сливной бак.

Дренаживание натрия из I контура в сливной бак практически не снизило мощности дозы γ -излучения в боксах размещения оборудования. Этот факт объяснялся тем, что долгоживущие радиоактивные элементы, попавшие в контур вследствие разуплотнения тепловыделяющих элементов, в основном высадились на внутренних поверхностях оборудования и трубопроводов.

Анализ пробы натрия, взятого из сливного бака после дренажирования теплоносителя из контура, показал наличие в нем цезия-137 и натрия-22.

Перед проведением дезактивационных работ контур обрабатывался паром для очистки внутренних поверхностей оборудования и трубопроводов от остатков натрия, оставшихся после дренажирования.

Пар подводился к верхним точкам контура: в газовые линии баков циркуляционных насосов и трубы активной зоны. В нижней точке контур подсоединялся к конденсатору для конденсации пара перед сбросом в сборники промывочных вод.

В целях избежания конденсации пара на холодных поверхностях оборудования и трубопроводов контур прогревался до температуры $t = 150^\circ\text{C}$ газовым и электрическим обогревом.

Для того, чтобы поток пара проходил по всем коммуникациям контура, обработка его осуществлялась отдельно по ниткам. Пар (давление $P = 1,3 \div 1,5$ ати, температура $\sim 130^\circ\text{C}$) проходил по коммуникациям контура, взаимодействуя с остатками натрия, конденсировался и сбрасывался в сборники промывочных вод. Для избежания повреждения обмоток электродвигателей циркуляционных насосов пар подавался непосредственно в баки насосов, а в кожухах электродвигателей поддерживалось давление азота на $0,1 \div 0,2$ ати выше давления пара.

Несмотря на большую разветвленность коммуникаций контура, наличие байпасных линий и тупиковых участков, никаких хлопков или ударов при подаче пара в контур не наблюдалось. В общей сложности через обе нитки I контура и трубу активной зоны было пропущено ~ 10 т пара.

В результате паровой обработки I контура мощность дозы γ -излучения в боксах размещения оборудования снизилась примерно вдвое, а в помещении расположения дренажного коллектора увеличилась более чем в 10 раз.

При выяснении причин повышения активности в помещении дренажного коллектора в одном из тупиковых участков была обнаружена "пробка" из тестообразной массы, образовавшаяся из продуктов взаимодействия пара с остатками натрия.

Радиометрический анализ пробы, взятой из этой "пробки", показал наличие в ней цезия-137 и циркония-95, причем 90% активности пробы определялось цирконием-95 и 10% цезием-137. Таким образом, паровая обработка I контура позволила удалить с внутренних поверхностей оборудования и трубопроводов вместе с остатками натрия часть цезия-137 и циркония-95.

Для определения полноты реакции остатков Na с паром I контур перед дезактивацией был заполнен холодным дистиллятом. Никакой бурной реакции, хлопков или разогрева участков контура не отмечено. После выдержки в течение одних суток дистиллят из контура был сдренирован. В дистилляте были найдены следующие радиоактивные примеси: цезий-137, цирконий-95, железо-59, кобальт-58 и кобальт-60. Какого-либо заметного снижения активности в боксах I контура не наблюдалось. К сожалению, создать циркуляции дистиллята на контуре было невозможно из-за неисправности циркуляционных насосов. Вторичное заполнение контура дистиллятом и выдержка его в течение ~ 6 суток также не дали заметного снижения активности. Это говорило о том, что основная активность контура определялась радиоактивными изотопами, хорошо удерживающимися на стенках трубопроводов I контура и слабо реагирующими с водой и паром.

Поэтому было принято решение о проведении промывки I контура азотной кислотой. Однако в связи с наличием в центральной трубе реактора никелевой корзинки пакетов ТВЭЛ, промывке кислотой были подвергнуты только отдельные нитки I контура. Центральная труба при последующей промывке была отсечена от контура арматурой и дезактивации не подвергалась.

Одна из ниток I контура была заполнена азотной кислотой 5%-ной концентрации, нагретой до $t = 70^\circ\text{C}$.

Сброс кислоты после 6 часовой выдержки дал снижение активности в боксах в 1,5 раза. Последующие еще две кислотные промывки той же нитки I контура с выдержками соответственно 12 и 16 часов раствора в контуре позволили снизить активность еще в два раза.

В дальнейшем дезактивация производилась в следующей последовательности:

1. Заполнение нитки контура 0,5%-ным раствором KMnO_4 , выдержка в течение 24 часов при $t = 70^\circ\text{C}$.

2. Заполнение нитки контура смесью 5%-го раствора азотной и 1%-го раствора шавелевой кислоты. Выдержка в течение 3 + 4 часов при $t = 70^\circ\text{C}$.

3. Заполнение контура дистиллятом. Выдержка в течение 1 часа при $t = 60 + 80^\circ\text{C}$.

Обе нитки I контура в отдельности были подвергнуты пятью таким циклам промывки. В результате произошло существенное снижение активности в боксах I контура (до ~10 мкр/сек), позволившее начать проведение ремонтных работ. Радиохимический анализ сдrenированных из I контура растворов показал, что активность растворов обязана в основном цирконию-95 и цезию-137.

Таким образом, было выяснено, что остаточная активность оборудования и трубопроводов I контура после слива натрия определяется твердыми осколками деления Pu, Zr и Cs. Это впоследствии подтвердили и радиохимические анализы внутренней поверхности участков труб, вырезанных из I контура, как не отмывавшихся, так и подвергнутых отмывке. Причем в последних содержание цезия было уже незначительным.

Заключительным этапом дезактивации контура была дистиллятная промывка для удаления остатков кислоты.

Сливной бак I контура после передавливания из него Na в специальную герметичную емкость (бак-могильник) был подвергнут дезактивации в том же режиме, как и контур.

Сушка I контура после промывки производилась вакуумированием с одновременным разогревом оборудования и трубопроводов до $t \approx 150^\circ\text{C}$, штатным электро- и газовым обогревом.

Сушка контура продолжалась до прекращения выделения влаги в холодной ловушке паров, установленной на всасе вакуумного насоса. После этого I контур был вновь заполнен свежим продистиллированным натрием.

Поскольку высаживание значительной долгоживущей радиоактивности на стенках оборудования и трубопроводов I контура сильно затрудняет доступ к оборудованию контура, а следовательно и эксплуатацию реактора, необходимо более тщательное изучение этого явления.

В настоящее время на I контуре реактора устроена специальная петля для изучения с помощью γ -спектрометра механизма активации контура, уточнения радиоактивных изотопов, высаживающихся на стенках оборудования и трубопроводов, а также способов очистки I контура от радиоактивности.

Как указывалось выше, проведенная кислотная дезактивация контура позволила провести необходимые ремонтные работы и подготовить I контур к пуску реактора. Массовая проверка герметичности ТВЭЛ рабочих пакетов и исследование состояния выборочных пакетов ТВЭЛ в горячей лаборатории позволили сделать заключение об удовлетворительном состоянии основной части плутониевых пакетов. Это послужило основанием для использования 80% работавших в течение трех лет плутониевых пакетов ТВЭЛ для дальнейшей работы

в реакторе. 29 марта 1962 года реактор БР-5 вновь был выведен на мощность.

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DISCUSSION

W.H. JENS: Can you describe the fuel-element failures in more detail? In particular, I should like to know whether the clad was ductile, and whether you are now confident that the 80% of fuel put back in the reactor will not be subject to further failures.

N.N. ARISTARKHOV (on behalf of A.I. Leipunsky et al.): In the paper it is pointed out that studies performed in a hot laboratory revealed the presence of longitudinal cracks in a number of fuel elements; in the bundles that were subjected to steam treatment these defects were clearly apparent. An analysis of the results of these tests suggests that the most likely cause of damage to the cladding was internal stress due to swelling of the fuel. The clearance between fuel and cladding was gradually reduced during burn-up, and our investigations showed that the failed elements were those which had had the smallest clearances.

Preliminary investigations have shown that the ductility of the stainless-steel cladding was not badly affected. In fact, bending tests gave results very similar to those obtained before the material was irradiated. The fuel remained in the solid state, and no effects such as internal melting or fusing were observed.

The methods used in these studies on BR-5 did not enable us to determine whether the elements loaded back into the reactor were absolutely tight. In fact, there appear to be some leaks: we have observed the presence of certain gaseous fission products (krypton and xenon) in the gas cavity of the reactor vessel, which suggests that some of the elements are not tight. However, the leaks have not led to any noticeable build-up of solid fission-fragment activity in the coolant circuit.

S. YIFTAH: In view of the operating experience acquired with the BR-5 reactor and the BN-50 and BN-250 50- and 250-MW(e) design studies reported at the Second Geneva Conference*, is the Soviet Union now contemplating the actual construction of a fast prototype power reactor fuelled with plutonium?

N.N. ARISTARKHOV: Various fast reactor designs are being considered in the Soviet Union, including a plant with an output of 1000 MW(e).

* Proc. 2nd UN Int. Conf. PUAE 8 (1948) 580.

As to the BN-50 design which was reported at the Geneva Conference, so far as I know it is not now planned to carry out this project.

C.A. PURSEL: I have two questions. Firstly, do the figures which you quoted for burn-up refer to peak or average burn-ups? Secondly, I understand that your pumps ran some 8000 hours without maintenance, which seems rather good. Are you doing any work on improved pumps, such as high-speed pumps or electromagnetic pumps?

N.N. ARISTARKHOV: The figures I quoted refer to the maximum burn-up attained in the reactor; the figure of 5.4% reflects the burn-up attained in the 80% of fuel elements that were subsequently loaded back into the reactor.

The operation of the pump in BR-5 has been limited, generally speaking, by the lifetime of the bearings, which so far has averaged about 8000 h. The bearings must then be replaced. The pump has undergone a number of structural changes designed to improve its operation in BR-5, such as elimination of gas transfer, improvements in the cooling of the upper bearing, and so forth. Electromagnetic pumps have been installed for use in the central experimental loop assembly, but their capacity is not particularly large.

J. LEDUC: You say that the PuO_2 fuel was studied in a hot laboratory. Were the same characteristics observed in the PuO_2 fuel after irradiation as have been noticed in the UO_2 , i.e. the formation of a central hole, a zone of grain enlargement, etc.? Was there any change in the clearance between fuel and cladding? Lastly, since a combination of uranium oxide and plutonium oxide is one of the possible fuels for the large reactors of the future, have you irradiated this type of mixture in BR-5?

N.N. ARISTARKHOV: In addition to the PuO_2 fuel elements, UO_2 elements of different enrichments (up to 90%) are being tested in the BR-5 reactor. These have not attained a sufficient burn-up, however, and for that reason we have not yet tested them. Not being a metallurgist, I am unfortunately not able to answer your question about alterations of grain structure.

W.H. JENS: In earlier papers on BR-5 you mentioned tests designed to determine the stability of the reactor and the possibility of a positive temperature coefficient. You stated that there was no prompt positive coefficient. Could you please say how the coefficient was measured — whether an oscillator rod was used — and what the shortest time constant of the coefficient was that could be measured in these tests?

N.N. ARISTARKHOV: The power characteristics of BR-5 were tested early in the reactor's history by means of transient power changes worked by an automatic regulator which governs the reactor's power level. These measurements were sufficiently accurate to allow the conclusion that there are no appreciable changes in the reactivity of the system associated with abrupt changes of power. A second series of power studies was begun early in 1963. Using essentially the same method, these investigations are designed to test the characteristics of the reactor at different power levels and different rates of coolant flow through the core. We hope then to be able to distinguish effects associated with the coolant from those associated with the fuel. The results are now being processed and will be published.

III

EXPERIENCE WITH SPECIFIC NUCLEAR POWER PLANTS (continued)

SHIPPINGPORT ATOMIC POWER STATION OPERATING EXPERIENCE, DEVELOPMENTS AND FUTURE PLANS

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Abstract — Résumé — Аннотация — Resumen

SHIPPINGPORT ATOMIC POWER STATION, OPERATING EXPERIENCE, DEVELOPMENTS AND FUTURE PLANS. This paper describes and evaluates five years of operation and test of the Shippingport Atomic Power Station and discusses the current technical developments and future plans of the Shippingport programme. This programme is directed towards development of the basic technology of light-water reactors to provide the basis for potential reduction in the costs of nuclear power.

The Shippingport reactor plant has operated for over five years and has been found to integrate readily into a utility system either as a base load or peak load unit. Plant component performance has been reliable. There have been no problems in contamination or waste disposal. Access to primary coolant components for maintenance has been good, demonstrating the integrity of fuel elements. Each of the three refuelling operations performed since start-up of Shippingport has required successively less time to accomplish. Recently, the third seed was refuelled in 32 working days, about one quarter the time required for the first refuelling. The formal requirements of personnel training, written administrative procedures, power plant manuals, etc., which have been a vital factor in the successful implementation of the Shippingport programme, are described.

The results obtained from the comprehensive test programme carried out at Shippingport are compared with calculations, and good agreement has been obtained. Reactor core performance, plant stability, and response to load changes, fuel element and control rod performance, long-term effects such as corrosion and radiation level build-up, component performance, etc., are discussed in this paper.

The principal objective of the current and future programmes of the Shippingport Project in advancing the basic technology of water-cooled reactors is discussed. This programme includes the continued operation of the Shippingport plant, and the development, design, manufacture and test operation of a long-life, high-power density second core - Core 2. At its design rating of 150 MW(e) gross and a first seed life of 10 000 effective full power hours (EFPH), Core 2 will have five and a half times the design energy output and twice the power density of Core 1. The Core 2 design is described and associated major developments in reactor physics, metallurgy, heat transfer and fluid flow, and fuel element manufacture, are summarized. Plans for decontamination of the reactor plant and for performing modifications to the plant in connection with the installation of the higher rated Core 2 are described.

LA CENTRALE NUCLÉAIRE DE SHIPPINGPORT, EXPÉRIENCE DE SON FONCTIONNEMENT ET PLANS POUR L'AVENIR. Les auteurs font le bilan des cinq années de fonctionnement et d'essai de la centrale nucléaire de Shippingport; ils examinent les progrès techniques actuels et l'avenir du programme de Shippingport. Ce programme vise à mettre au point la technologie fondamentale des réacteurs à eau légère pour permettre une réduction du prix de revient de l'énergie d'origine nucléaire.

La centrale nucléaire de Shippingport fonctionne depuis plus de 5 ans et s'est intégrée facilement dans un réseau d'interconnexion comme centrale de base ou comme centrale d'appoint. Les divers éléments de la

centrale se sont révélés d'un fonctionnement sûr. Il n'y a pas eu de problème de contamination ni d'élimination des déchets. On peut accéder facilement aux éléments du circuit primaire de refroidissement pour l'entretien et, ce faisant, on a constaté que les cartouches de combustible étaient intactes. Chacune des trois opérations de rechargement effectuées depuis le démarrage de la centrale a pris moins de temps que la précédente. Récemment, la troisième charge de combustible enrichi a été mise en place en 32 journées de travail, soit environ le quart du temps nécessaire pour le premier rechargement. Les auteurs exposent les conditions auxquelles doit satisfaire la formation du personnel et donnent un aperçu des instructions écrites en matière d'administration, d'exploitation, etc., qui sont autant de facteurs essentiels du succès du programme de Shippingport.

Les résultats fournis par la série complète des essais réalisés à Shippingport sont comparés avec les calculs; la concordance est satisfaisante. Les auteurs examinent aussi les performances du cœur du réacteur, la stabilité de l'installation et sa réponse aux variations de charge, les performances des cartouches de combustible et des barres de contrôle, les effets à long terme tels que la corrosion et la radioexposition, le fonctionnement des divers éléments constitutifs, etc.

Les auteurs examinent le principal objectif des programmes actuel et futur de Shippingport pour le développement de la technique fondamentale des réacteurs refroidis à l'eau. On prévoit notamment de continuer l'exploitation de la centrale et d'étudier, de fabriquer et d'essayer un deuxième cœur de réacteur à longue durée de vie et de grande puissance spécifique. Avec une puissance nominale brute de 150 MWe et une durée de vie équivalant à 10 000 h de marche à pleine puissance pour la première charge de combustible enrichi, le cœur N° 2 aura cinq fois et demie la production d'énergie prévue pour le cœur N° 1 et deux fois sa puissance spécifique. Les auteurs décrivent le cœur N° 2 et indiquent brièvement les principaux progrès qui ont été réalisés à cet égard dans les domaines de la physique des réacteurs, de la métallurgie, du transfert de chaleur, de la circulation du fluide caloporteur et de la fabrication des cartouches de combustible. Ils exposent les plans de décontamination de la centrale nucléaire et les modifications qu'entraînera l'installation du cœur N° 2 plus puissant.

ШИППИНГПОРТСКАЯ АТОМНАЯ ЭЛЕКТРОСТАНЦИЯ, ОПЫТ ЭКСПЛУАТАЦИИ, УСОВЕРШЕНСТВОВАНИЯ И ПЛАНЫ НА БУДУЩЕЕ. Дается оценка пяти годам эксплуатации и испытания Шиппингпортской электростанции, а также рассматриваются последние технические усовершенствования и программа на будущее. Эта программа направлена на усовершенствование основной технологии реакторов на обычной воде с тем, чтобы создать основу для уменьшения в будущем стоимости ядерной электроэнергии.

Шиппингпортская реакторная установка, эксплуатировавшаяся пять лет, признана годной для легкого подключения к системе энергоснабжения в качестве установки для базовой или пиковой нагрузки. Работа компонентов установки была надежной. Не возникало проблем, связанных с загрязнением или удалением отходов. Доступ к компонентам первичной системы охлаждения для их обслуживания был хорошим, что показывает целостность топливных элементов. Каждая из трех операций по замене топлива в реакторе с момента его пуска требовала все меньше и меньше времени. Недавно была установлена третья запальная сборка, на что потребовалось 32 рабочих дня, т.е. около одной четверти времени, потраченного на первую замену топлива. Описаны официальные требования в отношении подготовки персонала, письменных административных инструкций, руководств по энергетической установке и т.д., что явилось очень важным фактором в успешном осуществлении программы испытаний Шиппингпортской электростанции.

Результаты испытаний хорошо совпадают с расчетными данными. Рассматриваются работа активной зоны реактора, стабильность установки, чувствительность к изменениям нагрузки, работа топливных элементов и регулирующих стержней, повышение таких длительных эффектов, как коррозия и интенсивность излучения, работа компонентов и т.д.

Рассматривается основная цель текущей и будущей программ Шиппингпортского проекта в отношении усовершенствования основной технологии реакторов с водяным охлаждением. Эта программа включает продолжение эксплуатации Шиппингпортской установки, разработку, конструирование, изготовление и опытную эксплуатацию второй активной зоны (активная зона 2), имеющей длительный срок службы и высокую плотность энерговыделения. Имея проектную электрическую мощность брутто 150 мвт и продолжительность службы первой запальной сборки 10 тыс. часов работы на полной мощности, активная зона 2 будет в пять с половиной раз мощнее активной зоны 1 и обладать в два раза большей плотностью энерговыделения, чем последняя. Дается описание конструкции активной зоны 2 и связанных с ней основных усовершенствований в физике реактора, металлургии, теплопередаче, подаче теплоносителя и изготовлении топливных элементов. Изложены планы в отношении дезактивации реакторной установки и осуществления ее модификации в связи с установкой более мощной активной зоны 2.

CENTRAL NUCLEOELÉCTRICA DE SHIPPINGPORT: EXPERIENCIA ADQUIRIDA CON SU EXPLOTACIÓN Y PROGRAMA DE DESARROLLO. Los autores exponen y analizan los resultados obtenidos en cinco años de ensayo y explotación de la central nucleoelectrónica de Shippingport y examinan el programa de actividades en curso y los planes para el futuro. Con ese programa se proyecta perfeccionar la tecnología básica de los reactores de agua ligera, a fin de contribuir a la reducción de los costos de la energía nuclear.

La central nuclear de Shippingport viene funcionando desde hace más de 5 años y se ha comprobado que su incorporación a una red de distribución, sea como central para carga básica o como central para carga de cresta no plantea mayores dificultades. Los distintos componentes de la central se han comportado en forma satisfactoria. No ha habido problemas de contaminación ni de evacuación de desechos. Ha sido fácil el acceso a las partes del circuito primario de refrigeración, y se verificó el buen estado de los elementos combustibles. Cada una de las tres operaciones de reabastecimiento efectuadas desde la iniciación de la marcha exigió menos tiempo que la precedente. La tercera carga de material fértil se realizó recientemente en 32 días de trabajo, o sea menos de un cuarto del tiempo que exigió la primera. Detallan los autores las normas impuestas para la formación de personal, las instrucciones escritas en materia de procedimientos administrativos, los manuales de trabajo de la central, etc. que han contribuido decisivamente al éxito de la ejecución del programa de Shippingport.

Se comparan en la memoria los resultados del extenso programa de ensayos aplicado en Shippingport con los cálculos teóricos y se comprueba que concuerdan satisfactoriamente. Se examinan también el comportamiento del cuerpo del reactor, la estabilidad de la central y su respuesta a las fluctuaciones de la carga, el comportamiento de los elementos combustibles y de las barras de control, los efectos a largo plazo (corrosión, aumento de la intensidad de radiación), el comportamiento de las diversas partes de la central, etc.

Se examina en la memoria el objetivo principal del programa actual y futuro de Shippingport, para el desarrollo de la tecnología básica de los reactores refrigerados con agua. Dicho programa comprende la explotación de la central de Shippingport en forma continua, y el proyecto, diseño, fabricación y explotación experimental de un nuevo cuerpo de reactor de prolongada duración y elevada densidad de potencia, el denominado cuerpo N° 2. Con una potencia nominal bruta de 150 MW(e) y una duración equivalente a 10 000 h de funcionamiento a plena potencia, el cuerpo N° 2 dará una producción de energía 5,5 mayor que la del cuerpo N° 1 y su potencia específica será el doble de la de este último. Se describen las características de diseño del cuerpo N° 2 y se resumen los principales adelantos en materia de física de reactores, metalurgia, transmisión de calor, circulación de fluidos, y elaboración de elementos combustibles. Por último, se describen los planes para la descontaminación de la central nuclear y para introducir en la misma las modificaciones exigidas por la instalación del cuerpo N° 2, de mayor potencia.

1. INTRODUCTION AND SUMMARY

The purpose of this paper is to describe and evaluate the experience obtained during five years of operation and testing of the Shippingport Atomic Power Station and to discuss some of the more important current technical developments and future plans of the United States Atomic Energy Commission's (USAEC) Shippingport programme.

The Shippingport Project was authorized in July 1953 and actual plant construction started in March 1955 (Table I). The plant achieved initial criticality on 2 December 1957; the generator was synchronized on the Duquesne Light Company transmission system on 18 December 1957 and was operating at a full power of 60 MW(e) on 23 December 1957, accumulating 100 effective full power hours (EFPH) by the end of the year. The reactor portion of the Shippingport Plant was designed and developed by the Bettis Atomic Power Laboratory under the direction of, and in technical co-operation with, the Naval Reactors Group of the Commission's Division of Reactor Development. The Bettis Laboratory is operated for the Commission by the Westinghouse Electric Corporation. The Duquesne Light Company, which owns and operates the turbine-generator facilities of the plant, assumed \$5 million of the cost of the reactor portion of the

TABLE I
SHIPPINGPORT HIGHLIGHTS

Project history				
Date project was authorized	July 1953			
Date ground was broken	September 1954			
Date construction started	March 1955			
	Seed 1	Seed 2	Seed 3	Seed 4
Operation with Core 1	2-12-1957	12- 4-1960	7-10-1961	11- 1-1963
Date of initial criticality	23-12-1957	7- 5-1960	24-10-1961	30- 1-1963
Date plant reached full power (60 000 kW net)				
Lifetime (EFPH)	5806	7900	7329	6500*
Total electricity generated (million kWh gross)	388.5	514.3	475	86.2* #
Average load factor (%)	37	70	77	-
Average load factor, excluding testing and training (%)	75	97	97	-
Date refuelling started	2-11-1959	16- 8-1961	26-11-1962	-
Date refuelling was completed	11- 4-1960	6-10-1961	4- 1-1963	-
Core loading data				
Core 1: Seed - Uranium	75 kg U ²³⁵	90 kg U ²³⁵	90 kg U ²³⁵	90 kg U ²³⁵
Boron	0	170 g natural B	170 g natural B	170 g natural B
(Blanket - 34.2 short tons natural U)				
Core 2: Seed - Uranium	336 kg U ²³⁵	-	-	-
Boron	465 g B-10			
(Blanket - 18.7 short tons natural U)				
Total electricity generated and fuel burn-up data as of 31 March 1963				
Total electricity generated on Core 1 (gross output)				1464 million kWh
Approximate amount of electricity delivered to DLC system (net output)				1225 million kWh
Total operating time on Core 1				22 366 EFPH
Peak burn-up in Core-1 blanket fuel (natural UO ₂)				30 100 Mwd/t
Average burn-up in Core-1 blanket fuel				9000 Mwd/t

* Estimated
** As of 31 March 1963.

plant and is under contract to the Commission to supervise, operate, and maintain the entire plant and to purchase the steam produced by operation of the reactor. The Shippingport Project therefore represents in the United States a joint endeavour of the Government, a public utility company and an industrial concern.

From the very outset the Shippingport Project has been directed towards advancing the basic technology of light-water-cooled reactors through the design, development, building, testing and operation of a large power reactor as part of a public utility system. The mission of the project has been, and continues to be, to develop and advance the underlying technology so as to provide a sound base for potential reductions in nuclear power costs, rather than to reduce costs by merely optimizing existing technology. To achieve this most important purpose, the reactor plant and its cores were designed for flexibility of operation and extra instrumentation was incorporated to provide information of an experimental and developmental nature. Details of the design and construction of the plant can be found in papers presented at the Geneva Conferences of 1955 and 1958 and in the book entitled "Shippingport Pressurized Water Reactor" [1].

Information on the Shippingport Project is broadly disseminated to the nuclear industry as quickly as possible by means of unclassified periodic and topical reports and special technical interim reports. To date more than 1650 such reports have been issued [1a].

During its five years of operation the Shippingport Station has been readily integrated into a large utility company's electrical power system. The reactor has performed well and has been proved capable of continuous operation at rated power for extended periods. It has responded to load transients with complete safety and without any operating problems. This operating experience clearly shows that the plant is suitable for operation either as a base-load or a peak-load station.

The major plant components have performed well during this period, which has included three refuellings of the seed portion of the core. However, three main coolant pumps were replaced, modifications, including the plugging of some tubes, were made to the steam generators, and feed-water heaters were retubed. The turbine-generator governor valves and the moisture separator were redesigned. There were no significant problems in contamination or waste disposal. From a radioactivity standpoint, access to primary coolant components for maintenance work was good, indicating the advantage of high fuel-element integrity and of the lithium hydroxide water chemistry that has been used in the primary coolant from the very start of plant operation. Much was learned from the refuelling of the first seed [2] of this highly instrumented core, with the result that the refuelling of the second seed was performed in 44 working days - one-third the time that was required for the first refuelling; 32 working days were required for the third refuelling. These and other operations at Shippingport have emphasized the importance and necessity of thoroughly trained station personnel and the preparation of detailed operating and maintenance procedures. For example, 94 individual refuelling procedures, approved by the Commission, were used by Duquesne for the refuelling of Seed 3 with Seed 4.

Duquesne operating personnel are given classroom and on-the-job training by the company. These personnel must pass comprehensive written and oral examinations given by the Commission before they can be assigned supervisory or operating responsibilities at Shippingport. The detailed written procedures for operating the station are contained in the Shippingport Atomic Power Station Manual, which consists of four volumes containing over 100 chapters. This manual was prepared by Duquesne and the Bettis Laboratory and was approved by the Commission before station start-up. The manual is kept up to date, and all changes affecting the reactor plant or radioactive-waste-disposal system must be approved by the Commission. Further, in order to assure that station personnel will also perform effectively under emergency conditions, various emergency situations are periodically simulated by means of practice drills in which all personnel perform their prescribed duties.

The first seed of the Shippingport Core 1, containing 75 kg of 93% enriched uranium, was depleted in October 1959 after 5806 EFPH of operation, generating 338.5 million kWh gross. The second seed, containing 90 kg of 93% enriched uranium and 170 g of natural boron, was depleted in August 1961 after 7900 EFPH, generating 514.3 million kWh gross. The third seed, which had the same loading as the second, was depleted in November 1962 after 7329 EFPH of operation, generating 475 million kWh gross. Thus, at the start of operation with the fourth seed in January 1963, the core had generated a total of about 1400 million kWh gross, and the burn-up in the natural uranium, Zircaloy-2 clad blanket fuel elements had reached an average of 8450 MWd/t of uranium and a peak of 28 600 MWd/t. The total output at Shippingport since the start of operation would be sufficient to supply electric service to about 450 000 homes in Pittsburgh for one year.

Inspection of representative natural-uranium blanket elements after operation with two seeds showed them to be in excellent condition, i.e. there was no measurable distortion or swelling, very little crud build-up (average of 0.1 mil), and less than 80 ppm of hydrogen pick-up by the Zircaloy cladding. The examination of blanket elements removed after operation with three seeds is not yet finished; however, no distortion or swelling has been observed. A new core, which is discussed later in this paper, will be installed upon the depletion of the fourth seed in the early part of 1964. It is of interest to note that the first (Core 1) Shippingport core design specifications called for a total blanket life of 8000 EFPH and so far the blanket has operated successfully for over 21 000 EFPH.

The seed and blanket core concept used at Shippingport demonstrates:

- (1) That it is feasible in a light-water system to obtain about 50% of total core power from a blanket of natural uranium by using a seed, i.e. a small, highly-enriched core, as the driving element;
- (2) That the inventory of enriched fuel required can be significantly reduced;
- (3) That it is possible to control the entire reactor core by using few control rods located in the relatively small seed volume; and
- (4) That a seed and blanket core possesses a suitable negative temperature coefficient and a favourable dynamic response. In this latter connection, the presence of extensive instrumentation in the core enables detection and correction of any xenon-induced instabilities. These oscillations were easily controlled by the reactor oper-

ator by slight adjustment of the control rods, but it was most important that the operator be made aware of the oscillations so that the thermal capabilities were not exceeded in any local region of the core.

An extensive and comprehensive test programme has been followed as part of plant operations from the very start. Even though the plant operates most of the time as part of a utility company network, operations are determined largely by the needs of the test programme. For example, during Seed-2 and Seed-3 operation, over 270 tests were performed. This test programme has produced, and is producing, a wealth of data which can be used in future designs of nuclear power plants employing light-water-cooled reactors. Reactor physics calculational models and heat transfer and hydraulic calculations are verified; plant stability and response to load changes are demonstrated; fuel element and control-rod performance are observed; component reliability is checked; and long-term effects, such as corrosion build-up and radiation-level build-up, are measured and reported.

The operation of Shippingport with its first core is continuing with three main objectives:

- (1) Continuing the large-scale irradiation of natural UO_2 fuel elements with Zircaloy-2 cladding to determine the maximum MWd/t of uranium burn-up that these fuel elements can withstand;
- (2) Obtaining information on the physics characteristics and thermal performance of seed and blanket cores as a function of fuel depletion and demonstrating the continued generation of a high fraction of the power from natural uranium in such cores;
- (3) Obtaining data on the long-term behaviour of major plant components and thus acquiring a realistic basis for judging the adequacy of component design specifications. Further, this operational experience should pilot the long-term performance to be expected from similar components in plants now under construction or in early stages of operation.

The operation and testing of the Shippingport reactor, together with developments undertaken in connection with Core 1, have shown ways to develop a higher-power, longer-life core for the Shippingport Station. The manufacture of such a core is now in progress, with completion scheduled for the latter part of 1963. The principal objective of this advanced core programme is the development of a very high performance core in terms of both power density and lifetime. This core, designated as Core 2, will also be a seed and blanket core. It will have a rating of 150 MW(e) gross in the Shippingport plant, about five and one-half times the design energy output and about twice the power density of Core 1. Such performance should lead to reductions in both fuel-cycle costs and plant-capital costs.

The fuel elements for Core 2 will consist of compartmented, flat plates containing enriched UO_2 - ZrO_2 wafers in the seed elements and natural UO_2 wafers in the blanket elements. Such plates have been extensively tested in-pile and out-of-pile and appear suitable for operation at high heat fluxes and to large burn-ups. The fuel-element cladding material will consist of an improved zirconium alloy, Zircaloy-4, which has less susceptibility to hydrogen pick-up than Zircaloy-2 and is, therefore, expected to be more suitable for long-life cores. The seed fuel elements also will in-

corporate self-shielded burnable poisons and zones of different uranium concentration to reduce power peaking.

In addition to its higher power rating, Core 2 incorporates many significant changes from Core 1, such as change in geometry of the seed, increased size but reduced number of control rods, increase in length from 6 to 8 ft, a closure head of new design which incorporates a single central refuelling port, addition of a support flange to the pressure vessel which permits side exit of instrumentation leads rather than through the closure, and modification of the coolant flow pattern to include two passes through the natural uranium blanket. Major development efforts have been undertaken for this advanced core in the areas of reactor physics, computer codes, fuel poison and cladding materials, thermal and mechanical design (including core instrumentation), power distribution, and heat transfer. An example of the advances being made is the successful test irradiation of UO_2 plate-type elements to burn-ups of 75 000 MWd/t of uranium at Core-2 operating conditions.

The accommodation of Core 2 in the Shippingport plant will require certain plant modifications, specifically in the steam drums, main coolant pumps, and reactor control equipment. Since the core is rated at 150 MW(e), whereas the station turbine-generator is rated at 100 MW(e), additional heat removal capacity in the form of a heat-dissipation system is being installed to handle the power produced in the core in excess of the generator rating. This combination of turbine-generator and heat dissipation system will meet all the requirements for testing the 150-MW(e) reactor plant and its core.

To sum up, the Shippingport Station has proved well suited to integrated operation on an electric utility system and is providing a large amount of operational data and experience on the reactor and plant systems and components. In keeping with its primary goal of developing and advancing the basic technology of water reactors, the Shippingport programme is now concentrating on core development to provide the basis for significant potential reductions in nuclear power costs.

2. STATION OPERATING EXPERIENCE

Fig.1 shows a view of the Shippingport Atomic Power Station. A schematic diagram indicating the basic components of the reactor and steam plant is shown in Fig.2. The principal elements of the reactor plant are the reactor vessel, which contains the nuclear core, and the four main coolant loops which circulate the reactor coolant water between the core and steam generators. Each of the four main coolant loops contains a single-stage, centrifugal, canned motor pump, a steam generator, four isolation valves, a check valve, and interconnecting 18-in piping. The heat absorbed by the reactor coolant in passing through the core is given up to the steam plant water on the secondary side of the steam generator. The 600-lb/in² abs. steam which is produced is delivered to an 1800-rpm turbine-generator with rated capability of 100 MW. The core is a seed and blanket design consisting of two types of fuel elements assembled into a right circular cylinder. The active core is about 6-ft high and 6 ft 9½ in in diameter. The highly-enriched uranium which forms the seed is contained in 1914 plates clad with

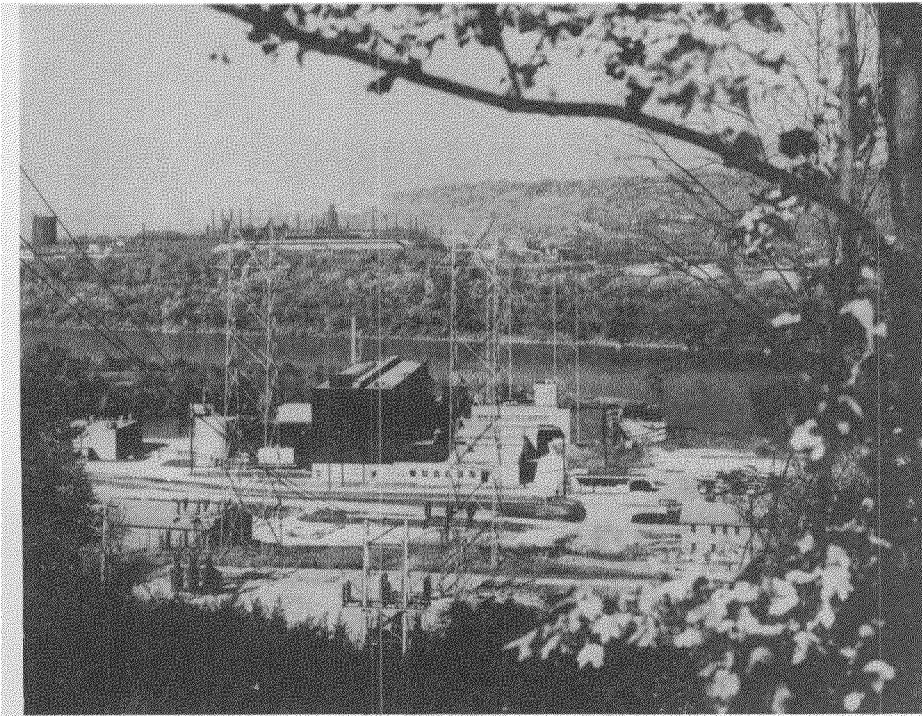


Fig. 1

View of Shippingport Atomic Power Station

Zircaloy-2 and the natural uranium oxide pellets which constitute the blanket are contained in 94 920 Zircaloy-2 tubes. There are 32 seed assemblies and 113 blanket assemblies. The core and its 32 control rods are housed in a reactor vessel which has an inside diameter of 109 in and an inside height of about 32 ft.

When the Shippingport programme was initiated, it was recognized that certain practical features of operation would be as important as competitive power costs in gaining wide acceptance of a nuclear station by the utility industry. Thus, the capability of a nuclear power station to be integrated into a major utility system, the station outage time because of refuelling and maintenance periods, the radiation hazard to personnel and environment and the station training and maintenance requirements are characteristics which are vital to any evaluation of a nuclear power station.

Over five years of operating experience at Shippingport, extending from the initial start-up in December 1957 to the beginning of Seed-4 power operation in January 1963, has demonstrated that none of these considerations will preclude the application of pressurized water reactors to central station power generation. Plant response to load transients are superior to modern conventional stations on the system. Continued long power runs, coupled with reduced refuelling outage time, have resulted in the achievement of a

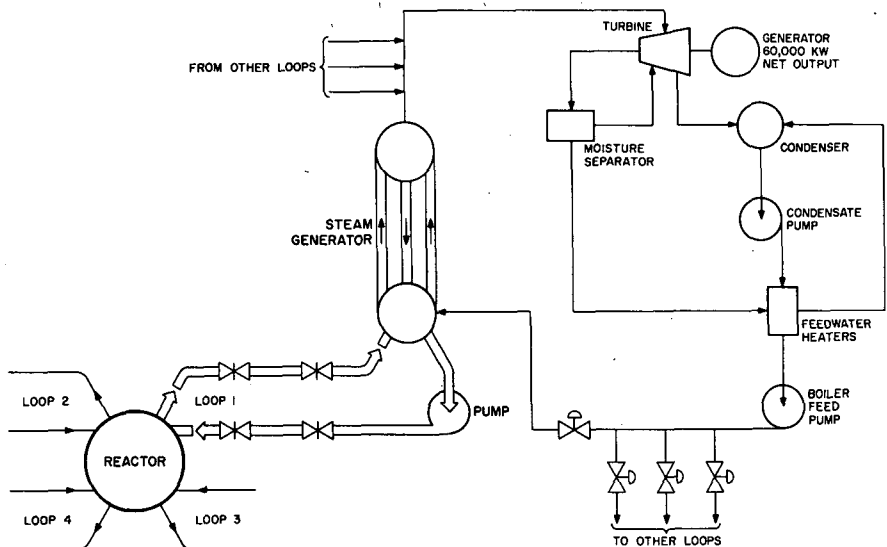


Fig. 2

Basic schematic diagram of the Shippingport Plant

high load factor, demonstrating that a high degree of availability can be achieved in this type of nuclear power station. The application of standard radiation control procedures has been more than adequate in maintaining effective radiation safeguards. Not a single contamination incident has occurred that would have presented even a minor hazard to the public and station personnel. It has been learned that, in comparison to a conventional station, efficient and safe operation of a nuclear power station requires more extensive concentration on detailed procedures and personnel training in operation and maintenance; however, the experience at Shippingport has shown that these requirements can be achieved.

Station-system compatibility

Design factors related to the use of the Shippingport Station as part of a utility system included: (1) the capability of operating as a base-loaded station for extended periods; (2) the capability of accepting its share of the load swings necessitated by system disturbance; and (3) the capability of operating on a peak-load basis when station-system efficiency conditions justified such use. The station was proved to be capable of performing all of these functions satisfactorily.

Throughout most of Core-1 life, the station has been operated as a base-load station. Two 1000-h, full-power runs were made during Seed 1; three 1300-h, full-power runs with Seed 2; and one 3000-h, full-power run and two 1500-h runs were completed during Seed-3 operation.

By operating the plant at rated power for long periods, Core 1 was subjected to rapid reactivity depletion and data were obtained to determine core power distribution, core stability and reactivity lifetime. During

these long power runs, the reactor was easily maintained at the desired power level by normal control-rod motion. During Seed-2 and Seed-3 operation, axial xenon oscillations occurred as a result of small changes in power production from the top to the bottom of the core when rod control was transferred from one group of rods to another; however, since these power oscillations were slow-moving (24-h period), small in amplitude and convergent, they did not affect the safety of the plant and did not significantly affect plant operability. Fig. 3 depicts the principal events and operating history of the Shippingport Station to date. It can be seen from the graph that much time was devoted to reactor plant and core tests during various stages of core life to further the understanding of water-cooled reactors. In addition, the plant was periodically utilized for training both Duquesne Light Company operating personnel and out-of-station trainees of the Nuclear Power Station Training Program conducted at Shippingport for the USAEC. Despite the large amount of low-power operation required by the testing and training programmes, the 514.3 million kWh gross generated during Seed-2 operation was equivalent to operating the plant at full power 70% of the time, and the 475 million kWh gross generated during Seed-3 operation corresponded to 77% full-power operation. If the time the plant was shut down for testing and training were excluded, it can be seen that the average load factor would be 97% for Seed-2 operation and 97% in the case of Seed 3.

The station could be operated as a base-load facility at full power for essentially the full reactivity lifetime of the core. 100% equilibrium xenon power was maintained for 5532 EFPH for Seed 1, 7528 EFPH for Seed 2, and 6935 EFPH for Seed 3. These figures represented approximately 95% of the core reactivity lifetime. The remaining 5% of reactivity lifetime was obtained as useful energy by operating at reduced power and decreasing the average reactor-coolant temperature 25° F.

Obviously operating reliability and continuity are essential if base-load operation is to be sustained. Fig. 4 indicates the progress made at Shippingport in this respect. The graph shows the total number of safety shut-downs (scrams) that occurred during the operation of Seed 1, Seed 2 and Seed 3 and a comparison of the shut-downs caused by personnel and those caused by other factors. The decrease in total number of shut-downs (from 23 to 14 to 9) during the operation of three seeds is indicative of increased operator familiarity with plant operation, as well as of equipment reliability.

To check the station's response to major system disturbances, system-load change patterns were simulated. These special tests demonstrated that the reactor plant met all of its design requirements. These requirements include step changes of +15 MW(e) and -12 MW(e); ramp changes of ± 15 MW(e) at 3 MW/s, and ± 20 MW(e) at 25 MW/min. Manual control-rod motion was used to maintain the magnitude of the plant temperature and pressure excursions during these load changes within operating limits.

At certain times throughout Seed-1, Seed-2 and Seed-3 life, the station was operated as a peak-load facility, as demanded by testing, training or by Duquesne Light Company system requirements. The station was required to follow system load-change requirements in the power range and was shut down and started up during off-peak hours at a relatively fast rate without difficulty. The ability of the station to follow the normal system load changes was demonstrated many times during the operating period. For a period

- LEGEND
- 1 REACTOR SHUTDOWN FOR TEST AND TRAINING.
 - 2 LEAK DISCOVERED IN STEAM GENERATOR.
 - 3 COOLANT LOOP PUMP FAILED.
 - 4 FIRST 1000-HR REACTIVITY LIFETIME TEST.
 - 5 SECOND 1000-HR REACTIVITY LIFETIME TEST.
 - 6 SECOND 1000-HR REACTIVITY LIFETIME TEST.
 - 7 SHUTDOWN FOR MAINTENANCE AND TESTING.
 - 8 MOISTURE SEPARATOR FAILURE DISCOVERED.
 - 9 PLANT REMAINED SHUTDOWN THROUGH JAN 1959.
 - 10 PERIODIC OPERATION FOR TRAINING PURPOSES.
 - 11 SHUTDOWN FOR MAINTENANCE AND TESTING.
 - 12 PERIODIC OPERATION FOR TESTING PLANT DYNAMIC RESPONSE.
 - 13 SHUTDOWN FOR REACTIVITY REDUCTION AND HOT SUB-POWER TESTS.
 - 14 FINAL REACTIVITY LIFETIME TEST. FINAL RUN AT FULL POWER FOR CORE 1 SEED 1.
 - 15 LOW-POWER PHYSICS TESTS: CORE LIFE TESTS.
 - 16 50 MW OPERATOR FOR XENON BUILDUP.
 - 17 SHUTDOWN BECAUSE OF XENON BUILDUP.
 - 18 FINAL POWER RUN AT 20 MW.
 - 19 SHUTDOWN TO REPLACE CORE SEED 1, FULL POWER LOW-POWER PHYSICS TESTING.
 - 20 LOW-POWER PHYSICS TESTING.
 - 21 MAINTENANCE - REPLACEMENT OF E-12 CONTROL ROD DRIVE MECHANISM.
 - 22 LOW-POWER PHYSICS TESTING.
 - 23 TESTING AND STUDENT TRAINING.
 - 24 RETURN OF H-8 LOOP TO SERVICE.
 - 25 LOW-POWER PHYSICS TESTING AND MAINTENANCE.
 - 26 LOW-POWER PHYSICS TESTING AND MAINTENANCE.
 - 27 TRANSIENT TEST START OF FINAL FULL-POWER RUN.
 - 28 END OF SEED 2 ABILITY TO OVERCOME EQUILIBRIUM XENON AT 20 MW.
 - 29 TESTING, MAINTENANCE, OPERATOR TRAINING.
 - 30 CORE PHYSICS TESTING - END OF SEED 2 LIFE TESTS.
 - 31 SEED 3 APPROACH TO CRITICALITY SEED 2.
 - 32 SEED 3 APPROACH TO CRITICALITY SEED 3.
 - 33 FULL-POWER OPERATION WITH SEED 3.
 - 34 LOW-POWER PHYSICS TESTS.
 - 35 3000-HR REACTIVITY LIFETIME TEST.
 - 36 3000-HR REACTIVITY LIFETIME TEST.
 - 37 STUDENT TRAINING, TESTING.
 - 38 3000-HR REACTIVITY LIFETIME TEST.
 - 39 OPERATIONS SUPERVISOR TRAINING.
 - 40 REDUCED OUTPUT ON GENERATOR DUE TO FEEDWATER EXHAUSTION.
 - 41 SHUTDOWN LOW-POWER PHYSICS TEST: ROD MECHANISM EXERCISE.
 - 42 SEED 3 ABILITY TO OVERCOME EQUILIBRIUM XENON AT 100% REACTOR POWER.
 - 43 OPERATION AT REDUCED POWER AND LOWER AVERAGE CORE LIFE.
 - 44 END OF SEED 3 OPERATION.
 - 45 SEED 3-SEED 4 REFUELING.
 - 46 INITIAL APPROACH TO CRITICALITY, SEED 4.
 - 47 PHYSICS TESTS.
 - 48 REACTIVITY LIFETIME TEST.

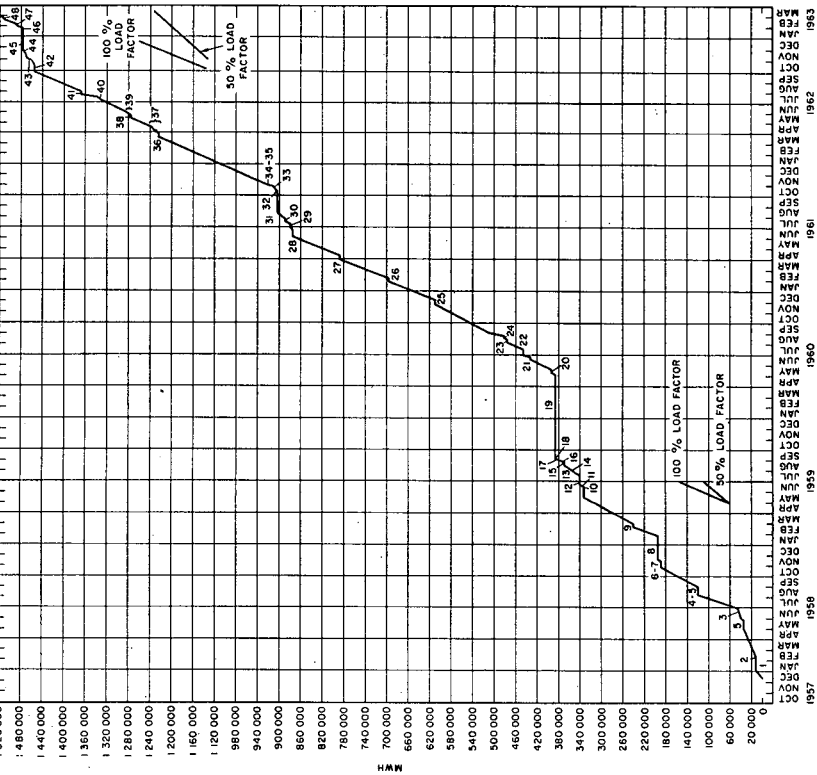


Fig. 3
Operating history of Shippingport Station

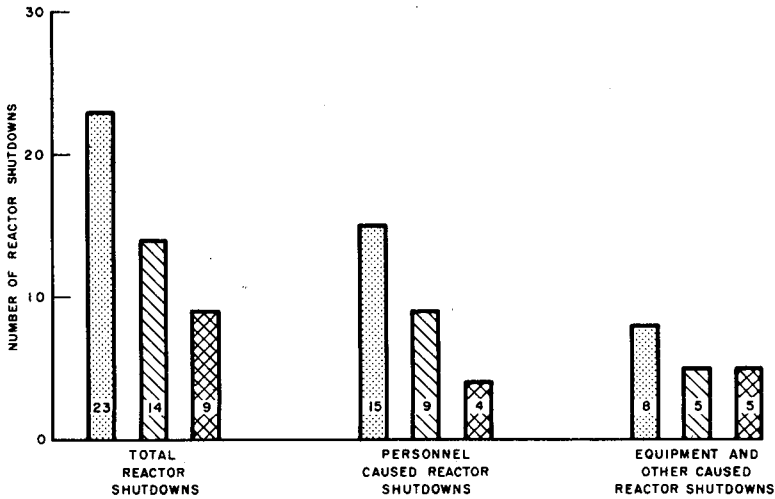


Fig. 4

Analysis of safety shut-downs
(Seed 1, Seed 2 and Seed 3)

of approximately three months, tests consisting of load dumps from various power levels and load swings in both directions were conducted. During the testing, system conditions were such that output was required in those periods of the day when the heaviest load existed, so that the load swings were factored into the system schedule. As a peak-load facility, the station followed all system load changes in the power range. It could also be shut down and started up under controlled conditions at a faster rate than that of any conventional, modern station on the Duquesne system. An example of this capability is shown in Table II, which compares Shippingport performance with that of the conventional, 175 MW(e) Elrama Station.

TABLE II

COMPARISON OF SHIPPINGPORT AND CONVENTIONAL STATION PERFORMANCE

Station output (MW(e))		Shippingport Time (min)	Unit output (MW(e)*)		Elrama Time (min)
From	To		From	To	
Zero (net)	20	1	Zero (net)	35	110
20	60	20	35	175	170
60	Zero (net)	20	175	Zero (net)	230

* Unit No. 4, 1800 lb/in² gauge 1000°F, 1½ in Hg.

Refuelling

After approximately two years of operation, the refuelling of the first seed of Core 1 began in November 1959. About five months were required for refuelling and one month for plant check-out and testing. Full power was achieved on 7 May 1960. Subsequently, the second refuelling commenced on 16 August 1961, and was completed 44 d later. After the necessary plant check-outs and the initial Seed-3 testing, full power was attained on 24 October 1961. The third refuelling of Core 1, i.e. replacing Seed 3 with Seed 4, was completed in 32 working days, beginning on 26 November 1962. By 30 January 1963, full power was attained with Seed 4. Throughout the three refuellings, the primary consideration was personnel and plant safety; no person received radiation exposure greater than the permissible dosage. The refuelling operations were performed by Duquesne personnel under the general supervision of the Commission, with Bettis Laboratory personnel serving as technical advisors to the Commission.

Detailed descriptions and comparisons of the first two refuellings have been published [2,3]. Fig.5 shows a general view of refuelling operations, including the fuel extraction crane positioned over the reactor vessel. In each refuelling operation the overall objective was to replace the 32 depleted seed fuel assemblies. The replacement was performed under water, one fuel assembly at a time, through penetrations in the reactor vessel head by means of the fuel extraction tool. Fig.6 shows an underwater photograph of the removal of a seed assembly with the extraction tool through a head penetration. To provide the necessary access inside the reactor vessel, the control-rod drive mechanism and associated shafting, along with much of the core instrumentation, had to be removed. This equipment was re-installed after the fuel was replaced. In each refuelling operation the refuelling canal at Shippingport proved to be an effective facility for removing and storing highly radioactive fuel without exposing personnel to radiation levels above background.

Each refuelling operation was easier because of the experience gained from the previous refuelling. The overall procedure was about the same in each refuelling, except that 20 d were utilized in the first refuelling to modify reactor-vessel-head test instrumentation. The principal lessons derived from the first refuelling were the significance of detailed planning and scheduling, the importance of thorough personnel training in all phases of the refuelling operation, the necessity of preparing detailed refuelling procedures, and the need for performing equipment check-outs to a greater degree than that required in conventional utility jobs. Centralized responsibility for the entire refuelling operation is a necessity and should include the development of plans and procedures for each operation. One hundred and thirteen individual refuelling procedures for each operation were prepared by Duquesne and the Bettis Laboratory and approved by the Commission for the Seed-2 refuelling operation. The implementation of this comprehensive programme, together with improvements in refuelling equipment design, led to a considerable reduction in the total refuelling time for Seed 2. Since maintenance and operational activities other than refuelling work were concurrently performed, it was essential that an over-

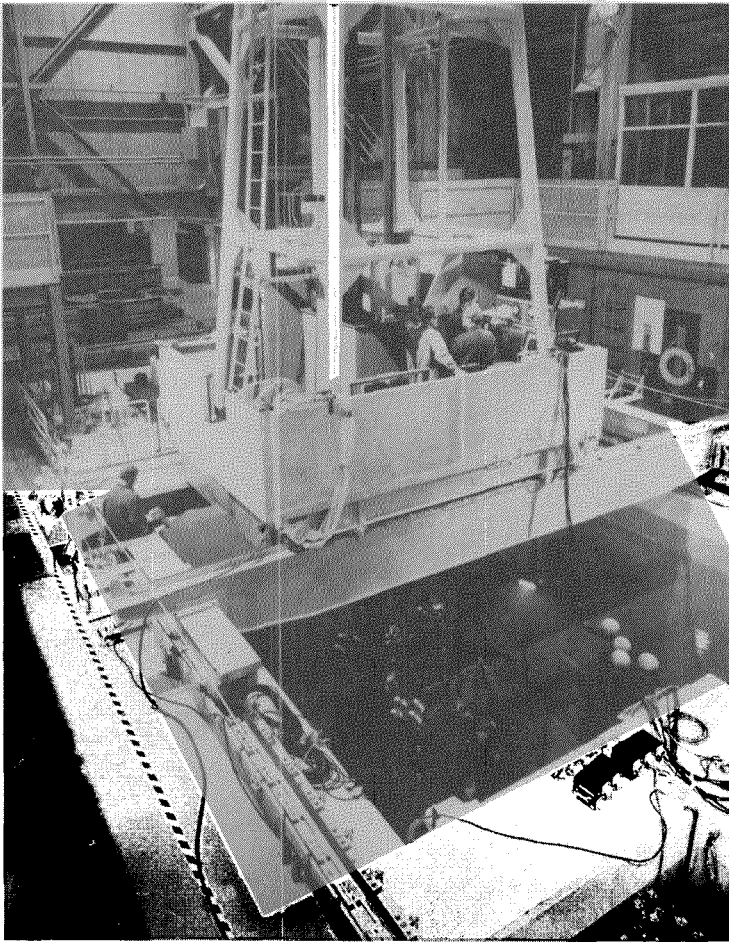


Fig. 5

Operations in fuel canal room during Shippingport refuelling

all supervisor co-ordinate these efforts with the refuelling programme, in order to avoid a conflict of operations and thereby minimize the time required to return the plant to full power. This co-ordination is particularly important in the latter stages of refuelling where non-refuelling work can impede progress. The effect of familiarity with refuelling operations and improved planning and training is reflected in the reduction of refuelling time from the second to the third refuelling - two practically identical operations. Table III shows a break-down of the refuelling operations and compares the actual time and the percentage of the total actual time for the three refuelling periods. The 32 work days required for the third refuelling are comparable to the down time for a normal overhaul of a modern, central station, high-temperature, high-pressure boiler.

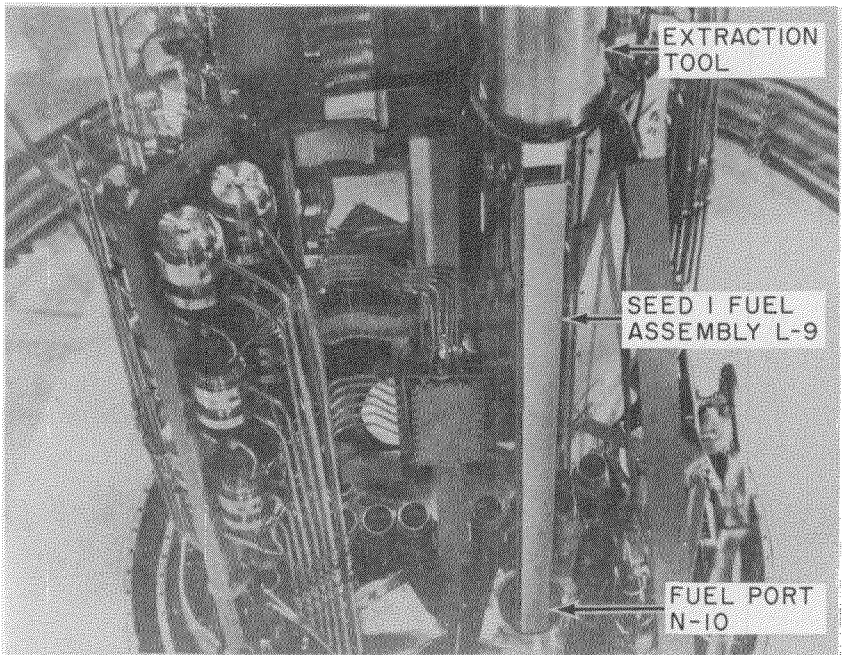


Fig. 6

Seed-1 fuel cluster removal by extraction tool

As shown in Table III, phases II and IV account for over one-half of the refuelling period. It is during these phases that a significant amount of time could be saved if core instrumentation did not have to be removed and replaced. In order to reduce refuelling time further and still retain the core instrumentation for use in the development test programme being carried out at Shippingport, a number of changes have been made in the design of the second Shippingport core. These are described in section 4.

Personnel and environmental safeguards

Operation of the Shippingport Station has provided a wealth of actual operating data on the effectiveness of specific systems and methods for controlling radioactive contamination and personnel radiation exposure both within and external to the station.

Personnel radiation and contamination control

The techniques of radiation and contamination control at Shippingport have evolved into a definite pattern of controlling access to exclusion or contaminated areas, periodic survey of radiation levels in all areas, and continuous monitoring of all personnel for levels of radiation exposure. The role of health physics became a permanent, added function of the station's operation in implementing these controls.

TABLE III

REFUELLING OPERATIONS

Operation	Time (24-h working days)					
	First refuelling		Second refuelling		Third refuelling	
	Actual time	% Total actual time	Actual time	% Total actual time	Actual time	% Total actual time
Phase I - remove external components	17.0	12.7	2.8	6.4	1.9	5.9
Phase II - cut seal welds, remove mechanisms, shrouds, instrumentation	28.0	20.9	16.7	37.9	11.0	34.3
Phase III - fuel replacement	17.0	12.7	4.2	9.5	4.3	13.4
Phase IV - replace mechanisms, shrouds, seal weld	44.0	32.8	11.6	26.4	7.4	23.2
Phase V - replace external components, hydrostatic test, prepare for station start-up	28.0	20.9	8.7	19.8	7.4	23.2
Total working days	134.0	100.0%	44.0	100.0%	32.0	100.0%

Since an essential part of dosage control is the measurement of the radiation received by each individual, all personnel at Shippingport are continuously monitored, and the dosage measurements are then used both as a final control on personnel dosage rates and as a record of dosages actually received.

The bar graph in Fig.7 shows the total dosage received by the personnel in different work groups. Since the radiation exposure per man may be reduced by having more men on a job, a graph totalizing the dosage is considered to be more meaningful in evaluating the requirements of overall radiation control than the conventional "per man hour" unit used in accident records. The comparatively high total dosage received by those under the classification "Others" during some quarters should be deemphasized in light of the very large number of persons involved and the inaccuracy of film badge determinations at low dosages. The graph shows that maintenance personnel received the most radiation dosage; a majority of this dosage was the result of many man-hours spent in moderate (10-20 mr/h) radiation fields. This radiation emanates almost entirely from radioactive corrosion products deposited in the primary coolant system.

As indicated on Fig.7, radiation dosages received by Shippingport personnel were very low during Seed-1, Seed-2 and Seed-3 operations. Office spaces and other normally occupied areas of the plant have shown only background radiation levels. Shut-down radiation levels in the access areas of the reactor plant have been essentially constant throughout core life. A radiation survey taken in August 1961 is shown in Fig.8. The numbers that are circled represent readings in mr/h 2 d after shut-down from power operation. Late in Seed-3 life the survey was repeated, and no significant changes in radiation levels were noted.

Radiation surveys of reactor plant components were made periodically throughout core life. Near the end of Seed-1 operation the radiation levels on the primary coolant piping were measured 2 h after a prolonged 100% power operation. Values of the magnitude of 60-70 mr/h were recorded. Repeat surveys at comparable times in Seed-2 and Seed-3 life indicated values on the order of 70-80 mr/h and 75-85 mr/h, respectively. These measurements indicate that corrosion product contamination does build up in the coolant system with time, but at a slow rate. Average radiation measurements on the reactor vessel head in the vicinity of the control-rod mechanisms were found to be about 800 mr/h during Seed-1 operation. This of course, was an important source of radiation during refuelling operations. However, periodic control rod exercises to flush out crud build-up in the mechanism region during Seed-2 and Seed-3 operations resulted in a reduction of the average radiation level to 400-500 mr/h and contributed significantly to reducing the general reactor-vessel-head radiation levels during refuelling.

One of the more important phases of radiation control is the prevention of accidental high exposures in the reactor plant while the station is at power. An effective control procedure was evolved. Areas in the primary plant container where high radiation levels exist, e. g. reactor chamber, pressurizer cubicle, and flash tank cubicle, were designated exclusion areas. Special warning signs and locked chains were placed across the entrances to these

OPERATIONS: OPERATORS, TEST PERSONNEL, CHEMISTS
 MAINTENANCE: MAINTENANCE PERSONNEL, HEALTH PHYSICS PERSONNEL
 CONTRACTORS: CONTRACTOR PERSONNEL
 OTHERS: OFFICE PERSONNEL, STUDENTS, MANAGEMENT, VISITORS
 AND OTHERS NOT NORMALLY EXPOSED TO RADIATION
 NUMBERS ADJACENT TO COLUMNS ARE THE TOTAL IN EACH GROUP
 (AVERAGE QUARTERLY)

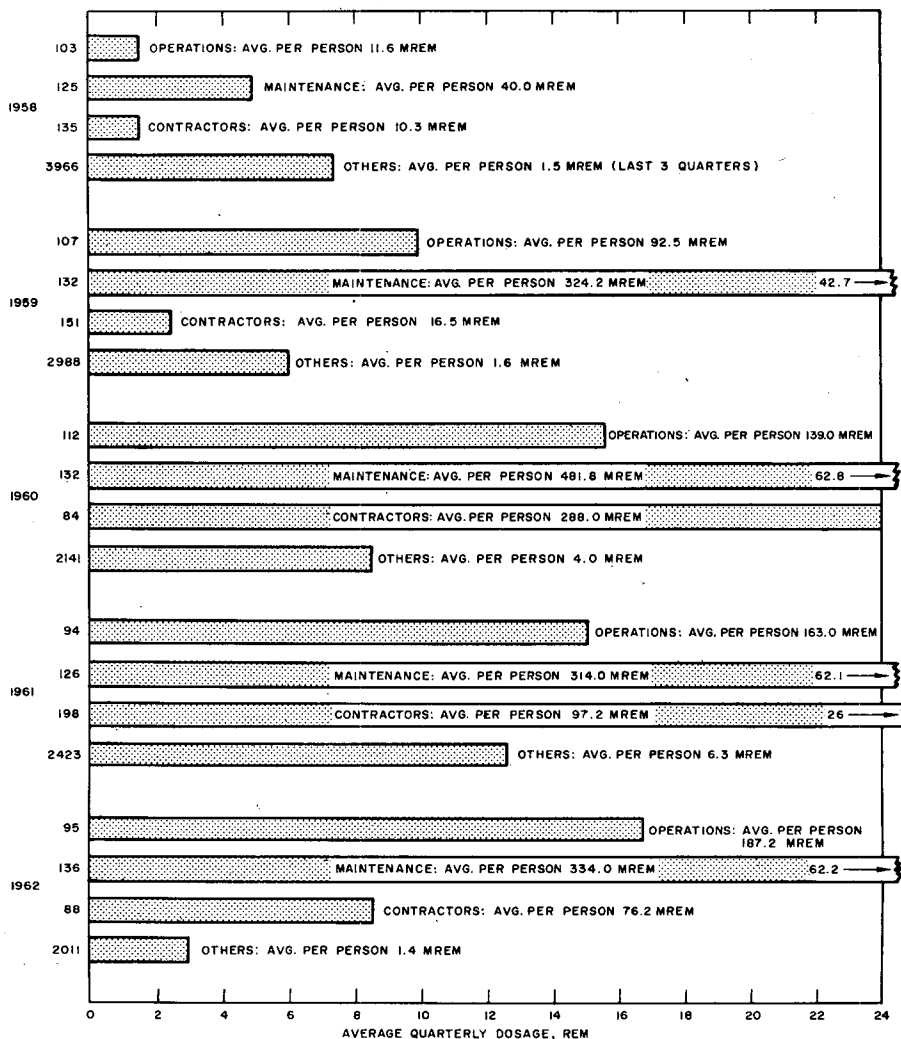


Fig. 7

Shippingport Atomic Power Station radiation dosage summary

areas. Before any of these chains could be removed, except in an emergency, a "radiation clearance form" had to be issued by the operations supervisor. The form prescribed the required radiation control measures and was posted at the area entrance. Before removing the radiation clearance, the areas had to be evacuated and the barriers had to be replaced.

Because of the presence of tritium and radioactive corrosion products in the reactor coolant, it is important to minimize coolant leakage from the primary system. (During Seed-1 and Seed-2 operations the reactor plant leakage was of the order of 35 to 50 gal/h.) This leakage is from primary coolant component discharge lines with most of the leakage coming from the four relief valves on the reactor vessel and the two relief valves on the pressurizer. The reactor coolant leakage is collected into the blow-off tank and then discharged to the waste disposal system, where it is processed. This coolant leakage has readily been made up by the high-pressure charging pumps, which have a total capacity of 3000 gal/h.

During the second refuelling four relief valves which were the principal source of primary coolant leakage were disassembled, inspected and their seats honed. Tests indicate that this maintenance on the relief valves has significantly reduced the plant leak rate. During Seed-3 operation, reactor plant leakage was of the order of 5 to 10 gal/h.

In conclusion, no health or operating hazards have been experienced during the four years of Shippingport operation.

Environmental monitoring

A continuous environmental monitoring programme has been conducted at Shippingport. The results obtained to date from this programme indicate there has been no measurable increase in activity due to Shippingport operations. These results confirm earlier expectations based on evaluations of the minimal amounts of activity discharged from the plant. In addition, it is concluded that, since they are less affected by environmental factors not connected with the plant, monitors at sampling locations closer to the site will provide information which is satisfactory and accurate. Accordingly, some distant monitoring locations were recently eliminated. Concomitant with this elimination, arrangements were made to provide to the Pennsylvania Department of Health some of the resulting excess monitoring equipment to be used in the vicinity of Shippingport in the state's radiation surveillance network.

Waste-disposal system

In the first four years of operation at Shippingport, the waste-disposal system has performed adequately over a wide range of conditions [4]. The criteria for the discharge of radioactive wastes are contained in a permit granted to Duquesne Light Company by the Commonwealth of Pennsylvania. Under the terms of the permit, a maximum of 6200 $\mu\text{C}/\text{d}$ and a yearly average of 1590 $\mu\text{C}/\text{d}$ of radioactivity may be discharged to the Ohio River. Tritium has an individual discharge limit of 10 c/d averaged over 365 consecutive days and a maximum of 50 c/d. In all cases the concentration of radioactivity in the final effluent discharged to the river or atmosphere must not be more than one-tenth the maximum permissible concentration, as given in National Bureau of Standards Handbook No.69. Concurrent with the revision of the Pennsylvania Radiation Safety Code in 1962, an increase of the daily permissible maximum and average discharge rates by a factor of 10 to 62 000 μC and 15 900 μC , respectively, was granted. The increase

was requested upon invitation by state authorities and not because of any operational need. Although an increase in discharge rate is permitted, Shippingport operations continue to use the former limitations as an operational policy and have consistently maintained the radioactive release well within those limits.

As shown in Fig.9, the waste-disposal system functions quite adequately to keep discharges to an acceptable minimum. The chart shows the total monthly discharge of radioactivity into the Ohio River. There are three periods of peak loads that occurred as a result of decontamination processing of clothing and equipment during the three refuellings. All of these quantities are considerably less than the maximum average allowable under the terms of the waste-disposal permit.

Although tritium is not shown in Fig.9, the average discharge over a four-year period was only one-sixtieth of the average allowable tolerance level under the terms of the waste-disposal permit. At the present time, tritium activity in the coolant has been reduced to an insignificant level. The recent two-months' average of tritium discharged was 0.02 c/d.

In December 1960 isotopically pure lithium (99.99% Li-7) was made available as lithium hydroxide for pH control of the reactor coolant. Since the primary source of tritium in the reactor coolant is Li-6, the concentration has been greatly reduced. A maximum concentration of tritium of 280 $\mu\text{c/l}$ was reached during Seed-1 operations; at the present time the concentration in the coolant is only 2 $\mu\text{c/l}$.

The operation of the Shippingport reactor has thus far been characterized by coolant fission-product activity levels considerably lower than those the waste-disposal system was designed to handle. The primary coolant-purification system and the waste-disposal system were designed to operate with as many as 1000 defected blanket fuel elements in the core. During Seed-1 and Seed-2 operations the fuel-element detection and location (FEDAL) system indicated that there probably were several slightly defected fuel rods in the blanket, and these were removed during the second refuelling for examination. The distribution level of fission products in the primary coolant has been so low that it could be accounted for entirely on the basis of the fission products produced by the $2\frac{1}{2}$ ppm of natural uranium which exists as an impurity in the Zircaloy cladding. Based on the performance of the waste disposal system to date, it is concluded that an adequate margin exists in the waste-disposal system to meet any processing requirements at Shippingport.

Emergency plans in the event of a radioactive release

Although it is highly improbable that a release of hazardous quantities of fission products to the environment or beyond the site boundaries will take place, it is considered prudent to be prepared for handling such an emergency. For this reason an emergency plan has been prepared to assure a rapid and orderly implementation of those steps which would be required to minimize any hazard to the surrounding population. The emergency plan covers both radioactive discharge into the atmosphere and radioactive discharge into the Ohio River. It includes the immediate and follow-up actions to be taken by Duquesne and Commission personnel in determining the

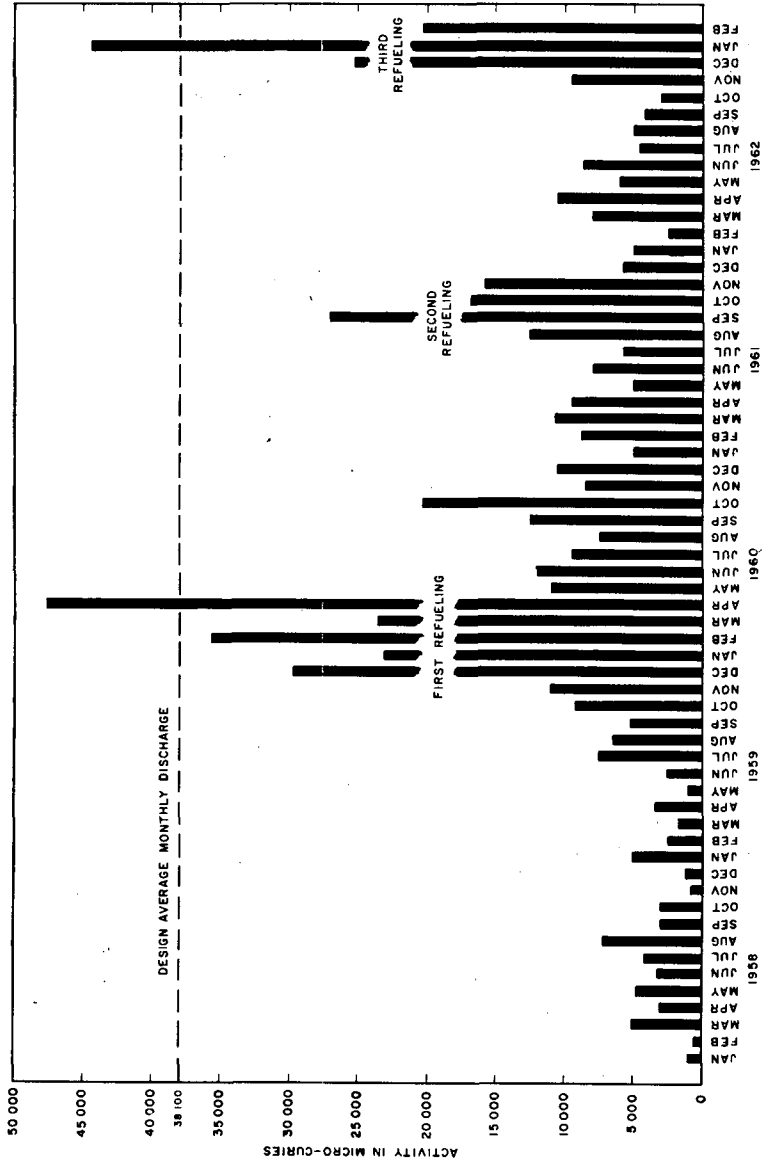


Fig. 9
Monthly activity discharged to Ohio River

amount of radioactivity released, in limiting its effects as much as possible and in securing any outside help required.

To assure that station personnel understand emergency procedures and will perform effectively under emergency conditions, various emergency situations are simulated by means of practice drills. These drills are conducted to reveal weaknesses in the procedures and to familiarize personnel with the actions they are expected to take during an emergency. As a result, many detailed changes have been made to further improve the effectiveness of these procedures.

Training, procedures and manuals

Formal personnel training and certification, written administrative procedures, power plant manuals, and other documents are considered necessary for the safe and effective operation of a nuclear power station. They have proved to be essential to the successful implementation of the Shippingport programme.

Shippingport Atomic Power Station Manual

A manual consisting of four volumes with over 100 chapters contains detailed procedures for operating and maintaining the station. It was prepared by Duquesne and Bettis Laboratory personnel and was approved by the Commission before station start-up. The manual is kept up-to-date to reflect any changes in procedures or plant modifications and any changes affecting the reactor plant or radioactive waste-disposal system are approved by the Commission.

Training and certification of Duquesne Station operating personnel

To comply with the requirement that operating personnel be certified by the Commission, personnel selected as candidates for station operation are given approximately six weeks of classes and eight to twelve weeks on-the-job training by Duquesne. When a trainee has satisfactorily demonstrated his ability as an operator or operating supervisor, an application is submitted to the Commission for the trainee's certification. The application contains a certificate of medical examination together with data which indicate that the applicant has completed the required training and is considered by Duquesne to have the proper qualifications. Upon acceptance of the application, the applicant must pass a comprehensive written and oral examination given by the Commission. After successful completion of both examinations the applicant is certified for two years, after which time he must be recertified. To date, seventeen supervisors and twelve operators are certified.

Station test programme

The Shippingport test programme was designed to extract and process technical data from the operation of the station and to make these available to the nuclear industry. The initial function of the test programme, that of proving out the installation and design of equipment and systems before and

after the initial operation of the station, was completed in 1958. The test programme since then has been conducted to obtain detailed information on the performance of the station, systems and equipment. Detailed written procedures for conducting tests and subsequent test evaluation reports are prepared by the Bettis Laboratory and Duquesne and are approved by the Commission. During Seed-2 and Seed-3 operation over 270 tests were performed.

Station administrative manuals

Manuals have been prepared by Duquesne to specify the organization and responsibilities of Duquesne personnel for operating the station, refuelling the reactor and performing maintenance operations on the reactor plant. In addition, these manuals prescribe the administrative procedures to execute these responsibilities. The responsibilities and duties of the various personnel involved in these activities are defined so as to provide complete instructions to station personnel for the safe conduct of station operations. The manuals have been approved by the Commission and complement the approved operating, refuelling and maintenance procedures.

3. CORE AND PLANT PERFORMANCE

The performance and high integrity of the reactor core, the reliability demonstrated by primary coolant components and the ease with which the reactor plant has been operated and maintained are highlights of the Shippingport Station operation to date. These results are attributed principally to design care, close manufacturing control, extensive proof testing of components and prototypes and detailed, carefully prepared installation procedures. This statement should not be construed to imply that no equipment malfunctions were encountered; there were a few primary plant components which had to be repaired or replaced. However, these equipment difficulties affected the station availability only to a minor degree.

Reactor instrumentation

In order to provide data for analysis and design verification of core performance, Core 1 has been extensively instrumented. Fig. 10 shows the various types and locations of the core instrumentation installed in Core 1, Seed 1. This instrumentation consists of seed instrumented fuel plates for measuring central fuel temperature [5], core thermocouples for measuring seed and blanket cluster inlet and exit coolant temperatures, flow meters for measuring flow distribution to individual clusters, a failed-fuel-element detection and location (FEDAL) system, and nuclear instrumentation located in the neutron shield tank at four corners outside the core.

The seed fuel thermocouples provided a means for measuring and verifying the axial power distribution in the core. For this purpose, each instrumented fuel plate contained six thermocouples, evenly spaced over the length in Seed 1, and four thermocouples in Seed 2. These instrumented plates

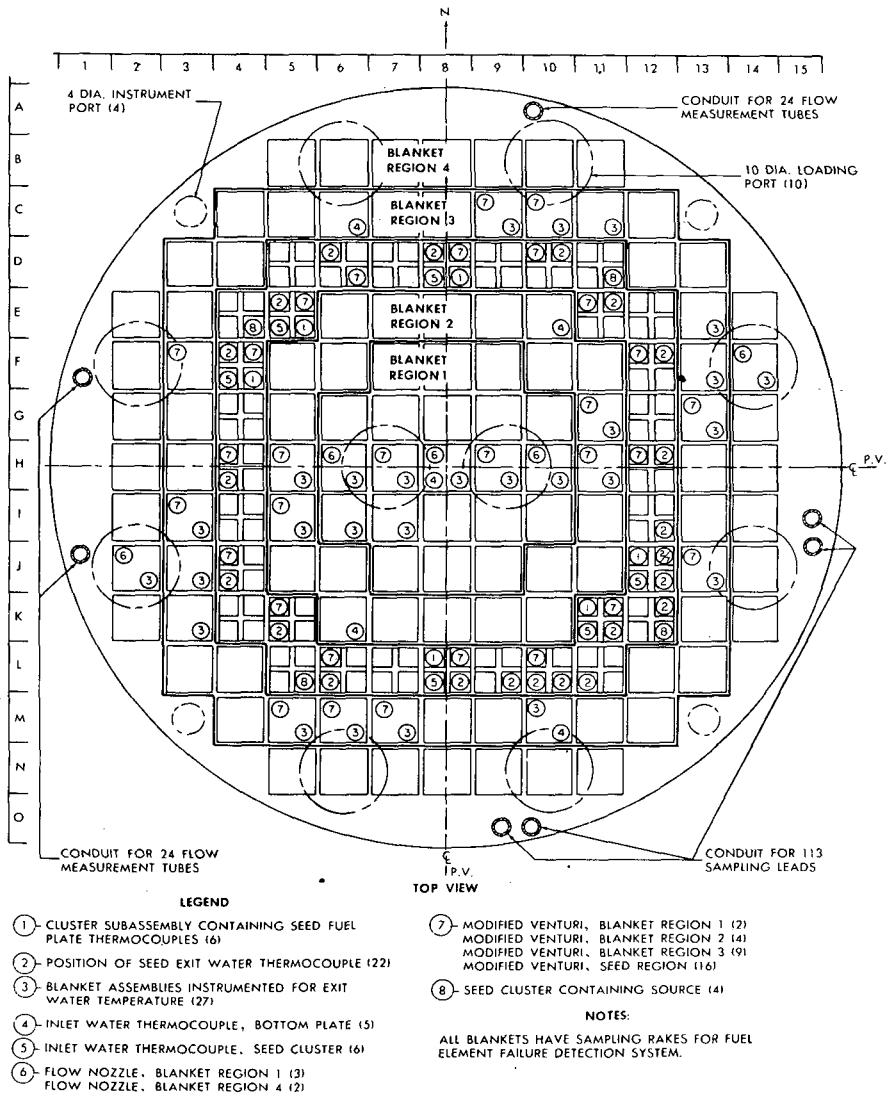


Fig. 10

Core-1 instrumentation (Seed 1)

were contained in six clusters which were located in symmetric regions of the core where high fuel temperatures were calculated to occur for Core-1 operation. In addition to measuring axial power distribution they were also used for monitoring the central fuel temperature to assure that core design limits were not exceeded.

The inlet and exit seed-coolant thermocouples provided a means for measuring and verifying the power generation in selected clusters by the

product of coolant temperature rise and measured flow rate (with known specific heat).

Flow measurements for individual seed and blanket clusters were obtained by measuring the pressure drop across a Venturi section located in the cluster inlet. The total reactor pressure drop was measured by a differential-pressure cell responding to pressure signals obtained from the hot and cold legs of two of the four main coolant loops.

The FEDAL system samples coolant from each blanket cluster by means of rakes located at the exit of each cluster. These samples are piped through a multi-port valve on the reactor vessel closure to a delayed neutron monitor located outside the reactor vessel. The multi-port valve operates continuously, successively sampling coolant from each cluster five times per day. (The frequency is adjustable.)

The nuclear instrumentation located at the four corners outside the core monitors the neutron leakage which is proportional to reactor power. It consists of four independent channels, each of which has a boron trifluoride (BF_3) detector for the source range and a compensated ion chamber (CIC) for the intermediate and power ranges up to 150% of full power.

Seed reactivity lifetime

The equilibrium xenon and samarium reactivity lifetime of Seed 1, which contained 75 kg of 93% enriched uranium, was 5532 EFPH as against a minimum design requirement of 3000 EFPH. The equilibrium xenon lifetime of Seed 2 which contained 90 kg of 93% enriched uranium and 170 g of boron was 7528 EFPH as against a minimum design requirement of 5000 EFPH.

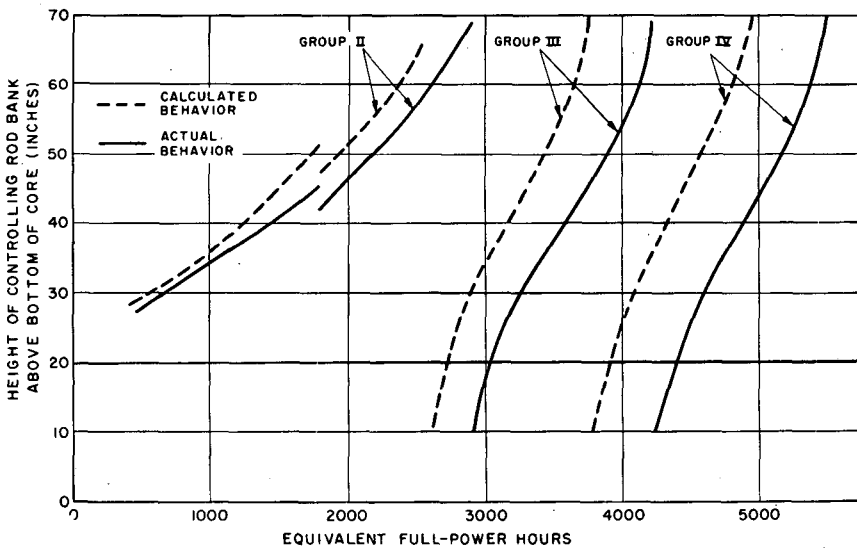


Fig. 11

Critical control-rod configurations versus Core-1 Seed-1
lifetime (equilibrium xenon condition)

Seed 3, which was identical to Seed 2, had a slightly shorter equilibrium xenon lifetime, 6935 EFPH, due to the slightly lower reactivity of the blanket.

Fig.11 shows the results of an analysis of Core-1 reactivity behaviour during life. The control-rod-bank critical position predicted in the calculation as a function of lifetime is compared with the actual core behaviour and is seen to underestimate the reactivity lifetime by about 500 EFPH. This value is equivalent to a difference of about 1% in reactivity. The discrepancy between measurements and calculations is not clearly understood. Additional analytical effort is being devoted to this area.

Measured and calculated values of core excess reactivity at the beginning of each seed life are shown in Table IV.

TABLE IV

COMPARISON OF MEASURED AND CALCULATED VALUES OF CORE EXCESS REACTIVITY

	Seed 1	Seed 2	Seed 3
Excess reactivity - inferred from rod worth measurements (% ρ)	14.8	13.9	12.8
Excess reactivity - calculated with one dimensional depletion code (% ρ)	15.5	14.7	13.2

The initial reactivity effect of increasing the seed fuel loading from 75 to 90 kg from Seed 1 to Seed 2 is more than compensated for by the effect of adding 170 g of boron. The fact that the initial excess reactivity is less for the third seed, compared to the nominally identical second seed, is indicative of a variation in the reactivity properties of the blanket region because of the formation of plutonium and fission products. It is interesting to note that one-dimensional calculations predicted a reactivity drop from Seed 2 to Seed 3 of 1.5%, compared to a measured drop of about 1.1%. Thus these calculations tend to underestimate the reactivity of the blanket after significant depletion.

Seed-blanket power sharing

One of the major objectives of the seed-blanket reactor concept is to obtain a substantial amount of energy from natural uranium in a light-water system. The actual power sharing between the seed and blanket of Core 1 in Shippingport was therefore of special interest and importance. Fig.12 shows that the power produced in the natural uranium blanket during the first three seed lives averaged about 54%.

The power produced in the blanket of a seed-blanket core is strongly sensitive to the blanket reactivity. Calculations on Core 1 indicate that the reactivity of the blanket increases as a function of Seed-1 life until approximately 5000 EFPH because plutonium build-up overcomes depletion and fission product effects. During the life of Core 1, to date, about one-half

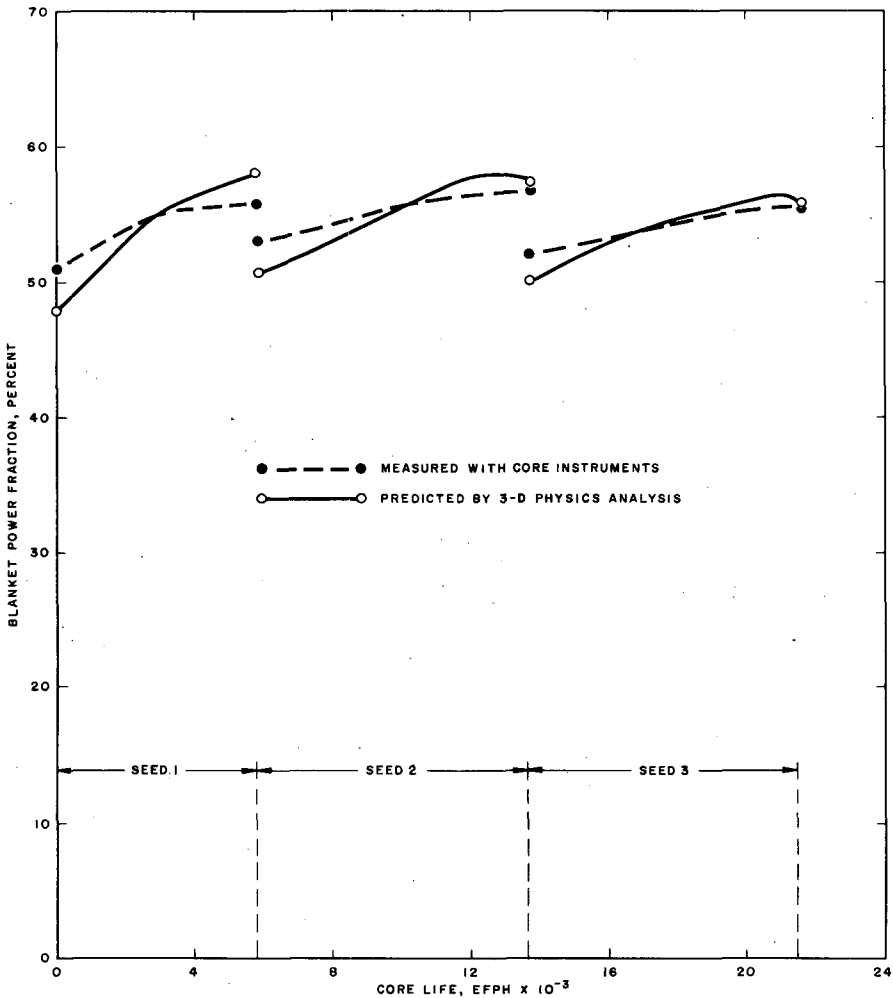


Fig. 12

Percentage of core energy from natural uranium blanket
during Core-1 life

of the blanket energy has been generated by burning plutonium in place and an additional 8% from direct fissioning of U^{238} . The balance of the blanket energy is from U^{235} , whose content is now only about one-third of the original. These calculated figures are consistent with measurements (Fig.13) of plutonium build-up in blanket fuel rods removed after the first and second seed lives.

Temperature coefficients of reactivity

An important feature of seed-blanket cores is that they afford an especially effective means of obtaining a suitable negative temperature

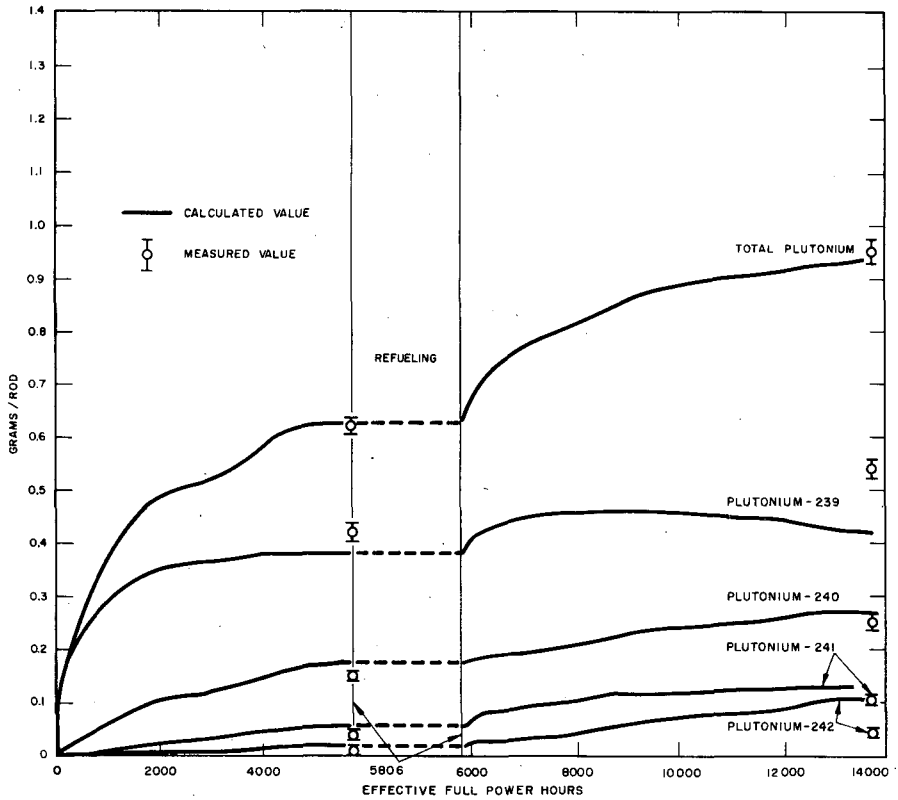


Fig. 13

Plutonium isotope content of rod irradiated for
two seed lives

coefficient in large power cores. The temperature coefficients for Core 1 with Seed 1 and Seed 2 were measured both at zero power and while operating at full power under equilibrium xenon conditions. Fig. 14 compares the measured temperature coefficient under operating conditions (500°F) as a function of core life with the calculated values for Seed-1 operation [6]. It is seen that the coefficient remained at a suitable negative value, thus providing ease of control and stability under load transients. The same basic characteristics of the temperature coefficient have been found during Seed-2 operation. The trend that the temperature coefficient becomes less negative with lifetime exists in both seeds. More precise analytical treatment is being sought to reduce the noted discrepancy between the measured and predicted values.

Power oscillations

Spatial power asymmetry (tilt) was first observed in Core 1 by the four channels of nuclear detectors during a rapid xenon depletion test [5]. The power tilt oscillated on a period of about 24 h, as shown in Fig. 15, and the

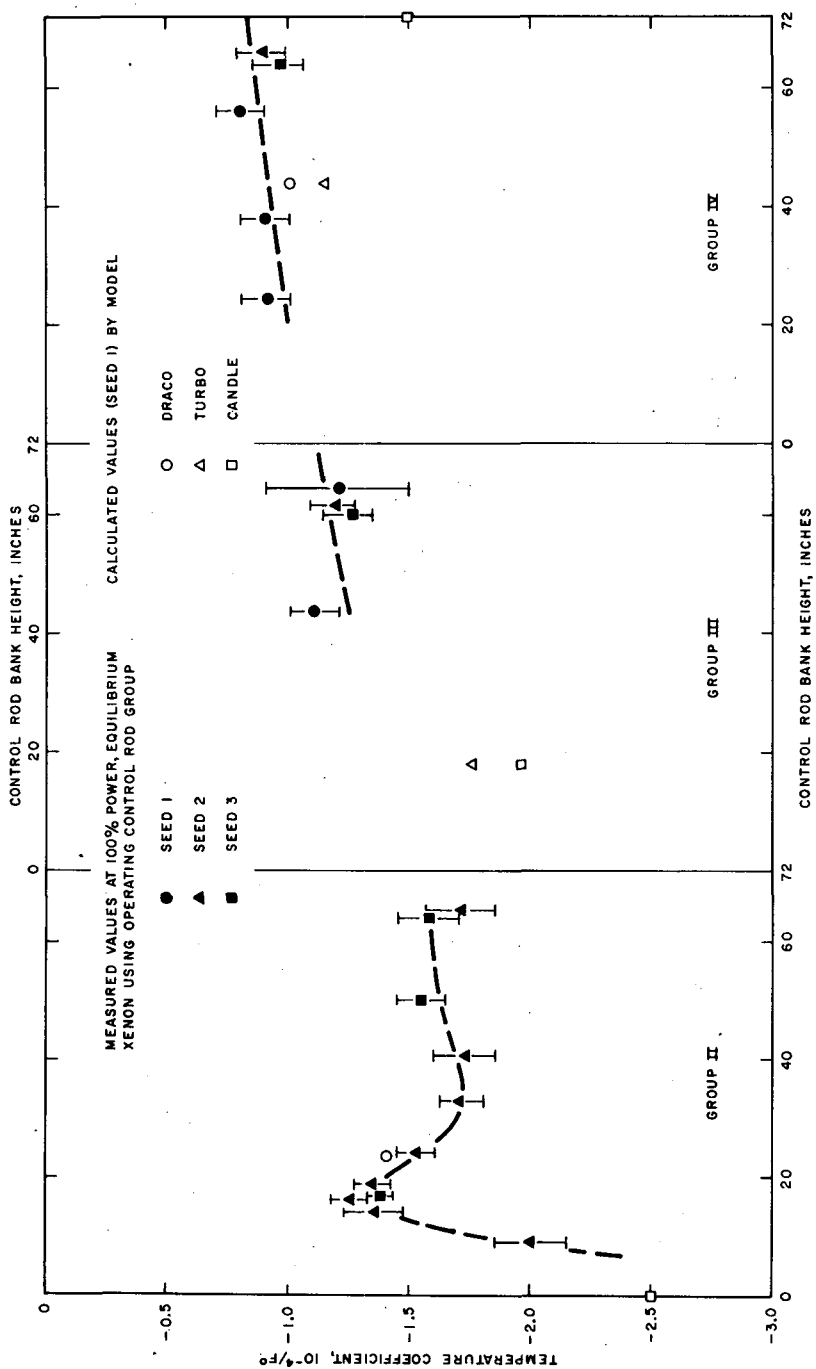


Fig. 14

Core-1 temperature coefficient for Seeds 1, 2 and 3

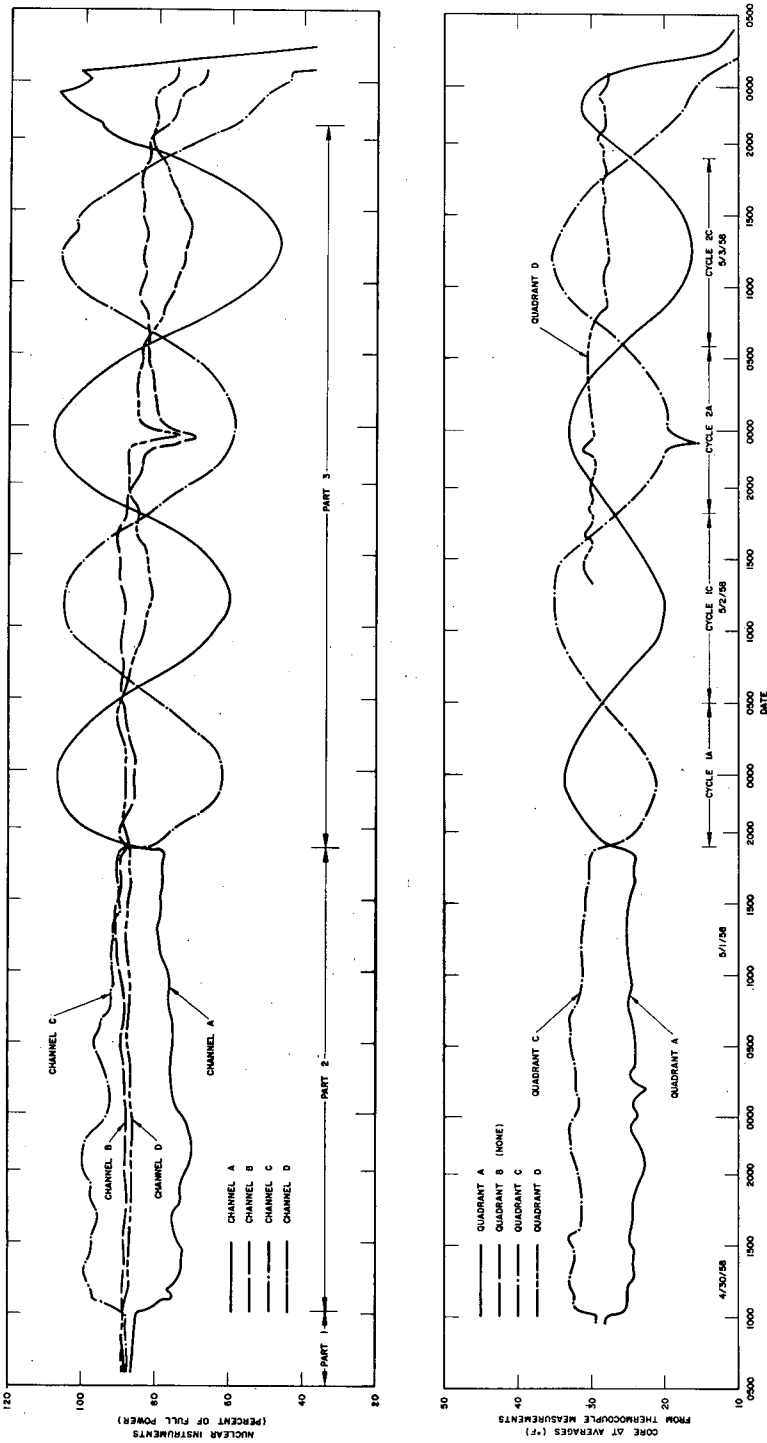


Fig. 15
Core-1 power oscillations (Seed 1)

amplitude of the oscillations increased with time. The measured coolant-temperature rise in seed clusters also showed an oscillation of the same frequency which was in phase with the oscillation indicated by the neutron detectors. This power oscillation was eliminated by repositioning of the controlling rods. Subsequently, a special test was conducted in which a power tilt was purposely introduced by asymmetric positioning of the controlling rods. When the rods were returned to their previous position, spatial oscillations resulted.

In addition to the above two tests, a tendency to develop spatial power oscillations was also observed in the course of a long full-power run. The oscillations were observed to converge at approximately 2900 EFPH and to disappear when the controlling group (Group II) of rods was fully withdrawn.

In order to cope with such power oscillations and protect any region of the core from exceeding design temperatures, a procedure was established which provided that, when the nuclear detectors indicated 5% above average power in any core quadrant, the controlling group of rods in the high-power quadrant was to be inserted until the power level in that quadrant was depressed to the average value.

During Seed-2 operation no spontaneous spatial power oscillations were observed. A deliberate attempt was made to induce oscillations by misaligning the rods during a test near the end of life of the seed. It was significant that at this time in core life, the power oscillations generated (Fig.16) were convergent, in contrast to the divergent behaviour found in oscillations purposely generated during tests with Seed 1. This result indicated that, with Seed 2 and the more highly irradiated blanket, the core quadrants behaved in a more closely coupled fashion than with Seed 1. Although no radial power oscillations occurred during Seed-3 life, they have begun again during early Seed-4 operation. Although these oscillations can be readily controlled, their behaviour is not well understood. Considerable analytical effort is currently being devoted to this area.

Centre-line seed fuel temperature

The axial power generation patterns can be inferred from centre-line fuel temperature measurements. Figs.17 and 18 show such patterns for two control-rod conditions and compare the predicted power distributions with those obtained from temperature measurements in the Core-1 seed fuel [6]. As these figures indicate, the axial power distribution is sensitive to the position of the controlling rod group and, particularly, to the rod position in the individual cluster. The relationship of the measured and the predicted values is considered a reasonably good verification of the core's calculated gross heat-generation pattern and heat-removal design.

Core hydraulics

Special tests were conducted to determine the amount of mixing in the reactor-vessel inlet plenum of the coolant flow from the operating loops. This knowledge is especially important to cold water accident studies. In the tests, with the reactor shut down, three reactor coolant loops were operated at 500°F and a fourth loop having a 10 to 40°F lower temperature

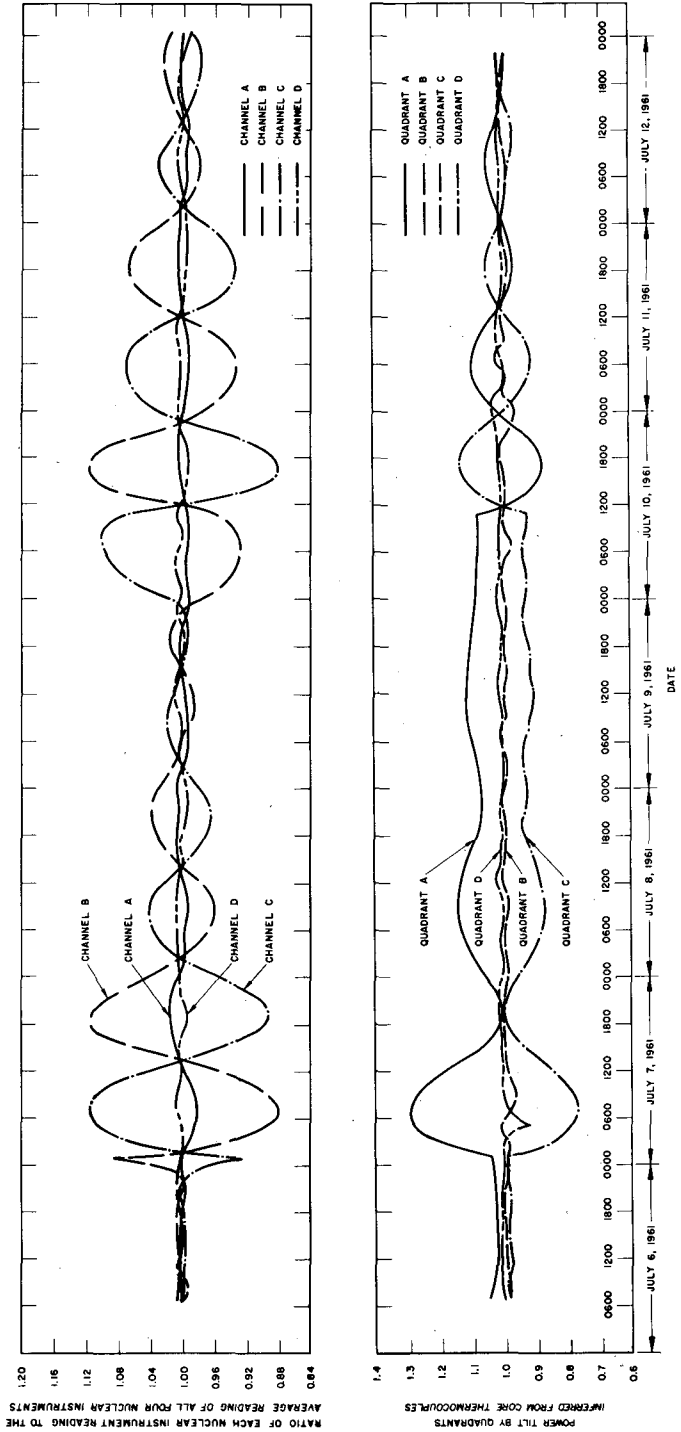


Fig. 16
Core-1 power oscillations (Seed 2)

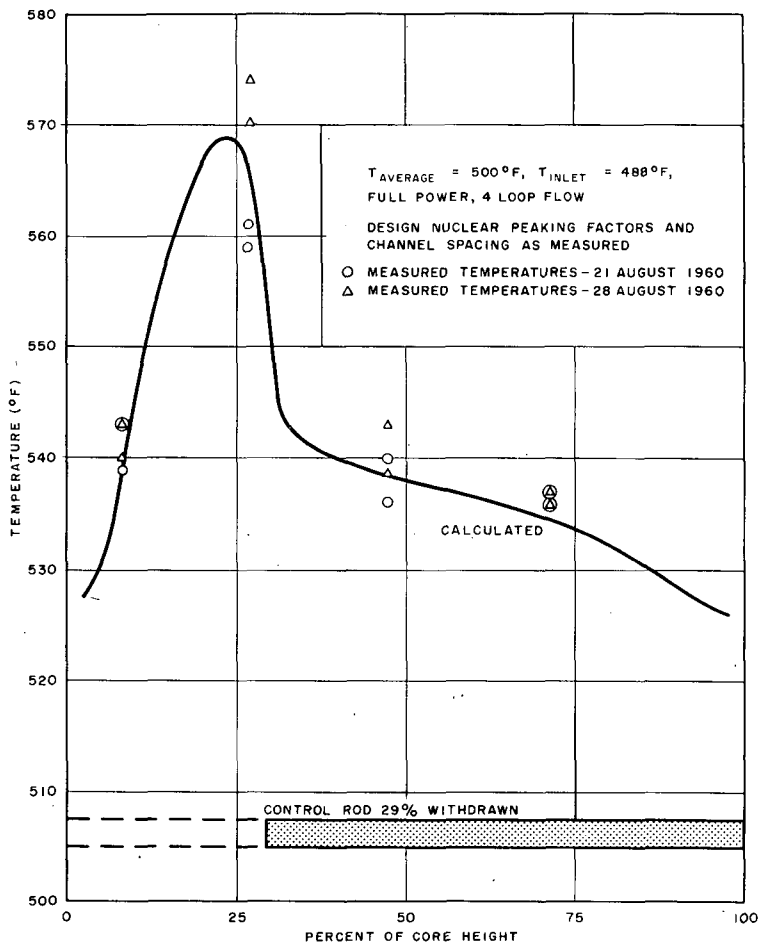


Fig. 17

Comparison of measured versus calculated centre-line
 seed fuel temperature of Core 1
 (Clusters containing partially withdrawn rods)

was brought into service. A comparison of measured and predicted values is shown in Fig.19. The agreement between the measurement and prediction is considered good, especially in view of the limited accuracy of such measurements [7] .

No significant increases in the pressure drop monitored across the reactor were observed as a function of time during Seed-1 and Seed-2 operations. This indication of the absence of fouling was further verified by inspection and crud measurements of seed and blanket clusters at the end of Seed-1 and Seed-2 operations.

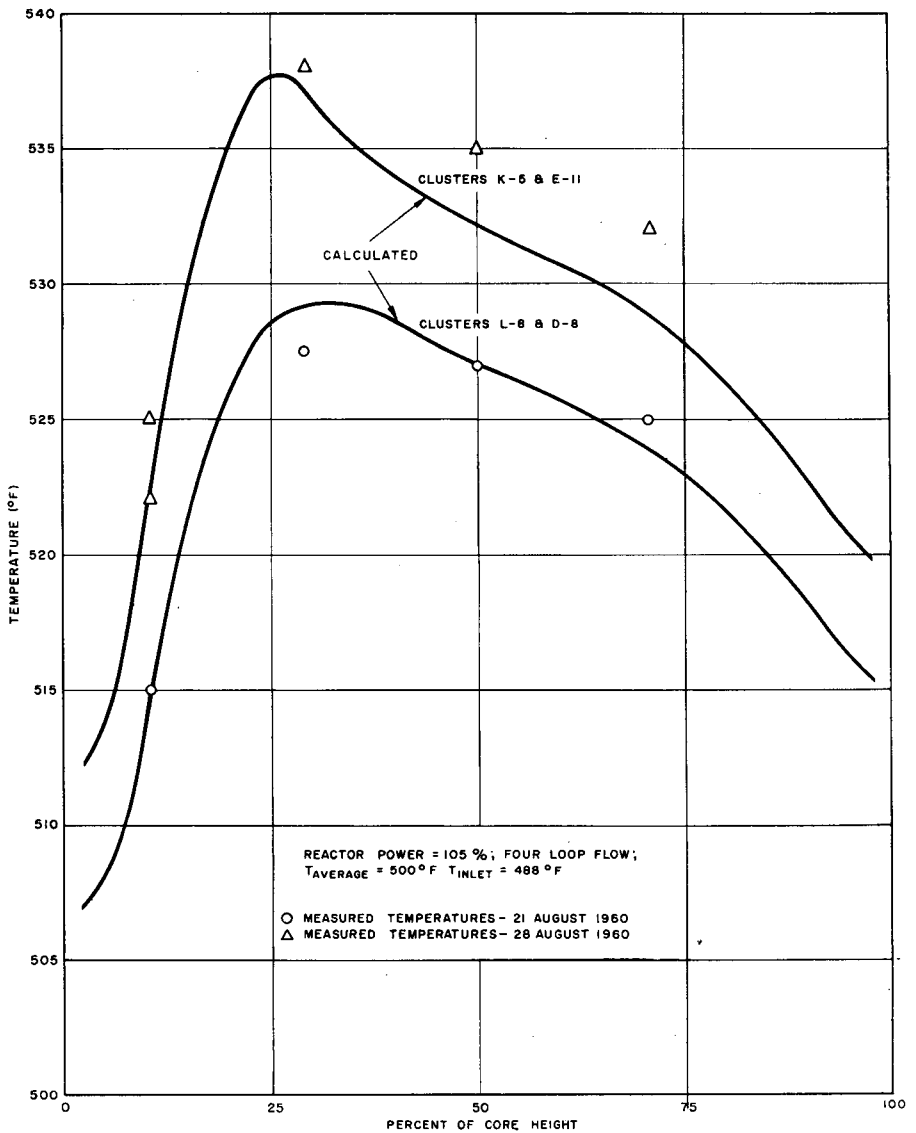


Fig. 18

Comparison of measured versus calculated centre-line
seed fuel temperature of Core 1
(Clusters containing fully inserted rods)

Core transient performance

An important PWR design requirement is that adequate thermal margin be available in the core in the unlikely event of a complete loss-of-flow accident. This thermal margin depends, to a large degree, upon the rate of flow coast-

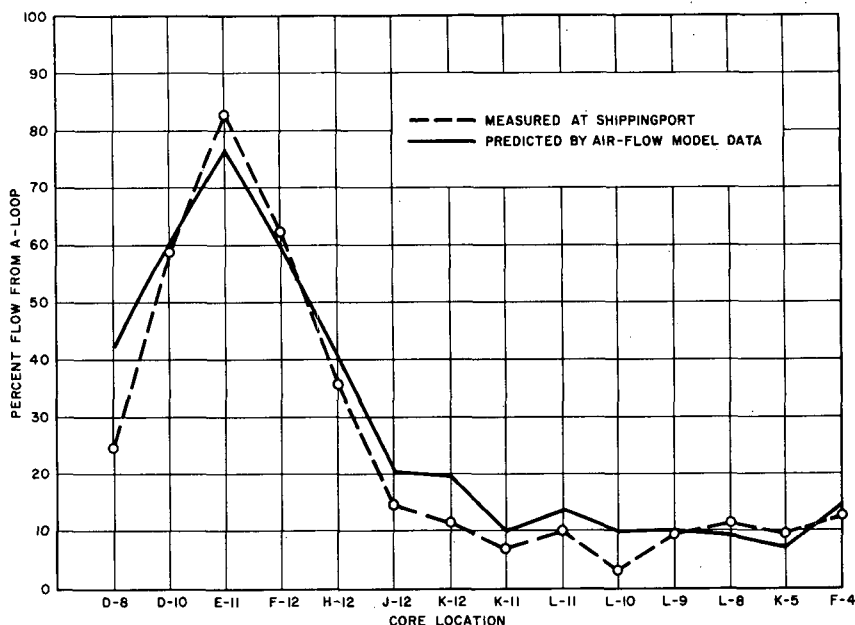


Fig. 19

Coolant mixing in reactor vessel inlet plenum

down following loss of pumps and the rate of power coastdown following the scram. Fig.20 shows good agreement between the predicted flow coastdown and measured flow coastdown during test runs with Seed 1 and Seed 2. Fig. 21 shows similar agreement between predicted power coastdown and measured power coastdown during tests with Seed 3. These measurements confirm the adequacy of the analytical methods used in the transient thermal design calculations.

An interesting measurement was made during the power coastdown test, as shown in Fig.22. The graph shows that blanket heat flux (as determined by core exit thermocouples) falls off much slower following a scram than does the neutron flux. This result is due to the low thermal diffusivity of the irradiated UO_2 fuel material in the blanket rods.

Natural UO_2 blanket irradiation effects

An important objective of the Shippingport programme is to determine, on a large scale, the maximum burn-up that the Core-1, natural UO_2 , Zircaloy-2 clad fuel elements can withstand. The results obtained by the end of Seed-3 life, after an average burn-up of 8450 MWd/t of uranium and a peak burn-up of 28 600 MWd/t of uranium, have been most encouraging.

Post-irradiation examination of blanket fuel-rod bundles removed after Seed-1, Seed-2 and Seed-3 operations has shown them to be in good con-

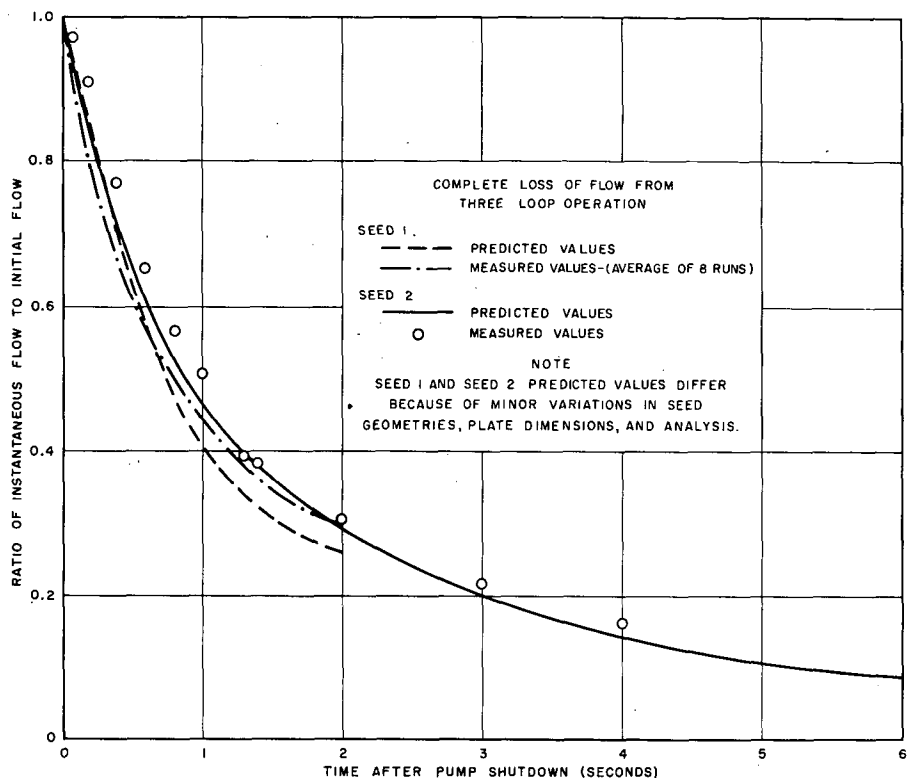


Fig. 20

Flow coastdown in Core 1 (Seed 1 and Seed 2)

dition. Some of the more important conclusions reached during these examinations are:

- (1) No significant dimensional changes occurred in the rod bundles selected from the highest depletion region, the intermediate flux regions, and the low-power, low-flow regions.
- (2) Fission gas release measurements, made on 15 fuel rods from the region of maximum depletion after Seed-1 operation, compared reasonably well with the rates computed [8] on the basis of solid-state diffusion theory. These data are also in agreement with release rates obtained experimentally in in-pile loop tests under comparable conditions.
- (3) After the first seed life the Zircaloy-2 cladding had an average hydrogen content of 60 ppm, based on measurements of five samples of Zircaloy-2 from each of three fuel rods. Individual values ranged from 50 to 100 ppm. These hydrogen values represent the sum of the hydrogen originally present in the Zircaloy-2 material (40 ppm) plus the amount absorbed from the reaction with water during corrosion testing of the fuel elements at the time of manufacture (10 ppm) and the amount absorbed during core operation (10 ppm). The

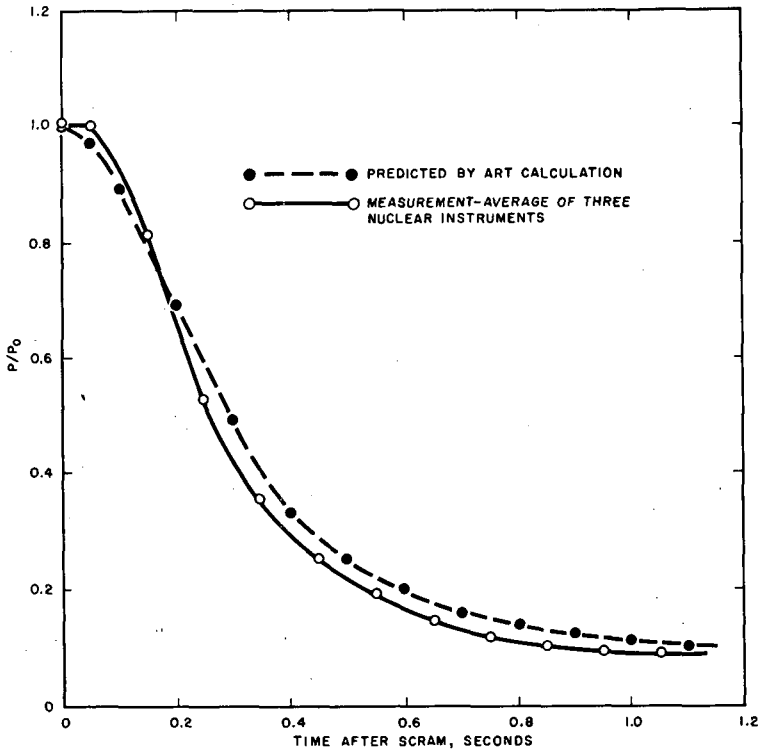


Fig. 21

Power coastdown following reactor scram (Seed 3)

levels are well below the 200-250 ppm level above which hydrogen embrittlement could be of concern.

(4) The maximum local deposit of crud after Seed-1 operation was approximately 160 mg/dm² (about 0.5 mil thick), which would cause a negligible rise in the surface temperature of the cladding.

(5) Detailed metallurgical examination of blanket UO₂ fuel material in a rod removed during the second refuelling indicated performance consistent with prediction[9]. This fuel rod received an estimated fission depletion of 4×10^{20} fissions/cm³ (16 100 MW d/t of uranium) and exhibited no distortion or changes in rod dimensions. The fuel showed no abnormal cracking or grain growth, as shown in Fig.23, and the 4-mil radial annulus between fuel and cladding was still intact. Hydrogen analysis of the Zircaloy-2 cladding showed the average hydrogen content of the cladding to be less than 80 ppm.

(6) Detailed examination of a similar high depletion rod removed during the third refuelling and having an estimated exposure of 23 000 MWd/t of uranium is in progress.

(7) A PWR Core-1 blanket assembly which operated for the first two seed lives was identified by the FEDAL system (described earlier) to have a small

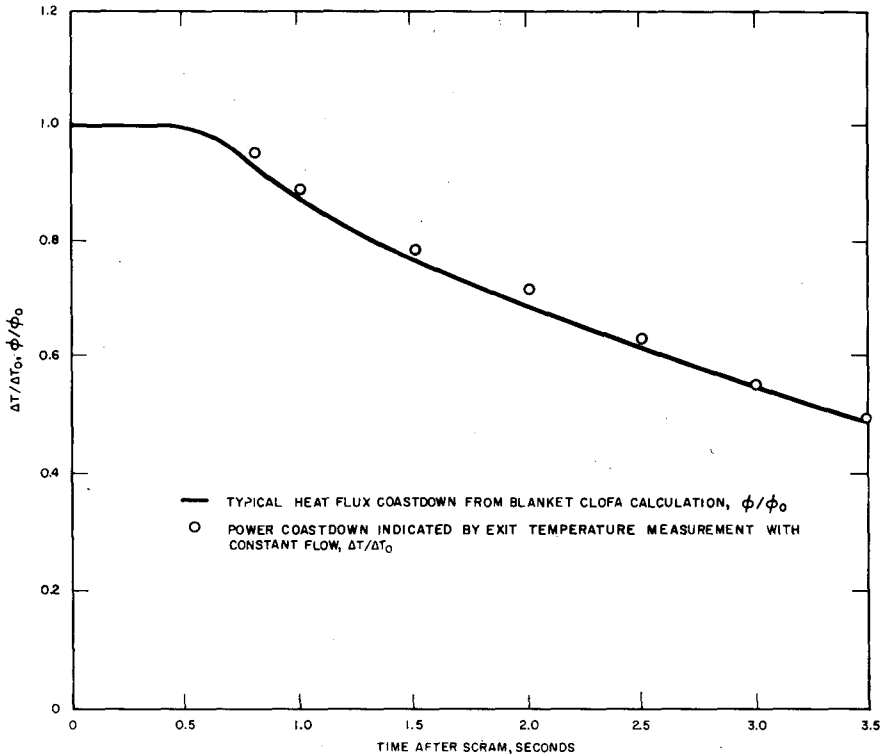


Fig. 22

Temperature coastdown in blanket following reactor scram (Seed 3)

cladding defect. The assembly was removed, even though it did not cause any plant activity problems, to determine the nature of the defect. A cladding penetration was found on one of the 840 fuel rods in the fuel assembly [10]. This defect was traced to a transverse tubing crack originating at the inside cladding surface which occurred during fabrication. The fact that the defect was present since start-up as determined by an examination of oxide film build-up, plus the fact that no significant fuel was lost from the rod, confirms the validity of the high corrosion resistance requirement which was established when selecting UO_2 as the fuel material for the PWR blanket.

Control-rod performance

Examinations have been carried out on samples of the hafnium control rods removed after each seed life. Fig. 24 shows the change in mechanical properties of hafnium as a result of exposure to a neutron flux during the life of the first two seeds [11]. The results indicate that the strength increase of hafnium with irradiation follows a two-step process, with an apparent saturation between 3.8 and 5.6×10^{21} n/cm² and an additional increase in strength at higher exposure, whereas the useful ductility decreases slightly.

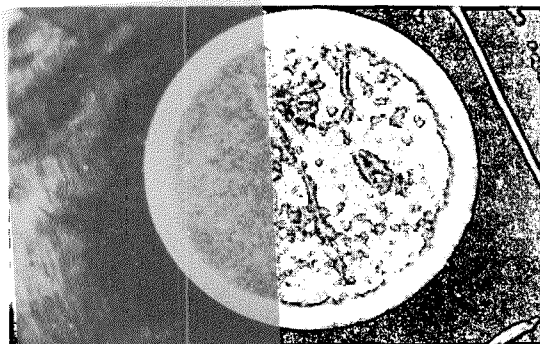
CORE 1 BLANKET FUEL ROD CROSS
SECTION AFTER TWO SEED LIVES

Fig. 23

Core-1 blanket microstructure after Seed-2 operation

Top

Transverse view of ground and etched UO_2 fuel pellet
from a highly depleted rod in PWR blanket after two seed lives
(4×10^{20} fissions/cm³; 16 100 MWd/t of uranium)

Bottom

Typical microstructure of UO_2
from a highly depleted rod in PWR blanket after two seed lives
(4×10^{20} fissions/cm³; 16 100 MWd/t of uranium)

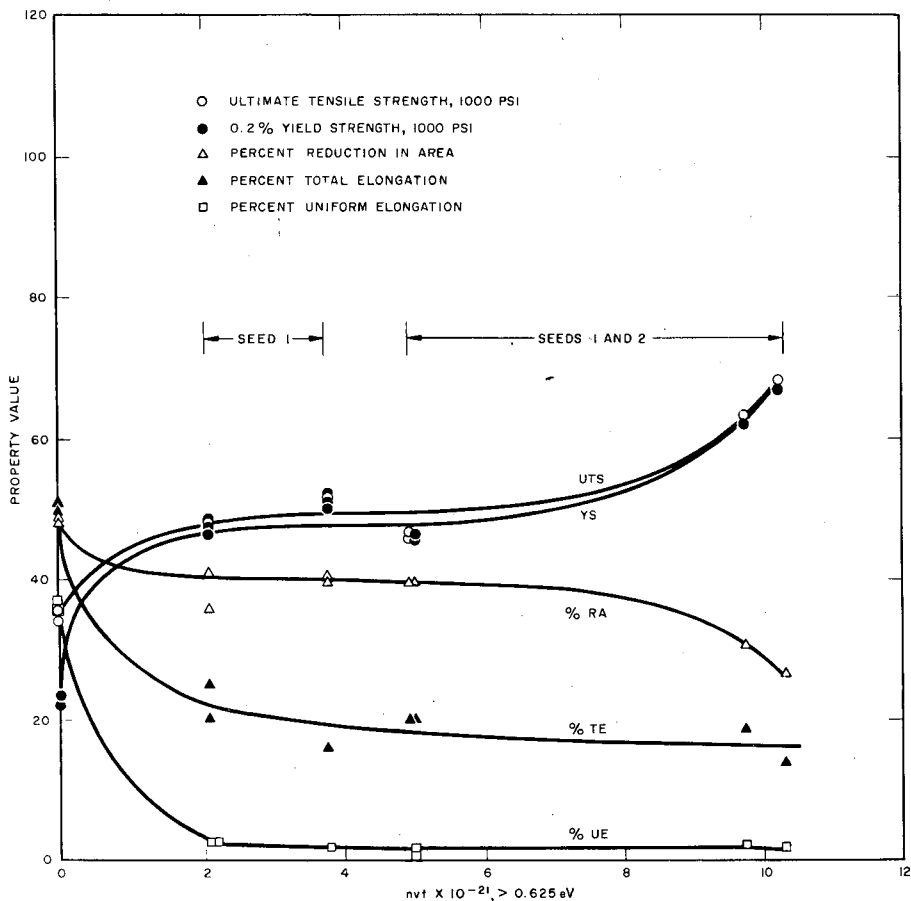


Fig. 24

Effect of neutron exposure on the 600°F tensile properties
of PWR hafnium control-rod after two seed lives

Also, isotopic examination of one of the control rods removed after Seed-1 operation revealed: (1) that only a minor alteration of the isotopic content (as compared to that of natural hafnium) had occurred; and (2) that the rod had lost less than 9% of its absorptive capacity. These results are consistent with measurements at Shippingport which indicate that the hafnium control rods used for three seed lives have retained their original reactivity worth.

Examination of the control-rod weld joint between the hafnium blades and their Zircaloy-2 extensions during Seed-1 refuelling indicated excessive corrosion. During the manufacture of these rods, repairs had been made to the weld joint because of difficulties encountered in welding hafnium to Zircaloy-2. In these repairs the weld metal became contaminated, which accounts for the observed deficiency in corrosion resistance.

As a result of this experience, improved Zircaloy-hafnium welding procedures were developed and used in the manufacture of a new set of control

rods which were installed with Seed 3. However, two of the original rods showing low corrosion at the joint were retained for operation during Seed 3 because of the interest in determining property changes in hafnium after extended in-pile exposure.

Reactor mechanical and structural components

The operation of the core structure, reactor vessel closure, control drive mechanisms, and the fuel assembly extraction tool has been very satisfactory. There have been no malfunctions of these components, except for the replacement of two mechanism stators because of low-resistance electrical insulation. Mating parts have exhibited no significant distortion, galling, or crevice corrosion. Examination of control-rod drive mechanisms components and scram shafts has revealed no evidence of excessive wear or other symptoms of adverse effects caused by service to date. The integrity of the numerous seal welds on the reactor vessel closure during operation was very satisfactory.

Because of a special concern with 17-4 pH stainless steel in reactor applications, a general review of this material was conducted by the Commission. Accordingly, a detailed investigation of the 17-4 pH stainless steel used in the Shippingport reactor for two seed lifetimes was undertaken. This material was chosen because of its high strength and excellent wear characteristics, with due regard to its susceptibility to stress-corrosion cracking. Two types of heat treatments were given to the 17-4 pH material: 4 h at 1100°F for scram shafts and 2½ h at 875°F for lead screws. New tests were instituted on scram shaft components made of 17-4 pH stainless steel which had been in service in Core 1 throughout Seed-1 lifetime.

As a result of these reviews, it was concluded that the 17-4 pH components were suitable for continued operation at Shippingport. This conclusion was based on the carefully controlled procedures used in the manufacture of these components and on the following considerations:

(1) At operating temperature, the minimum safety factor on the maximum working tensile stress in the assembly is 2, after application of a conservative stress concentration factor.

(2) The residual tensile stresses measured in the irradiated components were less than 30 000 lb/in². Samples of the material removed from the reactor were subjected to stress corrosion tests for 1000 h at stress levels up to 60 000 lb/in² and indicated no tendency toward stress-corrosion cracking.

(3) Irradiated tie rods, lead screws and spline shafts were impact-tested to produce nominal stresses of twice the value developed in operational full-height scrams without inducing defects detectable by liquid penetrant or magnetic particle inspections.

(4) Destructive metallurgical evaluation of material from the lead screws that were in the reactor indicated no change in the tensile properties compared to materials that had not been in the reactor. The hardness had increased from R_C 43 to about R_C 47. However, this had no adverse effect on the performance of the material as indicated by the engineering test results.

(5) A complete inspection of all lead screws was made during the Seed 2-Seed 3 refuelling. Dye check inspection disclosed a small defect in only one

lead screw, which was replaced and examined. It was determined that it was not a service-induced consequence. This inspection was repeated during the third refuelling and no defects were noted.

(6) No permanent damage to the core would result from failure of a scram shaft assembly.

Pressure-vessel steels have been found to suffer a loss in ductility, reflected in an increase in the ductile-brittle transition temperature as a function of the integrated fast neutron flux dosage. It is concluded, on the basis of present materials data, that the magnitude of this increase in transition temperature for the Shippingport reactor vessel does not limit operation at normal operating temperatures and will not do so throughout Core-2 life. However, at temperatures below 250°F, the increase in transition temperature as a result of fast-neutron dosage does place a limit on the permissible stress levels in the reactor vessel. Accordingly, to ensure vessel integrity, operating procedures have been established and are rigorously followed to control the pressure-temperature relationships during all plant start-ups and periodic hydrostatic testing. These relationships are so specified that the resultant stresses are well below the allowable values, if the irradiation-induced effects on the transition temperature of the material are taken into account.

Reactor protection system

The operational history of the reactor protection system indicates that the system has provided dependable protection without excessive maintenance or component failures. Component problems were largely confined to the bistable magnetic amplifiers, which have had a tendency to drift and required frequent adjustment and alignment. Modifications to the alignment procedures have been developed to reduce this problem during Seed-3 operation.

Two valid safety shut-downs occurred during Seed-2 operation. One was due to a low system pressure brought on by an operator error and the second was due to high coolant temperature during a test requiring rapid station shut-down. A number of other rod insertions and shut-downs occurred, but they were the result of component problems and operator errors. The safety shut-down set points of 114-118% of rated power have not proved to be restrictive.

Steam generators

Early in the operation of Shippingport, caustic stress corrosion caused failure of a few tubes and a leak rate of 35 gal/h in one of the steam generators. This defect was caused by a steam pocket forming on the secondary side as a result of insufficient circulation. The installation of two additional risers corrected the condition, and, to ensure a complete absence of "free" alkalinity in the boiler water, mono-, di- and tri-sodium phosphate were used for pH phosphate control. The phosphate concentrations have been 100 to 300 ppm and pH values between 10.6 and 11.0 have been maintained.

Improved control of boiler water chemistry was found to be necessary during periods of hot stand-by and cold lay-up conditions. This control was

accomplished by using an auxiliary heat exchanger, instead of the secondary steam system, as a heat sink for cooling down the plant.

Higher radiation levels have been experienced in the heat exchanger area than in the area around the main coolant piping. Tests on the inlet end of the heat exchanger having the highest level (600 mr/h) showed that the tube sheet, rather than the tubes, is the major contributor. During maintenance in the vicinity of the inlet to the heat exchanger, sheets of lead have been used to reduce exposure to maintenance personnel.

Main coolant pumps

The performance and reliability of the main coolant pumps during four years of Shippingport operation have been generally good. There have been two incidents of failures on the same pump design and one incident of start-up failure on a different design. All of these, by coincidence, occurred in the same loop but there is no apparent relation between them. In contrast, no failures or difficulties have been experienced with the other three pumps in the remaining three loops.

The first incident involved collapse of the can separating the stator coils from the reactor coolant, apparently as a result of operation at low primary-system pressure or a pinhole in the can, either of which could have produced a high differential pressure across the can. Since the time that this incident occurred, the minimum system pressure during pump operation has been increased from 100 to 300 lb/in² gauge to reduce the possibility of such failures in the event of low-pressure operations. Because of their satisfactory operation, no modifications were made to the original pumps in the other three loops.

The second incident involved a failure of the upper and lower radial bearings due to overheating of journal wear surfaces because of restrictions occurring in the internal cooling system of the pump. These restrictions are believed to have been caused by a mixture of free hydrogen and steam entering the pump cooling system via a common vent header to which the pressurizer was the major contributor. By rerouting the pressurizer vent line to the blowdown tank, this condition was corrected.

The third incident involved start-up failure of a pump of different design which was installed after failure of the second pump. The incident occurred after a period of hot stand-by service. It was believed to be the result of inadequate heat removal around the pump shaft during stand-by. Thus, a sufficient temperature differential developed to cause distortion and binding of the shaft and impeller. Attempts to correct the condition were unsuccessful. Accordingly, this pump was replaced by another pump, similar in design to that in the other three operating loops during the Seed 2 - Seed 3 refuelling.

Reactor coolant chemistry control

The reactor coolant contains demineralized water adjusted to a pH of 9.5 to 10.5 with lithium hydroxide. Hydrogen concentrations are specified as 15 to 60 cm³/kg H₂O and oxygen at less than 0.14 ppm. With these coolant conditions, corrosion of the reactor plant piping and components has been

minimized. The crud concentrations have been consistently low (about 5 pp 10³). The control of the pH by the ion exchanger has been excellent with infrequent additions of lithium hydroxide of less than 0.5 lb every 4 to 5 weeks. Hydrogen additions of approximately 100 ft³ per week have been necessary because of pressurizer relief valve leakage. This addition rate has recently been reduced as a result of repairs to the relief valves. No excursions of oxygen concentrations above 0.14 ppm have occurred during power operation.

Main turbine generator

The governor valve and the interstage moisture separator of the main turbine generator were replaced after successive repairs failed to eliminate deficiencies.

Failure of the No. 3 governor-valve stems prompted the replacement of all the governor valves by redesigned valves at an appropriate station outage during Seed-1 operation. These new valves have performed satisfactorily to date.

The turbine interstage moisture separator installed is a cyclone-type unit. The internals of this separator were found to have completely disintegrated when inspected early during Seed-1 operation. The cause and time of this failure could not be determined. Repairs to this separator resulted in a station outage of approximately two months, during which time an inspection of the turbine was also made. The turbine was found to be in a satisfactory condition. After the repairs, further difficulties with the separator were experienced, and it was concluded that the internals should be completely redesigned. The new design separator was installed during Seed 1 - Seed 2 refuelling and its performance to date has been satisfactory. It was inspected during Seed 2 - Seed 3 refuelling and no deficiencies were found.

Feedwater heaters

The turbine plant has been plagued with feedwater heater-tube leakage since October 1959. During the Seed 1 - Seed 2 refuelling shut-down, defective tubes were plugged in one of the three units and it was returned to service. After two and a half months of operation, severe leakage again developed and the unit was removed for retubing. The second feedwater heater continued to require tube plugging, necessitating frequent shut-downs. Finally, in July 1961 when the leakage became excessive, this unit was also removed. It was retubed and returned to service with Seed 3.

Lack of stress relieving at the tube bends and inadequate venting were identified as the principal causes of the feedwater heater-tube failures. These deficiencies have been corrected and stress-relieved Admiralty - Admiralty tubes in lieu of the original Admiralty tubes have been installed.

4. CURRENT DEVELOPMENTS AND FUTURE PLANS

The principal objective of the current and future programme of the Shippingport Project is to advance the basic technology of water-cooled re-

actors to achieve significant reductions in both capital costs and fuel costs. Advances in core technology in terms of lifetime capability, power capability and fuel-element fabrication techniques will have the most telling effect on reducing costs, for they will influence directly not only the fuel cost but also the plant capital cost per unit of heat removed. The latter derives from the fact that cores with high specific power capability will impose a lower burden on the plant per unit of heat removed from the core. The reduction in fuel cost derivable from increases in core lifetime or energy output, especially when combined with lower core manufacturing cost, is obvious.

Accordingly, the Shippingport programme is now primarily concentrated on core development and consists of two parts: (1) continued operation of the Shippingport Plant with the fourth seed in Core 1; and (2) the development, design, manufacture, and test operation of a long-life, high-power density, second seed and blanket core - PWR Core 2.

The continued operation of Core 1 with its first four seeds has been directed towards:

(1) The large scale irradiation of natural UO_2 fuel elements with Zircaloy-2 cladding to determine the maximum depletion that these fuel elements can withstand.

(2) The determination of the nuclear and thermal performance of seed and blanket cores as a function of fuel depletion, and the demonstration of the continued generation of a high fraction of power from natural uranium in such cores.

(3) The accumulation of further operating data and experience on major plant systems and components including especially the reactor core, thus providing information on the long-term performance characteristics needed to evaluate the adequacy of current component designs for service in water-cooled reactor plants now under construction or in early stages of operation.

The development of Core 2 represents an important step towards reducing the cost of nuclear power because at its design rating of 150 MW(e) gross and a first seed life of 10 000 h, Core 2 will have five and one-half times the energy output and twice the power density of Core 1.

High total energy, high-power density cores such as Core 2 require fuel elements which can withstand long irradiation exposure and operate at high heat fluxes. One of the major features of the Core-2 development programme is the effort directed toward producing a compartmented plate fuel element containing bulk UO_2 coupled with the development of Zircaloy-4 as a long-life cladding material. As a result of the Core-1 and Core-2 programmes, bulk UO_2 has been shown to have the greatest potential for stability and corrosion resistance under irradiation of any presently known fuel material for water-cooled reactor application. Application of this material in plate form to take advantage of greater heat-transfer area and lower fuel temperatures at high heatflux is the next logical step in UO_2 fuel element development.

Seed 4 programme

Full power operation at Shippingport with the fourth seed, which is expected to have a lifetime of about 6900 EFPH, began on 30 January 1963. It is, therefore, anticipated that this seed will be depleted early in 1964.

The results of the preliminary examination of sample blanket fuel elements at the end of three seed lives of operation give every indication that the Core-1 blanket will perform satisfactorily through additional seeds; however, it has been decided to remove Core 1 after Seed-4 operation to allow full-power testing of Core 2 at the earliest possible time.

The average depletion of the blanket fuel at the end of Seed-3 operations was 8450 MWd/t of uranium with a peak value of 28 600 MWd/t. At the end of Seed-4 life, the average depletion is expected to reach 10 860 MWd/t of uranium and the peak to reach 36 600 MWd/t.

An experimental fuel sub-assembly, which was representative of an early PWR Core-2 Seed-1 sub-assembly design, was inserted in the centre of the Core-1 blanket during Seed-3 installation. A second experimental fuel sub-assembly, representative of the final Core-2 Seed-1 sub-assembly design, was subsequently manufactured. Both of the sub-assemblies were made up of UO_2 - ZrO_2 fuel wafers in compartmented plate elements approximately 2-ft long. The reduced length makes it possible to test such a fuel assembly in the blanket of Core 1 without exceeding fuel thermal design limits. A third experimental fuel assembly was fabricated utilizing oxide rod configurations representing an advanced blanket design concept. During Seed-4 installation, the advanced blanket design was placed in the centre of the core and the two-seed fuel sub-assemblies were placed in adjacent, in-line, positions.

An interim inspection of the experimental fuel assembly which was tested during Seed-3 operation to a peak fuel depletion of 7.8×10^{20} fissions/cm³ of compartment volume was completed during refuelling. This inspection identified satisfactory design performance, and no observable dimensional changes were observed or noted. A view of a typical experimental fuel assembly is shown in Fig.25.

Core-2 design

Core 2 entails major developments in reactor physics and heat transfer to provide the basic data, analytical methods, correlations, and digital computer codes necessary to the design of high performance cores. Long reactivity lifetimes, high fuel densities, reduced thermal margins and operation in or near the boiling region, all combine to require an order of magnitude improvement in the detail and accuracy with which the nuclear and thermal characteristics are calculated. The determination of the power distribution in three dimensions and especially of the reactivity as a function of fuel depletion assume major importance in this design.

From a design standpoint, there were two key determinations in the development of Core 2: selection of the seed-blanket type core and selection of plate-type fuel elements. The principal considerations leading to retention of the seed-blanket concept for Core 2 as well as for Core 1 were:

- (1) The continued intrinsic worth and importance of obtaining a large fraction of the total energy from natural uranium in a light-water-reactor nuclear power station.
- (2) The achievement of a very long life core without the heavy inventory of enriched fuel that would otherwise be required.

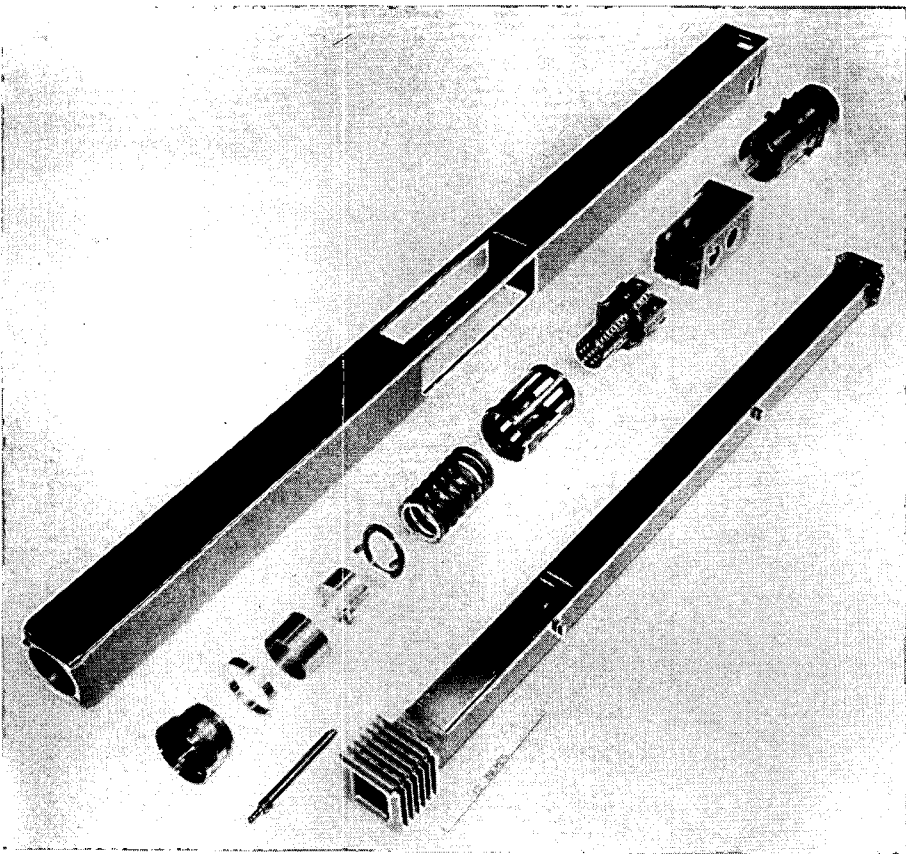


Fig. 25

Experimental fuel assembly tested during Seed-3 operation

(3) The demonstration during five years of operation in the Shippingport Station that the seed-blanket core concept possesses the performance characteristics with respect to dynamic load response, operating stability and simplicity of control that make the plant well suited to integration with an electric utility system.

(4) The reduction in the number of movable control elements required in a large, high-power, long-life core inherently afforded by using a relatively small, highly enriched seed as the driving element.

(5) The absence of any clear-cut technical or economic reasons for changing to a different core type.

In selecting the plate-type fuel element geometry for Core 2, special attention was given to the excellent performance of the rod-type UO_2 fuel elements in the Core-1 blanket, to the fact that rod elements are better able to withstand the internal pressures generated by fission gases and to the probability that rod elements are more suited to low-cost fabrication. Nevertheless, it was necessary to recognize that UO_2 rod elements of prac-

tical dimensions possess a number of undesirable properties, namely: very high temperature of the fuel, limited heat-flux capability, poor transient thermal response, limited ability to accommodate fuel-volume changes by clad deformation, and poor heat-transfer surface per unit core volume. These limitations of rod elements weighed heavily against their choice for meeting the high total energy, high-power density requirements of Core 2. In contrast, the superior heat-transfer properties of plate-type fuel elements make them inherently well-suited to these requirements. This is particularly true since the long-term irradiation stability and fission gas release rate of UO_2 are strongly sensitive to its temperature.

The choice, on the above basis, of plate-type fuel elements for Core 2 should not be interpreted to mean that UO_2 rod elements do not have any place in future water reactors. Rather, it means that the basic technology of both these fuel elements will have been developed by the total Core-1 and Core-2 programmes at Shippingport. As a result, an added dimension will be provided to the nuclear industry for optimizing future water-reactor design.

Core description

As shown in the cross-section view of Fig. 26, Core 2 is a seed-blanket core with an annular seed made up of 20 highly enriched fuel clusters, each

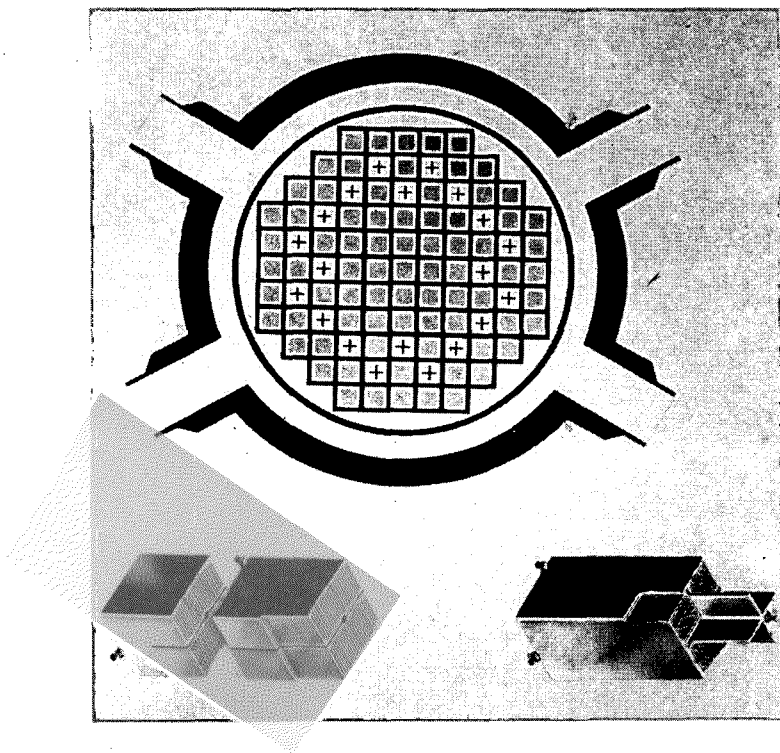


Fig. 26

PWR-2 core cross-section with cutaway seed and blanket

of which also contains a cruciform-shaped hafnium control-rod. A blanket of natural uranium consisting of a total of 77 natural-uranium fuel clusters surrounds the seed on both sides. The fuel clusters are arranged to form an approximate right-circular cylinder with an active fuel length of 8 ft and a mean core diameter of about 7 ft. This is illustrated in Fig. 27 which also shows a longitudinal section of the core and the principal structural components. The fuel clusters are positioned and supported within the core cage as shown in Fig. 28. They are held down against the bottom support plate by the twenty control-rod shrouds and a centre-support tube. In contrast to the Core-1 design, no top grid or fuel assembly latches are required. Individual fuel clusters can be removed without disturbing the remaining clusters

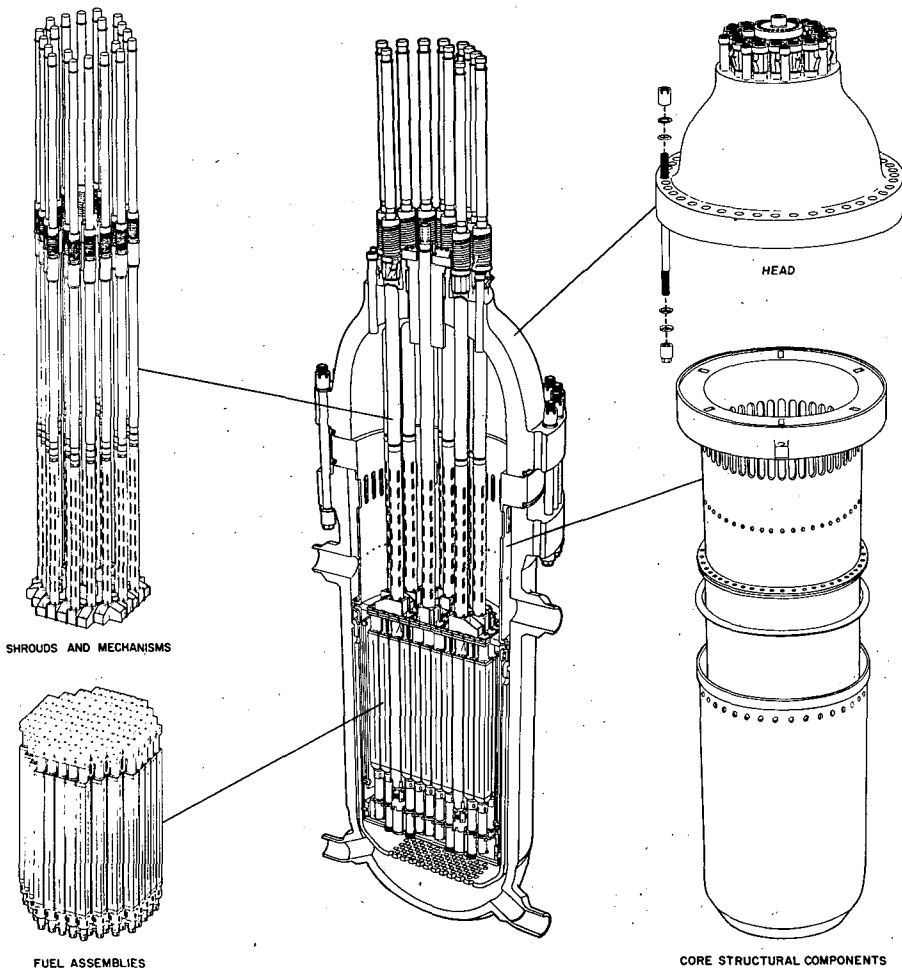


Fig. 27

Reactor component parts (Core 2)

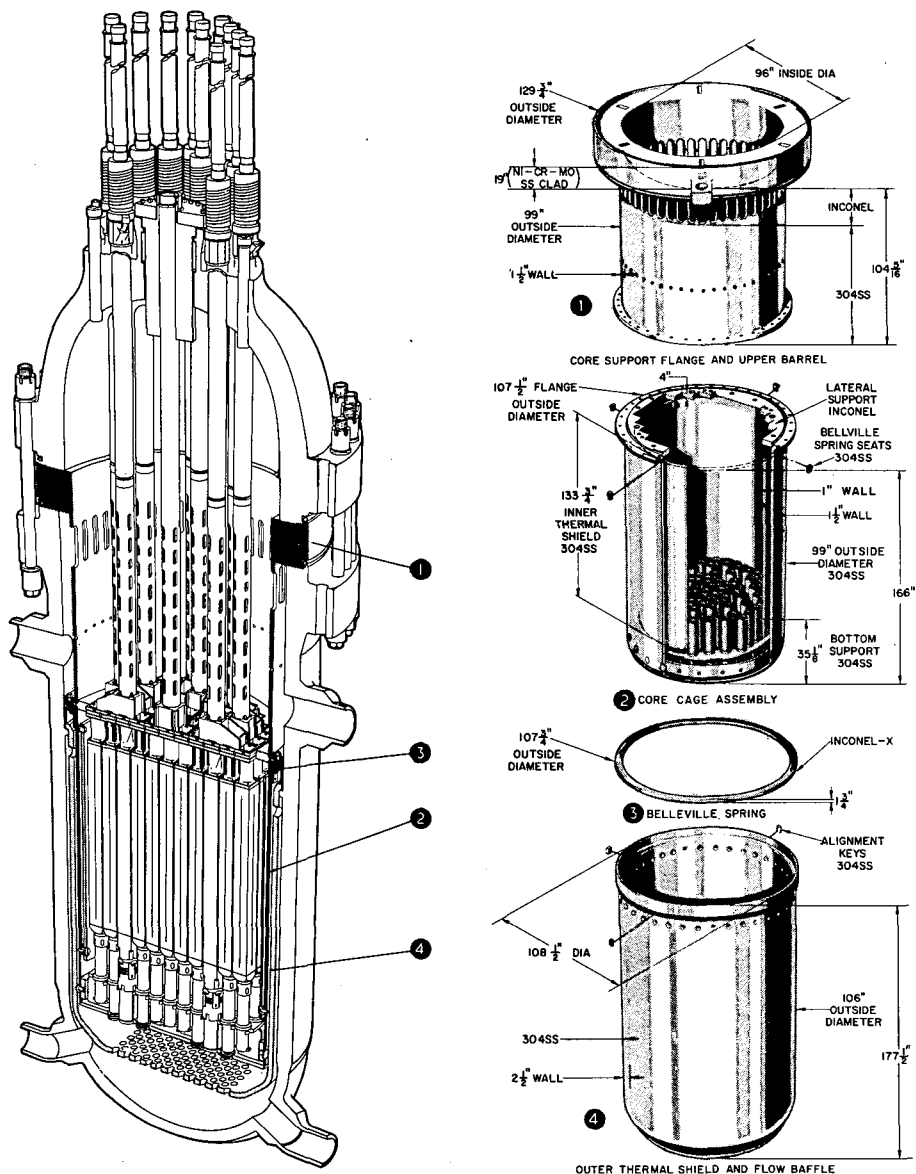


Fig. 28

Reactor structural components (Core 2)

through a central port in the reactor-vessel closure. The core cage assembly is shown in Fig.29.

Each blanket cluster is 7-3/8 in on a side and consists of a weldment of two sub-assemblies made up of compartmented plate fuel elements 3.64-in wide containing UO_2 fuel wafers 0.100-in thick clad with Zircaloy-4. Fig.30

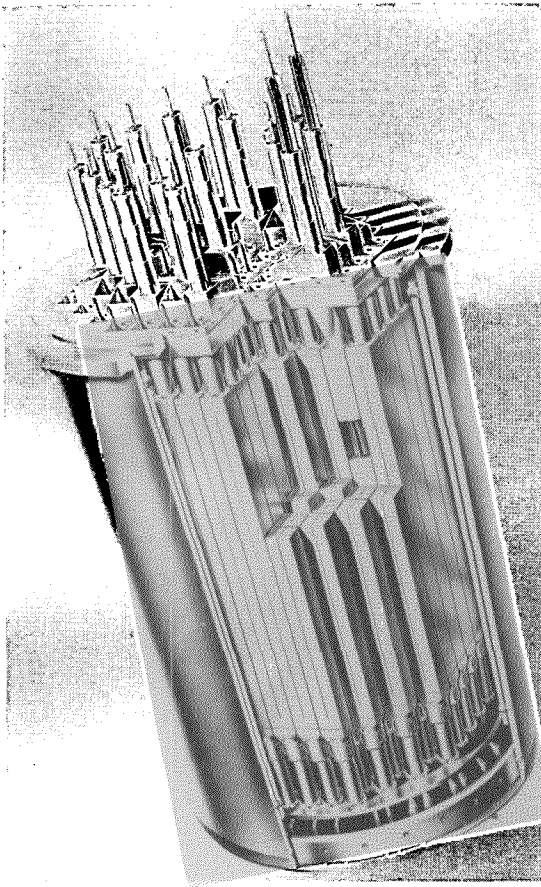


Fig. 29

PWR-2 Core-2 cage assembly

shows a section of the cluster and Fig.31 is a cutaway view of a blanket fuel plate.

Each seed cluster is $7\frac{3}{8}$ in on a side and consists of a weldment of four sub-assemblies arranged as shown in Fig.32 to form a cruciform-shaped control-rod channel. The seed fuel elements are compartmented plates similar to those in the blanket except that they are 3.44-in wide and contain enriched $\text{UO}_2 - \text{ZrO}_2$ wafers 0.036-in thick. Three different wafers with respect to weight percent of uranium are used in the seed plates, and these are arranged to provide three zones of fuel as shown in Fig.33. Lumped burnable poison wafers, B-10 in stainless-steel matrix, are located in each corner of the seed sub-assemblies to reduce the reactivity swing with fuel depletion and thereby the control-rod requirement; they also reduce the seed power peaking adjacent to the control-rod channel.

Both the seed and blanket oxide fuel plates are compartmented by internal ribs transversely and longitudinally to strengthen the cladding structure.

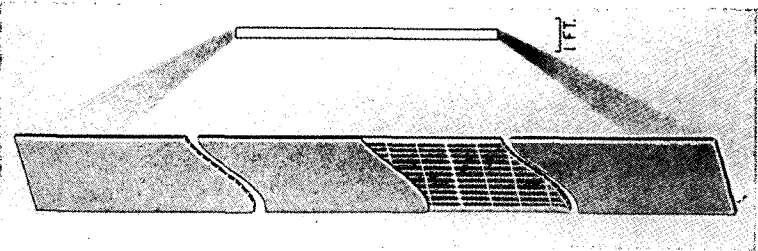


Fig. 31
Core-2 blanket oxide fuel plates

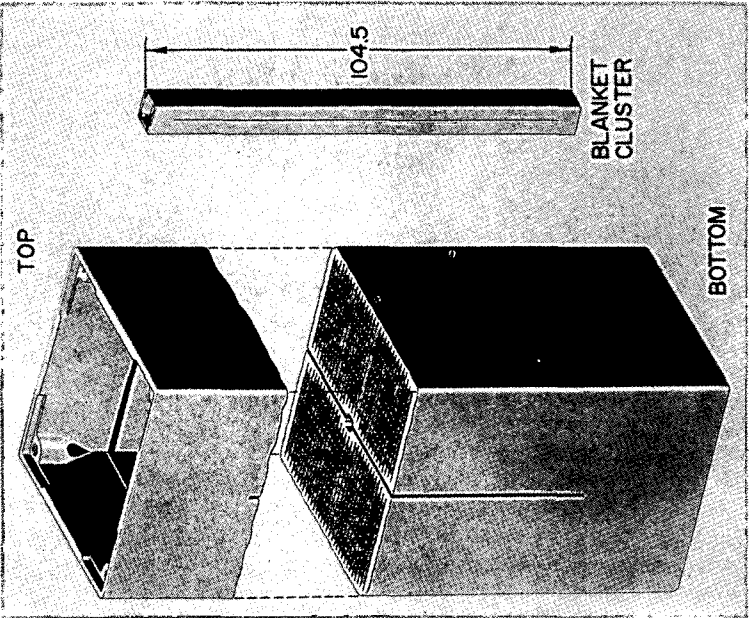


Fig. 30
Core-2 blanket fuel cluster

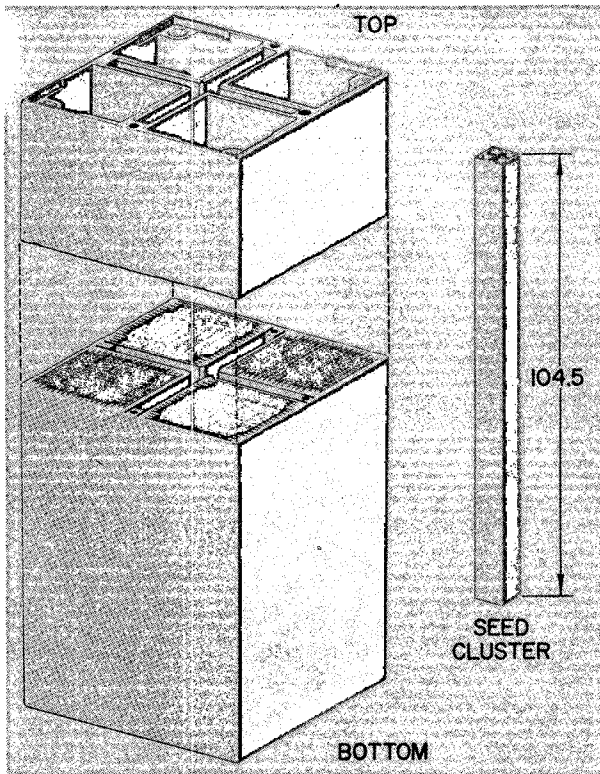


Fig. 32

Core-2 seed fuel cluster

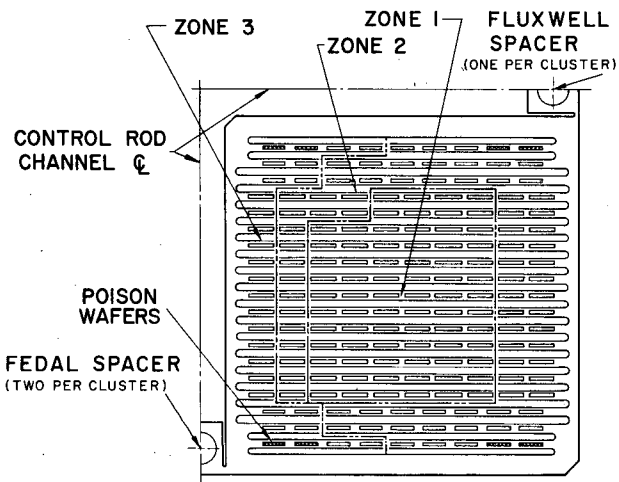


Fig. 33

Core-2 Seed-1 sub-assembly cross-section

To prevent waterlogging and bulging of the cladding sufficient to touch the adjacent plate, the longitudinal ribs are spaced 0.25 in apart. To limit the amount of fuel which can be leached into the coolant in case of fuel compartment failure and to assure rigidity and resistance of the plates to hydraulic forces, the spacing of the transverse ribs and/or shims has been set at about 1 in. The clad thickness of 0.020 in has been established based on plate strength, degree of fabrication control and corrosion requirements.

The location and type of instrumentation planned for Core 2 are shown in Fig.34. This instrumentation is provided to obtain data and is not required for operation. It consists of 62 thermocouples for measuring coolant temperature at cluster inlet and exit, 31 fuel-clusterflow-measuring devices, six flux wells, and a system for sampling coolant from the exit of each of the 97 seed and blanket clusters to detect and locate any fuel-element failures. To minimize interference with refuelling operations, more than 80% of the instrument leads are routed through the core support flange below the region from which refuelling operations will be conducted. The remainder of the instrumentation leads exit from the reactor vessel through the control-rod shroud and the closure head.

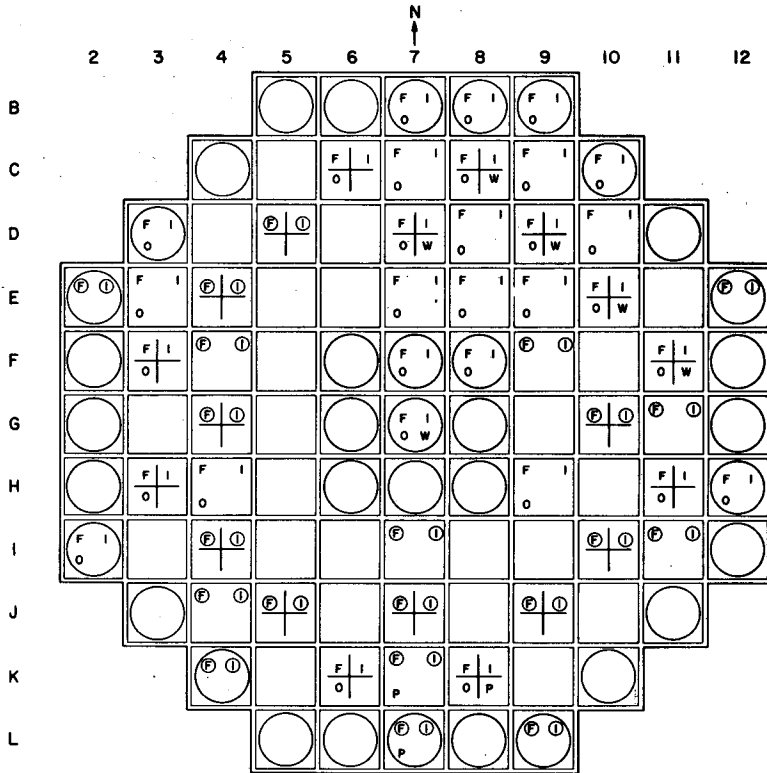
Most of the refuelling facilities provided for Core 1 will also be used for Core 2. The chief exception is a fuel-cluster installation and extraction tool of new design suitable for use through the central port in the reactor-vessel closure. As in the case of Core 1, the refuelling equipment for Core 2 will permit removal or installation of seed and blanket fuel assemblies, individually, under water through the closure head or with the closure head removed.

Except for the seed-blanket core concept, the design features of Core 2 represent a considerable departure from Core 1. Some of the more significant differences between the two core designs may be noted in Fig.35 and from the comparison given in Table V.

Nuclear design

Because of the long lifetime and high-power density requirements of Core 2, the core reactivity behaviour, power distribution and control characteristics must be determined with a very high degree of precision and geometrical detail. This has required an extensive computational effort and experimental physics programme and the development of new analytic models and digital computer programmes [12]. For example, adequate treatment of the detailed fuel and poison distribution as a function of depletion has required as many as 15 million calculational mesh points of the core.

Each increment of core depletion originates from the condition before that increment. As a result, the beginning-of-life condition of Core 2 needed to be accurately determined and verified by experiment to provide a basis for depletion studies as well as for setting the core loading. To this end, critical mock-ups were tested at room temperature, operating temperature and intermediate temperatures. The elevated temperature tests were conducted in a pressurized hot critical facility. Part of the mock-up fuel duplicated the nuclear properties of Core-2 design with high precision. The flexibility of the mock-up afforded a wide variety of critical-assembly configur-



ONE PASS SEED-TWO PASS BLANKET TYPE CORE

97 SQUARE MODULES SPACED 7-1/2 INCHES ON SQUARE ARRAY

- ⊕ SEED ASSEMBLY (20)
 □ FIRST PASS BLANKET ASSEMBLY (40)
 ○ SECOND PASS BLANKET ASSEMBLY (37)

SYMBOL AND TYPE OF INSTRUMENTATION	LOCATION			TOTAL	TOTAL SPARES
	SEED	BLANKET			
		FIRST PASS	SECOND PASS		
W-FLUX WELL	5	0	1	6	
I- INLET WATER THERMOCOUPLE	11 (9)	10 (7)	10 (5)	31	(21)
O-OUTLET WATER THERMOCOUPLE	11	10	10	31	
F- COOLANT FLOW (FMI)	11 (9)	10 (7)	10 (5)	31	(21)
P- REGIONAL PRESSURE DROP	1	1	1	3	
- FAILED ELEMENT DETECTION AND LOCATION (FEDAL)- ALL ASSEMBLIES	20	40	37	97	

NOTE-CIRCLED CHARACTERS DENOTE SPARE INSTRUMENTATION LOCATIONS AND QUANTITIES.

Fig. 34

Core-2 instrumentation

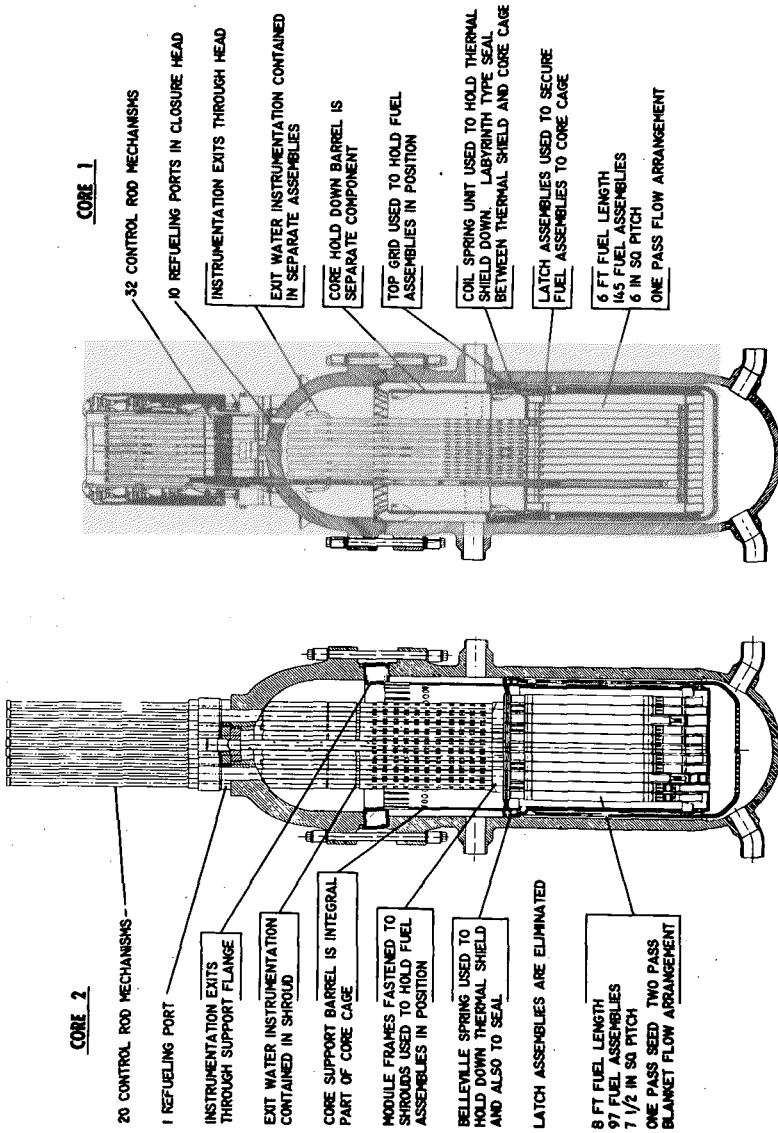


Fig. 35

Comparison of Core 1 with Core 2

TABLE V

COMPARISON OF CORE-1 AND CORE-2
DESIGN FEATURES

	Core 1	Core 2
Rating	231 MW(t) ; 67 MW(e) with 3 loops	505 MW(t) ; 150 MW(e) with 4 loops
Power density		
Seed	75 kW/l	157 kW/l
Blanket	25 kW/l	33 kW/l
Design core lifetime		
Seed	3000 EFPH	10 000 EFPH
Blanket	8000 EFPH	20 000 EFPH
Fuel elements		
Seed	Zircaloy-2 clad enriched metal alloy U-Zr plates	Zircaloy-4 clad enriched UO ₂ -ZrO ₂ compartmented plates
Blanket	Zircaloy-2 clad UO ₂ rods	Zircaloy-4 clad UO ₂ compartmented plates
Core pass		
Seed	1	1
Blanket	1	2
Core height	6 ft	8 ft
Nuclear control	32 rods and distributed burnable poison in the seed (Seeds 2, 3 and 4)	20 rods and lumped burn- able poison in the seed
Orifices	Fixed	Adjustable
No. of ports in reactor vessel head (for refuelling)	Ten	One

ations with different seed loadings and boron content to simulate the lifetime behaviour. Detailed measurements of excess reactivities, gross power distributions etc., were made with good consistency.

To provide the lifetime control for Core 2 by means of control rods alone would have required such a large number that the mechanical design would have become impractical. The use of a self-shielded lumped burnable poison was therefore incorporated into the design so that close match of poison depletion and fuel depletion could be achieved throughout seed-life. The calculated core reactivity at operating conditions is shown in Fig.36. The excess reactivity will increase by a maximum of $\sim 1.5\% \Delta k/k$ over the beginning-of-life value due primarily to build-up of plutonium in the blanket during this period. From this period on, the blanket reactivity decreases due to build-up of fission products.

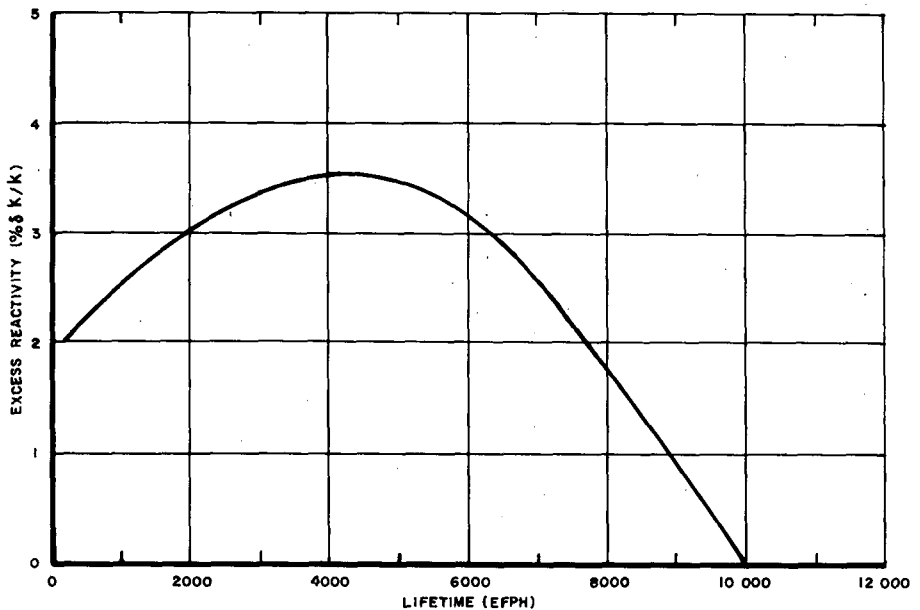


Fig. 36

Core-2 reactivity versus life (Seed 1)

The high-power densities and long life of Core 2 result in high uranium loading densities which aggravate the power peaks associated with discontinuities in the core lattice such as rod channels and seed and blanket cluster boundaries. Reductions in these power peaks were therefore necessary and have been achieved by providing three zones of fuel concentration within each seed sub-assembly. In addition, lumped burnable poison has been located in the corners of each sub-assembly to provide further significant reduction in the seed power peaking factors.

The axial distributions of the seed and blanket power generation using the TURBO-ZIP [13] code as computed on Philco S-2000 are shown in Figs. 37 and 38, respectively, for various times in core life. These axial distributions were selected for the hot channel at the core lifetimes indicated and are indicative of some of the core heat-removal problems.

The effect of fission products on long-term reactivity behaviour is being studied in a series of experiments in the Materials Test Reactor (MTR) and Engineering Test Reactor (ETR) to account for different fast to slow flux ratios. At various times during the irradiation, the test samples were removed and measurements of fuel and absorber content made in the Reactivity Measurement Facility (RMF) at MTR. In addition, long-term reactivity-gain experiments are being carried out to yield data on the fission product poison from plutonium and uranium.

The accumulation of stable fission products in Core 2 towards the end of Seed-1 life will amount to approximately 8% loss in reactivity. The seed fuel loading includes an amount to compensate for the effects of these poisons.

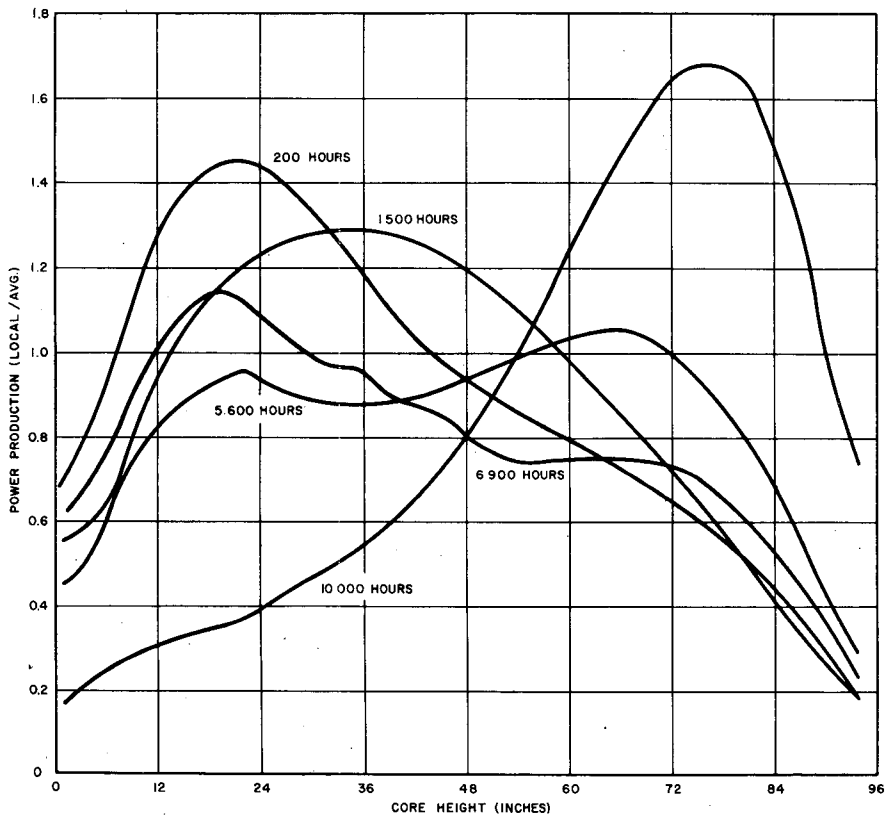


Fig. 37

Core-2 Seed-1 power distribution

In the blanket, the effects of the fission-product poison are not only balanced, but actually overcome by the plutonium production. During the life of the first seed, 45% of the blanket energy will be generated by plutonium produced and burned and 10% will be generated by fissioning of natural uranium. The balance of the blanket energy will be generated by fissioning of U^{235} of which only about one-half of the original content will remain at the end of Seed-1 life.

Thermal and mechanical design

The thermal design of Core 2 was developed on the basis that:

- (1) No departure from fuel element surface temperature associated with nucleate boiling should take place during normal or accidental transient conditions.
- (2) No flow oscillations should take place in the core under steady-state operating conditions. This restriction is necessary to prevent the occurrence of high-temperature cycling of the fuel, since the irradiation behaviour of uranium-oxide plate fuel elements at high depletions is sensitive to central fuel temperatures.

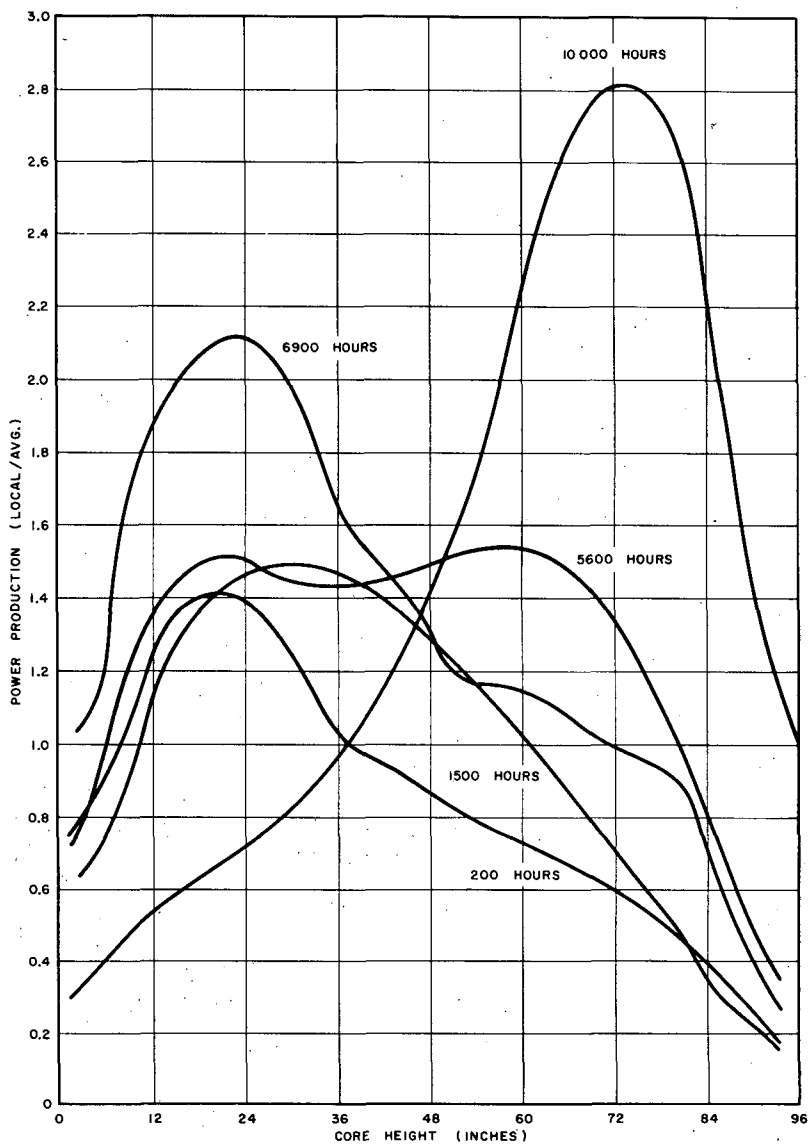


Fig. 38

Core-2 blanket power distribution

To study the limiting heat flux and coolant conditions applicable to Core 2, simulated fuel elements consisting of electrically-heated rectangular channels were inserted in high-pressure water test loops and the DNB (departure from the surface temperature associated with nucleate boiling) conditions determined over a range of design parameters. The tests covered a pressure range of 800 to 2000 lb/in² abs., coolant mass flow velocities from 0.2 to 5.0×10^6 lb/h ft²,

fluid enthalpies from about 500 BTU/lb to 1200 BTU/lb, channel lengths up to 96 in, and various axial heat-flux distributions. As a result of this work a new DNB correlation has been developed [14] in which mass flow plays a more important part than previously.

Similarly, an extensive test programme to measure pressure drops and quality in uniformly heated rectangular channels at 800 to 2000 lb/in² abs. in the local and bulk boiling regions was conducted which resulted in new methods for studying flow redistribution, flow oscillations and fast transients [15, 16, 17].

In the process of confirming the mechanical design, extensive reactor tests identified several unanticipated problems which are peculiar to the compartmented plate design. Each problem required analysis and in the final stages required minor redesigning to correct existing problems. This effort produced significant contributions to the basic engineering sciences. A good example is the phenomenon of fuel growth and the consequential weakening of the seed fuel plate which produces a tendency for the plates to bow when subjected to the hydraulic forces in the core.

In-pile irradiation tests showed that as the fuel material in Zircaloy compartments grew as a result of high depletion, the cladding became strained, causing a decrease in plate rigidity. Techniques were developed for analysing this loss in plate stiffness, including an experimental programme which showed the need for additional transverse ribs which would add sufficient rigidity to the plate to restrict plate bow at high fuel depletions. This change was incorporated in the design.

Photoelastic tests verified the calculated stress levels in both seed and blanket clusters when subjected to Core-2 mechanical loading conditions. A full-size fully instrumented carbon steel model of the core bottom support plate was also built and tested to ensure that the structure would meet design requirements on stress and deflection.

Tests of dummy fuel clusters with their associated control rods and support structures under conditions of flow, temperature and pressure, duplicating those expected in Core 2, have been satisfactorily completed. Additional tests on lead production components are planned.

Core materials development

Cladding

Experience with Zircaloy-2 cladding for Core-1 fuel elements has indicated that it is capable of extended service in high-temperature water with excellent corrosion resistance and mechanical stability. However, zirconium-base claddings are known to be capable of absorbing some hydrogen dissolved in the coolant or generated during the corrosion reaction between zirconium and water. Further, hydrogen in zirconium will migrate towards and concentrate in the relatively colder regions of the material. Fig.39 illustrates the effect of hydrogen migration towards the colder region of a sample plate fuel cladding where the dark etching rim denotes the precipitated hydride platelets. Experiments with Zircaloy-clad fuel elements have shown that the segregation and precipitation of zirconium hydrides can lead to sufficient embrittlement to cause clad cracking at average hydrogen content. To ensure that this does not occur, a limit of about 200 to 250 ppm in the cladding is

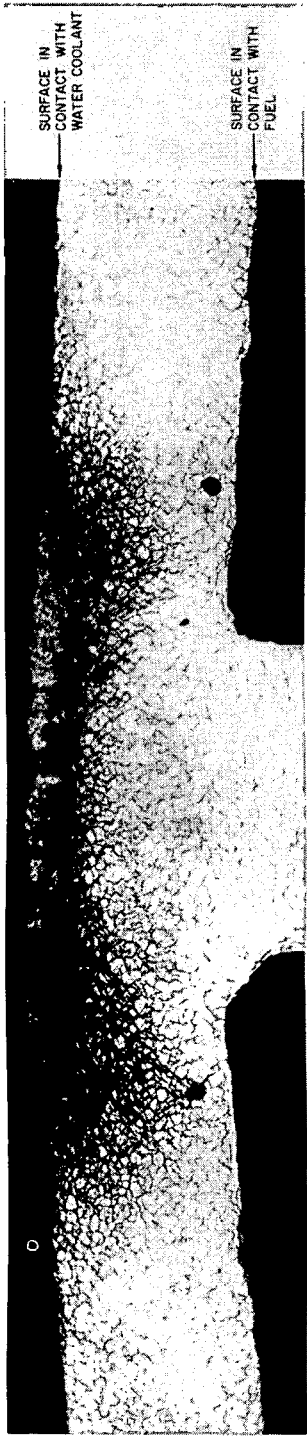


Fig. 39
Concentration of hydrides at cold sections of cladding (80x)
(Photograph shown half size)

maintained. Susceptibility to hydriding will probably prove to be the chief factor in determining the ultimate limits on the useful life of zirconium - base cladding in water-cooled reactors. Improvement in Zircaloy-2 with respect to hydriding has therefore been an important element of the Core-2 materials development effort.

This effort has led to development of a new alloy, Zircaloy-4, which is being used for fuel-element cladding in Core 2. Basically, Zircaloy-4 was derived by eliminating the 0.05% nickel addition specified for Zircaloy-2, thereby reducing the hydrogen pick-up during corrosion by a factor of three. This is illustrated in Fig.40 which shows that Zircaloy-4, in contrast to Zircaloy-2, is relatively insensitive to an increasing hydrogen content of the coolant.

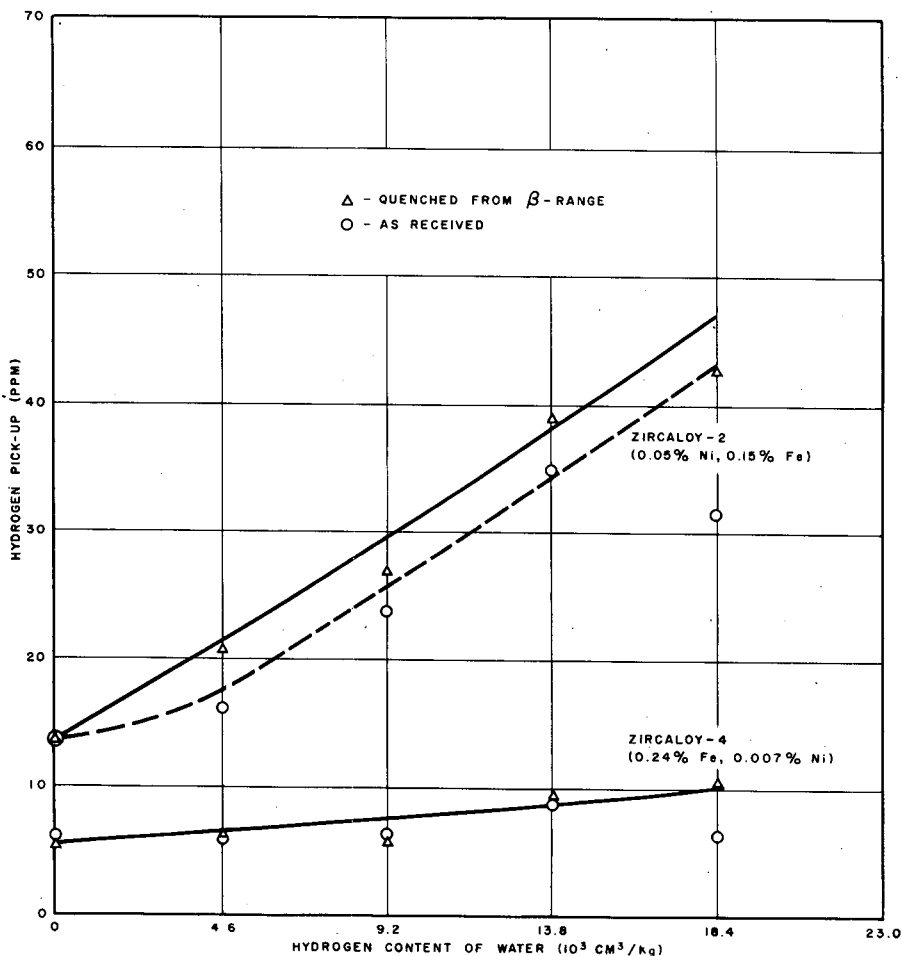


Fig. 40

Hydrogen absorption of Zircaloy-2 versus Zircaloy-4

Blanket fuel

As a result of the Shippingport programme, sintered UO_2 of high density has been demonstrated to possess high irradiation stability and corrosion resistance. The irradiation exposure of UO_2 on a large scale, as achieved by Core-1 operations to date, has produced no structural and property changes other than those induced by the operating temperature alone. However, the levels of irradiation reached in this way are still relatively low. Data from in-pile loop tests, on the other hand, do show that marked structural and dimensional changes can occur in UO_2 when exposed to high burn-ups. These tests were made on UO_2 wafers encased in plate cladding and exposed to fissioning depletions beyond 75 000 MWd/t of uranium. The tests covered a wide range of operating temperature, external pressure, cladding restraint, fuel density, void volume, and UO_2 composition [18]. Fig.41 shows the change in fuel structure in a UO_2 plate exposed to approximately 75 000 MWd/t of uranium at over 2000°F; the numerous voids generated are attributed to the internal precipitation of fission gases (more than 10% release as compared to less than 1% release at low burn-ups), whereas the white inclusions are tentatively ascribed to the agglomeration and precipitation of non-volatile fission product. X-ray crystallographic examination, on the other hand, has revealed that the crystalline structure of UO_2 is retained even at burn-ups of 75 000 MWd/t of uranium.

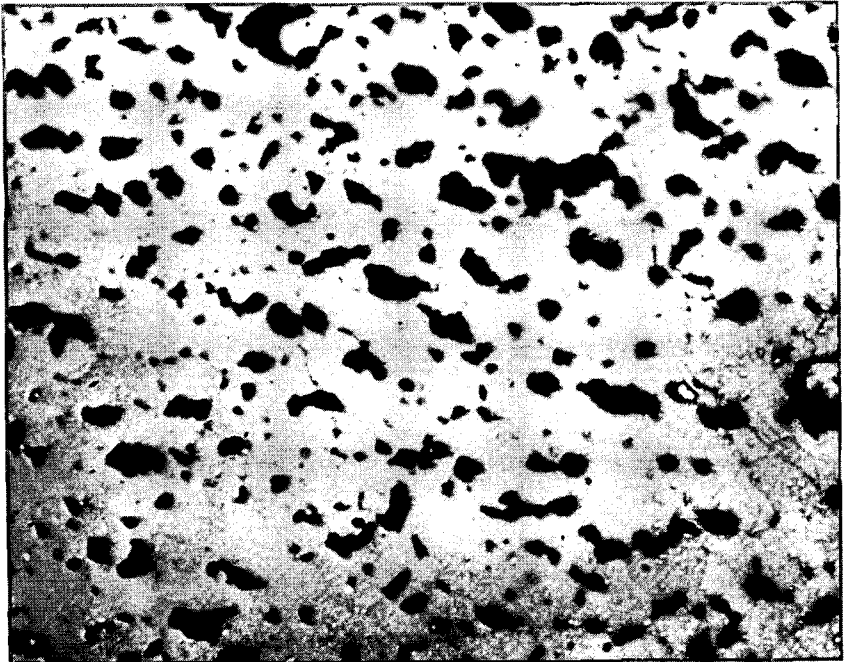


Fig. 41

Photograph of UO_2 irradiated to 75 000 MWd/t of uranium at a temperature greater than 2000°F (400X)

In addition, the thermal conductivity of UO_2 continuously deteriorates as burn-up progresses, as shown in Fig.42, resulting in a continuous increase in fuel temperature.

The most striking result of this irradiation programme is the identification of the strong dependence of the irradiation stability of high-density UO_2 fuel elements on the fuel operating temperature and restraint. For the case of Core-2 plate elements, this relationship is shown in Fig.43 which also indicates the margin of the most highly depleted Core-2 blanket fuel to clad rupture. This margin is considered adequate in meeting Core-2 design and material objectives.

Seed fuel

A requirement of the Core-2 seed design is the development of a fuel material capable of 25×10^{20} fissions/cm³ burn-up with adequate irradiation stability and corrosion resistance. After extensive in-pile and out-of-pile testing of various diluents of low thermal-neutron cross-section, the choice was narrowed to ZrO_2 because of its adequate corrosion resistance and relatively superior mechanical and dimensional stability under Core-2 irradiation exposure conditions.

The limitations of ZrO_2 - UO_2 fuel material as a function of burn-up and operating temperature[19] are shown in Fig.44, which is similar to Fig.43 for natural UO_2 . The behaviour of ZrO_2 - UO_2 differs from UO_2 in three major respects: (1) The initial porosity in ZrO_2 - UO_2 wafers has been observed to disappear completely at irradiation exposures less than 6×10^{20} fissions/cm³, as demonstrated in Fig.45, in contrast to porosity retention in UO_2 wafers even at 34×10^{20} fissions/cm³; (2) defected ZrO_2 - UO_2 fuel elements show no evidence of corrosion attack in either in- or out-of-pile tests in high-temperature water or steam and for very high fissioning rates including rates which have induced corrosion attack on UO_2 ; and (3) the thermal conductivity of ZrO_2 - UO_2 remains relatively unchanged with irradiation as shown in Fig.42.

The development programme on the Core-2 seed fuel material has, thus, culminated in obtaining a ZrO_2 - UO_2 fuel material which shows good promise of meeting the extremely high depletion requirements specified.

Poison materials

The nuclear design of Core 2 required, as discussed previously, that an appreciable fraction of the excess reactivity be controlled by the addition of lumped burnable poison. To obtain adequate matching of poison depletion to reactivity changes of the core, poison materials of high specific boron loading were required. It was further desirable that the lumped burnable poison be capable of being incorporated in the same fuel-element configuration as that utilized for the oxide plate seed. The austenitic stainless-steel alloy containing 0.8% B-10 additions has been selected because of its superior dimensional stability under irradiation and adequate corrosion resistance in high-temperature water.

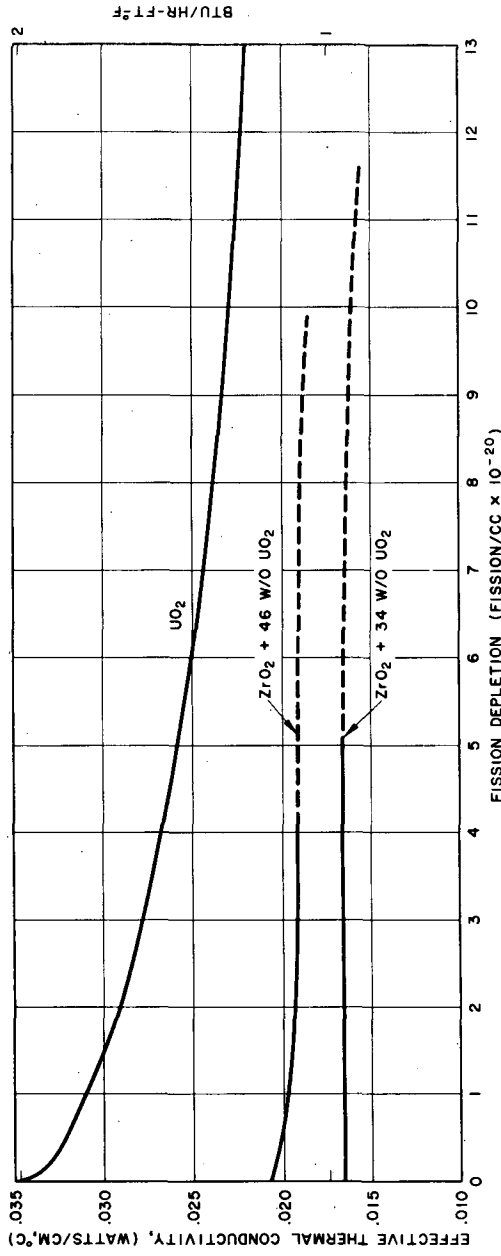


Fig. 42
Effective thermal conductivity of ceramic fuels during irradiation

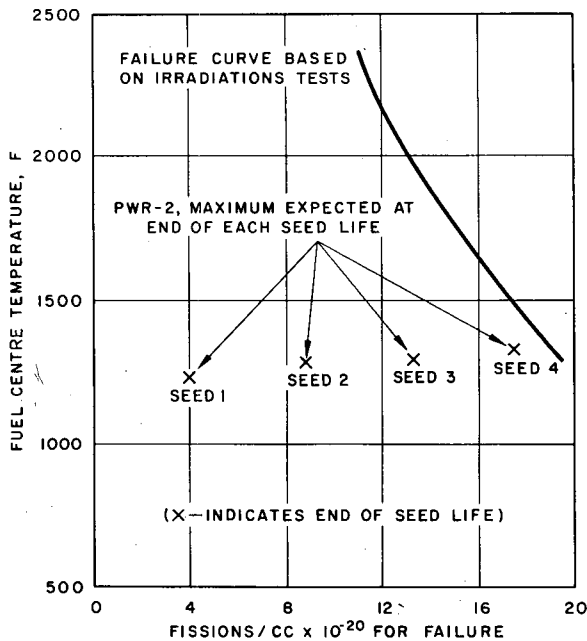


Fig. 43

Comparison of centreline fuel temperatures for failed UO_2 compartments with those expected in Core-2 blanket

Core-2 manufacture

The fuel elements for both the blanket and seed clusters of Core 2 consist of high-density ceramic-fuel wafers inserted in machined compartments of Zircaloy receptacle plates which are then pressure bonded to Zircaloy cover plates. Wafers of the B-10 stainless-steel alloy used as a lumped burnable poison are inserted in some of the seed fuel-element compartments in place of fuel wafers subsequent to the bonding process.

Manufacture of the natural UO_2 wafers for the blanket fuel elements follows essentially the same basic steps that were used in the UO_2 pellets for Core 1. These steps are (1) agglomeration, (2) compaction, (3) sintering, (4) surface preparation, and (5) pyrolytic carbon coating. The latter two operations are required to prevent the sintered UO_2 wafer from reacting with or diffusing into the Zircaloy cladding during clad bonding.

The seed fuel-wafer fabrication process is basically similar to the blanket wafer process. The differences in the two processes are due primarily to (1) the addition of the ZrO_2 powder to the enriched UO_2 along with the requirements to obtain a complete solid solution of the two, (2) the necessity to process limited quantities (maximum of 5 lb) of material at a time when using enriched uranium-oxide because of criticality requirements, and (3) the reduced thickness (0.036 as opposed to 0.100 in) of the seed wafers.

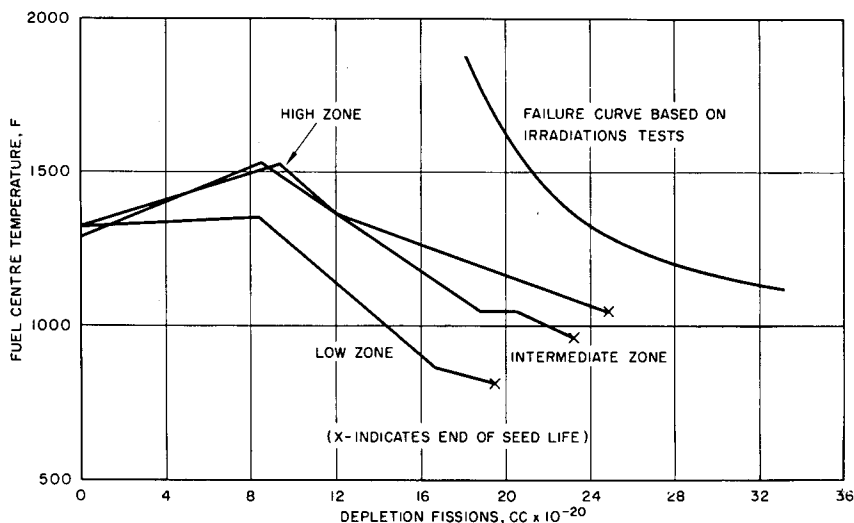


Fig. 44

Comparison of centreline fuel temperature in seed fuel elements
test with that expected for Core 2

The manufacture of the 0.8 wt. % boron stainless-steel alloy wafers follows standard alloy practices for stainless steel. The alloy actually consists of a uniform dispersion of very fine (5 μ m or less) particles of a complex iron-chromium-nickel boride in the stainless-steel matrix. Chromium coating was deposited on the alloy wafer surfaces to prevent the formation of low melting zirconium-base eutectics during the sub-assembly welding process.

A method of bonding which does not require extensive plastic deformation of the fuel components had to be developed for the Core-2 elements. A gas pressure-bonding process was chosen which effected a bond through the simultaneous application of 6500 (seed) and 10 000 lb/in² (blanket) gas pressure and 1580°F temperature. The combination of temperature and pressure forces the mating Zircaloy surfaces into intimate contact and allows solid-state bonding to occur by diffusion and subsequent grain growth across the Zircaloy interfaces.

The bonding temperature for optimum bond quality has been found to be 1580°F. However, extensive corrosion testing of samples and coupons selected from pressure-bonded development fuel elements indicated that bonding at 1580°F or higher with subsequent slow cooling from this temperature resulted in increased hydrogen pick-up and weight gains after corrosion testing. To restore the optimum corrosion resistance requires the adoption of a post-bonding heat treatment into the beta region (1875°F), holding for five minutes and rapidly cooling the fuel elements at a rate in excess of 200°F/min. A hydrogen removal treatment (vacuum annealing at 1450°F) and a hotflattening operation are necessary to complete the post-bonding thermal treatments.

The manufacturing status of Core 2 at this time may be summarized as follows: All fuel elements for both the seed and blanket have been com-

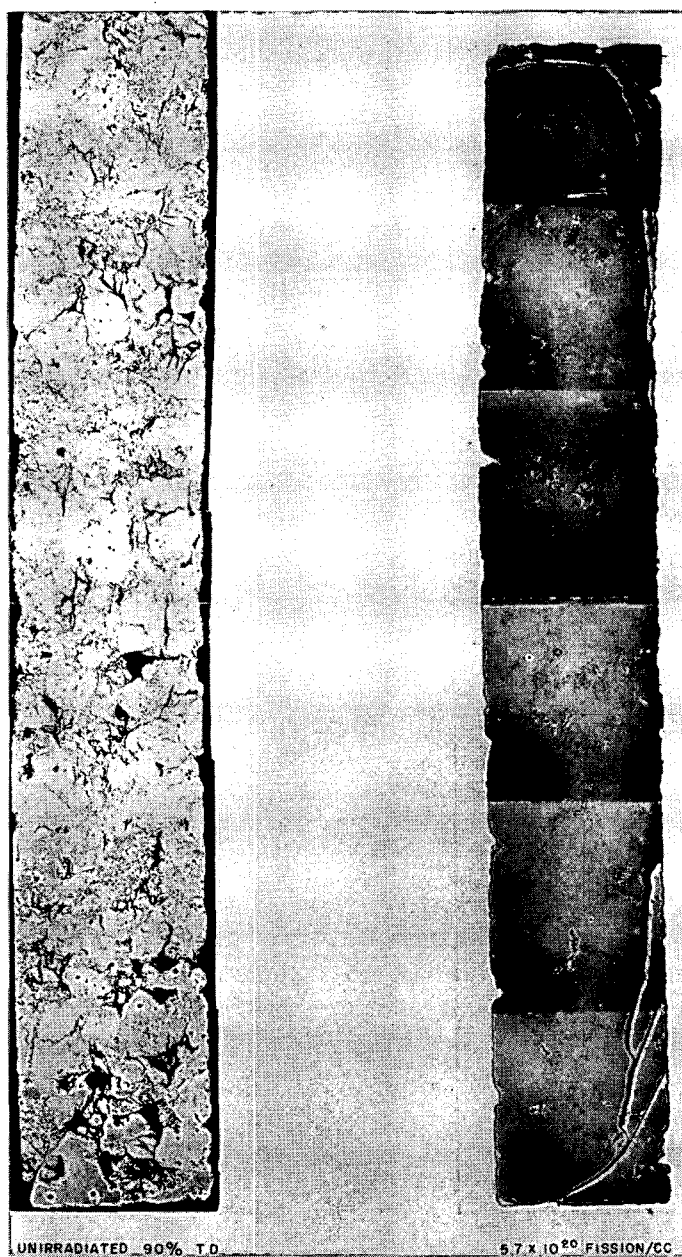


Fig. 45

In-pile densification of $\text{ZrO}_2 + 34 \text{ wt. \% UO}_2$ after short exposure (60x)

pleted and all of the blanket sub-assemblies have been welded and are in various stages of inspection. The first seed sub-assemblies have also been

satisfactorily welded. The structural components are about 75% complete; the reactor closure head has been shipped to the Station. Control drive mechanisms are approximately 95% complete and shipments have started. The core is expected to be completed in the latter part of 1963.

During the production of Core 2, new manufacturing problems arose which required development and, in some instances, core redesign. This type of problem is usually expected in the design and manufacture of a development core. The seed poison plates were redesigned to eliminate the adverse effects of irradiation-induced poison element swelling and to minimize the reaction of the poison material with the cladding during sub-assembly welding. In the development of electron-beam welding, considerable process development was required in order to obtain acceptable corrosion characteristics of the finished product. Development of fibre optic techniques for inspection of corrosion products and defects on fuel-element surfaces required considerably more effort than had been predicted, but the resulting inspection technique permits a very satisfactory inspection of the oxide plate characteristic of PWR Core 2. These are typical examples of the requirements usually experienced in the manufacture of a developmental core.

Plant modifications

The Shippingport Plant has an electric generating capacity of 100 MW(e). In order to demonstrate the 150 MW(e) gross capability of Core 2, a heat dissipation system consisting of water-cooled condensers and associated equipment will be added to the steam plant to dissipate the reactor power produced in excess of 100 MW(e) gross. Based on an economic study, the heat-dissipation system approach was selected over the alternative of increasing the generating capacity to 150 MW(e).

The demonstration of a 150 MW(e) gross capability of Core 2 also requires the modification of several components of the reactor plant, most notably the main coolant pumps, the steam generators, the primary and secondary relief valves, and modifications to the Safety Injection System.

The Core-2 flow requirements are approximately the same as those for Core 1 but, due to the higher pressure drop of Core 2, the pumping head and horsepower will have to be increased about 45%. New main coolant pumps are therefore being procured retaining the vertical single stage centrifugal design. The Core-1 pump volutes which were welded into the primary coolant system will also be retained, but are being machined in place to fit the new impellers.

Simulator studies of plant operating transients have demonstrated that the present pressurizer will be adequate for Core-2 operation. However, the analysis of an operating emergency case (step loss of the 150 MW(e) station load demand with no rod motion) has indicated a need for additional relief capacity. Hence, another pilot-operated relief valve of the same design as the existing valve and having the same control circuitry will be installed on the pressurizer so that both valves will operate in unison. Four additional self-actuated relief-valves will be installed on the secondary side of the steam generators and new springs will be installed in the existing secondary relief-valves to raise their set point.

To remove reactor decay heat during a loss of pumping power, a self-actuated secondary steam relief-valve has been utilized for Core 1. Natural convection flow through the main coolant loops transfers the decay heat from the reactor vessel to the steam generators. Operation of the secondary steam relief-valve dissipates the heat to the atmosphere. The higher decay-heat level of Core 2 requires the installation of a larger capacity relief-valve.

The Safety Injection System will supply cooling water to the core from the steam-plant feed pumps for protection against meltdown from decay heat generation in the unlikely event of a reactor plant boundary rupture resulting in a loss-of-coolant accident. Improvements in the Core-1 system will be provided by system modifications (including addition of a pump and valves and some piping changes) which permit utilization of the core-failed element detection and location piping to charge cooling water directly into each seed and blanket cluster.

It is expected that the plant modifications and installation of Core 2 at Shippingport will be completed during the last half of 1964. The Catalytic Construction Company will do the actual modification work and the procedures for refuelling are being developed by the Duquesne Light Company and Bettis Atomic Power Laboratory. Present planning calls for the decontamination of the primary plant before carrying out the modification. A decontamination solution of dilute potassium permanganate is expected to be circulated to oxidize the deposited radioactive corrosion product. This will be followed by an ammonium citrate solution rinse to remove the oxidized corrosion products. Developmental programmes are under way to determine the optimum conditions for the decontamination process. This will be the first time that a central station nuclear power plant will have been decontaminated after years of operation.

Reports on the overall modification programme, including the effectiveness of the decontamination, will be prepared and distributed to the nuclear power industry.

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INITIAL OPERATIONAL RESULTS OF THE FIRST BELGIAN NUCLEAR POWER STATION BR3

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Abstract — Résumé — Аннотация — Resumen

INITIAL OPERATIONAL RESULTS OF THE FIRST BELGIAN NUCLEAR POWER STATION BR3. The BR3 Nuclear Power Plant, property of the Centre d'étude de l'énergie nucléaire (CEN), is located at Mol-Donk on the site of the latter, and at a distance of 40 miles as the crow flies from Brussels. The operation of the plant is done by an Industrial Group. The reactor went critical, for the first time, on 29 August 1962. The paper gives the results of the preliminary tests and of the first months of operation. The reactor BR3 is of the pressurized, light-water moderated and cooled type, with a nominal thermal power of 40.9 MW. The core is heterogeneous, using UO_2 fuel, clad with stainless steel and has two regions. The outer region is enriched to 4.43% by weight, the inner one to 3.7%. Main coolant pressure is 140 kg/cm². Different "primary" circuits are in connection with the reactor.

The steam generator has a nominal capacity of 70 t/h. The turbine has two stages and three extraction lines and a nominal power of 11 700 kW. The personnel was selected from the experienced personnel of the conventional power plants and received a thorough training in both the nuclear and the classical field.

The paper gives the results of the practical determination of the plant characteristics: control-rod worth and boron worth determination, calibration of the control rods, temperature and pressure coefficients determination, determination of the dropping time of the control rods, etc.

The excess reactivity was calculated for the clean core. These tests were done at zero power and after 441 effective full power hours (EFPH) burn-up: 750 MWJ). A résumé is given of the load transient tests and the power coefficient determination.

Different problems encountered during operation are treated:

- (a) Radiation activity in the plant: we may conclude from the results that the habitual work in the normal areas is not affected by the radiation intensities present.
- (b) Activity in the air of the building: the only problem which appeared until now is the presence of radon in the reactor building and in certain auxiliary buildings. This radon comes from the concrete.
- (c) Leak-rate of primary water: about 100 l/h.
- (d) Mechanical difficulties, e. g. with the charging pumps, untightness of valves, etc.

RÉSULTATS INITIAUX DU FONCTIONNEMENT DE LA PREMIÈRE CENTRALE NUCLÉAIRE BELGE BR3. La centrale nucléaire BR3, qui appartient au Centre d'études de l'énergie nucléaire (CEN) est située, comme le Centre, à Mol Donk, à une soixantaine de kilomètres de Bruxelles à vol d'oiseau. La centrale est exploitée par un groupe industriel. Le réacteur a divergé pour la première fois le 29 août 1962. Le mémoire donne les résultats des essais préliminaires et des premiers mois de fonctionnement du réacteur. BR3 est un réacteur refroidi et ralenti à l'eau légère sous pression, d'une puissance thermique nominale de 40,9 MW. Son cœur, hétérogène, est composé d'éléments en oxyde d'uranium gainés d'acier inoxydable et comprend deux zones: la zone périphérique enrichie à 4,43% en poids et la zone centrale enrichie à 3,7%. La pression dans la partie principale du circuit de refroidissement est de 140 kg/cm². Divers circuits «primaires» sont reliés au réacteur.

Le générateur de vapeur a une puissance nominale de 70 t/h. La turbine, à deux étages, fournit du courant triphasé et a une puissance nominale de 11 700 kW.

Le personnel, choisi parmi des spécialistes ayant l'expérience de centrales classiques, a reçu une formation poussée dans le domaine nucléaire.

Les auteurs présentent les résultats de la détermination pratique des caractéristiques de la centrale: anti-réactivité des barres de contrôle et du bore, étalonnage des barres de contrôle, coefficients de température et de pression, temps de chute des barres, etc.

L'excédent de réactivité a été calculé pour le cœur non empoisonné. Ces essais ont été réalisés à la puissance nulle et après l'équivalent de 441 heures de marche à pleine puissance (combustion massique: 750 MW/j).

Les auteurs décrivent brièvement les essais en charge variable et la détermination du coefficient de puissance. Ils signalent divers problèmes qui se sont posés au cours de expériences:

- a) Activité des rayonnements dans la centrale: les résultats des expériences ont permis de conclure que dans les zones normales l'intensité des rayonnements n'affecte pas les travaux habituels.
- b) Radioactivité de l'air dans le bâtiment: le seul problème qui se soit posé jusqu'à présent a trait à la présence de radon dans l'enceinte du réacteur et dans certains locaux auxiliaires. Ce radon provient du béton.
- c) Fuite de l'eau du circuit primaire: environ 100 l/h.
- d) Difficultés mécaniques avec les pompes d'alimentation, défaut d'étanchéité des vannes, etc.

ПРЕДВАРИТЕЛЬНЫЕ РЕЗУЛЬТАТЫ ЭКСПЛУАТАЦИИ ПЕРВОЙ БЕЛГИЙСКОЙ АТОМНОЙ ЭЛЕКТРОСТАНЦИИ BR 3. Ядерная энергетическая установка BR 3, собственность Исследовательского ядерного центра (CEN), расположена в Моль-Донке, на площадке последнего, на расстоянии 40 миль по прямой от Брюсселя. Эксплуатация установки осуществляется промышленной группой. Реактор достиг критического состояния впервые 29 августа 1962 года. В данном докладе мы приводим результаты предварительных испытаний и первых месяцев эксплуатации. Реактор BR 3 является реактором с водным замедлителем и теплоносителем, номинальная тепловая мощность его 40,9 мвт. Активная зона - гетерогенного типа, с использованием в качестве топлива UO_2 , покрытие - из нержавеющей стали. В активной зоне имеется две зоны: Степень обогащения внешней зоны составляет 4,43 вес%, внутренней - до 3,7 вес%.

Давление основного охладителя составляет 140 кг/см². Различные "первичные" контуры связаны с реактором.

Номинальная мощность парогенератора составляет 70 т/час. Турбина имеет две ступени и три линии паросбора; номинальная мощность турбины 11 700 квт.

Персонал набран из числа опытных сотрудников обычных энергетических установок. Все получили тщательную подготовку в области атомной энергии, равно как и в области обычной энергетики.

Приводятся результаты практического определения характеристик установки: определение компенсирующей способности регулирующих стержней и поглощательной способности бора, калибровка регулирующих стержней, определение температурных коэффициентов и коэффициентов давления, определение времени падения регулирующих стержней и т.д.

Избыточная реактивность рассчитывалась для чистой активной зоны. Испытания проводились на уровне нулевой мощности и после испытания 441 эквивалентных часов работы на полной мощности (глубина выгорания 750 мвт·дн/т).

Приводятся выводы, полученные в результате испытания переходных колебаний нагрузки и определения коэффициентов мощности.

Рассматриваются различные проблемы, возникшие в ходе эксплуатации:

- a) активность излучения на установке: по результатам мы можем сделать вывод, что сотрудники в нормальных зонах не подвергаются воздействию имеющегося излучения;
- б) активность в воздухе помещения: единственная проблема, которая возникла до настоящего времени, заключается в наличии радона в помещении реактора и некоторых вспомогательных помещениях; радон попадает в атмосферу помещения из бетона;
- в) величина утечки воды из первичного контура около 100 л/час;
- г) трудности, связанные с отдельными механизмами, например с нагнетательными насосами, с клапанами и т.д.

RESULTADOS INICIALES DEL FUNCIONAMIENTO DE LA PRIMERA CENTRAL NUCLEOELÉCTRICA BELGA BR3. La central nucleoelectrica BR3, que pertenece al Centre d'étude de l'énergie nucléaire (CEN), está situada en Mol-Donk, en los terrenos de este Centro, a una distancia de 40 millas a vuelo de pájaro de Bruselas. De la explotación de la central se encarga una empresa industrial. El reactor alcanzó la criticidad inicial el 29 de agosto de 1962. Los autores exponen los resultados de las pruebas preliminares y de los primeros meses de funcionamiento. El reactor BR3 es del tipo moderado y refrigerado por agua ligera a presión y su

potencia térmica nominal asciende a 40,9 MW. El núcleo es heterogéneo, su combustible es de UO_2 con revestimiento de acero inoxidable y tiene dos regiones. El enriquecimiento de la región externa es de 4,43 por ciento en peso, y el de la interna, de 3,7% también en peso.

La presión del refrigerante alcanza a 140 kg/cm². El reactor está conectado con varios circuitos primarios.

La capacidad nominal del generador de vapor es de 70 t/h. La turbina es de dos etapas con tres líneas de extracción y una potencia nominal de 11 700 kW. El personal se ha seleccionado entre los efectivos que poseen una vasta experiencia en materia de centrales de tipo tradicional y ha recibido formación completa tanto en la esfera nuclear como en la clásica.

En la memoria se exponen los resultados de la evaluación experimental de las características de la central: determinación del valor de las barras de control y del boro, calibración de las barras de control, determinación de los coeficientes de temperatura y de presión, determinación del tiempo de introducción de las barras de control, etc.

Se ha calculado el exceso de reactividad del núcleo limpio.

Estas pruebas se han efectuado tanto a potencia cero como después de un tiempo de funcionamiento equivalente a 441 l a plena potencia (grado de combustión: 750 MWd).

Se describen brevemente las pruebas de carga transitoria y la determinación del coeficiente de potencia.

Se exponen diversos problemas planteados durante el funcionamiento:

- a) Radiactividad de la central: los resultados permiten concluir que la intensidad de las radiaciones no afecta al desarrollo de las operaciones habituales en las zonas de trabajo.
- b) Radiactividad del aire en el edificio: el único problema que ha surgido hasta ahora es la presencia de radón en el edificio del reactor y en algunos edificios auxiliares. El radón proviene del hormigón.
- c) Índice de escape del agua del circuito primario: unos 100 l/h.
- d) Dificultades de tipo mecánico, por ejemplo, las planteadas por las bombas de carga, la falta de estanqueidad de las válvulas, etc.

1. INTRODUCTION

The BR3 nuclear power plant, property of the Centre d'Etude Nucléaire (CEN) is located at Mol-Donk on the site of the latter, and at a distance of 40 miles as the crow flies from Brussels. The operation of the plant is carried out by an Industrial Group (Fondation Nucléaire). The interested companies have assigned qualified members of their personnel to form the operating crew. Some members of the CEN personnel have also been included in this group.

Criticality of the reactor was first achieved on 29 August 1962. The station was brought in parallel for the first time on 10 October 1962. Since that time until the end of February 1963 the station's output to the Belgian grid amounted to 10 126 000 kWh(e).

This paper gives the results of the preliminary tests and of operation during the first months.

2. SUMMARY OF THE PRINCIPAL CHARACTERISTICS OF THE BR3

The BR3 reactor is of the pressurized type, light-water moderated and cooled. The principal characteristics of the plant are as follows.

2.1. Reactor and main loop

The core of the reactor (Fig. 1) consists of 32 fuel assemblies of UO_2 of which 16 are enriched to 3.7% by weight (inner region) and 16 are enriched to 4.43% by weight (outer region). There are 12 control rods com-

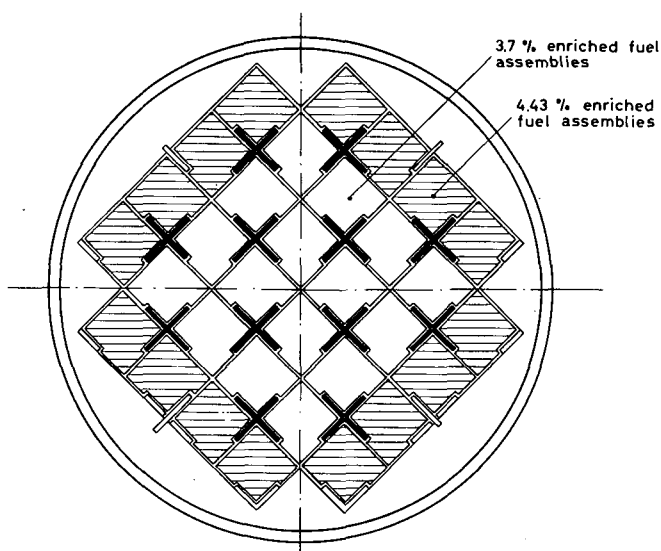


Fig. 1

Cross-section of the BR3 core

posed of 80% Ag+ 15% In+ 5% Cd. The total heat output is 40.9 MW and the estimated life-time of the core is 7000 EFPH (effective full power hours).

The average coolant temperature of the main loop is 262°C and the coolant flow rate is 2 250 t/h. The temperature difference between inlet and outlet of the reactor at full power is 14°C. The nominal system pressure is 140 kg/cm² maintained by a pressurizer (maximum heating capacity 153 kW).

2.2. Associated "primary" loops

A purification loop with a variable flow rate from 1 to 7 m³/h and provided with two demineralizers (anionic for H₃BO₃ and Mixed-Bed) functions at low pressure (7 kg/cm²) and in by-pass on the main loop, thus necessitating decompression stations and pumps. A surge tank is provided through which make-up water and hydrogen are added to the primary loop.

Demineralized water for the make-up water loop comes from the BR2/BR3 common water treatment station and is stored in a primary water storage tank. The make-up water is degasified in a vacuum de-aerator and transferred by two pumps to the charging system.

The shut-down cooling loop consists of two pumps and a heat exchanger to evacuate the calories of a sub-critical core (fission products), brought to low temperature and pressure.

There is also an emergency shut-down loop to evacuate the calories of a sub-critical core at normal operating temperature and pressure (fission products) and in the event of an emergency. In addition, there is a safety injection loop to compensate important losses of water due to a leak and to avoid the core being uncovered at any time.

The storage well loop consists of pump, heat exchanger and demineralizer to evacuate the heat produced by a depleted core stored in the storage well.

The component cooling loop consists of two pumps and two heat exchangers in a closed loop to evacuate the calories of the other heat exchangers and main coolant pumps. The two heat exchangers of this loop are cooled by water coming from a lagoon.

Finally, there are also the vents and drains loop, the liquid wastes loop (activity, contamination), and the ventilation and gaseous wastes loops (activity, contamination).

2.3. Measuring and control systems

Measurement of temperatures, pressures, flows, levels is by classical methods. There is also a nuclear instrumentation system which follows the evolution of the neutron flux and thus of the power of the reactor.

The control system of the reactor can be either manual or automatic and there is a radioactivity control system to protect personnel and plant.

2.4. Steam generator and turbine

The vertical steam generator contains 1400 hairpin tubes. At full load operating conditions, the generator steam flow amounts to 70 t/h and the steam pressure is 36.9 atm abs.; at no-load the pressure increases to 50 atm abs. The main steam line is provided with a turbine by-pass. The steam, supplied to a two-stage turbine with three extraction lines, provides a gross electrical generating capability of 11.7 MW at the turbine shaft. A moisture separator, placed between the two stages of the turbine, lowers the humidity of steam from 10% to less than 1% before entering the low-pressure part of the turbine. The condensation and pre-heating part is of a classical type. (Fig. 2)

2.5. Alternators

The principal alternator has a rated power output of 11.2 MW under 10.5 kV. An auxiliary generator is driven by the turbine generator by means of a reduction gear. This auxiliary generator feeds one of the primary pumps. In the event of a turbine trip, the auxiliary generator will continue for a while to supply this pump (7 min) as a result of the group inertia. This allows for the starting of the diesel engine and the emergency shut-down process (2 min maximum).

3. OPERATING PROBLEMS

3.1. Personnel and training

The personnel was selected from experienced people from classical power stations with a solid background in that field. Their experience was completed and oriented to the BR3 during the end of construction of the plant and during the many performance and functional tests of the different loops.

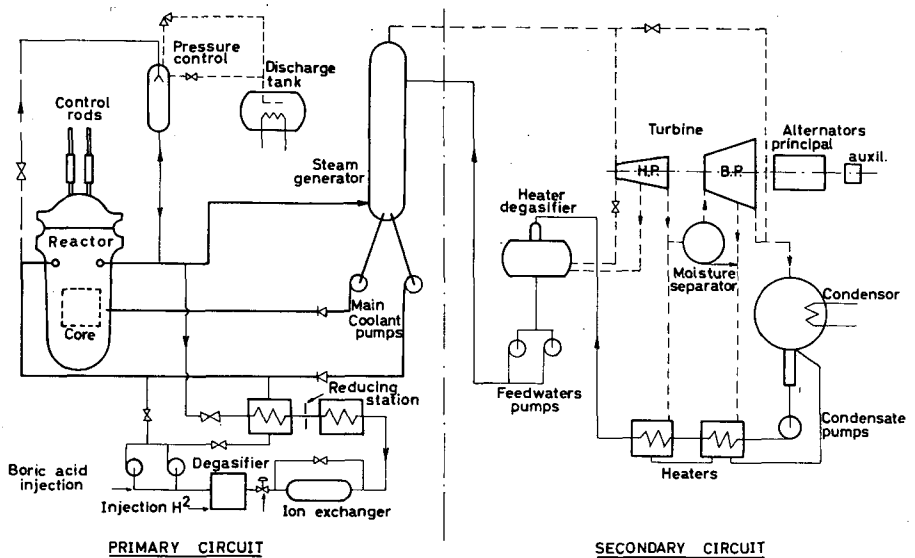


Fig. 2

BR3 schematics

Before this, nuclear training was given to them in universities or technical schools. This was extended on the spot by a complementary training given by the nuclear engineers, commenting in detail on the different characteristics of the BR3 reactor, with special emphasis on the safety aspects, instrumentation and the safety interlocks. In addition, the first operators in charge of running the reactor and its loops received training at the BR2 (a high-flux reactor for materials testing).

In collaboration with the Department of Applied Mathematics, an analogue computing machine was used to simulate some of the nuclear and thermal equations of the reactor. This simulator included controls and instruments, which allowed the BR3 operating crew to carry out and observe (a) start-up, the operation of a control-rod group with criticality approach and power increase; and (b) the behaviour of the reactor under disturbances such as a change of secondary load, with and without action of control rods, and a sudden variation of the position of one group of control rods.

The personnel also participated in the loading of the reactor, the critical tests at low power and the load transient tests.

Actually the plant is operating properly under the general supervision of the operation engineers and, for start-ups and tests, also of the nuclear engineer.

An important item which must be stressed is the knowledge of nuclear risks, particularly the problems involving contamination. Stringent rules and procedures were established by a crew of health physicists from CEN and are being properly carried out.

3.2. Practical determination of plant characteristics

3.2.1. Reactor

The operational safety of our heat source depends on its kinetic behaviour and on the knowledge of its kinetic characteristics, this knowledge allowing the establishment of safe procedures for plant operation. A list of the various parameters, giving reactivity modification, is given below. Obviously, a positive reactivity means an increase in the reactor power output, a negative reactivity a decrease of this power. The way in which the parameters work applies, of course, to the BR3 core only.

- (a) Water temperature, with a negative effect; if the temperature increases, ρ decreases;
- (b) Fuel temperature, with a negative effect; if the temperature increases, ρ decreases;
- (c) Pressure, with a positive effect; if the pressure increases, ρ increases too;
- (d) Control rods, with a positive effect when going up and a negative one when going down;
- (e) Neutron poisons such as fission products (or a chemical poison voluntarily added), with a negative effect; and
- (f) Fuel burn-up, with a negative effect.

It is very important to know the quantitative effect of these parameters on core reactivity. The following operations were performed.

(a) With the new core (burn-up = 0). 9-10 August 1962: loading of fuel assemblies into the pressure vessel, all control rods being inserted and poisoning of primary water assured by boric acid (boron concentration: 1000 ppm).

Besides the normal core instrumentation (with too low a sensitivity during loading because it is located, outside the core, in the neutron shield tank), three complementary neutron detector channels (BF_3 counters) were installed inside the vessel during loading in order to follow the thermal neutron flux.

11-21 August 1962: closure of the pressure vessel and replacement of the whole equipment in the reactor compartment (ventilation ducts, shielding slabs, etc)

29 August 1962: Initial criticality in cold condition (70°C), low pressure (21 kg/cm^2) with poisoning by 960 ppm boron. The reactor was just critical with all rods banked at a height of 83 cm.

31 August - 1 September 1962: Determination of control rods and boron reactivity worth (same conditions as above). The results of these measurements are given in Table I for boron and in Fig. 3, curve a, for the control-rods-worth.

The determination of reactivity worth of control rods for different critical heights was possible because of the modification of boron concentration, the reactivity worth not being affected by this concentration.

2-6 September 1962: Determination of pressure and water temperature reactivity coefficients with a boron concentration of 465 ppm (pressure and temperature varying from the values given above to the normal working

TABLE I

BORON WORTH

C_B (ppm)	θ (°C)	p (kg/cm ²)	$\partial p / \partial C_B$	H all rods (cm)
876	71	21	0.9×10^{-4}	72
685	74	26	1.0×10^{-4}	55
813	259	142	0.8×10^{-4}	120
519	261	142	0.9×10^{-4}	75
184	261	141	1.0×10^{-4}	52

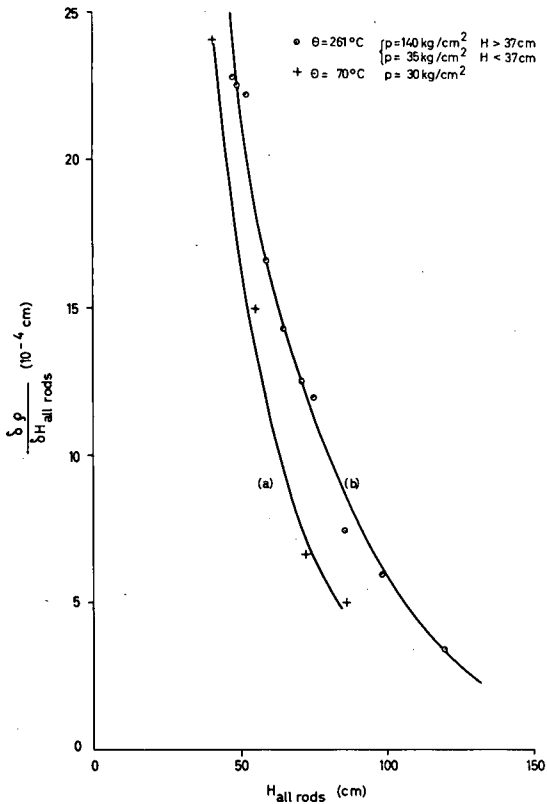


Fig. 3

$\partial p / \partial H$ all rods against H all rods

conditions, i.e. 140 kg/cm² and 262°C). The results are given in Table II for the pressure coefficient and in Fig. 4, curve a, for the temperature coefficient.

10-13 September 1962: Determination of control rods and boron reactivity worth at normal working pressure and temperature and of the critical height of control rods for various boron concentrations. The measurements relating to the control rods were made for different configurations (all rods banked, all rods out but one central group of two rods, etc.). The results are given in Table I (boron), in Fig. 3, curve b, and Figs. 5 and 6 (control rods).

TABLE II

PRESSURE COEFFICIENT

C _B (ppm)	θ (°C)	P (kg/cm ²)	∂ρ/∂p (kg/cm ³) ⁻¹
465	70	40	1.0 × 10 ⁻⁵
	214	45-138	2.0 × 10 ⁻⁵
	263	140- 95	3.1 × 10 ⁻⁵
		140-168	3.1 × 10 ⁻⁵
8.6 - 5.2	260	140-168	4.6 × 10 ⁻⁵
		140-115	4.4 × 10 ⁻⁵
		115- 90	4.5 × 10 ⁻⁵
1.7	213	140-115	3.0 × 10 ⁻⁵
		91-65	3.3 × 10 ⁻⁵
		70-40	3.0 × 10 ⁻⁵
0	40	42-120	1.6 × 10 ⁻⁵
	209	38-140	2.6 × 10 ⁻⁵

14-15 September 1962: Measurement of pressure and water temperature reactivity coefficients, without boron. See Table II for the pressure coefficient and Fig. 4, curve b, for the temperature coefficient.

22 September 1962: Measurement of control-rod drop time. Five tests were performed on each rod in normal working conditions for pressure, temperature, flow through the core, etc. The results varied between 840 ± 5 msec and 895 ± 5 msec, with an average of 870 ± msec.

(b) At 441 EFPH (burn-up = 750 MWd). 11-15 January 1963: Measurement of pressure and water-temperature reactivity coefficients around normal working temperature with a boron concentration of 475 ppm. Measurement

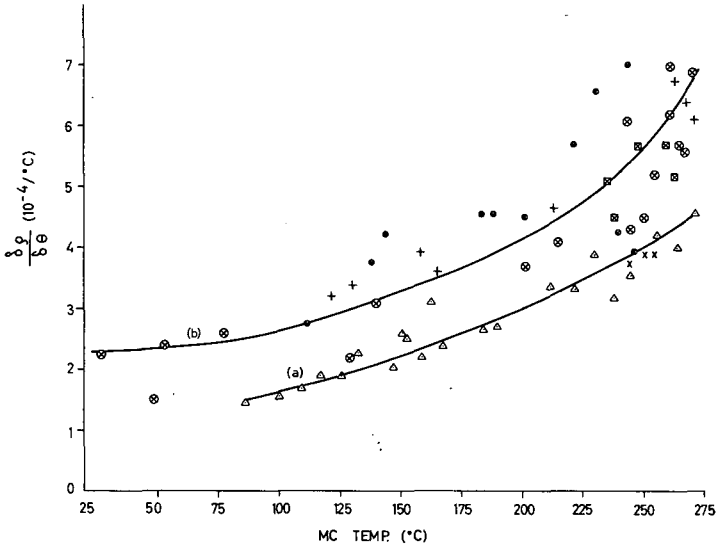


Fig. 4

$\frac{\partial p}{\partial \theta}$ against θ

Δ All rods banked $C_B = 465$ ppm	} 441 EFPH
$\square$ Programmed rods $C_B = 0$ ppm	
$\times$ All rods banked $C_B = 475$ ppm	} 0 EFPH
$\bullet$ All rods banked $C_B = 0$ ppm	
$\blacklozenge$ Programmed rods $C_B = 0$ ppm	

TABLE III

PRESSURE COEFFICIENT

C_B (ppm)	θ ($^{\circ}\text{C}$)	p (kg/cm^2)	$\frac{\partial p}{\partial p}$ (kg/cm^2) $^{-1}$
0	260	136-110	4.54×10^{-5}
0	260	110-136	4.7×10^{-5}
485	260	100-132	2.7×10^{-5}
485	260	132-105	3.04×10^{-5}

of critical height and reactivity worth of a central group of control rods, for various boron concentrations.

The results appear in Table III (pressure coefficient), in Fig. 4, curves a and b (temperature coefficient) and in Figs. 5 and 6 (control rods).

(c) At 1135 EFPH (burn-up = 1935 MWd). 11-13 March 1963: Measurement of water temperature and pressure reactivity coefficients around normal

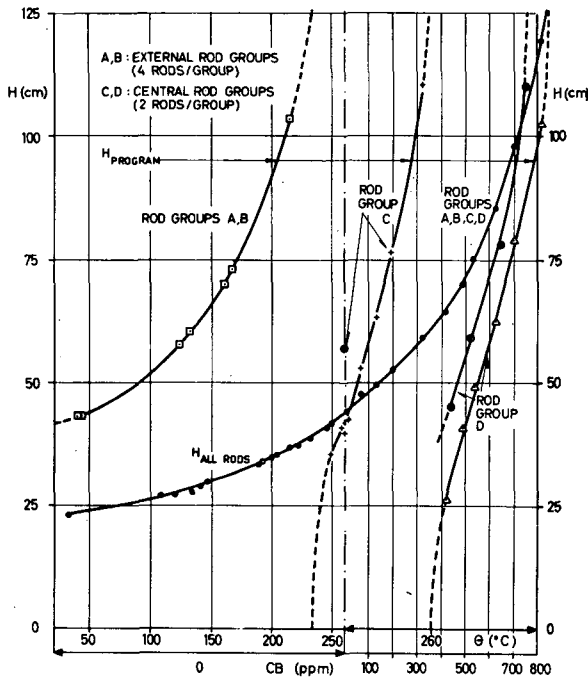


Fig. 5

$H_{\text{all rods}}$ and $H_{\text{programme}}$ against θ ($C_B = 0$) and against C_B ($\theta = 261^\circ\text{C}$)

□, + : 0 EFPH
 ⊗ : 441 EFPH

working temperature with different boron concentrations (400, 200 and 100 ppm). Measurement of boron reactivity coefficient at normal working temperature and at 100 ppm of boron. The result was $1.3 \times 10^{-4} \Delta k/k$ ppm. (d) **Conclusions.** The tests described under (a) allowed the determination of the excess reactivity without control rods of the new clean core at zero power and at normal working temperature and pressure. This excess reactivity is $8.5\% \Delta k/k$. This value was obtained in two different ways. First, from the boron reactivity worth which was around $1\% \Delta k/k$ per 100 ppm. For low boron concentration the boron reactivity coefficient is higher, as shown by the test performed at 1135 EFPH. The reactor was just critical in the desired conditions with all control rods completely withdrawn from the core with a boron concentration of 850 ppm, i. e. $8.5\% \Delta k/k$. Secondly from the integral

$$\int_{H_1}^{H_2} \frac{\partial \rho}{\partial H} dH = 8,5\% \frac{\Delta k}{k}$$

(where $H_1 = 45$ cm and $H_2 = 132$ cm) obtained with the curve B in Fig. 4 (45 cm is the critical height of all rods banked under these conditions, with-

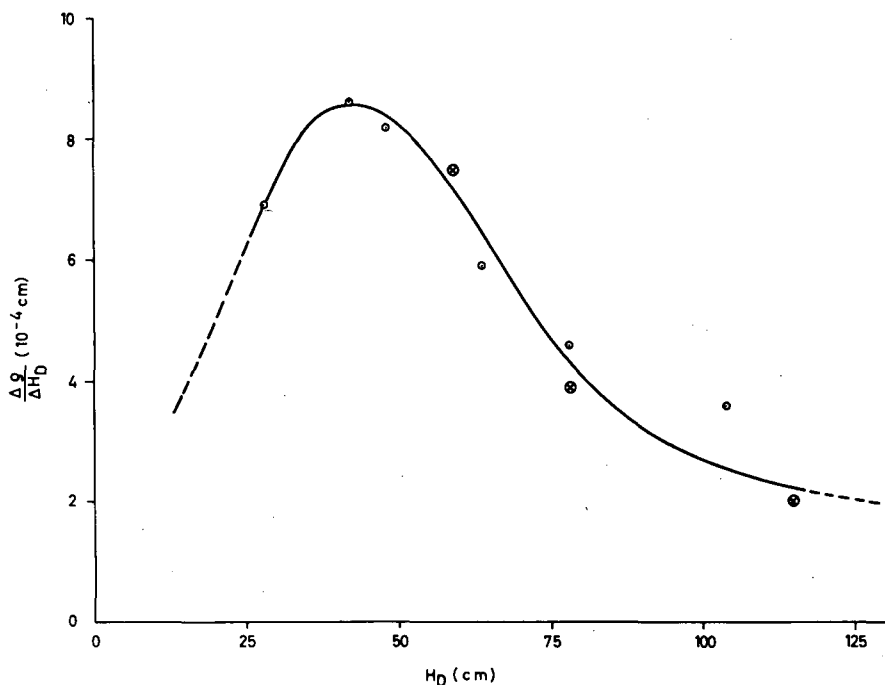


Fig. 6

$\partial \rho / \partial H_D$ against H_D
 (D = central rod group (2 rods))
 ○ 0 EFPH
 ⊗ 441 EFPH

out boron, 132 cm is the maximum height of the rods when they are completely out of the core). The two methods gave identical results.

We also determined the reactivity decrease due to the heating up of the core water; this variation was about 6% $\Delta k/k$. To obtain this number, we measured the variation in boron concentration to keep the reactor critical when going from cold to hot condition, without moving control rods or modifying any other parameter. The result was $C_B = 600 \text{ ppm} = 6\% \Delta k/k$.

The transient tests described below allowed the determination, from the average core water-temperature variations and the water-temperature reactivity coefficient, of the "power coefficient" introduced by the heating of the fuel. It was found to be equal to $1 \times 10^{-4} \Delta k/k$ per percent of full load, which gives a reactivity variation of 1% $\Delta k/k$ when going from zero to full power.

The reactivity taken by xenon poisoning at 85% of full power (we have not yet operated for a long period at full power because of a problem relating to turbine overspeed trip during a complete loss of load) is 2.35% $\Delta k/k$. The reactivity balance is given in Table IV.

The measurements at 441 EFPH and at 1135 EFPH showed that the pressure and water-temperature reactivity coefficients have not varied with the burn-up. They also allowed us to obtain a value for the reactivity taken by

TABLE IV

REACTIVITY BALANCE

Cold, clean reactor, at zero power:	14.5% $\Delta k/k$
Hot, clean reactor, at zero power:	8.5% $\Delta k/k$
Hot, clean reactor, at 85% of full power:	7.65% $\Delta k/k$
Hot reactor, eq. Xe, at 85% of full power:	5.3% $\Delta k/k$

burn-up and samarium poisoning around 0.95% $\Delta k/k$ at 441 EFPH and 1.55% $\Delta k/k$ at 1135 EFPH.

Based on the numbers obtained at 441 EFPH (750 MWd) we obtain, by subtracting the theoretical value corresponding to samarium poisoning (0.55% $\Delta k/k$), the reactivity taken by the fuel burn-up, i. e. 0.4% $\Delta k/k$. This result is corroborated by the test at 1135 EFPH.

The excess core reactivity when going over to power (hot reactor, equilibrium xenon, 85% of full power and samarium corresponding to 1135 EFPH at 85% power) is thus 3.75% $\Delta k/k$.

Taking into account a slight increase of samarium effect with irradiation time and increasing effects of xenon (0.15% $\Delta k/k$) and power coefficient (0.15% $\Delta k/k$) for working at full power, the excess core reactivity left at 1135 EFPH corresponds approximately to 3600 EFPH. Adding this to the value of 1135 EFPH, we obtain an expected core life of 4735 EFPH as against the 7000 EFPH calculated theoretically.

The difference may be gained by modification of the two-region-core configuration.

Evaluation of the reactivity subtracted by burn-up and samarium poisoning is given in Fig. 7 (0.441; 1135 EFPH). The values of water temperature reactivity coefficients against ppm of boron are given in Fig. 8.

(e) Notes. Some measurement results are not very accurate. This can be seen particularly in Fig. 4, which shows for water temperature coefficient against water temperature a rather large scattering of numbers. This lack of accuracy is chiefly caused by:

- (i) Inaccuracy in control-rod position measurement and slipping effects of the rods due to the actuating mechanism (a modification of the control circuits has in the meantime eliminated this slipping);
- (ii) Inaccuracy in period measurements because of lack of a special channel for those measurements; and
- (iii) Difficulty in obtaining a constant rate of warm-up or cool-down for determination of water temperature coefficients.

3.2.2. Turbine and reactor

After the determination of the nuclear characteristics of the reactor, the dynamic response of the latter was studied by modifying the secondary load. At the same time the extent of automatic control necessary has been checked. Information had also to be collected on the behaviour of the secondary plant, especially the turbine, during those transients.

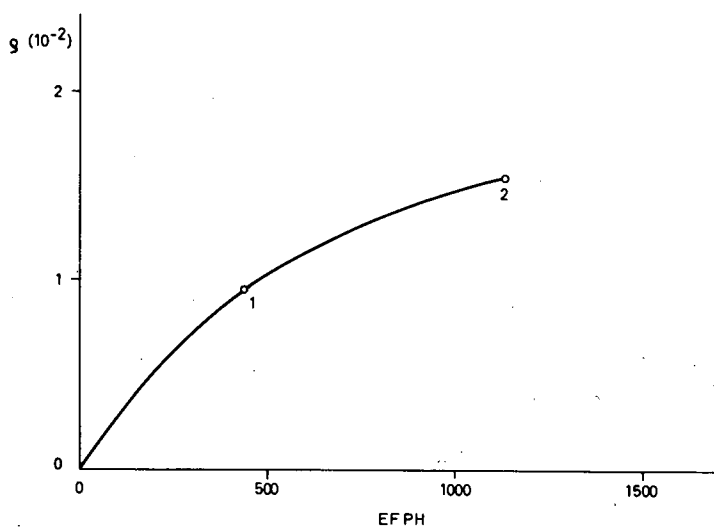


Fig. 7

Reactivity taken by the burn-up and the samarium as a function of EFPH

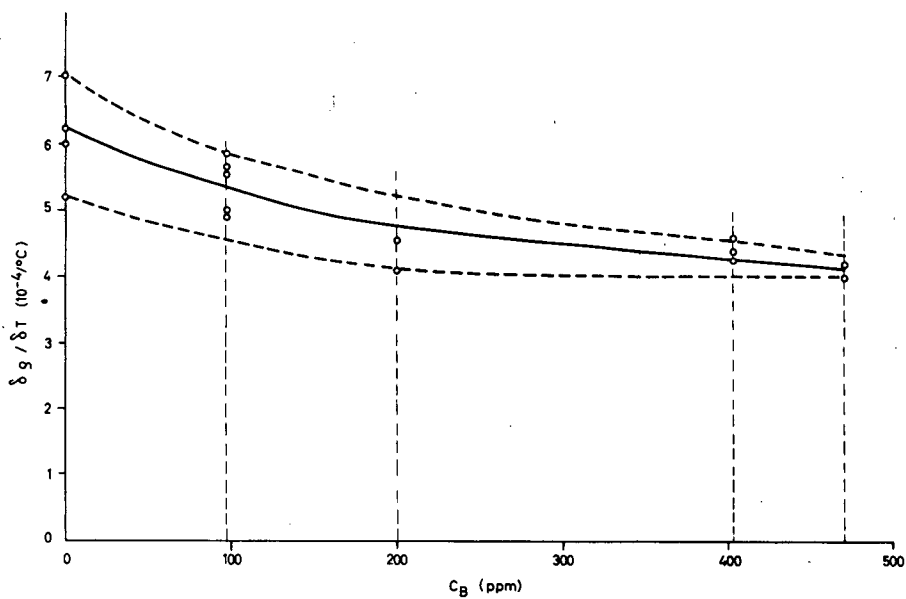


Fig. 8

Water-temperature coefficient as a function of the boron concentration (C_B)

Those transients were performed at increasingly higher loads, which were then shed by opening the tie-breaker to the outside grid. Before proceeding with a fresh test, the results of the foregoing ones were carefully analysed. During the load-transients the control rods were not moved and the automatic control of the reactor was put into service only once. This allowed, in addition, the determination of the power coefficient from the water-temperature coefficient. When the average temperature of the main coolant was stabilized, its nominal value was restored by moving the regulating control rods. Various possibilities were tested:

- (i) Transients by dropping the load to the load of the auxiliaries;
- (ii) Transients by dropping the load to zero load of the turbine;
- (iii) Transients under preceding conditions with and without opening the turbine by-pass to the condenser;
- (iiii) Transients with turbine trip, due to overspeed, and closing of the throttle valve, without opening the turbine by-pass.

The tests were started at 2 MW and continued at 4, 6, 8, 9, 10 and 11.2 MW(e) (11.2 MW = full power). In Figs. 9, 10, 11 and 12 are given graphically the results of the tests from 10 MW to 0, and from 11.2 MW to the auxiliaries.

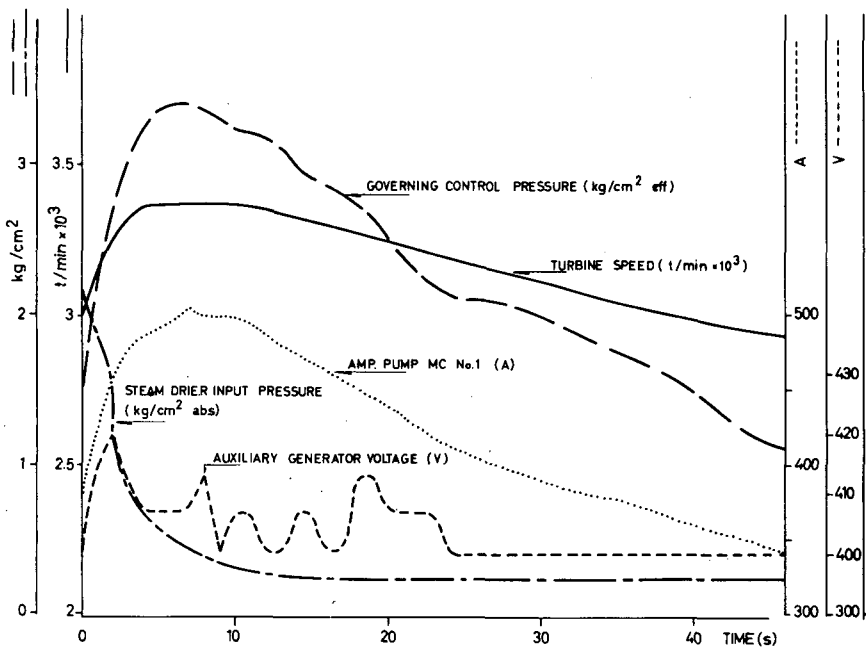


Fig. 9

BR3 Load drop test 10 MW → 0 MW: secondary parameters

The tests proved that the reactor may be regarded as capable of responding perfectly to all transients. The primary pressure and temperatures varied in a regular manner; the extreme values did not considerably

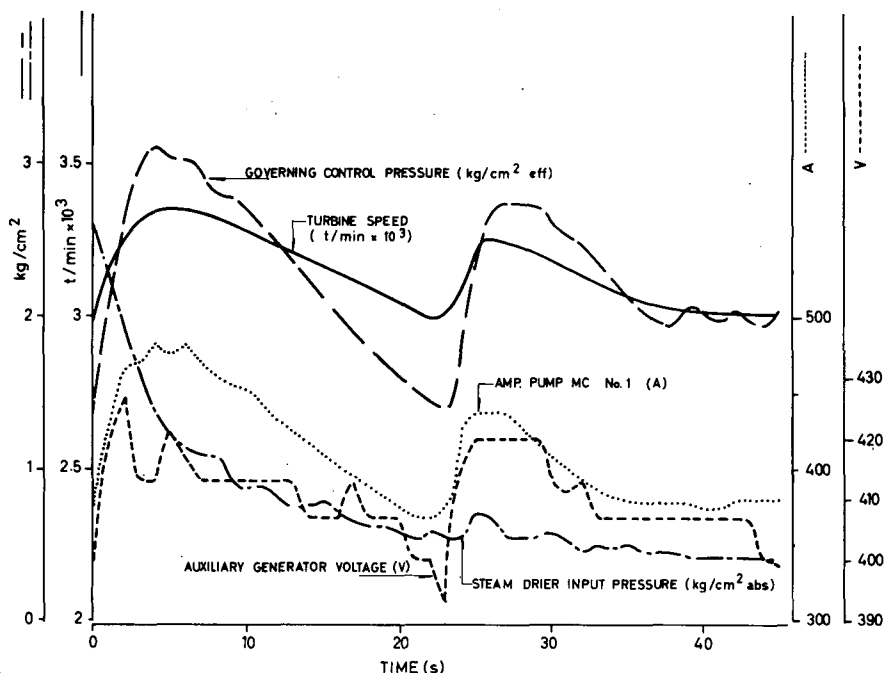


Fig. 10

BR3 Load drop test 11.2 MW $\rightarrow$ 0.38 MW: secondary parameters

overshoot the equilibrium values. The new equilibrium was attained between 8 to 15 min after opening of the circuit-breaker.

During the most severe transient for the reactor (load-drop from full power to 0, without opening the turbine by-pass) the hot-leg temperature increased from 269 to 279°C in 2 min. At that moment the temperature, although still rising, was almost stabilized. To avoid an automatic "scram" (set point at 279.5°C) an "all rods in" was initiated and the temperature decreased immediately. This indicates that automatic control of the reactor, which starts an insertion of all rods when nuclear power is 15% above steam power, is not absolutely necessary because the operator has time to intervene.

Other tests (increasing and decreasing load, normal operation etc.) showed that the automatic control of the reactor based on the change of the average temperature is not necessary.

Insofar as the influence of the load-drop tests on the secondary plant is concerned, an automatic load-drop anticipator is put into service following an important load drop. This device operates in the following simultaneous ranges: electrical output of the generator below 20%; and steam-drier pressure corresponding to mechanical turbine power above 60%. It temporarily closes the governing valves of the turbine to avoid a turbine trip due to over-speed. Practice has shown that certain modifications were necessary. The load-drop anticipator can be put on "automatic" or "manual" by a switch.

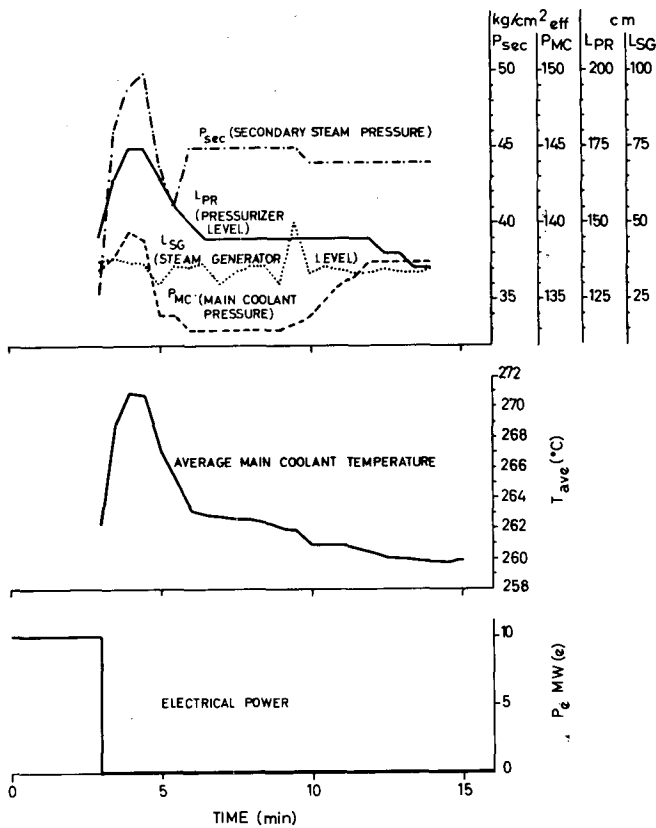


Fig. 11

BR3 Load drop test 10 MW $\rightarrow$ 0 MW; primary parameters
Automatic control; turbine trip by overspeed; turbine by-pass open

In the automatic position (in normal operation) the above-mentioned conditions are still valid, but in addition the device works immediately for 15 s when the tie-breaker opens. The manual position allows for intervention by the operator at any time.

Different parameters, such as turbine speed, regulation oil pressure, functioning of the load drop anticipator, pressure in the moisture separator and current of the primary pump on the auxiliary generator, were measured by means of a fast recorder. The following conclusions can be drawn:

- (i) For the load-drops from different loads to the load of the auxiliaries there was no tripping of the turbine by overspeed;
- (ii) For the load-drops from 10 MW and 11.2 MW to zero the turbine tripped by overspeed. For the test from 10 MW to 0, tripping occurred at 3350 rpm after 4 s, (overspeed trip device: 3350 rpm) for the test from 11.2 MW to 0 at 3380 rpm after 6 s.

Before the modification of the load drop anticipator we also had a turbine trip for the drop from 6 MW to 0 (3360 rpm after 6 s). During this increase

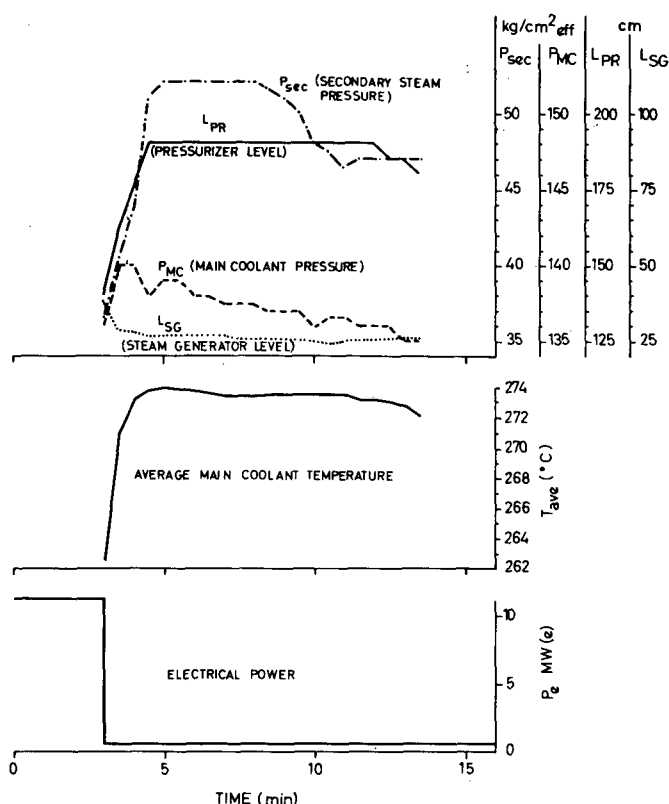


Fig. 12

BR3 Load drop test 11.2 MW $\rightarrow$ 0.38 MW: primary parameters
Turbine trip by overspeed; turbine by-pass open

of speed the voltage of the auxiliary alternator feeding a primary pump decreased to 200 V (normal voltage 380 V) in spite of the total short-circuiting of the excitation rheostat; 26 s after opening of the tie-breaker, the primary pump tripped through direct protection and caused a scram of the reactor, since above 20% of full power the loss of one pump provokes a scram. An analysis is in progress to study the means of decreasing the possibility of this tripping by overspeed.

During a first test from 11.2 MW to 0, a tripping of the turbine occurred by overspeed after 4.5 s and, the turbine by-pass failing to open, submitted the reactor to a severe transient, mentioned above.

3.3. Some facts about the power operation of the plant

3.3.1. Radiation levels in the plant

In normal operation at 7 MW(e) (60% of nominal thermal power) radiation intensities (survey meters) or doses (film-badges, dosimeters) have been determined.

(a) Normal working areas

Auxiliary building:

Basement: around the surge tank of the purification loop: 2-100 mr/h

Pipes-penetration into plant container: 1-8 mr/h

First floor: 0.4 mr/h

Second floor: 0.4 mr/h

Third floor: 0.4 mr/h (in the sampling room: about 2.5 mr/h when taking samples).

Ventilation building:

Suction fans chamber: 0.4 mr/h

Pulsion fans chamber: 0.4 mr/h

Dilution fans chamber: 0.4 mr/h

Plant container:

Operating deck: 0.75 mr/h (except around the reducing stations of the purification and in the neighbourhood of the access to the reactor-pit between the shielding blocks)

(b) Non-accessible areas during operation

All the integration detectors below the operating deck (steam generator compartment under radiation during approximately 110 h at an average load of 60% of nominal load were at full scale; this means 25 r for the gamma film-badges, 50 r for the gamma dosimeters and 120 mrem for the thermal neutrons dosimeters. For the same exposure time (110 h) the following levels were recorded:

On the shielding blocks of the operating deck in the container: 21 mr gammas, 20 mrem thermal neutrons;

Between the shielding blocks: 8 r gammas, 120 mrem thermal neutrons;

At the reducing stations of the purification: 90 mr gammas, 3 mrem thermal neutrons;

At the safety injection penetration: 110 mr gammas;

In the charging and circulating pumps room: 17 mr gammas;

In front of the demineralizers: 17 mr gammas.

These results indicate that the habitual work in the normal areas is not affected by the radiation intensities present.

The hottest spot is located around the surge tank of the purification, with about 100 mr/h, in contact, at the upper part of the purification surge tank due to accumulated radioactive gas, and quickly reduced with distance. Therefore, operating times in this area are strictly controlled.

3.3.2. Air activity in the buildings

The presence of radon was detected in the reactor building and in certain auxiliary buildings, probably due to the concrete. The same problem appeared at the BR2 reactor where the same materials were used at the same time as at BR3.

When the ventilation of the reactor building is stopped, the activity shows a twofold increase. The measured concentrations (ventilation in operation) are: beta gamma: $8 \times 10^{-10} \mu\text{c}/\text{cm}^3$

alpha : $3 \times 10^{-10} \mu\text{c}/\text{cm}^3$

In some buildings where air contamination was observed, measures were taken to avoid it: better separation between rooms, leaktightness of ventilation ducts and of water drain pipes, etc. Air contamination also occurred due to sampling of the primary water. Tight quick disconnects would do a good job in the sample trains.

3.3.3. Water chemistry

Primary loops. The principal chemical prescriptions for the primary water are the following:

pH : 9.5 to 10.5 (LiOH)
Impurity content: 1-2 ppm
Chlorides : 100 γ /l
Silica : 500 γ /l
Oxygen: 140 γ /l (by adding N_2H_4 when cold; H_2 when hot).

The demineralized make-up water is furnished by a central epuration station for BR2 and BR3. It then passes through a vacuum degasifier before being sent to the primary circuits. Various difficulties have arisen with this degasifier. However, after several interventions, e.g. modification of the top of the degasifier and of the aspiration of the incondensable gasses by the ejectors, and correcting and improving the control circuits, satisfactory operation was obtained.

During the tests with boric acid, we operated at pH = 7 to avoid an exaggerated increase of the LiOH concentration.

The use of hydrogen to remove oxygen by radiolytic combination necessitates precautions to avoid accidents, among others the regular analysis of the hydrogen concentration in the different accumulation tanks of the loops. We also noticed a NH_3 formation by radiolytic combination of nitrogen and hydrogen.

We obtained for the activity of the primary water, since the start-up of the station at power, a value of $0.2 \mu\text{C}/\text{cm}^3$ at full power. The spectrometric analysis of the water sampled has shown that besides the presence of activated corrosion products there were also fission products. These probably result from a surface contamination of the fuel elements.

We noticed that the mixed-bed demineralizer retained practically all non-gaseous products and that the gaseous ones were taken out of the water by a degasifying action in the purification system surge tank, from whence they are sent to the stack after dilution. This evacuation, done only under adequate meteorological conditions, is controlled by detection. The operation is closely followed and samples are analysed to determine the exact quantity of I^{131} released to the atmosphere.

The following example will give an idea of this type of evacuation. On 26 December, 1962 we evacuated 220 mc the greatest amount of which was Rb^{88} (half-life: 17 min). The I^{131} concentration was $1.5 \times 10^{-7} \mu\text{C}/\text{cm}^3$, which is below the concentration for the permissible 13 weeks dose to the population.

Secondary loops. The principal chemical components of the secondary water are given in Table V. The demineralized water is provided by the same central station as the primary water. Some difficulties have arisen because

TABLE V

PRINCIPAL CHEMICAL COMPONENTS
OF THE SECONDARY WATER

	Low- and medium- pressure loops	High- pressure loops	Steam
pH	9.2	10 - 11	9.2
O ₂	0 γ/l	0 γ/l	0 γ/l
Cl ₂	50 γ/l	500 γ/l	0 γ/l
SiO ₂	50 γ/l	15 mg/l	-
Fe _T and Cu _T	10 γ/l	500 γ/l	10 γ/l
Free OH	-	0°F	-
PO ₄ ³⁻	-	100-300 mg/l	-
H ₂	-	-	10 γ/l
Total solubles	-	300-500 mg/l	-

the different circuits were insufficiently clean when starting up the plant. Important quantities of iron oxide were collected on the filters of the condensate pumps. The iron and silicon concentrations in the steam generator water were decreased to permissible values by energetic blow-down. Some priming of the steam generator was noticed once or twice. Lowering of the steam generator level and decreasing the phosphate concentration in the feedwater restored normal conditions.

3.3.4. Primary water leaks

The various primary water leaks result primarily from (a) "leak-offs" of valves; (b) the safety valves; and (c) the circulating and charging pumps. Leak-tests were performed with and without the purification loop in operation. The addition of water and its loss were measured by following the level changes in the pressurizer and in the purification surge tank. A first series of tests gave (a) leaks with purification loop in operation: 110 l/h; (b) leaks with purification loop isolated: 65 l/h. A part of these leaks were collected in the discharge tank. This tank collects the leak-offs of the valves and the leaks of the primary plant safety valves and of the steam generator. The most important leaks originated at the plunger packings of the pumps in the purification loop. After replacing the packings of the pumps and after careful inspection of valves and safety valves the leak-rate dropped to 30 l/h.

These leaks, when important, represent serious disadvantages by requiring the frequent addition of make-up water and frequent control of the primary water quality and the addition of chemicals.

3.3.5. Performance of different components and major difficulties

Before criticality hot-run tests were run, many other tests were performed on components and loops. An important source of trouble was leaking valves and safety valves. After a few months of operation the plant had to be shut-down to cold condition to repair leaking safety valves on the pressurizer and in the purification loop.

A high level of noise in the nuclear instrumentation made it necessary to move the pulse integrator of the latter from the control room to a location close to the container of the reactor.

The two canned-motor main coolant pumps have operated over 4000 h with no special difficulties. The magnetic jack type control-rod mechanisms have been working satisfactorily. For same time we noticed that the control rods were slipping. We built in a device that energized the stationary coil before de-energizing the movable gripper and since then we have not observed any further slipping.

The reducing stations in the purification loop have been and still are an almost permanent source of trouble. Unwanted changes in outlet pressure open the safety valves and in that way create primary water losses. As mentioned above, many interventions were necessary on leaking valves.

The circulating and charging pumps, which re-inject the purification water into the main coolant loop, also present many imperfections. The piston packings have to be replaced regularly (± 700 h) to reduce primary water leaks. The pistons were seriously damaged after 610 h (charging pump) and 1200 h (circulating pump) of operation. The breakdown of the speedchanger belts caused a shut-down. The charging pumps speedchanger shaft ruptured after 1720 h of operation.

There was some trouble with damaged bearings due to dirt in the turbine oil circuits.

3.3.6. Scram analysis

Since initial criticality and up to 15 March, 1963 a total of 20 scrams has been recorded. Seven of these were initiated voluntarily for test purposes (e. g. checking of the interlocks and the automatic actions following a scram). Four of them were due to failure of the instrumentation (e. g. periodmeter, flowmeter in the main coolant loop, protection relay of the primary pump No. 1, etc.), three to an operator error, one by injecting too large a quantity of relatively cold water into the steam generator, one by tripping a primary pump during a load drop test, one by short period during a test, one by testing the nuclear instrumentation, one at the end of an "all rods in" by low-pressure signal and one by a loss of direct current to the control rods. It is worth noting that the frequency of scrams has now been seriously reduced.

CONCLUSIONS

Up to now the BR3 plant has been working in a satisfactory manner. The difficulties we have had were primarily with the "classical" equipment.

Furthermore BR3 is an outstanding installation for training people for larger nuclear stations.

A broad test programme has also been elaborated: operation without demineralizers, lowering of the pH, testing new types of fuel elements, etc. Also incidents will be simulated: loss of compressed air, loss of direct or alternative current, etc.

Apart from this, a lot of useful data will be obtained from the BR3 plant, in that way emphasizing the efforts which have been accomplished by the CEN and Belgian Industry.

DISCUSSION

A. WECKESSER: Can you tell me the response time of the by-pass valves? And what is the high-power "scram" setting?

M. POTEMANS: The turbine by-pass valve opens on a "High steam pressure" signal. When, for example, the turbine is tripped from full power (11.2 MW) to zero, the pressure increases from the normal operating value (36 kg/cm²) to the value at which the turbine by-pass valve opens (50 kg/cm²) in about one minute. The by-pass valve itself (pneumatically operated) takes about one second to open. The high-power "scram" setting is 120%.

Z. PELLED: It seems that you have been using boron concentration as a sort of independent variable. How accurately can you measure the boron concentration, and how do you remove specified amounts of boron from the circuit when necessary?

M. POTEMANS: The boron was removed by a "feed-and-bleed" operation. The accuracy of the measurement of boron concentration is of the order of 1 to 2 ppm.

OPERATING EXPERIENCE WITH INDIAN POINT NUCLEAR ELECTRIC GENERATING STATION

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Abstract — Résumé — Аннотация — Resumen

OPERATING EXPERIENCE WITH INDIAN POINT NUCLEAR ELECTRIC GENERATING STATION. Indian Point Station Unit No. 1 consists of a 585 MW(t) pressurized-water reactor, four primary coolant loops with horizontal heat exchanger boilers, two 1.1 million lb/h oil-fired superheaters and a 275 000 kW turbine generator. The reactor fuel is a mixture of fully enriched U^{235} oxide and Th^{232} oxide.

The station is located on the Hudson River about 24 miles north of New York City. Because of this proximity to New York, exceptional safeguards against the occurrence of a reactivity excursion as well as against the radiation effects of such an excursion were incorporated into the station design.

Construction was completed in May 1962. Fuel loading was accomplished in June and the reactor was taken critical for the first time on 2 August 1962. Low power testing up to 5 MW(t) at ambient and at elevated temperatures was done during August, and the turbine generator was first phased into the Consolidated Edison system on 16 September 1962.

Testing at reactor power levels up to 50% extended into November and was marked by frequent automatic shut-downs, a large number of which were initiated in the conventional plant. Control-rod-drive control system difficulties were the heaviest contributor from the nuclear plant to automatic rod insertion operations and to delays in recovery from automatic trips.

On 14 November 1962 the station was shut down for scheduled piping changes in the conventional plant and for modifications and additions to the control-rod-drive system. The latter included the installation of a dry nitrogen purging system for the control-rod-drive housings designed to minimize the effects of seal water leakage into the rod-drive housings. This appears to have been the major cause of the false indications encountered with the reactor control system.

The unit was returned to service on 1 January 1963. Testing at reactor power levels up to 100% under steady load conditions was completed on 27 January 1963. Test results have followed closely the predicted performance for the reactor. Further testing of response to load transients is to continue into the Spring of 1963.

The Indian Point Station has proved to the satisfaction of our Company the feasibility of a combination nuclear and oil-fired plant to produce competitive power. The United States Atomic Energy Commission has been requested to issue a construction permit for the Company to build another nuclear plant, similar in design, but of 1000 MW capacity, to be located at Ravenswood which is within the corporate limits of the City of New York. Our Company is prepared to finance the venture with its own funds without subsidy of any kind as was the case with our Indian Point Station.

EXPÉRIENCE D'EXPLOITATION DE LA CENTRALE NUCLÉAIRE D'INDIAN POINT. La centrale N° 1 d'Indian Point se compose d'un réacteur à eau sous pression de 585 MWt, quatre circuits de refroidissement primaires avec des échangeurs de chaleur horizontaux, deux surchauffeurs au mazout de 500 000 kg/h et un turbogénérateur de 275 000 kW. Le combustible du réacteur est un mélange d'oxyde d'uranium complètement enrichi en ^{235}U et d'oxyde de ^{232}Th .

L'installation se trouve au bord de l'Hudson à une quarantaine de kilomètres au nord de New York. En raison de la proximité de cette ville, on a prévu des dispositifs de sécurité intrinsèques exceptionnels contre le risque d'un emballement du réacteur et contre les rayonnements qui pourraient en résulter.

La construction a été terminée en mai 1962. Le combustible a été chargé en juin et le réacteur est entré en divergence pour la première fois le 2 août 1962. On a ensuite procédé, dans le courant du mois d'août, à des essais à faible puissance (jusqu'à 5 MWt), à la température ambiante et à des températures élevées. Le turbogénérateur a été intégré pour la première fois au réseau de la « Consolidated Edison », le 16 septembre 1962.

Les essais aux puissances allant jusqu'à 50% de la puissance théorique se sont prolongés jusqu'en novembre et ont été marqués par de fréquents arrêts automatiques dont un grand nombre avaient leur origine dans le

secteur «classique» de la centrale. Ce sont surtout des défauts du système de commande des barres de contrôle qui, dans le secteur nucléaire, ont contribué à la chute automatique des barres et aux retards dans la remise en marche après arrêt.

Le 14 novembre 1962, on a arrêté la centrale pour procéder aux changements prévus dans les tuyauteries de la partie classique et pour apporter des modifications ou additions aux organes de commande des barres de réglage (installation d'un purgeur à azote sec pour les logements des organes de commande, afin de réduire les effets des fuites d'eau à travers les joints). Il semble que c'était là la cause principale du mauvais fonctionnement du système.

La centrale a été remise en service le 1er janvier 1963. Les essais, jusqu'à 100% de la puissance théorique à charge constante, étaient terminés le 27 janvier 1963. Les résultats concordent parfaitement avec les prévisions. De nouveaux essais pour étudier la manière dont le réacteur répond à des variations subites de la demande se poursuivront au printemps 1963.

ОПЫТ ЭКСПЛУАТАЦИИ ИНДИАН-ПОЙНТСКОЙ ЯДЕРНОЙ ЭЛЕКТРОСТАНЦИИ. Блок № 1 Индиан-пойнтской ядерной электростанции состоит из реактора с водой под давлением мощностью 585 мвт, четырех первичных контуров охлаждения с горизонтальными теплообменниками, двух работающих на нефти пароперегревателей производительностью 1 100 тыс. фунтов/час и турбогенератора мощностью 275 тыс. квт. Топливом в реакторе служит смесь полностью обогащенной окиси U^{235} и окиси Th^{232} .

Станция расположена на реке Гудзон, примерно в 24 милях севернее Нью-Йорка. В связи с такой близостью от Нью-Йорка в проекте станции были предусмотрены дополнительные предохранительные устройства для предотвращения отклонения реактивности и радиационных эффектов такого отклонения.

Строительство закончено в мае 1962 года. Загрузка топлива осуществлена в июне, а 2 августа 1962 года реактор впервые достиг критичности. В течение августа проводились испытания на малой мощности до 5 мвт при температуре окружающей среды и повышенной температуре. 16 сентября 1962 года турбогенератор был впервые фазирован с системой Консолидейтид Эдисон.

Испытание реактора на мощностях до 50% проводилось до ноября и прерывалось частыми автоматическими остановками, большая часть которых была вызвана неполадками в неядерной части станции. Неполадки в системе управления приводов регулирующих стержней были самыми серьезными помехами со стороны ядерной части станции для работы с автоматическим введением регулирующих стержней и причиной задержек с пуском реактора после автоматических остановок.

14 ноября 1962 года станция была остановлена для плановой замены трубопроводов в неядерной части станции и модификаций и дополнения системы проводов регулирующих стержней. Последнее включало установку системы очистки сухого азота для кожухов приводов регулирующих стержней, предназначенных для сведения к минимуму эффектов воды, просачивающейся через уплотнения в кожухи регулирующих стержней. Оказалось, что это является основной причиной неправильных показаний системы управления реактором.

Блок был вновь пущен 1 января 1963 года. Испытание реактора на мощностях до 100% в условиях стабильной нагрузки было закончено 27 января 1963 года. Результаты испытаний очень хорошо согласуются с расчетными характеристиками работы реактора. Следующие испытания чувствительности реактора к переходным колебаниям нагрузки должны быть продолжены весной 1963 года.

EXPERIENCIA ADQUIRIDA CON LA EXPLOTACIÓN DE LA CENTRAL NUCLEOELÉCTRICA DE INDIAN POINT. El grupo № 1 de la central de Indian Point consta de un reactor de 585 MW(t), de agua a presión cuatro circuitos refrigerantes primarios con intercambiadores de calor horizontales, dos sobrecalentadores a petróleo de 500 000 kg/h y un turbogenerador de 275 000 kW. El combustible del reactor consiste en una mezcla de óxido de ^{235}U totalmente enriquecido y de óxido de ^{232}Th .

La central se encuentra a orillas del río Hudson, a uno 40 km al norte de Nueva York. Debido a la proximidad de esa ciudad se adoptaron dispositivos especiales de seguridad intrínseca para evitar saltos bruscos de reactividad y prevenir sus efectos radiológicos.

La construcción quedó terminada en mayo de 1962. Se cargó el combustible en junio y el reactor alcanzó la criticidad inicial el 2 de agosto de 1962. En el curso del mismo mes, se efectuaron los ensayos de baja potencia, hasta 5 MW(t), a la temperatura ambiente y a temperaturas elevadas, y el 16 de septiembre de ese año el turbogenerador se conectó por primera vez a la red de la Consolidated Edison.

Hasta entrado el mes de noviembre, prosiguieron los ensayos a potencias de hasta 50% del valor teórico, con frecuentes detenciones automáticas, originadas en gran parte por perturbaciones en la sección «clásica» de la central. En cuanto a la sección nuclear, fueron sobre todo las fallas del mecanismo de mando de las

barras de control que contribuyeron a la caída automática de las barras y a las demoras en la reiniciación de la marcha después del paro.

El 14 de noviembre de 1962 se detuvo la marcha de la central para proceder a los cambios previstos en las tuberías de la sección «clásica» y para modificar y perfeccionar el mecanismo de mando de las barras de control. Esta última operación comprendía la instalación de un dispositivo de purga a base de nitrógeno seco para las cajas de los mecanismos de las barras de control, a fin de reducir al mínimo los efectos de los escapes de agua, a través de las juntas. Al parecer, dichas pérdidas eran la causa principal de las indicaciones erróneas registradas en el sistema de control del reactor.

La central se volvió a poner en servicio el 1º de enero de 1963. El 27 de enero de 1963, se dio fin a los ensayos del reactor, con potencias hasta el 100%, en condiciones de carga constante. Los resultados de los ensayos concordaron satisfactoriamente con las previsiones. En la primavera de 1963 se continuarán las pruebas, para estudiar la manera en que el reactor responde a fluctuaciones rápidas de la demanda.

1. STATION DESCRIPTION

Indian Point is a 275 000 kW electric generating station located on a 250-acre site on the East bank of the Hudson River approximately 24 miles upstream from the northern limits of the City of New York. It employs nuclear fuel for production of saturated steam and takes advantage of the increased capacity and added economy derived from superheated steam by using oil-fired superheaters. The station output is delivered to the 138 kV transmission lines of the Consolidated Edison Company system which serves a 600 mile² franchise area having a load density rated among the highest in the world. The electric operating territory and principal facilities of this system are shown in Fig. 1.

2. ENVIRONMENTAL CONSIDERATIONS

While the station is in a relatively sparsely settled section of the area served by the Company, there are approximately 62 000 inhabitants within a five-mile radius of the site. Accordingly, the station is designed to minimize the effects of any credible radiation incident.

The containment vessel is a steel sphere, 160 ft in diam., designed as a 25 lb/in² gauge pressure vessel in accordance with Section VIII of the Boiler and Pressure Vessel Code of The American Society of Mechanical Engineers. It is set approximately to its equator in a solid rock foundation. A cylindrical wall of ordinary concrete 5½-ft thick with a dome 2 ¾-ft thick surrounds the upper half of the containment vessel and serves as an outer biological shield. It is so designed that even in the event of a maximum credible incident, radiation exposure at the site boundary would not exceed 0.057 mr/h during the first week following the incident. Thus, evacuation of the local area would be unnecessary, and station equipment on the site could continue normal operations with plant personnel in attendance.

The top of the superheater stack is 470 ft above the Hudson river. The containment ventilation and waste-gas discharge duct enters the side of the stack about 70 ft below its top and extends upward on the stack centre-line approximately 10 ft above the top of the stack. The combustion gas resulting from the oil-fired superheaters discharges to an annulus around the ventilation duct thereby aiding in the acceleration and dispersion of the waste gases. No gaseous waste is released until it has been in storage in holdup tanks for about three months to assure the decay of iodine isotopes.

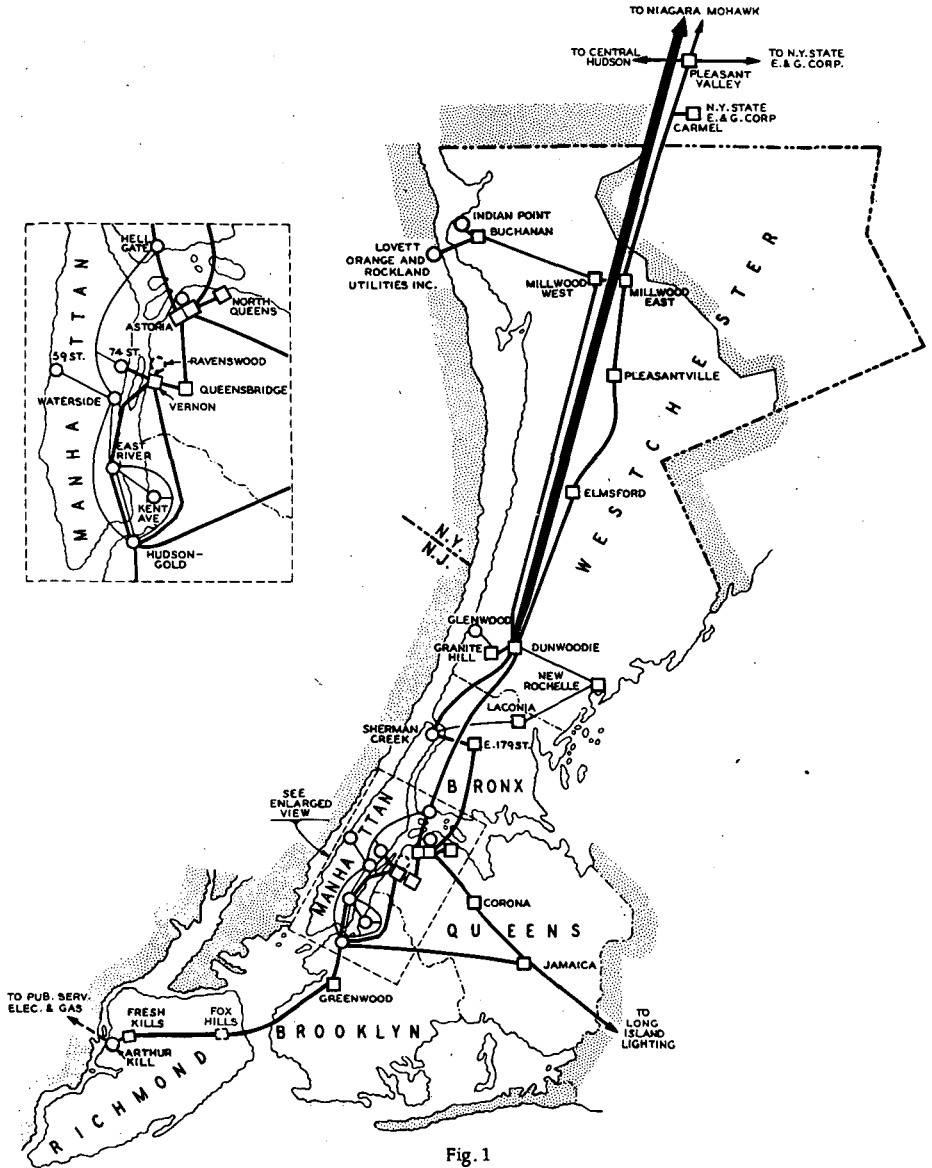


Fig. 1

Electric operating territory and principal facilities
(31 December 1951)

- Generating stations
- Principal sub-stations
- ▨ Operating area
- 345 kV tie feeders
- 138 kV tie feeders
- Other tie feeders
- - - Indicates projects under construction

Liquid wastes containing radioactivity too high in concentration for direct discharge to the river are evaporated in a system designed for the treatment of chemical wastes. The condensed distillate is checked for its level of radioactivity and pumped to the plant discharge canal where it is diluted by the main condenser circulating water and discharge water from other station uses. The evaporator sludges are removed to storage tanks from which they are periodically pumped to a concrete mixing plant. This concrete mixture is placed in drums for off-site burial. Other solid waste is either baled or drummed for off-site disposal.

All equipment in the chemical treatment systems is conservatively designed to assure adequate capability under peak load conditions to meet the requirements of the United States Code of Federal Regulations Title 10, Part 20 for the release of waste to the contiguous environment.

3. NUCLEAR PLANT

The nuclear facility includes a 585 MW pressurized light-water reactor with four primary-coolant loops having horizontal U-tube heat exchangers as steam generators. The major components of the primary system were designed and fabricated by The Babcock and Wilcox Company. The reactor vessel is of SA 212 Grade B carbon steel, clad on its internal surface with 304 stainless steel. It is designed for 1800 lb/in² gauge, 650°F in accordance with Section I of the Boiler and Pressure Vessel Code of The American Society of Mechanical Engineers. The reactor vessel is about 38-ft high with an inside diameter of 9 ft 9 in.

The core is an approximate right cylinder 7-1/2 ft in diam., consisting of 120 fuel elements, with associated shim rods and flux depressors. Each fuel element is made up of 195 fuel rods in a Zircaloy container. Each fuel rod is a stainless-steel tube 0.304 in in diam., containing sintered pellets of a mixture of fully enriched uranium oxide and thorium oxide to give an active fuel length of approximately 98.5 in. The elements are furnished with three different fuel loadings to provide for a three-zone core.

21 movable control rods of 95.4% hafnium are provided. Each control rod is positioned by a drive mechanism, through a drive line assembly, consisting of the control rod, a Zircaloy-2 follower rod, a snubber shaft and housing, a seal shaft and buffer seal housing and the drive mechanism. Each drive line assembly is mounted to a flanged nozzle on the bottom of the reactor vessel.

An electric drive motor in the drive mechanism turns two parallel lead screws on which a carriage runs up or down depending upon the direction of motor rotation. A coupling spacer attached to the seal shaft is held up against the carriage by hydraulic oil pressure applied to a piston at the lower end of the shaft. Thus, the assembly moves the control rod up out of the core, or down into the core, in response to signals to the drive motor from the control system. The control rod moves rapidly downward into the core under the combined forces of gravity and reactor pressure when the hydraulic pressure is released by a solenoid operated scram valve in response to a signal from the reactor safety system.

Each control-rod drive mechanism is enclosed in a housing designed as an 1800 lb/in² gauge pressure vessel. This arrangement is to ensure retention of primary coolant in the event of a failure of the seal water pressure which might be followed by excessive leakage of primary coolant water along the shaft.

Mechanical levers, cams, and shafts within this housing actuate small switches in electric circuits of the drive control system. The engagement of the coupling spacer with the carriage, and the upper and lower limits of rod travel are thus monitored to provide signals for the rod control system.

Four or five rods symmetrically distributed among the elements constitute a rod group. There are five rod groups which are withdrawn sequentially for power operation. Symmetrical power distribution is expected at all flux levels throughout core life.

Nuclear instrumentation includes 25 channels whose detecting elements are external to the reactor vessel. There are no in-core monitors. The detectors are mounted in tubes immersed near the inner cylindrical surface of the neutron shield tank.

Two low level channels use BF₃ proportional counters. Two channels have U₃O₈ fission counters and provide response overlap of about two decades of power with the proportional counters. Three channels have compensated ion chambers for power level indication up to 100% and overlap the fission counters by about two decades. These seven detectors are located at the midplane elevation of the core.

The remaining 18 channels have uncompensated ion chambers as detectors. These chambers are arranged in three planes of six detectors each, one at the top, one at the middle and one at the bottom of the core. In each plane, the detectors are spaced 60° apart. They provide neutron flux information up to 150% power.

There are two 16 000 gal (US)/min canned motor pumps in each primary coolant loop to circulate the primary coolant through the stainless-steel tubes of the heat exchanger designed to transfer 500 million BTU/h. Each boiler is rated to deliver 550 000 lb/h saturated steam at 405 lb/in² gauge. All four loops are wholly within the containment vessel.

There are approximately 150 piping penetrations of the containment vessel. Each pipe penetration has two valves installed as containment isolation valves located a short distance outside the containment vessel. Every welded pipe joint within the containment vessel and to the isolation valves outside the containment vessel was X-rayed to ensure acceptable welds.

A plant controller is provided to maintain 405 lb/in² gauge steam pressure at the boiler outlet by automatic operation of the control-rod drive controls. Parameters fed into this controller include the steam pressure at each boiler outlet, the main steam flow and the average of the neutron flux seen by three of the uncompensated ion chambers at the midplane.

Holding steam pressure constant for all loading conditions results in a rising characteristic in both reactor inlet and reactor outlet temperatures as the power level increases. The 450°F zero-power-level temperature is designed to increase to 486.5°F inlet temperature and 519°F outlet temperature when the reactor reaches 100% power.

4. CONVENTIONAL PLANT

In the conventional plant, the steam leads from the four boilers connect to an inlet header for the two oil-fired superheaters. Each superheater is designed to raise the temperature of 1.1 million lb/h of dry and saturated steam at 405 lb/in² gauge from 449 to 1000°F. Four steam leads convey the steam from the superheater outlet mixing chamber to the turbine. The turbine generator is rated at 275 000 kW when supplied with 2 256 000 lb/h steam at 370 lb/in² abs. and 1000°F, and 1.0-in Hg abs. exhaust pressure.

A controlled start line is installed to by-pass steam from either superheater outlet directly to the main condenser. This arrangement permits the matching of the superheater outlet steam temperature with the metal temperature of the turbine casing when preparing to roll the turbine.

The major items of the conventional plant facilities are shown on the plant heat balance diagram Fig. 2.

5. PRE-CRITICAL OPERATIONS

Preliminary testing of the nuclear plant systems started in August 1961. In November the primary coolant system was flushed with primary grade water at full pressure and temperature. Individual 50- μ m filters were installed in the inlet ends of the heat exchanger tubes for the flushing operation. The primary coolant pumps and their discharge check valves were tested for their flow characteristics at this time.

Full-scale testing of the auxiliary systems of the nuclear plant continued into March 1962. This work was accomplished by the permanent station operating crew supervised by the 19 licensed reactor operators who had passed the written and oral examinations given by the United States Atomic Energy Commission examiners in December 1961.

An 18-month Provisional Operating Licence was issued on 26 March 1962 and construction of the facility was completed in May of that year.

Fuel loading was accomplished in June using the remote fuel handling system. This method was employed for two reasons: first, to train the personnel in the technique of handling fuel remotely under water and secondly, to test the apparatus before handling the spent fuel elements during a core reloading. During the operation minor defects in the fuel handling equipment and the fuelling procedures were observed and corrected.

A loading sequence for the core had been developed during an assembly of the core components at The Babcock and Wilcox Company's Critical Experiment Laboratory in the summer of 1961. To ensure that the results followed closely those of the previous assembly, two temporary internal BF₃ counters were placed in the vessel and count rates were taken as each control rod, shim rod or fuel element was installed. Closed circuit television and direct vision were used to determine that the core components were installed in the correct location and with the proper orientation. A plan view of the core is shown in Fig. 3.

The central control rod and its surrounding four fuel elements, each containing a polonium-beryllium neutron source, were the first to be installed. Loading of the core then proceeded with the installation, one at

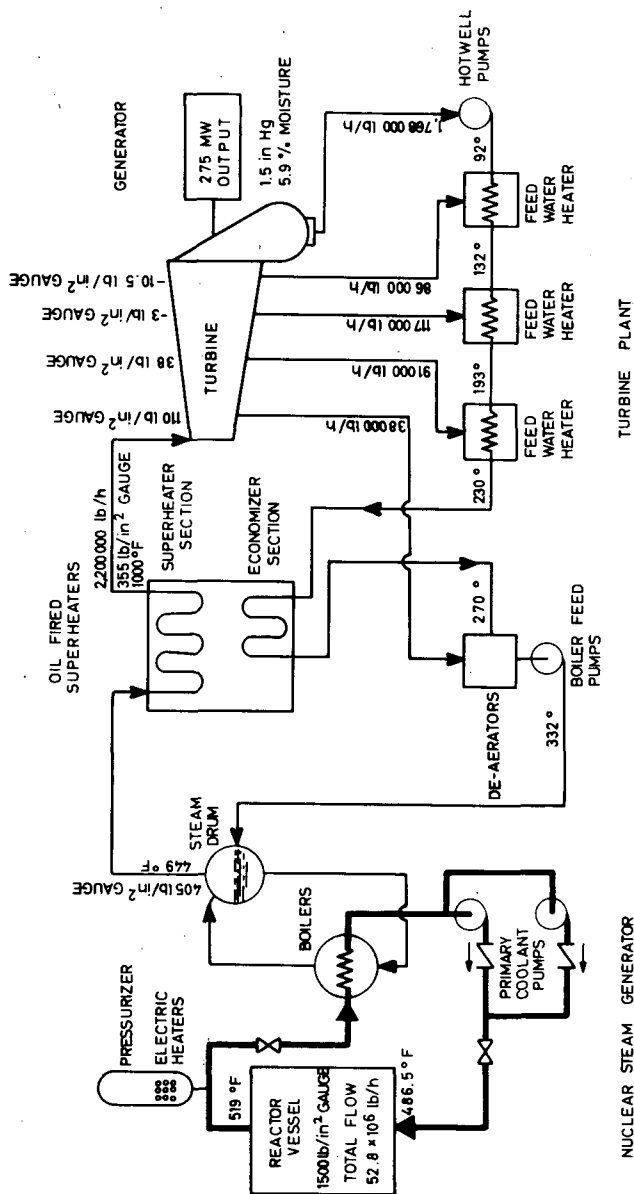


Fig. 2

Indian point heat balance 275-MW nuclear plant

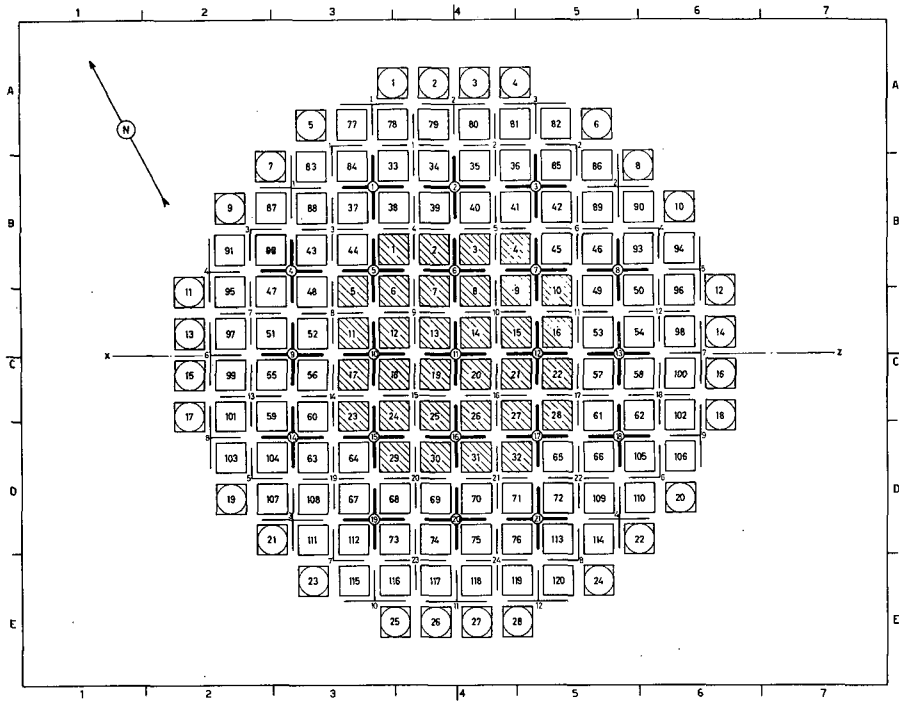


Fig. 3

CETR core hardware position designations

-  Fuel element position zone I
-  Fuel element position zone II
-  Fuel element position zone III
-  Hold-down column position
-  Movable-rod position
-  Fixed-rod position
-  T-shaped flux depressor
-  Angle-shaped flux depressor
-  Cruciform-shaped flux depressor

FE Fuel element
CR Control rod
FR Follower rod
FSR Fixed shim rod
FFR Fixed filler rod

FDC Flux depressor, cruciform
FDA Flux depressor, angle
FDT Flux depressor, T-shaped
HDC Hold-down control
FES Fuel element with source

a time, of a control rod or fixed shim with its adjacent four elements to complete a series of concentric rings. No attempt was made at any intermediate step to make the reactor critical.

The water treatment plant and storage system had been unable to provide water of sufficient clarity to give the required visibility in the fuel handling pools. The temporary installation of a portable wood cellulose filter which removed particles down to 2- μ m size solved this problem. Despite these excellent final conditions, however, complete indexing of the core internals was recorded for future reference.

The design of the reactor head joint employs two gaskets plus welding for tightness if required. Tests at the time of flushing indicated that these gaskets were adequate for leak-tight closure. Therefore, when the head joint was bolted after fuelling, the seal ring inside the bolt circle was welded to the head only and not to the body of the vessel. No leakage has been detected through all of the temperature cycles which have followed.

Provision had also been made in the design for seal-welding other bolted joints on equipment in the primary system. One pressurizer vent valve bonnet has been seal-welded, and the hinge pin covers on all eight discharge check valves for the primary coolant pumps have been seal welded after two leaked at the hinge pin packing during heating and cooling cycles. In each case the leakage had stopped without any required maintenance operations, but since coolant leakage could be expected to contaminate a large area if permitted to continue for a long period, it was decided that all covers should be seal welded.

6. LABORATORY TESTS

The assembly of the core components at The Babcock and Wilcox Company's Critical Experiment Laboratory in 1961 had been followed by a series of experiments undertaken to minimize the extent of the testing necessary at the Indian Point Station. The major determinations were:

(1) The mechanical compatibility of the core components. This was confirmed by the ease of assembly at Indian Point.

(2) The reactivity in the core and the shut-down margins of various control-rod configurations. Methods used for this included partial water height, variation of boric acid concentration, and a pulsed neutron technique. The average value of the various methods indicated a core reactivity of 0.113 at 68°F.

(3) The boric acid concentration required to ensure a shut-down margin of 0.056 when in the cold, clean condition with all control rods fully inserted. This concentration was 930 ppm as boric acid.

(4) The number and final design of the fixed shim rods needed to hold down excess reactivity during the first half of core life. Twelve shim rods have been installed. All are expected to be removed in Autumn 1963.

(5) The flux distributions for various rod patterns. Since there is no in-core monitoring equipment, symmetrical power distribution is important. Measurements and extrapolations of data to power conditions indicated conformance with calculations to a high degree. About one-sixth of the core is scheduled for examination by a gamma-scanning technique during the Autumn 1963 outage to study the power distribution further.

7. INITIAL CRITICALITY AT INDIAN POINT STATION

In July 1962, the primary coolant system was raised in pressure and temperature to operating conditions for the operational and scram time tests of the control rods and drives. Every rod scrammed from the fully withdrawn position to the two-thirds inserted position in less than the required 1.0 s.

For the initial approach to criticality the system was cooled to 107°F and reduced in pressure to 125 lb/in² gauge. The coolant remained borated at 960 ppm. The low level period fast insertion and scram values were set at 100 s and 80 s respectively instead of the design values of 35 and 10 in order to ensure conservative operator actions. No primary coolant pumps were in service.

The approach to criticality started at 4.15 p.m. on 2 August 1962. As each increment of group rod withdrawal was completed, the advance in count rate was measured, normalized to the initial count rate, and plotted as an inverse function. The curve developed by this method is shown in Fig. 4.

Criticality was achieved at 5.42 p.m. 2 August 1962, with two control-rod groups fully withdrawn and the third group withdrawn 15.8 in.

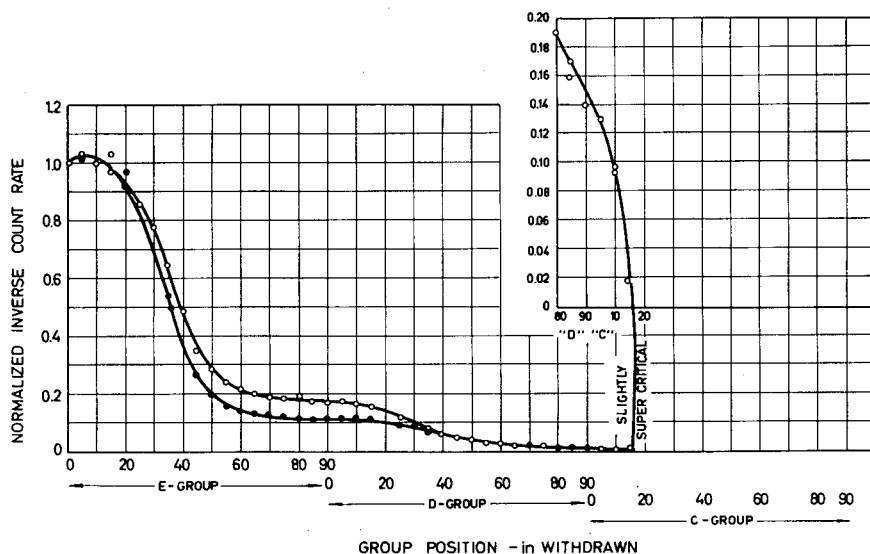


Fig. 4

Curve of normalized inverse count rate against rod withdrawal position
for initial approach to critical (3 Aug. 1962)

Base count rate with all rods in:

channel 1 = 175 counts/min

channel 2 = 163 counts/min

● Channel 1

○ Channel 2

Further testing with the same coolant conditions demonstrated the negative reactivity effect of a 2°F rise in coolant temperature as the first sensible heat was developed.

8. LOW-POWER TESTING

For the remainder of August 1962, testing at power levels up to 5 MW(t) was continued. The moderator temperature coefficient was measured with and without boric acid at various temperatures and pressures up to 450°F and 1485 lb/in² gauge. This coefficient was found to be negative under all measured conditions, becoming increasingly negative for the higher temperatures. Reduction of boric acid concentration caused it to become still more negative. The measured coefficient at 450°F with no boric acid was $-1.73 \times 10^{-4} \Delta k/k \cdot ^\circ F$ whereas the calculated value at 500°F was $-1.73 \times 10^{-4} \Delta k/k \cdot ^\circ F$.

The pressure coefficient was measured by doubling time tests at 1450 and 1650 lb/in² gauge. The measured value was $+1.31 \times 10^{-6} (\Delta k/k)/(lb/in^2)$ as opposed to $+2.0 \times 10^{-6} (\Delta k/k)(lb/in^2)$ as calculated.

The boric acid coefficient of reactivity measured at 450°F was -2.16 ± 0.1 cent/10 ppm, which was in close agreement with the value of -2.25 ± 0.1 cent/10 ppm established at the Laboratory at 68°F in 1961.

Since a design criterion required that the clean core be sub-critical at 425°F with the two most worthy rods jammed in the fully withdrawn position, measurements were made with no boric acid in the primary coolant system at temperatures down to 375°F and extrapolations indicated that the core would just be critical at 275°F.

9. POWER-LEVEL TESTING

The reactor power level was carried at 4% for about two weeks in August and September 1962 to produce sufficient steam from the nuclear boilers to flush the main steam and condensate systems.

De-aerated water was then available to the nuclear boilers through the main boiler feed piping. While the steam and condensate lines had been cleaned with citric-formic acid and phosphoric acid with subsequent passivation, there were sporadic indications of phosphate and chlorides during the flushing. As the condensate system became cleaner, the reactor power level was increased to about 15% for the initial trials of the superheaters and the controlled start line.

The turbine generator was synchronized to the Consolidated Edison System for the first time at 7.37 p. m. on 16 September 1962. The machine was loaded to about 40 MW.

Power level tests under steady load and equilibrium xenon conditions were made at 25 and 50% by 5 October. Deficiencies in conventional plant piping prevented operation at reactor power levels over 50% until after a six-week outage starting 14 November. Upon return to service 1 January 1963 the tests at 75, 90 and 100% rated reactor power were run. The maximum generator load developed at 100% power was 277 MW with 2 223 000 lb/h of steam at 365 lb/in² gauge, 998°F, and 0.5 in Hg abs. exhaust pressure.

Steady-state power testing was completed in February. Transient tests including ramp changes up to 25 MW/min and step changes of 50 MW are scheduled for completion in Spring 1963.

As the testing progressed the reactor core responded as predicted. There were, however, 187 automatic shut-downs in the first six months of operations due to malfunctions of associated equipment and to the conservative settings of the nuclear instrumentation during the testing period. At no time has it been necessary to initiate a manual fast insertion or scram for reasons of reactor safety.

Of the automatic shut-downs, about 20% were from the nuclear instrumentation system for the reason noted above, about 25% were due to malfunctions of the control-rod drive control system and about 30% occurred as fast insertions as the result of malfunctions in the conventional plant equipment and were often followed by scram action in the nuclear plant. The remainder were due to miscellaneous causes, generally of a non-repetitive nature such as isolated component failures.

10. MAJOR MAINTENANCE AND DESIGN CHANGES

Control-rod drives

During the early testing it became evident that some of the minor items in the control system for the control-rod drives would require redesign. The totally enclosed housings were subjecting the electrical circuits and small shaft bearings in the drive mechanism to a highly humid atmosphere. As a result, small switches dependent upon slight mechanical movements were not providing proper information as to rod position, or were falsely transmitting signals of unsafe conditions with consequent operation of the reactor safety system.

Fabrication and procurement of the necessary replacements were completed in time for the work to be done during the outage arranged for required changes in the conventional plant. The following replacements were made for the twenty-one mechanisms:

- (a) An angle bar in the system which monitors the engagement between the carriage and coupling spacer was replaced with a heavier section;
- (b) Small shaft bearings were replaced with sealed bearings lubricated by a grease which is less susceptible to deterioration in a humid atmosphere;
- (c) Switches in the engagement and lower limit circuits were replaced by switches of a slightly different design; and
- (d) The mounting brackets for these switches were replaced by brackets permitting better adjustment.

These minor changes became the controlling item of the maintenance outage because the two sections of each drive mechanism had to be completely disassembled to provide access to the areas of work.

As an additional precaution to guard against a recurrence of the high humidity condition, a purge system was installed to provide a supply of dry nitrogen to the drive housings to carry off the moisture.

The sum total of the correction measures indicated above has been virtual elimination of troubles with the rod-drive system.

11. CONVENTIONAL PLANT PIPING

Two piping changes in the conventional plant were necessary to permit operation at power levels over 50%. The controlled start line had been designed with its cold end terminating in a small flash tank instead of a large manifold so that sufficient steam could not be passed through the line to place both superheaters in service. Also, the condensate inlet piping to the twin de-aerators was such that the second de-aerator could not be put in service because of excessive steam and water hammer. Both of these deficiencies were corrected during the outage in November and December.

There have been other components of the station which have required corrective measures, but none of these has had the long-lasting limiting effects on the plant operations which were contributed by the above two considerations. An oil whip developed in the sleeve bearings of two of the three 22-stage vertical centrifugal pumps supplying the control-rod-drive buffer seal water and primary make-up water. Motors with anti-friction bearings have been installed and there has been no recurrence of this trouble.

Canned motor-pump units in the chemical systems have required continued maintenance. Some of this trouble appears to have been caused by hydraulic unbalance, which has been corrected. A few pumps are still undergoing tests to determine what changes in design are required.

12. RADIO CHEMISTRY

During the testing up to 50% reactor power the gross gamma activity in the primary coolant averaged $0.2 \mu\text{c/ml}$. During operations between 75 and 100% power the gamma activity ranges between 0.2 and $0.3 \mu\text{c/ml}$. Activation products such as Mn^{56} account for a large portion of this gamma activity. Because of the short half-life of Mn^{56} , the activity varies rapidly with power level and reduces to approximately $3 \times 10^{-3} \mu\text{c/ml}$ shortly after shut-down.

The primary blowdown and purification system has operated satisfactorily. The delay time in the blowdown storage tanks accounts for a reduction to approximately $10^{-2} \mu\text{c/ml}$ in the average activity in the water going to the purification system. At the outlet of the cation exchangers the activity is approximately $3 \times 10^{-3} \mu\text{c/ml}$ in the undegassed sample. This is all anion activity. At the outlet of the mixed bed exchangers the activity is about $2 \times 10^{-3} \mu\text{c/ml}$ due to Xe^{133} , Xe^{135} , Kr^{87} and Kr^{88} passing through the mixed bed plus Xe^{133} and Xe^{135} from the decay of I^{133} and I^{135} on the anion resin. Degassed mixed bed effluent is about $10^{-5} \mu\text{c/ml}$.

Effluents from the waste treatment systems range in the order of 10^{-7} to $10^{-6} \mu\text{c/ml}$.

There is no measurable leakage of primary coolant into the secondary side of the steam generators. Similarly, there has been no measurable leakage into the fresh-water cooling system serving equipment such as the primary-coolant-pump cooling-water jackets.

The chemical performance of all systems in general has been in accordance with expectations.

13. HEALTH PHYSICS

While there has been an occasional exposure of personnel to fields as high as 25 - 30 mr/h, no one has been subjected to such fields long enough to absorb dosages even approaching allowable limits as shown by pocket dosimeters or routine development of film badges at monthly intervals.

Measurements of air and water samples on and off site have not shown any increase in activity due to the plant operations. No gaseous waste from the chemical systems has been discharged to date, and ventilation of the containment vessel to purge it before access has required no exceptional precautions.

Gross gamma measurements at four locations in and around the Indian Point Station from August 1962 through January 1963 averaged 0.02 mr/h. The background measurements for the previous five years ranged from 0.015 to 0.023 mr/h.

14. OPERATING ORGANIZATION

The basic plant organization is shown in Fig. 5. Because of the unusual nature of the station, during start-up there have been modifications which are expected to be of relatively short duration.

The General Superintendent presently has an Assistant General Superintendent, and the Superintendent title is presently held by two men, a Reactor Engineer and a Conventional Plant Superintendent. Each of the men in these titles holds an United States Atomic Energy Commission operator's licence.

The function of shift Watch Foreman is being performed by two men, one of whom is a licensed reactor operator. The other is a qualified conventional-plant foreman training for his licence. Similarly, the function of the shift Operator A is being performed by two men, one of whom is a licensed operator. The other is a conventional-plant operator training for his licence. In each case the job will be covered by one man when the trainee becomes licensed.

15. COSTS AND PERFORMANCE

The actual construction costs came to \$ 126 million. In addition \$ 10.8 million was spent on research and development.

The first core production costs are estimated on the basis of an 80% load factor at 6.0 mill/kWh excluding fixed charges on the construction costs. Of this, fuel costs are 5.1 mill and operation and maintenance are 0.9 mill.

The station has not yet achieved the 80% load factor for a normal reporting period. For the early operations summarized as of the end of December 1962, the production cost was 2.039 cent/kWh. For the month of January 1963, when the load factor was 43.7%, the cost was 8.19 mill, broken down as 5.95 mill for fuel, 1.12 mill for operations, 0.86 mill for maintenance and 0.26 mill for other supplies.

As of 31 March 1963 the station had produced 291 million net kWh and had operated for 58.5 equivalent full power days.

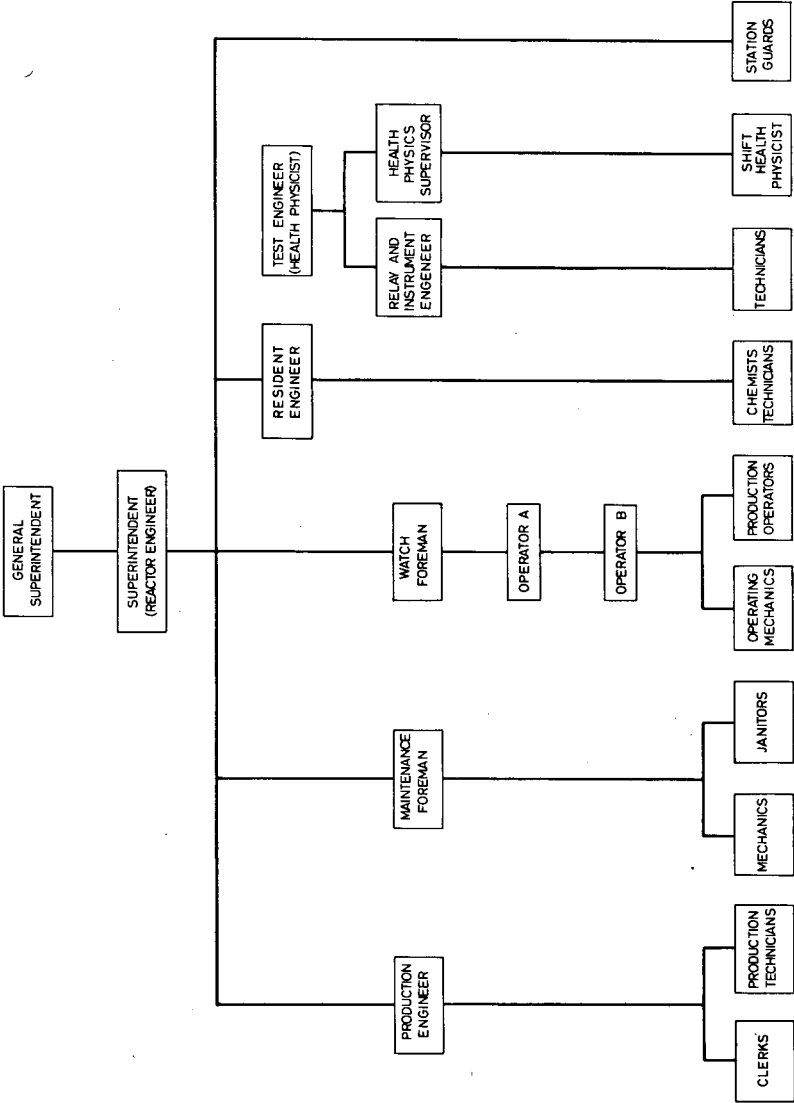


Fig. 5
Indian Point Station organization chart

16. THE SECOND AND THIRD CORES

Our Company has entered into contractual agreement with The Westinghouse Electric Corporation for delivery of the second and third cores. The fuel will be slightly enriched uranium and elements will be arranged in three concentric regions of forty elements each. Fuel costs are expected to be reduced to 5 mill/kWh for the second core and to less than 3 mill/kWh for the third core.

17. CONCLUSION

The Indian Point Station has proved to the satisfaction of our Company the feasibility of a combination nuclear and oil-fired plant to produce competitive power. The United States Atomic Energy Commission has been requested to issue a construction permit for the Company to build another nuclear plant, similar in design, but of 1000 MW capacity, to be located at Ravenswood which is within the corporate limits of the City of New York. Our Company is prepared to finance the venture with its own funds without subsidy of any kind as was the case with our Indian Point Station.

DISCUSSION

I. PAULICKA: How was the commissioning programme of the Indian Point power station worked out? I am mainly interested in knowing how many staff members worked on the preparation of the programme, and for how long. By commissioning programme I mean all experiments which take place after the completion of construction work, and I include the actual start-up and the programme for the gradual rise to power.

W. BEATTIE: I cannot give a definite answer to this question because the programme was conducted in two phases - The Babcock and Wilcox Company, who designed the reactor, did a lot of the preliminary work on a prototype reactor at Lynchburg so that there were months of work involved there mainly because of the use of thorium oxide in some of the elements. The work at Lynchburg allowed us to install the elements in the reactor, in the specified manner, within a month. The maximum number of personnel involved in getting the Indian Point plant ready, once the parts were assembled, was in the neighbourhood of 185. At the present time the number of personnel is 140.

U. ZELBSTEIN: I would like to put four questions to Mr. Beattie: First, could you give me more precise information about the economic advantage of linking a fossil-fuelled superheater to a nuclear reactor? Secondly, does this combination entail special technical difficulties? If so, what are the problems you have encountered? Thirdly, could the information derived from a fossil-fuelled superheater be used in forming conclusions about nuclear superheat? And, finally, what is the cost per supplementary kilowatt hour produced by fossil-fuelled superheat compared to the cost per kilowatt hour produced in a conventional plant using an equal quantity of fuel?

W. BEATTIE: The economic study at the time of designing the Indian Point Station definitely favoured the installation of fossil-fuelled superheaters in series with the nuclear reactor steam and the combination did not entail any special technical difficulties. The answer to your third question is, Yes. We need superheated steam from reactor operation in order to have a more efficient turbine plant cycle. As regards comparative costs I am afraid I have no figures with me.

G. B. SCURICINI: Can you tell me why you changed your plant over from a thorium to a uranium cycle? Is the reason purely economical or did you expect trouble from the use of thorium? Have you considered the cost penalties involved in changing the type of fuel?

W. BEATTIE: The reason is purely economical and we did not expect to have any trouble with the thorium cycle. There is, of course, a change-over cost involved. Westinghouse, the new contractors, are at present building a second and third core. It is hoped that by the time the third core is installed, and taking into account the change-over cost, we will be down to a 4 mill cost figure for fuel, which is going to be comparable to some of our other plants. The cost of production, at Indian Point, for the month of April was 0.735 cents per kWh - this figure is just for operation and does not include any capital depreciation. The average for the Consolidated Edison System was 0.632 cents per kWh, so you can see that Indian Point is getting into a competitive range with the other conventional plants.

You may be interested in an estimated comparison of capital costs for a nuclear plant and a conventional plant. At the Ravenswood site, which I referred to in my paper, we are installing a 1000-MW conventional unit and it is going to be oil-fired. We estimate that it will cost \$112 per kW installed. We are trying to get permission to install a 1000-MW nuclear plant at the same location in 1970 and here we estimate that the cost will be \$175 per kW installed.

As the two stations are for installation at the same location, the same wage rates and the same overheads will apply. If nuclear superheat comes into the picture, then I would hope that, by 1970, the \$175 will be down to \$130 - but this of course is just an estimate.

S. YIFTAH: The intention of the Consolidated Edison Company to build, following the experience in the design and operation of the Indian Point station, a 1000-MW power reactor in Ravenswood, New York City, has raised questions in the minds of many people about the safety of locating such a big reactor in such a big city. Can you say something about the special means to be taken to minimize the effects of any credible radiation accident at this reactor?

W. BEATTIE: The basic concept of the shielding at Ravenswood will be the same as that described in my paper on the Indian Point plant.

The proposed reactor container will be cylinder 150 ft in (inside) diameter with a domed end giving an inside height of 167 ft. The inner steel shell is 1/4-in thick with 2 ft of pervious concrete enclosing it. The outer steel shell is also 1/4-in thick and enclosed by a 5½-ft thick outer reinforced-concrete wall.

I see nothing in the operation of the Indian Point plant that would lead me to believe that it is unsafe to build such a reactor in a city.

OPERATING EXPERIENCE AT THE YANKEE ATOMIC ELECTRIC COMPANY NUCLEAR POWER STATION

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Abstract — Résumé — Аннотация — Resumen

OPERATING EXPERIENCE AT YANKEE ATOMIC ELECTRIC COMPANY. Operation of the Yankee plant began in November 1960 and since that time the station has generated over 2×10^9 kWh gross. The overall use factor from first generation of power in 1960 is just over 67%. The economic picture has also been very encouraging. The Yankee plant is not only continuing to demonstrate its ability to supply base-load power for the New England area but is also providing new information of benefit to the economics of all closed-cycle reactors. Problems have been encountered, some of a fairly significant nature, but every indication to date is that these are correctable and that plant operation will become even more dependable with continually improving economics as time goes on.

EXPÉRIENCE PRATIQUE DU FONCTIONNEMENT DE LA CENTRALE NUCLÉAIRE YANKEE. La centrale Yankee a commencé à fonctionner en novembre 1960; depuis, elle a fourni plus de deux milliards de kWh bruts. Son facteur d'utilisation depuis sa mise en service est légèrement supérieur à 67%. Au point de vue de la rentabilité, les résultats obtenus sont très encourageants. Non seulement la centrale s'est révélée capable de fournir la charge de base pour la région de la Nouvelle-Angleterre, mais elle permet aussi de recueillir des données nouvelles très utiles touchant la rentabilité de tous les réacteurs à cycle fermé. Des problèmes parfois assez importants se sont posés, mais tout semble indiquer à ce jour qu'il est possible de les résoudre et qu'avec le temps la centrale fonctionnera toujours mieux et dans des conditions de rentabilité toujours meilleures.

ЭКСПЛУАТАЦИОННЫЙ ОПЫТ КОМПАНИИ "ЯНКИ АТОМИК ЭЛЕКТРИК". Эксплуатация электростанции "Янки" началась в ноябре 1960 года, и с тех пор станция выработала более двух миллиардов киловатт-часов брутто. Общий коэффициент использования, начиная с момента производства первой энергии в 1960 году, составляет более 67%. Экономическая картина также является очень обнадеживающей. Электростанция "Янки" не только продолжает показывать свою способность снабжать энергией основную нагрузку район Новой Англии, но и позволяет получить новые данные об экономической пользе всех реакторов с замкнутыми циклами. В процессе эксплуатации возникали проблемы, некоторые из которых носили довольно серьезный характер, но в настоящее время все говорит за то, что они разрешимы и что со временем работа станции станет даже более надежной и ее экономичность будет непрерывно повышаться.

EXPERIENCIA PRÁCTICA EN LA YANKEE ATOMIC ELECTRIC COMPANY. La central Yankee comenzó a funcionar en noviembre de 1960 y desde entonces ha generado más de dos mil millones de kilowatts-hora brutos. Desde que se inició la producción en 1960 el factor de utilización medio es algo superior al 67%. Los aspectos económicos son asimismo muy alentadores. La central Yankee no sólo sigue demostrando que es capaz de cubrir la carga fundamental en la zona de Nueva Inglaterra, sino también está proporcionando nuevos datos sobre los factores económicos de todos los reactores de ciclo cerrado. Se han presentado problemas, algunos de ellos graves, pero hasta el presente todo indica que son subsanables y que el funcionamiento de la central seguirá mejorando tanto técnica como económicamente.

The Yankee Atomic Electric Company reactor is of the pressurized-water or closed-cycle type, presently rated at 170 MW(e) gross. The plant was designed and constructed by Westinghouse Electric Corporation and Stone & Webster Engineering Corporation for a group of ten of New England's major utility companies.

The YANKEE reactor has four primary coolant recirculating loops operating at a nominal pressure of 2000 lb/in² gauge and an average temperature

TABLE I
DESIGN INFORMATION
YANKEE ATOMIC ELECTRIC COMPANY

Licensed Rating: 540 MW(t) ; 170 MW(e) gross; 160 MW(e) net					
Core (uniformly enriched at 3.4%)					
Composition at 514°F	UO ₂	GAP	SS	Zirc-2	H ₂ O
Volume (in ³)	143 025	6 725	48 219	12 137	201 065
Weight (lb)	52 190	-	13 588	2 855	5 675
Dimensions at 514°F	Diameter: 75.68 in		Height: 92.3 in		
Heat Transfer					
Coolant: H ₂ O		System volume: 24 000 gal (US)			
Pressure: 2 000 lb/in ²		Average temperature: 514°F			
Flow: 40.8 × 10 ⁶ lb/h		Velocity in core: 14.9 ft/s			
Active surface area: 15 500 ft ²					
Average heat flux: 115 700 BTU ft ⁻² h ⁻¹					
3.02 kW/ft					
Maximum heat flux 439 600 BTU ft ⁻² h ⁻¹					
11.50 kW/ft					

of 514°F. The four canned motor main coolant pumps are rated at 24 000 gal (US)/min. The entire reactor system is contained in an elevated steel sphere 125 ft in diameter. A more complete listing of design data may be found in Table I.

Operation began in November 1960 and since that time the station has generated over 2×10^9 gross kWh. The overall use factor from first generation of power in 1960, including the first refuelling, is just over 67%. This record, we think, is most gratifying, especially since there is every hope that it can be improved as time goes on.

The economic picture has also been more encouraging than any one had dared to predict. YANKEE was constructed purely as a demonstration plant, well before the days when a nuclear plant could be expected to be competitive with a conventional plant. Average power costs from the first core were less than 9.5 mill/kWh, which compares to conventional costs in New England of about 8 mill/kWh for a plant of similar size built at the same time. In economics, as in operation, we have hopes that future costs from this plant can be improved, possibly even to the point of being truly equivalent with conventional plants.

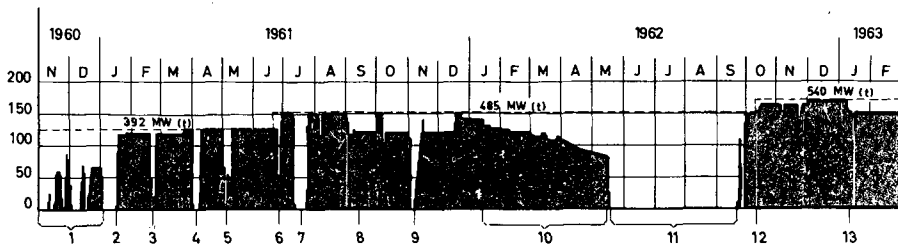


Fig. 1

Average daily generation (MW gross)

- 1 Initial test operation
- 2 Turbine vibration corrected
- 3 Valve repairs
- 4 Scheduled physics testing
- 5 Modified pressurizer instrumentation
- 6 Power increase to 485 MW (t)
- 7 Scheduled physics testing; turbine governor
- 8 Boron test
- 9 Scheduled physics testing
- 10 "Stretch-out"
- 11 Refuelling
- 12 Power increase to 540 MW (t)
- 13 Boron test

The entire history of plant operation is shown graphically in Fig. 1.

Initial power testing was carried out in November and December 1960. During this "shake-down" period we had a variety of minor mechanical difficulties. The most serious was a vibration in the turbine which was caused by a steam lift arising from bottom entry of steam with insufficient space between the first and second stages for adequate distribution to take place. This was corrected in January 1961 by a modification of the blade rings.

Following this repair, the plant was taken to 120 MW(e) gross, which was the power level authorized at that time.

In February a plant shut-down was necessary in order to make a number of valve repairs and modifications. We now realize that plant reliability depends to a considerable degree on the quality of a number of relatively small valves which connect a variety of auxiliary systems to the primary system. The valves which had been employed were inadequate in several respects and have been a source of rather continuous annoyance as we proceed with a gradual programme of replacement or repair.

The first of a series of scheduled physics tests on the first core took place in April 1961. These tests were intended to check the effects on nuclear characteristics of the gradual build-up of plutonium throughout core lifetime. Further checks were made in July and November 1961 and in May 1962 just before refuelling. As it turned out, neither the power nor the temperature coefficients of reactivity changed appreciably.

In the YANKEE reactor we are able to evaluate local power levels throughout the reactor core by means of a combination of exit water thermocouples and flux-wires. The "flux-wires" are inserted into the centre of certain

strategic fuel assemblies, then withdrawn by remote control and passed by a radiation counter. The counter provides a profile of the activation of the wire which can be converted into a profile of the power level along that particular fuel assembly. Since reactor power level is limited by the maximum local power rather than the average, the ability to determine experimentally the variations in these levels has enabled us to apply for and to receive authorization for two power increases since the station went into operation. The first of these was in June 1961 when we went to 150 MW(e) gross and the second in October 1962- up to 170 MW(e) gross.

During July 1961, while shut down for scheduled physics testing, a number of minor modifications were made in the turbine governor system and to other plant components.

On 18 January 1962 all control rods had been fully withdrawn and the plant was no longer able to carry 120 MW at the normal reactor temperature of 514°F. During the next four months, however, an additional 320 million kWh were generated by taking advantage of the reactivity made available by gradual reductions in reactor power level and temperature. This additional generation amounted to approximately one quarter of the total generation from the first core. By the time the plant was finally shut down for refuelling on 18 May, turbine throttle pressure had dropped from 485 to 250 lb/in² gauge and the load was approximately 80 MW(e). During this period the plant was on the line without interruption for 147 d.

To sum up, most of the problems during the 18 months of Core I operation had to do with the secondary plant, - primarily the turbine. Even in the nuclear portion of the station the problems were primarily with conventional equipment such as valves. In fact, with the exception of physics testing, the reactor system was capable of producing power 96% of the time.

The YANKEE fuel consists of brazed bundles of stainless-steel tubes, each tube containing uranium oxide pellets. Since Core II was ordered before Core I had accumulated any appreciable service, it is essentially a duplicate of the initial core. The appearance of the fuel assemblies after 18 months of operation was excellent, with no indications of any problem whatsoever. Seventy-four of the seventy-six fuel assemblies were replaced-two of the high burn-up assemblies being retained as a test of future operations with two-region cores. These two test assemblies will have reached burn-ups of approximately 18 000 MWd/t by the time Core II is replaced.

All other reactor internals and core components were checked with no evidence of problems with the exception of the control-rod drive chain. Some wear was found at the adapter between the control-rod absorber section and the control-rod drive shaft. This had been anticipated and sufficient spares were on hand to replace all shafts and absorber sections. However, wear was also found in the joint between the stainless adapter at the bottom of the absorber section and the top of the Zircaloy follower. This necessitated a rush order for an additional set of followers, modified so that the joint is now between two stainless adapters, rather than from stainless to Zircaloy. This replacement and re-design caused a delay in the overall refuelling operation of approximately six weeks.

The next problem to appear was also connected with the control rods. These are an alloy of silver, indium and cadmium, covered with a metallurgically-bonded nickel plate. They had operated quite satisfactorily during

the life of Core I with only negligible levels of silver in the main coolant. During refuelling, however, under the combination of cold, borated, oxygen-saturated water, the nickel plate deteriorated rather rapidly. This, in turn, exposed large areas of the silver alloy. Unfortunately, a few grams of this material, dissolved in the refuelling water and subsequently plated out on all submerged metal surfaces, is all that is necessary to create a severe radiation problem. This condition caused an additional delay of about ten days while various decontamination agents were tried out and while temporary shielding was being devised and put to use during the replacement of the reactor head.

Since that time, an intensive search has been made for a more satisfactory control-rod material. There seems to be no ideal solution, considering the requirements for structural strength, neutron absorbing effectiveness, corrosion resistance, availability, proven technology and cost. The most suitable answer to YANKEE's particular requirements seems to be either hafnium or the same alloy of silver - clad this time in a much thicker layer of Inconel.

One of the encouraging aspects of refuelling was the turbine. After a complete check it was found to be in excellent condition - at least as good as units operating with more conventional steam conditions.

On 20 September 1962 the plant was returned to operation. After resolving some minor turbine governor problems, base-load operation was resumed.

During the final three months of 1962 the plant generated 370 951 000 kWh, at a use factor for this period of over 97%.

Presently the Yankee plant employs boron in the form of boric acid as a supplemental control for cold shut-down of the plant. During normal operation at power, the only water additive present is an overpressure of hydrogen so as to reduce oxygen levels to a minimum.

Core III will be a two-region core made up of a half-loading of new, more highly enriched (4.1%) fuel, surrounding a central portion made up of the least burned assemblies from Core II. Such a core will have a somewhat higher initial reactivity than the earlier, single region, 3.4% enriched, cores. This will be compensated for by using boric acid during the first few days of operation while xenon poisoning is taking effect. We feel this is a logical approach to the eventual use of boric acid as a continuous operating shim control, with its attendant benefits of longer-lived and higher-powered cores.

As contributing steps in demonstrating the feasibility of operation at power with boric acid present, YANKEE has twice operated for periods of several days with up to 400 ppm of boron in the primary system. The most recent of these tests took place this past February.

From a nuclear standpoint these tests were inconclusive in that during the first test we noted a slight loss of reactivity and during the second we found a slight gain. From an operational standpoint, however, the tests demonstrated that boric acid causes no difficulty. We were able to add the poison and, later, to remove it without incident. All reactivity changes were slow - on the order of hours or even days - and well within the capability of the control systems.

On 1 March 1963, plant load was returned to 170 MW(e). It is expected that all control rods will be withdrawn and another period of core stretch-

TABLE II
OPERATING STATISTICS

		Core I	Till 30 April 1953
Gross generation	(kWh)	1 330 521 000	2 137 270 800
Station service	(kWh)	107 560 431	157 984 674
Net generation	(kWh)	1 222 960 569	1 979 286 126
Station service	(%)	8.08	7.39
Plant operating factor	(%)	68.59	66.92
Time critical	(h)	13 247	18 555
Average burn-up	(MWd/t)	8 349	5 182 (Core II)
Plutonium formed	(kg)	99	66

out begun before the end of May. Core II refuelling is scheduled to begin about the middle of August 1963. A summary of plant performance statistics, to the end of April 30, is presented in Table II.

In conclusion, the Yankee plant is not only continuing to demonstrate its ability to supply base-load power for the New England area but is also providing new information of benefit to the economics of all closed-cycle reactors. Problems have been encountered, some of a fairly significant nature, but every indication to date is that these are correctable and that plant operation will become even more dependable with continually improving economics as time goes on.

DISCUSSION

A.K. HANNERZ: I would like to ask Mr. Minnick for a break-down of this production cost of 9.5 mill/kWh. What is the fuel cycle component — is the fuel use-charge included or, in the case of this demonstration reactor, is it supplied free of charge? What is the annual capital charge and what is the operating cost?

L. MINNICK: The figure I quoted was 9.5 mill/kWh for the first core. It is based on a price of \$30/g for the plutonium produced and no use charge for the uranium in the core. On that basis the fuel cost is 4 mill or less per kilowatt hour.

For the second core the kilowatt hour cost is averaging about 10 mill and this is on the basis of plutonium at \$9.50/g, so the change has not been great. In future cores we feel that, still on the basis of plutonium at \$9.50/g, we can reduce fuel costs to 3 mill or even less and include the use charges for the uranium.

F.S. ASCHNER: My question refers, in general, to nuclear power plants using saturated steam at the turbine inlet, and in particular to the Shippingport, Yankee, Dresden and Kahl plants.

Is the steam wetness in the turbines limited, even in winter when cooling-water temperatures are low, to the 12% or 13% generally accepted as the maximum admissible in conventional steam power stations?

After how many operating hours have the most recent turbine inspections been made? Have erosions been found? Are special materials used for the wettest turbine stages? What are the arrangements for water separation inside the turbine?

L. MINNICK: After the Yankee plant had operated for 18 months we found practically no erosion. In the last stage of the turbine we did, however, have to replace stellite strips - which, as in most turbines, are used in the last two stages on the leading edges of the blades - on four blades, a small proportion of the total number used. There are provisions throughout the turbines (slinger rings, drains, etc.) for the removal of moisture and there are also moisture separators between the high- and low-pressure turbines which reduce the moisture entering the low-pressure turbine to almost zero. I understand that the moisture is in the 10% range at the exit.

G. B. SCURICINI: With reference to the boron test mentioned in Mr. Minnick's paper I have been told that some boron will be retained in the "crud" or as a hard deposit on some surfaces and I would like to know if this boron deposit could have any effect on reactivity. Did you find any change in the reactivity balance before and after the boron shim tests?

L. MINNICK: Our concern used to be that boron in combination with lithium hydroxide would form lithium metaborate, which has a reverse solubility and could deposit on the core. However, we do not have lithium or any other salt in the reactor when we have boron; we thus only find boric acid and there is no reverse solubility effect, though there is a possibility that there could be a deposit on the core. The observed changes in reactivity are very small. We found that in the first test the reactivity did indeed decrease, which could be interpreted as a deposition of boron, and in the second test the reactivity increased, which could be interpreted as the removal of some poison already in place in the core, if indeed the reactivity effect arises from that source. On looking at the fuel after operation, the fuel is the cleanest that anyone has ever seen. There is no crud and there is no film but it is slightly discoloured and that is the only effect.

M. GUEBEN: How long was it, after the reactor vessel was opened, before $\text{Ag}^{110\text{m}}$ contamination was detected on the walls of the loading channel?

L. MINNICK: We knew something was wrong very shortly after beginning the refuelling because the radiation levels around the pit, at the floor where the operators stand during the refuelling, were of the order of 30 mr and the designer had told us that they should be about 2 mr. As we lowered the water level in the pit and uncovered the stainless wall, before installing the reactor head, the activity level in the vapour container continued to rise until we were experiencing levels as high as 2 r near the wall down inside the pit.

M. GUEBEN: How long is it, from the moment the primary loop is cold, before you begin to refill the loading channel preparatory to reloading?

L. MINNICK: Altogether something like a week, of which one day is spent actually removing the studs.

A. WECKESSER: Did you make any chemical or physical test to find out if there is any boron content in the fuel due to physical absorption?

L. MINNICK: This will be one of the aspects of the Westinghouse examination. I personally do not hold much hope for it because of the lapse of time and the large number of environments through which the fuel has passed and will pass before it is examined.

A. WECKESSER: How often can you use the O-rings which seal the reactor flange?

L. MINNICK: We think we could use them almost indefinitely and, although we did replace them this last refuelling, we have had no trouble with them at all.

A. WECKESSER: How many spurious scrams occurred during the whole period of operation?

L. MINNICK: A total of only 33 scrams have occurred to date. I am not sure of the percentage of spurious scrams.

J. STORRER: Did you detect any undue activity in the primary water which might be attributed to uranium contamination on the surface of the fuel assemblies or to fuel failure? Did you make any contact measurements of the activity of the vessel's internal components during the first refuelling? And what was the water pH value when you operated with 400 ppm boron?

L. MINNICK: We did experience some fission product activity in both the first and the second core. The levels were such ($<1\%$ of design values) that we've had a running argument within the organization as to whether this is external contamination or whether this actually represents one or two very small fuel leaks. My own opinion is that we have one or two small leaks.

No contact measurements were made since the items concerned were stored under water and their activity was not a matter of importance.

The pH of the water during the boron tests is unadjusted and therefore runs slightly on the acid side with 400 ppm of boron.

F. R. BELOT: My question refers to the instrumentation of the YANKEE reactor, and especially to the placing of flux wires to determine the distribution of flux in the core. This device is likely to be of interest to the operators, but it involves both a technical complication and an extra expense. Could you give us some idea of the effect of this complication on operation as well as the magnitude of the expense involved? Could you also tell us if authorization to increase power could have been obtained as quickly in the absence of flux wires?

L. MINNICK: The core instrumentation unfortunately cost much more than we felt that it should have done. We did not decide to install it until approximately a year and a half before plant operation and this necessitated excessive engineering and fabrication costs in order to fit it in. The cost was something like \$500,000. There are two schools of thought as to the desirability of the core instrumentation: the pure operator says it is quite unnecessary – the scientists and management of the company say it has been very valuable. It alone has allowed us to go up in power as we have been able to do, and going up in power is the best and quickest single way to reduce the cost per kWh from an existing plant – the capital costs being spread over more kWh. As for the day-to-day problems that we have had with instrumentation; as far as I know it has never caused a shut-down or delayed the operation of the reactor; it is, however, a major problem during refuelling.

B. SAITCEVSKY: Is the cost of materials replaced during operation, such as valves and control rods, included in the cost per kWh? What is the total amount of such replacements? And what is the cost of a control rod?

L. MINNICK: The answer to the first question is, Yes, the figures I quoted do include all the replacement parts used; in particular the 10 mill/kWh figure for Core II includes the cost of the pieces that were added between Cores I and II. The total cost was rather over \$500 000, of which the 24 Zircaloy followers accounted for \$200 000. The control rods cost something like \$20 000 each.

A. NOVIKOV (Chairman): Are there burnable poisons in the first YANKEE core? And what is the reactivity balance in regard to temperature effect, power effect, burn-up and finally poisoning?

L. MINNICK: There are no burnable poisons or other means of reactivity control except the control rods and boric acid in solution. The boric acid was used on the first core only, for supplementary control during a cold shut-down of the plant.

The total excess reactivity on the first core was measured as $15.5 \pm 1.5\%$ and is divided up approximately as follows:

Temperature effect (hot to cold)	= 5%
Power effect (zero to full power)	= 3%
Burn-up effect	= 5%
Poisoning (Xe & Sm)	= 2.5%

THE FIRST TWO YEARS OF OPERATING EXPERIENCE OF THE KAHL NUCLEAR POWER STATION

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Abstract — Résumé — Аннотация — Resumen

THE FIRST TWO YEARS OF OPERATING EXPERIENCE OF KAHL NUCLEAR POWER STATION. Kahl, being the first European private atomic power station, has been operating at load since June 1961. Kahl is equipped with a boiling-water reactor, operating in an indirect cycle and with natural circulation. Its electrical net capacity is 15 MW, and the power produced until February 1963 amounts to 140 million kWh. In the paper the operating experience, particularly the extensive testing programme such as concerning transient behaviour and gamma-scanning, will be reviewed. Results about the operating performance of certain plant components such as control-rod-drive system, off-gas system and turbine will be presented. After this testing programme the plant has been operated at base load for some time in order to obtain realistic experience on fuel performance. After completion of the base-load phase of operation a nuclear superheating test loop will be installed and run in the Kahl reactor.

EXPÉRIENCE ACQUISE PENDANT LES DEUX PREMIÈRES ANNÉES DE FONCTIONNEMENT DE LA CENTRALE NUCLÉAIRE DE KAHL. La centrale de Kahl, la première en Europe dont l'exploitation soit assurée par une entreprise privée, alimente le secteur depuis juin 1961. Elle est équipée d'un réacteur à eau bouillante, en cycle indirect, à circulation naturelle. Avec une puissance électrique nette de 15 MW, elle avait produit, jusqu'à février 1963, 140 millions de kWh. Le mémoire rend compte de l'expérience acquise concernant le fonctionnement de la centrale, notamment du programme étendu des essais effectués sur son comportement en régime transitoire et sur la détection gamma. Il donne, en particulier, des renseignements sur le fonctionnement de certains éléments de la centrale tels que le mécanisme de commande des barres de contrôle, le dispositif d'échappement des gaz et la turbine. A la suite de ce programme d'essais, on fait fonctionner depuis quelque temps la centrale comme centrale de base afin de rassembler des données pratiques sur les performances du combustible. A l'expiration de la période prévue pour le fonctionnement comme centrale de base, on aménagera, dans le réacteur, une boucle expérimentale destinée à l'étude de la surchauffe nucléaire.

ОПЫТ ПЕРВЫХ ДВУХ ЛЕТ ЭКСПЛУАТАЦИИ АТОМНОЙ ЭЛЕКТРОСТАНЦИИ В КАЛЕ. Атомная электростанция в Кале, первая в Европе частная атомная электростанция, эксплуатируется под нагрузкой с июня 1961 года. На электростанции установлен реактор с кипящей водой, косвенным циклом и естественной циркуляцией. Чистая электрическая мощность реактора составляет 15 мвт. До февраля 1963 года количество полученной энергии составило 140 млн. квт.ч. Рассмотрен опыт эксплуатации, в частности расширенная программа испытаний: например, поведение реактора при переходном процессе и контроль гамма-излучения. Будут представлены результаты изучения эксплуатационной характеристики некоторых компонентов установки, например системы привода регулирующих стержней, системы удаления газа и турбины. После осуществления этой программы испытаний установка в течение некоторого времени эксплуатируется с базовой нагрузкой с целью приобретения соответствующего опыта о работе топливных элементов. По завершении стадии эксплуатации с базовой нагрузкой на реакторе будет установлена испытательная петля для изучения ядерного перегрева.

EXPERIENCIA ADQUIRIDA EN LOS PRIMEROS CUATRO AÑOS DE FUNCIONAMIENTO DE LA CENTRAL NUCLEOELÉCTRICA DE KAHL. La central de Kahl constituye la primera central nucleoelectrónica europea de propiedad privada, y funciona en carga desde junio de 1961. Está equipada con un reactor de agua hirviente

que trabaja en ciclo indirecto por circulación natural. Su capacidad eléctrica neta asciende a 15 MW y hasta febrero de 1963 había producido 140 millones de kWh. La memoria revisará la experiencia adquirida durante su funcionamiento, ante todo con el extenso programa de ensayos sobre el comportamiento transitorio y la exploración gamma. Presentará datos acerca del resultado que han dado en funcionamiento ciertas partes de la central, tales como el dispositivo de accionamiento de las barras de control, el sistema de purificación de los gases de escape y la turbina. Una vez terminado el programa de ensayos, la planta se explotó en carga básica durante algún tiempo a fin de reunir datos sobre el rendimiento del combustible en la práctica. Una vez completada esta fase, se instalará en el reactor de Kahl un circuito experimental de sobrecalentamiento nuclear.

1. INTRODUCTION

In 1958 the Rheinisch-Westfälische Elektrizitätswerk AG (RWE) and Bayernwerk AG, two large utility companies in the Federal Republic of Germany, ordered a 15 000 kW(e) nuclear power station. It consists of an indirect-cycle boiling-water reactor with a conventional secondary steam-turbine generator system (Fig.1). Other than the Elk River Reactor Plant, it is the only one of this type in operation today. The steam for the turbine comes from an intermediate heat-exchanger rather than from the reactor. Since this was the first nuclear power plant in Germany, it was decided that the licensing procedures should be expedited by not taking the radioactive steam outside the reactor containment building. This decision was partly based on the fact that there is a high population density in Germany, making exclusion areas as a safety measure hardly feasible.

The indirect-cycle plant – appropriate for the experimental predecessor of larger nuclear power stations – was to demonstrate the safety and feasibility of the direct-cycle BWR.

The order to build the Kahl Plant on a turn-key basis was placed with the Allgemeine Elektrizitäts-Gesellschaft (AEG) as prime contractor. AEG let sub-contracts to the International General Electric Company (IGE) of the United States, the German civil engineering firm Hochtief and many other companies. IGE provided the conceptual design of the nuclear steam supply system, rendered engineering services during plant construction and start-up and supplied the fuel elements, control rods and drives and parts of the reactor safety system. All other components were manufactured in Germany.

The power plant produces 60.4 MW(t) and 16 MW(e) gross. The reactor steam system operates at 71.3 atm gauge constant, and the turbine steam system operates at 45.6 atm gauge at rated power. The initial U^{235} enrichment of the Zircaloy-clad UO_2 -fuel elements is half 2.3% and half 2.6%. The layout of the plant has been described several times before [1-7].

The average fuel burn-up hitherto attained is 4500 MWd/t of uranium. To date 150 million kWh of electric power have been produced by the Kahl station.

2. EXPERIENCE OF PLANT OPERATION

The Kahl Nuclear Power Station is now operated as a base-load plant to collect information on fuel performance as much and as soon as possible.

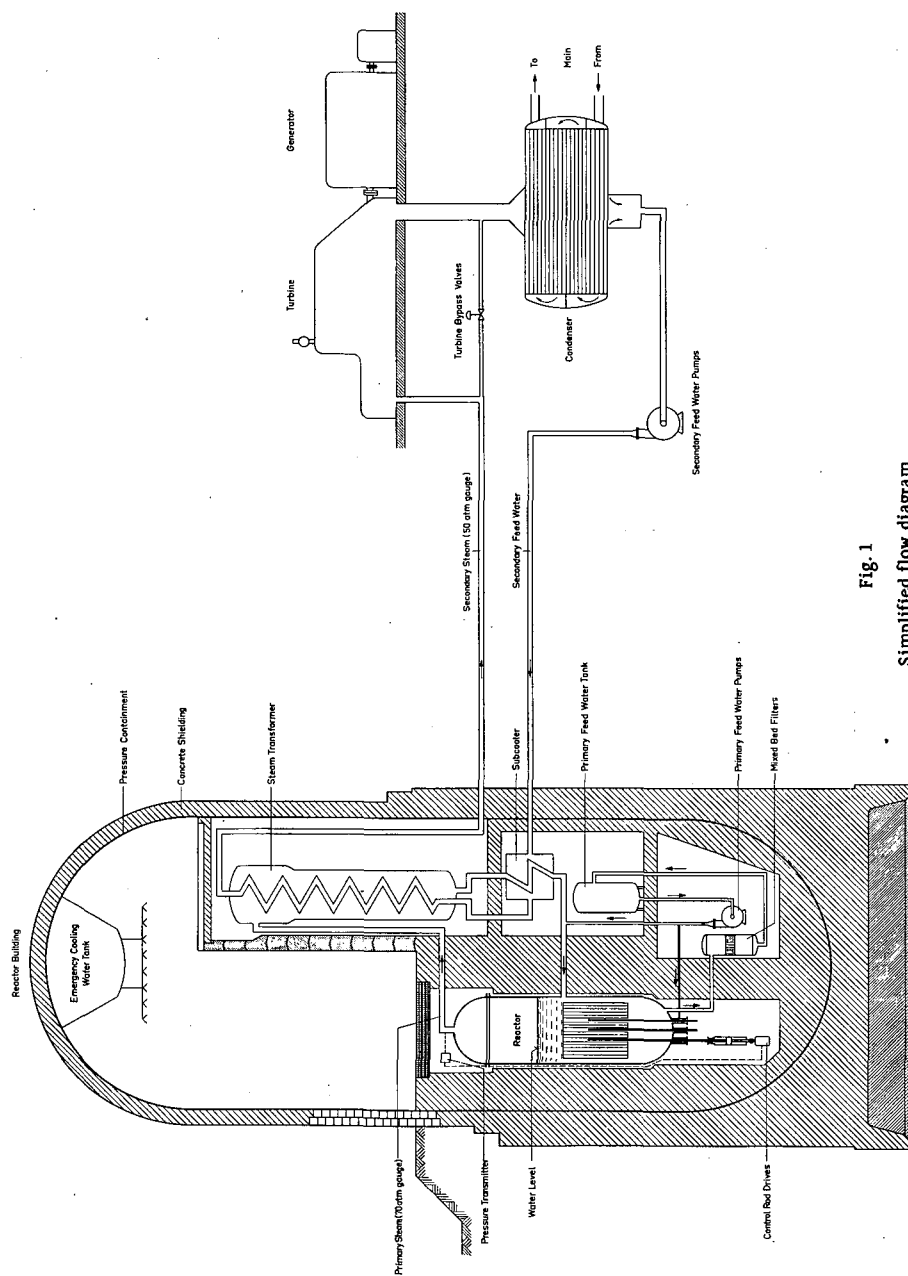


Fig. 1
Simplified flow diagram

Before this, the plant performance under various operating conditions, e.g. subject to load changes, has been extensively studied, as they are of significance from the point of view of power-plant operation in a compound electric net-work. This programme of testing of plant performance has included tests of start-up, transient behaviour, shut-down and low power operation.

2.1. Reactor start-up experience

The time required to run the plant from ambient temperature and pressure up to the full operating condition is only determined by the permissible temperature difference between the wall and the head flange of the reactor pressure vessel. This means that the temperature rise in the walls of the pressure vessel is limited to about 35°C/h (Fig.2). Consequently it takes 10 h to reach the full operating temperature and pressure. Compared to that, the time required for reactor criticality and for supply of the nuclear heat production is insignificant.

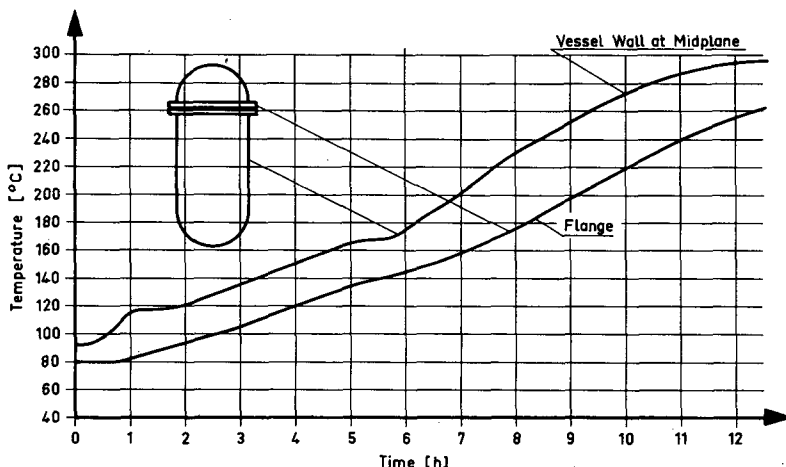


Fig. 2

Heating-up curve of reactor vessel

Fig.3 shows a plot of time against thermal reactor power during start-up. 18 min after withdrawing the first control rod the reactor becomes critical at about 6 kW(t). The transition from the sub-critical state to the super-critical one occurs so smoothly that it cannot be observed on the power plot at all. After about 40 min the reactor output is attained which corresponds to the permissible heat-up rate. Initially the reactor output is about 600 kW(t); it rises to 2 MW(t) by small steam production at higher temperature. The reactor operating pressure is reached seven to eight hours later so that the reactor can then be rapidly brought up to full power.

Fig.4 demonstrates this operation both without and with the turbine connected. If in the first case steam is directly transmitted to the condenser via the by-pass, the reactor power can be raised from 5 to 75% within 10 min.

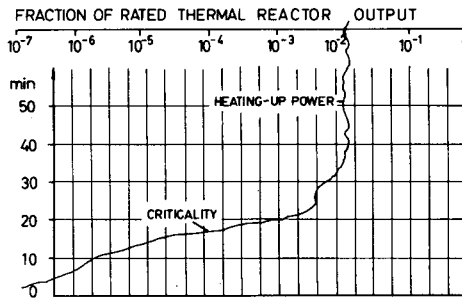


Fig. 3

Plot of reactor output at start-up

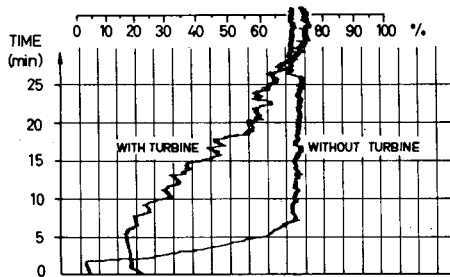


Fig. 4

Change of reactor power
with and without connected turbine

If the turbine is also connected, this will determine the rate of power increase so that it takes twice this time for the same operation. Consequently, the reactor assumes any power level between zero and full load within a few minutes. It is important to note that the problem of low-load operation, which is quite crucial for conventional boilers, does not exist.

2.2. Plant availability attained

Fig.5 shows the electric power production and the availability of both the entire power plant and the reactor for the period from June 1961, when power operation commenced, until February 1963, when this paper was written. In 1961 the power production was limited to 50 - 75% of the total power of the plant because of licence requirements. After a period of half a year, during which the performance tests required by the licence were carried out, full power operation was authorized at the beginning of 1962. It should be emphasized that the availability of the reactor exceeds that of the entire power plant. On an average, plant availability has amounted to 75%. The balance to 100% results from both scheduled and unscheduled plant shut-downs which are discussed in the following sections.

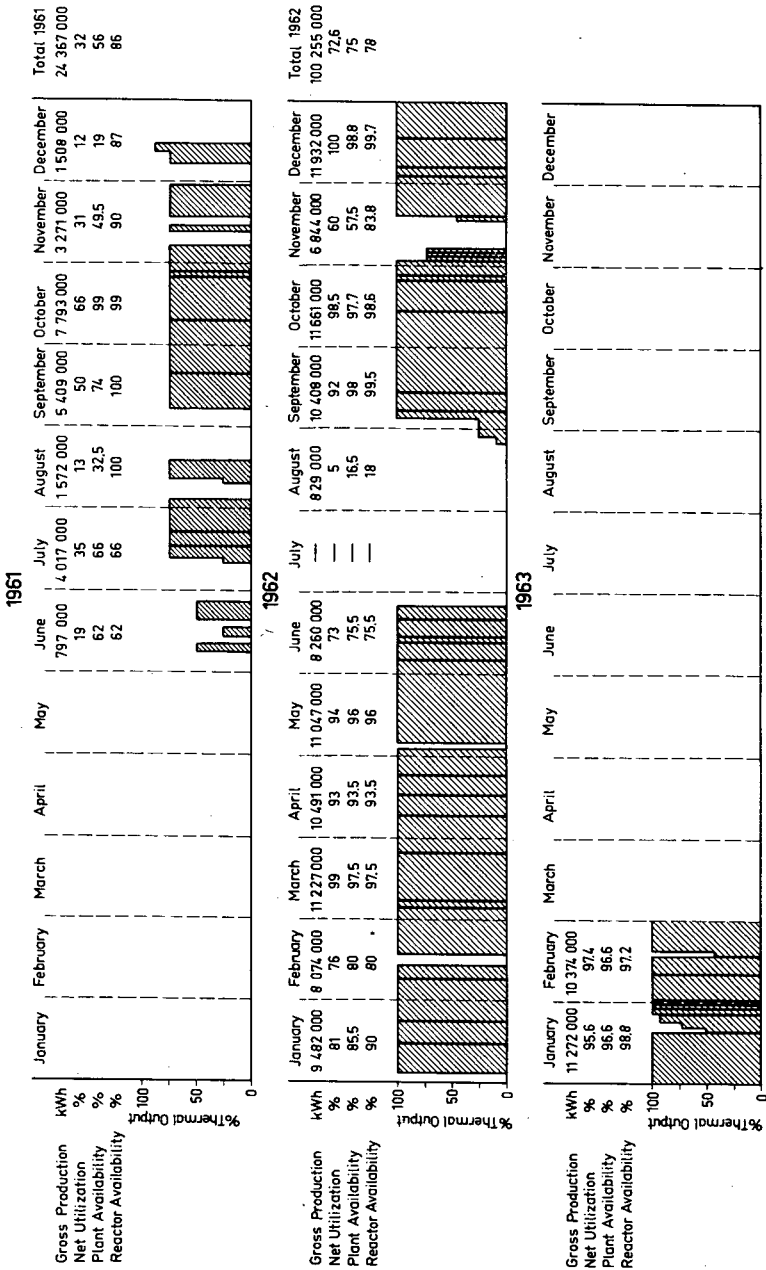


Fig. 5
Output diagram

2.3. Scheduled plant shut-downs

There were two scheduled plant shut-downs: (a) June - August 1961; and (b) July - August 1962. During period (a), the plant was shut down several times in order to measure the flow and neutron-flux distribution throughout the reactor. The reactor vessel had to be opened for this purpose. This programme also included measurements such as that of the recirculation flow, which were demanded only by the licensing authorities. The results have clearly shown that there is good agreement between the calculated and measured values for the Kahl core so that the authorities should now be able to facilitate the licensing procedure for future boiling-water reactors.

In order to compensate the initial excess reactivity of the reactor, 60 of the total of 88 fuel elements carry shrouds of borated steel. In the course of reactor operation the steel channels will be replaced by Zircaloy channels. Consequently the plant had to be shut down for period (b) to replace 30 borated steel channels by Zircaloy channels. This interruption of plant operation was also used to replace the scram valves of the reactor safety system and to check the metallurgical condition of the control rods.

2.4. Unscheduled plant shut-downs

Unscheduled plant shut-downs occurred for several weeks in November and December 1961, and in February, May and November 1962. These plant

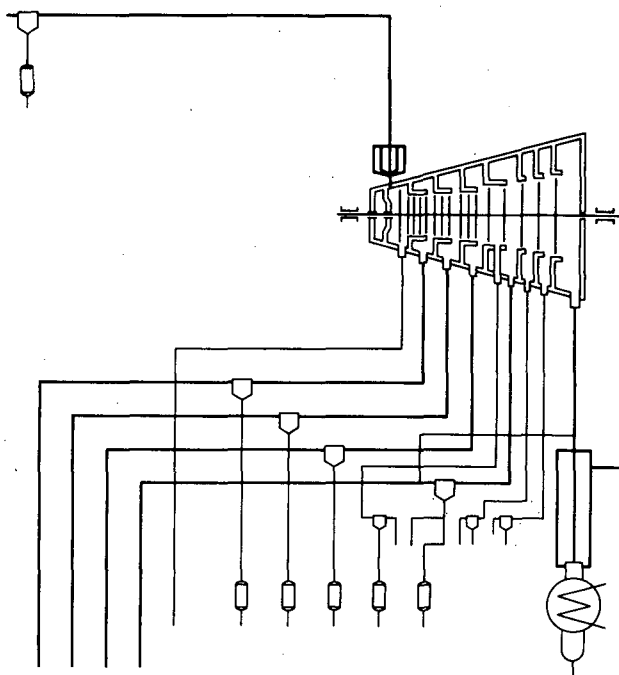


Fig. 6

Turbine layout with extraction lines

interruptions were essentially caused by difficulties in turbine operation and in the mechanical parts of the control-rod drive system.

2.4.1. Difficulties in turbine operation

The difficulties experienced in the operation of the turbine are evidently due to the fact that the turbine is operated with saturated steam. The moisture content of the live steam is below 1%. At a steam flow rate of 100 t/h about 10 t/h of water are entrained from the turbine at eight extraction points (Fig.6). During operation the turbine casing is covered internally by a water layer which evaporates upon shut-down. This evaporation obviously gives rise to an unfavourable temperature distribution in the casing which may have been the reason for leakage in the turbine flange.

In addition to the normal flash tanks in the extraction system, tanks were installed to measure the amount of water extracted. When the generator is tripped, the pressure in the turbine decreases, causing the water to flash to steam which continues to drive the turbine. Fig.7 shows the condition when the main valve of the turbine is closed and the generator is still on the line. It can be seen that 12 s are required before the generator reaches zero power. Fig.8 shows the situation when the generator is tripped from the grid and the main turbine valve is also closed. In this case, the turbine overspeeds to 3380 rpm, which is about 12% overspeed, and exceeds the set point for turbine trip. These difficulties were overcome by using electrically operated relief valves in the lines to the extraction tanks. These valves operate automatically when a generator trip occurs.

2.4.2. Difficulties in the reactor scram system

Frequent functional maloperations of the mechanical parts of the reactor scram-system caused several short unscheduled reactor shut-downs. The air control and sealing system for the control rods is shown in Fig.9. During normal operations the control rods are driven by electrical motors. At scram the control rods are operated by pressurized air. The control blades are connected to the air cylinder by a seal shaft and connecting shaft. Through the opening of two electro-pneumatic valves mounted in series, air is allowed to enter the control-rod air-cylinder causing the rapid scram insertion of the poison blade into the core. The valve response, calculated to meet safety system performance, on the average should not be greater than 100 ms. With the original valves, it was difficult to keep the time response constant. One of the reasons was the breaking friction between the dynamic O-rings and the seal surface in the valve. For this reason, the valves were replaced. In this connection, it is interesting to note that the operational reliability of a nuclear power plant may be limited to a large extent by conventional equipment and engineering problems.

Also several spurious scrams occurred by induction in the extremely sensitive reactor safety system due to external electrical sources. Such disturbances are highly undesirable from the standpoint of reliable power supply to the grid. It is therefore of the utmost importance to exercise extreme care in the installation of the wiring and to pay meticulous attention to the sparkless functioning of electrical components like relays.

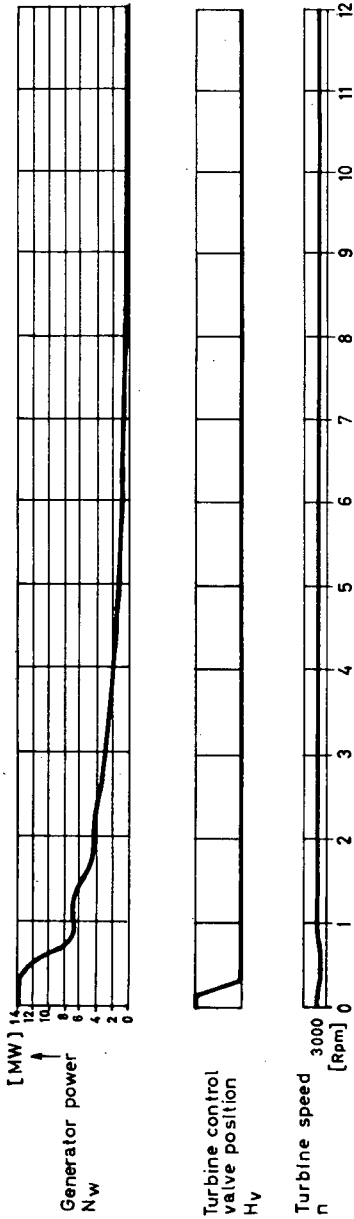


Fig. 7

Turbine trip load rejection
Conditions: Turbine valve tripped at 14 MW(e),
Generator on line until tripped by reverse current,
Steam flow 200 000 lb/h.

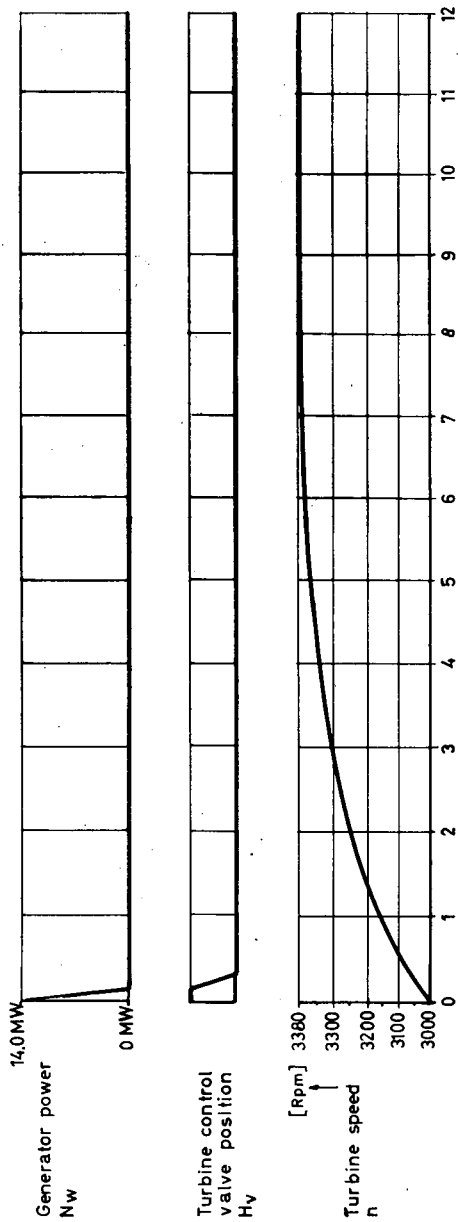
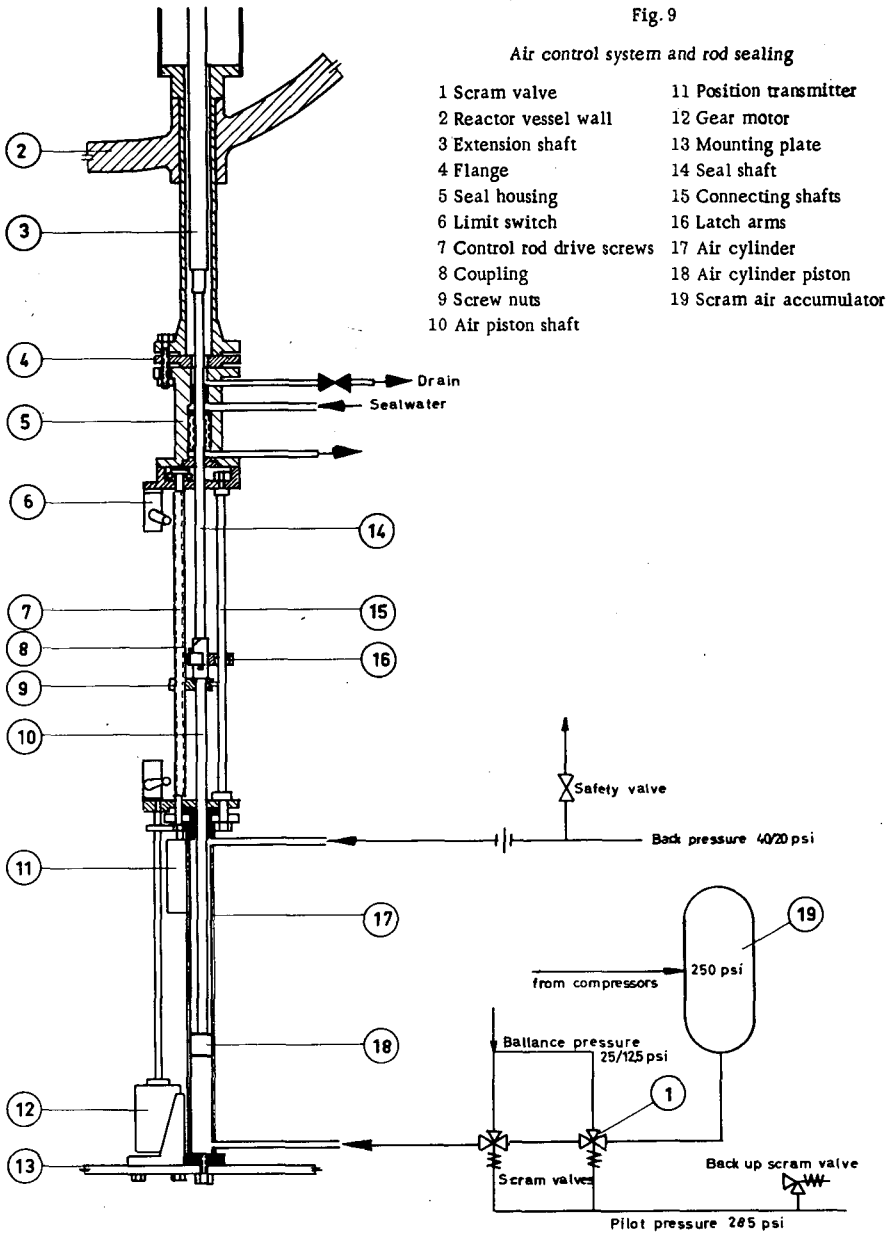


Fig. 8

Generator and turbine trip
The load was rejected by tripping both the generator from the grid and the turbine stop valve at 14 MW(e)

Fig. 9

Air control system and rod sealing



On the other hand, there is always the possibility of a grid problem that could cause a generator trip, which in turn may initiate a reactor scram for safety reasons. With the plant at full power, this would, of course, be the most severe transient that the system could experience under normal operating conditions. In case of a generator trip, the secondary steam to the turbine is automatically by-passed to the main condenser. Fig.1 shows the principal steam and water circuits. It was of special interest to know the relationship between the by-pass valve response, which requires about 8 s to actuate, and the entire plant response. Tests were conducted to show this effect and also the effect on reactor stability. Fig.10 shows the result of the load rejection test at 80% of full power. At time 0, the turbine control valves are closing to avoid a turbine overspeed. The by-pass valves are still closed. The secondary steam flow decreases very rapidly. The results of this can be seen by the increase in secondary and primary steam pressure.

The increase in reactor pressure causes a decrease in void content in the core and corresponding increase in flux which is a typical response for a boiling-water reactor. In our case the flux increase was about 10%. Since this flux increase was below the scram point, no scram resulted. The by-pass valves opened after about 8 s and the reactor stabilized itself without control-rod adjustment.

Fig.11 shows the response of the system after a full power load rejection with control-rod adjustment. In this particular test, the automatic reactor pressure regulator operated two control rods to control the reactor pressure. It should be noted that there was no flux increase, but that an immediate flux decrease resulted. The system again attained equilibrium in about 100 s.

2.5. Experience with the off-gas system

Fig.12 shows the off-gas system. This is essentially the first experience with this type of off-gas system in connection with boiling-water reactors. The off-gas is collected through a baffle system on the primary side of the steam generator. The off-gas contains steam, hydrogen, oxygen and a small amount of nitrogen. If a fuel-element failure occurs, krypton and xenon appear as non-condensable gases. In the recombiner, H_2 and O_2 are recombined and condensed in the gas cooler and then returned to the primary system. This condensate will take with it a certain amount of activity to the feed-water storage tank. The non-condensable gases flow from the gas cooler through the vacuum pumps to the stack. The entire off-gas system including the feed-water storage tank and feed-water pumps is under vacuum. It has proved to be very difficult to completely seal such a large system with so many possibilities for leakage. All leaks in the system would yield an apparent increase in the off-gas flow. At this time the leakage rate is between 100 and 200 l/h which is in excess of the designed storage capacity. Unfortunately, the inleaking air is mixed with the non-condensable activated off-gas from the primary circuit. However, the activity of the off-gas is so extremely low that the radiation level at the stack is far below the upper licence limit. Because of this low level we are permitted a temporary exception to the licence requirement of two days' storage time for all off-gas before discharge.

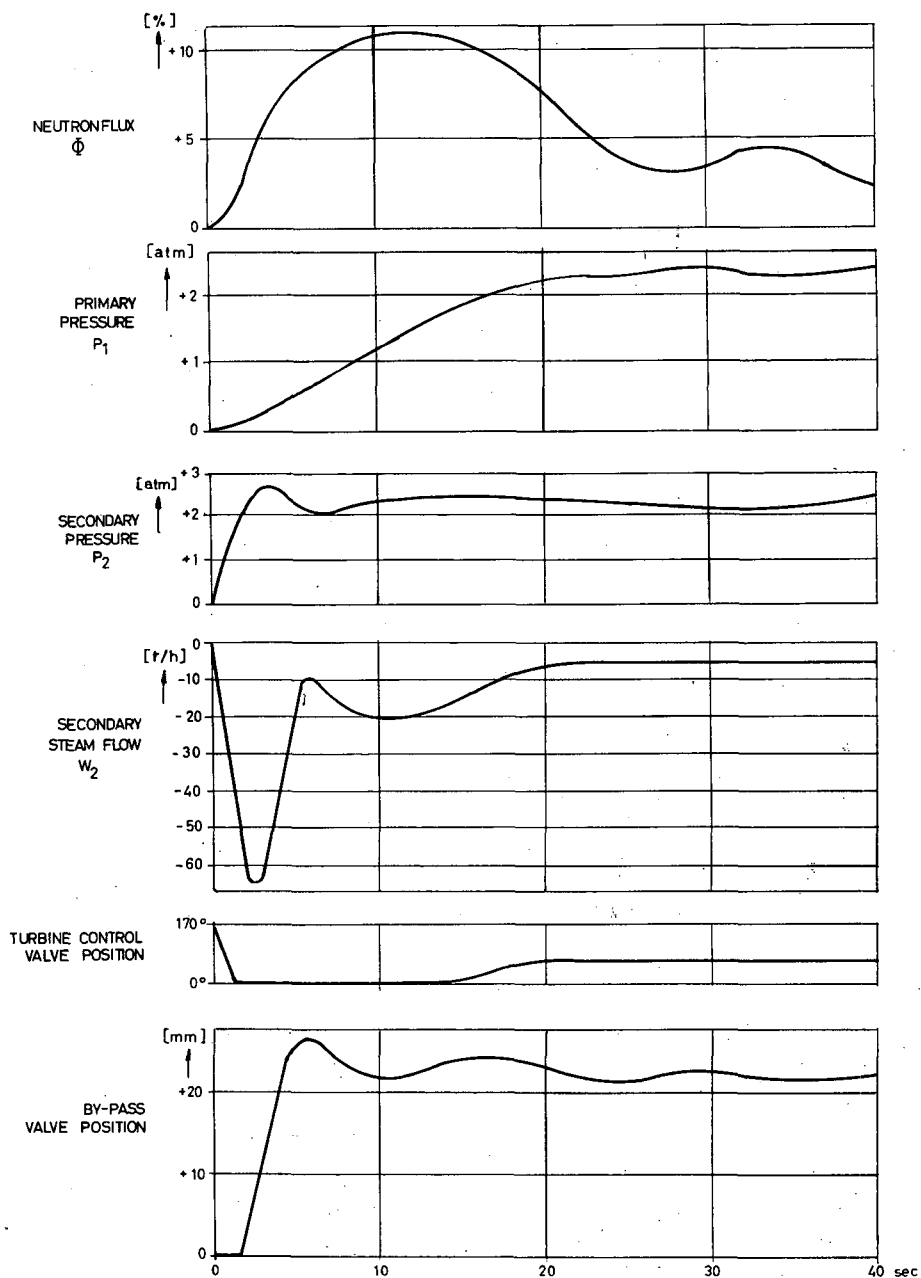


Fig. 10

Load rejection at 12.8 MW(e) turbine power
Starting values:

$\phi = 88\%$
 $p_1 = 70$ atm gauge
 $p_2 = 54$ atm gauge

$w_2 = 86$ t/h
vacuum = 89%

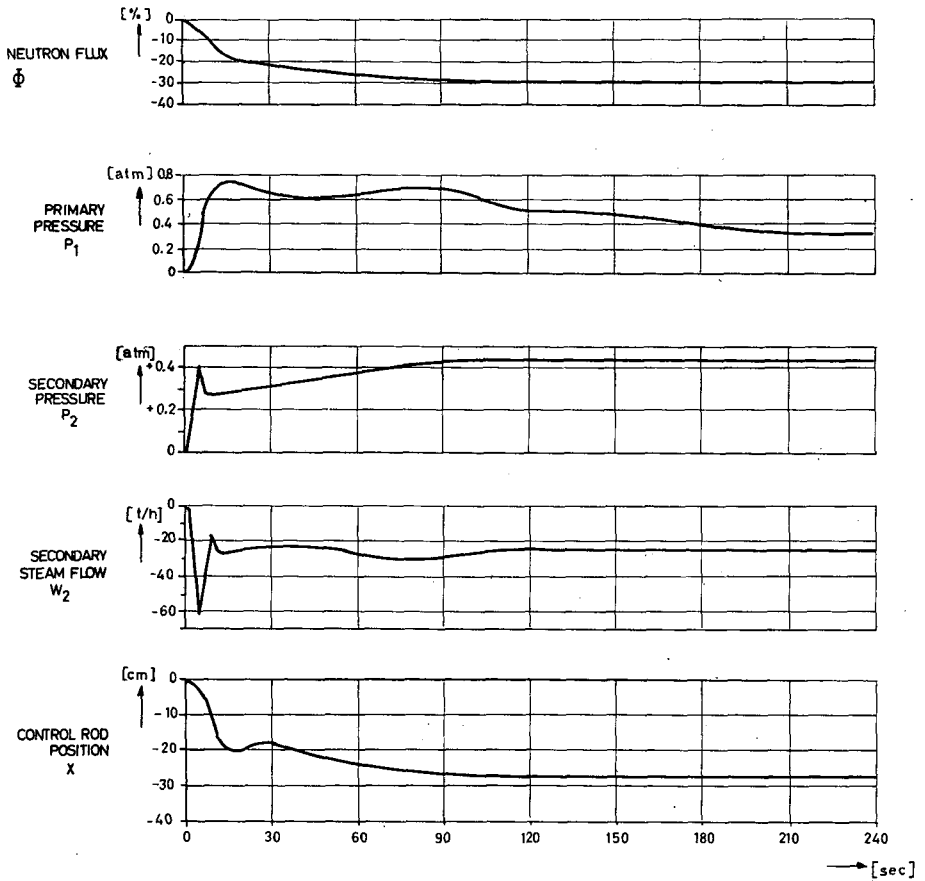


Fig. 11

Load rejection at 16 MW(e) turbine power
(Reactor pressure regulator in operation)

Starting values:

$$\phi = 95\%$$

$$p_1 = 70 \text{ atm gauge}$$

$$p_2 = 52 \text{ atm gauge}$$

$$w_2 = 94 \text{ t/h}$$

$$x = 79 \text{ cm}$$

$$\text{proportional band} = 60 \text{ cm/atm}$$

$$\text{reset position} = 0.36 \text{ (cm/s) / atm}$$

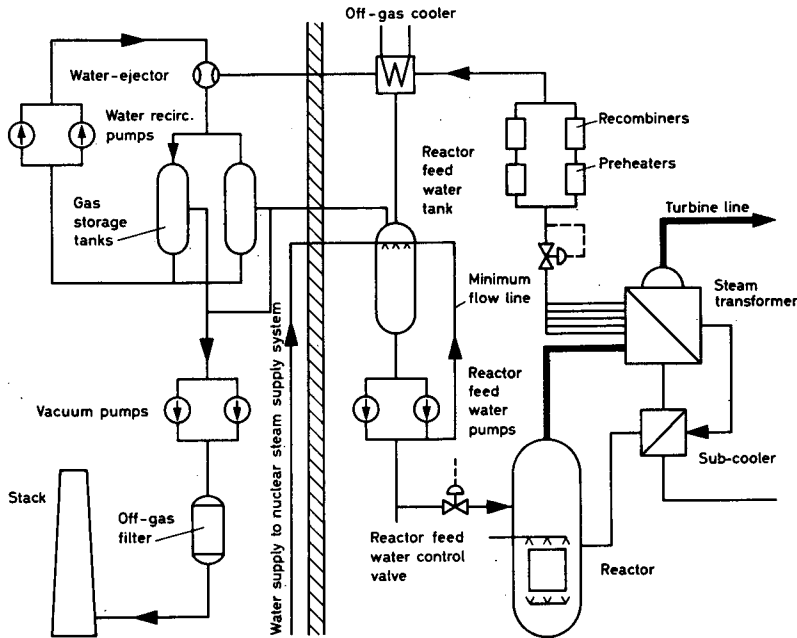


Fig. 12

15-MW(e) Kahl Nuclear Power Station
Off-gas system

2.6. Environmental release of activity

The licensing authorities have postulated narrow limits for the environmental release of activity from the plant. So, an amount of air activity of 8 c/h at normal weather conditions and 400 mc/h at inversion conditions may be discharged from the stack after a delay time of two days. The off-gas activity at the outlet of the stack presently amounts to 3 to 8 mc/h, i.e. 1% of the upper permissible limit. Hence, it is almost impractical to prove an influence of the discharged air on the environment. Fig.13 shows the environmental airborne activity measured for 1961 and 1962. It can be seen that certain atom-bomb tests in fall 1961 caused an increase of the activity of the environmental air by a factor of about 400. Under these circumstances it happened for several times that the activity of the outlet air, after passing through filters, was much lower than that of the intake air.

The same situation holds for the environmental activity release to the Main River. The accumulated amount of water activity can be deduced from Fig.14. The licence permits an activity release of 0.6 mc/week. It can be seen that activity release has kept well below this upper limit.

2.7. Radiation survey

Fig.15 shows the radiation levels as measured at the different floors in the reactor building. Two numbers are given: the first giving the total

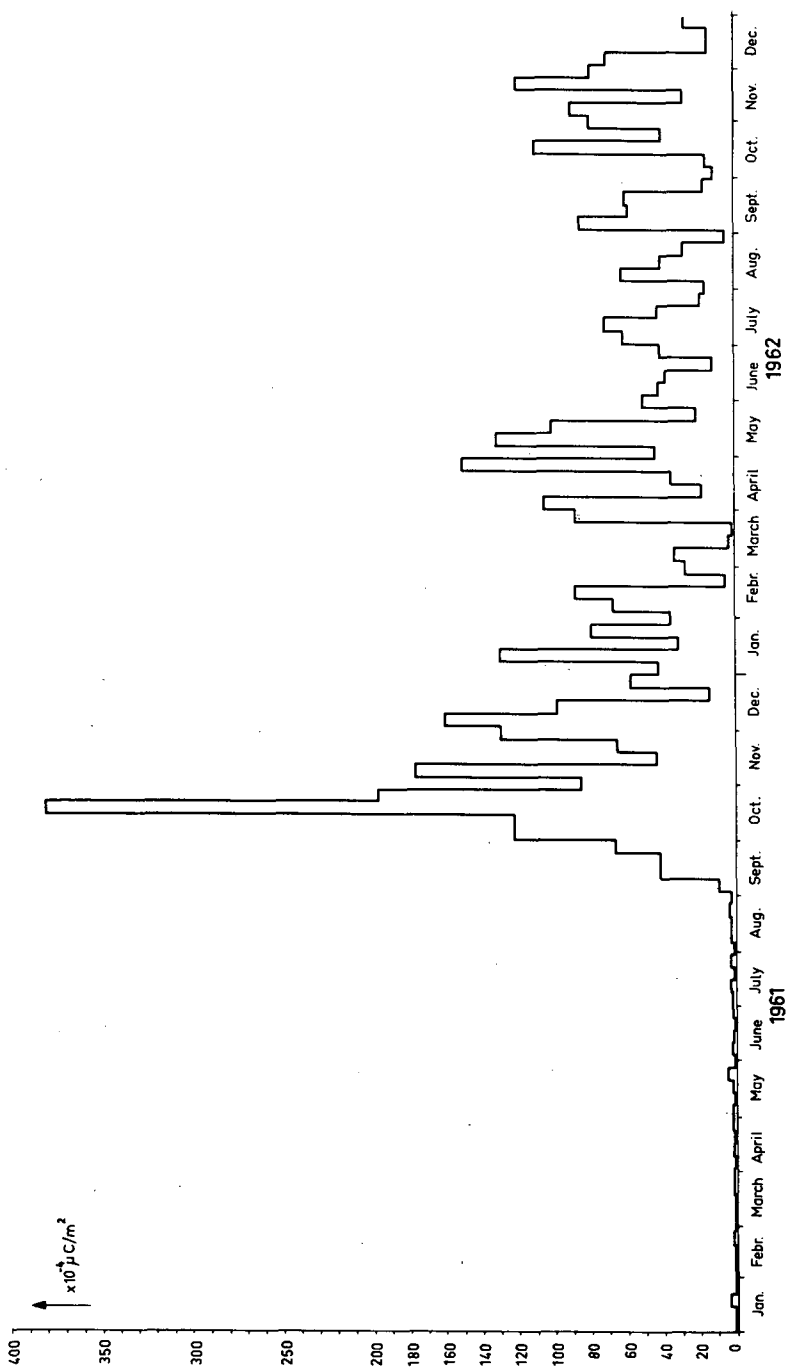


Fig. 13
Environmental airborne activity

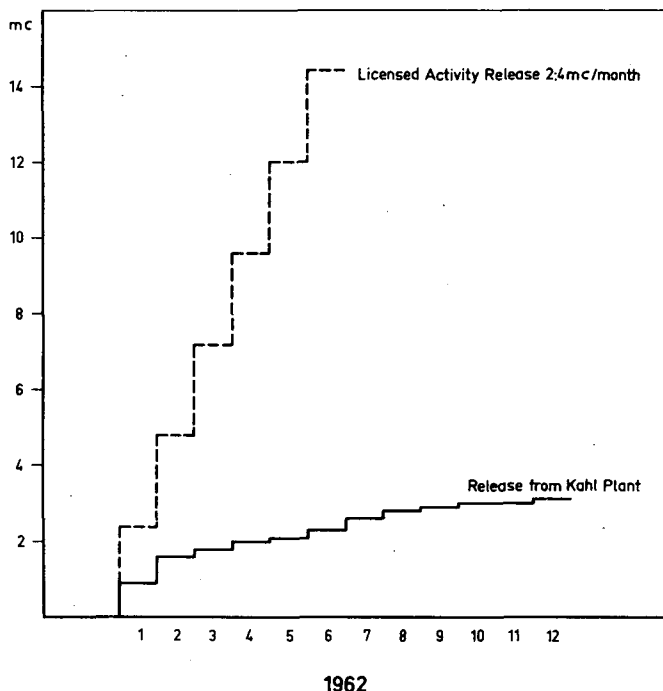


Fig. 14

Accumulated release of activity
to Main River in 1962

radiation, the second in parentheses the gamma fraction. As can be seen, the neutron radiation, consisting of fast neutrons to 90%, predominates. At normal operation no personnel is in the reactor building because the entire plant is controlled from the main control room. For occasional inspections the staircases outside the biological shielding are used. Along this path, the radiation level is below 10 mrem/h.

2.8. Conclusions from the performance of the plant

The following conclusions can be drawn from the first two years of operating experience at the Kahl Nuclear Power Station. At no time was the safety of both the plant and the environment at stake. No instabilities were observed in the course of the various operating conditions tested. The plant can be operated satisfactorily by normally trained personnel. This very favourable operating experience has contributed considerably to the positive decision recently taken by RWE and Bayernwerk to order a 250-MW(e) dual-cycle boiling-water reactor from the same group of companies that built the Kahl Plant. This first full-size nuclear power station in Germany is already under construction.

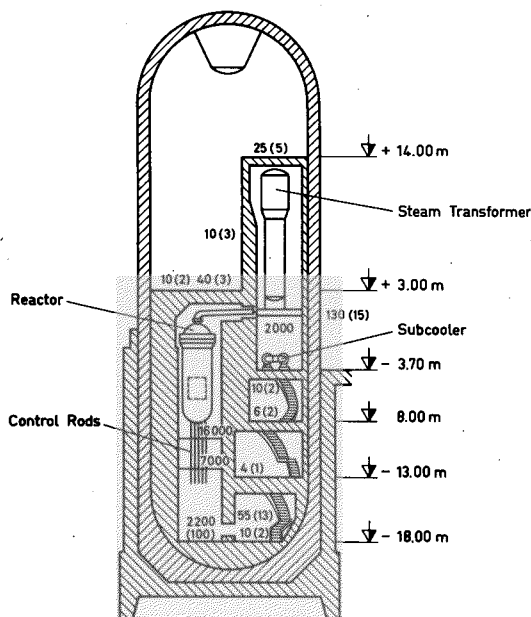


Fig. 15

Radiation levels in the reactor building

First figure $\alpha n + \gamma$ mrem/hSecond figure in parentheses $\beta \gamma$ mrem/h

3. FUTURE UTILIZATION OF THE KAHL NUCLEAR POWER STATION

3.1. Collection of fuel performance experience

Now the immediate objective of the further operation of the Kahl Station is to demonstrate that the fuel burn-up of 8800 MWd/t of uranium guaranteed for the first core charge will be attained. In order to reach this burn-up target it will be necessary both to reshuffle the fuel elements of the first core charge and to replace a quarter of the total core by new elements. This occasion is being used to give the domestic German fuel industry a chance to supply and to test their fuel elements. Accordingly 22 new fuel element assemblies have been ordered from German companies.

3.2. Installation and operation of a nuclear superheat test loop

Within the scope of the German atomic programme for advanced and medium-sized reactors, which is supported by the Federal Ministry of Scientific Research, a nuclear superheat test loop is about to be installed in the Kahl reactor. Saturated steam generated in the reactor will be transmitted to the loop after drying. The loop is designed for the simultaneous testing of four full-size superheating elements. The superheated steam leaving the test elements will then be transmitted to filters in order to ex-

tract possible solid impurities present. Thereafter the superheated steam will pass a measuring section and a throttle valve before entering a heat exchanger where it will be condensed. The condensate will finally be returned to the reactor pressure vessel by a booster pump.

The power extracted from the loop amounts to less than 10% of the total rated thermal reactor output. This means that the reactor will continue to operate at full power, while the loop tests will be run. The steam flow rate is expected to be 2 t/h each for two test elements and 9 t/h each for the other two elements. Successful results of the preceding series of tests with single rods and tubes provided, eventually steam temperatures of about 500°C at about 70 atm should be attained.

The operation of the nuclear superheating loop is to commence towards the end of this year. The results will form a substantial asset for the construction of an integral nuclear superheating prototype reactor. From this it becomes manifest that the Kahl Nuclear Power Station has not only provided valuable experience in the construction and operation of a boiling-water reactor, but that it is now rendering an important contribution to the further development of this reactor type.

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DISCUSSION

J. STOLZ: Figure 5 in your paper shows the net utilization as being often higher than the availability coefficient, for example during December 1962. What is the explanation of this apparent contradiction?

H. J. BRÜCHNER: This is not a contradiction but a matter of definition. In December 1962, for instance, where we have shown a net utilization of 100% and a plant availability of 98.8%, the plant was operated at full power for the whole time that it was available and it was available for 98.8% of the month. In other words, the product of net utilization and plant availability is the "plant load factor".

G. ZORZOLI: Which techniques did you use for measuring the flow and neutron flux distribution? And when you speak of good agreement between calculated and experimental values what exactly do you mean?

H. J. BRÜCHNER: The techniques used for flow and neutron flux measurements at Kahl are described in detail in an article by Kühnel and Misenta.* This article also gives some indication of the agreement between measurements and calculations.

* Reference [7] to H. J. Brüchner and A. Weckesser: "The first two years of operating experience of the Kahl Nuclear Power Station", these Proceedings.

M. POTEMANS: You said that electrically-operated valves are placed in the extraction lines to prevent the turbine overspeeding. What is the response time of these valves and at what turbine speed does tripping occur?

A. WECKESSER: The response times of the valves are less than one second, and the trip speed of the turbine is 3250 rpm.

G.B. SCURICINI: What are the differences between the Kahl and KRB plants? Will the control rod driving system at KRB be the same as at Kahl, or do you plan to use a hydraulic drive as at Dresden and SENN?

H.J. BRÜCHNER: Concerning the differences between the KRB plant at present under construction at Grundremmingen and the Kahl station, the following information may be of interest: The Kahl station has an indirect-cycle boiling-water reactor and the Grundremmingen plant has a dual-cycle boiling-water reactor. Whereas, in the Kahl station, the steam-water separation devices are in the steam transformer, at Grundremmingen they will be in the reactor. There are further differences in the design of the core. The KRB core will consist of 368 fuel elements containing a total of 51 t of uranium with an average U^{235} enrichment of 2.5%. The specific reactor rating therefore amounts to 15.7 kW/kg of uranium compared to only 10.9 kW/kg of uranium at Kahl. Since we have eliminated the separate steam drum as used in the Dresden and SENN plants and since we have devised a more compact component arrangement in the reactor containment, the volume of this containment can now be reduced from 110 m³/MW(t) at Kahl to 44 m³/MW(t) at Grundremmingen.

H. KLEIJN: What is the minimum reactor period allowed at Kahl during changes from low to high power?

A. WECKESSER: During the load-drop tests, no period readings were taken. Normally the period "scram" point is set at 10 s.

PERFORMANCE CHARACTERISTICS OF THE EXPERIMENTAL BOILING WATER REACTOR FROM 0 TO 100 MW(t)

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Abstract — Résumé — Аннотация — Resúmen

PERFORMANCE CHARACTERISTICS OF EBWR FROM 0-100 MW. On 25 May 1962 the Argonne National Laboratory received approval from the USAEC to operate EBWR to a power level of 100 MW. Administrative approval to proceed was granted by the International Atomic Energy Agency safeguards system on 11 July 1962. On 15 November 1962 an operating power level of 100 MW was reached.

The EBWR 100 MW Reactor Experimental Program was completed on 6 December 1962. One of the major goals of this project was to instrument the reactor extensively in order to obtain data and information on the performance characteristics of this reactor type. The programme was the first of its kind to be undertaken and the first to be completed. Many new instrumentation techniques had to be developed for obtaining the desired data. The goal was successfully achieved, and many new data were obtained on the performance characteristics of a natural circulation boiling-water reactor.

The data derived from this programme provided information on recirculation flow rates, vapour liquid separation limits (steam carryunder in the downcomer and liquid carryover in the effluent steam), subcooling, location of the true interface in the reactor and its relation to the water column level, steam collapse rates in the downcomer, void coefficients, H_2BO_3 worth, temperature coefficients, use of boron strips for control purposes, use of spike elements, transfer functions, noise analysis, some flux measurements, stability, etc. In addition, data were obtained on the behaviour and integrity of certain reactor components and systems such as boric acid control, radiation levels, corrosion product distribution, equipment malfunctions, fuel and control rods, etc.

EBWR's performance characteristics were governed almost exclusively by steam carryunder in the downcomer, liquid carryover in the effluent steam, and indirectly by the location of the true interface in the vessel. Carryunder was the dominating factor in the lower power range. Above 65 MW the reactor's performance characteristics changed radically. The steam disengaging velocity reached 1 ft/s and the steam dome height decreased to 3 ft. Under these conditions liquid carryover occurred and increased rapidly with increasing power. The reactor no longer behaved as a direct-cycle boiling-water reactor; in a sense it functioned as a natural-circulation dual-cycle reactor.

PERFORMANCES DE L'EBWR DE 0 A 100 MW. Le 25 mai 1962, le Laboratoire national d'Argonne a reçu l'autorisation de la Commission de l'énergie atomique des Etats-Unis (CEA-EU) d'exploiter le réacteur expérimental à eau bouillante (EBWR) jusqu'à une puissance de 100 MW. Dans le cadre de l'administration du système de garanties, l'Agence internationale de l'énergie atomique a autorisé à aller de l'avant le 11 juillet 1962. Le 5 novembre 1962, on a atteint la puissance de 100 MW.

Le programme expérimental, réalisé avec le réacteur EBWR de 100 MW, a été achevé le 6 décembre 1962. L'un des principaux objectifs du projet était de pourvoir le réacteur de tous les instruments de mesures nécessaires pour obtenir des données et des renseignements sur les performances de cette filière. C'est le premier programme de ce genre qui ait été entrepris. On a mis au point plusieurs techniques d'instrumentation nouvelles pour obtenir les données voulues. Cet objectif a été atteint avec le plus grand succès, et on a pu recueillir beaucoup de données nouvelles sur les performances d'un réacteur à eau bouillante à circulation naturelle.

Ce programme a permis d'obtenir des indications sur les points suivants: débit de recirculation; limite de séparation de la vapeur et du liquide (entraînement de vapeur dans le tube d'eau et de liquide par la vapeur); sous-refroidissement; localisation de l'interface vraie dans le réacteur et son influence sur le niveau de la colonne d'eau; taux de condensation de la vapeur dans le tube d'eau; coefficients cavitaires; antiréactivité de l'acide borique; coefficients de température; emploi de bandes de bore à des fins de contrôle; emploi d'éléments enrichis; fonctions de transfert; analyse des bruits; certaines mesures de flux; stabilité, etc. En outre, on a pu savoir comment se comportaient certaines pièces et parties constitutives du réacteur et si elles restaient intactes,

notamment le système de contrôle de l'acide borique, le niveau des rayonnements, la distribution des produits de la corrosion, les vices de fonctionnement du matériel; les barreaux de combustible et les barres de commande, etc.

Les performances de l'EBWR dépendent presque exclusivement de l'entraînement de la vapeur dans le tube d'eau, de l'entraînement du liquide par la vapeur qui se dégage, et, indirectement, de l'emplacement de l'interface vraie dans le caisson. C'est l'entraînement de la vapeur dans le tube d'eau qui domine pour les puissances inférieures. Au-dessus de 65 MW les performances du réacteur sont radicalement modifiées. La vitesse de dégagement de la vapeur atteint 33 cm/s et la hauteur du dôme de vapeur descend à 1 m. Dans ces conditions, il se produit un entraînement de liquide par la vapeur qui augmente rapidement lorsqu'on augmente la puissance. Le réacteur cesse alors de se comporter comme un réacteur à eau bouillante à cycle direct; en un sens, il fonctionne comme un réacteur à deux cycles, en circulation naturelle.

РАБОЧАЯ ХАРАКТЕРИСТИКА ЭКСПЕРИМЕНТАЛЬНОГО КИПЯЩЕГО РЕАКТОРА EBWR ПРИ МОЩНОСТИ 0 - 100 МГВТ. 25 мая 1962 года Аргоннская национальная лаборатория получила разрешение КАЭ США на эксплуатацию реактора EBWR на мощности 100 мвт. Административное разрешение на эксплуатацию реактора было предоставлено системой гарантий Международного агентства по атомной энергии 11 июля 1961 года. 15 ноября 1962 года был достигнут уровень мощности в 100 мвт.

6 декабря 1962 года экспериментальная программа была закончена. Одной из основных целей ее была тщательная проверка реактора для получения данных и информации рабочей характеристики этого типа реактора. Эта программа была первой программой такого рода и первой выполненной программой. Для получения нужных данных необходимо было разработать многие новые приборы. Цель была достигнута, получено много новых данных о рабочей характеристике кипящего реактора с естественной циркуляцией.

Так, например, получена информация относительно скорости потока циркуляции в замкнутом цикле, пределов сепарации жидкого пара (выделение пара в осадок в спускной трубе и унос жидкости эфлуентом пара), недогрева, локализации действительной поверхности раздела в реакторе и ее связи с уровнем водной колонки, скорости разрушения пара в спускной трубе, пустотных коэффициентов, реактивной способности H_2BO_3 , температурных коэффициентов, использования стержней из бора для целей контроля, использования свежих топливных элементов, передаточных функций, анализа шума, некоторых измерений потока, стабильности и т.д. Кроме того, были получены данные о поведении и целостности некоторых реакторных компонентов и систем, таких, как борнокислая контрольная реакция, уровни радиации, распределение продуктов коррозии, выход из строя оборудования, топливо и регулирующие стержни и т.д.

Рабочая характеристика реактора EBWR определялась почти исключительно по выделению пара в осадок в спускной трубе, по уносу жидкости эфлуентом пара и косвенно, путем локализации действительной поверхности раздела в корпусе реактора. Выделение пара в осадок было доминирующим фактором в диапазоне более низких энергий. При мощности свыше 65 мвт рабочая характеристика реактора резко менялась. Скорость отделения пара достигла 33 см/сек и высота парового пространства уменьшилась до 1 м. При таких условиях происходил унос жидкости, который быстро увеличивался с увеличением мощности. Реактор больше не вел себя как кипящий реактор с прямым циклом; в некотором роде он вел себя как реактор с двойным циклом естественной циркуляции.

RENDIMIENTO DEL REACTOR EXPERIMENTAL DE AGUA HIRVIENTE (EBWR) ENTRE 0 Y 100 MW. El 25 de mayo de 1962, el Laboratorio Nacional de Argonne fue autorizado por la USAEC a poner en funcionamiento el EBWR con una potencia de 100 MW. En el marco de la administración de su sistema de salvaguardias el Organismo Internacional de Energía Atómica dio su aprobación el 11 de julio de 1962. El 15 de noviembre del mismo año, el reactor alcanzó la potencia de 100 MW.

El programa experimental ejecutado con el reactor EBWR de 100 MW quedó completado el 6 de diciembre de 1962. Uno de los principales propósitos del mismo consistía en dotar al reactor de los instrumentos necesarios para obtener datos e informaciones sobre el rendimiento de este tipo de reactor. Constituye el primer programa de este género que se haya llevado a la práctica. Para reunir los datos buscados, fue preciso idear muchas técnicas de instrumentación nuevas. Se logró el propósito perseguido y se obtuvieron copiosos datos nuevos acerca del rendimiento de un reactor de agua hirviente con circulación natural.

Este programa permitió obtener indicaciones sobre los siguientes puntos: caudales de recirculación, límites de separación entre vapor y líquido (arrastre de vapor en el tubo de descenso y arrastre de líquido por el vapor efuyente), subenfriamiento, posición de la interfase real en el reactor y su relación con el nivel de la columna

de agua, coeficientes de condensación del vapor en el tubo de descenso, coeficientes de cavitación, antirreactividad del H_2BO_3 , coeficientes de temperatura, utilización de cintas de boro para la regulación, empleo de elementos combustibles enriquecidos, funciones de transferencia, análisis de los ruidos, algunas mediciones de flujos, estabilidad, etc. Además se obtuvieron datos sobre el comportamiento y el estado de ciertas partes y circuitos del reactor, tales como el sistema de control de ácido bórico, los niveles de radiación, la distribución de los productos de corrosión, los defectos funcionales de ciertos equipos, las barras de combustible y de control, etc.

Se comprobó que el rendimiento del EBWR depende casi exclusivamente del arrastre de vapor en el tubo de descenso, del arrastre de líquido por el vapor efluente e, indirectamente, de la posición de la interfase real en el recipiente. En el intervalo de bajas potencias, el factor preponderante es el arrastre de vapor en el tubo de descenso. Al superar los 65 MW, el rendimiento del reactor cambió radicalmente. La velocidad de desprendimiento del vapor alcanzó a 33 cm/s, y la altura de la cúpula de vapor se redujo a 1 m. En estas condiciones se registró un arrastre de líquido que se intensificó rápidamente al aumentar la potencia desarrollada. El reactor deja entonces de comportarse como reactor de agua hirviendo de ciclo directo; en cierto modo, funciona como reactor de ciclo doble y circulación natural.

1. INTRODUCTION

The Experimental Boiling Water Reactor (EBWR) was originally constructed to demonstrate the feasibility of a direct-cycle, natural circulation, boiling-water reactor integrated power plant. The reactor was designed to produce 20 000 kW of heat in the form of 600-lb/in² gauge saturated steam which was fed directly to a turbo-generator, generating about 5000 kW of electricity. This power level was considered as a minimum value for producing useful information which could be extrapolated and used in the design of large, central-station power plants. The reactor achieved criticality in December 1956 and began full-power operation at 20 MW(t) on 29 December 1956.

Subsequent experiments at power levels ranging from 20 to 40 MW(t) indicated that stable operation at powers as high as 66 MW(t) was possible with the initial 4-ft diam. core. This was affirmed in 1958 by the short-term operation of the plant at 61.7 MW(t). Further power increase at this time was precluded by the feed-water pumps which were operating at maximum capacity.

A detailed study of EBWR stability was made by taking a series of transfer function measurements, relating flux or power level to reactivity input. Data at various power levels, steam pressures and control-rod positions were accumulated, analysed, and extrapolated to predict reactor performance at higher power levels. The results of this study were the subject of a paper presented at the 1958 Geneva Conference [1].

Subsequent analyses and projection of these data to a 5-ft diam. core indicated that with some modification of the core structure and pressure vessel internals, and with additional heat-removal equipment, EBWR could operate at or near 100 MW(t). In July 1959 the reactor plant was shut down preparatory to the onset of proposed modifications.

The modification of EBWR represented another successful pioneering endeavour at the birthplace of nuclear reactors. In addition to the system changes and additions that were effected, the core was extensively instrumented. This entailed development of unique instrumentation techniques to ensure qualitative data on the performance characteristics of the reactor at elevated power levels.

On 25 May 1962 permission was granted by the United States Atomic Energy Commission (USAEC) to operate EBWR at power levels up to 100 MW(t). Administrative approval by the International Atomic Energy Agency was granted on 11 July 1962.

Upon completion of the modifications the reactor was subjected to a series of stability tests at successive power level increases up to 70 MW(t). After careful analysis and scrutiny of the data, it was decided that operation at 100 MW(t) could be achieved safely.

This paper summarizes the significant observations and reactor performance data evolved during the period July 1959 to 6 December 1962. The latter date marked the shut-down and termination of the water-cooled reactor programme at the Laboratory. The majority of the data were obtained during the last six months of operation.

The pertinent reactor design data are summarized in Table I. Table II lists the reactor-generator operation and output accumulated at the time of shut-down.

TABLE I

REACTOR DESIGN DATA

Maximum power	100 MW(t)			
Maximum operating pressure	600 lb/in ² gauge			
Number of elements	147			
Fuel sub-assembly				
Plate	115 (Metallic U)			
Spike	32 (UO ₂)			
Clad	Zircaloy-2			
Total fuel content	6.2 t (5700 kg) ~ 90 kg U ²³⁵			
Active core diameter and height	5 ft x 4 ft			
Total fuel heat-transfer area in core	2480 ft ²			
Calculated heat-transfer data	Central thin enriched zone	Spike zone	Outer thin enriched zone	Thick enriched zone
Average power density (kW/l)*	113.3	130	92.8	77.1
Average heat flux (BTU ft ⁻² h ⁻¹)	225 353	147 160	185 000	118 000
Maximum heat flux (BTU ft ⁻² h ⁻¹)	346 000	246 100	283 605	172 000
Power removed (%)	31.25	23.27	11.37	34.11
Equivalent diam. of coolant channel (in)	0.781	0.45	0.781	0.653

* kilowatt per liter of coolant in core

TABLE II
ACCUMULATED REACTOR AND GENERATOR OPERATION

	Total up to 3/7/59	Total up to 6/12/62
Reactor		
h	11 225	15 665
MWh	211 509	294 028
Generator		
h	8 745	10 603
MWh	35 006	39 696

2. NATURE OF MODIFICATIONS

2.1. Core

2.1.1. Fuel assemblies

The core consisted of the original 115 plate-type fuel assemblies plus 32 enriched spike assemblies. Four variations of plate-type assemblies were employed (Fig. 1): the plates were thin (0.212 in) or thick (0.280 in) and contained natural or enriched (1.44% U^{235}) uranium. The natural uranium plates comprised a heat-treated ternary alloy of 93.5 wt. % natural uranium (0.72% U^{235}), 1.5 wt. % niobium, 5 wt. % zirconium with Zircaloy-2 cladding. The side plates were made thick enough (1/16 in) to weld to the fuel plates and were perforated to allow stretching (creep) in the event of fuel-plate elongation under irradiation. The enriched-uranium plate assemblies were structurally similar but contained uranium enriched to 1.44% in U^{235} .

The spike fuel-assembly concept was based on the desire to minimize research and development of new fuel assemblies and to utilize the maximum number of the original EBWR fuel assemblies. Each spike assembly (Fig. 2) comprised 49 fuel rods. Each fuel rod consisted of UO_2 fuel (> 90% enriched in U^{235}) dispersed in a $ZrO_2 + CaO$ matrix pellet and clad with Zircaloy-2. The gap between the pellet and the clad was filled with helium. Some flexibility in the worth of each spike was obtained by adding 1.1% boron stainless-steel poison strips to the spike frames. These strips were easily removed when additional reactivity was required. The worth of a spike element containing boron strips and located in the spike zone was as follows:

Spike assembly containing two boron strips	0.14% Δk
Spike assembly containing one boron strip	0.23% Δk
Spike assembly without boron strips	0.32% Δk

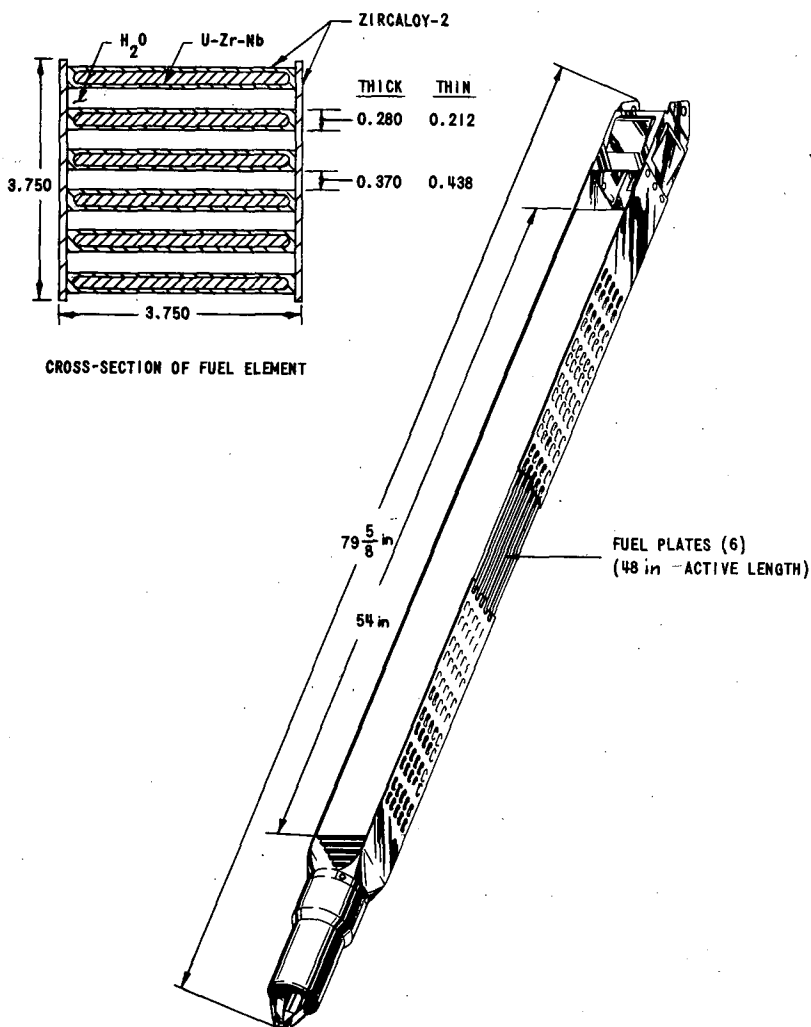


Fig. 1

EBWR plate-type fuel assembly

A stepwise approach in loading the reactor to the final configuration required a number of core changes. These included fuel rearrangement, fuel additions and the removal of some burnable poison strips from the spike assemblies; consistent with the established loading criteria. These criteria were:

- (1) The system must always have the capacity to be made subcritical at any time by use of only eight of the nine control rods and boric acid; and

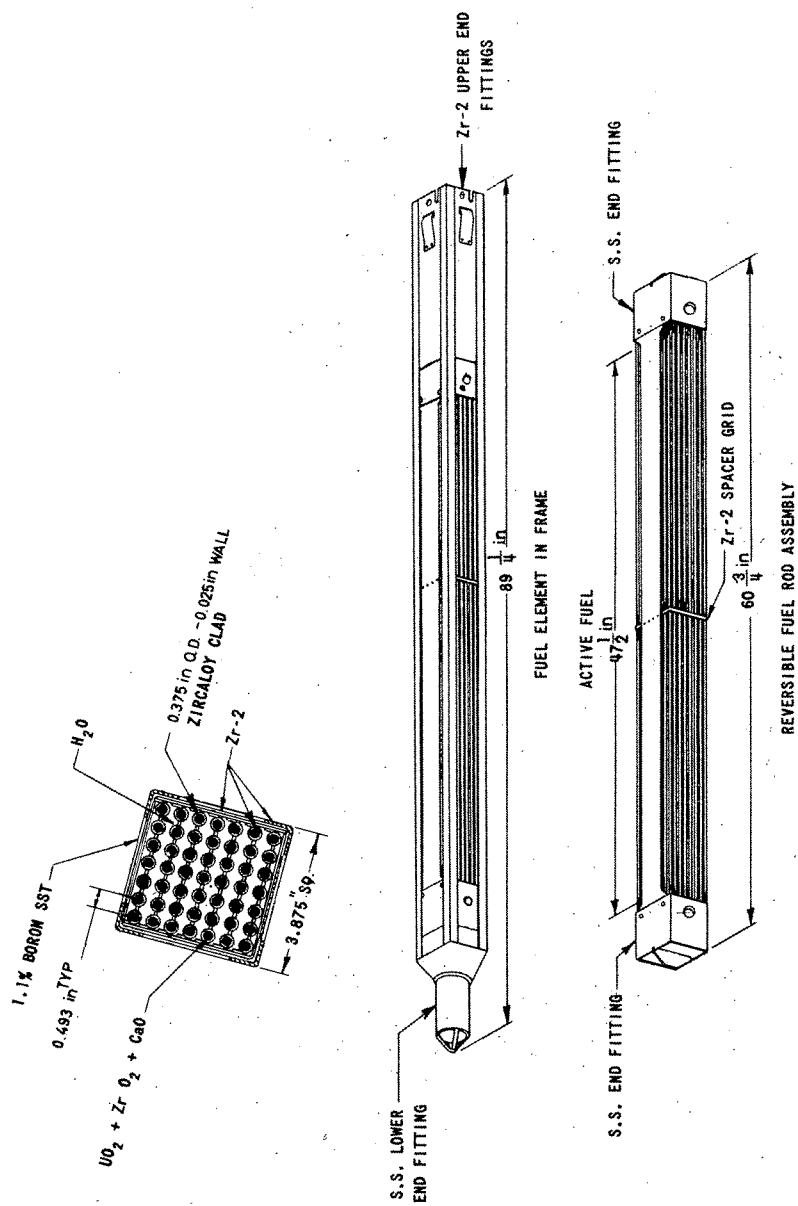


Fig. 2
EBWR spike fuel-rod assembly and frame

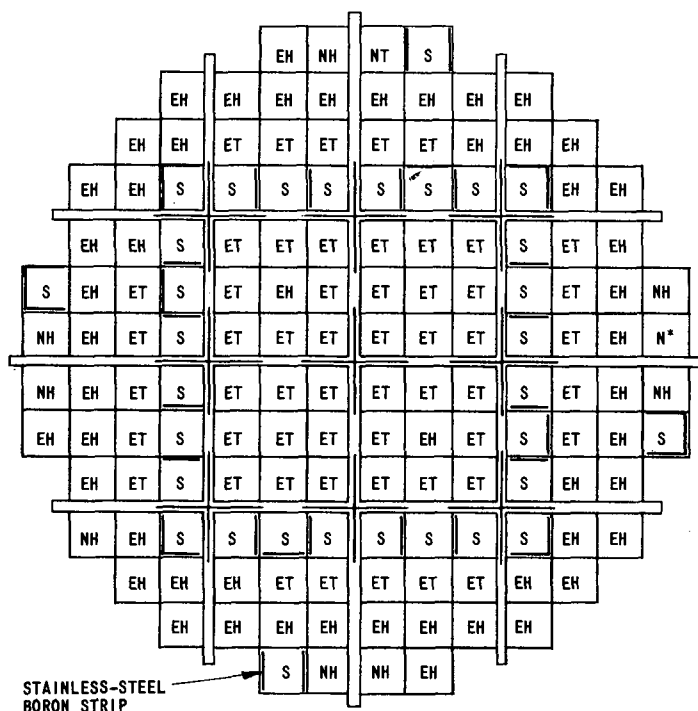


Fig. 3

Final EBWR 100-MW(t) core loading

52-EH = Enriched heavy

54-ET = Enriched thin

8-NH = Natural heavy

1-NT = Natural thin

32-S = Spike

1-N* = Source

- (2) No loading was to be installed which would be critical with all nine rods fully inserted and with no boric acid in the reactor core.

Fig. 3 shows the final arrangement of fuel assemblies in the core for operation at 100 MW.

2.1.2. Control

(a) Control rods. Primary reactor control was effected by nine cross-shaped rods. Each rod was 60-in long, 10-in wide, 0.25-in thick and fabricated from Type 304 stainless steel containing 2 wt. % boron. The 10-in wide portion of the control-rod follower was made of Zircaloy-2. These rods were identical to the original set of rods placed in the EBWR in 1956. The central control rod was replaced by a 4-ft long hafnium rod. This rod was used in conjunction with an oscillator mechanism to obtain transfer function data.

(b) Soluble poison (boric acid). In addition to the nine control rods, boric acid solution was used with the final core loading. This was done to provide a sufficient range of reactivity control to ensure a clean, cold shut-down margin and yet permit reaching a power level of 100 MW(t). The final core loading required 1060 ppm of boric acid in the reactor water in order to meet the 8-rod shut-down criteria. The reactor was also subcritical without boric acid and with all nine rods fully inserted. The void co-efficient remained negative at all concentrations up to 2640 ppm, which was the maximum range of interest. Each gram of boric acid per gal* of reactor water (264 ppm) was sufficient to hold down about 1% ρ .

Boric acid having a concentration of 182 g/gal was stored in a 200-gal tank. The solution could be pumped into the reactor feed-water line under pressure, pumped into the condenser hotwell, or into the low-pressure boric-acid spray ring located within the reactor vessel. Positive displacement pumps were utilized and could be energized from the control room. The maximum pumping rate was 2.8 gal/min of 182 g/gal solution. The reactor vessel has a normal water inventory of about 4000 gal.

2.2. Core structure and pressure vessel internals

The following modifications were designed primarily to increase reactor recirculation flow rates by approximately a factor of two in order to minimize the steam volume fractions in the core and, hence, loss of reactivity.

2.2.1. Core riser

The new riser arrangement (see Fig. 4) basically consists of an addition to the original system, which was essentially the control-rod guide superstructure. The total height is approximately 7.5 ft above the shroud. This height was chosen as the maximum feasible for maintaining the steam-water interface within a desired level range and for minimizing the liquid carry-over. The upper portion of the riser was necked down so that the increased downcomer area and resultant reduced water velocity would aid de-entrainment of steam in the downcomer.

In addition, a cylindrical shell was installed around the core to provide a tight closure between the riser and the core shroud. These modifications were intended to prevent any short circuiting of the fluid from the downcomer to peripheral riser sections and core due to static pressure differences inherent in such systems.

2.2.2. Feed-water injection ring

The effectiveness of the riser was enhanced by installing a new feed-water injection ring just below the top of the riser. With this arrangement, steam carried under in the downcomer would be mixed with the influent cold feed water and quenching would be initiated rapidly.

* US gallons throughout

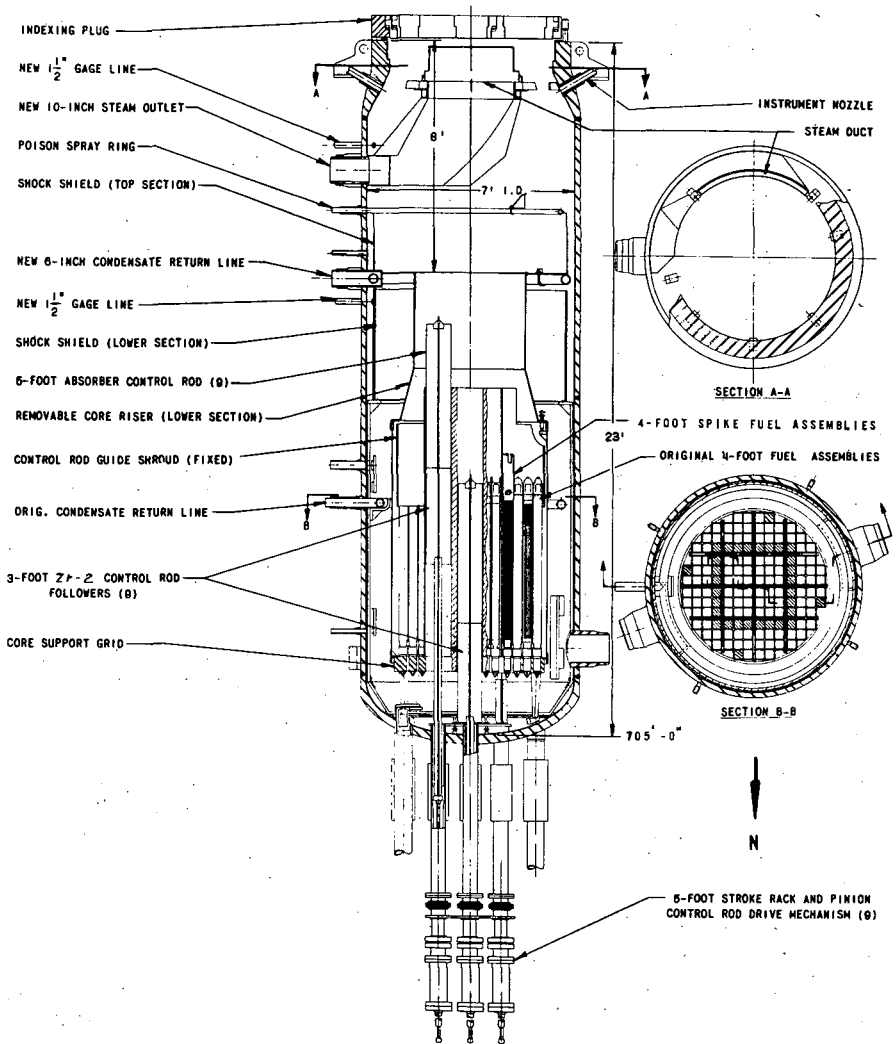


Fig. 4

Modified reactor vessel and internals

2.2.3. Steam discharge system

The original steam discharge ring and 6-in outlet nozzle were removed from the pressure vessel and the vessel penetration was enlarged to accommodate a 10-in discharge nozzle. The larger size outlet was necessary because of the increased steam loads at the higher powers. At 100 MW, the steam load is approximately 300 000 lb/h. The steam is collected from the very top of the reactor vessel by a duct arrangement. The duct arrange-

ment was installed to accommodate the high liquid levels brought about by the addition of the taller riser.

2.3. Auxiliary equipment

Additional equipment installed within the containment shell included two feed-water filters, a de-aerator, sub-cooler, two feed-water pumps and an instrument air compressor. A 16-ft long gauge board was installed in the control room. A major portion of the equipment was located external to the containment building (see Fig. 5).

2.4. System flow cycle

Fig. 5 is a simplified flow diagram of the 100 MW(t) reactor plant design. The reactor was operated as a direct-cycle, boiling-water reactor cooled and moderated by natural circulation of light water. Except for minor modifications, the 5 MW(t) turbo-generator plant and associated equipment was operated in essentially the same manner as originally designed. The Reboiler Plant was operated in parallel to handle the increased heat output. Distribution of the thermal power output at 100 MW was as follows: 20 MW(t) was utilized by the Turbine Plant, and the balance (up to 80 MW(t)) was absorbed by the Reboiler Plant.

The Reboiler Plant is divided into primary, intermediate, and secondary systems. The primary system is operated at a pressure of 560 lb/in² gauge saturated, the intermediate system at 350 lb/in² gauge saturated and the secondary system at 200 lb/in² gauge. The secondary system operates at the same pressure as the Laboratory steam system. An intermediate system was incorporated to preclude radioactive steam from entering the Laboratory heating system. The Reboiler Plant can supply steam to the Laboratory or dissipate the energy through air-cooled heat exchangers. Details of the system and components are described by MATOUSEK [2].

A number of problems were encountered with new equipment during start-up of the plant. The most predominant was maloperation of valves of all types. For the most part, the difficulties experienced were attributable to poor quality-control inspection by the makers, or to improper installation by the contractor.

Pumps and controls also presented some operational problems. For example, the contractor had installed a pump cut-off switch at the same level as the flash tank low-level alarm. This condition could have damaged the large, intermediate feed-water pumps but fortunately was discovered in time and rectified. All four sleeve bearings on both intermediate pumps required reworking. Insufficient lubrication caused the bearings to exceed normal operating temperatures. This situation was corrected by cutting grooves in the Babbitt metal bearings to enhance oil lubrication of the pump shafts. Pipe lines to the level controller, sight gauge and level alarm on both primary reboilers were revised to prevent particulate matter from plugging the lines and giving false indications.

After these and other equipment problems were corrected, plant performance was satisfactory. The modified plant was never subjected to shake-down operation in advance of the experimental programme. As a result,

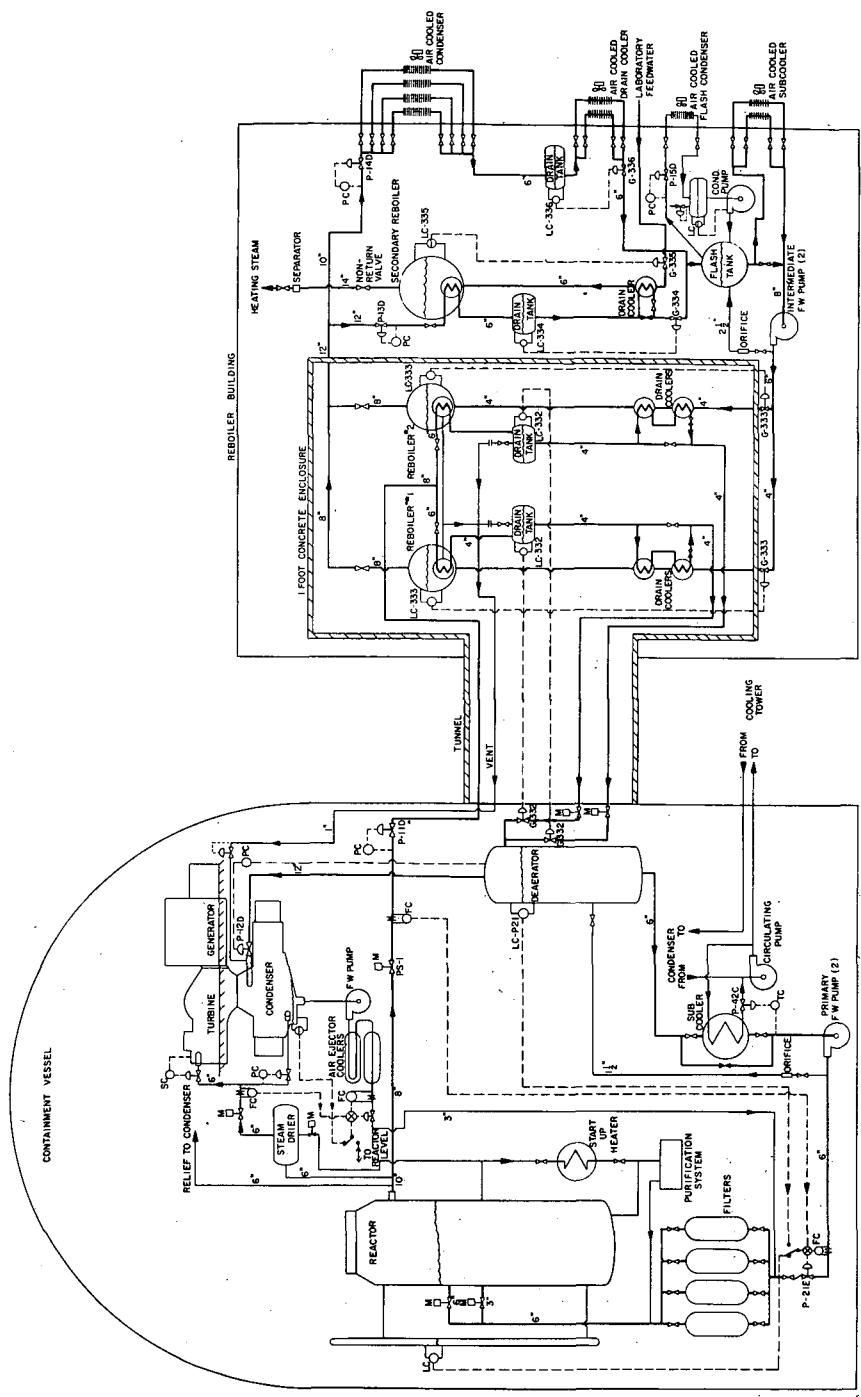


Fig. 5
EBWR 100-MW(t) flow diagram

the shake-down and the experimental programme were carried out simultaneously in order to complete the programme on schedule.

3. TEST PROGRAMME RESULTS AND SPECIFIC OBSERVATIONS

3.1. *Thermal hydraulic performance*

The thermal hydraulic test programme on EBWR was focussed primarily on the vapour-liquid separation process and its effect on reactor performance. By adding the taller riser and raising the operating water level, the steam-water separation problem was greatly aggravated. This problem may be categorized into three main parts: (1) steam carry-under in the downcomer; (2) liquid carry-over in the effluent steam; and (3) maintaining the actual vapour-liquid interface within a fixed height range.

Because the degree of primary steam separation that would take place within the reactor vessel was virtually unknown and because of the tremendous effect this could have on reactor performance, a decision was made to instrument the reactor extensively. The instrumentation installed was designed to provide data or information on the recirculation flow rate; volumetric and weight fraction of vapour carry-under in the downcomer; reactor sub-cooling; riser void fractions; rate of quenching of entrained vapour in the downcomer; location of the true two-phase mixture interface within the reactor; and liquid carry-over in the effluent steam.

3.1.1. Experimental techniques

A complete description of the experimental techniques and a description of the thermal hydraulic instrumentation are given in Ref.[4]. Briefly, the data were acquired in the following manner.

(a) Static pressure differentials. Static pressure differentials were measured with specially designed probes located at the positions shown in Fig. 6. Three probes were located in the downcomer, two in the riser and two above the riser. The static pressure differentials are readily converted to steam volume fraction since frictional and acceleration pressure drops may be considered as negligible. The three probes in the downcomer provided data on the magnitude of steam carry-under, how rapidly it was quenched and the loss of driving head due to the voids present in the downcomer. The steam volume fraction measurements derived from the riser probes were used as a basis of comparison with the calculated performance characteristics of the reactor. The two probes located above the riser provided data on the vapour hold-up and aided in locating the actual vapour-liquid interface.

(b) Total recirculation flow rate. Because of the occurrence of carry-under, the recirculation flow rate can no longer be determined by the sub-cooling technique. An impact meter was, therefore, selected for measurement of the recirculation velocity; more specifically, a stauschiebe tube,

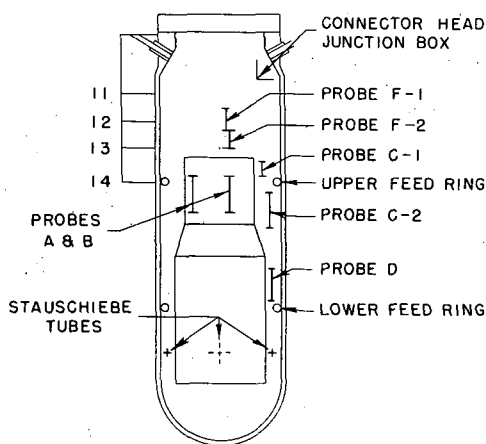


Fig. 6

Location of instrumentation in EBWR pressure vessel

because it yields an approximately 50% greater reading than the normal impact meter such as a pitot tube. Three stauschiebe tubes were placed in the downcomer approximately 120 degrees apart and at three different radii to obtain equal area readings.

(c) Sub-cooling. Sub-cooling measurements were made with insulated, chromel-alumel thermocouples sheathed in a tube (1/16 in O.D.). Six couples were located at various positions in the downcomer near the bottom of the reactor core. Three of the couples were attached to the stauschiebe tubes.

(d) Vapour-liquid interface. The vapour-liquid interface was located by a series of differential pressure readings taken from five taps penetrating the reactor vessel in the region of riser discharge (see Fig. 6, positions 11, 12, 13 and 14). By comparing the series of readings with equivalent lengths of pure steam and saturated water, the vapour-liquid interface can be pinpointed within a limited height range.

(e) Steam carry-under. The steam carry-under was calculated from the measured recirculation flow rates and sub-cooling by means of the following heat balance:

$$\frac{W_T}{W_T - W_I} (h_f - h_{in}) - \frac{W_I}{W_T - W_I} (h_f - h_I) + X_D (h_{fg}) = X_R (h_f - h_M),$$

where

W_T = total recirculation flow rate (lb/h)

W_I = ion exchanger flow (lb/h)

h_f = enthalpy of saturated liquid (BTU/lb)

h_{in} = enthalpy of inlet core coolant (BTU/lb)

h_i = enthalpy of ion exchanger effluent (BTU/lb)

X_D = steam quality in downcomer

h_{fg} = heat of vaporization

X_R = steam quality in riser

h_M = enthalpy of feed water (BTU/lb).

In the event that liquid carry-over occurs in the effluent steam, the heat balance must be modified and the following equation results:

$$\frac{X_D}{X_R} = \frac{(W_M/W_T)(h_f - h_M) - (h_f - h_{in})}{(Q/W_T) - (h_f - h_{in})},$$

where:

Q = reactor power (BTU/h)

W_M = feed water flow rate (lb/h)

W_T = total recirculation flow rate (lb/h)

$$X_D/X_R = \frac{\text{pounds steam in downcomer}}{\text{pounds steam in riser}}$$

(f) Liquid carry-over. Data on the magnitude of the liquid carry-over were obtained from three sources. The principal and most accurate source was a series of heat balances on the secondary and primary heat dissipation systems. A heat balance on the secondary system establishes the true reactor power. Knowing the true reactor power, the liquid carry-over can then be computed by making a heat balance on the primary reactor system. In the case of EBWR, the heat balance was made on the primary reboiler which has as its heat source the reactor steam:

$$X = \frac{W_{IE} h_{fg,IE}}{W_{PR} h_{fg,PR}},$$

where

W_{IE} = intermediate heat exchanger flow rate (lb/h)

$h_{fg,IE}$ = heat of vaporization, intermediate heat exchanger

W_{PR} = primary reboiler flow rate (lb/h)

$h_{fg,PR}$ = heat of vaporization, primary reboiler.

Liquid carry-over data were also obtained from the discrepancy between the measured feed water and steam discharge flow rates. When liquid carry-over occurs, more feed water is injected into the reactor than steam is withdrawn. This discrepancy is actually a measure of the liquid carry-over rate. The major uncertainty in this data is the steam flow rate. With large amounts of water in the steam, the measured steam rate is undoubtedly in error. The magnitude of the error is unknown. However, the carry-over data obtained in this manner check well with values obtained by the heat balance method.

The third method consisted of taking steam samples from sampling probes located 6 in, $14\frac{1}{4}$ in and $22\frac{1}{2}$ in below the steam discharge duct. The samples were analysed for the sodium-24 content. The results were compared with a sample of reactor water obtained from a sampling point in the downcomer. These data were used to establish carry-over gradients as a function of steam dome height. Extrapolation to the height of the steam duct yields the actual liquid carry-over.

3.1.2. Discussion of results

A broad generalization of the results is as follows:

The reactor performance characteristics are governed directly by steam carry-under in the downcomer, liquid carry-over in the effluent steam and, indirectly, by the location of the true water-steam interface in the vessel.

The effect of steam carry-under and liquid carry-over on reactor performance is reflected in the core inlet sub-cooling, and hence, mean core moderator density. As the magnitude of the steam carry-under increases, the sub-cooling and moderator density decreases, since the mean steam volume fraction increases. The effects are reversed in the case of liquid carry-over. In order to maintain a constant liquid inventory, an excess of cold feed water is injected into the reactor. This increases the reactor sub-cooling and, hence, the reactivity. The excess water is expelled in the effluent steam. The carry-over phenomenon radically alters the performance characteristics of the reactor. Under these conditions, it is not a normal direct-cycle boiling-water reactor; in a sense, it becomes a natural circulation, dual-cycle reactor. The steam carry-under and liquid carry-over is very sensitive to the true two-phase mixture interface level within the reactor vessel. As the interface is lowered, the carry-under rises rapidly and the carry-over decreases. As the interface level is raised, the effects are opposite: carry-under decreases and carry-over increases.

(a) Total reactor recirculation flow rates. Fig. 7 is a plot of the recirculation flow rate as a function of power. The magnitude of the flow rate in the lower power range is in agreement with earlier core analyses and extrapolation from EBWR data. Analyses indicated a two-fold increase in flow rate could be expected with the additional riser height. However, the rate of increase of the recirculation velocity with power was not as large as predicted. The lower rate of increase of flow is attributed to loss of sub-cooling and, hence, increased frictional resistance in the core resulting from steam carry-under in the downcomer. At power levels above 40 MW, the flow rates tend to level off and are affected slightly by the true water-

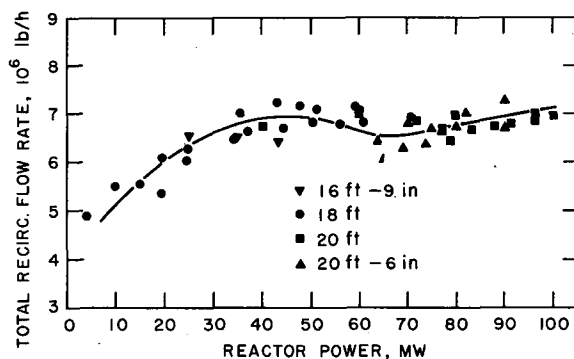


Fig. 7

Total recirculation flow rate as a function of reactor power at 600 lb/in² gauge and various true mixture interface levels

steam interface level. This is evidenced by the data scattering on the figure. This behaviour is a result of the indirect effect of the interface height on the reactor sub-cooling through its effect on the liquid carry-over and vapour carry-under.

These trends are more clearly shown on Fig. 8. The core inlet velocity data were obtained from turbine type (Potter) flow meters installed in an instrument assembly near the centre of the core [5]. Note the branching of the velocities as a function of the true interface height. The measured recirculation flow rates at power levels from 5 MW to 100 MW ranged from 4.7 to 7.25×10^6 lb/h. These correspond to a velocity ranging from 1.2 to 1.9 ft/s in the upper downcomer.

When the make-up water was injected in the lower feed-water ring, the reactor appeared to have reached an equilibrium state. Over the power range of 5 to 20 MW, the equilibrium recirculation flow rate was $\sim 4.7 \times 10^6$ lb/h. The equivalent velocity in the upper downcomer region

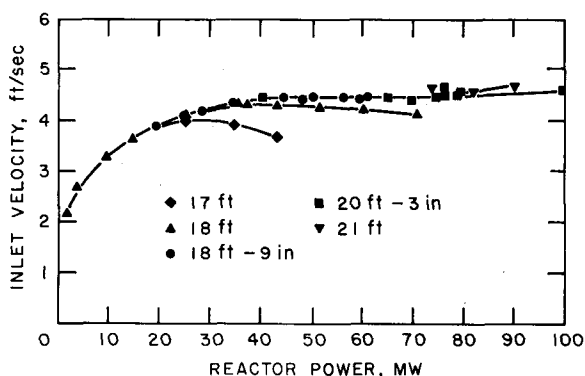


Fig. 8

Core inlet velocity as a function of reactor power and various true mixture interface levels

is 1.2 ft/s. Under this mode of operation, the steam voids carried under are not collapsed until they reach the lower feed-water ring located below the riser. Therefore, the reactor stabilizes at a total flow just above the critical value for the initiation of steam carry-under. When the critical value is exceeded and steam carry-under commences, small amounts of steam produce large steam volume fractions, since the relative velocity of the two phases is essentially zero. This, in turn, immediately tends to reduce the net driving head by decreasing the density in the downcomer and equilibrium conditions are rapidly achieved. The measured equilibrium downcomer velocity of 1.2 ft/s agrees very well with laboratory loop data which shows the threshold velocity for the initiation of steam carry-under at 600 lb/in² to be 1.3 ft/s.

(b) Sub-cooling. Sub-cooling is, perhaps, the most important measurable reactor parameter since it reflects the effects of carry-over and carry-under, and the variation of other parameters such as recirculation flow, interface height, etc. The effects of the vapour carry-under, liquid carry-over and true interface height are clearly evident in the sub-cooling "map"

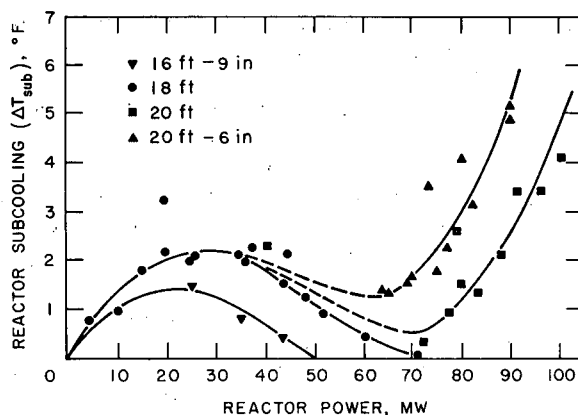


Fig. 9

"Map" of reactor sub-cooling as a function of reactor power and various true mixture interface levels of EBWR (Fig. 9). The inception of liquid carry-over can virtually be pinpointed. The seriousness and magnitude of the vapour carry-under is readily apparent. The magnitude of the liquid carry-over can also be easily deduced.

As evidenced by Fig. 9, reactor sub-cooling is extremely sensitive to the true mixture interface height. The family of curves obtained as a function of interface height merge at the low reactor powers. At an average interface height of 16 ft 9 in, the sub-cooling increases slightly with power, peaks at about 20 MW, and then diminishes rapidly toward zero at about 50 MW. This pattern of sub-cooling is a result of the steam carry-under in the downcomer. For normal reactor operation, if no steam carry-under is present, the sub-cooling increases continuously with power. At zero sub-cooling, the carry-under is 34.6 wt. %.

As the interface height is increased, the sub-cooling increases slightly and the peak occurs at a higher power level. This behaviour is due to the fact that the carry-under is a function of the interface height. For the same power and flow, the carry-under decreases as the interface height increases, therefore the sub-cooling is greater. The sub-cooling becomes zero at about 70 MW.

At an interface height of 20 ft, the trend with power is the same in the lower range. However, at a power level of approximately 70 MW, the sub-cooling increases sharply. The height of the steam dome has decreased to a point where, coupled with a superficial vapour velocity of 1.25 ft/s in the steam dome, liquid carry-over occurs. As the power is increased beyond this point, the vapour velocity also increases and the quantity of liquid carried over rises rapidly. The identical trend is seen in the data taken at an interface height of 20 ft 6 in. The difference is that the sub-cooling is again slightly higher and the point of carry-over is reached at a lower power level. This is due to the fact that the steam dome height is smaller.

(c) Steam carry-under. The steam carry-under in the reactor was calculated from a heat balance utilizing the measured sub-coolings and flow rates, as discussed previously. The steam carry-under as a function of power and interface height is plotted in Fig. 10. For these runs, the make-

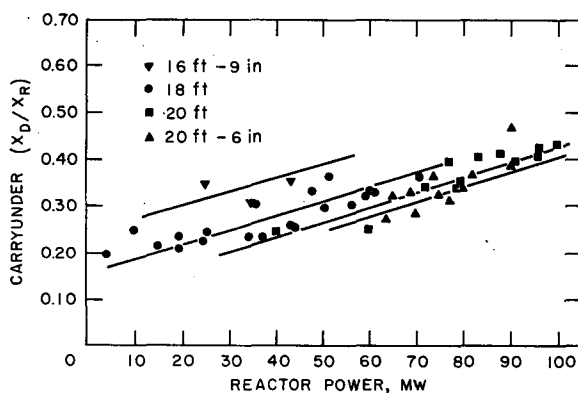


Fig. 10

Steam carry-under as a function of reactor power and true mixture interface levels

up water was injected in the upper feed-water ring. The quantity X_D/X_R on the ordinate is equivalent to the fraction weight per cent of carry-under. The carry-under is substantial, ~20%, at the lower powers. This is due to the fact that the recirculation flow rates are of such magnitude that the threshold velocity for initiation of carry-under is exceeded even at the very low powers. As mentioned previously, the data indicated that the threshold water velocity for the initiation of carry-under is 1.2 ft/s. As the power is increased, the carry-under increases gradually to a value of ~40% at 100 MW for an intermediate mixture interface height. That is, 40 wt.% of all steam generated in the core is being entrained in the downcomer. The average water velocity in the downcomer at 100 MW was 1.9 ft/s.

It is interesting to note that the carry-under has exceeded the theoretical value of 34.6% (which corresponds to the point where all sub-cooling is eliminated and recirculation of steam commences). This is due to the occurrence of liquid carry-over.

The magnitude of vapour carry-under in the downcomer is somewhat startling, but corroborating evidence is gained from an analysis of the physics data as described in a later section.

Laboratory loop studies [6] showed that as power is increased for a fixed recirculation velocity, the percentage of carry-under decreases, reaches a minimum and then begins to increase. This sequence did not occur in the reactor, since the recirculation velocity continued to increase slightly with power. Once the threshold velocity for carry-under is passed, substantial increases in carry-under occur with small increases in velocity. In EBWR, the expected decrease of carry-under with increasing power apparently was nullified by the increase in the recirculation velocity. The over-all effect was that the percentage of carry-under continued to increase with power.

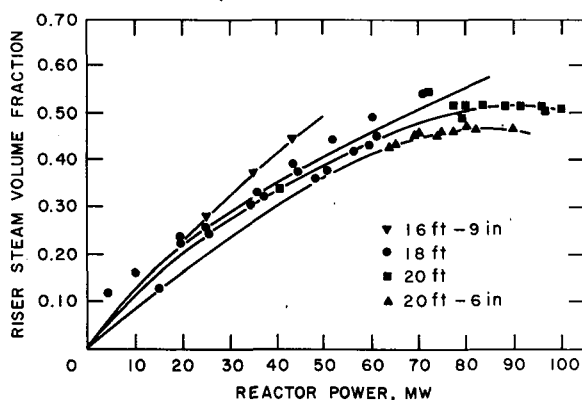


Fig. 11

Steam volume fraction at periphery of riser as a function of reactor power at 600 lb/in² gauge and various true mixture interface levels

(d) Steam volume fractions in riser and downcomer. The steam volume fractions in the riser and the downcomer are shown in Figs. 11 to 13. The variation of the riser steam volume fraction with power can also be scrutinized for the effects of liquid carry-over and steam carry-under. In the lower power region (10-40 MW) where the steam carry-under is predominant and the loss of the sub-cooling is rapid, the riser steam volume fraction rises rapidly. In the middle power range (60-80 MW) where the threshold of liquid carry-over is reached, the void fraction tends to level off. In the very high power range (> 80 MW), the steam volume fraction tends to decrease. This behaviour in the high-power region is expected, since the sub-cooling is increased rapidly with power due to carry-over.

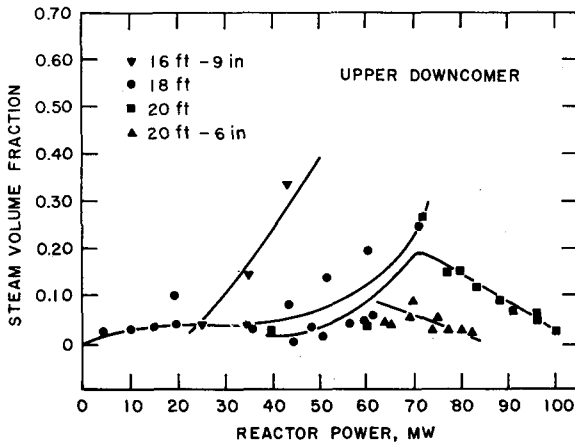


Fig. 12

Steam volume fraction in upper region of downcomer as a function of reactor power and various true mixture interface levels

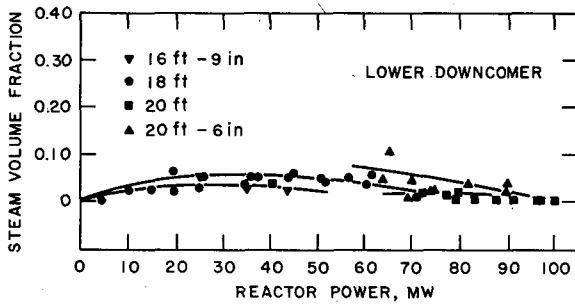


Fig. 13

Steam volume fraction in lower region of downcomer as a function of reactor power and various true mixture interface levels

An indication of the change in the void content can be gained from a simple heat balance on the core:

$$Q = W_T (\Delta h + X h_{fg})$$

or

$$X = \frac{(Q/W_T) - \Delta h}{h_{fg}}$$

where X is the average mixture quality at the core exit. The average core

and riser void fractions are roughly proportional to the average exit core mixture quality X . Therefore, the change in quality and, hence, change in voids was computed over the power intervals 70-100 MW.

The results of this analysis indicate that the rate of void fraction increase decreases as the power is increased. Between 80 and 90 MW, there is virtually no change in exit quality, hence, the core and riser void fractions tend to remain nearly constant. Between 90 and 95 MW, it appears that the void content actually decreases, hence, the reactivity increases. This behaviour is shown in Fig. 11 and was corroborated by the control rod movements made over this power interval, as is discussed in the Physics Section.

Data on the rate of quenching of the steam carried under were obtained in the lower downcomer, and the upper downcomer where the riser was necked down (see Figs. 12 and 13).

An insight into the collapse-rate pattern of the steam bubbles can be gained from a rough theoretical analysis of the problem. Assuming that the collapse of a vapour bubble in a sub-cooled liquid is governed primarily by the rate of heat transfer at its surface during the major fraction of its lifetime and that the bubble is moving in a quiescent fluid, a theoretical collapse rate can be computed. Undoubtedly, in a highly turbulent region the collapse rate of a bubble would be much higher than predicted. However, the qualitative effect of sub-cooling on the collapse rate is considered fairly accurate.

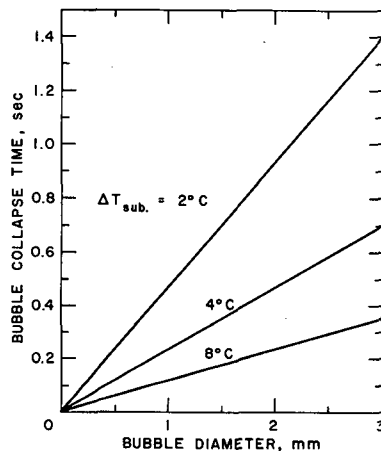


Fig. 14

Effect of sub-cooling on bubble collapse time

Fig. 14 shows the results of some rough calculations which illustrate the strong effect of sub-cooling on the collapse rate of bubbles of various diameters. As the sub-cooling increases, the time required to collapse the vapour bubble decreases. Thus at the top of the downcomer where the cold feed water is injected, the sub-cooling is maximum, the collapse rate is likewise at a maximum and one would expect a sudden large drop in the void content.

In an EBWR-type reactor where cold feed water is injected into the downcomer, the sub-cooling at core entry would vary from one degree at ~ 10 MW to $\sim 12^\circ$ at 100 MW, assuming no carry-under. At low powers, the collapse rate is at a minimum, thus a large quantity of steam carry-under would reflect a high steam volume fraction. As power is increased, the theoretical sub-cooling increases and thus the bubble collapse rate increases. Hence, for the same amount of steam carry-under, the steam volume fraction would be lower. In the very high power range (> 75 MW) where excess cold water is injected due to the liquid carry-over, the collapse rate is even greater and, despite maximum carry-under, a low steam volume fraction would be expected. Moreover, because of liquid carry-over, the steam volume fraction would decrease with power. Fig. 15 illustrates the type of void pattern that was expected in the upper region of the downcomer. In the lower downcomer region virtually all the steam voids would be condensed because of the relatively long transient time involved. This is especially true in the very high power range where an excess of feed water is injected.

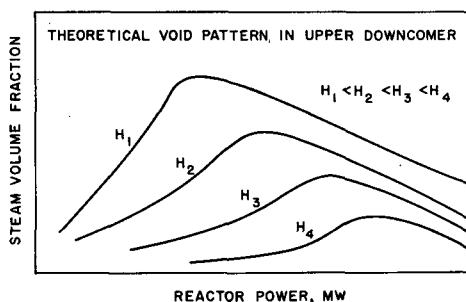


Fig. 15

Steam volume fraction as a function of reactor power
(Showing void pattern expected in upper region of downcomer)

The steam volume fraction data obtained from the upper and lower downcomer are shown in Figs. 12 and 13, respectively. At first glance, the data appear to be erratic with excessive scatter. However, upon closer scrutiny trends similar to those just described can be deduced which apparently substantiate the foregoing analysis. The major deterrent to establishing conclusive patterns is that the data were not taken over the entire power range at a fixed interface height. However, as shown in Fig. 12 at the lowest powers and lowest interface heights, the steam volume fraction in the upper section of the downcomer is maximum. As the interface height is raised and the power is increased, the steam volume fraction decreases in a manner similar to that predicted. In the lower region of the downcomer, the steam volume fraction is virtually zero under all conditions. (See Fig. 13)

(e) True liquid level. The difference between the level as indicated by the water column and the actual vapour-liquid interface level in the reactor is shown in Fig. 16. The height differential is a result of the steam voids present in the core, riser and downcomer and the "bubble bed" above

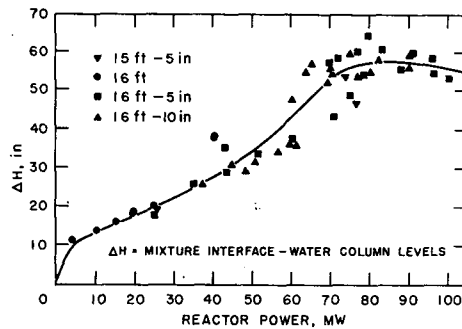


Fig. 16

Height differential between actual interface level and water column level as a function of reactor power and various true mixture interface levels

the riser. The voids that are formed in the reactor core, flow through the riser and are entrained in the downcomer, displace an equal volume of water. This causes an increase in the mixture height. The water level above the riser is, in turn, further expanded by the vapour flowing through it, which creates a two-phase mixture or "bed". As can be seen in Fig. 16, the discrepancy between the two levels can be quite large. The height differential is primarily a function of the reactor power and increases as the power is increased. In the very high power range where liquid carry-over occurs, the height differential levels off and begins to decrease. This is a result of the sudden increase in sub-cooling and, hence, decrease in core and riser void content. The scatter of the data increases with power. This is probably due to the fact that the interface becomes more turbulent and less defined. The maximum error in establishing the average interface level was estimated to be 9 in.

(f) Liquid carry-over. The three methods employed to determine the liquid carry-over were described earlier. The results are compared in Fig. 17. They represent the first set of carry-over measurements taken. Accordingly, these data are preliminary since the true mixture interface levels were not determined at this time. The data do, however, serve to point up the agreement among the three techniques.

Only limited data were obtained from the sampling probes because of time considerations and termination of the project. However, some interesting trends are illustrated in Fig. 18 where the percentage of liquid carry-over is plotted as a function of the distance above the true interface. A family of curves is obtained for varying superficial vapour velocities; hence, it is readily apparent that carry-over rates cannot be expressed in terms of superficial vapour velocities alone. The height of the steam dome must also be specified. As the actual steam dome height decreases the carry-over increases rapidly.

The liquid carry-over data derived by the heat balance technique are the most accurate and probably of greatest interest since they represent the actual amount of water carried over into the turbine by-pass line and the

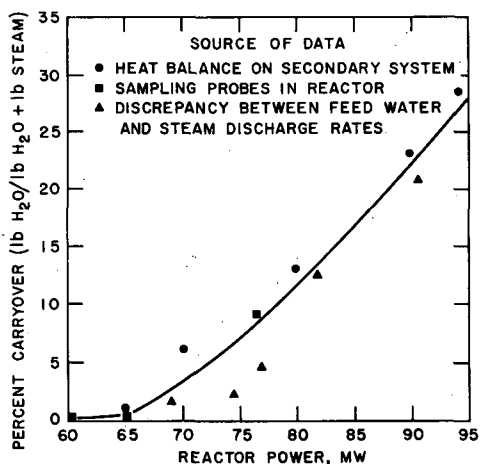


Fig. 17

Percentage of liquid carry-over as a function of reactor power
(Showing comparison of data obtained from three sources)

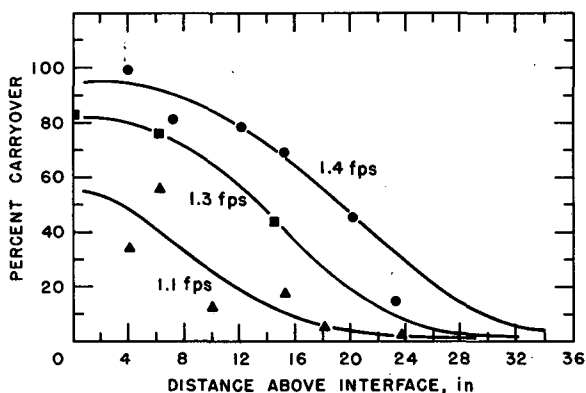


Fig. 18

Percentage of liquid carry-over as a function of distance above vapour interface for various superficial vapour velocities

primary reboilers. As illustrated in Fig. 19, the initiation of carry-over occurs at different power levels for different interface heights. At 60 MW, the carry-over starts at a superficial velocity of 1.1 ft/s and with a steam dome height of 30 in. As the steam dome height is increased, the superficial vapour velocity required to initiate large amounts of carry-over also increases.

A comparison of the data derived from the reactor, with loop data obtained under static conditions (no fluid recirculation), shows that the magni-

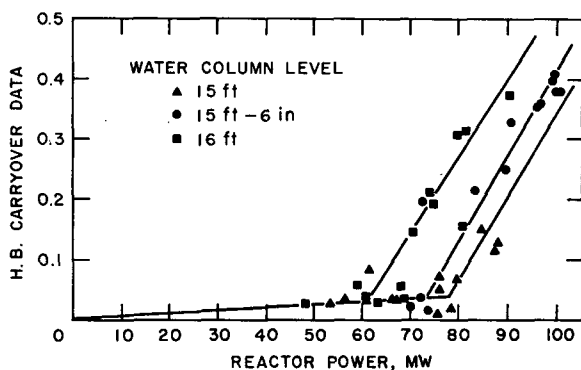


Fig. 19

Liquid carry-over as a function of reactor power and various water column levels
(Data from heat balance on primary reboiler)

tude of the carry-over is larger for the reactor under the same conditions of vapour velocity and dome height. The amount of liquid carry-over measured in the reactor was much larger than had been expected.

3.2. Physics

The results of the detailed physics studies that were made on EBWR are given in Refs. [7 and 10]. However, some pertinent physics data and parameters are presented below which corroborate the effect of the vapour-liquid separation process on reactor performance.

3.2.1. Functions of spike fuel assemblies

The merits of and the reasons for the selection of the 100-MW(t) core loading shown in Fig. 3 are described in detail in Ref. [10]. One of the prime considerations which govern the arrangement of the spike fuel assemblies was to achieve as flat a power distribution as was possible. Other considerations included the use of boron stainless-steel strips to provide excess reactivity through the use of a burnable poison. However, owing to termination of the water reactor programme, this latter objective was no longer of importance. Consequently, the primary functions of the strips attached to the spike fuel assemblies were: (1) to control the reactivity of the core; and (2) to provide the desired flexibility in effecting increases in reactivity as required.

In planning the power operation of the reactor, the number of boron strips per spike was initially reduced to an average of 1.57. With this configuration ~5% reactivity was available for steam void formation within the core.

3.2.2. Core parameters

Before making power runs, the control-rod system (with Zircaloy-2 followers) was calibrated. The results are plotted in Fig. 20. In addition,

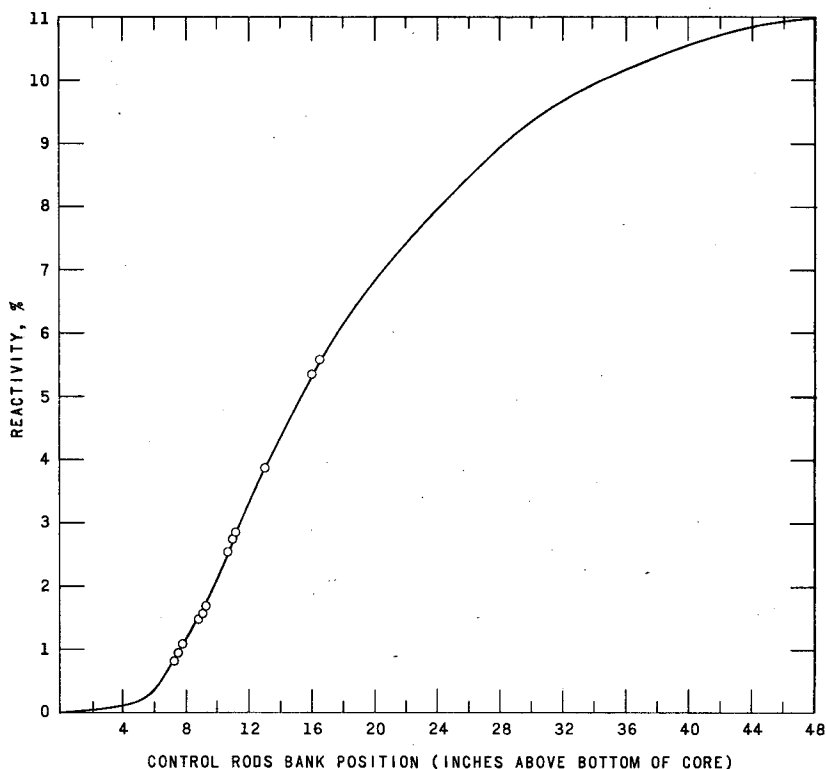


Fig. 20

Bank control-rod calibration

some void coefficient measurements were made with the aid of four void simulators installed in the fuel assemblies. Owing to dimensional limitations, the void simulators were confined to the thin enriched plate assemblies. The measurements made with H_3BO_3 in the reactor were of particular interest. Extrapolation of these data indicated a negative void coefficient for concentrations as high as 10 g H_3BO_3 /gal H_2O in the core (2640 ppm).

(a) Reactivity distribution. A series of runs were then made over a power range from 0 to 60 MW. Control of reactivity was achieved primarily by control-rod movements for the operating range of 0-40 MW; at 40 MW and with equilibrium xenon, the rods were all out. Above 40 MW, control was achieved by reducing the H_3BO_3 concentration in the core water. At the 60-MW power level, $\sim 0.5\%$ ρ was available to effect further increase in reactor power.

In order to obtain powers greater than 65 MW, it was necessary to remove additional strips from the spike elements. The core reactivity was increased $\sim 2\%$ by reducing the number of boron strips to one strip per spike in the central ring of 28 spikes. This gave a total of $\sim 10.5\%$ core reactivity after deducting 0.6% for burn-up during these power runs. This allowed

$\sim 1.5\% \rho$ to increase the power above 65 MW. The maximum reactivity that could be added was governed by the established shut-down criteria.

A summary of reactivity distribution in the core is given below:

Reactivity required from cold to hot, zero voids	$\sim 1\% \rho$
Reactivity tied in steam voids for 60-MW operation	$\sim 5\% \rho$
Additional reactivity for operation beyond 60 MW	$\sim 1.5\% \rho$
Xenon poisoning at 80 MW	$\sim 2.8\% \rho$
Total reactivity	$\sim 10.5\% \rho$
Control-rod worth (cold reactor)	$11\% \rho$
Reactivity worth of H_3BO_3 for 1 g of $\text{H}_3\text{BO}_3/\text{gal H}_2\text{O}$ (264 ppm)	$\sim 1.0\% \rho$

(b) Power coefficient. The effect of steam carry-under and liquid carry-over can readily be deduced from power coefficient measurements made over the power range of 0 to 80 MW for a fixed water column level. Power coefficients were obtained by determining the change in reactivity effected by control-rod movements and by removal of boric acid. These data are shown in Fig. 21. In the lower power range, the power coefficient increases with

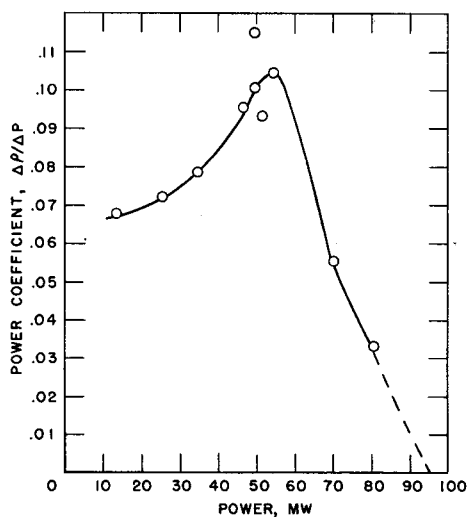


Fig. 21

Power coefficient against reactor power

increasing reactor power. It peaks at about 55 MW, begins to decrease and tends toward zero. The peak at about 55 MW corresponds roughly to the power level where liquid carry-over commences and reactor sub-cooling begins to increase. Below 60 MW, the carry-under phenomenon is predominant; this is reflected in the increasing power coefficient. Above 60 MW,

liquid carry-over becomes the dominating factor and the power coefficient decreases. An extrapolation yields a zero value at ~ 95 MW. Moreover, the extrapolation indicates that above ~ 95 MW, the power would continue to increase without introducing additional reactivity.

Such a sequence of events actually occurred in the EBWR, as evidenced by the data in Table III. With the rod bank at 34.4 in and the centre rod at 40 in, the reactor was at a power level of 67.7 MW. The rod bank was raised to 38 in and the power increased to 84.9 MW. However, the reactor did not reach an equilibrium power level, but increased over a 10-min interval to a level of 94.4 MW. At this point the whole rod bank was reinserted to 36.7 in. After a series of rod manipulations an equilibrium power of 90.7 MW was achieved with the rod bank at 37.7 in. This rod bank position at 90.7 MW is lower than the corresponding position at 84.9 MW. This unique behaviour,

TABLE III

CONTROL-ROD POSITIONS VERSUS POWER
(Boric acid content = 1 g/gal)

Time	Rod positions		Power (MW(t))
	1-8 (in)	Centre rod (in)	
1245	34.4	40	67.7
1338	38	40	84.9
1348	38	40	94.4
1404	36.7	36.7	82.0
1430	36.7	36.7	80.6
1510	37.7	37.7	90.7

attributed primarily to liquid carry-over, was undoubtedly aided, to some extent, by the expulsion of boric acid from the reactor through the mechanism of carry-over.

The same reactor behaviour pattern can be seen in the reactivity versus power level plot shown in Fig. 22. Below 60 MW, a single-value functional relationship exists between the two parameters. The branching of the curve at 60 MW is a result of the carry-over mechanism. It is apparent that the maximum power that could be attained (all rods out and boric acid concentration < 40 ppm) was governed directly by the liquid level. For a water column level of 14 ft 10 in the maximum power level was 72 MW. When the water level was raised to 16 ft, the design power level of 100 MW(t) was readily achieved with $\sim 0.65\%$ ρ remaining.

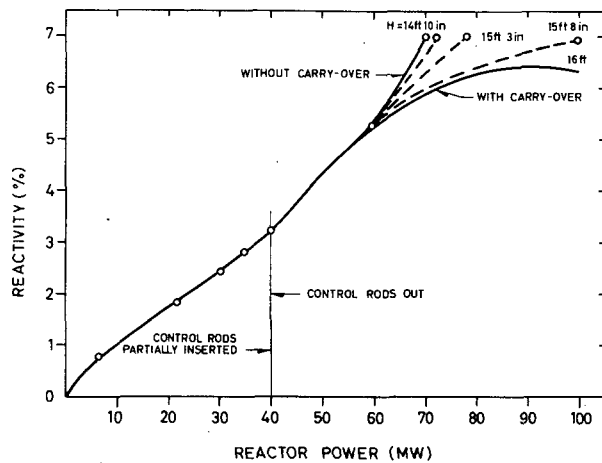


Fig. 22

Reactivity against reactor power

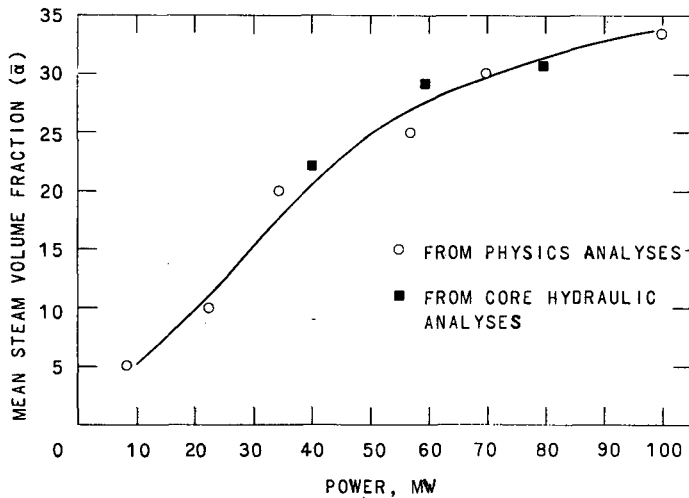


Fig. 23

Mean core steam volume fraction against reactor power

(c) Void coefficient of reactivity. The void coefficient ($\Delta k/\Delta V$) of the core was computed to be 0.195 for a change of 0-10% in the average void content of the core. This value was obtained from rough preliminary calculations. Detailed calculations, which will take into consideration the increased void content, are in progress. Based on this preliminary analysis, an estimate of the average void content in the core was made as a function of power and is shown in Fig. 23. Included are the average void fractions computed from thermal hydrodynamic analyses using the measured reactor

sub-coolings. As can be seen, the average core coolant void fraction derived from physics and thermal hydraulic considerations are in fair agreement.

(d) Neutron flux distributions. A method was developed for, and used in, the EBWR for characterization of the neutron flux as a function of location in the core, reactor power and reactor control parameters. Bare and cadmium-covered cobalt foils were introduced and accurately positioned for irradiation in selected locations in the core. After irradiation, they were removed without affecting reactor operation.

The activation detector foils were prepared as small spheres ground to precision tolerances. This shape and close dimensional control eliminated individual weighings and obliquity effects in the isotropic flux, and greatly simplified automatic analysis. The activated foils were analysed with conventional instrumentation and the thermal and resonance neutron flux components were calculated. Figs. 24 and 25 show typical thermal neutron flux distribution isogram plots constructed from several axial plots to show both axial and radial distribution in one quadrant of the core. The left margin of each plot represents the centre-line of the core.

The power level for both isogram plots was 40 MW. The boric acid concentrations were 25 ppm and 1200 ppm for Figs. 24 and 25, respectively. It will be noted that the peak power in the presence of soluble poison control is considerably lower (highest isogram is $2.6 \times 10^{13} \text{ n cm}^{-2} \text{ s}^{-1}$) than the peak power in the absence of a significant concentration of soluble poison (highest isogram is $3.75 \times 10^{13} \text{ n cm}^{-2} \text{ s}^{-1}$). The effective core size is much larger and better distributed in the presence of soluble-poison shim control and should result in a more uniform burn-up of the fuel throughout the core.

Isogram plots were also prepared to show the effect of reactor power level on thermal and resonance neutron flux distribution, and to show the effect of operation with the control rods in an unusual non-banked configuration. (See Ref. [8]).

3.3. Specific observations

3.3.1. Fuel assemblies

The original EBWR core contained 114 fuel assemblies surrounded by 32 aluminium dummy fuel assemblies to fill the grid plate. This core remained essentially undisturbed for about three years in the reactor vessel. During this time, deposits varying in thickness (up to 0.008 in) had built up on the fuel plate surfaces. Attempts to remove the deposit proved not practical. Analysis of sample scrapings revealed a composition of 67 wt. % boehmite ($\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$), 25 wt. % nickel and 8 wt. % iron. Consequently, the aluminium dummy fuel assemblies were judged to be the principal source of the scale. The exact mechanism of deposition has not been resolved. However, it is fairly well established that the mass transfer is from the dummy fuel boxes, to the coolant and then to the active fuel plates, primarily on the boiling areas of the plate surfaces. All aluminium dummy fuel assemblies were removed from the core in order to provide space for the additional spike fuel assemblies of the high-power runs.

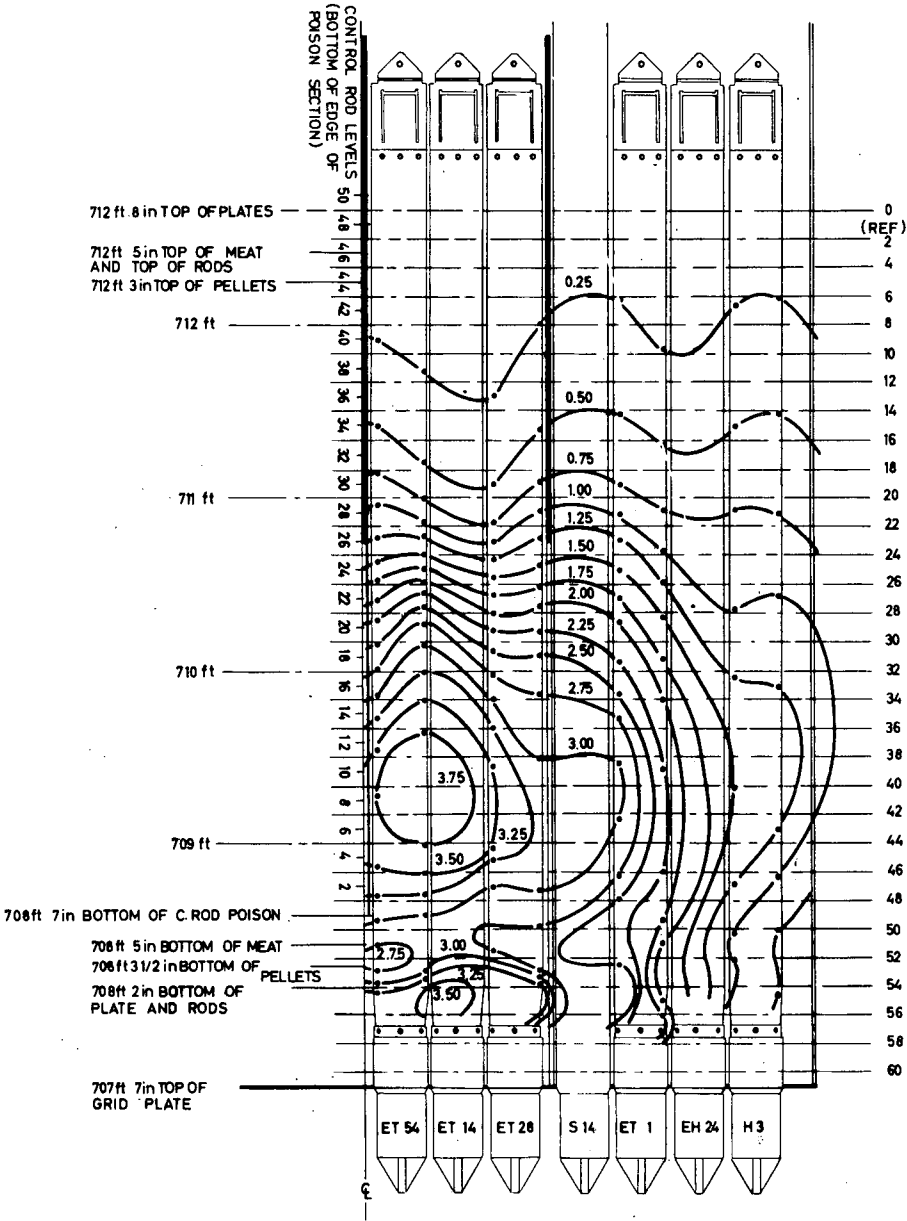


Fig. 24

Thermal neutron flux isogram plot for reactor power of 40 MW(t) and soluble poison concentration of 25 ppm boric acid

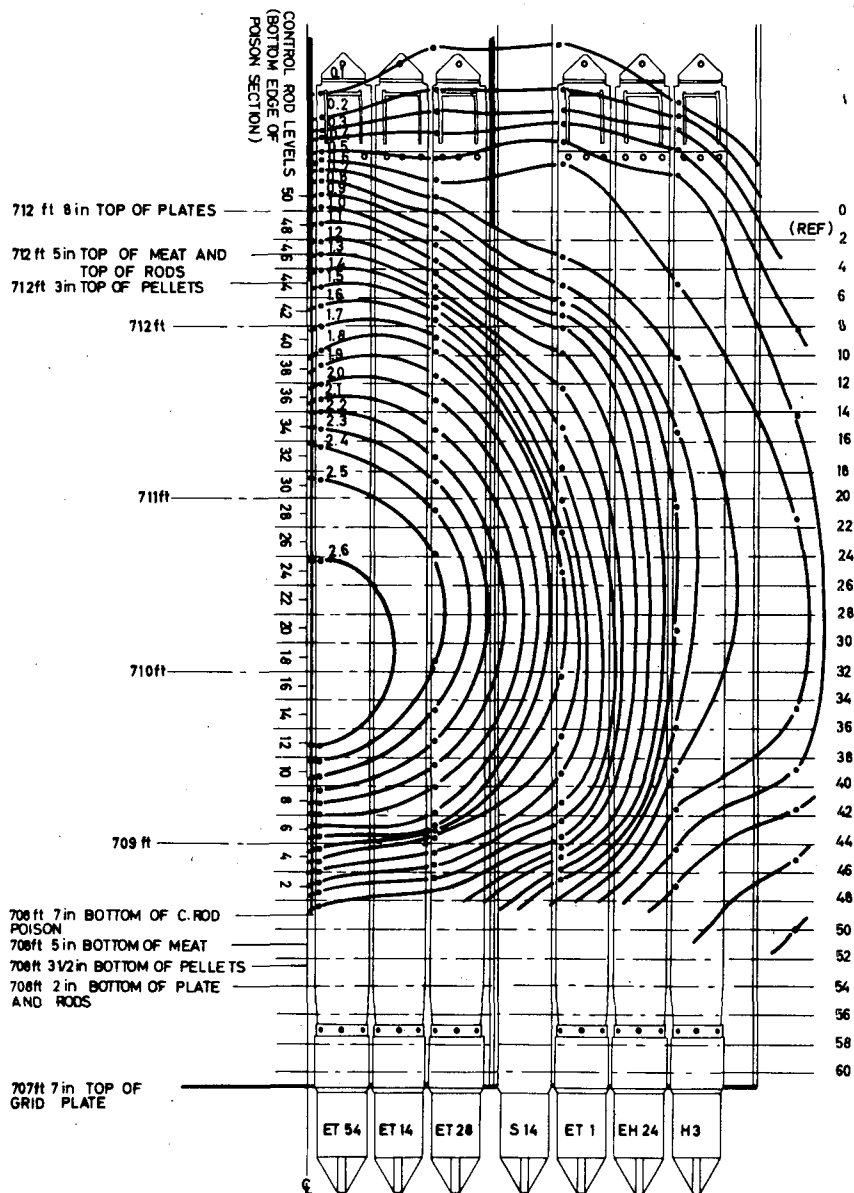


Fig. 25

Thermal neutron flux isogram plot for reactor power of 40 MW(t) and soluble poison concentration of 1200 ppm boric acid

This scale was a matter of deep concern during operation at elevated powers. The existing scale on the fuel plates represented a barrier to effective heat transfer. The thermal conductivity of the scale had been estimated to be less than 1 (BTU ft⁻²h⁻¹)(°F/ft). At elevated powers, the scale could promote a central fuel temperature of ~1125°F. At this temperature, the creep strength of the uranium alloy would be reduced and the fission gas trapped in the fuel plates could cause considerable swelling. It was feared that some fuel plates would eventually rupture.

Initial evidence of fuel growth was observed upon completion of the 85 MW(t) power level phase of the programme. cursory examination of the plate assemblies (under the core water) revealed the side plates were slightly distorted, with some loss of the scale. There were no detectable physical defects in the fuel plates.

After shut-down from 100 MW, some plate and spike fuel assemblies were removed for visual examination. There were no perceptible changes in the spike assemblies; however, definite changes were observed in the plate-type assemblies. The perforated side plates had buckled about 1/16 in between spot welds (in the horizontal position and not between fuel plates) Maximum deformation occurred in the region of highest flux and diminished toward the ends of the fuel plates.

The following comments are subject to confirmation by current destructive and non-destructive tests on these assemblies. The buckling may be attributed to the difference in expansion coefficients of the Zircaloy-clad metallic fuel and the Zircaloy side plates. However, the fuel plates did not buckle because the clad was bonded to the fuel. During the high-power runs, the side plates may have exceeded their elastic limit and, as a consequence, buckled as the metallic fuel plate contracted after reactor shut-down. With respect to the loss of some scale: evidently at some power above 60 MW(t) the scale was subjected to a shearing stress induced by expansion of the fuel plates. Undoubtedly, the descaling process improved heat transfer between fuel plates and the primary coolant. At this time, it can only be speculated that a redistributed deposition of the spallings would have occurred with continued operation.

3.3.2. Radiation levels

During reactor operation and/or subsequent shut-down, authorized personnel were permitted to enter the containment shell and the reboiler building for purposes of performing experiments or maintenance on system components. The following sections describe briefly the radiation levels that were encountered, the contributing factors and the corrective measures that were taken.

(a) Transport of corrosion products. Scale spalling from the fuel plates presented some radiation problems, particularly in the sub-reactor room. The settling and accumulation of this scale in control-rod rack housings and flanged appendages for the original forced circulation nozzles created radiation levels up to 20 r/h at contact. Blow-down of these nozzles (to a dump tank) effected a reduction in surface radiation to 3-5 r/h.

At power levels above 65 MW, where water carry-over with the steam was encountered, fine particles of scale were entrained with the water. These particles did not plate out on any surfaces external to the reactor vessel, but were eventually collected in the full-flow condensate filters. Radiation levels external to the filter vessels normally are 10-15 mr/h, but with carry-over the levels increased to 20 r/h. Irradiated particles of scale were also collected in the reactor water purification filters and resin beds. These vessels are shielded with lead (4-in thick); therefore, no significant radiation levels were encountered.

A fine coating of iron oxide (rust coloured) was deposited on all the surfaces that were contacted by the primary fluid. It is believed that the bulk of the iron came from the carbon-steel condenser shell and some of the carbon-steel piping.

(b) Plant activities. Radiation levels in the containment shell and in the reboiler building can be attributed primarily to N^{16} (half-life = 7.4 s) except for liquid carry-over of irradiated corrosion products. The amount of liquid carry-over is a function of the reactor power and is especially sensitive to interface level in the reactor vessel. With no carry-over, radiation levels in the containment shell were essentially the same as was reported for 20-MW operation [1]. This was true whether the reactor was operated at 20 MW or at 80 MW. The original 20-MW plant and the reboiler plant were operated in parallel; steam in excess of 20 MW was shunted to the reboilers. Since N^{16} is the primary contributor of activity and decays through several half-lives before it re-enters the containment shell, it retains only very minute amounts of activity. Table IV lists typical radiation levels registered in the containment shell.

One unexpected source of radiation was detected in the sub-reactor room. An intense neutron beam was emitted from each of the two, large, forced-circulation piping nozzles that extend through the shielding beneath the reactor. Neutron streaming which occurred in the annuli between the 12-in pipes and their respective 20-in sleeves was attributed to enlargement of the initial 4-ft-diam. core to a 5-ft-diam. core. As a consequence: (1) the core was in closer proximity to the nozzles; and (2) the shielding afforded by the water in the core downcomer was diminished. Further neutron streaming was precluded by filling the annuli with water to a height that extended ~ 1 ft into the biological shield.

Fig. 26 shows the radiation levels registered at various locations in the reboiler building. The primary system components are enclosed by a concrete wall (1-ft thick). The maximum radiation level at the outer surface was 15 mr/h.

Upon termination of the experimental programme, the turbine was dismantled for routine inspection. The maximum radiation level on any surface was 1 mr/h at contact. The inside surfaces of the turbine had been plated with 0.005 in of nickel by the Kanigen process. Most of this plating had eroded, especially in areas where steam velocity was highest. Otherwise the turbine appeared to be in good condition.

(c) Fission gas release. As a point of interest, fission gas release due to fuel-element failure has not been experienced throughout the history

TABLE IV
RADIATION LEVELS IN CONTAINMENT SHELL (mr/h)

Monitor location	Power distribution		
	22 MW(t) to turbine 55 MW(t) to reboilers (31/10/62)	15 MW(t) through turbine by-pass 75 MW(t) to reboilers (1/11/62)	3.3 MW(t) through turbine by-pass 53 MW(t) to reboilers (1/11/62)
Turbine	46	10	6
Plant main floor area	0	0	0
Steam dryer	1100	1100	400
Air ejector	1700	2000	80
Condenser hotwell	110	280	50
Plant air exhaust duct	70	50	30
Air ejector gas filter	100	70	40

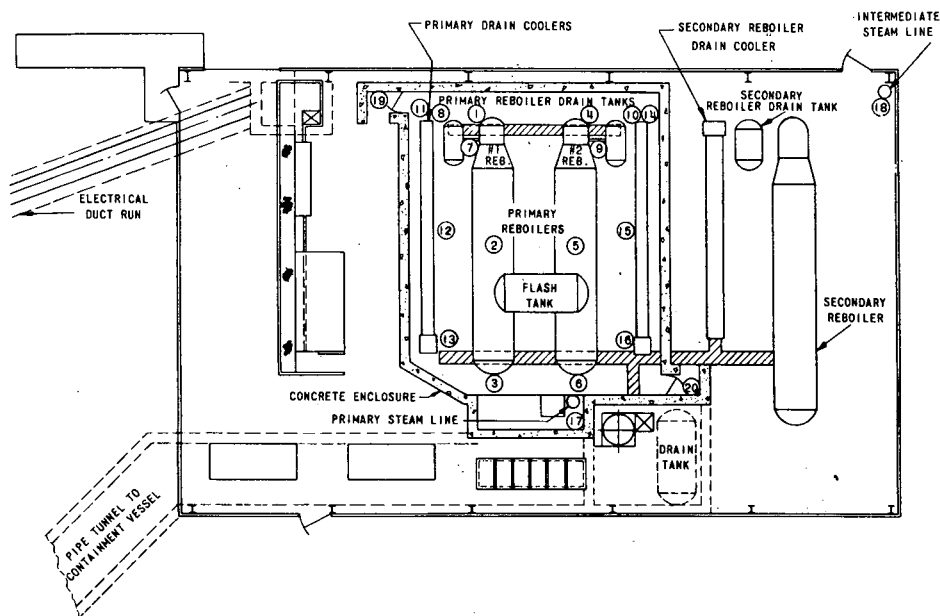


Fig. 26

Location of radiation measurement points in reboiler building

Legend

* r/h

1. No. 1 primary reboiler head end	1.1
2. No. 1 primary reboiler centre bottom	1.5
3. No. 1 primary reboiler back end	0.45
4. No. 2 primary reboiler head end	1.1
5. No. 2 primary reboiler centre bottom	3.1
6. No. 2 primary reboiler back end	0.45
7. No. 1 reboiler drain tank line	2.7
8. No. 1 reboiler drain tank bottom	1.0
9. No. 2 reboiler drain tank line	4.5
10. No. 2 reboiler drain tank bottom	1.6
11. No. 1 primary drain cooler back end	0.6
12. No. 1 primary drain cooler centre	0.3
13. No. 1 primary drain cooler head end	0.43
14. No. 2 primary drain cooler back end	0.6
15. No. 2 primary drain cooler centre	0.38
16. No. 2 primary drain cooler head end	0.5
17. Primary steam line	0.9
18. Intermediate steam line	0
19. West doorway to cell	0.2
20. East doorway to cell	0.25

*Maximum readings obtained when reactor was operating at 90 MW(t) with 70 MW(t) being sent to the reboilers.

Readings were 2 in from surfaces except in doorways where a general background level is given.

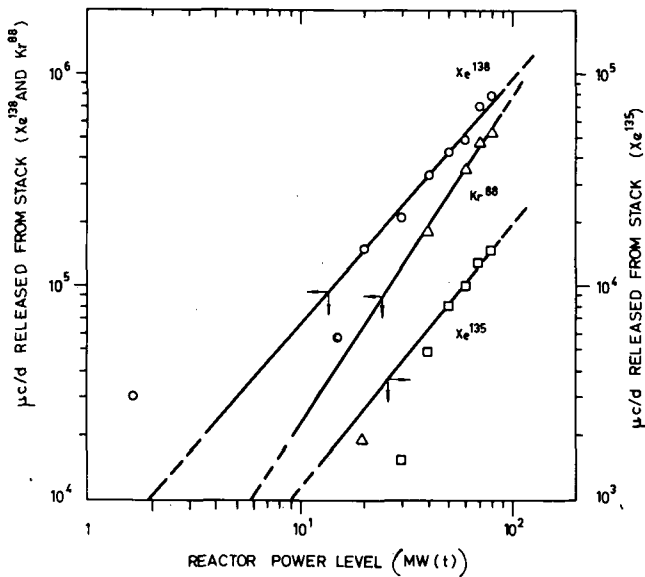


Fig. 27

EBWR fission-gas release against reactor power

of EBWR operation. The fission gases that were detected (Fig. 27) are attributed to fission of tramp uranium that adhered to the plate surfaces during fabrication. Minute amounts of Xe^{133} were also detected but are not shown in the figure.

Gas samples from the exhaust stack and, also, downstream of the air ejector were filtered and drawn into evacuated collectors.

Two methods of gas analysis were employed. Briefly, a chemical method was used for gases decaying into a radioactive particulate daughter. The activity of the daughter is determined and the activity of the gaseous parent can be calculated from equilibrium information. This method is satisfactory for Xe^{138} and Kr^{88} . The second method involved the use of a single-channel, pulse-height analyser. This method is satisfactory for determining the longer half-life products such as Xe^{135} and Xe^{133} . A detailed description of both methods is given in Ref. [3].

3.3.3. On the use of soluble poisons (boric acid)

From time to time, questions are raised about the use of boric acid for augmentive control in an operating reactor. The term "hideout" is used occasionally. Other questions pertain to improper or incomplete mixing, or to operational difficulties such as crystallization, reliability of monitoring the boric acid concentration etc.

(a) Hideout. The EBWR has had many months of operation using boric acid in the reactor with concentration in general not exceeding 5 g/gal. Hideout has not been experienced. If it does exist, the quantities must be very

minute and do not present a problem. Saturated steam leaving the reactor (assuming no water carry-over) carries a boric acid concentration of one part in seventeen of that present in the reactor water. This is due to volatilization and not carry-over. This means that the concentration in the condenser hotwell is approximately 6% of the concentration in the reactor vessel. This presents no problem since equilibrium is established within a few minutes during reactor operation. However, where large amounts of water are carried over with the steam, entrained boric acid is carried over with the water. For example, during a power run at 91.5 MW, the boric acid concentration in the hotwell was 186 ppm, whereas it was 422 ppm in the reactor vessel. Without carry-over and with a concentration of 422 ppm in the reactor, the normal concentration in the hotwell would be about 25 ppm.

(b) Dilution. During cold critical experiments, the boric-acid concentration in the reactor vessel was easily varied by dilution. This simply involved adding clean water, mixing and then draining the vessel to a pre-determined level.

Dilution of the poison by the addition of clean water is relatively slow. As far as reactivity hold-down is concerned, boric acid presents no real hazard with one possible exception. Mixture of boric-acid solution with the reactor water may be a problem when there is no natural convection. This will not prevail during operation or if sufficient core decay heat is available to induce natural circulation.

To obtain good mixing in the EBWR, air was bubbled through the reactor water for 15 min after the addition of boric acid (with the vessel pressure at atmospheric conditions). There was no indication of crystallization or settling out of the crystals once a homogeneous solution was obtained, even with concentrations up to 2640 ppm.

(c) Removal. During reactor operation, removal of the soluble poison was accomplished with the reactor water purification system in conjunction with an anion-resin-bed ion exchanger. Water was pumped from the reactor at about 8 gal/min, cooled and passed through the anion resin bed (9 ft³). Subsequently, the water was circulated through a standard mixed-bed ion exchanger (1.2 ft³ resin) to ensure neutral water, reheated and pumped back to the reactor. In this manner, the boric-acid concentration in the reactor was reduced by a factor of two in about five to six hours.

(d) Monitoring. Two primary monitoring methods were used to determine the boric acid concentration: (1) continuous monitoring of water flowing through a sample cell in the purification system; and (2) bi-hourly determinations in an apparatus designed for individual samples. Both methods are based on neutron absorption by the poison. A chemical titration method was used periodically as a check on the monitors.

The neutron absorption method is used on the B¹⁰ isotope which has a high cross-section for thermal neutron capture. The apparatus consists of a plutonium-beryllium source (10⁷ n/s) and a detector arranged in a cell of suitable geometry. Reactor water circulates continuously through the cell. The concentration of boron determines the thermal neutron flux attenuation between the source and the detector.

In EBWR, a signal from the neutron ionization detector chamber was measured with a vibrating-reed electrometer in a circuit incorporating feedback stabilization and control. A continuous recorder in the control room indicated the boric-acid concentration.

3.4. Reactor transfer function measurements

3.4.1. Methods

EBWR stability was determined by experimentally measuring the reactor transfer function at selected power levels and extrapolating the results of these measurements to obtain an upper limit for stable operation. Measurements were made at 10, 20, 40, 60 and 71 MW. The data indicated a maximum stable power of 120 MW.

(a) Sinusoidal reactivity input. The power transfer function was obtained by exciting the reactor with a sinusoidal reactivity input and measuring the amplitude and phase shift of the neutron flux output with respect to the control-rod input at selected frequencies. The inherent noise of the boiling process tended to reduce the accuracy of the measurements, and therefore, noise rejection techniques based on cross-correlation were used.

Cross-correlation of the control-rod position (input) and neutron flux (output) allowed the reactor transfer function to be calculated with the desired degree of accuracy. Two methods were employed to compute the cross-correlation. One method used an analogue computer to multiply the output signal by the sine ωt and cosine ωt , and to integrate the resulting products. The resulting vector is the magnitude and phase angle of the output signal. The other method employed a digital-recording system which converted the analogue input and output signals to digital data and stored these data on a magnetic tape. The magnetic tape was then used to enter the experimental data into an IBM-704 computer. The digital-computer programme was written to compute the cross-correlation of the input-output signals and the amplitude and phase shift of the reactor transfer function.

The measured power transfer functions are shown in Fig. 28. The gain is given by

$$20 \log_{10} [(\Delta n/n_0)/(\Delta k/\beta)],$$

where Δn = peak value of sinusoidal component of neutron flux; n_0 = average neutron flux; Δk = reactivity; and β = delayed neutron fraction.

The absolute gain of the power transfer function was not measured directly because of the uncertainty in control-rod worth in the presence of voids. Therefore, the data of Fig. 28 have been normalized to the zero-power transfer function by providing a best fit to the three highest frequency points. This type of normalization is valid provided the feedback is negligible at these frequencies.

Reactor stability was examined by calculating the closed loop gain and phase margin. The block diagram of the reactor with feedback is shown

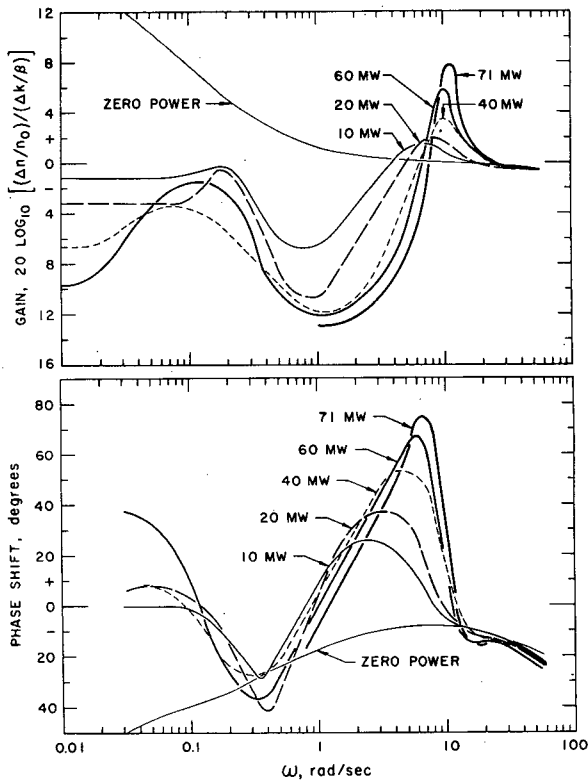


Fig. 28

Reactor transfer functions

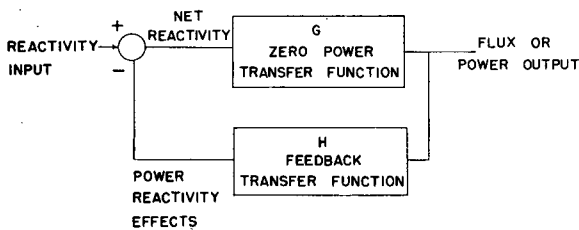


Fig. 29

Block diagram of reactor with feedback

in Fig. 29. The power transfer function is given by

$$P = \frac{G}{1 + GH} = \frac{\text{Flux output}}{\text{Reactivity input}},$$

where P , G and H are complex quantities and functions of frequency. Instability exists when $GH = -1$, or when $|GH| = 1$ with the simultaneous con-

dition of $\angle GH = -180^\circ$. The terms GH can be extracted from the experimental data by substituting G and P in the equation:

$$GH = \frac{G}{P} - 1.$$

The zero-power reactor transfer function (G) is measured at very low power before the onset of boiling.

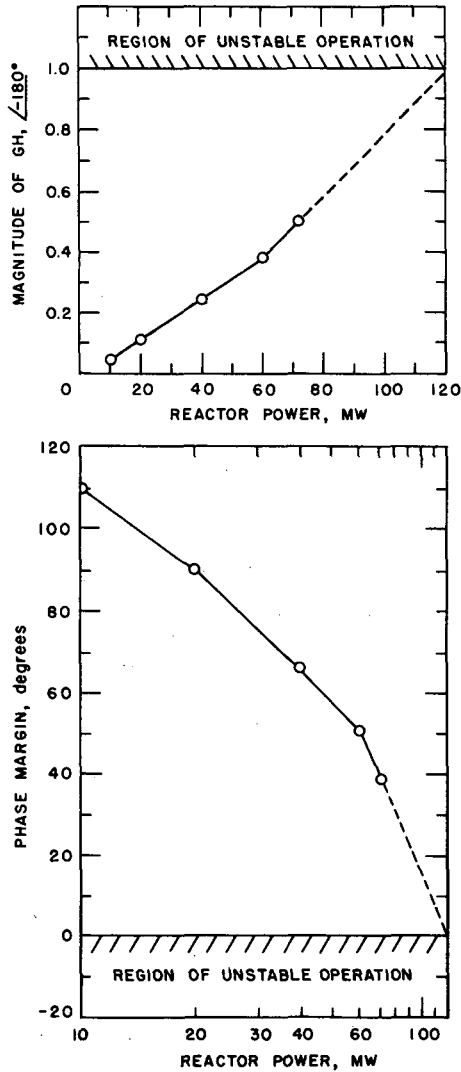


Fig. 30

Reactor gain and phase stability

A plot of $|GH|$ at -180° and of phase margin as function of power is shown in Fig. 30. Phase margin is the additional phase shift that must be added to the $|GH|$ to obtain a total phase shift of -180° at $|GH| = 1$. Extrapolation of the gain and phase margins indicates a maximum stable power of 120 MW(t) for the EBWR.

The maximum power achieved with all control rods withdrawn and with $\sim 2\%$ carry-over was 60 MW(t) (before destripping $\sim 2\%$ worth). To increase power above 60 MW(t), the mode of reactor operation was modified and operation up to 100 MW(t) was achieved by increasing feed-water flow to the reactor. This increased the sub-cooling which, in turn, provided additional reactivity.

(b) Auto-correlation. Operation at the higher powers required all control rods to be withdrawn; consequently, the centre rod was not available for transfer function measurements. At powers above 70.7 MW(t), only noise data were taken; auto-correlation functions and power-density spectra were computed.

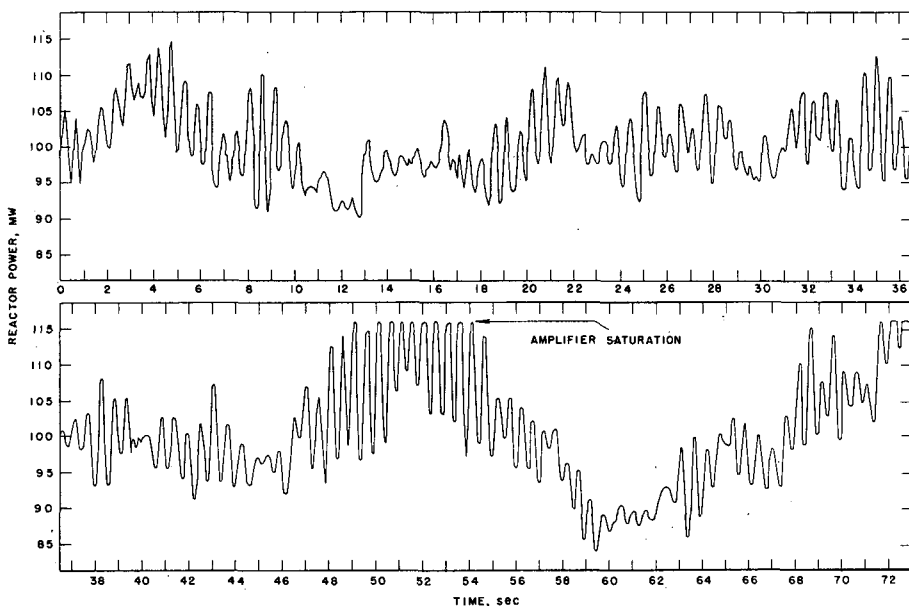


Fig. 31

Reactor power against time as recorded directly from ionization chamber output

Fig. 31 shows a time record of the neutron flux at 100 MW(t). The auto-correlation function is defined as

$$\phi(\tau) = \int_{-\infty}^{\infty} f(t) f(t+\tau) dt.$$

Fig. 32 shows the autocorrelation function at 100 MW(t).

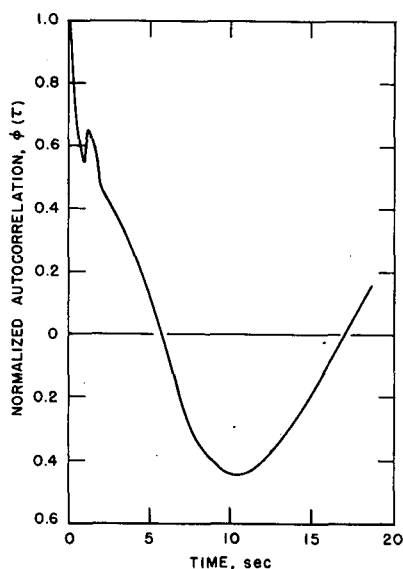


Fig. 32

Auto-correlation function at 100 MW(t)

The square root of $\phi(0)$ gives the root mean square (RMS) flux noise. Table V lists the RMS noise under various conditions of power level, water level, pressure, boric acid concentration, control-rod position and whether upper or lower feed-water ring was used.

(c) Noise frequency spectrum. The Fourier transform of the auto-correlation function gives the energy density or power spectrum:

$$\Phi(\omega) = \frac{1}{2\pi} \int_{-\infty}^{\infty} \phi(\tau) \exp(-j\omega\tau) d\tau$$

Fig. 33 shows the 100 MW(t) power spectrum. The 100 MW(t) auto-correlation and power spectrum were computed on the IBM-704 digital computer, using a total of 2000 samples taken at 0.096-s intervals. The maximum shift (τ) used in computing the auto-correlation function was 200 samples.

(d) Random reactivity input. To utilize the digital techniques available more completely, an experiment was performed in which the reactor was excited with a random reactivity. The randomness was of a particular nature referred to as fixed-interval binary noise. Fig. 34 shows a signal of this type. The control-rod position is varied about a mean position in a random manner, but the change in position is allowed to take place only at fixed time intervals. The control-rod mechanism on EBWR was velocity limited, consequently, the actual input waveform appeared as in Fig. 35.

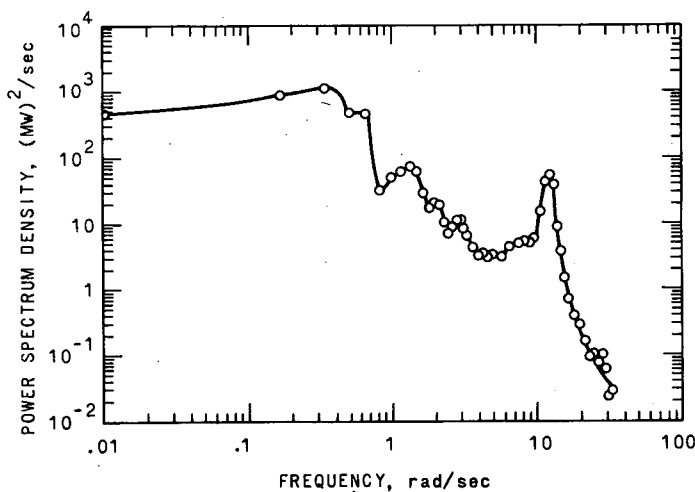


Fig. 33

Noise frequency spectrum at 100 MW(t)

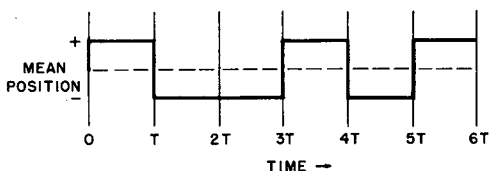


Fig. 34

Discrete-interval binary noise

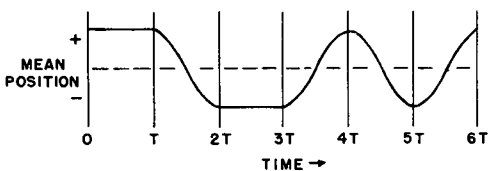


Fig. 35

Velocity-limited, discrete-interval, binary noise

The frequency spectrum for the EBWR velocity-limited, discrete-interval, binary noise signal is shown in Fig. 36. The frequency response of the reactor is obtained from the cross-correlation function and power spectrum calculations. Fig. 37 shows the frequency response of the reactor obtained by using a binary noise input. The high-frequency portion of the

TABLE V
ROOT MEAN SQUARE NOISE UNDER VARIOUS REACTOR OPERATING CONDITIONS

Pressure (lb/in ² gauge)	Power (MW)	Water level (ft)	Boric-acid concentration (g/gal)	Feed-water ring	Rods 1-8	Rod 9	RMS noise	
							Analogue	Digital
300	10.0	16.0	4.0	Lower	27.3	23.0	0.97%	0.91%
300	10.0	16.0	0.1	Upper	17.8	14.5	1.06%	1.01%
300	20.0	16.0	4.0	Lower	32.2	23.0	1.72%	1.89%
300	20.0	16.0	0.2	Upper	22.7	14.5	3.20%	3.18%
600	11.5	16.0	5.0	Lower	29.1	23.0	0.81%	0.80%
600	10.1	16.0	0.2	Upper	18.7	14.5	0.90%	1.23%
	10.0	16.0	0.2	Lower	18.4	14.5	0.77%	0.78%
600	13.4	16.0	0.2	Upper		14.5	1.31%	1.38%
600	20.1	16.0	3.9	Lower	30.0	23.0	0.97%	1.01%
600	20.0	16.0	0.1	Upper	22.4	14.5	1.88%	1.96%
600	40.0	16.0	0.2	Upper	29.2	21.5	3.27%	3.20%
600	40.0	16.0	2.7	Upper	48.0	30.0	4.28%	4.01%
600	60.0	16.0	0.5	Upper	48.0	30.0	5.64%	4.96%
600	70.7	16.0	1.1	Upper	40.6	30.0	6.26%	7.43%
600	71.0	16.0	1.0	Upper	34.7	39.8	4.72%	4.55%

TABLE V (cont.)

Pressure (lb/in ² gauge)	Power (Mw)	Water level (ft)	Boric-acid concentration (g/gal)	Feed-water ring	Rods 1-8	Rod 9	RMS noise	
							Analogue	Digital
600	79.3	16.0	1.1	Upper	35.2	39.8	5.20%	5.15%
* 600	74.8	16.0	1.7	Upper	37.0	30.0	4.86%	4.93%
* 600	75.8	16.0	1.7	Upper	34.6	34.6	4.11%	4.04%
* 600	76.8	16.0	1.2	Upper	35.0	35.0	4.19%	
* 600	72.0	16.0	1.2	Upper	34.0	40.0	3.66%	3.45%
* 600	82.0	16.0	1.2	Upper	36.7	36.7	6.88%	5.78%
* 600	90.7	16.0	1.2	Upper	37.7	37.7	8.47%	6.57%
* 600	100.0	15.75	0.1	Upper	48.0	48.0	6.01%	6.77%
* 600	60.0	16.0	0	Upper	30.0	40.0	2.94%	2.65%
* 600	56.0	15.25	0	Upper	32.0	40.0	1.38%	1.35%
* 600	60.7	15.25	0	Upper	34.0	40.0	1.56%	1.47%
* 600	78.1	15.25	0	Upper	48.0	48.0	2.65%	2.59%
* 600	74.7	16.0	0	Upper	34.0	40.0	3.51%	3.50%
* 600	82.0	14.25	?	Upper	48.0	48.0	-	2.21%
* 600	84.6	15.25	?	Upper	48.0	48.0	-	3.35%

* Automatic

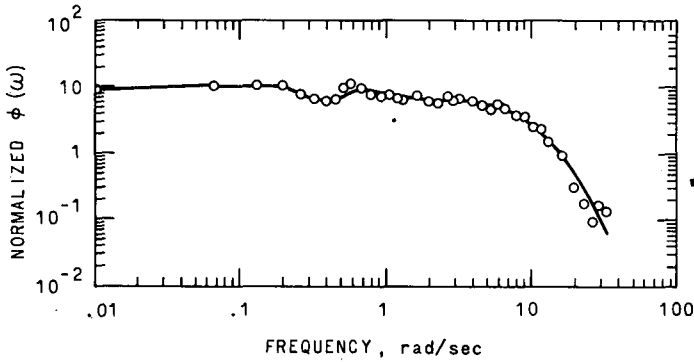


Fig. 36

Velocity-limited, discrete-interval, binary noise frequency spectrum

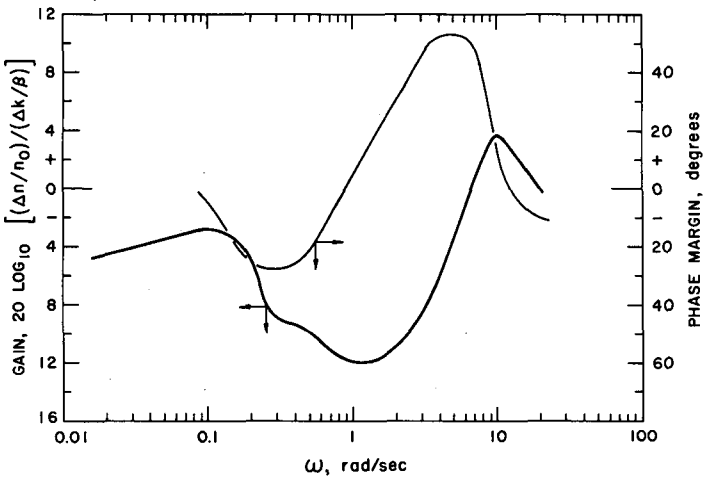


Fig. 37

Transfer function at 40 MW(t) obtained by cross-correlation of neutron flux output with binary noise reactivity input

figure was obtained with a binary time interval of 0.25 s and 9000 digital samples taken at 0.096 s per sample. The low-frequency portion of the record was obtained with a binary time interval of 2.0 s and 9000 digital samples taken at 0.768 s per sample.

The amplitude and phase shift of Fig. 37 is in close agreement with the 40-MW(t) measurement shown in Fig. 28, except for frequencies above 20 rad/s. The velocity limitation imposed by mechanical considerations on the control-rod mechanism resulted in the upper usable frequency limit of 20 rad/s. This frequency limit does not allow the data of Fig. 37 to be normalized to the

zero-power transfer function, consequently, the binary noise technique was not used at higher powers. It should be noted that the high-frequency response portion of Fig. 37 was obtained in 15 min of reactor experimentation time, as compared to 15 min for a single frequency point when using a sinusoidal excitation.

3.4.2. Summary

Reactor transfer functions are useful tools in studying reactor performance. Through extrapolation they allow predictions of stability at higher powers so that increases in power can be made with confidence. The power at which instability will develop also can be predicted. These data can be used to develop a dynamic model which combines all the design parameters—reactor physics, heat transfer and hydraulics—for the specific reactor type under investigation. The dynamic model is more descriptive of reactor performance than are static measurements, e.g. reactivity per unit power or unit void per unit power. The development of a correct model then assures success in optimizing the performance on the next generation of similar reactors.

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IV

EXPERIENCE WITH SPECIFIC NUCLEAR
POWER PLANTS (continued)

THE DEVELOPMENT OF CALDER HALL AND CHAPELCROSS AS BASE LOAD NUCLEAR POWER STATIONS

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Abstract — Résumé — Аннотация — Resumen

THE DEVELOPMENT OF CALDER HALL AND CHAPELCROSS AS BASE-LOAD NUCLEAR POWER STATIONS. The initial design conception of Calder Hall and Chapelcross was as dual-purpose power reactors with the major emphasis on production of plutonium for military purposes. During the design stage optimization was carried out to enable the best possible results to be achieved from the electricity generation point of view. Both stations are integral parts of the National Electricity Grid supply system as base-load power stations and supply about 15% of the demand in the regions in which they are located. Their performance in this capacity has been developed considerably during the seven years that have elapsed since Calder Hall was officially opened in 1956.

The power output of the reactors has been increased by over one-third above the initial design figure. This has been largely achieved by careful research into the methods of temperature assessment of the fuel elements and the use of control rods as movable absorbers to supplement the fixed absorbers used for flux shaping. The criterion on which temperature control is based has been developed over the years in order to allow the maximum output to be obtained while still retaining very adequate safety margins. One result of this has been that the turbo-alternator sets have been modified to increase their outputs from the original design figure of 21 MW(e) to 27/30 MW(e) in order to cope with the extra steam.

Calder Hall and Chapelcross reactors are now achieving overall load factors in excess of 92% in spite of the fact that refuelling is carried out off-load. Increases in load factor have been largely achieved by marked reductions in the time taken up by refuelling and by careful planning of essential maintenance work which involves shutting down reactors. In this respect gains have been made by the application of method study and critical path techniques. Losses due to failed fuel elements have been very small.

In addition to the operation of the Station for electricity generation, a large number of special irradiation experiments are carried out to improve the design of fuel elements, both for the Calder Hall and Chapelcross reactors and for civil reactors in the United Kingdom and Overseas. It has been found possible, by careful planning, to mount such experiments with negligible loss in production and without encroaching upon operating safety margins.

The future operating policy of these stations depends upon two main factors, namely, the possibility of further increases of reactor power output and an extension of burn-up times of reactor fuel. Advances are expected under both these headings, particularly the latter, and these should have appropriate beneficial effects upon the efficiency and economics of the station.

AMÉNAGEMENT DES CENTRALES NUCLÉAIRES DE CALDER HALL ET CHAPELCROSS EN CENTRALES NUCLÉAIRES DE BASE. Calder Hall et Chapelcross avaient été conçus à l'origine comme des réacteurs de puissance à double fin, destinés surtout à produire du plutonium à des fins militaires. Au cours de l'élaboration des plans des mesures ont été prises pour obtenir les meilleurs résultats possibles du point de vue de la production d'électricité. Les deux centrales sont intégrées au réseau national d'électricité comme centrales de base et fournissent environ 15% de la demande dans les régions où elles sont situées. Leur fonctionnement comme telles s'est considérablement développé au cours des sept années écoulées depuis la mise officielle en service de Calder Hall en 1956.

La puissance des réacteurs a été accrue de plus d'un tiers par rapport au chiffre initial prévu. On y est parvenu surtout grâce à des recherches minutieuses sur les méthodes d'évaluation de la température des éléments combustibles, et grâce à l'emploi de barres de contrôle comme absorbeurs mobiles complétant les absorbeurs fixes utilisés pour la mise en forme du flux. Les critères régissant le contrôle de la température ont été mis au point peu à peu, au cours des années, de façon à obtenir la puissance maximum tout en gardant une marge

de sécurité très suffisante. Il a fallu en conséquence, modifier les groupes turbo-alternateurs pour porter leur production du chiffre de 21 MWe, prévu originellement, à 27/30 MWe, afin d'utiliser la valeur supplémentaire.

Les réacteurs de Calder Hall et de Chapelcross atteignent maintenant des facteurs de charge globaux de plus de 92%, malgré les arrêts nécessaires pour le chargement du combustible. Les accroissements du facteur de charge ont été obtenus en grande partie grâce à des réductions considérables du temps consacré au rechargement et à une programmation minutieuse des travaux d'entretien essentiels qui nécessitent l'arrêt des réacteurs. A cet égard, on a obtenu des gains de temps par l'emploi de méthodes judicieuses et rationnelles. Les pertes dues au fonctionnement défectueux d'éléments combustibles ont été très faibles.

Outre la production d'électricité, de nombreuses expériences spéciales d'irradiation ont été effectuées afin d'améliorer la conception des éléments combustibles, à la fois pour les deux réacteurs de Calder Hall et de Chapelcross et pour des réacteurs «civils» tant au Royaume-Uni qu'à l'étranger. On a réussi, grâce à une programmation soignée, à faire ces expériences sans perte appréciable pour la production et sans dépasser les marges de sécurité du fonctionnement.

L'exploitation future de ces centrales dépend de deux facteurs principaux: la possibilité d'augmenter encore la puissance des réacteurs et celle de prolonger la durée d'utilisation du combustible. On s'attend, dans ces deux domaines, en particulier dans le deuxième, à des progrès qui devraient entraîner une amélioration correspondante de l'efficacité et de la rentabilité des centrales.

РАЗВИТИЕ АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ В КОЛДЕР-ХОЛЛЕ И ЧЕПЕЛКРОССЕ КАК АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ С ОСНОВНОЙ НАГРУЗКОЙ. Первоначальная концепция при проектировании атомных электростанций в Колдер-Холле и Чепелкроссе состояла в строительстве энергетических реакторов двойного значения с основным упором на производство плутония для военных целей. Однако на стадии проектирования возникли предположения о возможности достижения наилучших результатов в случае использования реактора для производства электричества. Обе станции являются составной частью национальной системы снабжения электричеством в качестве электростанций с базовой нагрузкой, они производят около 15% всего необходимого электричества в тех районах, где они расположены. В течение семи лет, прошедших с начала работы станции в Колдер-Холле в 1956 году, значительно улучшена их работа в этом направлении.

Выход мощности реакторов увеличился по сравнению с первоначальной проектной цифрой более чем на одну треть. Это в значительной степени достигнуто в результате тщательного исследования методов температурной оценки топливных элементов и использования управляющих стержней в качестве подвижных поглотителей в дополнение к неподвижным поглотителям, используемым для формирования потоков. Критерий, на котором основывается контроль температуры, разрабатывается в течение ряда лет с целью получения максимального выхода мощности при одновременном обеспечении вполне надлежащего коэффициента безопасности. Одним из результатов этого является изменение систем турбоальтернаторов с целью повышения их мощности с 21 до 27 - 30 мвт (эл.) для обработки дополнительного пара.

На реакторах в Колдер-Холле и в Чепелкроссе в настоящее время величина общего коэффициента загрузки превышена на 92%, несмотря на то, что перегрузка осуществляется без дополнительного увеличения количества топлива. Увеличение коэффициента загрузки достигнуто в основном в результате значительного сокращения времени при перегрузке топлива и тщательного планирования основной работы по ремонту, которая связана с остановкой реактора. В этом отношении увеличение коэффициента загрузки достигнуто в результате проведения методического изучения и применения методики критической траектории. Потери в результате разрушения топливных элементов очень невелики.

Помимо эксплуатации станции для производства электричества, проводится большое число специальных опытов по облучению с целью улучшения конструкции топливных элементов как для реакторов в Колдер-Холле и Чепелкроссе, так и для гражданских реакторов в СК и за океаном. Признано возможным при тщательном планировании увеличить число таких опытов с незначительной потерей в производстве и без превышения эксплуатационного коэффициента безопасности.

Направление эксплуатации этих станций в будущем зависит от двух основных факторов, а именно: возможности дальнейшего увеличения мощности реактора на выходе и увеличения периодов выгорания реакторного топлива. Ожидается, успехи в работе в обоих направлениях, особенно в последнем, и это окажет положительное влияние на эффективность и экономические показатели станции.

EL DESARROLLO DE CALDER HALL Y CHAPELCROSS COMO CENTRALES NUCLEOELECTRICAS PARA LA CARGA BASICA. Calder Hall y Chapelcross fueron concebidos en un principio como reactores de potencia

de doble finalidad, principalmente destinados a producir plutonio con fines militares. Al diseñarlos, se adoptaron medidas con miras a obtener resultados óptimos desde el punto de vista de la producción de electricidad. Ambas centrales forman parte de la red nacional de distribución en calidad de centrales para la carga básica, y suministran aproximadamente el 15 por ciento de la demanda en las regiones en que están situadas. Su rendimiento para este tipo de producción se ha mejorado considerablemente en los siete años transcurridos desde que Calder Hall entró oficialmente en servicio en 1956.

La potencia de los reactores ha sido incrementada más de un 33% con respecto a la cifra inicialmente prevista. Esto se ha logrado principalmente merced a numerosos estudios de los métodos para determinar la temperatura de los elementos combustibles, y empleando las barras de control como absorbedores móviles para complementar a los absorbedores fijos con que se configura el flujo. El criterio en que se basa el control de las temperaturas se perfeccionó paulatinamente para alcanzar una potencia máxima manteniendo al mismo tiempo un margen de seguridad suficiente. Los turboalternadores se han modificado para aprovechar el exceso de vapor producido incrementando su capacidad desde 21 MW(e) que daba el diseño original hasta 27/30 MW(e).

Actualmente, los reactores de Calder Hall y Chapelcross alcanzan factores de carga globales superiores a 92% a pesar de las interrupciones necesarias para cargar el combustible. Los incrementos del factor de carga se deben en gran parte a las considerables reducciones del tiempo necesario para renovar el combustible y al planeamiento cuidadoso de todos los trabajos de conservación esenciales que exigen la detención de los reactores. A este respecto, se ha ganado tiempo aplicando métodos más eficaces y racionales. Las pérdidas debidas a elementos combustibles defectuosos han sido muy pequeñas.

Además de explotar la central para la producción de electricidad, se realizan en ella numerosos experimentos especiales de irradiación para mejorar el diseño de elementos combustibles, tanto de los reactores de Calder Hall y de Chapelcross como de otros reactores del Reino Unido y de ultramar destinados a fines civiles. Gracias a un planeamiento meticuloso, se ha logrado efectuar estos experimentos sin pérdidas apreciables para la producción y sin reducir excesivamente los márgenes de seguridad.

La futura explotación de las centrales depende de dos factores principales: posibilidad de aumentar nuevamente la potencia de los reactores y de prolongar el período de utilización del combustible. En estos dos campos, y especialmente en el segundo, cabe esperar progresos que se traduzcan en un aumento de la eficacia y rentabilidad de las centrales.

INTRODUCTION

Calder Hall and Chapelcross Works together comprise eight gas-cooled, graphite-moderated, natural-uranium fuelled reactors, the last three reactors constructed (at Chapelcross) being fitted with graphite sleeves. The first reactor at Calder was made critical in May 1956 and the whole station was completed early in 1959. At Chapelcross the first reactor achieved criticality in November 1958 and the station was operating at full power early in 1960 [1]. The initial design figure for reactor power was 180 MW(t) but development of operating techniques has enabled considerable power increases to be achieved. Present reactor powers vary from over 230 MW to 250 MW, the highest powers being achieved on the graphite sleeved reactors. In addition, refuelling and maintenance times have been considerably reduced since 1956 and overall load factors of 94% are now being achieved. Between refuelling shut-downs load factors of well over 98% are being regularly achieved.

Each station has eight turbo-alternator sets, the initial ratings of which were 21 MW(e). In order to cope with the excess steam made available by the increased reactor powers, six of the Calder turbines have been rebladed to allow generation of 27 MW(e) per set. In addition, large quantities of process steam are being provided for the Windscale plant. The saving of coal thereby achieved amounts to about 50 000 t/yr. At Chapelcross four

TABLE I

CALDER HALL AND CHAPELCROSS REACTORS
Improvements over design conditions

	Design conditions	Present performance
Reactor circuits		
Maximum fuel-element can temperature	408°C	430°C
Gas temperature at heat-exchanger inlet	336°C	345°C
Gas temperature at heat-exchanger outlet	135°C	138°C
Coolant flow per heat exchanger	490 lb/s	635 lb/s
Steam circuits		
High-pressure steam flow per heat exchanger	99 000 lb/h	126 000 lb/h
High-pressure steam pressure	195 lb/in ² gauge	235 lb/in ² gauge
High-pressure steam temperature	315°C	325°C
Low-pressure steam flow per heat exchanger	29 650 lb/h	46 000 lb/h
Low-pressure steam pressure	48 lb/in ² gauge	60 lb/in ² gauge
Low-pressure steam temperature	180°C	185°C
Reactor power	180 MW	240 MW
Gross electrical generation	42 MW(e)	55 MW(e)
Net electrical generation	34.5 MW(e)	45 MW(e)

of the turbines have been rebladed to permit generation of 27 MW(e) per set and four to permit generation of 30 MW(e) per set.

Some details of the means by which these power increases have been achieved, while still operating within the safety limitations of the plant, will be given later. However, the facts show that both Calder Hall and Chapelcross have functioned as highly efficient production units on a normal operating basis from start-up, and have not been used simply as prototypes. This type of reactor has been used in the United Kingdom as a basis for the first phase of the civil programme for the generation of electricity using nuclear reactors. Present operating conditions at Calder Hall and Chapelcross compared with design values are shown in Table I.

The reactors are operated to provide electrical base load, one of the aims being to achieve maximum availability and to demonstrate that reactors have a high standard of reliability. In order to do this, every effort has been made to reduce planned shut-downs such as refuelling and maintenance periods. The overall load factor has been increased from 67% in the very early days of operation to 94% in 1962, the most striking improvement arising

from a cut in the refuelling times from 22.5% to 4%. The extent of the improvement has been such that refuelling is no longer the controlling factor in so far as outage times are concerned. Maintenance requirements are now predominating and special attention is being given to the planning of such work — this will be referred to later in the paper. The reliability of the operating performance can be illustrated by the fact that during the winter periods of 1960, 1961 and 1962 the maximum generated load during peak-load periods was maintained by both stations.

As base-load stations, both Calder and Chapelcross are integral parts of the National Grid system and each supplies about 15% of the load in the areas in which they are situated. Fig. 1 shows their positions in the grid system and it can be seen that Calder is connected to the National Grid by much longer transmission lines than Chapelcross. The effects of this on operating experience are evidenced by an incident in 1960 when both stations were subjected to very severe electrical storms when lightning struck the overhead transmission lines from Calder to Harker [2]. The transmission line protection operated and isolated the section, but the voltage transient which occurred was sufficiently severe to stop the main blower drive motors. The reactor protection was not affected by the voltage variation and correctly tripped the reactors as coolant mass flow fell. Action has been taken to improve the stability of essential electrical equipment during such faults so that all important auxiliary plant has a stability under electrical voltage variations equal to or better than the stability of the coolant circulator driving motors, which are linked to the safety circuits.

At Chapelcross, however, the effect of such storms is much less marked. Chapelcross has four transmission lines which approach the Works from different directions and, except for a section of approximately 14 miles long, are geographically separated. The lines carried on the double circuit section emanate from different sub-stations. It is possible that a lightning strike on the Chapelcross system, unless directed at the 132-kV busbars, would leave the Works still connected to two other sub-stations and retaining at least 50% of its grid connecting capacity. In fact, of the six recorded occasions of lightning storms affecting Chapelcross, five have resulted in the tripping of single lines and one has resulted in three lines tripping due to the incorrect operation of the protection circuits on two of them. In the case of Calder, the two transmission lines to the North, which represent two-thirds of its grid connecting capacity, are carried on double circuit towers. They are, therefore, simultaneously vulnerable to lightning disturbances along their path. As nuclear power stations at present normally have to export their power via long transmission lines, an important feature to prevent shut-downs must be the matching of reactor protection and coolant circuits to the transients which may occur during fault clearance by transmission line protection.

The power developed by the stations and the electrical load factors achieved since start-up are shown in Fig. 2. The excellent progress shown in these diagrams is in part due to an improvement in skill and knowledge of the personnel operating and maintaining the plant. This is further evidenced in Table II which shows how the numbers of automatic shut-downs at Chapelcross have decreased each year since start-up. Similar progress has been achieved at Calder.

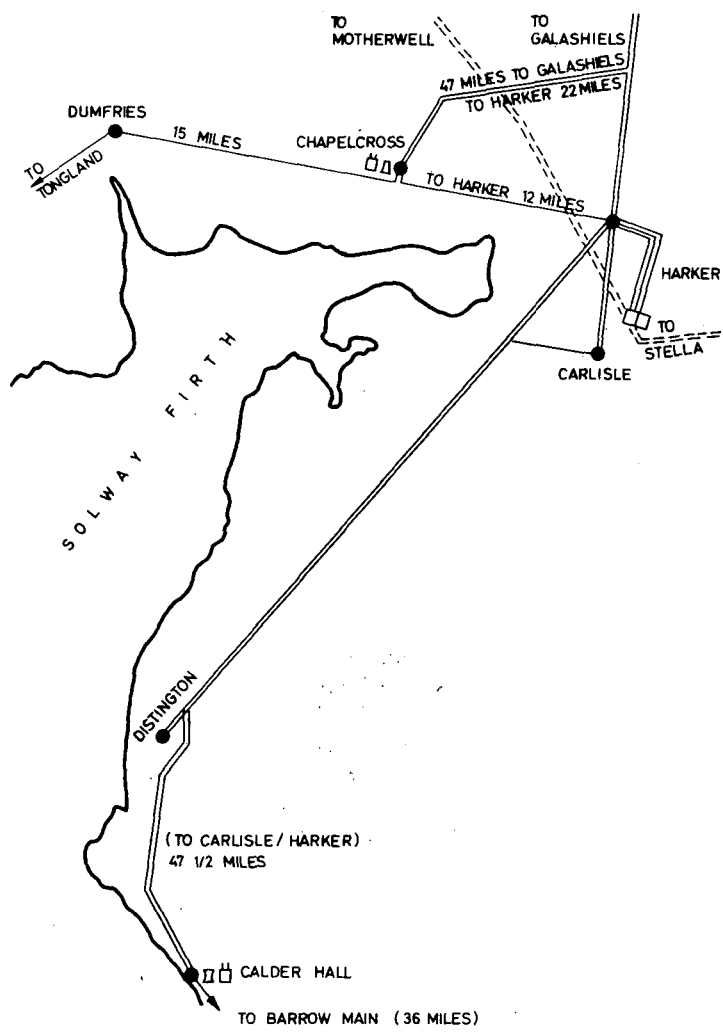


Fig. 1

Geographical position of Calder Hall and Chapelcross

- 132-kV transforming station
- ▢ 275-kV transforming station
- 132-kV transmission line
- 275-kV transmission line.

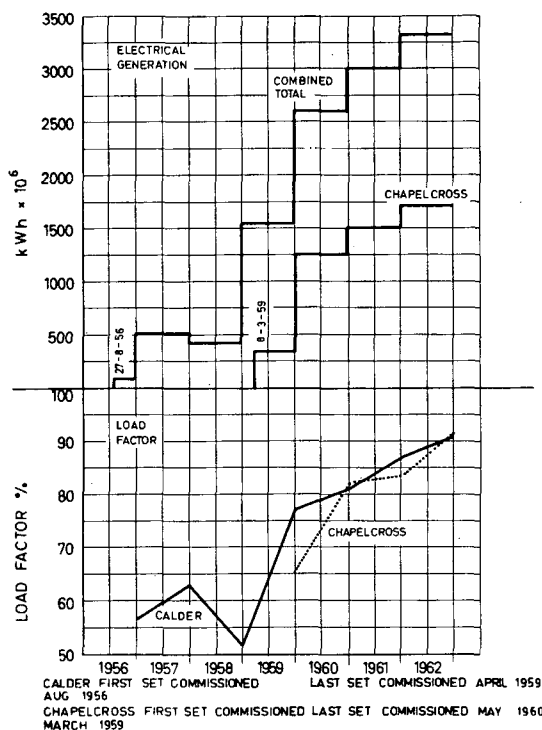


Fig. 2
Electrical generator and load factor

TABLE II

CHAPELCROSS REACTORS
Automatic shut-downs

Year	Equipment* faults	Operational errors	No. of reactors at power	Trips/reactors/year
1959	5 (10)	3	2	9
1960	4 (14)	11	4	7
1961	2 (2)	4	4	2
1962	2 (1)	1	4	1

* Figures in brackets indicate the number of "flashovers" on the main gas blowers - a special type of fault which has now been virtually cured.

INCREASING REACTOR POWER

Reactor output is limited by various factors, the most important being the assessed safety margins concerned with the reactor core and the "design" limitations of the various items of plant. These in turn lead to a list of operating limitations which the reactor operator is bound to observe at all times. However, this does not prevent the margins left for uncertainty in a design from being examined and plant operating conditions adjusted to effect improvement [3].

Improvements in performance of the Calder and Chapelcross reactors have followed from practical experience of the effects of different absorber patterns and configurations of partially-inserted control rods under different operating conditions. The practical aspects of the optimization of power distribution in these reactors, which have no facilities for on-load fuel changing, can be considered under two major headings: variables which must be fixed before the reactor is taken to power, such as absorber patterning and fuel-element loading; and other factors, such as control-rod insertion, which can be altered while the reactor is at power. Since the power losses arising from errors in absorber patterning may be heavy, there is a strong incentive to develop a method of predicting the temperature distribution associated with a particular reactor charge [4]. In order to achieve accurate results, a computer programme has been produced [5], the object of the programme being to predict fluxes and temperatures for all channels in the reactor. In addition, it is required to assess the effects of replacing fuel channels with channels of absorber as core irradiation proceeds. As information on the differences between predicted and measured temperatures has accumulated, corrections have been applied to the programme. The net result of this work has been to allow optimization of power by the choice of an initial absorber pattern when the reactor is loaded. During the irradiation of the fuel, further adjustments to the absorber pattern are made by using selected control rods as movable absorbers - this is particularly important in these reactors, where changes in fixed absorber loading can only be made after shutting down the reactor. As control-rod insertion into the core increases (with increasing irradiation) flux and temperature distributions are adversely affected and reactor power is reduced. An optimized temperature distribution can be achieved, and maintained for considerable periods, by inserting some of the control rods further into the core. Increases in reactor power of up to 20 MW have been achieved by this method.

In the Calder-type reactors the coolant flow is matched to the flux shape by gagging of fuel channels to give the desired maximum fuel-element can temperature. In practice only twenty-one different gag sizes are used, so that many of the channels in the reactor may be as much as 40°C low. In the operating reactor the temperature distribution is inferred from a limited number of thermocouple readings, e. g. in the Calder and Chapelcross reactors there are 104 thermocouples loaded, two in each of four channels on 13 separate radii. The channels into which thermocouples can be loaded are limited and it is, therefore, difficult to obtain a really good estimate of the maximum can temperature in the reactor. In fact, the reactor is controlled by taking the mean maximum indicated can temperature on each of the

instrumented radii and increasing reactor power until one of these temperatures attains the limit.

The use of smaller diameter thermocouples has enabled more temperature measurements to be made and thus more information obtained about the random and systematic variations in fuel-element temperatures. A new method of temperature assessment has been defined based on the statistical probability of any fuel-element temperature exceeding the melting point of Magnox during the temperature transient which would arise following a fracture of one of the main gas circuits with complete loss of coolant, but with control-rod insertion. This new system of fuel-element temperature assessment has now transferred control of reactor power to the operating limit on the temperature of the steel of the top dome of the pressure vessel. Further research and development work is now in progress to re-assess the safety margins involved in the setting of this limit.

An interesting, and unexpected, limit on reactor power arose from the observation that temperatures in diagonally opposed quadrants of the reactor were higher than those in the other two quadrants after one of the heat exchangers had been returned to circuit. The effect was observed on one of the Calder reactors in 1959 and it was possible to carry out tests during the commissioning of Chapelcross Reactor 4 to demonstrate that with four blowers in circuit two flow patterns are possible. The solution was also found: raising power should be carried out with two diagonally opposed blowers running 100 rpm in advance of the other pair. Changes of up to 10 MW in reactor power have been achieved by this means.

INCREASING LOAD FACTORS

Overall load factors at Calder Hall and Chapelcross are now over 94%; even in its first full year of operation, the thermal load factor at Chapelcross was 79%. Table III shows the improvements at Chapelcross since 1959.

A summary of outages during 1962 is given in Table IV.

It can be seen that the largest single item is refuelling and it is in this field that the greatest gains have been made. At Chapelcross the first refuelling of a whole reactor core took 35 d; recently a two-thirds reactor refuelling was carried out in 9 d. This improvement has been achieved by careful attention to:

- (a) Plant;
- (b) Procedures;•
- (c) Planning.

(a) The refuelling equipment, of course, suffered from teething troubles; these were dealt with as matters of urgency and all components with a relatively short life have been redesigned to improve reliability and also to permit rapid replacement [6].

The use of television cameras has proved invaluable, both for inspection purposes and as a means of assessing the action to be taken when fuel elements have suffered damage which prevented their being discharged by the standard methods. Much development work has been done to improve the

TABLE III
CHAPELCROSS REACTORS
Overall load factors

Year	Thermal load factor (%)	Electricity exported (Units $\times 10^6$)	Electrical load factor (%)
1959	79.3	258.759	* 66.0
1960	86.8	995.524	82.5
1961	85.3	1202.262	83.8
1962	94.3	1403.660	93.6

* Six generating sets were commissioned during this period.

$$\text{Thermal load factor} = \frac{\text{MWd produced}}{\text{MWd produced} + \text{MWd lost}}$$

$$\text{Electrical load factor} = \frac{\text{Units exported}}{\text{Hours} \times (\text{target generation capacity} - \text{target own usage})}$$

TABLE IV
CHAPELCROSS REACTORS
Summary of outages - 1962

Blower maintenance		0.53%
Other maintenance		0.21%
Automatic shut-downs		0.09%
Physics experiments		0.20%
Failed fuel discharges		0.58%
Irradiation experiments		0.67%
Refuelling		<u>3.43%</u>
Total		5.71%

robustness and versatility of the television equipment. By making use of annular hoses for carrying coolant to the camera, it is now possible to use the equipment in the core at temperatures of about 150°C for several hours. Several types of grab have been designed which operate directly from the nose of the camera to remove small loose articles. The importance of this equipment cannot be overstressed [7].

As far as possible, plant maintenance is carried out between refuelling shut-downs. In certain cases, plant modifications have been carried out to enable this to be done, for example, extra isolating valves have been fitted to the humidifiers to provide double isolation so that they can be worked on safely with the reactor at power.

(b) Fuel-handling procedures and techniques have also received careful attention. Specialist teams have been formed to carry out the tasks of fuel handling and also the detailed planning of fuel movements. This planning is carefully integrated with the maintenance work to be carried out on the reactor. After each refuelling operation a careful analysis is carried out with the object of identifying areas where further improvements can be made. Method study techniques have been applied which have resulted in an increase in discharge machine utilization from 50 to 70%.

Considerable advantage was also gained from a simple modification to the fuel-element thermocouple leads. Originally these elements had to be discharged separately into a special container and then handled separately at the cartridge cooling pond where the thermocouple cable was cut by a remotely operated mechanical device. The modification consists of a small length of reduced diameter on the cable near the element, a sharp pull severs the cable at this point, the cable is wound onto a shielded drum and the fuel element discharged in the normal way. These "weak link" thermocouple cables have now replaced the original type.

(c) The improvements in refuelling techniques have reached the point where the outage time due to refuelling shut-downs depends on the speed with which essential maintenance and inspection work can be carried out. Method study techniques are being employed here also, in conjunction with Critical Path Planning to identify the jobs where attention is most urgently required. In all project or outage planning, the jobs required to accomplish the project must be determined. This is usually easy to do, but finding how they interrelate and constrain each other is not so easy. The critical path method does this graphically by denoting each job or activity by an arrow, and constructing a network of these on the basis that any job cannot start until its necessary predecessors are complete. Once the network diagram is complete, and time estimates given to each job, by previous experience, time study or estimation, the longest sequence of jobs from beginning to end of the network - the critical path - can be found [8]. The jobs lying on this path control the total duration of the projects. From the network, jobs can be found which can be done concurrently, thus saving time. In addition, effort can be diverted from jobs not on the critical path further along the network to earlier jobs which are on the critical path. In simple networks, earliest and latest possible starting times and latest possible finishing times for any job can be easily calculated by manual methods, but where the number of jobs runs into some hundreds, a digital computer may be used to calculate minimum costs, optimum manning, etc. Fig. 3 shows a sample network: this is an actual example and indicates how this method of appraisal can direct attention where it is needed. Before carrying out this analysis it was thought that the more complicated task of electrical overhaul was the "holding" item, rather than bellows inspection.

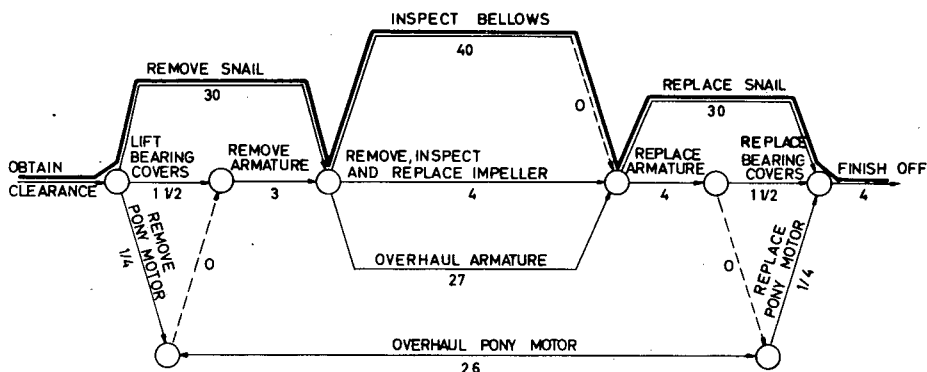


Fig. 3

Critical path method - gas circulator overhaul

— Critical path

Numbers indicate time taken in hours to complete each job.

FUEL-ELEMENT PERFORMANCE

The excellent performance of the fuel in the Calder and Chapelcross reactors has also helped considerably in maintaining high load factors. Of the several hundred thousand standard fuel elements loaded into the Calder and Chapelcross reactors, only seven sudden failures have occurred. These were detected by a sharp rise in the gaseous activity of the bulk coolant, and in each instance the reactor was shut down immediately. The failures occurred in newly loaded elements after a short period of irradiation. Subsequent examination showed in all instances a leak in the end cap weld at one end of the element, and a crack or split in the can wall below which was a mound of oxide. The end cap leaks were undetected manufacturing defects or the results of damage during loading. Oxidation began when the element was loaded, the coolant leaking in through the hole, some distance from the fuel itself, until the mound of oxide on the fuel caused rupture of the can itself.

The mechanism of failure suggested that the rate of fission-product emission might be increased by creating a pressure differential across the leak, which can be done by steady reduction of circuit pressure with the reactor at low power. This method has proved very successful and since its use to prove all new fuel immediately after loading, there has not been an unscheduled shut-down on any reactor caused by such faulty fuel. Subsequent investigations have shown that enhanced emission of fission products can also be obtained by power and temperature oscillations; the resulting differential expansion between canning and uranium is thought to have the same effect as a pressure step.

The only systematic mode of failure experienced has resulted from a tendency for cavitation and porosity to develop in the Magnox cladding of the cooler fuel elements in a channel; at this temperature the ductility is sometimes inadequate to accommodate the small dimensional changes of the

uranium under irradiation. With the original type of Magnox cladding intended only for low irradiation, cavitation developed in some cartridges after about a year, but others have satisfactorily withstood four years in a reactor [6].

CALDER AND CHAPELCROSS AS EXPERIMENTAL REACTORS

In addition to operating the eight Calder and Chapelcross reactors as electricity production units, a major activity of the Production Group of the Atomic Energy Authority is the manufacture and marketing of reactor fuel. It is anticipated that more than 100 000 fuel elements per year will be supplied over the next few years to the first generation of civil nuclear power stations [9].

The uranium fuel rods for civil reactor fuel elements are almost identical with those employed in Calder reactors but the fuel-element can designs vary considerably. Various forms of helical finning with longitudinal "splitter" blades replace the simple circumferential fins and short support braces of the Calder fuel elements. New metallurgical treatment has improved the low-temperature ductility of the cans, which have been designed to withstand on-load charging, and the effects of higher irradiation levels. These design changes depended initially on extensive out-of-pile tests. To support these a comprehensive testing programme was mounted in the Calder and Chapelcross reactors to test design details before bulk manufacture.

In this respect the Chapelcross sleeved reactors have provided considerable versatility in accepting sleeved elements and other elements of greater than normal diameter.

The Calder and Chapelcross production reactors are particularly suitable for these tests since they can permit the irradiation of relatively large numbers of fuel elements, thus providing the necessary statistical accuracy. Minor modifications to fuel handling equipment were necessary, but considerable effort has been necessary to increase in-pile instrumentation. An essential part of the irradiation testing programme has been the detailed metallurgical examination of the irradiated fuel by the A. E. A. Reactor Group's Windscale Laboratories [10].

In practice, the capacity of a reactor for experimental work is limited by the conflicting requirements of different experiments. There is, therefore, considerable advantage in having a number of reactors for large scale fuel testing under differing environments. Substantial numbers of fuel elements with different forms of heat transfer surface and having different metallurgical treatments have now received irradiations up to channel averages of over 3000 MWd/t (metric). The Berkeley fuel, loaded into a Chapelcross reactor in 1959, is at present leading the fuel operating in the Berkeley Civil Station by nearly 3000 MWd/t [9].

In service, the civil reactor fuel elements will experience rapid thermal changes on being loaded and the sector control rods will make frequent significant movements to counter the reactivity effects of the insertion of the refuelling machine grab. On loading a fuel element may suffer changes of about 100°C/min and channels near the sector rods will experience some 2000 thermal cycles of over 10°C during their irradiation life.

To test the effect of on-load charging, an element was inserted into a fuel channel at power and irradiated at normal temperature before being withdrawn. Examination showed that the element had suffered no damage as a result of this full-power charge and discharge procedure. In addition, a reactor has been operated for many months with two diagonally opposite control rods at the outer edge of the reactor oscillating with similar amplitudes but out of phase. Reactor power was hardly effected and no major significant thermal changes were observed. However, fuel near the oscillating rods suffered cycling of about 30°C. Fuel operating at a mean temperature of about 350°C has experienced 30 000 cycles and other cartridges at a mean temperature of 450°C have received 15 000 cycles without failure.

Such irradiation experience under actual operating conditions is unique, and represents a convincing demonstration of the expected performance and long term endurance of uranium metal, magnesium-alloy clad fuel elements. It seems likely that operators of British Magnox designated reactors will eventually be expecting fuel irradiation lifetimes of 4000 to 5000 MW d/t, with a consequent reduction in fuelling costs [9].

Many experiments have been performed to measure the main parameters affecting the safe and efficient operation of the plant. Most of the experimental techniques have a direct application in the civil nuclear stations, and in certain cases the results obtained have reduced the need for similar measurements on the civil reactors. Experiments of particular value to the civil programme were performed during long irradiation runs at both Calder and Chapelcross.

From the standpoint of nuclear safety the most important reactor parameters are those which determine the net reactivity of the system. This is the algebraic sum of positive and negative reactivity contributions and must be assessed for all operational conditions. Nuclear safety requires a similar assessment under possible fault conditions, which involves temperatures outside the range of normal reactor operation. In the initial stages of fuel irradiation the overall temperature poisoning is negative so that cold conditions are the most reactive and hence, for shut-down capacity requirements, the most stringent. As irradiation proceeds the moderator temperature coefficient becomes progressively more positive, until at about 650 MWd/t and normal operating temperature, the hot conditions become the most hazardous. Since it is not possible to measure reactor parameters at temperatures considerably in excess of operating values, the semi-theoretical expression for moderator temperature coefficient has been extrapolated to reactor temperatures close to the melting point of uranium, showing that the Calder-type reactors would remain shut down even in this extreme physical state. Experiments performed on power reactors must be less precise than those performed under laboratory conditions. Such full-scale plant experiments are costly and there is, therefore, an economic limit to the accuracy to which parameters should be measured. Beyond this limit further experiments cannot normally be justified, but at Calder and at Chapelcross some attempt has been made to achieve greater precision of measurement than is justified by plant requirements.

Due to significant differences in size and mode of operation between the civil and the Calder-type reactors, some measurements of the main

reactivity parameters will be required on the civil reactors. The extent to which experimental work of this kind can be fitted into the operational routine will inevitably be restricted by base-load generation requirements [11].

CONCLUSIONS

Thirty-seven reactor years of base-load operation at Calder and Chapelcross have been accompanied by marked increases in power output, exceedingly high load factors and steady improvements in fuel element performance. This record, in conjunction with the almost complete absence of faults, has established great confidence in the operators of these stations; more important, it firmly establishes this type of reactor plant as capable of routine operation in a National Grid system.

Further improvements will be achieved at Calder and Chapelcross. To some extent increases in power output must be controlled by the temperature limit imposed on the pressure vessel steel; a research programme is under way to establish this limit with more accuracy. However, there are other ways of extracting more heat from the core and several are under consideration, e.g. improvements in fuel-element design, improvements in the heat-transfer properties of the coolant gas and the possibility of modifications to the heat-exchanger system. All such proposals must be approved on economic grounds in order that capital spent and generation time lost should be recovered with a suitable margin of profit. Nevertheless, the prospects for further increases in output are considered to be good.

Increases in load factors will be difficult to obtain and it is of the greatest significance that, even with off-load charging of fuel, outage times are governed by plant maintenance requirements – not by the need to remove fuel elements.

Off-load fuel handling is by no means as uneconomic as it seemed when Calder Hall was first constructed. As the Calder and Chapelcross stations move to higher fuel "burn-ups" a batch discharge programme is being developed which will minimize electricity production costs and which will also enable reactivity requirements to be met without reducing the present load factors [12].

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DISCUSSION

S. YIFTAH: The development of Calder Hall and Chapelcross as base-load power stations has been strongly emphasized, and one cannot but be impressed by the excellent performance of these reactors. The question of economics naturally comes to mind, however, and I wonder how seriously the dual nature of Calder Hall and Chapelcross affects the economics of power production. Can one quote any meaningful cost per kWh for reactors such as these or does every consideration of economics have to be discarded as irrelevant?

J. McCRICKARD: The first consequence of this dual purpose is a lower overall efficiency than could be attained in nuclear stations designed only for electrical power production. Your surmise that considerations of economics must be put aside is therefore quite correct; because of the special commitments which these two stations have, I can quote no figures that would allow a meaningful comparison with the cost of other stations. Needless to say, Calder Hall and Chapelcross are operated as economically as possible under the circumstances.

O. ČERNÝ: How do you determine the location of CO₂ leaks into the environment, and of water leaks into the CO₂?

J. McCRICKARD: The loss of CO₂ is measured during each shift by noting the amount of CO₂ supplied to maintain vessel pressure at a constant level. This quantity – no more than 1 t/d – is not measurable in the local environment and the contamination hazard is negligible; nor is the loss expensive, since the coolant gas costs only £11/t.

Water leaks into the coolant gas are first detected by the infra-red gas analysers installed in each circuit and afterwards are confirmed by chemical analysis of the coolant gas. The actual point of leakage is then found by draining the steam side and testing for CO₂ through the header caps, again with an infra-red gas analyser.

O. ČERNÝ: How is the CO₂ that leaks into the plant's rooms removed, and its activity reduced?

J. McCRICKARD: I have already pointed out that the amount of CO₂ lost is very small and of negligible activity. Nevertheless, all parts of the plant are equipped with a plenum ventilation system; fresh-air blowers are available and, for emergencies, breathing sets are located in various parts of the plant.

O. ČERNÝ: How much moisture do the humidifiers allow in the CO₂?

J. McCRICKARD: The moisture content of the circuit gas is normally maintained at about 10 ppm by the operation of the humidriers. However, the purity of the gas supplied for the system is so high that it has been found possible to dispense with the use of the humidriers except during the first few weeks of normal operation after fuel charging. Similarly, the activity of the coolant gas is kept at such a low level that the bypass filters and cyclones can be taken out of service, thus allowing a higher rate of gas flow through the core for a given blower power.

I. PAULIČKA: Have operations at Calder Hall had any measurable effect on the surrounding region? I should also like to know how much radioactive waste is discharged from Calder Hall in the course of 24 h and what its specific activity is.

J. McCRICKARD: The effect on the surrounding countryside of operations at these two plants is so small as to be scarcely measurable; in fact, it is rather obscured by the far more significant effect of fall-out from atomic weapons tests. The second part of your question is best answered by considering the effluent discharged from the Chapelcross plant. We are officially authorized to discharge a certain amount of liquid waste - about 20 c/month - from the pond where irradiated fuel is stored. This is not a large amount, but in point of fact we have never exceeded 10% of it. That the discharge of waste can be kept at this low level is largely due to the good integrity of the Magnox fuel-element cladding; there is very little wear and corrosion when the elements are immersed in reasonably clean water.

THE PERFORMANCE OF MAJOR PLANT ITEMS AT CALDER HALL

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Abstract — Résumé — Аннотация — Resumen

THE PERFORMANCE OF MAJOR PLANT ITEMS AT CALDER HALL. For over six years Calder Hall has been operated as a base-load power station and has sustained faults similar to those experienced by conventional power plants. All the faults have occurred in the conventional plant and none in the nuclear reactors themselves. Experience has demonstrated that from a reactor safety viewpoint, reactors and major plant items, together with the Grid connections, must be treated as a whole.

No significant changes have been detected which suggest that the reactor pressure vessel or the graphite moderator will limit plant life. Asymmetric temperature distribution exists round the outlet gas ducts and now limits reactor power. The initial fitting of more thermocouples, strain gauges and provision for visual examination at high stress and temperature areas is recommended. The humidifiers are no longer used during normal operation and consequently a small increase in reactor power has resulted.

Recommendations are made for the frequency of surveys of heat exchangers, main gas ducts and bellow units. A small increase in steam generation has resulted from reducing gas bypassing.

Only minor difficulties have been experienced with the centrifugal type of gas circulators. A small loss in production occurred initially because of commutator flashovers on the highly stressed blower drive motors and generators.

The provision of duplicate stand-by auxiliaries, together with the interconnection of steam and water systems, has been amply justified.

The BCDG equipment has proved extremely satisfactory. Because the channel group sensitivity is higher than necessary, modifications to achieve shorter cycle times at the expense of some sensitivity are being contemplated.

The re-arrangement made to the power supplies for reactor instrumentation and safety circuits is discussed. The grading of alarms and sequence of importance and a "first up" feature are recommended. The two-guard-line system has proved adequate and the need for three or more is questioned. Statistics have shown that the data-logging equipment is extremely reliable.

More standardization of fuel handling equipment is advocated. The use of special flasks and special discharge equipment are time-wasting and should be discouraged. Modifications made to the charge/discharge machines and auxiliary plant are described. Careful handling of fuel during loading is emphasized. Fuel channel television cameras and special grabs have enabled discharge times to be reduced.

Only minor faults have been experienced with the control-rod mechanisms and associated control equipment.

As far as possible plant controls should be centralized and important control centres and items of plant should be protected from accidental damage from external sources. Automatic start-up and paralleling features for the emergency diesel plant are considered unnecessary but the plant could, with advantage, be centralized.

FONCTIONNEMENT DES ÉLÉMENTS PRINCIPAUX DE LA CENTRALE DE CALDER HALL. Depuis plus de six ans, Calder Hall fonctionne comme centrale de base et a subi des défaillances semblables à celles qu'on peut constater dans les centrales électriques classiques. Toutes ces défaillances se sont produites dans la partie classique de la centrale; aucune n'a été relevée dans les réacteurs proprement dits. L'expérience a montré que du point de vue de la sécurité du réacteur, il faut traiter comme un tout les réacteurs et les principaux éléments de la centrale ainsi que les connections avec le réseau.

On n'a constaté aucune altération importante donnant à penser que le caisson du réacteur ou le ralentisseur de graphite limiteront la durée de vie de la centrale. La distribution de la température autour des conduites de sortie du gaz est asymétrique, ce qui limite actuellement la puissance du réacteur. Il y aurait eu intérêt à aménager au début un plus grand nombre de thermocouples, d'instruments de mesure des efforts et de dispositifs permettant un examen visuel des zones soumises à des contraintes et des températures élevées. Les dessiccateurs

ne sont plus employés au cours du fonctionnement normal et il en résulte une légère augmentation de la puissance du réacteur.

Le mémoire formule des recommandations pour la fréquence des inspections des échangeurs de chaleur, des conduites de gaz principal et des soufflets. On a obtenu une légère augmentation de la production de vapeur en réduisant le débit du gaz dans les circuits de dérivation.

Les appareils centrifuges pour la circulation du gaz n'ont causé que des difficultés minimales. Il y a eu, au début, une perte légère de production due à des courts-circuits dans les commutateurs des générateurs et des moteurs de ventilateurs soumis à un effort excessif.

On s'est aperçu que l'on avait eu raison de prévoir des auxiliaires de secours en double et d'interconnecter les circuits de vapeur et d'eau.

L'appareillage de détection de rupture de gaine s'est révélé extrêmement satisfaisant. Comme la sensibilité par groupe de canaux est plus élevée qu'il n'est nécessaire, on envisage des modifications qui raccourciront la durée du cycle au prix d'une certaine perte de sensibilité.

L'auteur examine les mesures que l'on a prises pour aménager l'alimentation électrique de l'appareillage du réacteur et des circuits de sécurité. Il formule des recommandations concernant le système d'alarme. Le système à deux lignes s'est révélé satisfaisant et la nécessité d'en avoir trois ou davantage ne semble guère s'imposer. Les statistiques montrent que les appareils d'enregistrement des données sont d'un fonctionnement très sûr.

L'auteur recommande une standardisation plus poussée des appareils de manutention du combustible. L'emploi de récipients et d'appareils de déchargement spéciaux fait perdre du temps et est déconseillé. L'auteur décrit les modifications apportées aux machines de chargement et de déchargement et aux installations auxiliaires. Il souligne l'importance du soin à apporter à la manipulation du combustible au cours du chargement. L'emploi d'appareils de télévision dans les canaux à combustible, et de pinces spéciales, a permis de réduire les temps de déchargement.

On n'a enregistré que des défaillances mineures dans les mécanismes complexes de commande des barres de réglage et le matériel de contrôle connexe.

Dans toute la mesure du possible le contrôle de l'usine doit être centralisé et les principaux centres de contrôle ainsi que les éléments importants de l'usine doivent être protégés de tout dommage accidentel pouvant provenir de l'extérieur. Pour la génération diesel de secours, il ne semble pas nécessaire de prévoir le démarrage automatique et des circuits doubles, mais une centralisation ne serait pas inutile.

ХАРАКТЕРИСТИКА ОСНОВНЫХ УЗЛОВ УСТАНОВКИ В КОЛДЕР-ХОЛЛЕ. В течение свыше шести лет большая установка в Колдер-Холле эксплуатируется как энергетическая установка с базовой нагрузкой. На ней возникали такие же повреждения, как и на обычных энергетических установках. Все повреждения имели место в обычной установке и ни одного - в самих ядерных реакторах. Опыт показал, что с точки зрения безопасности реактор и основные узлы установки вместе с приводами системы необходимо рассматривать в целом.

Не обнаружено никаких существенных изменений, которые бы давали повод думать о том, что корпус высокого давления реактора или графтовый замедлитель ограничат срок эксплуатации установки. Асимметричное распределение температуры имеет место вокруг выходных газовых трубопроводов, и в настоящее время это ограничивает мощность реактора. Рекомендуется установить вначале дополнительное количество термопар, контрольно-измерительные приборы для определения напряжения и обеспечить визуальное наблюдение за областями высокого напряжения и температуры.

В ходе нормальной эксплуатации осушители больше не применяются, и соответственно в результате этого достигнуто небольшое увеличение мощности реактора.

Даются рекомендации относительно периодичности осмотров теплообменников, основных газовых трубопроводов и нижних узлов. Небольшое увеличение в производстве пара достигнуто в результате уменьшения перепуска газа.

Небольшие трудности возникли лишь при эксплуатации циркуляторов газа центробежного типа. Отмечено небольшое снижение производства вначале в результате кругового искрения коммутаторов при высоком напряжении на моторах вентиляторов и на генераторах.

Оправданной является установка двоярного вспомогательного оборудования вместе с объединением паровой и водной систем.

Оборудование В.С.Д.Г. оказалось в высшей степени удовлетворительным. Ввиду того, что чувствительность группы каналов выше чем необходимо, в настоящее время планируются изменения с целью ускорения циклов за счет некоторой чувствительности.

Обсуждаются мероприятия по переоборудованию системы подачи энергии для приборов реактора и контуров безопасности. Рекомендуются классификация степени опасности и важности и устройство по первому выключению реактора. Выясняется необходимость иметь систему с двумя защитными линиями и устанавливать три или более таких линий. Статистика показала полную надежность оборудования регистрации данных.

Оправдывается необходимость большой стандартизации оборудования по перегрузке топлива. Использование специальных бадлонов и специального оборудования по разгрузке требует много времени, и от этого следует отказаться.

Описываются изменения, внесенные в механизмы по загрузке/разгрузке и во вспомогательную установку. Подчеркивается необходимость тщательной обработки топлива во время загрузки. Применение телевизионных камер в топливных каналах и специальных захватов позволило сократить время разгрузки.

Небольшие неисправности отмечены лишь при эксплуатации сложных механизмов управляющих стержней и соответствующего оборудования по контролю.

Насколько это возможно, основные системы управления установки должны быть централизованы, и основные центры управления и узлы управления установки должны быть защищены от случайного повреждения при облучении внешними источниками. Признаются необходимыми автоматические пусковое и параллельное устройства для аварийной дизельной установки, однако установка могла бы быть централизована с пользой для дела.

RENDIMIENTO DE LOS PRINCIPALES ELEMENTOS DE LA CENTRAL DE CALDER HALL. Desde hace más de seis años, Calder Hall viene funcionando como central para la carga básica, habiendo sufrido averías similares a las que se registran en las centrales clásicas de electricidad. Todas las fallas se han producido en la parte clásica de la central y ninguna en los reactores propiamente dichos. La experiencia ha demostrado que, desde el punto de vista de la seguridad de los reactores, estos mismos y los principales elementos de la central, así como las conexiones con la red, se han de considerar en conjunto.

No se ha observado ningún cambio importante que sugiera que el recipiente de presión del reactor o el moderador de grafito limitarán la vida útil de la central. La distribución de temperatura alrededor de los conductos de salida de gas es asimétrica, lo cual limita actualmente la potencia del reactor. Se ha recomendado montar más pares termoelectrónicos, así como instrumentos para medir deformaciones y dispositivos que permitan observar visualmente zonas de tensión y temperaturas elevadas. Durante el funcionamiento normal, no se emplean ya los desecadores Humidrier y gracias a ello se ha logrado un ligero aumento de la potencia del reactor.

Se formulan algunas recomendaciones respecto de la frecuencia de las inspecciones de los intercambiadores de calor, de los principales conductos de gas y de las juntas de fuelle. Se alcanzó un ligero aumento en la producción de vapor reduciendo el caudal gaseoso en los circuitos de derivación.

Los aparatos centrífugos para la circulación del gas no han ocasionado más que dificultades de poca importancia. Al principio se produjo una ligera pérdida de producción debida a cortocircuitos en los conmutadores de los motores de las bombas y de los generadores, sometidos a esfuerzos excesivos.

Ha quedado plenamente justificada la instalación de motores auxiliares por duplicado y la interconexión de los circuitos de vapor y agua.

Los aparatos para detección de defectos en las vainas han dado resultados altamente satisfactorios. Como la sensibilidad por cada grupo de canales es mayor de la necesaria, se está estudiando la conveniencia de acortar los ciclos a expensas de la sensibilidad.

El autor examina las medidas adoptadas para reordenar el sistema de alimentación eléctrica de los instrumentos del reactor y de los circuitos de seguridad. Formula recomendaciones sobre el sistema de alarma; el de dos líneas demostró ser satisfactorio y no parece imponerse la necesidad de disponer de tres o más líneas. Las estadísticas muestran que el equipo de registro de datos funciona perfectamente.

Aconseja normalizar más el equipo de manipulación del combustible y evitar el empleo de recipientes y de aparatos de descarga especiales que suponen pérdidas de tiempo. El autor describe asimismo las modificaciones introducidas en las máquinas de carga y descarga y en las instalaciones auxiliares. Subraya la importancia de manipular cuidadosamente el combustible durante la carga. La instalación de cámaras de televisión en los canales para el combustible, así como de pinzas especiales, han permitido reducir el tiempo de descarga.

Los complejos mecanismos de las barras de control y el equipo auxiliar sólo han sufrido averías menores.

En lo posible, habría que centralizar los controles de la planta y proteger los centros de control y los elementos más importantes de la central contra daños accidentales que puedan provenir de fuentes externas.

Para el generador diesel de emergencia, no parece necesario prever un arranque automático o circuitos paralelos, si bien una centralización no sería superflua.

1. GENERAL PERFORMANCE

Because this paper highlights faults, difficulties and unusual experiences, it is only right to state at the outset that the performance of Calder Hall has been really outstanding. Although the Station was designed as an off-load discharge, dual-purpose nuclear plant producing plutonium as its primary product, it has been operated virtually as a base-load power station at maximum load day and night, throughout the six-and-a-half years that it has been in operation. During this period the thermal output of the reactors has been increased by a third, and in order to make use of the excess steam produced, five of the eight 21/24 MW turbo-alternators were rebladed to produce 27 MW by November last year and the remaining three machines will be rebladed during the next few months, thus increasing the total electrical generating capacity to 216 MW. Steam is also now supplied to the process plants at Windscale where all the coal-fired boiler plants have been shut down. The generation plant annual load factor based on the enhanced electrical outputs has also progressively risen from 78 to 91% (in 1962, in addition to maintaining an annual load factor of 91.2%, steam equivalent to the burning of 31 000 t of coal was supplied to the Windscale Works).

2. PRINCIPAL LESSONS FROM FAULT INCIDENTS

2.1. *Fault incidents*

The plant has sustained a number of faults such as could be experienced by any conventional power station connected to a national grid system. All the faults have occurred in the conventional plant or grid system, none in the nuclear reactors themselves. Five of these incidents have demonstrated that, because the reactors, heat exchangers, electrical blower drives, turbo-alternators and grid connections are interconnected, the system must be considered as a whole from a reactor safety viewpoint. Further, experience has shown that all potentially hazardous occurrences, even those which do not cause a fault, should be thoroughly investigated for it is surprising how such investigations can bring to light weakness in operating procedures or protective systems.

These incidents, which have already been described by the author [1] and also by MARSHAM and GOODWIN [2], are summarized as follows:

- 2.1.1. Feed-water failure to the heat exchangers, as a result of the tripping out of a transformer supplying the feed pumps and the condenser cooling water pumps during testing, caused the temperature of the return coolant gas to the reactor, and hence the temperature at the bottom of the reactor, to rise 10% before appropriate corrective action was taken by the operators.
- 2.1.2. The disintegration of a turbo-alternator due to overspeeding because of the presence of fine cast-iron shot, inadvertently left in the steam system

after construction, prevented the high-power governor valve and high-power emergency stop-valve from closing, when the alternator was tripped on load.

2.1.3. An internal electrical system voltage disturbance occurred due to the build-up and probably intermittent breakdown of a high resistance film on the contact surfaces of the carbon brushes of the pilot exciter of one of the alternators. This caused the main gas circulators to stop and the operation of the control-rod tripping circuits to be delayed for a short time. A short transient of 20°C was experienced by the fuel elements before the reactor was tripped.

2.1.4. An external grid voltage disturbance caused by lightning strikes on the grid lines about fifty miles from Calder Hall led to a severe and prolonged voltage disturbance. This caused all the induction motors associated with the main Ward Leonard motor drives to slow down past the point of no return and trip. Consequently a short shut-down of all reactors and turbo-generating plant occurred. A somewhat similar incident occurred a year later, but Calder Hall continued operating, although the Station became completely isolated from the main grid system through transmission line faults.

2.1.5. Whilst a reactor was shut down a drop in feed-water pressure occurred as a result of the faulty isolation of feed-water supply in Calder "B" through a human error. This caused the standby feed pumps to start automatically, run without water and eventually seize. Water was then supplied by other pumps in the station through the installed cross-connectors.

2.2. Principal lessons learned

The principal lessons learned from these incidents, and which have now been applied as modifications to the plant, are briefly described below, and should be read vis-à-vis the appropriate foregoing paragraphs.

2.2.1. Although the emergency pumps were started and corrective action taken promptly, subsequent investigations showed the desirability for providing interconnections between the feed-water suction and discharge ranges of the two reactors feeding each turbine house. This modification subsequently proved to be of great value when incident 2.1.5 occurred. Arrangements have also been made whereby feed-water can be made available from other external sources in an emergency for the removal of reactor decay heat in the event of complete failure of all electrical supplies to the feed pumps, or the complete loss of feed-water supply following a serious fracture in the normal feed mains.

2.2.2. Additional steam-circuit cleaning was carried out during the commissioning of the remaining reactors, namely acid cleaning of the heat exchangers and prolonged blowing out of the steam pipework to atmosphere at full pressure and flow and checking for the presence of cast-iron shot by means of etching plates.

The turbine governor and runaway stop valves have also been fitted with hardened nitro-alloy guides. Larger clearances were provided between the spindles and the guides to make them more resistant to hard particles and back seating provided to seal off the gland when the valve was fully open.

The electrical protection system installed for the automatic tripping of the alternator main circuit breaker as a consequence of the tripping out of

a reactor has been disconnected. Experience has shown that there is ample time in which to off-load the alternators associated with the tripped reactor because as the steam pressure falls, the turbines automatically unload themselves.

As damage to pipework in the turbine house or damage to the cooling tower ponds by flying fragments from a disintegrating machine could cause flooding, which could affect essential electrical equipment or the air-cooling system for the main gas circulators, ramps and cuttings have been provided to divert the flow of water from critical plant. For similar reasons duplicate carbon-dioxide supply mains have been installed to each reactor. In order to safeguard personnel and equipment in the reactor control-rooms and carbon-dioxide plant, suitable protection has also been provided at these areas.

2.2.3. The reason for the formation of a high resistance film on the carbon brushes has never been satisfactorily explained, but it is known that a similar phenomenon has occurred elsewhere. To prevent recurrence of this type of fault the frequency of examination of the brushgear of the alternators and exciters was increased.

The protection system for the electrical drive motors for the gas circulators was modified, so that the opening of either circulator motor or generator field breaker of the associated Ward Leonard set would open the 11-kV O.C.B. to the induction drive motor, and also trip the reactor and start the pony motor. Provision has now also been made in the reactor control-room for the emergency starting of the pony motors. This supplements the automatic starting facilities which come into operation on loss of the main drive motor and also the manual starting arrangements installed in the blower houses.

2.2.4. The large grid voltage drop which occurred at Calder Hall was due to the slowness of opening of some grid circuit breakers under fault conditions. Following this incident the speed of operation of these breakers was increased. No deficiencies in the safety circuitry or emergency arrangements were revealed and the modifications referred to in paragraph 2.2.3 worked as intended. The complete shut-down of all four reactors and associated generating plant, together with the severance of all main grid supplies to the Station on account of the transmission line faults, did, however, show up a number of minor deficiencies and the desirability of introducing modifications which are briefly described later.

A somewhat similar incident occurred the following year, which demonstrated that the stability of the induction motor drives for the Ward Leonard generators for the blower drives could be significantly increased by reducing the load on them. On this occasion faults on the grid transmission lines caused Calder to be completely isolated from the Cumberland ring, but the Station continued to supply nearby towns until the 132-kV grid ring connections were restored. As a result of this experience, whenever the Station is informed that there is a very high probability of lightning in the Cumberland area, the speed of the blower motors on one reactor is reduced by about 10% and on the other three by about 4%. The loss in production as a consequence of such blower speed reductions is extremely small and on several occasions this method of increasing stability has proved worthwhile, e.g. during one weekend this last winter the Station success-

fully sustained more than forty grid surges, some of which were quite severe, as a result of flashovers which occurred on the grid insulators due to salt spray deposits left by storm-force winds with gusts up to 90 mph.

A further precaution which is taken to ensure continuity of supplies to important plant at Windscale during extreme weather conditions or in the event of severe overloading of the national grid system, is to disconnect two turbo-alternators from the grid system and to use them as "house sets", supplying essential local loads and Station auxiliaries.

The complete shut-down of all four reactors and associated generating plant, together with the complete severance of all grid supplies to the Station on account of the transmission line faults prompted a number of safety studies and also showed up a number of minor faults, such as:

(a) Although the availability of important reactor and turbine house auxiliaries had been ensured by the provision of standby plant supplied from lead acid storage batteries in the reactors and turbine houses, the complete cessation of power supplies highlighted certain omissions which, in this instance, did not cause any embarrassment, for example, important auxiliaries in the CO₂ storage house and Health Physics air sampling and counting equipment were without electricity supplies. Additional emergency lighting in certain areas was also necessary.

(b) A safety study showed that because very large and prolonged voltage depressions on the AC supply mains will cause the speed of the gas circulators to fall below the point of no return and consequently shut down the reactors, the operation of safety circuits from the AC mains at Calder enhances the overall reactor safety. A proposal that safety circuits should be supplied from a self-contained constant voltage supply to eliminate the risk of spurious reactor trips as a result of mains-borne surges was therefore rejected. The ability of individual safety instruments to withstand short line voltage surges was, however, checked by tests and showed that most safety instruments will withstand a short voltage depression of 30% without tripping.

(c) The method for supplying certain control-room recorders was also considered and it was concluded that they should be fed from battery-driven motor alternators so that in the event of a loss in mains supply, a record would be available of the ensuing temperature and power transients.

(d) A sequence operations recorder with "first-up" indicator for the protective relays would have speeded up post-fault analysis by removing the uncertainty which existed in determining the order in which events occurred.

(e) Whenever new reactors are being commissioned, a test should be carried out to determine the effects on the reactor-blower-turbo-alternator system of a complete loss of electrical power because, in spite of the very careful consideration given during the design of the plant to the provision of emergency power supplies to essential services, or the installation of battery-fed standby plant, important services can be easily overlooked.

(f) There are eight emergency diesel sets at Calder Hall, two on each reactor. It is considered that they could have been, with advantage, centralized away from the reactors. By this arrangement the total numbers could have been reduced, but duplicate feeders to each of the reactors following alternative routes would have been necessary.

Experience has shown that the automatic start-up facilities provided with the emergency diesel sets and auto-paralleling features for working

with the emergency battery are not absolutely necessary, although desirable, for even under fault incidents there should be ample time for one of the plant operators to start one or other of the diesel sets. The emergency battery provided in each reactor has sufficient capacity to supply the reactor emergency auxiliaries for about an hour.

2.2.5. There are six boiler feed-pumps in Calder "A" and also in Calder "B" turbine houses arranged in groups of three at each end of each turbine house. The two groups are interconnected by common suction and discharge pipes, vide incident 2.2.1.

Normally four or five feed pumps are in use in each turbine house depending upon the load, leaving the remaining pump or pumps as standby. In the event of failure of a feed pump, the standby pump or pumps start-up automatically to prevent loss of reactor cooling. The start-up is initiated automatically by a relay which works by the reduction in feed-water pressure.

A total failure in feed-water supply will, however, cause all the running feed pumps to seize, and subsequently the standby pumps also, since they will start-up automatically. Although the reactor decay heat under these circumstances can be removed by using water from the site fire mains (by means of pipework installed following incident 2.2.1), the reactors cannot be restarted until either the feed pumps are repaired or replaced. A considerable loss in revenue would, therefore, result.

In order to prevent such an incident occurring of which incident 2.1.5 is a variant, a pressure switch is fitted on each pump suction immediately after the isolating valve and connected to trip the pump motor in the event of loss of suction to that pump. A pressure switch is also fitted to each common suction manifold and connected so that an alarm will be given at the turbine operating floor on the loss of pressure in the feed-pump suction range. The pressure switches connected to the "alarm units" operate at a higher pressure than those connected to the "trip units", so that an early warning of water-supply failure will be given.

3. PRESSURE CIRCUIT WITHIN THE BIOLOGICAL SHIELD

3.1. *Pressure vessel*

Internal and external direct visual examination of the mild steel pressure vessel has not been possible even when the reactors have been shut down and charged with new fuel because of the high radiations which come from the steel parts.

The impossibility of carrying out such examinations was appreciated when the pressure vessel was being designed and therefore considerable care was taken during fabrication and subsequent stress relief. Material test specimens including welded sections were also loaded into the reactor core to serve as monitoring specimens, and some of these were given accelerated neutron dosages. Tensile and bend tests on the irradiated specimens have not shown any changes in either the plate or weld metal. Impact tests have, however, not shown any significant increase in the brittle/ductile transition temperature. This is not surprising because other specimens which have been subjected to accelerated neutron dosages have only shown a

very small increase in the transition temperature and indicate that a shift of perhaps not more than 20°C is to be expected during twenty years.

In the early days of the plant the temperature of the pressure vessel was kept above 110°C when the reactor was shut down for refuelling because there were no insurmountable difficulties in doing this, even though it was appreciated at the time that this limiting temperature was very much higher than necessary, even making generous allowances for aging and irradiation embrittlement. The minimum pressure-vessel temperature has now been reduced to 80°C for operational convenience but even this is still higher than necessary to safeguard a shut-down reactor pressure-vessel against brittle fracture.

Other material tests have shown that creep is unlikely to cause a failure. Excessive creep will, however, cause movements of the standpipes which are attached to the top dome of the vessel and this could possibly limit the operational life. The distances of all standpipes and main gas ducts are therefore measured regularly from the concrete biological shield. So far no significant evidence of creep has been observed.

To monitor the circuit for any defect the leak rate is measured continuously in operation and any large increase is promptly investigated. If the reason is not readily determined a carbon dioxide balance* is carried out using infra-red gas analysers to check that the increased leakage must be taking place outside the biological shield. Tests have demonstrated that a small leak from the pressure vessel direct into the void would be detected by this method. So far tests of this kind have not revealed any signs of leakages from the pressure vessel and the parts of the pressure circuit which lie within the concrete biological shield, thus confirming the continued integrity of these parts of the pressure circuit.

The upper limit at present placed on the temperature of the top dome and around the cooling ducts for metallurgical reasons, also places a limit to the reactor output with the present design of fuel-element heat-transfer surfaces. Recently the outlet gas temperature was used as a measure of these steel temperatures. Investigations have now shown that variations in temperature distributions can exist around the metal forming the main gas ducts and that the metal temperature can be as much as 10°C above the gas outlet temperature as measured by a centrally located thermocouple in the gas duct.

It has been shown that these variations are related to the distribution of gas outlet temperatures from the reactor core channels and hence to flux distribution. Some reduction in spread of duct skin temperatures can thus be effected by suitable flux shaping. Investigation of this problem is continuing. At present the skin temperature of the outlet gas is being limited to 345°C.

There is no evidence of any damage or deterioration to the pressure vessel, which would suggest that this item will limit the life of the plant. However, methods for inspecting the high-temperature and stress areas are being developed to supplement the information obtained from irradiated

* The total volume of carbon dioxide in the air sucked into the void by the thermal shield ventilation fans is compared with the total volume in the air passing up the shield ventilation chimneys.

material specimens. The application of non-destructive tests such as ultrasonic inspection and X-ray radiography is also being studied.

Some photographs have also been taken of the internal surface of a reactor pressure-vessel directly, Fig. 1, and also by means of a television

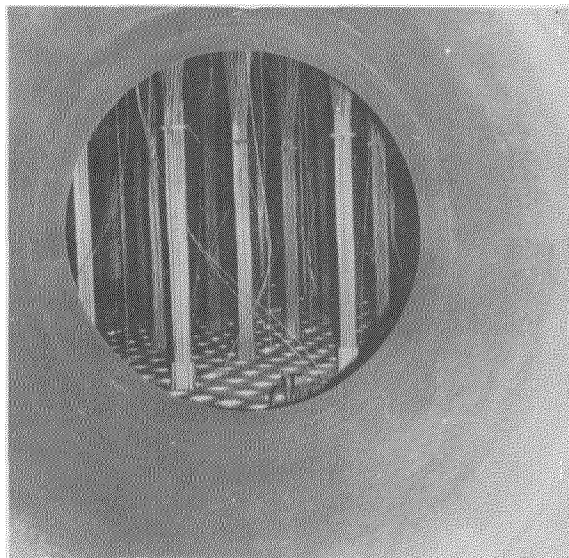


Fig. 1

Photograph taken through a top duct on Reactor 4 during a refuelling shut-down in March 1963, showing top of reactor charge pans; burst-cartridge detection pipes and trailing lead thermocouples

camera. The pictures obtained are sufficiently clear to have shown up a gross defect if one had existed. This technique, however, is of limited value at Calder Hall because the design of the pressure vessel allows only very limited visual examination.

There is a considerable monetary incentive to increase outlet gas temperature even a few degrees, and therefore the installation of additional thermocouples and strain gauges during construction at regions of high temperature and stress with viewing facilities would give the designers and operators a better knowledge of the conditions of the pressure vessel and their cost could be very easily justified.

3. 2. Graphite core

The surfaces of the channel walls in the graphite core have been examined yearly by means of a television camera, but there have not been any signs of cracking or spalling. Due to the irradiation growth of the graphite core bricks normal to the wall of the channel, some closure of the Wigner gaps between the core bricks is to be expected. Regular measurements of these gaps by means of a television camera unit with a rotating

head fitted with a suitable measuring device has shown that the biggest closure which has occurred is only $0.03 \text{ in} \pm 0.010 \text{ in}$ in a gap size 0.25 in , and consequently no serious difficulties are expected to arise. Measurements of Wigner gaps are, however, being continued.

Measurements have also been made once a month of the clearances between the graphite core and the reactor pressure vessel by means of eight probes which can be moved up to the graphite structure and then against a stop on the vessel, but no significant changes in these clearances have been observed.

At each of the major reactor shut-downs, graphite specimens have been trepanned from the reactor core bricks and have been used to determine changes in the physical properties and stored energy. Extrapolation of the results from these specimens will give sufficient advance warning for the need of a thermal anneal should this become necessary.

4. PRESSURE CIRCUIT EXTERNAL TO THE BIOLOGICAL SHIELD

4.1. *Heat exchangers*

The heat exchangers, pressure-circuit main gas ducts and the bellows units on the first reactor were designed on the assumption that after commissioning it would not be possible to enter them, and therefore only the very minimum of manholes required for construction purposes was provided. After the first reactor had been operated for about six months, personnel wearing dust respirators and protective clothing and surgeons' boots showed that it was possible to enter these parts of the pressure circuit under Health Physics' surveillance, and also to work inside them safely, carrying out such operations as welding and grinding. The surface contamination on the steel was of the order of 600-1000 counts/s beta gamma by probe and 300-600 counts/s gamma by swab, but the airborne activity depended on the type of work carried out and could reach $30\,000 \text{ disintegrations/m}^3$ of air per minute. Since then additional manholes for inspection purposes have been provided and modifications carried out in all sixteen of the heat exchangers, and over 50 internal surveys have been made. Health Physics' control has proved adequate and no person has been seriously contaminated.

After over six years of operation there is no evidence in the gas side of corrosion and the internal surfaces are virtually in the same condition now as when the reactors were commissioned, except for some slight discolouration. Further, there had been no increase in the contamination level and experience indicates that a simpler form of protective clothing would be adequate. The only defects which have been found have been a small blister in the bottom dome which was due to surface laminations in the steel plate, some minor cracking in the welds supporting the fairings and some slight distortion of the tube-element support plates.

Only one defect has occurred in the 32 separate steam circuits. Shortly after the last reactor was commissioned a pin hole appeared at a tube bend external to the heat-exchanger vessel. There has not been any satisfactory explanation for this failure in spite of metallurgical examinations and non-destructive tests which have been carried out on similar bends. A number

of tubes have also been examined internally with the aid of an introscope but no deterioration of any kind has been seen in them. The defect is attributed to faulty material.

There has not been any fall-off in the thermal performance of the heat exchangers. About a year ago it was suspected that some of the carbon dioxide gas might be bypassing the tube banks through larger-than-necessary clearances around the various platforms and baffle segments, which are fitted in the stagnant gas spaces. By blanking off these clearances the thermal output of the reactors has been increased by a small amount.

Because the reactor pressure vessel must be kept filled with carbon dioxide at a pressure slightly above atmospheric, about 2-in W.G., whenever it is opened up for refuelling (to keep moisture out of the vessel as it could cause rusting to occur in the mechanisms of the control-rod operating motors), complete or double isolation is provided in all pipe connections which could feed carbon dioxide into the heat exchangers and/or the main gas ducts opened up for examination, in order to ensure safety of personnel. Double isolation is obtained in the main gas ducts by closing the 54-in duct valve and by inflating a special balloon inside the main duct, but some distance away from the valve. The interspace is then vented and kept purged with fresh air. This ensures that air alone can pass into those parts of the pressure circuit where men are employed. Where only single isolation is possible, maintenance personnel are required to wear air-line masks.

Very considerable painting effort is necessary to maintain the outdoor heat exchangers and surrounding structural steelwork in a satisfactory condition. Deterioration of paint and corrosion of exposed steelwork is aggravated at Calder Hall by the fine droplets of water which rain down from the cooling-tower plumes, scrubbing salt from the atmosphere and depositing it on plant near the towers. It would have been more economic and easier for maintenance if the heat exchangers had been initially housed in weather-proof structures.

4. 2. Main gas ducting and bellows units

The additional access manholes which have been provided in the main gas ducts and the new design of removable fairing which has been fitted over the convolutions of the bellows units enable internal surveys to be carried out far more easily than has hitherto been possible. Inspections, which have been made so far, have not revealed any defect, corrosion or deterioration. A few convolutions in three bellows units were found to have closed slightly when compared with other convolutions in the same unit. When they were re-examined two years later and found not to have altered, it was considered that this small closure was present when the units were installed.

4. 3. Main gas circulators

The centrifugal-type gas circulators have regularly been dismantled in order to inspect the impellers which are manufactured of forged steel plates riveted together, Fig. 2. Of the 30 impellers inspected so far, loose rivets and some distortion and rubbing of the inlet labyrinth seal have been found on only one, and are attributed to a bent impeller shaft possibly damaged

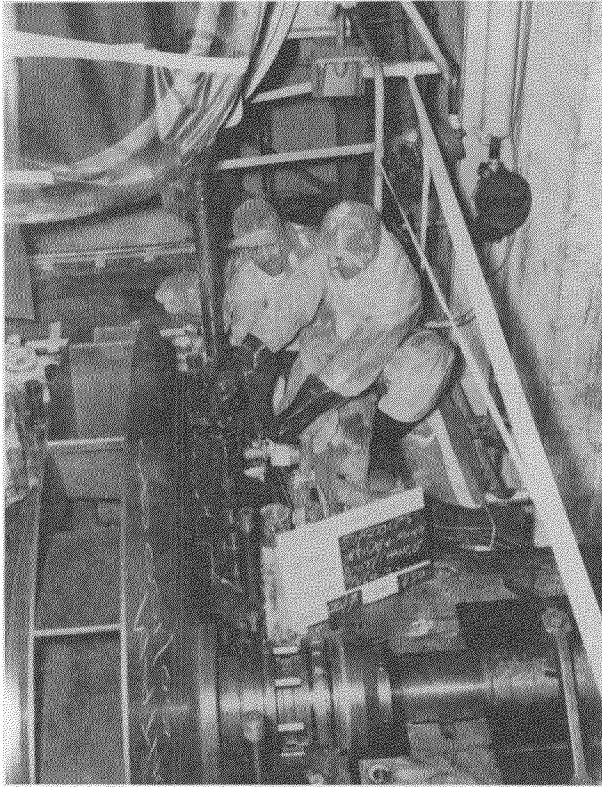


Fig. 2

Gas circulator opened for survey and maintenance

during erection. Experience has shown that the standstill seal of silicone rubber hardens and deteriorates after two or three years in use. The replacement of this seal can only be effected by lifting the top casing and although operational experience has shown that replacement is not essential (for the system can be run without a standstill seal) investigations are being pursued to find a more durable seal. The running seals have been remarkably free from trouble, the temperature of this seal being a good guide of its running performance. On two occasions, however, some escape of gas occurred due to wear of guides and spring components which are associated with the floating portion of the seal.

A large proportion of the carbon-dioxide loss from a reactor is due to leakages from the various gas circulator joints, viz. the O-rings under the diffuser nuts which harden and set after a period at temperature and cease to function as a gas-tight seal. Also leakages appear at the casing joints no matter how carefully the joints are assembled and tested. These leakages can be easily detected using infra-red gas analyser techniques and by the application of soap-water solution. In order to rectify carbon-dioxide leakages of any consequence, the heat exchangers have to be taken out of

service and as this causes a loss in production there is no financial incentive to remake the joints. On some occasions, however, gas leaks have virtually healed themselves during operation, possibly through flexing of the flanges.

The blowers are checked regularly for vibrations and concentricity of the impeller with the inlet eye is checked yearly.

4. 4. Frequency of inspections

To date the gas and steam side of all the heat exchangers have been surveyed every two years and the steam side in addition has been hydraulically tested every four years. The main gas ducts and bellows units have been surveyed at least once every five years.

The continuation of these survey frequencies, especially those for the gas side of heat exchangers and bellows units is under review, for after nearly seven years of operation, no deterioration has been observed, but this is to be expected for the carbon-dioxide coolant is extremely pure and the moisture content is only 10-100 ppm. Further, unlike furnace-fired boilers the tubes are not subjected to erosion by fly ash or corrosive gases. The moisture in the coolant gas is monitored continuously by installed moisture-in-CO₂ instruments and if any rusting did occur on the internal surfaces of the heat exchanger, evidence would be seen in the bypass filters installed in the main gas circuit and also in the filters associated with the burst-cartridge detection gear.

It is considered that the present frequency of gas-side surveys is not justified and that it should be sufficient to enter and examine annually the gas side of only one of the sixteen heat exchangers at Calder Hall.

Entry into the pressure circuit even when free from carbon dioxide involves some hazards to personnel who, wearing dust respirators and protective clothing, are required to descend some 80 ft down each heat exchanger four times and to pass through more than 50 access doors which are fitted in the platforms and the tube bank supports in the dead gas spaces.

On the steam side no deterioration has been observed on the visible water surfaces of the boilers and in the tube banks of the heat exchangers. Consideration has been given to altering the present survey frequency because of the purity of the treated feed water which is continuously sampled and is common to all four heat exchangers, but as steam-side surveys can be carried out without hazard, there is not the same motive to reduce the frequency.

5. INSTRUMENTATION

5. 1. Safety and alarm circuitry

Nuclear measuring and recording instruments and also electronic, electrical and physical instruments associated with the reactor safety circuits have proved extremely reliable. Examination of records for over six years since the reactors were commissioned show that on an average four reactor trips occurred each year per reactor from all causes, and that during this period the number of trips due to faults in instruments amounted to only

0.4 trip per reactor per year. Since the teething troubles were overcome, the reliability of these instruments has improved. For instance, only two trips have occurred in the past four years due to instrumentation. This number is far fewer than the number arising from faults in power supplies, auxiliary circuits associated with electrical switchgear, blower drives and human errors.

The reactor emergency shut-down circuit consists of several relay contacts in series with the two tripping relay coils and are connected across a 110-V AC supply derived from the site mains. This circuit is duplicated and so arranged that any of the nine monitored parameters* can cause the emergency shut-down instruments to open the appropriate series relay contacts in both circuits and shut down the reactor. Most of the shut-down instruments are duplicated and for maintenance purposes are provided with a shorting device, the plug for which is designed to prevent the simultaneous isolation of two identical instruments. During normal operation these shut-down instruments can be checked or replaced by jacking out their respective relay contacts in the appropriate emergency tripping circuit. Excess power and fuel-element temperature tripping instruments are triplicated and work on the "two-out-of-three" principle, i. e. operation of any one unit gives an alarm but if two operate they cause an emergency shut-down. In view of this circuit arrangement these instruments have not been provided with shorting-out facilities but maintenance and replacement of faulty units can be carried out with the reactor at power with only a very small risk of a trip. In fact only one such trip has occurred since commissioning and this was due to a defect in one temperature tripping instrument whilst maintenance work was in progress on another.

The emergency shut-down circuits and their associated tripping relays can only be tested when a reactor is shut down for maintenance because an open circuit on either of these circuits will cause the reactor to trip. During all shut-downs, however, it is an operational safety requirement to keep two control rods out of the core and connected to the safety instrumentation so that they can be inserted in the event of any accidental divergence, e. g. arising from the inadvertent movement of absorbers. To cope with this safety requirement during shut-downs when the main protection circuits are being maintained, a portable safety circuit is used to hold the two safety rods out of the core.

This practice of using only two guard lines and a portable safety circuit system for maintenance during major shut-downs for refuelling has proved entirely satisfactory at Calder Hall where the reactors have been shut down once a year for statutory surveys of heat exchangers etc. and for refuelling.

* The nine parameters which will cause immediate shut-down if unsafe levels are reached are:

- (1) High neutron flux at power;
- (2) High neutron flux during start-up;
- (3) Excessive rate of change of neutron flux at low power;
- (4) High neutron flux in the sub-critical condition;
- (5) High fuel-element temperature;
- (6) Loss of coolant pressure;
- (7) Excessive coolant pressure;
- (8) Coolant flow unbalanced between circuits (for circulator protection);
- (9) Loss of power supplies to coolant circulator motors.

The use of three guard lines arranged in the two-out-of-three principle which enables maintenance to be carried out on one of the guard lines whilst the reactor is at power was considered and rejected as an unnecessary elaboration contributing little to reactor safety. This system may, however, be justified where reactors are required to run continuously for over a year and where spurious trips are to be avoided at all cost. The use of the two-out-of-three principle for electronic shut-down instruments has, however, avoided a number of spurious trips and has been justified. This requirement may, however, not be necessary with the introduction of reliable, fully transistorized, instruments.

The original guard-line circuits had both trip relay coils connected at the earthed end of each series chain. This now has been modified by putting one of the two trip relay coils at each end of the series chain because such an arrangement eliminates the possibility of a "live-line" fault within the series chain holding in the safety trip circuit, even though series contacts within the guard line have been opened by one or other of the shut-down devices. The possibility of such a fault is, however, extremely remote and has never been experienced at Calder Hall.

The alarm system consisting of a number of flag relays with a single audible bell alarm proved to be unsuitable, primarily because it was difficult to pick out a new alarm when a number of other alarms were showing, and also because the system was subjected to spurious operations and difficult to maintain. This system of flag relays has now been replaced by a winking light system which has proved to be reliable. The new system could, however, be improved still further, e.g. alarms both visual and audible could be graded in degree of importance and a device could be incorporated to indicate the sequence of operation of the more important alarms.

5.2. Ion chambers and battery supplies

The 600-V dry batteries used to polarize the ion chambers have proved reliable and the ion chambers themselves have given continuous service without fault or deterioration. In the very early days it was found that random changes in the internal impedance of individual cells in the high-tension batteries during aging caused sudden and short random changes in the voltage applied to the ion chambers. This caused the sensitive period meters to behave erratically but did not affect other nuclear instrumentation, namely shut-down amplifiers, to the same degree. Since then the dry batteries are given a short soak test and measurements are made of their internal resistance. Batteries with higher internal resistances are not used for period meters but experience has shown that they can be used satisfactorily with other ion-chamber instruments. This procedure has completely eliminated the erratic behaviour of period meters from this cause. All the batteries are therefore checked every six months and the batteries supplying the period meters in particular are checked before major refuelling.

5.3. Temperature scanning and on-line data reduction equipment

Data loggers, or more precisely temperature scanning and alarm monitoring equipments, have been installed on all reactors to handle the large

number of temperature measurements required, namely coolant gas, fuel, graphite, concrete, pressure-vessel steel etc. Some form of data reduction is obviously desirable to reduce the volume of information presented to the operators. In the past this information has had to be collected on to log sheets from multipoint recorders and indicators. Experience with the data logging equipment has been good and suggests that the use of this type of equipment could, with advantage, be extended to monitor other limited conditions such as burst-cartridge detection-gear signals, coolant pressure etc. and also to provide automatic trip facilities.

The basic specification for three of these equipments is as follows: (The equipment on the fourth reactor, in addition to data logging, is also provided with limited facilities which can be used for reactor operational computations such as heat balances, flux plotting, temperature survey analyses etc.)

- (a) 450 thermocouple inputs are scanned sequentially at 1.2 s per point, i. e. it requires approximately 10 min for a complete scan;
- (b) The measuring range is 0-600°C with an accuracy of $\pm \frac{1}{2}\%$ of the full scale range, namely $\pm 3^\circ\text{C}$;
- (c) 13 high-level and one low-level alarm references are provided with setting range of 0-600°C in 5° steps;
- (d) As each point is scanned the thermocouple is first checked for continuity and then the machine compares the temperature measurement made with the alarm level (set point). If the temperature is above the alarm level, visual and audible alarms are given and the point number, temperature and time are printed by the excess alarm strip printer. The machine with its electrical typewriter can also provide a complete log of temperatures at each input point either on demand or automatically at pre-set times.

Important units in the equipments such as measuring channels, output typewriters etc. are duplicated and this has enabled high availabilities to be achieved following initial commissioning. Except for the measuring amplifiers the equipments are fully transistorized.

Most of the faults experienced have been mechanical in nature (breaks in wires, connectors, bad contacts) rather than electrical or electronic. A summary of breakdowns on the three identical units is shown in Table I.

The proven reliability of these installed data loggers makes the use of solid state devices look most attractive for nuclear instrumentation and reactor safety circuitry.

5.4. Burst-cartridge detection gear

The burst-cartridge detection gear which is based on the short-lived, double-beta decay fission-product precipitation system has proved to be extremely reliable. It has detected signals which have risen very slowly ("cavitation"-type fuel failures giving doubling times varying between 10 and 100 d) and the few that have risen extremely rapidly ("fast" failures giving doubling times varying between minutes and an hour). Experience, however, has shown that this detection system is insensitive to fuel failures which have a long leak path, but operational procedures known as "pressure steps" and "blower cycling" have been developed which enable the burst-cartridge

TABLE 1

BREAKDOWN SUMMARY

Reactor No.	1	2	3
Availability (%)	98.7	99.6	99.8
Breakdown cause expressed as percentage of total numbers			
1. Wiring, plugs, connectors etc.	24.0	14.8	6.3
2. Typewriters	16.7	29.6	23.4
3. Line printers	7.4	3.3	4.3
4. Logic circuits	18.5	19.6	16.0
5. Relays and uniselectors	11.5	16.4	20.2
6. Measuring circuits	18.5	12.2	12.8
7. Miscellaneous	3.4	4.1	17.0

detection gear to detect such incipient failures by squeezing fission products out through any minute hole which might exist in the can wall or end cap weld.

The equipment is complex and a number of operations are required to take place in sequence within the 27-s cycle-time of a scan. These operations, namely wire movement, valve movement, ratemeter counting, reset and recorder print have to be repeated continuously and reliably. A complicated cam relay and switching system is fitted for controlling and timing the sequence and interlocking of these operations. Most of the day-to-day faults occur in these switches and relays, but are easily rectified by cleaning and/or physical adjustment of the contact surfaces. Because there are 44 separate units which are required to work day and night continuously at the Station, an almost continuous maintenance effort is required.

Experience has shown that certain changes and additions can be made to the existing equipment. Some comments on the existing equipment are as follows:

(a) At present groups of four fuel channels are scanned at 27-min intervals. This, however, provides too large a signal at too low a scan frequency. The gas from each of the main ducts is examined every $4\frac{1}{2}$ min but as the sample represents approximately one-quarter of the total number of fuel channels, it provides too small a signal at a high scanning frequency. The system is too insensitive to give early warning of individual channel failures.

It would therefore seem reasonable to sacrifice some sensitivity by increasing the channel groupings to say twelve, to achieve a shorter cycle time of 10 min. Any modification designed to achieve this should still enable the gas from any single suspect channel to be examined continuously. Designs to achieve these requirements have been prepared but are costly.

(b) Provision of testing facilities which will permit overall tests to be carried

out on each of the precipitator systems using the same source of fission products preferably from the reactor. This will allow comparative tests to be made on the precipitators with their associated pipework, precipitation and counting systems. Experience has shown that due to the layout and design of the pipework from the reactor to the precipitators, variations do occur in the signals as measured on different precipitators.

(c) It is desirable that the burst-cartridge detection gear should be incorporated into the reactor protection circuits so that in the event of a severe fuel-element failure the reactor will be shut down automatically. The precipitators and counting equipment have not proved sufficiently stable for this modification.

(d) The 54-point graphical form of signal presentation requires considerable attention and maintenance.

(e) In the present state of the art, there is value in designing the system to give flexibility in channel grouping and in scanning frequency.

6. FUEL-HANDLING EQUIPMENT

6.1. *General comments*

During the first fuel discharges difficulties were encountered with the fuel-handling equipment and consequently the loading and unloading of a fuel charge of 10 000 fuel elements in a reactor became excessively long, namely ten weeks.

Not long after the first reactors were raised to power it was found that the burst-cartridge detection gear did not give early indication of failures in fuel elements which were characterized by long leak paths and that such failures gave rise to sudden and rapid increase in the burst-cartridge detection-gear signal. Post-irradiation examination of the first few failures of this type revealed that they were due to leak paths through the end welds and that they had been the result of either a manufacturing defect or damage during handling or loading into the reactor.

Consequently, in the early days of the plant much attention was directed to:

(a) Improving the reliability of the fuel-handling equipment and devising additional ancillary equipment in order to reduce the discharge and charge times.

(b) Handling methods and equipment with a view to eliminating the possibility of damage to the fuel-element tops and welds during loading operations.

Some of these modifications and improvements are mentioned below.

6.2. *Charge machine*

(a) It was realized at a very early stage that direct viewing of fuel elements whilst they were being removed from the basket in the charge machine would assist in preventing damage to the elements during the grabbing and lowering operations, and would also allow the fuel elements to be checked immediately before loading into the reactor. An armour plate-glass was therefore fitted into the machine door.

- (b) The holes in the machine through which the fuel elements are lowered were radiused to prevent damage occurring to the tops and end welds.
- (c) The grabs and grab connections were modified to enable the grabs to be quickly changed when damaged and also to give higher grab availability.
- (d) Because it was discovered that certain obscure faults could allow fuel elements to be released from a grab during loading or unloading without the knowledge or intervention of the operator, the design of the grab and associated electrical equipment was modified.
- (e) The grab cable was sometimes damaged due to piling up of the cable on the reeling drum. Modified layering and tensioning devices were fitted to overcome this defect on both charge and discharge machines.

6. 3. Discharge machine

- (a) It was found desirable to have a viewing facility to examine the positioning of discharged fuel elements in the holding basket. An introscope was therefore fitted on each discharge machine and incidentally has proved of some use in viewing and photographing suspected faulty fuel. It was found that the lenses discoloured very quickly and therefore they are now made from cerium stabilized glass.
- (b) The hydraulic system used for deflecting the fuel elements into their respective positions in the discharge basket gave considerable trouble but these teething troubles were gradually eliminated.
- (c) It was soon evident that not enough attention had been given to the siting of the fuel-handling instruments and alarm signals, e. g. it was necessary to reposition the "grab-open" and "fuel-element-in-grab" indicators so that both were within the field of view of the machine operator, because it was found that some fuel elements had been mal-handled. Greater study should therefore be directed to the grouping of important instruments.
- (d) On occasions the discharge machine has not been able to remove a fuel element from the reactor either because the lifting top had been damaged during loading or because the top had unscrewed itself by the aerodynamic forces of the coolant gas and had become detached. Under such circumstances the television camera provided for viewing fuel channels has proved extremely useful in determining the reasons for failure to lift. It was soon realized that the usefulness of the camera would be greatly extended by the addition of different forms of remotely controlled grabs and a whole series of these was therefore devised, Fig. 3. The dwell time of the television camera inside the reactor was, however, restricted to about 15 min because the picture quality deteriorated due to heating up of the camera which was in an ambient temperature of about 140°C. Although the camera was cooled by an external supply of carbon-dioxide gas, the latter, on its way through the reactor charge tubes to the camera, got warm. A new design of television camera cable has now been introduced which permits more effective cooling of the camera.

6. 4. Ancillary equipment

Generally the ancillary equipment associated with fuel handling, namely hydraulic jack for the discharge machine door, the discharge machine well-

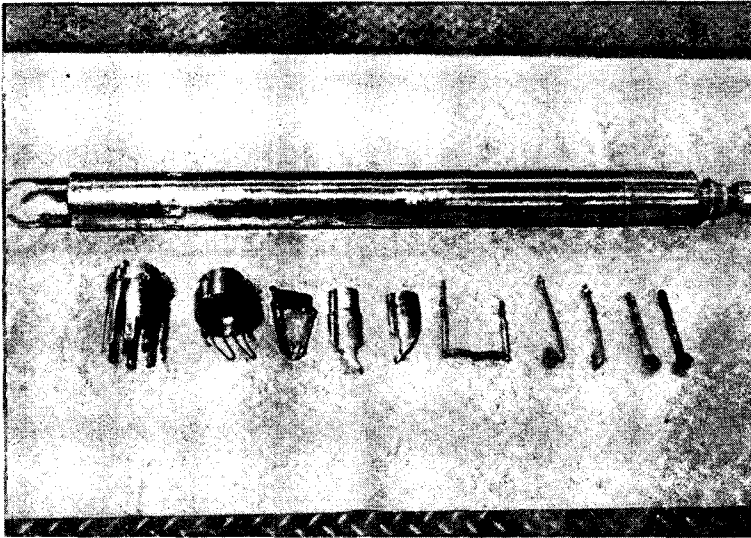


Fig. 3

Channel television camera with an assortment of grab fittings

hoist and coffin bogie, have worked satisfactorily after the initial teething troubles were overcome and minor modifications introduced, e.g. the indicators provided for the operator of the discharge-machine shield-door jack were inadequate and consequently on a couple of occasions the hydraulic cylinder for the jack was fractured. A new foolproof interlocking system had therefore to be devised. Also a radiation control interlock had to be fitted to the doors giving entry into the well-hoist chamber to ensure that operators could not enter the chamber if, for any reason, irradiated fuel elements were exposed.

The practice of using a number of special flasks for removing and transporting certain irradiated fuel elements wasted valuable reactor refuelling time. Normally the irradiated fuel elements are despatched to the cooling ponds in standard water-filled flasks - if, however, a suspect or experimental fuel element was required to be examined in the "caves", it has had to be loaded into a special dry handling flask which, incidentally, was also required for handling thermocoupled fuel elements. With the development of a thermocouple with a breakable weak link, it now appears that a minimum of two types of flasks will suffice - namely standard fuel-element flask and a special flask for handling graphite specimens.

The provision of a number of special motorized vehicles for coffin transportation is not necessary. Universal trailers which could be towed by general purpose tractors would have been adequate.

The foregoing describes only some of the modifications that were made to the fuel-handling equipment. The effect of these improvements has been to reduce shut-down time required for discharging and charging a reactor with a full charge of 10 000 fuel elements from ten weeks to a little under three weeks, and the latter allows time for loading experimental fuels with

additional thermocouples, steel and graphite specimens and television inspection of channel walls and Wigner gaps between the core bricks.

7. AUXILIARY PLANT ITEMS

7.1. *Control-rod system*

The control-rod system has performed satisfactorily under normal and fault conditions. Although this equipment is designed to fail safe and hence can easily be inadvertently tripped, there have been only a few spurious trips. Under operating conditions no control rod has ever failed to drop into the reactor when it tripped either automatically or manually. The complex frequency convertor system has presented some difficulties to plant operators who, for maintenance requirements, are required on a routine basis to change over the low-frequency supplies to the control rods from one frequency convertor set to the other without tripping the reactor. A simpler system of control is desirable.

In the very early days of the plant two control rods failed to drop into the reactor core on a shut-down reactor whilst it was undergoing pre-start-up trials. One of these failures was due to corrosion which had occurred in the ball bearings in the control-rod actuating mechanism whilst the shut-down reactor was full of air, consequent upon the refuelling operations which had been in progress. The other was due to rubbing between an eddy-current braking disc and the pole pieces of the magnet because a holding screw had worked loose.

Even though subsequent tests showed that the control rod suspended from these two mechanisms could have been lowered into the reactor core using the sine potentiometers to drive the control-rod motors, the following operational procedures were introduced:

- (a) During refuelling operations when the reactor is at times opened to the atmosphere, it is kept filled with CO₂ at a positive head of 2-in W.G, and the moisture content measured at the top dome is kept below 1000 ppm.
- (b) During maintenance of the mechanisms a check is made to ensure that the bearing races are not fitted too tightly on to the shaft for tests have shown that even if the ball bearings have seized due to corrosion build-up, the rods will still fall under gravity alone with the shaft rotating relative to the roller bearing inner race.
- (c) The frequency for maintaining control-rod mechanisms in the workshops was reviewed and it was decided that at least 50% of the mechanisms in a reactor should be overhauled each year.
- (d) The friction resistance is measured and a velocity diagram recorded of the fall of the control rod for all mechanisms both in the workshops after maintenance overhaul and also in the reactor building before their installation in the reactor itself.

Further, before a reactor is started up the control-rod mechanisms are again carefully tested in the reactor by checking the time for insertion.

Since these procedures were introduced no further troubles have been experienced.

The control-rod mechanisms operate in an extremely dry CO_2 atmosphere and are subjected to some radiations from the reactor core from which they are shielded by a concrete plug. Experience has shown that the molybdenum disulphide dry lubricant used in the ball bearings and gear trains tends to harden and slow down the fall of the control rods. Experiments with a high-temperature grease, 701 APL, have proved successful in and out of the reactor and consequently the use of this type of grease is now being extended to all control-rod mechanisms.

7.2. Feed-water regulators

The single-element, float-operated, automatic feed-water level regulators (direct and also servo-acting) installed in the high-power and low-power steam systems have proved to be quite satisfactory when the boilers are steaming above minimum load. They tend, however, to be slightly sluggish in following load changes and do not maintain a precise level. Their dead band width of a few inches is acceptable and with only occasional adjustment and little maintenance they work extremely well under the steady load conditions which prevail. At very low loads their performance is not so good.

These regulators cannot cope with large load changes such as occur when heat exchangers are being taken out and brought back into circuit, but trials have shown that no other regulator is able to cope with this situation satisfactorily except a three-element type (water level, steam and water flow).

When large load reductions of 50% or more are anticipated, manual control is found to be adequate, i.e. the feed-water flow is adjusted manually so that the regulator is nearly fully open and then the operator controls the flow to maintain the water level slightly below half glass.

Unexpected large load reductions cannot be satisfactorily handled by simple automatic regulators and are handled only inadequately by manual operation. Until the new steam-flow conditions are established, the operator controls by "overshoot methods". This involves heavy blowdowns because steam-flow and feed-flow metering has not been installed at the feed-control station.

Trials have been carried out using three different types of two-element feed regulators and also one type of three-element regulator. The two-element regulators behaved adequately except at low loads. The three-element type behaved very much better at low load conditions, but the additional expense was not considered to be justified at Calder Hall where 32 regulators would be needed, and because the reactors are normally operated under base load conditions.

One advantage of the servo-acting type of feed regulator is that the mechanism need not be at drum level where higher radiation fields exist. This is an important consideration when maintenance has to be carried out on a reactor at load.

7.3. Gas-circulator drives

Each of the four centrifugal-type gas circulators installed on each reactor circulates coolant gas at about a quarter of one ton per second and is

driven by a variable speed 2200-HP, 600-V, 940-rpm DC motor which is supplied from an associated 1730-kW, 600-V, 740-rpm Ward Leonard DC variable-voltage generator. The Ward Leonard generator is driven by an 11-kV direct on-line start squirrel-cage induction motor. The main drive motors and Ward Leonard generators are fitted with commutating poles and pole-face compensating windings.

The speed of the gas-circulator main-drive motors can be controlled individually or alternatively collectively from the reactor control room.

An overcurrent DC relay is fitted between the Ward Leonard generator armature and main-drive motor armature which, in the event of severe overcurrent between the motor and generator, trips out the 11-kV circuit breaker for the induction motor. Intertripping is provided between the four gas-circulator main-drive motors to eliminate the possibility of backdriving, and protection is so arranged that the failure of any one of the main gas-circulator drive motors will trip the reactor and automatically start the auxiliary pony motors. A single flashover on any machine can therefore trip a reactor, and cause approximately one day's loss of production.

Although the machines met the usual guarantees, troubles were initially experienced due to flashovers, burn marks on commutators, random and sometimes rapid wear of the carbon brushes, copper picking* and commutator glazing.

The frequent flashovers which used to occur in the early days on these "limit motors and generators" (maximum power speed product $\text{kW} \times \text{rpm} = 1.6 \times 10^6$) are ascribed to copper drag** and now have been virtually eliminated by the selection of a carbon brush grade which produces little or no copper drag and by careful maintenance. Experience has shown that these powerful "limit" DC machines can now be operated continuously at 600 V (20 V per segment average) with outages determined primarily by the need to replace worn brushes, provided that they are carefully maintained at periodic intervals and copper drag is removed when seen. The total loss in production on account of these DC motors and generators has now been reduced to about 0.5%. The early difficulties experienced with this form of blower drive have been fully described by the author[3].

8. MISCELLANEOUS TOPICS

8.1. Oil in reactors

The first of the four occasions on which oil entered the reactors occurred shortly after Reactor 1 was loaded with fuel but before it went on power for the first time. On this occasion, because of a human error during a preliminary test on the oil system for the gas-circulator running seal, the bypass valve across the oil-pressure regulator in one of the circuits was inadvertently opened and consequently the increased oil/gas-pressure differential at the running seal forced about 140 gal of oil into the gas circuit. The oil

* Copper picking - plating of parts of the brush contact surface with fine copper particles.

** Copper drag - the formation of globules of copper on commutator segment edges or the pulling over of the surface of the commutator segments sometimes with the formation of easily detachable whiskers of copper.

was then distributed throughout the reactor core and the gas ducting connected to the affected gas circulator. The incident was first noticed by the large reduction in oil tank level and later when the reactor was opened up, oil could be seen lying on the charge pans in large quantities. Considerable amounts of oil were removed by the filters in the affected gas circuit and by mopping up oil lying on the charge pans, bottom gas fairings and deflectors and also in the gas ducts. After this incident a flow restrictor was fitted into the oil-pressure circulator bypass circuit. The reactor was then put on load and operated, and no deleterious effects have since been observed; this has given more confidence in the use of oils and greases as lubricants for reactor components.

On the other three occasions oil entered the gas circuit only in very small quantities and occurred during shut-downs as a result of the vacuum which was created in a gas circuit whilst the gas circulator was running at low speed in order to clear a heat-exchanger circuit of CO_2 preparatory to the entry of maintenance personnel. Under these conditions the oil-pressure regulating valve on the running-seal oil-supply circuit was not able to maintain the necessary oil/gas-pressure differential and consequently oil was sucked into the gas side of the main circulators, the drainage system from which became overloaded and choked. In all these incidents, which were not noticed until after the reactor had been started up, the blower speeds were not increased at full gas pressure until the temperature of the fuel had been raised to 350°C , but only small quantities of oil entered the reactor core. The presence of oil in the gas circuit was found by chemical tests which are referred to later. Following these incidents the use of the exhauster for purging the gas ducts and heat-exchanger circuits for maintenance purposes is allowed only when the circulator is on standstill seal and the oil pumps are stopped.

The following tests were instituted after the first incident to determine if oil has inadvertently been introduced into the reactor.

(a) A sample of the coolant gas is bubbled through carbon tetrachloride which is then examined for the presence of oil by infra-red spectrometry. This test is carried out shortly after a reactor start-up following maintenance, and also weekly. This method will detect oil contamination in the test sample as low as 5 ppm, but as this test sample may not be truly representative of the bulk gas due to sampling difficulties, the values obtained can only be used quantitatively, i. e. any abnormal rise above the background figure is indicative of oil ingress.

(b) Infra-red gas analyses are carried out continuously using the installed CO-in- CO_2 meters - any sharp increase in carbon monoxide will indicate the presence of oil breakdown products or CO from other sources.

(c) A daily check is made to determine the residual gas content in the coolant by passing a measured quantity through a potassium hydroxide solution and measuring the unabsorbed portion. The value of residual gas normally follows a set pattern on each reactor, being dependent upon the amount of CO_2 leakage from it. Any abnormal increases are indications of oil breakdown. Experience may, however, show that the reliability of the instruments referred to in paragraph (b) will permit a relaxation in the frequency of carrying out this daily check.

(d) At infrequent intervals, and also if any unusual results are obtained

from the foregoing tests, the coolant gas is examined by a mass spectrometer to determine the methane and ethane constituents. A build-up of methane above background, viz. 15-20 ppm and the appearance of ethane greater than about 5 ppm would indicate the presence of oil-breakdown products.

8.2. Moisture in reactors

No sign of corrosion has been observed in the heat exchangers and gas circuits which could be assigned to water leaks.

Following the discharge of fuel during which the reactor is opened up to atmosphere, the subsequent moisture levels at the start-up of the reactor range from 100-200 ppm w/w, but within a few weeks, fall to 10 ppm w/w as determined by chemical tests - the only exception is Reactor 1 where the moisture level has not fallen below 25 ppm from the time when large quantities of oil entered the reactor.

Two humidriers are provided on a reactor, each of which is connected by pipework to the inlet and outlet of a blower so that at full gas pressure and maximum blower speed, approximately 2% of the mass flow through the blower passes through the humidrier.

In the first four years the humidriers were kept working continuously but as they are only designed to remove moisture efficiently down to 100 ppm (and this has been confirmed by tests) they are now taken out of circuit after a reactor has been in operation for a few days. It has been found that although the humidriers are not in use, the moisture level continues to fall to 10 ppm.

The moisture content in the coolant gas in the reactor is monitored by infra-red gas analysers which will respond to large increases in moisture in CO₂ but are insensitive to small changes, viz. below 50 ppm. Daily analytical checks of moisture content are made by a gravimetric procedure which is also used to check the infra-red gas analyser instrumentation. Improvements which have been made in the moisture-in-CO₂ instrumentation in the last two years have shown that it is safe to operate with the humidriers out of service as any large increases would immediately be recorded. Thus 2% of the total mass flow which previously was passed through the humidriers is now passed through the reactor and consequently the thermal power has been slightly increased.

Maintenance of the humidriers has to be carried out under Health Physics' surveillance as the alumina concentrates tritiated water.

9. CONCLUSION

All the early problems and faults occurred in conventional plant or were of an engineering nature, primarily because the plant was either the first of its kind or because it was designed to run at maximum load day and night continuously for month after month without a shut-down. Except for teething troubles experienced with the heat-exchanger recirculating pump systems and some minor plant faults, the reliability and performance of the turbo-alternators, turbine house auxiliaries, electrical switchgear etc. have been

excellent. The reactor itself has proved an extremely reliable heat source and annual electrical load factors of 90.0% have been maintained.

Although the Station contains many more engineering plant items than in a conventional steam power station, it has not proved any more difficult to run, but plant operators need additional training and technical assistance. The radiation shielding provided has proved both effective and adequate. Special precautions are, however, necessary when maintaining plant in direct contact with the coolant gas, but these have not proved restrictive. The average radiation exposure of Calder Hall staff has only been 1.5 rem/yr, and no one has received more than 5 rem/yr.

Up to recently the Station has been operated primarily as a plutonium producer; in consequence almost complete charges of fuel have had to be discharged annually. In spite of this, extremely high electrical load factors have been achieved. The Station in future will be operated primarily as an electrical power station and therefore only comparatively small quantities of fuel will have to be discharged annually. This means that the shut-down times of the reactors will be less than a fortnight each year and will be dictated by the time required for statutory examination of the heat exchangers and maintenance of conventional plant items, and not by refuelling.

Electrical plant annual utilization factors of the order of 94% based on the enhanced electrical output are anticipated.

ACKNOWLEDGEMENT

The high availability and increased output obtained from this prototype nuclear power station has been due to the application of technical knowledge and skill of the design, construction, operation and maintenance staffs associated with Calder Hall over many years, and the author wishes to acknowledge their work which has produced such a successful power station.

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DISCUSSION

W. D. J. GESTEL: I gather that some material test specimens have been put into the reactors in connection with pressure-vessel inspection. Are these test specimens subject to the same stresses as the reactor vessel plates themselves, because if they are not, surely they cannot be taken as representative of the condition of the pressure-vessel plates?

E. L. DESBRUSLAIS: Thus far, no stressed steel specimens have been tested in the Calder Hall reactors, although this is to be done very soon.

Stressed specimens have been loaded into Chapelcross reactors, however. They will be analysed in due course.

B. SAITCEVSKY: Have the filters in the CO₂ circuit been replaced? How high is the content of dust in the CO₂, and what is its composition?

E. L. DESBRUSLAIS: The filters in the main CO₂ circuit have not been changed. In the early days we used to take samples from the filters and analyse them; the quantity of dust was so small, however, and its composition so consistent that I recently decided not to keep the filters in the circuit during normal operation but bring them into use only if abnormal conditions were suspected. So far we have not had occasion to put them back in the circuit. I cannot tell you exactly what the composition of the dust is but it consists mainly of graphite and iron.

M. R. SRINIVASAN: Mr. Desbruslais has given us a very lucid account of the experience gained with various plant components at Calder Hall. I should like to know what improvements of design could be made in the light of this experience - particularly in the safety systems and instrumentation - with a view to further simplifying this type of plant and reducing both capital costs and the cost of power production per kilowatt hour.

E. L. DESBRUSLAIS: This matter has already been informally considered. One of the reasons for the success of Calder Hall is that the plant is already simple in design and can be operated with no greater difficulty than conventional power plants. It has no complicated servo loops, no on-load charge and discharge machinery, and the maintenance procedures are essentially simple. There are a few changes which I would advocate, however, if Calder Hall were to be redesigned: replacement of the existing DC blower-drive system, which is very expensive; a new design for the reactor core which would allow higher output with a higher inlet-gas temperature; a redesigned burst-cartridge detection system, as mentioned in my paper; the use of solid-state devices instead of the relays and contactors now employed in the safety circuitry (this is in itself really quite simple and would not admit of much further rationalization); and the centralization of plant controls in order to reduce the number of operators required.

I do not know what the cost per kilowatt installed of a redesigned Calder Hall would be, but I can refer your query to the consortium of manufacturers.

QUATRE ANS DE FONCTIONNEMENT DES RÉACTEURS G2 ET G3.

F. CONTE

CENTRE D'ÉTUDES NUCLÉAIRES DE MARCOULE, FRANCE

Abstract — Résumé — Аннотация — Resumen

THE G2 AND G3 REACTORS AFTER FOUR YEARS OF OPERATION. The G2 and G3 gas-graphite reactors at Marcoule have been in operation for four and three years respectively. This paper reviews the main problems raised by the operation of the reactors from the standpoint of technology, safety, performance and organization. The solutions adopted and the results obtained are presented here as a contribution to the study of future power reactor programmes.

QUATRE ANS DE FONCTIONNEMENT DES RÉACTEURS G2 ET G3. Les réacteurs graphite-gaz G2 et G3 de Marcoule fonctionnent respectivement depuis 4 et 3 ans. La communication présentée passe en revue les principaux problèmes soulevés par l'exploitation de ces réacteurs au point de vue technologie, sécurité, performances et organisation. L'auteur expose les solutions adoptées et les résultats obtenus afin de contribuer à l'étude de programmes futurs de réacteurs de puissance.

ЧЕТЫРЕ ГОДА РАБОТЫ РЕАКТОРОВ G2 И G3. Реакторы с графитовым замедлителем и газовым теплоносителем G2 и G3 в Маркуле работают соответственно 4 и 3 года. В предлагаемом сообщении дается обзор основных проблем, вызванных эксплуатацией этих реакторов, с точки зрения технологии, безопасности, эксплуатации и организации. Принятые решения и полученные результаты используются для более глубокой разработки будущих программ энергетических реакторов.

CUATRO AÑOS DE FUNCIONAMIENTO DE LOS REACTORES G2 Y G3. Los reactores de grafito y anhídrido carbónico G2 y G3 de Marcoule funcionan desde hace cuatro y tres años respectivamente. La memoria revisa los principales problemas planteados por su explotación desde los puntos de vista tecnológico, de la seguridad, del rendimiento y de la organización. Informa sobre las soluciones adoptadas y sobre los resultados obtenidos para contribuir al estudio de programas futuros de reactores de potencia.

I. INTRODUCTION

Les réacteurs G 2/G 3 du Centre de Marcoule sont du type uranium naturel - graphite - gaz. Ces piles sont horizontales, possèdent 1200 canaux chargés de 120 t d'uranium. La pression de service est de 15 kg/cm², le gaz caloporteur est l'anhydrique carbonique. Le soufflage est assuré par des turbosoufflantes. Une description très complète de ce réacteur a fait l'objet d'une publication [1].

La puissance nominale du projet était de 200 MW. Une centrale de récupération d'énergie est associée à ces réacteurs et débite sur le réseau général d'Electricité de France.

Certaines options audacieuses ont été adoptées pour leur réalisation: a) le caisson est en béton précontraint et b) le chargement du combustible s'effectue en marche.

La montée en puissance de G 2 a eu lieu en avril 1959, celle de G 3 en mars 1960. Ces réacteurs constituent la première réalisation française de la filière graphite - gaz pour les réacteurs de puissance. Leur destination est avant tout plutonigène, mais devant l'intérêt que cette filière re-

présente au point de vue énergétique, le développement de cet aspect est poursuivi par l'Electricité de France, sur le site de Chinon.

La somme d'expérience acquise à Marcoule en 4 années est considérable et il paraît intéressant de publier les résultats obtenus, surtout en ce qui concerne les réalisations originales de nos réacteurs.

La présente communication mettra l'accent sur les performances obtenues au cours de ces 4 années de fonctionnement, entre autres sur les augmentations successives de la puissance de ces réacteurs.

Le chargement et déchargement en marche de G 2 et G 3 sera particulièrement détaillé en mettant en relief les principales modifications qui ont permis de réaliser le tonnage actuel de 3 t/j.

Il sera fait mention de quelques incidents sur les installations dites « classiques » et on essaiera d'en tirer un enseignement.

Trois compléments importants sont joints à ce rapport. Ils traitent de la détection de rupture de gaines de G 3, de la surveillance automatique des températures, de l'organisation et de l'instruction du personnel [2].

Le caisson en béton précontraint des piles G 2 et G 3, précurseur heureux en ce domaine, est traité à part dans une autre communication [3].

L'aspect neutronique et les enseignements qu'il est possible d'en tirer ont été exposés au Colloque de Bornemouth en avril 1962, [4] et à la conférence actuelle dans une communication traitant de l'ensemble de ces problèmes pour la filière graphite - gaz dans le programme français [5].

II. PERFORMANCES DU RÉACTEUR

Lors de l'établissement de l'avant-projet, la puissance de 150 MW avait été fixée comme puissance nominale. Au fur et à mesure de l'avancement des études, on pouvait envisager une puissance de 200 MW en utilisant le même cœur, grâce à une répartition du flux correspondant à un aplatissement de la zone centrale et à un relèvement de ce flux dans la zone périphérique par le chargement d'un combustible de plus grand diamètre.

La divergence de G 2 a eu lieu le 21 juillet 1958 (fig. 1)

Après les expériences neutroniques habituelles et les dernières mises au point, la montée en puissance débutait le 5 avril 1959. Dix-huit jours après, la puissance de 150 MW était atteinte et la puissance nominale de 200 MW était obtenue le 27 mai de la même année. La centrale de récupération mise en route entre-temps, débitait sur le réseau français le 22 avril 1959.

Le réacteur G 3, mis en route 11 mois après G 2, suivait sensiblement la même progression dans la montée en puissance. Les dates correspondantes sont:

- montée en puissance: 7 mars 1960,
- puissance nominale de 200 MW: 22 avril 1960,
- couplage de l'alternateur: 4 avril 1960.

Pendant la première période de fonctionnement des deux réacteurs, la limite de pilotage a été la température de la gaine la plus chaude. Une série d'une cinquantaine de thermocouples de gaine avait été répartie dans toutes les zones des réacteurs afin de dresser une carte des températures.

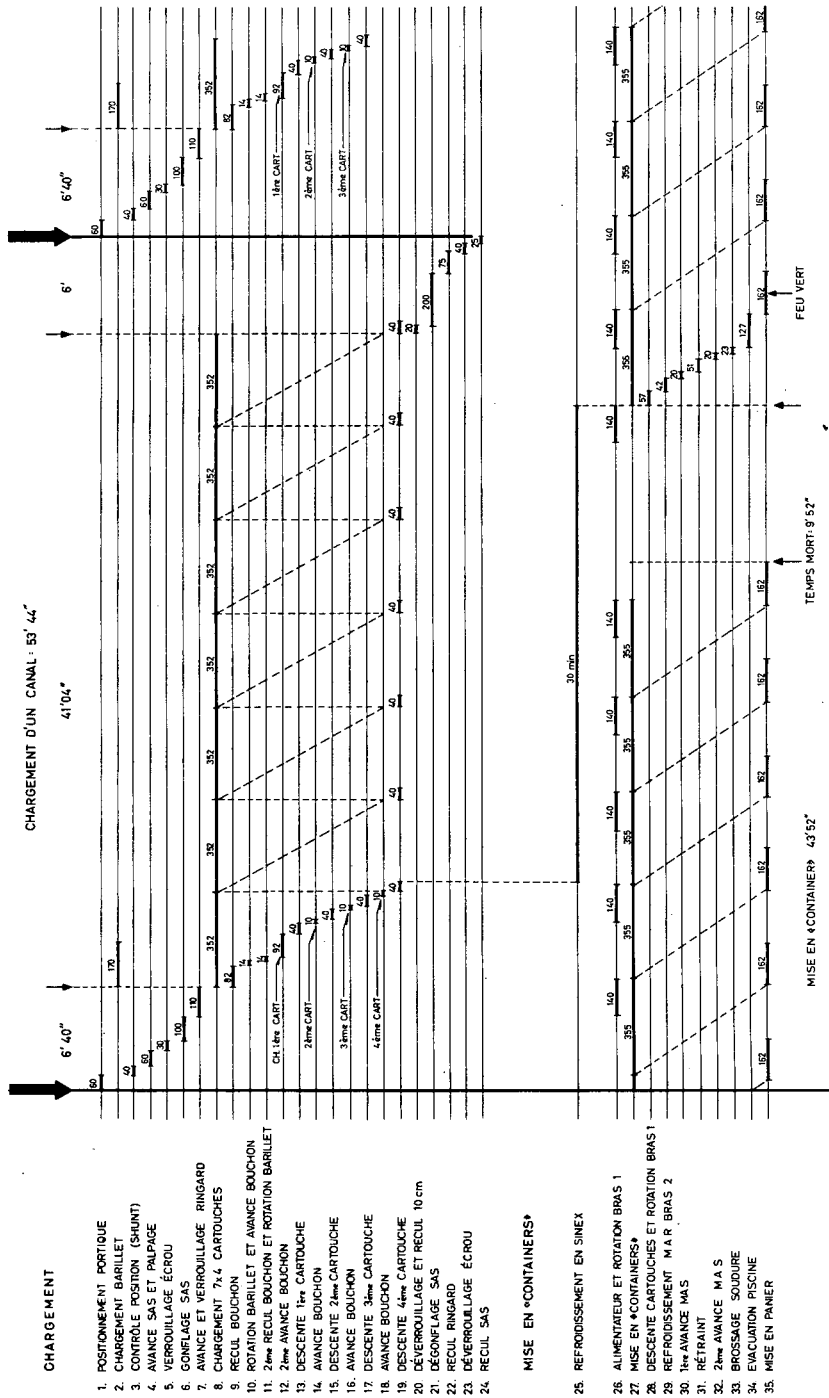


Figure 1
Diagramme des temps pour les réacteurs G 2/G 3.

permettant, par la suite, de s'abstraire de ces mesures de températures incompatibles avec un chargement en marche. Ces séries de mesures furent suffisantes pour déterminer les relations entre les températures des gaz, celles des gaines et le flux neutronique. La température de gaine était fixée à 415°C.

Lors du premier arrêt de chacun des réacteurs, les thermocouples de gaines furent enlevés et ne furent pas remplacés. Depuis cette époque, par conséquent, la température de pilotage est la température du gaz de sorties des canaux. Rappelons que tous les canaux sont équipés de thermocouples CO₂. Les relations établies pendant la première période ont permis de conserver la validité de la température de la gaine la plus chaude à 415°C même pendant des fonctionnements avec des températures d'entrée de gaz modifiées.

Les températures d'entrée de gaz ont subi, en effet, plusieurs changements depuis le démarrage des réacteurs. Rappelons que nos réacteurs possèdent 2 zones d'entrée de gaz. En premier lieu, les points chauds découverts dans le béton de G 2 ainsi qu'il est relaté plus loin, ont imposé une période de marche anormale, la température de la zone périphérique intéressant environ 400 canaux a été baissée au maximum (110°C). Dès la remise en ordre des circuits internes, cette température a été ramenée à une valeur plus correcte (130°C).

Ces températures d'entrée ont été, à nouveau, modifiées par la suite de façon à mesurer l'influence exacte sur l'énergie Wigner dans les 2 zones d'entrée à températures différentes. Par la suite et en raison des résultats obtenus, ces températures ont été uniformisées et les deux réacteurs fonctionnent avec une température d'entrée en toutes zones de 140°C.

A la suite de ces modifications de températures d'entrée et au prix d'un léger changement dans le réseau d'absorbants (possible en marche), l'on a pu envisager un chargement de combustible différent. Jusqu'alors, la zone centrale (800 canaux) était chargée avec des barreaux d'un diamètre de 28 mm, la zone périphérique (400 canaux) étant chargée en diamètre 31 mm. Pendant l'année 1961, ces diamètres furent uniformisés à 31 mm. La puissance atteignit alors 220 MW.

La fonction avant tout plutonigène de nos réacteurs, n'imposant pas nécessairement une économie de combustible, nous a amenés à repenser les températures limites initialement fixées pour ce combustible et le gainage. Il a paru possible d'augmenter ces valeurs assez sensiblement, sans risques, le combustible et les gaines ayant une tenue parfaite.

Au début de l'année 1962, la température de gaine maximum fut fixée à 455°C, la puissance atteignait alors 240 MW.

Cette température de gaines ne tient pas uniquement compte de la tenue du combustible en pile. Sa valeur est donnée par le calcul de l'échauffement possible en cas d'accident de dépressurisation rapide par rupture de grosse tuyauterie à l'entrée du réacteur, échauffement ne devant pas dépasser la température de fusion et de combustion du magnésium. Il semble que les hypothèses de calcul laissent encore une marge de sécurité.

Par ailleurs, la tenue du gainage depuis cette augmentation de température est toujours aussi bonne et aucune tendance néfaste n'a été décelée. Des essais se poursuivent avec des températures encore plus élevées et nous permettent d'espérer pour l'avenir.

Une autre possibilité d'augmentation de puissance est donnée par l'augmentation des températures dans les zones relativement froides. L'aplatissement du flux dans le sens radial est obtenu par un réseau de canaux d'absorbants conservant sensiblement une forme de flux en cosinus dans le sens longitudinal. Par un chargement hétérogène de ces canaux d'absorbants, il a été possible de déplacer le maximum du flux vers la zone d'entrée de gaz. Ceci s'est traduit par un aplatissement de la courbe des températures le long des canaux, rendant possible un relèvement général sans que le maximum dépasse la valeur précédemment admise.

L'augmentation de puissance corrélative a été supérieure à 10 MW, la puissance dépassant ainsi 250 MW dès la fin de l'année 1962.

On verra dans le chapitre traitant du chargement en marche les possibilités offertes par cette technique. Si l'augmentation des performances maxima est agréable en soi, il paraît encore plus rentable d'augmenter la puissance moyenne en la maintenant le plus près possible de ce maximum. C'est ici qu'intervient l'agrément d'un changement possible, en marche, du combustible et des éléments absorbants. La dispersion des puissances unitaires des canaux peut être réduite par un ajustement permanent du chargement. Dans nos réacteurs, ce réajustement est habituellement appliqué. Une telle facilité permet donc d'obtenir une puissance moyenne proche de la puissance maximum. L'on peut dire que 250 MW est dans l'état actuel la puissance minimum que nous obtenons sur les réacteurs G 2/G 3.

La puissance récupérée par la centrale bénéficiait parallèlement des performances du réacteur et augmentait dans le même temps pour atteindre à ce jour une puissance proche de 40 MW, c'est-à-dire la saturation du turbo-alternateur.

Rappelons que la puissance nécessaire au soufflage est prélevée au niveau des échangeurs par emploi de la vapeur MP sur des turbosoufflantes. La puissance récupérée est donc nette et peut être distribuée sur le réseau d'Electricité de France en quasi totalité.

III. MODIFICATIONS INTERNES

La mise en route du premier réacteur (G 2) menée à la fois avec prudence et rapidité, permettait d'atteindre 150 MW le 23 avril 1959, soit 18 jours après le «top» de départ. Les petits incidents habituels en un tel cas furent vite surmontés mais il restait quelques grosses anomalies dans les températures à l'intérieur du caisson, anomalies auxquelles il était impossible de remédier sur l'heure.

En trois points, en effet, le béton dépassait très sensiblement la température de 50°C prévue prudemment par le projeteur. Des perturbations dans les courants du gaz de refroidissement amenaient hors des circuits normaux un gaz très chaud et provoquaient des échauffements néfastes.

1. Echauffements dus aux circuits «primaire» et «secondaire»

Il est expliqué dans une autre communication [3] les raisons pour lesquelles, dans cette première réalisation en béton, le refroidissement du caisson est assuré par un circuit spécial appelé «circuit secondaire». Le

bloc actif proprement dit est balayé par le «circuit primaire». La co-existence à l'intérieur du caisson étanche de 2 circuits à destinations différentes et à température et débits également différents, posait des problèmes de fuites entre circuits. Afin de contrôler ces fuites, elles furent rendues volontaires et dans un sens tel, que les gaz «primaires» chauds ne pouvaient se mélanger au gaz «secondaire» froid. La compensation de ces fuites est faite dans les circuits extérieurs au réacteur. En fait, la section de fuite volontaire s'est avérée trop importante et la hauteur de l'empilement créait un effet de cheminée dans les capacités de répartition des gaz. La fuite volontaire s'est révélée inversée dans le sommet du bloc graphite, le gaz chaud venant se mélanger au circuit secondaire provoquait un réchauffement dans la partie haute du caisson.

Par ailleurs, le circuit secondaire intéresse à l'intérieur du caisson, un volume considérable et le débit devenait par endroits trop faible pour évacuer les calories supplémentaires amenées par la fuite inversée.

Une poche de gaz chaud s'est ainsi formée le long de la génératrice haute du cylindre de béton, réchauffant la peau d'étanchéité et par suite le béton lui-même. Les températures atteignirent et dépassèrent 80°C sur la tôle intérieure. Les contraintes dans le béton, mesurées avec le soin qu'on imagine, donnèrent également des valeurs relativement élevées mais n'atteignirent jamais un seuil inquiétant. Le réacteur fonctionna à pleine puissance dans ces conditions pendant plus de 3 mois.

Un premier remède provisoire fut apporté en abaissant les températures d'entrée de gaz de la zone périphérique, mais cette méthode ne pouvait s'appliquer indéfiniment en raison de l'accumulation de l'énergie Wigner malgré le flux relativement faible de cette zone.

Le réacteur G 3 n'ayant pas encore fonctionné à cette époque, des essais en température furent entrepris afin de trouver un remède efficace à ces anomalies. En fin de compte, plusieurs modifications furent décidées et appliquées:

- a) la section de la fuite volontaire entre circuits fut considérablement diminuée par la mise en place de tôles de fermeture jouant un rôle de diaphragme.
- b) le balayage de la partie haute du cylindre fut amélioré par la pose de tuyauteries amenant directement du gaz frais dans cette zone.
- c) la circulation générale du circuit secondaire à l'intérieur du caisson fut rendue plus cohérente et plus efficace en tous points par la pose d'un «déversoir» imposant un cheminement plus homogène au gaz.

Il convient de rappeler ici que le réacteur G 2 avait fonctionné à pleine puissance pendant 3 mois. Le travail à l'intérieur du caisson pouvait paraître impossible. Il n'en fut heureusement rien et malgré la gêne apportée par les précautions draconiennes imposées au personnel, ces travaux furent effectués dans de parfaites conditions de sécurité et de rapidité.

Le réacteur G 3 ne posa pas de problème semblables et il fut possible d'apporter quelques modifications supplémentaires ce qui constitue une des rares différences existant entre les 2 réacteurs.

2. Echauffements dus au déchargement

Le déchargement du combustible irradié s'effectue par gravité grâce à des tubes collectant les canaux à la sortie du graphite et conduisant les cartouches dans des tours de ralentissement extérieures au caisson en béton. Une extrémité de ces tubes est donc en communication avec les capacités d'entrée des gaz chauds et l'autre extrémité avec les tours appelées «toboggans» contenant du gaz froid.

Un thermosiphon s'établit entre ces deux capacités par l'intermédiaire des tubes. Ces tubes traversent le béton du caisson. Les gaz chauds apportèrent donc des calories à ce béton et sa température dans la zone intéressée atteignit et dépassa 100°C.

Le remède fut ici relativement simple puisque intéressant une partie de circuit extérieur au caisson proprement dit. Il a consisté en la pose sur les tubes de déchargement de clapets interrompant la circulation parasite du gaz. Ces clapets doublés par des vannes d'isolement sont évidemment ouverts à l'occasion des déchargements.

3. Echauffements dus aux prélèvements de gaz

Les prélèvements de gaz pour la détection de rupture de gaine s'effectuent à la sortie de chaque canal, à l'intérieur du caisson. Les tubes correspondants traversent la coupole de béton constituant la face pile et se regroupent dans les salles de détection.

En raison des différences de pression existant entre les canaux, un courant s'établissait entre ces différents tubes réchauffant ainsi le béton de la face pile. Ce phénomène était accéléré en particulier sur le réacteur G 2 en raison de la technique de la détection de rupture de gaine. Le gaz est ici à haute température puisque prélevé à la sorties des canaux. Le béton atteignit des températures encore plus élevées que précédemment.

Le remède fut de deux sortes: les vannes rotatives de la détection de rupture de gaine de G 2 furent modifiées de façon à supprimer toute communication entre les arrivées de tubes de prélèvement. Sur les deux réacteurs, des clapets antiretour furent posés sur ces mêmes tubes de façon à ne laisser subsister qu'un sens d'écoulement des gaz chauds et uniquement pendant les périodes de prélèvement.

Les anomalies citées ci-dessus furent donc supprimées dans la première année de vie du réacteur G 2. Ces phénomènes n'atteignirent jamais des valeurs inquiétantes. En dehors des leçons à tirer concernant l'étude et la conception des circuits de gaz dans un réacteur à caisson en béton, il convient de souligner l'illustration que constituent ces phénomènes en ce qui concerne la sécurité de ces caissons. Malgré les hautes températures atteintes (plus de 100°C), le matériau constitutif n'a pas souffert et le caisson de G 2 a montré ainsi la marge considérable que laisse un tel ouvrage dans les valeurs habituelles d'une exploitation normale.

IV. AMÉLIORATION DES INSTALLATIONS DE CHARGEMENT

Le chargement des réacteurs de Marcoule se fait en marche, c'est-à-dire que ces opérations effectuées en service continu n'affectent pas le

fonctionnement de la pile elle-même. Rappelons-en brièvement les principes généraux: G 2 et G 3 sont des piles horizontales, l'introduction du combustible neuf a lieu par la face extérieure où débouchent les 1 200 canaux sur lesquels vient se fixer l'appareil de chargement. Le combustible introduit pousse les cartouches du canal et provoque à l'autre extrémité la chute du combustible irradié. Celui-ci descend de lui-même par gravité jusqu'aux installations de mise en « container » et est ensuite dirigé vers la piscine de stockage. L'ensemble de ces opérations est automatique ou télécommandé.

Le chargement proprement dit s'effectue par l'intermédiaire de 2 sas de chargement. Un sas étant accouplé au canal à charger, les pressions sont égalisées, le bouchon est déverrouillé et retiré du réacteur. On dispose alors le combustible neuf devant ce même bouchon qui est remis en place dans le canal, entraînant les cartouches dans son mouvement.

L'appareillage des sas comporte sept mouvements pour assurer l'ensemble des opérations. Ces mouvements sont automatiques et les reports visibles, mécaniques ou électriques, permettent de contrôler à tout instant les positions et l'avancement du combustible en cours d'enfournement. Nous relaterons ci-dessous les principaux incidents survenus dans la mise au point et les modifications apportées.

1. Défauts d'étanchéité au sas de chargement

Des ruptures d'étanchéité se sont produites sur le sas de chargement. La sanction était en général toujours la même: le sas étant ramené brusquement à la pression atmosphérique par la fuite, le bouchon en cours de mouvement était repoussé violemment par la pression intérieure du réacteur et projeté au fond du sas. Il provoquait des dégâts importants à l'appareillage. Une première parade a consisté à rendre le mouvement de commande du bouchon irréversible et capable de contenir des poussées importantes. De cette façon, une rupture d'étanchéité dans l'état actuel provoquerait une perte momentanée de CO_2 mais l'appareillage ne souffrant pas et étant toujours disponible, il deviendrait possible de refermer le canal immédiatement.

Les bouchons fermant les canaux ont une longueur de 8 m environ; ils sont articulés mais, afin de constituer une protection biologique suffisante, ils sont introduits dans les boisseaux avec un jeu très faible. On conçoit dans ces conditions, que certains bouchons aient pu difficilement reprendre leur place normale. Précisons que ces incidents ont surtout eu lieu au début. Au bout de plusieurs manœuvres l'opération se reproduit sans difficulté, sauf cas exceptionnel de grippage. A plusieurs reprises, il a été impossible de verrouiller le bouchon. Nous avons été dans l'obligation, lors d'un de ces incidents, de découper au chalumeau l'avant du sas de chargement pour l'abandonner sur la pile. Cette opération nous a montré la modification à apporter à l'appareil. Actuellement, le sas possède une partie avant démontable qu'il est possible d'abandonner sans ennui sur un canal rétif. Quelques opérations simples permettent ensuite de le récupérer puisqu'il devient accessible, la partie gênante de l'appareil étant reculée.

Une autre possibilité de refermer un canal malgré tout nous est aussi donnée grâce à un bouchon spécial qui est en permanence dans le sas de chargement. Ce bouchon à verrouillage automatique a la longueur de deux

cartouches et occupe un alvéole réservé au combustible. En cas d'ennuis, il suffit de pousser ce bouchon comme une cartouche ordinaire et il vient se verrouiller de lui-même sur le canal, en assurant la fermeture.

Les incidents relatés ci-dessus ont eu, en général, des conséquences semblables pendant la période de mise au point, avant modification: le dégagement du sas de chargement étant impossible puisque les canaux ne pouvaient être fermés, il a fallu décompresser complètement le réacteur pour désaccoupler la machine.

Ces incidents ont rapidement ouvert les yeux sur les remèdes urgents à apporter, dont certains ont été décrits ci-dessus. Un principe primordial peut être dégagé: pouvoir, en toute circonstance, étancher un canal, l'abandonner et dégager l'appareil de chargement.

2. Positionnement défectueux des pièces à l'intérieur de l'appareil

Certains incidents ont eu pour cause une connaissance imparfaite des positionnements des pièces à l'intérieur de l'appareil. Cette mauvaise connaissance provenait autant de la précision mécanique insuffisante des pièces elles-mêmes que de la mauvaise fidélité des organes reproduisant ces mouvements à l'extérieur.

Les principales modifications ont porté sur le maintien des pièces en mouvement. On a cherché à guider positivement, à tout instant, chaque pièce dans n'importe quelle position, chaque arrêt par contre étant confirmé par un verrouillage mécanique positif et précis.

On devient ainsi sûr à 100% qu'un organe arrive bien à l'endroit qui lui est assigné, s'y arrête et ne bouge plus. Ceci paraît très simpliste mais ne paraissait pas évident lors de l'établissement du projet. Notre expérience nous a ainsi dicté cet autre principe: mécanique simple mais précise, toujours guidée ou verrouillée positivement.

3. Défauts dus à des transpositions hâtives du réacteur G 1

La rapidité de la conception et de la construction de nos réacteurs, a amené certaines transpositions du réacteur G 1. Toutes n'ont pas donné de bons résultats; par exemple la serrure des bouchons des canaux, quelques ennuis, mais surtout quelques craintes nous ont fait décider leur remplacement. Les fonctions de cet organe ont été séparées, les pièces de verrouillage ne subissant plus d'effort pendant les avances ou recul du bouchon. Certains principes ont été adoptés, en particulier un auto-verrouillage provoqué par la poussée de la pression intérieure. La résistance mécanique de l'ensemble a été considérablement augmentée.

Avant de généraliser le changement des serrures sur les 2 400 canaux des 2 réacteurs, des essais très sévères ont été effectués. Notre satisfaction n'a été complète qu'après un nombre de manœuvres représentatives des serrures prototypes correspondant à un service continu de 4 000 ans sans un seul incident.

Nous avons été amenés d'ailleurs en d'autres occasions, à appliquer systématiquement la règle suivante: un organe essentiel n'est mis en pile qu'après avoir subi, sans le moindre incident, sous forme de prototype, des essais représentant 2 000 ans de service au minimum.

4. Hall d'essai pour les pièces importantes

L'entretien systématique et le réglage des appareils ont été étudiés très minutieusement. Un hall d'essai faisant, en quelque sorte, fonction de « station service » a été aménagé. Tous les appareils importants sont révisés à intervalles fixes et essayés dans des conditions représentant exactement celles de leur travail normal. Ceci s'applique aussi bien aux petites pièces qu'aux grands ensembles tels que les sas de chargement. En effet, ceux-ci peuvent être remplacés facilement et transportés d'une seule pièce dans le hall d'essai, où un canal expérimental en pression permet tous les essais et mises au point possibles.

La mise en œuvre de cette « station service » a été immédiatement bénéfique. Actuellement, un appareil, quel qu'il soit, réglé dans cette station peut être employé directement sur le réacteur sans réglage ultérieur. Le cas des machines à souder les « containers » est ici typique, puisque ces machines complexes sont changées à distance sans intervention manuelle.

Un fonctionnement régulier et sûr est assuré par un entretien et un réglage systématique dans des conditions intégralement représentatives.

5. Nouveaux moyens de mise en œuvre pour le déchargement et le chargement

Signalons par ailleurs, et à titre d'information, de nouveaux moyens mis en œuvre, à titre d'essai, pour le déchargement et le chargement de ces réacteurs par la face avant. Un ringard souple et enroulable constitué de feillard de tôle s'agrafant automatiquement, pénètre dans le cœur. Ce ringard muni d'une pince électrique à son extrémité peut ainsi retirer les cartouches une à une, les transférer dans d'autres canaux où les évacuer ou les évacuer par cercueils spéciaux. Il y a aussi possibilité, par simple poussée de ce ringard, de déchargement par la voie normale sans être tenu de remplacer le combustible comme dans le cas habituel et ainsi d'obtenir aisément des canaux vides. Le système complet est contenu dans une petite enceinte sous pression de faible encombrement. La longueur totale de ringard qui peut être déployée atteint 30 m.

V. AMÉLIORATION DES INSTALLATIONS DE DÉCHARGEMENT

Le combustible irradié tombe par gravité hors du réacteur et est récupéré, toujours sous pression, dans des couloirs vibrants horizontaux qui permettent de le stocker et de le refroidir pendant quelques instants avant la mise à l'air libre. Le passage du CO_2 à 15 kg/cm^2 à l'air à pression atmosphérique s'opère dans un sas équipé d'un couloir vibrant qui évacue les cartouches par groupe de 4. Ces 4 cartouches sont disposées dans un tube d'aluminium (container) qui est rétreint à ses extrémités et soudé. Le « container » est ensuite déposé dans une conduite hydraulique qui l'achemine en sous-sol jusqu'à la piscine de désactivation où le combustible est emmagasiné.

L'ensemble de ces opérations est automatique et cet automatisme ne nous a jamais créé d'ennuis. Les quelques incidents que nous ayons eu à déplorer proviennent de coincements mécaniques des « containers » dans les machines. Un remodelage des pièces de guidage est venu à bout de la majorité des cas.

Comme dans le cas du chargement, une chaîne d'essai extérieure a été réalisée. Les machines de mise en « containers » sont périodiquement évacuées, entretenues et réglées. Leur interchangeabilité est complète. Un entretien systématique a ainsi remplacé le dépannage qui est le propre de toute installation prototype.

Les systèmes à vibrations enfermés dans des enceintes étanches et sous pression nous ont occasionné quelques déboires. La seule parade efficace a été, à nouveau, une précision accrue et un contrôle strict et systématique de l'installation.

Tous les autres incidents ont été surmontés facilement en accroissant les moyens de contrôle visuel et les moyens d'intervention à distance. Les machines et le combustible étant ici à l'air libre, il a suffi de disposer des hublots à grande visibilité et de mettre en œuvre des télémanipulateurs de grande capacité.

Il semble utile de mentionner quelques considérations d'ordre général en ce qui concerne la mise en « containers » et l'évacuation du combustible irradié.

Cette installation était une de celles qui nous inquiétait le plus à l'époque du projet. Trier un combustible sortant de pile, le mettre en boîtes étanches et l'expédier à la piscine de désactivation, ceci à cadence industrielle, semblait à l'époque d'une inquiétante complexité.

Grâce à une mise au point assez poussée, en inactif, la mise en service en actif a été relativement facile malgré les petits ennuis inévitables mentionnés ci-dessus. L'exploitation industrielle de cette installation qui fonctionne depuis 4 ans permet des cadences doubles de celles du chargement détaillées plus loin.

Par ailleurs, beaucoup de précautions prévues à l'époque du projet sont inutiles et une mise en « containers » moderne serait plus simple. Certaines opérations seraient supprimées, par exemple le refroidissement intermédiaire du combustible avant soudage.

Actuellement, le déroulement des opérations est le suivant:

- déchargement du canal,
- refroidissement durant une demi heure dans du CO₂ en pression (ce temps peut être fortement réduit sans inconvénient),
- mise à l'air libre du combustible,
- mise en boîte,
- rétreint des « containers »,
- soudage,
- transport hydraulique à la piscine.

Depuis la mise à l'air libre des cartouches jusqu'à l'arrivée en piscine, il s'écoule moins de 6 min. Le transport par conduit hydraulique souterrain supprime toute manutention, il est sans aléas et d'une parfaite régularité. Trente secondes sont suffisantes pour le transport d'un « container » jusqu'aux installations de pesage, triage et stockage sous l'eau de la piscine.

Là aussi, tout se passe par télécommande, sans heurts.

Le côté spectaculaire de ces opérations provient surtout de cette facilité et de cette sûreté déconcertantes qu'il n'est pas habituel de voir pour une telle manutention.

Nous avons longtemps considéré la mise en «containers» comme étant un impératif embarrassant. Il nous semble actuellement que les avantages considérables qu'en tire le stockage en piscine compensent très largement les investissements de cette installation.

Il convient de remarquer également l'agrément supplémentaire que procure une eau de piscine non contaminée, où le temps de séjour du combustible importe peu puisqu'il est à l'abri de toute attaque extérieure.

VI. INFLUENCE DU CHARGEMENT EN MARCHE

Le chargement en marche de nos réacteurs a deux conséquences au point de vue puissance:

- a) la première, d'effet immédiat, provenant du mouvement du combustible ou des absorbants.

Le gaz de refroidissement tend à repousser les cartouches vers la face de chargement quand le bouchon, étant reculé, ne les maintient plus en place. Ceci provoque peu d'effet quand il s'agit de combustible, mais provoque des fluctuations de puissance plus importantes quand on charge ou décharge des absorbants. Dans ce cas, les variations peuvent atteindre si l'on n'y remédie par les barres de contrôle, ± 10 MW autour de la puissance normale. En fait, le contrôle suit aisément ces fluctuations et, dans le cas d'absorbants, les manœuvres sont faites assez lentement pour ne pas créer de cyclages du combustible.

- b) la deuxième conséquence, d'intérêt considérable, est la possibilité d'adapter les zones d'irradiations différentes aux zones de températures ou de flux différentes et ainsi de profiter d'un gain de puissance supplémentaire au prix d'une programmation des chargements très étudiée. De plus, l'irradiation générale de la pile peut ne pas influencer sensiblement sur le contrôle du réacteur, l'introduction ou l'enlèvement d'absorbants étant possible à tout instant.

Il convient de mentionner un autre aspect des avantages du chargement en marche de ces réacteurs, il s'agit de l'homogénéité de l'irradiation. La circulation du combustible dans les canaux permet, suivant le nombre de cartouches chargées et déchargées, une économie certaine de combustible. Ceci peut provenir d'une plus grande homogénéité dans l'irradiation imposée, ou de la possibilité d'irradier au maximum un plus grand nombre de cartouches. Cet avantage supplémentaire, d'un grand intérêt, découle en partie de la position horizontale des canaux des réacteurs G 2/G 3.

VII. PERFORMANCES DU CHARGEMENT

Les sections précédentes font état de modifications dans la mécanique et la géométrie des installations de chargement avec un accroissement de

la sécurité comme but principal. Un deuxième but était atteint par la même occasion, il s'agit de la rapidité des opérations.

Par ailleurs, des améliorations supplémentaires ont été apportées afin de concourir cette fois au seul impératif des performances.

Les sas de chargement sont mis en pression ou en dépression à chaque opération. Il avait été prévu un circuit complexe de mise en pression et de mise sous vide de l'appareil. Ces opérations étaient fort longues. Le circuit a été modifié, la compression est faite à partir de gaz neuf, la mise sous vide (pour éviter des entrées d'air en pile) est remplacée par un balayage. Le gain de temps a été considérable. De plus, l'ancien circuit était à commande automatique. Ces opérations ne faisant prendre aucun risque particulier et restant très simples sont maintenant manuelles et ont permis de simplifier l'installation.

L'appareil de chargement positionne à chaque manœuvre du bouchon 2 cartouches. Après quelques mois d'exploitation, il a paru possible d'introduire un nombre supérieur de cartouches sans inconvénient pour les températures ou les débits à l'intérieur du canal. Le fonctionnement actuel normal admet 4 cartouches à chaque manœuvre de bouchon, permettant une meilleure cadence.

Un diagramme des temps pour une opération type est donné à la figure 1.

Il ressort de ce diagramme la possibilité de décharger près de 27 canaux complets en 24 h, soit 750 cartouches, soit 3 t de combustible. Ceci pour chaque réacteur.

Il s'agit là de performances correspondant à un fonctionnement normal, et non à des cadences de pointe. Les entretiens et remplacements systématiques cités aux sections précédentes, trouvent facilement leur place pendant les creux du programme, la cadence ci-dessus peut être en effet tenue en permanence mais dépasse bien sûr la demande.

On pourra objecter que ces performances sont bien inutiles pour des réacteurs de puissance à forte irradiation tels qu'ils sont envisagés pour les centrales futures. Mais trop de bien n'a jamais nui.

De plus, les réacteurs en construction, ayant un nombre de canaux important, demandent des performances de chargement élevées afin de profiter au maximum des avantages d'une telle installation.

Rappelons rapidement ici les principaux de ces avantages:

- économie de combustible,
- homogénéité de l'irradiation permettant la permanence du réseau d'absorbants,
- ou, au contraire, constance de la réactivité par modification du réseau,
- homogénéité des températures,
- possibilité de l'aplatissement longitudinal du flux, commode,
- gain énorme de temps,
- accroissement du facteur de charge, surtout si les arrêts pour entretien d'autres installations sont réduits.

VIII. L'AUTOMATISME ET LA CENTRALISATION DU CONTRÔLE

Il est frappant de constater l'étonnement des visiteurs entrant dans la salle de commande de nos réacteurs, devant l'importance des installations de

contrôle. Ces salles de commande ont été conçues comme le cerveau unique de l'installation où aboutissent toutes les mesures et d'où partent tous les ordres. Les opérateurs ont sous les yeux, en permanence tous les éléments d'appréciation dans tous les domaines: réacteur proprement dit, refroidissement et soufflage, centrale électrique, alimentation électrique, détection de rupture de gaine, épuration des gaz, alimentation en eau et en air comprimé, etc. Seul le chargement, qui possède sa propre salle de commande, n'est connu que par ses valeurs principales.

En ce qui concerne la détection de rupture de gaine de G 3 et la surveillance des températures, le contrôle n'est pas seul centralisé. L'automatisation est ici complète et les opérateurs n'ont qu'un rôle de supervision. On pourra lire dans les compléments 1 et 2 à cette communication, le détail de ces deux installations et les résultats obtenus dans ces domaines.

Des réalisations antérieures, mettant l'accent sur la simplification, avaient réduit le rôle de la salle de contrôle centrale, laissant à des contrôles locaux le soin d'assurer la bonne marche des sous-ensembles.

Des réalisations en cours de construction, au contraire, centralisent la totalité du contrôle et tendent à automatiser au maximum l'ensemble des manœuvres simplifiant par contre-coup l'aspect de la salle de commande, par suppression d'un grand nombre d'indicateurs et d'enregistreurs inutiles, ou remplacés par des téléscripteurs.

Il apparaît donc que les réalisations de G 2 et G 3 sont une étape intermédiaire entre les deux conceptions. Les résultats obtenus ne sont pas étrangers à la décision de passer à l'étape suivante, celle de l'automatisme intégral. En effet, la souplesse permise par la centralisation s'est rapidement avérée un élément essentiel dans la sécurité de l'exploitation. Toute modification des paramètres de pilotage a des effets dans tous les domaines et les automatismes partiels existants sont rapidement modifiés puisque tous les organes sont géographiquement rassemblés. De plus, toute action d'un opérateur étant connue des autres agents à l'instant même où elle a lieu, certaines conséquences néfastes peuvent être annulées immédiatement, les liaisons étant instantanées.

Chaque fois qu'il a été possible, et jugé raisonnable, l'automatisation a été poussée sur des organes existants permettant ainsi de contrôler la validité des arguments avancés. Ceci est particulièrement mis en valeur par l'exemple déjà cité de la détection de rupture de gaine de G 3 et de la surveillance des températures.

Ainsi, la conception des salles de commande des réacteurs G 2/G 3 a permis à la centralisation complète d'assurer les rôles qui lui étaient impartis. Souplesse et sécurité de l'exploitation.

De plus, cette conception a ouvert la voie à l'automatisme intégral, base d'une sécurité et d'un rendement accrus.

IX. INCIDENTS SUR LES INSTALLATIONS CLASSIQUES

Le nombre et l'importance des problèmes nouveaux posés par la construction des réacteurs nucléaires a pu faire oublier trop souvent que les installations dites classiques pouvaient à l'occasion créer des difficultés

imprévues. Les réalisations de Marcoule n'ont pas entièrement échappé à ce défaut.

Il serait fastidieux de passer en revue les ennuis mineurs tels que chacun en a connu dans des conditions semblables. Ce propos se limitera à un exemple caractéristique concernant les réfrigérants à eau. Les réacteurs G 2 et G 3 comportent chacun 6 réfrigérants principaux:

- 2 réfrigérants du circuit secondaire, circuit nécessaire au refroidissement du caisson en béton ainsi qu'il est expliqué plus haut.
- 2 réfrigérants du circuit primaire destinés au circuit principal jusqu'à une puissance de 40 MW et dont le rôle est primordial en cas d'arrêt du réacteur et en période de démarrage.
- 2 réfrigérants du circuit de compensation rétablissant les débits entre les 2 circuits précédents à la suite des fuites volontaires à l'intérieur du caisson comme il est exposé à la section III.

Ces appareils sont constitués par quelques milliers de tubes véhiculant le gaz de refroidissement, l'eau circulant autour de ces tubes. Disposant à profusion de l'eau du Rhône il n'a pas paru utile de faire une circulation en circuit fermé avec aéroréfrigérants. L'eau circule en circuit ouvert par pompage et restitution directes dans le Rhône. Il a toujours semblé acquis que ces appareils bien connus par ailleurs, ne poseraient pas de problèmes particuliers et rien n'avait été prévu pour leur emplacement éventuel. Leur emplacement en a découlé de telle sorte qu'ils sont en partie inaccessibles pour de grosses réparations.

Au bout de 4 ans de fonctionnement, des fuites se sont déclarées dans les faisceaux de tubes. Après contrôle, la situation a paru très sérieuse car les faisceaux étaient complètement encrassés et les tubes eux-mêmes corrodés de telle façon que des centaines d'entre eux n'avaient plus qu'une épaisseur réduite de 80% par endroits. Le pire était à craindre en cas de lâchage général d'un grand nombre de ces tubes.

Le remède paraissait simple: changer les tubes et conditionner l'eau. Si la deuxième partie de ce programme ne nécessitait qu'une mise de fonds et un délai raisonnable de quelques mois, le changement des tubes avariés était autrement délicat. Surtout si l'on tenait à conserver les réacteurs en marche avec une sécurité suffisante. Un planning détaillé fut établi et finalement l'opération fut entreprise dans des conditions assez acrobatiques pour concilier tous les impératifs de la production. Cette réparation, simple en elle-même, s'étage sur 1 an et demi et est actuellement en bonne voie de réalisation.

Il est inutile d'insister sur les difficultés de l'opération pour qui connaît la complexité des tuyauteries de nos réacteurs. Quand tout sera terminé, une grande partie de ces tuyauteries aura été déposée et reposée pour permettre la réparation. Il convient ici de dire que cette complexité a eu un côté relativement agréable, car sans les possibilités de souplesse qu'elle offrait en contrepartie, l'opération n'aurait peut-être été possible qu'au prix d'un arrêt complet de longue durée des réacteurs.

Les incidents sur de telles installations sont particulièrement irritants. Le projeteur considère de telles techniques sans aléas et les inclut dans l'ensemble, sans y apporter une attention particulière, retenu par ailleurs par des problèmes bien plus vastes et plus graves. Mais il est décevant,

après 4 ans de fonctionnement, de constater que les principaux ennuis proviennent d'appareillages connus et utilisés par l'industrie depuis toujours, alors que les ensembles ressortant des techniques nouvelles fonctionnent sans à-coups à la satisfaction de tous.

Il semble bien que rien n'est entièrement classique dans nos réalisations et que chaque détail doit être étudié et essayé longuement toujours comme une nouveauté, avant de prendre place dans un ensemble qui se veut parfait et toujours à l'avant-garde de toutes les techniques.

X. CAISSON EN BÉTON PRÉCONTRAIT

On ne traitera pas ici en détail de ce problème capital. En effet:

- l'aspect sécurité de ces ouvrages a été exposé lors d'une précédente conférence [6] et les caissons de G 2 et G 3 ont fait l'objet d'une communication spéciale;
- l'étude, la construction, l'exploitation sont traités dans une communication à la présente conférence [3].

Il convient néanmoins de rappeler ici les principaux avantages de telles enceintes, avantages mis en lumière pour la première fois, et avec un plein succès, par la réalisation encore unique au monde des réacteurs G 2/G 3 de Marcoule:

- possibilité d'atteindre des volumes importants,
- pressions de service très supérieures à toute autre méthode,
- sécurité quasi absolue en raison de l'impossibilité pratique d'explosion ou même de fuite importante,
- facilité de mise en œuvre,
- possibilité d'essais sur maquettes représentatives,
- commodité du contrôle des déformations, contraintes, températures,
- exploitation sans aléas et, en particulier, suppression du risque de rupture fragile.

XI. CONCLUSION

D'un point de vue très général, deux solutions peuvent être envisagées pour l'étude et la mise en œuvre d'ouvrages aussi révolutionnaires à leur époque que les réacteurs G 2/G 3 de Marcoule. Le projet peut être poussé dans les détails, retardant la construction, elle-même encore retardée par des améliorations incessantes en cours de travaux. L'espoir dans ce cas est que la mise en route sera sans aléas, rapide et concluante. Ou bien, et c'est la solution adoptée ici, l'avant - projet étant définitivement figé quant à ses grandes options, commencer à traiter des sous - ensembles sans coup férir et mettre en œuvre immédiatement. Ceci sous-entend que des marges assez considérables ont été prises dans les techniques nouvelles ou audacieuses de façon à ne pas brider inconsidérément les améliorations ultérieures. La construction suit alors à quelques jours près (et quelquefois précède) la mise au point des détails.

Cette façon de faire promet quelque souci lors de la mise en route et l'on considère à priori que la première année de fonctionnement est hypo-

théquée de la moitié du temps. En fin de compte cette demi année est gagnée sur la solution exposée en premier et bien souvent le gain est encore supérieur. Par ailleurs, la formation morale et intellectuelle et l'ambiance créée sont à mettre complètement au crédit de la deuxième solution. De plus, les modifications apportées avant la mise en forme définitive sont plus près de la perfection parce que raisonnées et s'appuyant sur un début d'expérience.

Il y a donc tout lieu de se féliciter des options prises pour nos réacteurs, tant au point de vue technique que moral. Il est évident que la réussite enregistrée ici en découle en grande partie.

Dans les circonstances présentes, la conclusion ne saurait, malgré les précédentes considérations, qu'être technique. En ce qui concerne l'expérience de l'exploitation des réacteurs de Marcoule depuis 4 ans:

La technique des réacteurs nucléaires est à son début, ce qui explique les augmentations de performances assez spectaculaires acquises sur de telles installations, ceci en raison des précautions prises à l'étude, par honnêteté intellectuelle.

Le béton précontraint a tenu ses promesses et il paraît vain de répéter les avantages de cette solution puisque la conviction est unanime, démontrée par les réalisations en cours.

Le chargement en marche, après une période d'adaptation et de mise au point, présente peu de difficultés supplémentaires par rapport au chargement à l'arrêt et les avantages sont trop considérables par ailleurs pour prêter à discussion.

L'automatisme complet est la voie à suivre à l'avenir et assurera sécurité et rendement.

Les installations « classiques » sont à surveiller de près.

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LA DÉTECTION DE RUPTURE DE GAINÉ AU RÉACTEUR G3

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Abstract — Résumé — Аннотация — Resumen

SLUG-BURST DETECTION IN THE G3 REACTOR. The author explains the principles underlying slug-burst detection and describes the construction of the apparatus concerned. The main features are a) fully automatic operation, b) centralization of data in the control room and c) measurement by electrostatic collection on a turntable.

LA DÉTECTION DE RUPTURE DE GAINÉ AU RÉACTEUR G3. Dans ce mémoire, l'auteur expose les principes sur lesquels est fondée la détection de rupture de gaines et il décrit la réalisation des installations. Les caractéristiques principales sont a) l'automatisme intégral, b) la centralisation des informations dans la salle de commande et c) mesure par collection électrostatique sur plaque tournante.

ОБНАРУЖЕНИЕ РАЗРЫВА ОБОЛОЧКИ В РЕАКТОРЕ G3. Излагаются принципы, на которых основано обнаружение разрыва оболочки, описывается конструирование установок. Основные характеристики таковы: а) интегральный автоматизм, б) централизация информации в командном зале и в) измерение путем электростатического собирания на поворачивающейся пластинке.

DETECCIÓN DE FALLAS DEL REVESTIMIENTO EN EL REACTOR G3. El autor expone los principios en que se basa la detección de las fallas en los revestimientos de los elementos combustibles y describe las características principales de la instalación, que son: a) automatización integral, b) centralización de las informaciones en la sala de mandos, y c) medición por recolección electrostática sobre una placa giratoria.

I. INTRODUCTION

Le but de la détection de rupture de gaine (DRG) est de vérifier, en permanence, l'intégrité de la gaine des éléments combustibles du réacteur. Une rupture d'étanchéité de cette gaine entraînerait, d'une part la pollution du fluide de refroidissement par les produits gazeux libérés au cours de la fission de l'uranium et, d'autre part, risquerait de créer un «point chaud» sur la cartouche par formation d'oxyde au contact uranium-gaz carbonique. La condition, pour que ces risques de pollution et d'élévation de température soient réduits au minimum et restent en tout état de cause dans des normes admissibles, est évidemment d'éliminer l'élément combustible incriminé, le plus rapidement possible. Cela est rendu possible par le chargement en marche et grâce à l'installation de détection de rupture de gaines.

Cet examen du rôle de la DRG permet de définir les deux critères essentiels auxquels devra répondre cette installation:

- a) Sensibilité très poussée afin de déceler une fuite de gaine dès son début.
- b) Rapidité de localisation de l'élément défectueux ou, tout au moins, du canal contenant cet élément.

Nous allons voir comment l'installation réalisée à G3 répond à ces principes, par la mise en application de solutions originales dont l'efficacité

a été continuellement améliorée au cours de ces quatre années d'exploitation.

II. PRINCIPE DE LA DÉTECTION

La détection d'une rupture de gaine se fait par la mesure de l'activité du gaz de refroidissement.

Les gaz de fission émis au cours d'une rupture de gaine, sont entre autres, du xénon (Xe) et du krypton (Kr) ayant une période de décroissance de 16 et 2,4 s. Ces gaz prélevés avec le CO_2 de refroidissement sont, après filtrage, envoyés dans une capacité appelée «volume de reconstitution» où, par émission β et γ , ils se transforment en atomes de césium et de rubidium, ionisés positivement, que l'on précipite sur une électrode à l'aide d'un champ électrostatique. Ces ions solides de césium et de rubidium sont eux-mêmes radioactifs - émission β - et permettent par l'intermédiaire d'un scintillateur et d'un photomultiplicateur, d'obtenir un signal électrique proportionnel à l'activité du gaz (fig. 1).

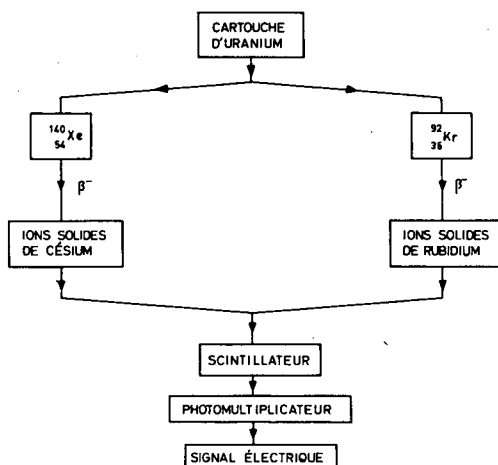


Figure 1

Schéma illustrant le principe de la détection de rupture de gaine.

De façon à mettre en évidence l'évolution du signal en fonction du temps, un dispositif appelé «évolumètre» permet de donner la différence entre la valeur du signal au moment de la mesure et une valeur antérieure, prise pour origine. Cette valeur origine est très simplement mise en mémoire, au moment voulu, par injection d'une tension sur des potentiomètres de tarage.

Le signal recueilli est, par définition, proportionnel au nombre de fission de l'uranium et, donc, au flux de neutrons, c'est-à-dire à la puissance de la pile. Il est donc nécessaire de corriger la valeur du signal au cours d'une évolution de puissance de la pile, par exemple, pendant la période de démarrage. Ceci peut se faire manuellement, par recalage volon-

taire, ou automatiquement de façon continue par l'intermédiaire d'un «pilote» délivrant une tension proportionnelle à la puissance.

Ce pilote est constitué par un détecteur mesurant l'activité de l'azote 16 contenue dans le CO_2 de refroidissement.

III. DESCRIPTION DE L'INSTALLATION

L'installation de détection de rupture de gaine de G3 se compose essentiellement de trois parties:

- a) Circuits pneumatiques comportant tous les circuits de prélèvement de gaz.
- b) Installation électromécanique regroupant tout le relayage nécessaire au fonctionnement automatique de l'installation.
- c) Chaîne électronique avec la chaîne de mesure et de traitement analogique des informations.

1. Circuits pneumatiques

Les 1200 canaux sont répartis à égalité entre 12 groupes de détection identiques, indépendants les uns des autres, sauf en ce qui concerne les séquences à réaliser qui sont asservies à un cadenceur général.

Les canaux d'un même groupe sont groupés par 5 pour former 20 faisceaux par groupe. Au total la répartition est la suivante: 12 groupes de 20 faisceaux de 5 canaux (fig. 2).

Chaque groupe de détection comprend 2 étages de mesures:

En marche normale, l'étage «prospecteur» seul en service, permet d'analyser successivement les 20 faisceaux du groupe à la cadence d'un faisceau par minute. Le temps total de balayage est donc de 20 min.

En cas de besoin particulier: signal en évolution, déchargement de canal, etc. on peut au moyen d'un jeu d'électrovannes passer sur l'étage «suiveur» qui permet, soit d'explorer successivement les 5 canaux d'un faisceau donné (l'intervalle entre deux mesures sur un même canal est alors réduit à 5 min), soit d'analyser de façon continue un canal donné (une mesure par minute).

Deux compresseurs centrifuges - l'un en service, l'autre en réserve - placés en fin de circuit, renvoient les prélèvements de CO_2 dans la pile (fig. 3).

2. Installation électromécanique

La sélection des canaux est purement électromécanique et se fait au moyen d'électrovannes à 2 ou 3 voies commandées par des relais et contacteurs classiques.

L'appareillage comprend des organes généraux communs aux 1200 canaux et des organes particuliers identiques pour tous les groupes.

Organes généraux

- Alimentation haute tension des détecteurs et armoire générale d'alimentation.

G_3

D'
3

C'
5

B'
6

A'
6

M
6

A
6

B
6

C
5

D
3

RÉPARTITION
DES
GROUPES

	S	R	Q	P	O	N	M	L	K	J	I	H	G	F	E	D	C	B	A	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q	R	S	
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Figure 2

Répartition des groupes.

- Cadenceur général rotatif qui rythme les cycles d'exploration, la rotation des tambours d'affichage et des sélecteurs.
- Relais de faisceaux qui pilotent les contacteurs d'électrovannes.
- Circuits de sécurité vérifiant le fonctionnement correct de l'installation et relais de signalisation correspondants.
- Répartiteur distribuant les signaux de l'armoire générale aux armoires de groupes, aux évolumètres, etc.
- Deux groupes de rechange complets permettant de suppléer n'importe lequel des 12 groupes normalement en service.
- Ensemble de commutation des groupes rechange.

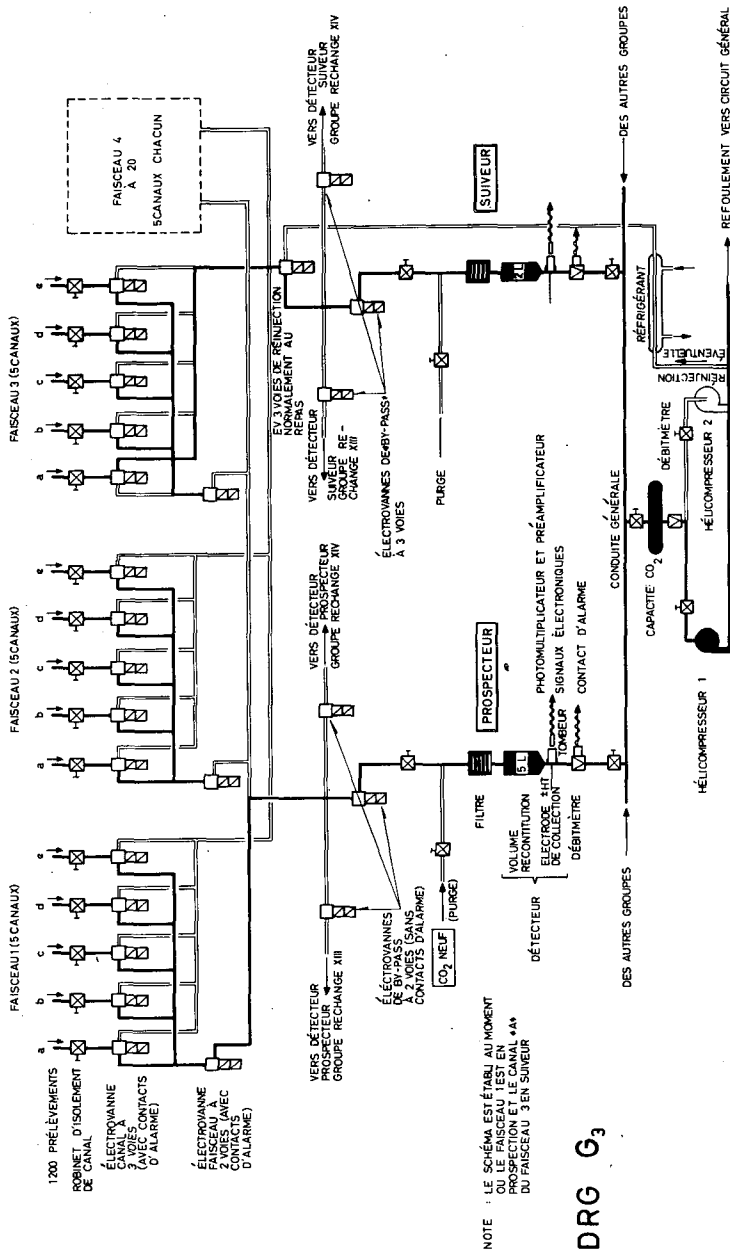


Figure 3
Schéma des circuits pneumatiques.

Organes particuliers

Ces organes relatifs à chaque groupe comportent essentiellement:

- Les relais et contacteurs de commande des électrovannes (les relais mémoires correspondant aux préalerte et alerte).
- Les circuits de sécurité et la signalisation correspondants.
- Les tableaux d'affichage et de signalisation.

3. Chaîne électronique

Elle comprend, par groupe et par étage, prospecteur ou suiveur:

- Un ensemble scintillateur - photomultiplicateur - préampli.
- Un ensemble ampli-discriminateur-octade-intégrateur.
- Un élément annexe constitué par les évolumètres.
- Un voltmètre à lampe et un enregistreur de type miniature.

IV. FONCTIONNEMENT PRATIQUE

1. Détection

Qu'il s'agisse de la chaîne prospecteur ou de la chaîne suiveur, la mesure se fait en trois temps (fig. 4 et 5):

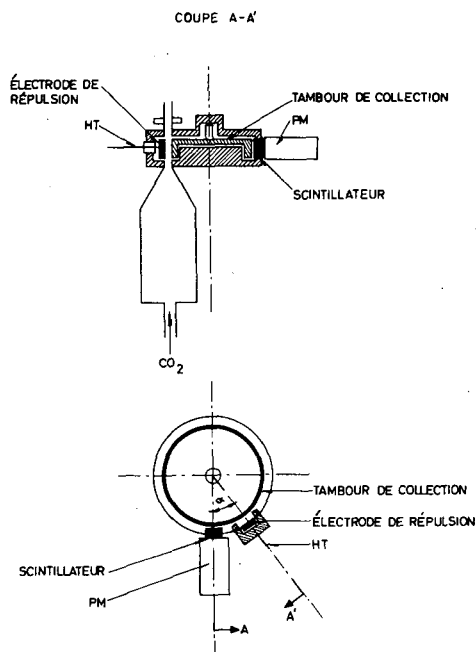


Figure 4

Schéma de principe du détecteur.

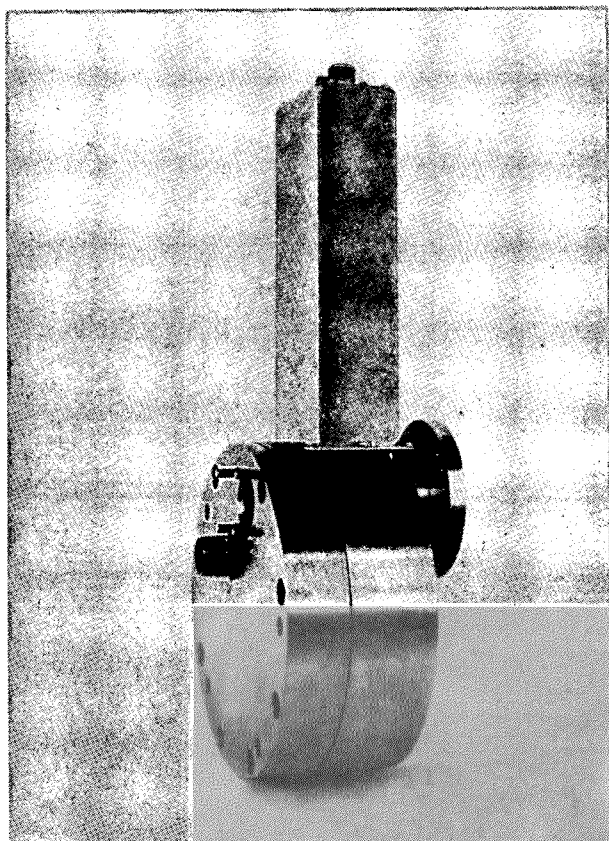


Figure 5

Photo du détecteur.

Premier temps

Le scintillateur mesure la pollution résiduelle ΔN du tambour en un point. La durée de la mesure est de 7 s. L'électronique enregistre un signal qui est la somme de la pollution du tambour, de l'activité des gaz restant sous le scintillateur, et du bruit de fond du scintillateur. Ces deux dernières valeurs sont sensiblement constantes.

Deuxième temps

La collection des ions positifs se produit au cours de ce temps. Le tambour tourne de 18° ($1/20^e$ de tour) et la plage considérée ci-dessus reste environ 35 s dans la chambre de précipitation. L'électrode de collection est portée à + 5000 V. Le champ électrique ainsi créé repousse sur le tambour les ions de rubidium et de césium.

Troisième temps

Le scintillateur mesure la nouvelle collection $N + \Delta N$ du tambour. La plage considérée du tambour revient se placer sous le scintillateur.

Pendant 7 s environ l'électronique mesure, en plus du signal mesuré au 1^{er} temps, l'activité des produits de fission collectés sur le tambour pendant le 2^e temps. Un montage électronique fait la différence des mesures du 3^e et du 1^{er} temps, soit $(N + \Delta N) - \Delta N$. Cette différence N constitue le signal utile correspondant à l'activité du CO_2 du faisceau ou du canal observé. Le tambour comportant 20 plages de mesures chaque plage repasse en mesure toutes les 20 min. Grâce à la vie courte des éléments considérés la pollution résiduelle du tambour a fortement décru pendant ce laps de temps.

2. Étage prospecteur

A. Marche normale

Seuls les étages «prospecteurs» des 12 groupes sont en service continu, en régime normal.

En cas de défaillance de l'un d'entre eux les étages «prospecteurs» des groupes de rechange XIII ou XIV doivent être mis en service de façon à ne pas interrompre la surveillance DRG.

Rappelons que la limite admise pour une interruption totale ou partielle de cette surveillance est de 30 min.

B. Passage sur évolumètre

Nous avons vu que le dispositif de lecture des signaux peut, soit fournir ceux-ci en valeur absolue, soit en indiquer l'évolution par rapport à un signal de référence mis en mémoire au moyen du dispositif «évolumètre» (fig. 6).

Cette opération de mise en mémoire s'appelle le tarage. Elle se fait à volonté sur un groupe quelconque et dure 20 min.

3. Étage suiveur

A. Passage volontaire

Au moyen d'un clavier d'appel groupe faisceau et d'une touche «Passage sur suiveur» l'opérateur peut, à partir de la salle de commande, appeler en surveillance sur l'étage «suiveur» un faisceau quelconque de 5 canaux. La carte de prospection des canaux DRG permet de trouver rapidement l'adresse d'un canal quelconque.

B. Passage automatique

Il se produit en cas de «dépassement» au niveau de l'étage «prospecteur». Il y a dépassement lorsque le signal (ou son écart par rapport au signal de référence si l'évolumètre est en service) atteint un seuil d'alerte

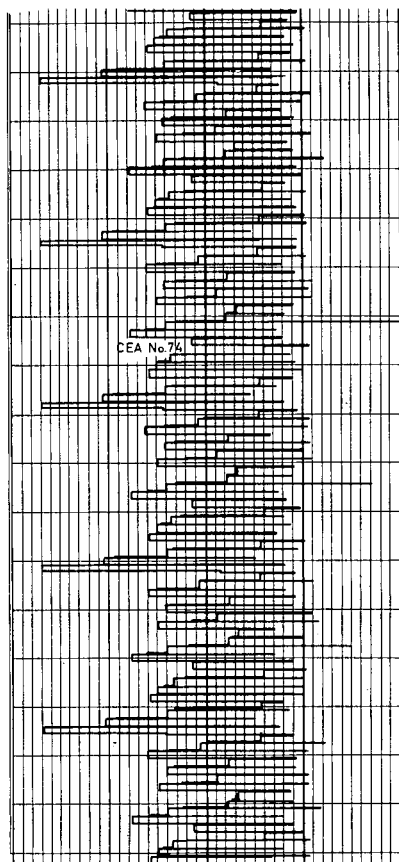


Figure 6

Type d'enregistrement avec évolumètre.

préalablement fixé. Cela correspond à un accroissement de l'activité du faisceau qui vient d'être observé.

Ce faisceau est repris automatiquement par l'étage «suiveur». La signalation « faisceau en observation » apparaît.

Si l'étage « suiveur » est déjà utilisé pour l'observation d'un autre faisceau la signalisation « faisceau actif non suivi » s'affiche.

C. Faisceau en suiveur cyclé

Grâce aux électrovannes canal à 3 voies les 5 canaux constituant le faisceau repris par l'étage suiveur sont analysés successivement, le cycle d'observation d'un canal étant de 60 s comme celui d'un faisceau.

Cette analyse séparée des 5 canaux d'un faisceau permet de déterminer le canal dont le signal a évolué, provoquant le dépassement initial.

D. Canal en suiveur continu

Ce canal en évolution étant repéré, il est intéressant de le suivre d'une façon continue. A cet effet une touche « canal sur suiveur » permet de bloquer en position excitée l'électrovanne à 3 voies du canal en cours d'analyse.

L'analyse cyclée des 4 autres canaux est abandonnée et le faisceau correspondant, privé du canal conservé en observation continue, est remis en prospection normale. On peut suivre de la même façon un canal en cours de déchargement normal, ou en cours de réglage de boisseau.

Une touche « remise en identification » permet de reprendre l'observation du faisceau dont on avait extrait un canal. Enfin une touche « suiveur disponible » permet de remettre en prospection normale le canal ou le faisceau en observation.

E. Suiveur en dépassement

Le faisceau dont le signal dépasse le seuil d'alerte fixé pour l'étage « prospecteur » est repris automatiquement par l'étage « suiveur ».

En marche normale les gammes en service sur les étages prospecteur et suiveur sont les mêmes (à titre indicatif gamme 250 chocs/s, pile à puissance normale).

Tandis que la gamme « prospecteur » ne peut être modifiée que sur intervention manuelle à partir de la salle de commande la gamme « suiveur » monte automatiquement si le signal du canal suivi atteint le seuil d'alerte de l'enregistreur « suiveur ». En même temps apparaît la signalisation « rupture de gaine ».

V. EXPÉRIENCE D'EXPLOITATION DE LA DRG

1. Essais préalables

La mise en service effectif de la DRG a été précédée d'un certain nombre d'essais et de vérifications. En particulier le détecteur a été mis au point sur une installation pilote mise en place sur le réacteur G2 [1, 2, 3, 4]. Le prototype du détecteur, essayé au préalable sur EL2 à Saclay, a été monté en dérivation sur un circuit de rechange de la DRG de G2. Le circuit pneumatique et l'ensemble de détection reproduisaient, à peu de choses près, les conditions d'utilisation de l'installation définitive. Ces essais ont permis, d'une part:

- a) de déterminer la valeur de la tension donnant les résultats optimum de collection des ions,
- b) de déterminer la pollution rémanente du tambour,
- c) d'étudier la décroissance des produits collectés et d'établir ainsi de façon précise la répartition des temps du cycle élémentaire de mesure.
- d) de connaître approximativement la valeur des signaux en fonction de la position de la rupture de gaine dans le canal.

D'autre part, des défauts minimes ont été mis en évidence et ont pu être éliminés grâce à des modifications de détails sur les appareils définitifs devant équiper la DRG de G3.

2. Modifications avant mise en service

En plus des modifications mineures et des mises au point de l'appareillage électronique, d'importants aménagements des circuits pneumatiques ont été réalisés avant la mise en service de l'installation. Ces modifications dont la nécessité est apparue au cours du fonctionnement de la pile G2 ont eu pour rôle essentiel de réduire l'échauffement du béton de la face Nord de la pile, traversée par les 1200 tubes de prélèvement. Plusieurs dispositifs ont été retenus et réalisés:

- a) Adjonction aux boisseaux, à la surface desquels est ménagée la rainure servant à canaliser le CO_2 prélevé dans le canal, d'anneaux flottants, permettant d'augmenter la surface de contact entre le gaz prélevé et le boisseau, au niveau de la zone secondaire.
- b) Pose d'un clapet anti-retour sur chaque tube de prélèvement. Ce clapet permet la suppression d'une circulation continue de CO_2 entre les tubes de prélèvement des différentes zones, circulation provoquée par les différences de pression existant entre différents canaux.
- c) Diminution du débit de prélèvement par canal, 12 g/s en moyenne au lieu de 30 g/s.
- d) Réinjection de CO_2 froid lorsqu'un canal est en suiveur permanent, à contre courant dans le tube de prélèvement. Dans ce cas, le prélèvement de mesure donne 1 min et la réinjection 1 min également.

3. Mise en service

La mise en service, au démarrage du réacteur G3 a permis dans l'ensemble de confirmer les résultats préalables obtenus sur l'installation pilote provisoire à G2. Après un mois de fonctionnement, le réacteur étant à sa puissance nominale, les constatations suivantes pouvaient être faites:

- a) Sensibilité: La valeur des signaux donnés par les différents groupes est parfaitement cohérente et du même ordre de grandeur pour tous les canaux. La sensibilité du prospecteur - pour la puissance nominale - est d'environ 2 chocs/s par mm^2 d'uranium nu (utilisation de cartouches témoins avec jauge DRG). Cette sensibilité est évidemment variable avec la puissance:

$$N = f \frac{\Phi \text{ flux}}{Q_{\text{débit CO}_2}}.$$

Modifications apportées pendant la période de mise au point:

- b) Modification des cadences de détection,
- c) Pilotage des évolumètres par le détecteur d'azote 16.

Ce détecteur donnant un signal proportionnel à la puissance du réacteur injecte une tension proportionnelle au comptage sur les potentiomètres mémoire des évolumètres.

4. Exploitation normale actuelle

L'exploitation normale de la DRG est assurée par un seul agent technique dont le poste normal de travail est en salle de commande. C'est là, en effet, que sont regroupées:

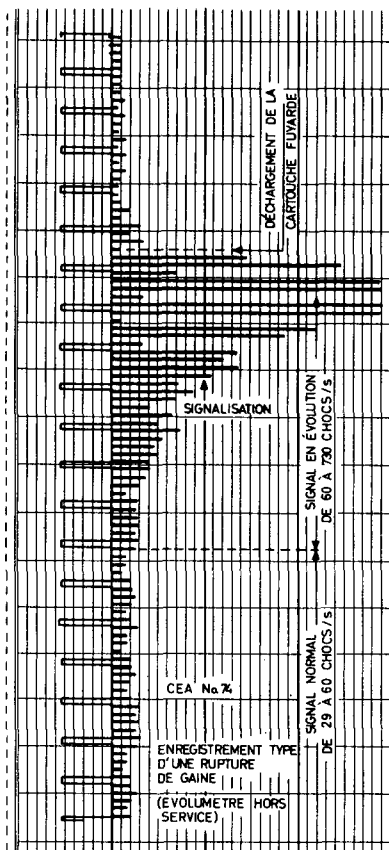


Figure 7

Enregistrement type d'une rupture de gaine.

- Les informations: les signaux correspondant à chaque faisceau (prospecteur) ou à chaque canal (suiveur), sont reportés sur des enregistreurs miniatures.
- Les signalisations de défaut de l'installation ou de préalerte et d'alerte dus à une augmentation importante d'un signal quelconque.
- Les commandes de commutation des groupes normaux sur les groupes de secours.
- Les commandes permettant la sélection en manuel d'un faisceau ou d'un canal.
- La commande des changements de gamme de mesure.

D'ailleurs, l'automatisme très poussé de l'installation réduit considérablement l'action de l'agent de quart qui n'assure pratiquement plus qu'un rôle de surveillance.

A titre d'exemple du fonctionnement de la DRG nous donnons (fig. 7), un enregistrement type de rupture de gaine permettant de voir l'évolution du signal et son atténuation brutale après déchargement de la cartouche fuyarde.

5. Modifications réalisées en cours d'exploitation

Elles sont essentiellement de deux sortes: a) amélioration de l'exploitation par un automatisme plus poussé, b) modifications techniques pour une meilleure sensibilité.

Amélioration de l'exploitation

- a) Changement automatique de gamme sur les suiveurs.
- b) Report du changement de gamme en salle de commande pour les prospecteurs.
- c) Mise en place d'une gamme de mesure allant jusqu'à 32 000 chocs/s sur les groupes normaux.
- d) Mise en place d'une gamme de mesure allant jusqu'à 64 000 chocs/s sur les suiveurs de secours.
- e) Montage d'un clavier d'appel de faisceau du suiveur, directement à partir de la salle de commande.

Modifications techniques

- a) Réduction de l'épaisseur du scintillateur pour diminuer le bruit de fond. En passant de 15 mm à 5 mm le bruit de fond diminuera de moitié. Des essais sont actuellement en cours sur les scintillateurs qui, après deux à trois mois de fonctionnement, présentent un clivage. Ces essais devraient permettre de déterminer l'épaisseur optimum, vraisemblablement encore plus faible que l'épaisseur actuelle, compatible avec la tenue du scintillateur et d'obtenir une mesure non altérée.
- b) Modification du cycle de mesure: la stabilité du signal a été augmentée en réalisant la mesure du bruit de fond sur le même secteur de tambour que la mesure du canal ou du faisceau.
- c) Montage de contacts rotatifs en lieu et place de microrupteurs.
- d) Adaptation des photomultiplicateurs PM 150 AVP au préamplificateur initialement prévu pour des 51 AVP.
- e) Modifications du montage des électrovannes (fig. 8): celles-ci étaient à l'origine toutes soudées sur les tuyaux, dans le souci de garantir la meilleure étanchéité. En cours d'utilisation, le fonctionnement de ces électrovannes s'est trouvé altéré par la présence de vapeur d'huile dans le CO₂ prélevé et la nécessité de leur nettoyage est apparue. Pour le réaliser, toutes les électrovannes ont été rendues démontables, réacteur en pression, par une fixation à bride à joint.

VI. RÉSULTATS OBTENUS

Les quatre années d'exploitation de la DRG constituent un test particulièrement probant de la qualité de cette installation. Elle a permis de déceler dans d'excellentes conditions de sécurité, les 12 ruptures de gaines qui se sont manifestées depuis le démarrage du réacteur.

Quatre ruptures de gaines ont été particulièrement rapides et violentes. Les taux de comptage maximum enregistrés ont été, dans ces cas, de 1000,

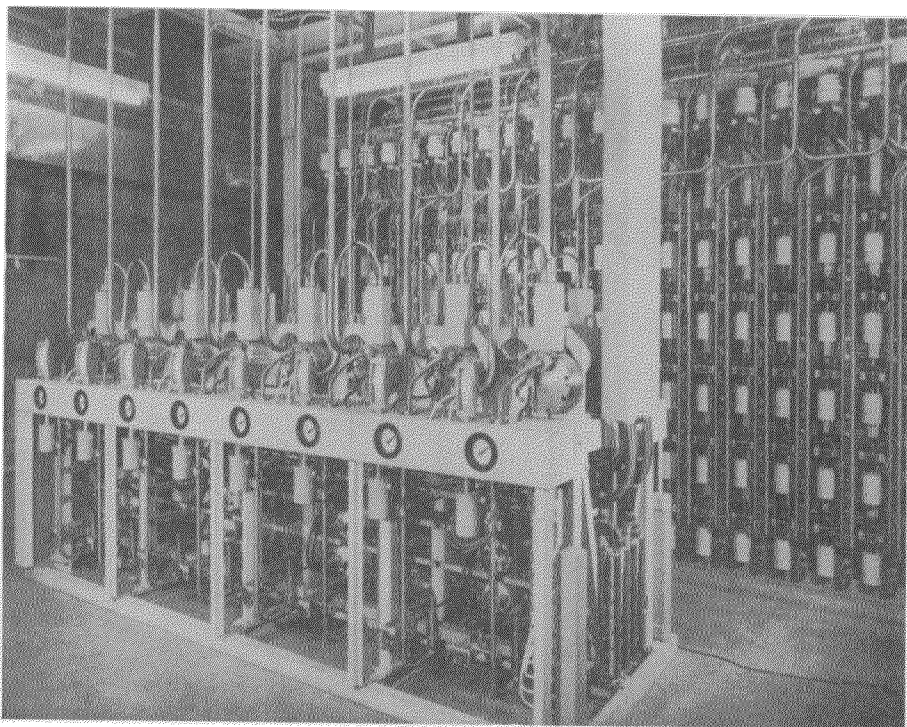


Figure 8

Baie d'électrovannes. Au premier plan, châssis supportant les détecteurs, les débitmètres et les indicateurs de pression.

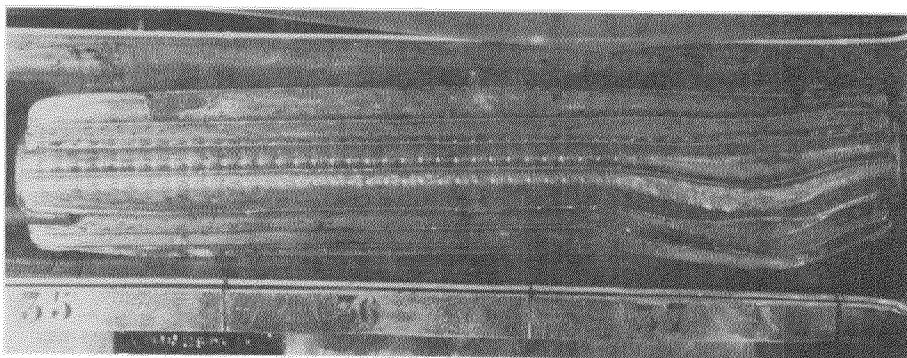


Figure 9

Cartouche avec rupture de gaine. Le gonflement de la gaine est dû à la présence d'oxyde d'uranium.

1540, 5550 et 10 000 chocs/s, pour des évolutions allant de quelques secondes à 10 min. Dans chaque cas, la détection et l'identification du canal ont été suffisamment rapides, pour qu'après déchargement, aucune pollution

résiduelle du réacteur ne soit constatée. La figure 9 représente la cartouche ayant donné un comptage de 5550 chocs/s.

Les huit autres ruptures de gaines enregistrées ont eu des évolutions beaucoup plus lentes - de 1 h à 4 j - et les signaux obtenus sont toujours restés inférieurs à 750 chocs/s.

VII. ENTRETIEN

Circuits pneumatiques

L'ensemble des circuits pneumatiques nécessite une fréquence d'entretien extrêmement réduite. Les électrovannes sont, en moyenne, démontées et nettoyées une fois tous les deux ans.

Les cartouches filtrantes placées sur le circuit CO₂, n'ont jamais accusé d'augmentation d'activité mesurable et n'ont été remplacées qu'une fois depuis la mise en service de l'installation.

Les groupes hélico-compresseurs sont soumis à des révisions mécaniques au cours des arrêts normaux d'entretien du réacteur.

A titre de «maintenance» un ensemble complet de rotors avec ses garnitures, est disponible.

Installation électromécanique

L'entretien systématique des armoires de relaying est réalisé semestriellement et n'appelle aucune observation particulière.

Chaîne électronique

L'installation électronique, présentée sous forme d'éléments débrochables - tiroirs amplificateurs, alimentation, etc. - permet un entretien systématique préventif particulièrement simple et efficace.

Les ensembles scintillateurs-photomultiplicateurs ont donné un certain nombre de difficultés, qui ont pu être levées, en partie, par la réduction d'épaisseur des scintillateurs.

A ce jour, la statistique des pannes relevées sur l'installation de DRG donne environ:

- 80% d'incidents d'une durée inférieure à 5 min,
- 20% d'incidents d'une durée comprise entre 5 et 15 min.

Ces incidents n'entraînent, en général, qu'une indisponibilité partielle de la DRG (un groupe intéressé sur 12).

Depuis la mise en service de l'installation un seul cas de panne, supérieure à 30 min, a nécessité l'arrêt volontaire du réacteur, conformément aux consignes de sécurité en vigueur.

VIII. CONCLUSION

Réalisée à l'état de prototype en 1958, la DRG de G3 a démontré, depuis, que les principes fondamentaux qui avaient guidé ses réalisateurs, sont par-

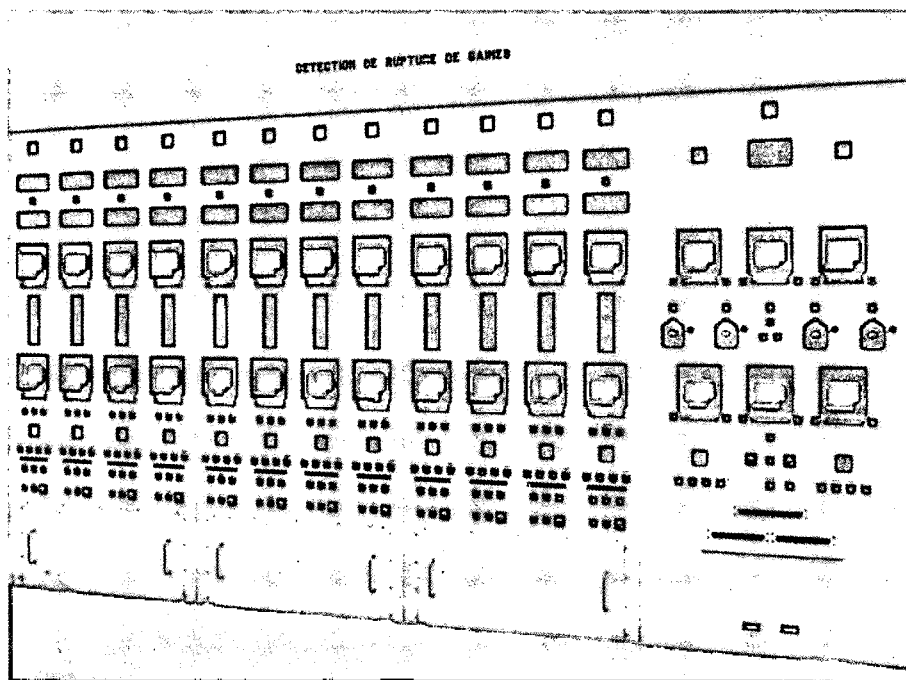


Figure 10

Vue d'ensemble de la DRG en salle de commande.

faitement adaptés aux besoins de sécurité et de rapidité des réacteurs modernes.

Ses caractéristiques principales: a) automatisme intégral, b) centralisation des informations en salle de commande (fig.10) et c) principe de mesure par collection électrostatique sur électrode tournante, restent encore à la base des réalisations en cours pour les prochains réacteurs et seul le traitement analogique des informations qu'elle réalise pourrait être abandonné au bénéfice de la technique plus récente et aux possibilités beaucoup plus grandes du traitement des informations par des ensembles digitaux.

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SURVEILLANCE AUTOMATIQUE DES TEMPÉRATURES AUX RÉACTEURS G2 ET G3

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Abstract — Résumé — Аннотация — Resumen

AUTOMATIC TEMPERATURE CONTROL IN THE G2 AND G3 REACTORS. The G2 and G3 reactors are equipped with electronic computers which process the data received from 1200 thermocouples at the channel exits. The author describes the various parts of the thermometric and data-processing systems.

SURVEILLANCE AUTOMATIQUE DE TEMPÉRATURES AUX RÉACTEURS G2 ET G3. Les réacteurs G2/G3 sont équipés de machines à calculer électroniques traitant les informations reçues de 1200 thermocouples disposés à la sortie des canaux. L'auteur décrit les différentes parties de l'appareillage de prise de température et de traitement des données.

АВТОМАТИЧЕСКОЕ НАБЛЮДЕНИЕ ЗА ТЕМПЕРАТУРОЙ РЕАКТОРОВ G2 И G3. Реакторы G2/G3 оборудованы электронными счетными машинами для обработки информации, получаемой с 1200 термопар, расположенных на выходе каналов. Дается описание различных частей аппаратуры по определению температур и обработке данных.

VIGILANCIA AUTOMÁTICA DE LAS TEMPERATURAS EN LOS REACTORES G2 Y G3. Los reactores G2 y G3 están equipados con calculadoras electrónicas que elaboran las informaciones recibidas de 1200 termopares dispuestos a la salida de los canales. El autor describe las diferentes partes del sistema de registro de temperatura y de elaboración de los datos obtenidos.

I. INTRODUCTION

Les réacteurs G2/G3 sont équipés de machines à calculer électroniques pour le traitement des informations reçues des 1200 thermocouples disposés à la sortie des canaux.

Ces appareils sont de 2 types:

- a) un équipement de surveillance: installation de la Compagnie industrielle des téléphones (CIT)
- b) un équipement de traitement des informations: installation de la Société d'études et de recherches électroniques (SEREL).

L'équipement de surveillance assure le contrôle des températures de CO₂ à la sortie des 1200 canaux et ceci à une cadence qu'il est pratiquement impossible d'atteindre manuellement (65/s). La surveillance est donc pratiquement continue et cette fonction de la machine est liée à une alarme lumineuse et sonore à 2 seuils réglables; le contrôle de la valeur se fait sur machine à écrire automatique.

L'équipement de traitement des informations est un véritable calculateur électronique qui permet le traitement et la présentation des températures des 1200 canaux d'une façon séquentielle déterminée par l'exploitation. D'autres paramètres liés au fonctionnement du réacteur sont traités de la même façon.

Les liaisons du calculateur avec «le monde extérieur» sont beaucoup plus étendues que celles de l'appareil de surveillance, mais elles ne sont pas automatiques. Ce sont essentiellement des présentations de cartes reproduisant la topologie de la face de chargement du réacteur et comportant des températures brutes données par l'appareil de surveillance, des résultats de calculs sur ces températures ou sur d'autres paramètres (activité et irradiation des canaux). Toutes ces valeurs sont présentées par l'intermédiaire d'une machine à écrire automatique.

Les 2 équipements sont techniquement assez semblables. Leur principe de fonctionnement est basé sur la technique des impulsions et l'organisation logique des circuits électroniques transistorisés qui transforment et traitent les informations en code binaire (pur ou décimal).

II. DESCRIPTION DES PRISES DE TEMPÉRATURE DU CO₂

On a placé un thermocouple cuivre-constantan dans le CO₂ à la sortie de chaque canal. Ces 1200 thermocouples sont reliés à une armoire de répartition reproduisant la topologie de la face Nord du réacteur et placée sur la plate-forme de chargement. A chaque thermocouple correspond une douille dans laquelle on peut brancher un enregistreur à une voie permettant de procéder à un contrôle sur place d'une température et éventuellement de suivre les opérations faites sur un canal.

Correction des températures brutes

Il est apparu que les valeurs des températures brutes étaient entachées d'une légère erreur due à une perturbation du débit pendant les prélèvements de la Détection de rupture de gaine. Une correction systématique est alors introduite dans la mesure qui, sans elle, serait erronée par défaut. Cette correction est déterminée canal par canal et conservée en mémoire dans le calculateur. Ces corrections sont très variables d'un canal à l'autre, mais toujours positives, elles sont donc nécessaires et d'ailleurs vérifiées par des thermocouples de contrôle que l'on dispose provisoirement de façon différente à la sortie du canal.

III. ÉQUIPEMENT DE SURVEILLANCE

1. But de l'appareil

Cet appareil est prévu pour 2 fonctions distinctes:

- a) surveillance aveugle des températures et impression automatique des mesures sur téléscriptrice en cas de dépassement et
- b) appel de certaines voies de mesure sur demande de l'opérateur jusqu'à concurrence de 16 résultats.

2. Disposition pratique générale.

1200 paires Cu-K sont raccordées sur l'armoire de répartition décrite ci-dessus et ramenées dans une salle dans laquelle sont disposés dans l'ordre de la chaîne (fig. 1):

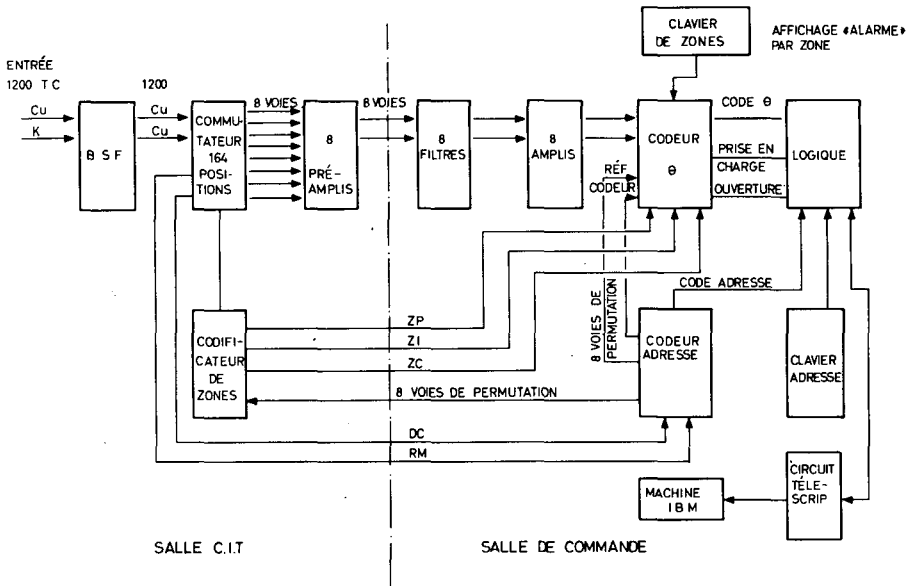


Figure 1

Synoptique de l'appareil de surveillance.

- une boîte de soudure froide (BSF) (fig. 2),
- une armoire de régulation de température de la BSF,
- un commutateur mécanique rotatif (fig. 2),
- 8 préamplificateurs de signaux température,
- un codificateur de zones.

La boîte de soudures froides est une enceinte thermostatée où sont groupées toutes les soudures froides des thermocouples. La température y est maintenue à $120 \pm 0,2^\circ\text{C}$.

La commutation des forces électromotrices de couples s'effectue en 2 stades:

- une commutation mécanique à bas niveau des voies de mesure par groupe de 8 (fig. 3),
- une commutation électronique à niveau élevé de chacune des 8 voies de mesures.

Le temps de commutation total est 65,6 s. Les 8 signaux sont amplifiés une première fois (gain 100).

Le codificateur de zone effectue la répartition des 8 canaux d'une position quelconque du commutateur entre les 3 zones du réacteur.

Les signaux ainsi amplifiés et répartis sont envoyés dans la salle de commande principale du réacteur où l'on trouve:

- 8 filtres d'entrée,
- 8 amplificateurs,
- un commutateur codeur adresse,
- un codeur température,
- un circuit «logique»,

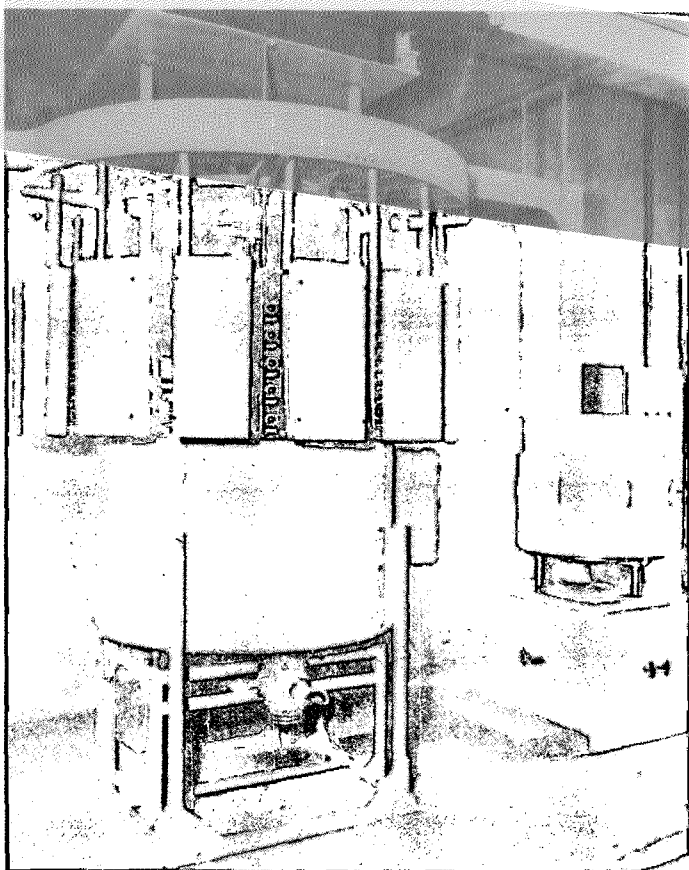


Figure 2

Appareil de surveillance,
(au premier plan: boîte de soudure froide, à droite: commutateur rotatif).

- un circuit de sortie téléscriptrice,
- des circuits annexes.

A l'entrée des armoires de la salle de commande, les tensions de mesures sont filtrées et amplifiées à nouveau (gain 100) avant d'être dirigées vers les circuits de commutation électronique et de codage (pour les températures normales de fonctionnement, la tension de sortie est alors de l'ordre de 100 V).

- Le commutateur codeur électronique remplit 2 fonctions principales:
- l'élaboration du code d'adresse des canaux auxquels est affectée la représentation alphanumérique de l'adresse de ces canaux,
- la fourniture des signaux de commutation rapide des 8 voies en niveau élevé et le codage de celles-ci.

Le codeur température transforme la tension analogique donnée par les amplificateurs et proportionnelle à la température en une information binaire-décimale. Ce codeur est donc une combinaison de machine analo-

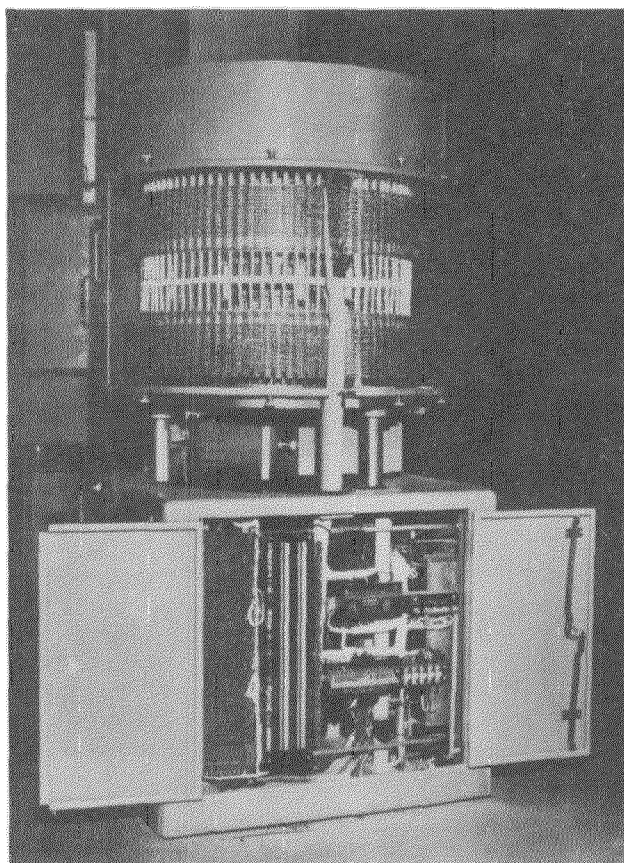


Figure 3

Commutateur rotatif.

On distingue en haut le raccordement des thermocouples par groupes de 8 et, dans le socle, les préamplificateurs et les organes de codificateurs de zone.

gique et de machine numérique; il traduit électriquement la fonction «tension-degré» d'un thermocouple Cu-K laquelle est la somme d'une droite et d'une parabole à partir de 120°.

Le circuit logique est le circuit «pensant» de l'explorateur. C'est lui qui prend la décision de la suite des opérations suivant le type d'information qui se présente à lui: passage ou disparition d'un canal en alarme, appel ou effacement d'un canal au clavier. Le circuit logique reçoit toutes les informations et tous les codes (adresse et température) qu'il garde en mémoire dans des tores à ferrite et se tient à la disposition des circuits de la téléscriptrice lorsque cette dernière devra communiquer avec le «monde extérieur», c'est-à-dire inscrire en clair les grandeurs codées et stockées dans les mémoires adresses et températures.

3. Exploitation de l'appareil

L'appareil de surveillance comporte un clavier de zone sur lequel on peut inscrire par zone du réacteur (centrale, intermédiaire et périphérique) une température de référence qui peut être, par exemple, la température limite de sortie du CO₂ dans un canal de la zone considérée.

Lorsqu'un canal dont la température de sortie est supérieure à la référence, est détecté par le commutateur, un voyant vert «ALARME» s'allume. Si ce canal dépasse de 20°C la température de référence, un voyant rouge «ALERTE» s'allume. Ces allumages sont doublés d'un coup de klaxon.

L'adresse et la température du canal ainsi détecté s'inscrivent automatiquement en rouge sur la téléscriptrice. Si le canal reste en alarme ou en alerte, sa température réapparaît en rouge toutes les minutes sur la téléscriptrice.

C'est la fonction «surveillance» de la machine.

L'appareil comporte un 2^{ème} clavier, appelé clavier d'adresse sur lequel l'opérateur peut inscrire l'adresse d'un canal dont il désire connaître la température. Dès sa détection par le commutateur, l'adresse et la température de ce canal s'inscrivent en noir sur la téléscriptrice.

La température réapparaît en noir toutes les minutes jusqu'à effacement de l'adresse du canal.

C'est la fonction «contrôle» de la machine.

Le nombre total de canaux pouvant être pris en charge par la machine par dépassement ou appel manuel, est de 16. L'appareil reçoit les tops horaires d'une pendule électrique extérieure. L'inscription d'une adresse ou d'une température sur la téléscriptrice est toujours précédée de l'inscription de l'heure (heures et minutes). L'erreur absolue sur les températures mesurées est inférieure au degré, mais la machine ne donne la valeur qu'à 1°C près sans tenir compte des décimales. Un spécimen des résultats obtenus sur la téléscriptrice est donné ci-après (fig. 4).

IV. ÉQUIPEMENT DE TRAITEMENT DES INFORMATIONS

1. But de l'appareil

L'appareil de traitement des informations (type 1001) est un calculateur numérique qui, suivant un programme bien défini, est chargé:

- a) de stocker un certain nombre d'informations données par l'appareil précédent et de paramètres introduits manuellement par l'opérateur;
- b) d'effectuer des opérations sur ces données et communiquer ensuite les résultats suivant des présentations définies par l'opérateur (fig. 5 et 6).

2. Programme actuellement en service

a) Paramètres d'entrée

- 1200 températures en code binaire données par l'appareil de surveillance;
- code horaire, heures et minutes en binaire données par l'appareil de surveillance,

Instant de commutation		h min	Fonction <<SURVEILLANCE, ALARME, ALERTE>>	Fonction <<CONTROLE, APPEL CLAVIER>>
09 35	E26A		← Canal en dépassement <<ALARME>> (seuil à 365)	
09 36	<u>368</u>		← Température correspondante	← Canal appelé au clavier
09 37	<u>367</u>			← Température correspondante
09 38	<u>366</u>			
09 39			← Le Canal E26A n'est plus en alarme	
09 40	W13L		← Canal en dépassement <<ALERTE>> (seuil à 365 + 20)	
09 41	<u>386</u>		← Température correspondante	
09 42	<u>385</u>			
09 43	<u>383</u>		← Température correspondante, le canal reste en <<ALARME>>	
09 44	<u>380</u>			
09 45	<u>375</u>			
09 46	<u>370</u>			
09 47	<u>366</u>			← Le canal E190 passe en <<ALARME>> (seuil à 333)
09 48			← Le canal n'est plus en <<ALARME>>	
09 49				
09 50				← Le canal E190 n'est plus en <<ALARME>>
09 51				
09 52				
09 53				← Le canal a été effacé au clavier
09 54				

Nombre total de canaux en <<ALARME, ALERTE OU APPEL CLAVIER>>

Figure 4

Spécimen des résultats obtenus sur la téléscriptrice de l'appareil de surveillance.

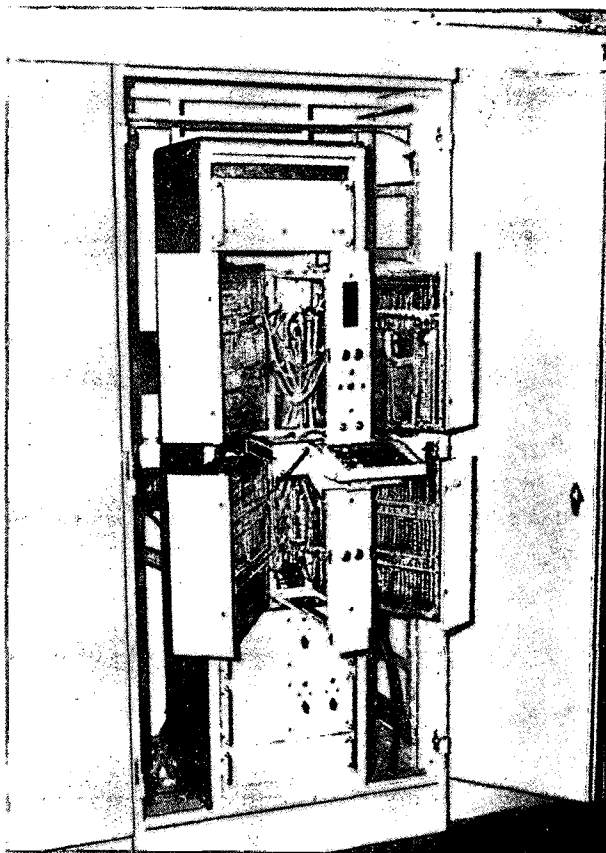


Figure 5

Appareil de traitement des informations.

- 1200 corrections introduites manuellement (valeur positive de 0 à 30°);
- 1200 numéros de zones introduits manuellement et assurant la répartition des 1200 canaux en zones (5 maxim.) suivant la configuration adoptée sur le réacteur;
- 5 températures de références de zones introduites manuellement et fixant un seuil pour chaque zone;
- 2 températures d'entrée du CO_2 en pile (θ_e) introduites manuellement et correspondant aux 2 zones d'entrée du réacteur;
- 1 origine de calcul statistique des écarts: début du tracé des histogrammes, introduit manuellement;
- 2 échelles de calcul statistique des écarts: élongations des histogrammes introduites manuellement;
- 1 coefficient d'échauffement relatif (r); paramètre liant la température de sortie à la température de gaine et introduit manuellement;
- 3 débits moyens de gaz (q), (zones centrale, intermédiaire et périphérique) introduits manuellement;

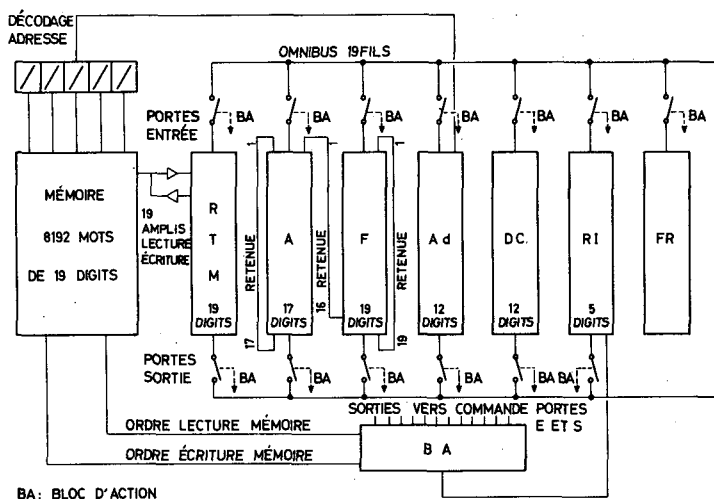


Figure 6

Synoptique du calculateur.

- 2 coefficients d'enthalpie (entrée et sortie) introduits manuellement et permettant la transposition température-enthalpie à l'entrée et à la sortie du réacteur;
- 1 coefficient de correction de puissance, introduit manuellement et permettant de tenir compte approximativement des fuites inaccessibles de puissance dans le circuit secondaire;
- 1200 coefficients d'activité DRG (A) introduits manuellement et qui correspondent à la contamination ou surface d'uranium nue, contenue dans les parois de chaque canal;
- 1200 irradiations (I) introduites manuellement.
- 20 coefficients d'irradiation (K_4), un par couronne de canaux autour du centre, introduits manuellement et permettant la tenue à jour de l'irradiation totale;

b) Résultats à la sortie

Les résultats sont présentés sous forme de tableaux, de courbes ou de cartes (topologie de la face du réacteur) dont les spécimens sont donnés ci-après. La machine permet d'obtenir les résultats suivants:

- carte de valeurs brutes, valeurs données par l'appareil de surveillance et traduites en décimal;
- carte des corrections, contrôle des valeurs introduites manuellement;
- carte des valeurs corrigées, addition algébrique des valeurs brutes et des corrections;
- carte des numéros de zones, contrôle des valeurs introduites manuellement;
- carte des écarts bruts, soustraction de la valeur de référence d'une zone déterminée et de la valeur brute;

- carte des écarts corrigés, soustraction de la valeur de référence d'une zone déterminée et de la valeur corrigée (fig. 7);
- carte des températures de gaine, calcul de la température de gaine fictive suivant la formule: $\theta_g = r(\theta_s - \theta_e) + \theta_e$ dans laquelle θ_s est la température de sortie en valeur corrigée;
- carte des dérivées de températures, soustraction algébrique des températures en valeurs corrigées entre 2 dates déterminées.

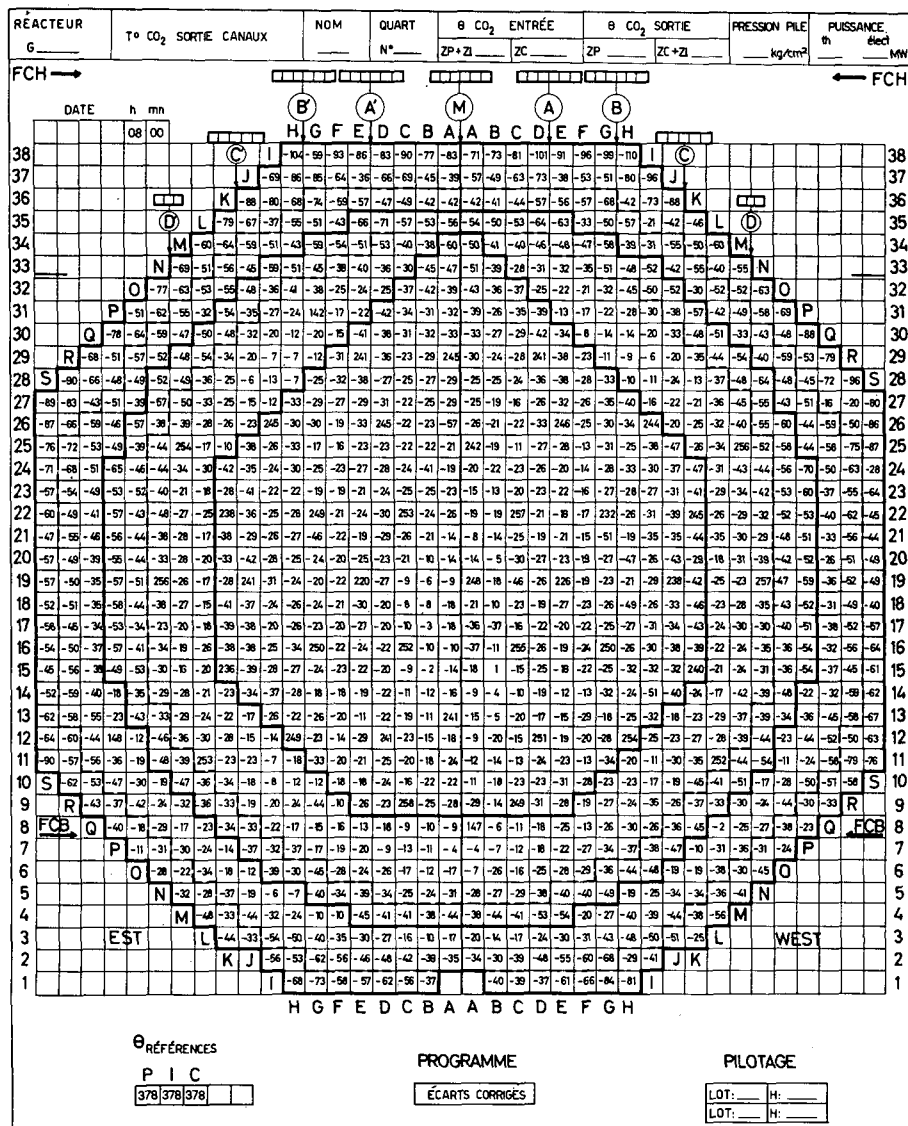


Figure 7

Spécimen d'une carte des « écarts corrigés » obtenue par l'appareil de traitement des informations.

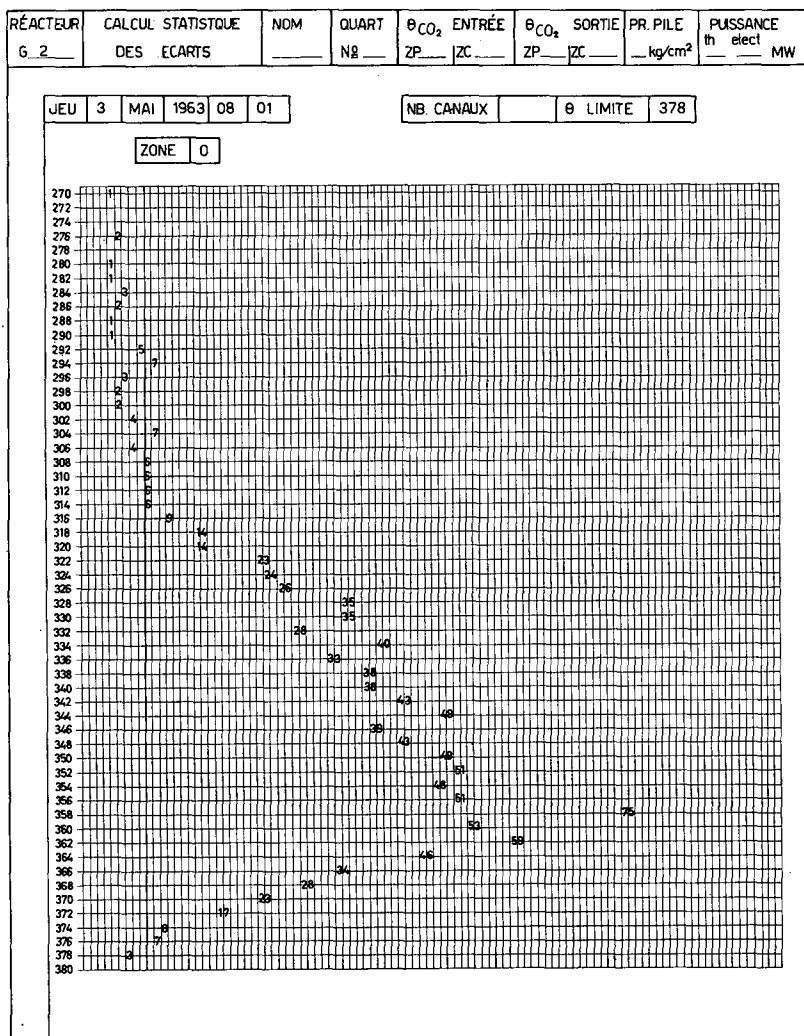


Figure 8

Spécimen d'un histogramme obtenu par l'appareil de traitement des informations.

- carte des activités DRG, signaux de bruits de fond donnés par la DRG, pour une puissance totale et un débit canal, déterminés suivant la formule: $N = AW/Q$, Q étant un débit volume prédéterminé par la machine dans ce même programme à partir des paramètres q et θ_s ;
- carte des irradiations moyennes, irradiations moyennes des 28 cartouches par canal se trouvant dans le flux à une énergie intégrée déterminée par l'opérateur;
- calcul statistique des écarts, calcul du nombre de canaux par zone ou pour l'ensemble des zones dont la température en valeur corrigée, est

comprise dans une fourchette déterminée. Les résultats sont présentés sous forme de courbe en cloche ou histogramme (fig. 8); moyenne des températures par zone et par quadrant, calcul de la moyenne arithmétique des températures en valeurs corrigées, des canaux d'une zone et d'un quadrant déterminés, ainsi que du quadrant complet. Présentation sous forme de tableau (fig. 9);

RÉACTEUR G 2

MOYENNES DES θ_{CO_2} SORTIE CANAUX

					H	mm			
					11	28			
					ZONE 1	ZONE 2	ZONE 3	MOYENNES	
					QUADRANT EH	322 21	345 49	353 73	340 47
					QUADRANT WH	326 76	345 07	354 33	342 05
					QUADRANT EB	335 85	347 46	355 09	346 13
					QUADRANT WB	336 60	344 64	353 83	345 02

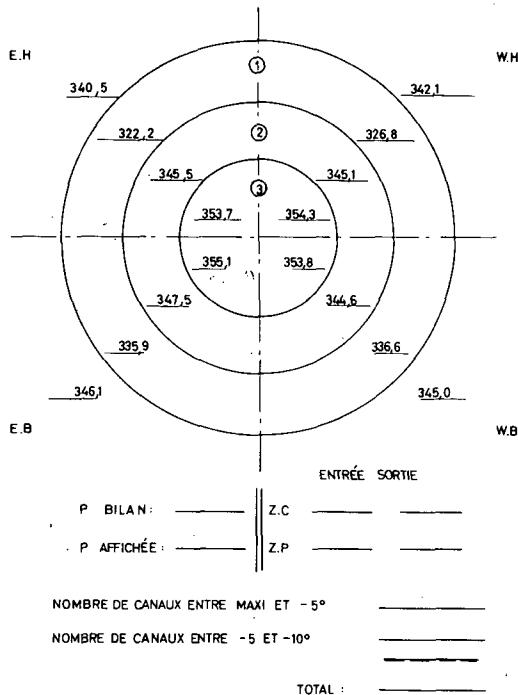


Figure 9

Spécimen des moyennes de températures obtenues par l'appareil de traitement des informations.

- moyenne des irradiations par zone et par quadrant, calcul de la moyenne arithmétique des irradiations des 28 cartouches par canal d'une zone et d'un quadrant déterminés, ainsi que du quadrant complet. Présentation sous forme de tableau;
- diagramme de répartition de puissance; calcul de la puissance thermique dégagée par couronne et par quadrant et de la puissance thermique suivant la formule;

$$W_R = q_R (H_{\theta_s} - H_{\theta_e})$$

dans laquelle W_R est la puissance dégagée dans un canal situé au rayon R et H_{θ_s} et H_{θ_e} les enthalpies de sortie et d'entrée correspondant aux températures θ_s et θ_e de sortie et d'entrée et précalculée par la machine dans ce même programme. Ces résultats sont présentés sous forme de courbes déterminant l'allure du flux thermique radial dans les diamètres verticaux et horizontaux de la face du réacteur (fig. 10);

- démarrage automatique, calcul sur le code horaire pour démarrer l'enregistrement «valeurs brutes» et établir la carte «écarts corrigés» automatiquement à des heures prédéterminées sans risque d'oubli de la part de l'opérateur.

On peut à volonté changer le programme principal de fonctionnement et un programme de contrôle secondaire permet de le vérifier en permanence sur téléscriptrice.

3. Disposition pratique générale

Le calculateur se présente sous la forme d'une baie électronique qui reçoit les informations codées en provenance de l'appareil de surveillance (fig. 5): les adresses, les températures, l'heure, le signal de début de cycle de commutation, le signal de référence du codeur de l'appareil de surveillance.

Ces informations sont reçues en parallèle sur 39 fils de liaison. Le calculateur est à programme enregistré avec exécution des opérations sur tous les chiffres en parallèle.

Il se compose d'un certain nombre de sous-ensembles fonctionnels classiques d'un calculateur industriel (fig. 6): un bloc mémoire et ses circuits annexes, les organes de calcul et de manipulation de l'information, le bloc de commande avec son clavier d'exploitation associé à la disposition de l'opérateur et les organes périphériques.

Un dispositif d'entrée à bande perforée est associé au calculateur augmentant considérablement la capacité de la mémoire électronique et permettant le stockage de programmes de calcul utilisés peu fréquemment.

a) Bloc mémoire

La mémoire est à tores de ferrite et d'une capacité de 8192 mots de 19 digits. Elle peut contenir aussi bien des données d'adresse que des résultats de mesures.

Les opérations sont exécutées en binaire sur ces mots de 19 digits (plus 1 digit de sécurité): 13 digits représentent l'adresse et 7 digits représentent l'instruction d'opération (dont 1 digit de sécurité).

b) Organes de calcul

Ils comprennent 8 registres spécialisés chacun pour des opérations élémentaires déterminées;

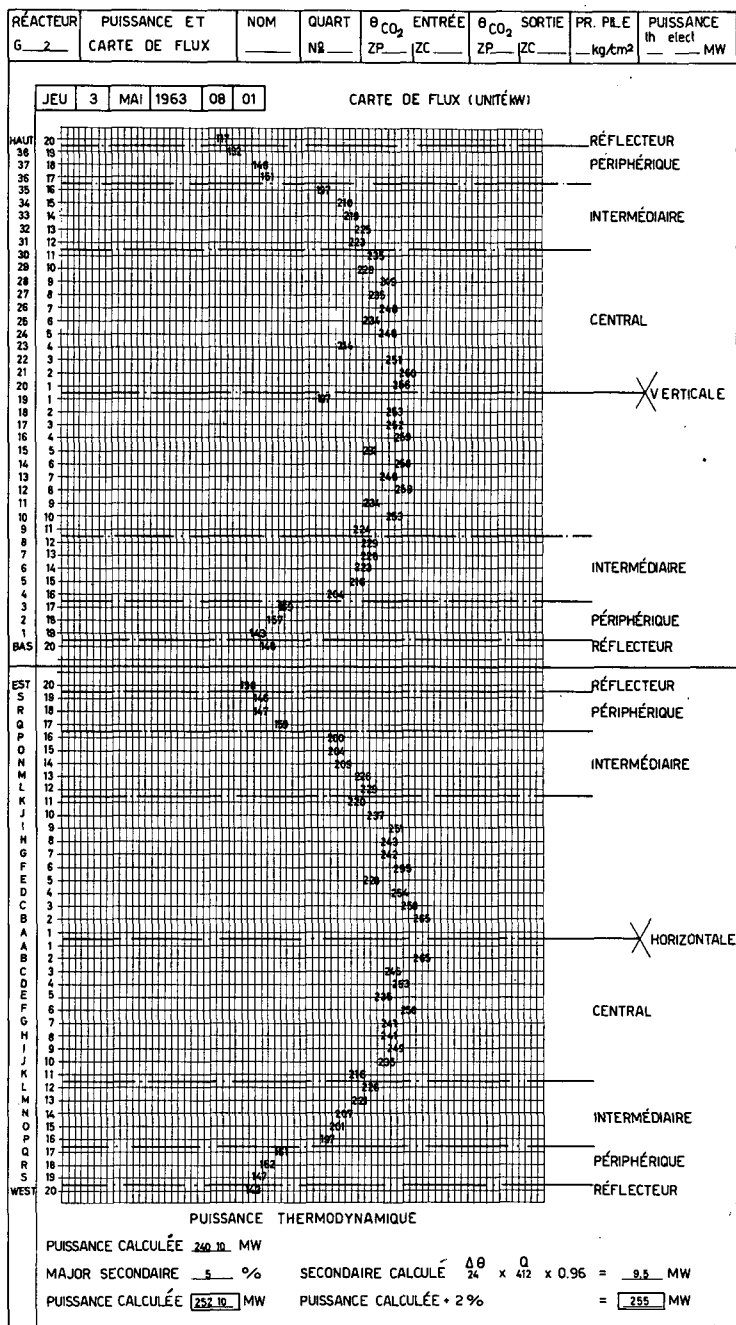


Figure 10

Spécimen d'un diagramme de répartition de puissance thermique
 obtenu par l'appareil de traitement des informations.

Cette courbe représentant des moyennes de puissances unitaires fait apparaître des dispersions
 dues à la présence des canaux absorbants dans les couronnes considérées.

Registre	: additions
Registre F	: filtrages
Registre DC	: décalages
Registre I	: instructions de calculs
Registres P et N:	(faux registres) compléments
Registre RTM	: tampon mémoire
Registre Ad	: adresses

Ces 2 derniers registres sont en liaison directe avec la mémoire, les autres y sont branchés suivant le calcul à effectuer par le bloc d'action ou le bloc de commande. Les calculs sont effectués en virgule fixe, les facteurs varient de 0 à 64 000 en valeur absolue.

c) Bloc de commande

Il ouvre et ferme les portes d'entrée et de sortie des 6 premiers registres et envoie les ordres d'écritures et de lecture à la mémoire.

Le clavier d'exploitation contient l'électronique de périphérie correspondant aux touches de composition d'adresses, de valeurs, de numéros de zone et de choix de programme. Cette électronique effectue la transposition de ces compositions manuelles en code binaire assimilable par le calculateur.

d) Organes périphériques

Il contiennent l'électronique chargée de l'adaptation des entrées et des sorties du calculateur: adaptation de l'appareil de surveillance et commande téléscriptrice et sont destinés à l'entrée des programmes et à la sortie des résultats.

e) Adaptation « bande perforée »

Une machine FLEXOWRITER, type SDP 12 pouces, équipée d'un lecteur de bande 8 trous, peut s'adapter sur l'un ou l'autre des calculateurs de chaque réacteur afin de leur introduire les informations programmées sur les bandes. Ces programmes sont composés par l'opérateur sur une machine à écrire électrique et transposés immédiatement sur bande par un perforateur 8 trous intégré à la FLEXOWRITER.

La fréquence du générateur de rythme est de 100 kHz ce qui donne une idée de la rapidité d'exécution des opérations élémentaires:

Addition	: 60 μ s
Soustraction	: 80 μ s
Division	: 500 μ s

V. FONCTIONNEMENT ET ENTRETIEN

L'appareil de surveillance n'a donné lieu à aucune panne importante sauf des dérèglages des circuits d'étalonnage dus aux variations importantes de température extérieure. Les transistors utilisés sont en effet très sen-

sibles à ces variations et supportent difficilement des températures élevées (supérieures à 25°C).

Au premier mois de fonctionnement, plusieurs pannes ont été relevées sur l'appareil de traitement des informations: quelques-unes sur l'électronique même, défaut de montage ou faiblesse des éléments de circuits; la plupart sur la téléscriptrice qui suit difficilement la cadence imposée par le calculateur. Ces défauts ont été à peu près repris et l'exploitation de l'appareil est devenue normale et satisfaisant vers le 1^{er} avril 1962.

Après un an d'utilisation pratiquement continue, on note:

- a) des petites pannes provoquant une indisponibilité de l'ordre de la journée (défaut de fonctionnement des bascules des registres provoqué par le déséquilibre dû au vieillissement d'un des deux transistors, électromécanisme de la téléscriptrice); et
- b) des grosses pannes provoquant une indisponibilité de plusieurs jours de l'appareil (dérégulation des courants mémoires, chauffage excessif et séchage des condensateurs dans les alimentations, chauffages excessifs des transistors provoquant leur détérioration en série).

De fortes perturbations sont apparues dans les mémoires de calculs lors des variations importantes de température ambiante (janvier - février 1963) avec détérioration d'un grand nombre d'éléments de circuits (transistors). Une étude de thermostatage intégral est actuellement en cours; la température idéale est comprise entre 18 et 22°C.

On peut estimer à 99% le facteur de disponibilité de l'appareil et à 95% celui de l'appareil de traitement des informations.

Ce facteur de disponibilité ne peut être atteint qu'avec une équipe de techniciens (2 à 3 personnes) connaissant parfaitement les circuits électroniques miniaturisés et la technique des impulsions.

VI. PERSPECTIVES D'AVENIR

Il est évident que disposant d'un calculateur ayant d'immenses possibilités de calculs, les différents modes d'exploitation définis ne sont pas limitatifs et nous pouvons déterminer des analyses complémentaires et des calculs statistiques qui peuvent s'avérer nécessaires pour la conduite des réacteurs.

On peut, d'autre part, envisager l'amélioration de la présentation des résultats sur carte. Les cartes de température à la sortie des canaux sont actuellement colorées à la main de façon à faire apparaître la dispersion de ces températures. A l'heure actuelle l'existence de tubes cathodiques «trichromes» permet d'envisager la sortie de cartes de températures, directe en couleur. L'obtention de ce résultat est réalisée au moyen d'un balayage spécial du tube qui superposera à un balayage «ligne» du type télévision; un balayage sinusoïdal à haute fréquence qui couvre un «carré» représentant un canal. Les informations de «couleur» proviennent de la combinaison de tension de 3 wehnelt conformément à un code choisi. Les écarts de température sont lus directement dans une mémoire à tores auxiliaires, à une cadence d'un mot toutes les 15 μ s permettant la présentation d'une carte en un temps approximatif de 1/50^{ème} de seconde, évitant ainsi le scintillement.

VII. CONCLUSION

Les deux appareils non prévus dans le projet des réacteurs G2/G3, sont devenus rapidement indispensables pour l'exploitation rationnelle de ces réacteurs.

En effet, bien que les températures de sortie des canaux ne représentent pas la totalité des informations nécessaires à la conduite d'un réacteur, elles constituent en régime stable, l'accès facile à un grand nombre de paramètres importants du cœur de la pile, paramètres pratiquement inaccessibles directement.

D'autre part, les temps de réponse de ces appareils étant relativement courts, ils sont encore très utiles en régimes transitoires du réacteur où les déplacements des zones de températures sont fonction du mouvement des barres de contrôle.

ORGANISATION ET INSTRUCTION DU PERSONNEL AUX RÉACTEURS G2 ET G3

F. CONTE

CENTRE D'ÉTUDES NUCLÉAIRES DE MARCOULE, FRANCE

Abstract — Résumé — Аннотация — Resumen

ORGANIZATION AND TRAINING OF THE G2 AND G3 REACTOR PERSONNEL. This paper contains a detailed description of the organizational system associated with the G2 and G3 reactors (Operations Section, Maintenance and Works Service), together with a review of personnel training.

ORGANISATION ET INSTRUCTION DU PERSONNEL AUX RÉACTEURS G2 ET G3. On trouvera dans ce mémoire une description détaillée de l'organigramme des réacteurs G2 et G3 (Section d'exploitation, Service de l'entretien et des travaux etc.) ainsi qu'un aperçu concernant l'instruction du personnel.

ОРГАНИЗАЦИЯ И ИНСТРУКТАЖ ПЕРСОНАЛА, ЗАНЯТОГО НА РЕАКТОРАХ G2 И G3. В докладе дается подробное описание организационной структуры реакторов G2 и G3 (Отдел эксплуатации, Служба содержания и работ и т.д.), а также обзор инструктажа персонала.

ORGANIZACIÓN Y FORMACIÓN DEL PERSONAL EN LOS REACTORES G2 Y G3. La memoria describe detalladamente la organización de las operaciones en los reactores G2 y G3 (sección de explotación, sección de conservación y trabajos, etc.), así como una reseña del programa de adiestramiento del personal.

I. ORGANISATION DU PERSONNEL

Une remarque préliminaire s'impose avant de détailler l'organisation du personnel des réacteurs G2/G3.

Ces réacteurs sont implantés sur un site contenant des services aussi différents que le réacteur G1, les réacteurs G2/G3, l'usine d'extraction du plutonium, la station de traitement des effluents, l'atelier de taillage du graphite, l'empilement critique Marius, etc.

A ces services, groupés sous l'autorité d'une seule Direction, sont adjoints des services communs, à l'échelon du centre lui-même. Il s'agit des services généraux, administratifs, des programmes, immobiliers, de protection, de sécurité, médicaux, etc.

La présente communication ne traitant que de l'organisation des réacteurs G2/G3, il ne sera pas fait mention des services communs dont l'existence ne doit toutefois pas être perdue de vue, si l'on veut que l'exposé suivant reste compréhensible.

1.1. Service G2/G3

La totalité du personnel atteint 400 personnes, dont 18 ingénieurs (voir l'annexe I).

Trois sections principales se répartissent ce personnel ainsi qu'un groupe administratif. Deux sections semblables correspondent à l'exploitation proprement dite, assurant la marche des deux réacteurs. Une section d'entretien et de travaux répartit ses activités sur l'ensemble des deux réacteurs.

Groupe de travail

L'ensemble des ingénieurs, en plus de leurs travaux routiniers, forme un groupe de travail sous la direction du Chef de Service. Ce groupe de travail effectue toutes les études théoriques ressortant de la physique, de la neutronique, de la thermodynamique, nécessités par les besoins de l'exploitation. La plupart du temps, ces travaux sont menés par l'ingénieur dont la spécialité se rapproche le plus de la question traitée sous la forme «d'ingénieur chargé».

1. 2. Section exploitation G2 et G3 des réacteurs

Chaque réacteur est dirigé par un chef d'ensemble assisté de 4 ingénieurs et a sous ses ordres 105 personnes dont 50 techniciens.

En plus du secrétariat et de quelques agents en service de jour, la section est avant tout constituée par du personnel en service continu, chargé du fonctionnement du réacteur et des opérations de chargement.

A. Service continu au réacteur

Ce service comprend 5 équipes composées de la façon suivante:

- 1 chef de quart,
- 1 adjoint au chef de quart,
- 1 conducteur de pile,
- 1 conducteur des machines,
- 1 technicien électricien,
- 1 technicien électronicien,
- 1 spécialiste de la détection de rupture des gaines (DRG)
- 2 mécaniciens rondiers,
- 1 électricien rondier,

soit en tout 10 agents par équipe.

B. Service continu du chargement

Ce service comprend 5 équipes composées de la façon suivante:

- 1 chef d'équipe,
- 1 technicien du chargement,
- 1 technicien électricien,
- 1 ouvrier électricien,
- 2 ouvriers mécaniciens,
- 1 magasinier,
- 1 aide,

soit en tout 8 agents par équipe.

C. Roulement des équipes

Les changements de quart ont lieu pour les deux services ci-dessus aux heures suivantes: 5h 30, 13h 30, 20h 30.

Il suffit de 4 équipes pour assurer ce service, la quatrième étant au

repos quand les 3 autres travaillent. On peut donc poser la question de l'utilité de la 5ème équipe.

Il est nécessaire de disposer en plus des équipes de service continu du personnel de jour chargé de travaux de petit entretien, de réglages, mises au point, étalonnages, etc. ainsi que de travaux périodiques ressortant directement de l'exploitation, par exemple: cartes de flux, essais périodiques de sous-ensembles, permutation de certains organes. Il a paru intéressant de remplacer en partie cette équipe de jour importante par du personnel des services continus par roulement d'une semaine. Il en découle plusieurs avantages:

- toutes les 5 semaines, le personnel à poste fixe est «changé d'air» et peut quitter son pupitre ou son tableau,
- les agents circulant dans l'installation se mettent mieux au courant des événements passés ou des modifications en cours,
- les conducteurs de pupitres visitent ainsi obligatoirement les machines qu'ils commandent habituellement à distance.

De plus, la spécialisation trop poussée qui serait la leur, est atténuée et l'on peut disposer ainsi d'agents polyvalents.

Comme l'on verra plus loin, il est de plus, possible de distraire quelques heures dans la «semaine de jour» pour des réunions d'instruction et de formation professionnelle.

1.3. Section entretien et travaux

Cette section regroupe tout ce qui n'est pas directement la surveillance et l'exploitation des réacteurs.

Le chef de cette section est assisté de 5 ingénieurs, et a sous ses ordres 160 agents environ, dont 60 techniciens, contremaîtres ou chefs d'équipes.

A. Etudes

Le bureau d'études comprend 15 techniciens, il est assisté d'un bureau de dessin avec 10 dessinateurs.

Ce bureau d'études entreprend toutes les modifications ou les travaux d'aménagements en mécanique, électricité et électronique.

Il contrôle les fabrications, les essais, et les mises en route avant la livraison à l'exploitation.

B. Mécanique

Ce groupe très important comprend 70 personnes dont 10 contremaîtres ou chefs d'équipes.

Il comprend: un atelier et le personnel correspondant, des équipes d'entretien volantes et des équipes de services généraux.

C. Electricité et électronique

Ces groupes sont constitués de la même façon que le précédent, avec respectivement 35 et 25 personnes.

1. 4. Groupe administratif

Un cadre administratif groupe sous ses ordres 10 employés chargés de la gestion du personnel, des achats, du contrôle des travaux et des facturations, de l'établissement des prévisions.

II. INSTRUCTION DU PERSONNEL

2. 1. Connaissance des installations

La presque totalité des ingénieurs et techniciens employés actuellement sur nos réacteurs a suivi et souvent dirigé la construction, les essais et la mise en route. Par conséquent, la question capitale de la connaissance des installations n'a pas jusqu'ici posé de problèmes, l'encadrement du personnel étant parfaitement au courant de l'ensemble des appareillages.

Il convient de signaler malgré tout que des cours théoriques et pratiques ont été dispensés à ce personnel pendant toute la période de construction, cours très complets et circonstanciés. Mais ces cours s'adressant à des agents ayant sous les yeux les installations en cours de montage ne pouvaient pas être pris comme base d'un cours d'information destiné à du personnel «neuf». Nous n'en ferons donc pas état plus longuement.

2. 2. Instruction permanente

Il a été expliqué au chapitre précédent comment le roulement des équipes d'exploitation permet toutes les 5 semaines de disposer d'un certain temps pour l'instruction de l'équipe en «semaine de jour».

Trois matinées sont, à cette occasion, consacrées à l'instruction théorique. Les équipes d'exploitation en service continu bénéficient donc d'une moyenne de 10 h de cours par mois.

Ces cours donnés en général par les ingénieurs eux-mêmes traitent de la physique nucléaire, de thermodynamique et des spécialités intéressant ces agents.

L'annexe II donne à titre d'exemple le programme des cours de printemps 1962.

2. 3. Simulateur

Des cours d'instruction plus spécialisés que les précédents peuvent être institués chaque fois que l'occasion s'en fait sentir et en particulier avant ou après des expériences ou des manoeuvres spéciales.

De plus, des séances pratiques de manipulations sont effectuées par les agents chargés de pupitres ou de tableaux sur un simulateur de pile. Ce simulateur reproduit le pupitre principal de la salle de commande avec la totalité des instruments réels. Il est évident qu'il serait impensable de simuler intégralement tous les appareils en raison de leur nombre considérable. Seuls les principaux organes de commande (barres de contrôle, puissance, températures, débits, rotation des soufflantes, etc.) sont simulés

par un appareillage électrique et électronique programmé. Les multiples indicateurs ou organes auxiliaires sont simplement animés manuellement.

Il s'est avéré, à l'usage, que le personnel conserve à l'esprit la philosophie des manœuvres nécessitées par l'exploitation correcte des réacteurs. Mais, après la période de mise en route de 1959 et 1960, les manipulateurs se trouvaient devant des pupitres à regarder, sans manœuvres à effectuer, en raison de la stabilité déconcertante de ce type de pile. Il peut se passer des semaines entières sans qu'aucune manœuvre manuelle soit nécessaire.

On pouvait se demander si, à la longue, ces agents conserveraient le «doigté» du bon manipulateur.

C'est surtout pour cette raison, pour faire manœuvrer les agents des pupitres, que ce simulateur a paru nécessaire.

Les séances de simulation sont donc surtout des séances de «manipulation». Le grand nombre d'organes en cause rend ces séances du plus haut intérêt et le volume du pupitre permet, de surcroît, l'admission d'un nombre assez grand de «spectateurs» accroissant ainsi le rendement de l'instruction.

2.4. Cours de conducteur de pile

L'Institut national des sciences et techniques nucléaires à Saclay, a institué 2 sessions annuelles de cours spéciaux à l'usage des conducteurs de pile. Ces cours s'adressent aussi bien aux conducteurs de piles à graphite, qu'aux conducteurs de réacteurs à eau légère ou lourde.

Ces cours, malgré leur nom, sont en fait destinés à des chefs de quart en raison de leur niveau assez élevé. Ils s'adressent à du personnel déjà formé sur place et connaissant parfaitement les installations.

Précédés de 2 semaines d'«entraînement de l'esprit» sur place à Marcoule, ces cours durent 4 semaines. Le programme détaillé est donné en annexe III.

Ces cours, uniquement théoriques avec quelques manipulations sur des piles école, sont suivis d'un stage pratique de 4 autres semaines sur les réacteurs de Saclay ou de Grenoble. L'étudiant prend alors place aux commandes de ces réacteurs, différents des réacteurs de puissance de Marcoule, et peut ainsi illustrer la théorie des semaines précédentes.

En résumé, ces cours de conducteurs de piles, après 4 semaines d'études théoriques et 4 semaines de stage pratique, forment des agents aptes à piloter des piles à eau lourde, des piles piscines et des piles de puissance au graphite.

ANNEXE I

Dans cette annexe on trouvera une descriptions de l'organigramme du service G2/G3 (fig. 1).

ANNEXE II

Dans cette annexe on trouvera un aperçu de l'horaire réservé à l'instruction du personnel des réacteurs G2/G3 (Tableau I)

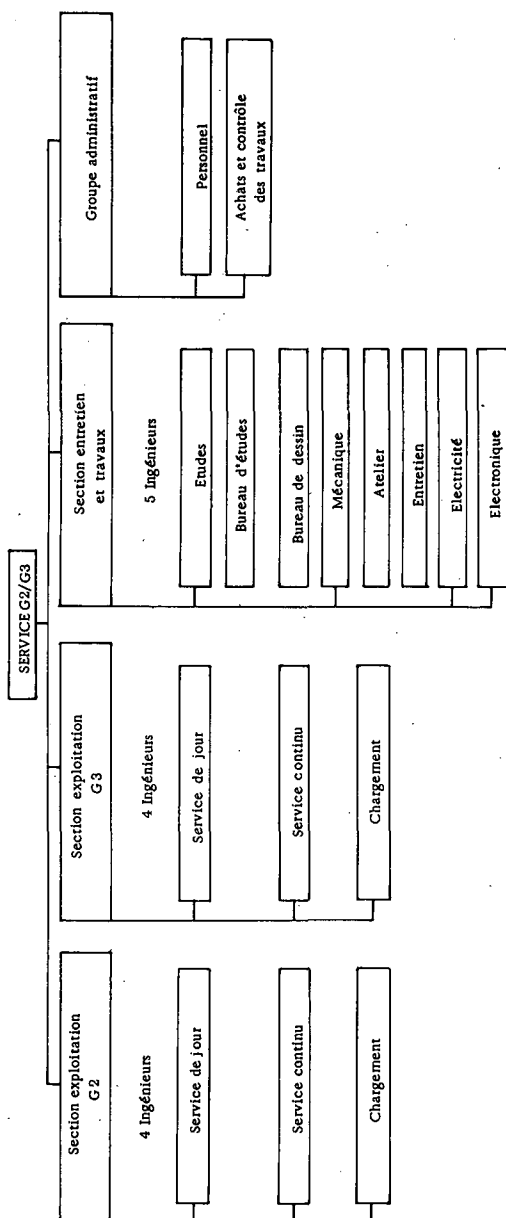


Figure 1

Organigramme du Service G2/G3

TABLEAU I
INSTRUCTION DES ÉQUIPES DE JOUR

	Lundi	Mercredi	Vendredi
1ère semaine Éléments de physique nucléaire	L'atome; la radioactivité, la fission	Les réacteurs nucléaires	Protection contre les radiations
2ème semaine Installations électro- niques	Rappels de la 1ère semaine	Mesures nucléaires et sécurité	Mesures thermiques : CIT et SEREI
3ème semaine Installations mécaniques	Rappels de la 2ème semaine	Circuit CO ₂ et machines	IRE
4ème semaine Installations électri- ques et chargement	Rappels de la 3ème semaine	Distribution électrique et automatismes	Chargement - déchargement
5ème semaine	Rappels de la 4ème semaine	DRC (G3)	DRG (G2)

Note: Les cours ont lieu de 8 h à 22 h dans la salle des conférences. La première semaine débutera le 22 Mars.

ANNEXE III

COURS DE CONDUCTEURS DE PILE A L'INSTN DE SACLAY

I. Notions fondamentales de physique nucléaire (3 séances)

1. Constitution de l'atome et du noyau.
Particules et rayonnement.
2. Réactions nucléaires - fission (généralités).
3. Transmutations en pile
(plutonium et formation de radioéléments.
Activation (sources naturelles ou artificielles).

II. Neutronique - généralités - Sections efficaces (6 séances)

1. La vie des neutrons dans la pile (2 séances)
Classification des neutrons d'après l'énergie.
Ralentissement - Absorption - Multiplication - Réaction en chaîne - Réflecteur.
Bilan neutronique sommaire - Réactivité.
Taille et masse critiques.
2. Description technologique d'une pile (1 séance)
3. Flux neutronique dans une pile (1 Séance)
Carte de flux - Mesures - Aplatissement - Déformations locales (barres).
4. Cinétique de la pile (1 séance).
Prompte criticalité.
Neutrons retardés - Période - Courbe de Nordheim.
Principe du contrôle : barres de contrôle et niveau d'eau, matériaux absorbants et poisons consommables.
5. Bilan de réactivité et évolution de la réactivité (1 séance).
Xénon - Samarium - Température - Vide - Pression.
Pureté des matériaux - Introduction des matériaux étrangers.
Evolution du combustible.

III. Les problèmes technologiques dans les réacteurs (7 séances)

1. Eléments combustibles - Gains - Température du combustible (2 séances).
2. Transfert de chaleur aux divers fluides caloporteurs utilisés (2 séances).
Répartition des températures - Points chauds en relation avec la courbe de flux.
Ebullition dans les piles à eau.
3. Echauffements locaux: protection, radioéléments, dispositifs en pile, échauffement γ (1 séance).
4. Problèmes chimiques dans les réacteurs. (2 séances)
Corrosion - Résistivité de l'eau.
Recombinaison - Epuration.

IV. Mesures (4 séances)

1. Etude générale des détecteurs de rayonnements (1 séance).
2. Les chaînes de mesure neutronique (2 séances).
Principe des chaînes à impulsion.
Sensibilité - Gammes de mesure - Recouvrement - Intégrateurs et différentiateurs.
Principe des chaînes à courant continu: chaînes linéaires - chaîne log.
Précisions comparée des chaînes.
Périodemètre - Fluctuations - Temps de réponse.
Enregistreurs.
Chaînes de santé.
3. Mesures de température et de débit (1 séance).
Puissance d'une pile - Relation flux-puissance - Etalonnage thermodynamique.

V. Dispositifs assurant la sécurité du réacteur (6 séances)

1. Dispositifs d'alerte.

Boucle de sécurité - Verrouillages.

Maintenance électrique.

2. Barres de sécurité.

3. Protection contre la libération de réactivité - Retrait d'absorbeur.

4. Protection contre les accidents thermiques.

5. Détection de rupture de gaine dans les divers réacteurs.

6. Sécurité des expériences - Les auxiliaires.

VI. Pilotage d'un réacteur (6 séances)

1. Démarrage (2 séances).

Niveau de source.

Réacteur sous-critique.

Divergence.

Montée en puissance.

Stabilisation (problèmes qui s'y rattachent).

Pilotage automatique.

Dans les exposés de ce paragraphe, il paraît souhaitable d'attirer l'attention sur les points suivants :

- incertitude sur l'état initial du réacteur,
- degré de confiance à accorder aux appareils de mesure (périodemètre, pseudo-période),
- influence de la vitesse de remontée des barres sur le niveau atteint à la criticalité,
- dangers de cette phase à cause de l'absence de contre réaction de température.

2. Marche en puissance (2 séances).

Stabilité intrinsèque du réacteur,

Evolution dans le temps du coefficient de température,

Evolution de la réactivité dans le temps.

Arrêt du réacteur - redémarrage après un court arrêt.

3. Evolutions accidentelles (1 séance).

Causes possibles provoquant l'accident de réactivité.

Réponse d'un réacteur à un saut de réactivité.

Accident, froids - vidange accidentelles du modérateur et du fluide réfrigérant.

4. Conférence sur les accidents «historiques» (1 séance).

VII. PROTECTION AUTOUR D'UN RÉACTEUR (4 séances)

1. Diffusion et fluctuation des rayonnements (2 séances)

Arrêt des rayonnements.

Matériaux de protection et écrans de pile.

Activation des fluides de refroidissement.

2. Dangers biologiques des rayonnements (1 séance).

Normes et doses - Détecteurs portatifs.

3. Protection lors des déchargements et des travaux sur les piles (1 séance).

Contamination atmosphérique.

DISCUSSION

U. ZELBSTEIN: I should like to take advantage of M. Conte's presentation to put a rather general question. It appears that one can use remote-handling equipment in combination with television in order to deal with certain more or less predictable defects in a reactor. Has this in fact been done at Marcoule?

F. CONTE: Yes. Several months ago a very small television camera - 35 mm in diam. - was described in a memorandum by the Commissariat à l'énergie atomique (CEA). This instrument can be used in combination

with all sorts of remote-handling equipment, including grippers, mirrors and so forth, for operations many metres within the reactor. The work is performed under the guidance of the operator and can be carried out at distances of up to 60 m, so that practically speaking every nook and corner of the reactor is accessible.

O. ČERNÝ: How are the fuel elements cooled after being withdrawn from the core?

F. CONTE: The film you saw a few minutes ago showed the vibrating conveyors which carry the fuel to the pond for storage. The elements remain in these ducts for 30 min and are cooled by circulation of cold CO_2 at a pressure of 15 kg/cm². However, this period of 30 min could be considerably reduced, since the time actually required to transport the elements to the pond is only 6 min. Not infrequently we have put the elements in containers and sent them straight to the pond immediately after their extraction from the reactor.

J. TERPSTRA: I should like to know, firstly, what special considerations led you to adopt the type of structure employed in the G2 and G3 reactors.

F. CONTE: The principal advantage of the horizontal channels lies in the greater ease of charging and discharging fuel, for the two operations can be carried out simultaneously. Nevertheless, I ought perhaps to make it clear that our choice would not necessarily be the same if we were to build these reactors again today.

J. TERPSTRA: Thank you. If I may turn to another question, how much gas leaks out during loading and unloading?

F. CONTE: At the normal rate of fuel discharge we lose about 3 t/d, whereas the loss of gas due to loading varies from 500 to 800 kg/d.

J. TERPSTRA: Lastly, is there any leakage of gas from one channel to another during the operation of these reactors?

F. CONTE: No. The horizontal structure of the graphite beds leaves no play between the bricks, and consequently gives no leakage of gas between the channels.

INTÉRÊT DE L'EXPÉRIENCE NEUTRONIQUE ACQUISE A MARCOULE ET A CHINON POUR LE PROGRAMME DE CENTRALE A GRAPHITE

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Abstract — Résumé — Аннотация — Resumen

EXPERIENCE IN NEUTRON PHYSICS ACQUIRED AT MARCOULE AND CHINON: ITS VALUE FOR THE GRAPHITE-REACTOR PROGRAMME. The entry into service of the first French power reactor - G1, G2 and G3 at Marcoule and EDF1 at Chinon - has provided fundamental experience for the further development of this reactor type. This experience has accrued both from start-up tests and from power operation.

The most important start-up tests consisted of : (a) Progressive replacement experiments, which made it possible to perfect the methods of calculation for G2 and EDF1; and (b) Fixed absorber and control rod-tests.

Through the operation of G2 and G3, a better adjustment of the reactivity balance under power has been achieved, taking into account effects of temperature and xenon poisoning. Similarly, experiments carried out with the reactors during operation under power have made it possible to check the validity of the kinetic models used to study transient phenomena in this type of reactor.

The experience acquired with the first French graphite reactors - though it still has to be completed by measurements of a more basic and systematic kind on assemblies specially designed for the purpose, such as MARIUS and CESAR - has thus proved to be of extreme value both for future projects and for studies on the operation of large power stations, where problems of control and kinetics are especially important.

INTÉRÊT DE L'EXPÉRIENCE NEUTRONIQUE ACQUISE A MARCOULE ET A CHINON POUR LE PROGRAMME DES CENTRALES A GRAPHITE. La mise en service des premières piles de puissance françaises (G1, G2, G3 à Marcoule, EDF1 à Chinon) a permis d'acquérir une expérience fondamentale pour la poursuite de la filière, tant par les essais de démarrage proprement dits que par le fonctionnement en puissance.

Parmi les essais de démarrage, les plus importants ont été a) les expériences de remplacement progressif, qui ont permis de mettre au point les méthodes de calcul pour G2 et EDF1; b) les essais d'absorbants fixes et de barres de contrôle.

Le fonctionnement de G2 et G3 a permis un recalage du bilan de réactivité en puissance (effets de température et de xénon). Des expériences effectuées sur les piles en puissance ont également permis de vérifier les modèles cinétiques utilisés pour l'étude des phénomènes transitoires dans ce type de pile.

L'expérience acquise dans les premières piles à graphite françaises, bien qu'elle doive être complétée par des mesures plus fondamentales et plus systématiques dans des ensembles spécialement conçus à cet effet (MARIUS, CÉSAR), s'est donc avérée fondamentale aussi bien dans l'élaboration des projets ultérieurs que pour l'étude du fonctionnement de centrales de grande puissance où les problèmes de contrôle et de cinétique jouent un rôle primordial.

ЗНАЧЕНИЕ НАКОПЛЕННОГО В МАРКУЛЕ И ШИНОНЕ ОПЫТА РАБОТЫ С НЕЙТРОНАМИ ДЛЯ ПРОГРАММЫ ЭЛЕКТРОСТАНЦИЙ С ГРАФИТОВЫМИ РЕАКТОРАМИ. Эксплуатация первых французских энергетических реакторов (G1, G2, G3 в Маркуле, EDF1 в Шиноне) позволила приобрести основной опыт для дальнейшей разработки реакторов этого типа как в процессе пусковых испытаний, так и в процессе работы реакторов на полной мощности.

Среди опытов с пуском наиболее важными оказались а) опыты по прогрессивной замене тепловыделяющих элементов, которые дали возможность разработать методы расчета для G2 и EDF1, и б) опыты с неподвижными поглотителями и управляющими стержнями.

Работа реакторов G2 и G3 позволила составить баланс потенциальной реактивности (воздействие температуры и ксенона). Проведенные на энергетических реакторах опыты позволили также выверить кинетические модели, использованные для исследования переходных явлений в этом типе реактора.

Опыт, приобретенный на первых французских графитовых реакторах, хотя его и следует дополнить путем более фундаментальных и систематических измерений на специально созданных для этого уста-

новках (MARIUS, CESAR), оказался, тем не менее, существенным как в разработке последующих проектов, так и для изучения работы электростанций большой мощности, где проблемы контроля и кинетики играют первостепенную роль.

INTERÉS DE LA EXPERIENCIA NEUTRÓNICA ADQUIRIDA EN MARCOULE Y EN CHINON PARA EL PROGRAMA DE CENTRALES DE GRAFITO. La puesta en marcha de los primeros reactores de potencia franceses (G1, G2 y G3 de Marcoule, EDF1 de Chinon) permitió adquirir una experiencia fundamental para la prosecución del desarrollo de este concepto, tanto por los ensayos de puesta en marcha propiamente dichos como por el funcionamiento en régimen normal.

Los principales ensayos de puesta en marcha fueron: a) los experimentos de sustitución progresiva, que han permitido poner a punto los métodos de cálculo para los reactores G2 y EDF1; b) los ensayos de absorbedores fijos y de barras de control.

El funcionamiento de los reactores G2 y G3 ha permitido establecer con más precisión el balance de reactividad en régimen normal (efectos de la temperatura y del xenón). Además, experimentos realizados con los reactores en marcha han permitido verificar los modelos cinéticos empleados para el estudio de los fenómenos transitorios en los reactores de ese tipo.

La experiencia adquirida con los primeros reactores de grafito franceses, aunque falte todavía completarla con mediciones más fundamentales y sistemáticas en conjuntos especialmente concebidos a este efecto (MARIUS, CESAR), ha demostrado su importancia capital tanto para la elaboración de los proyectos ulteriores como para estudiar el funcionamiento de centrales de gran potencia en las que los problemas de control y de cinética desempeñan un papel primordial.

INTRODUCTION

Les expériences de neutronique effectuées soit au démarrage, soit au cours du fonctionnement d'une pile de puissance ont comme objectif premier d'améliorer la connaissance de l'installation proprement dite: réserve de réactivité, distribution des sources de chaleur dans différentes configurations du fonctionnement, anti-réactivité des barres de contrôle, comportement cinétique, etc.

Il serait cependant très long, dispendieux, et au demeurant aléatoire, de prétendre étudier expérimentalement tous les aspects neutroniques du fonctionnement d'une telle installation. On cherchera donc au cours de ces essais à vérifier, parfois à corriger, les méthodes de calcul couramment utilisées. Il sera ainsi possible, à partir d'un nombre limité d'essais, de prévoir d'une façon très générale le comportement de l'installation.

Les connaissances ainsi acquises peuvent, dans une certaine mesure, être utiles pour les études générales sur l'ensemble de la filière: nous en verrons des exemples avec les mesures de pression critique dans G2 ou G3, d'aplatissement radial du flux dans les piles de Marcoule et dans EDF1, d'efficacité de barres de contrôle et de déformations du flux dans EDF1, et également dans un autre ordre d'idées avec les expériences, liées à la sûreté, de baisse ou d'arrêt de débit.

Il est cependant possible d'effectuer, à l'occasion de la mise en service de grandes installations, des expériences dans le but spécifique d'améliorer les connaissances générales. On peut citer en particulier: les expériences de remplacement progressif effectuées pour la première fois dans G1 en 1956, et reprises en 1958 dans G2; les mesures d'indice de spectre (^{239}Pu , ^{235}U , Lu , Mn) effectuées à 40 et 160°C dans EDF1 en avril de cette année; les expériences de perturbation de la puissance effectuées dans G2 et G3 à divers niveaux d'irradiation pour vérifier les modèles cinétiques utilisés dans les études d'instabilité spatiales des grandes centrales.

Enfin, le fonctionnement proprement dit de l'installation apporte des renseignements très importants pour les études de projet, en particulier en ce qui concerne l'évolution de la réactivité. C'est ainsi qu'on a pu vérifier, sur G3, les méthodes de calcul utilisées grâce à l'évolution globale de la réactivité de la pile. D'autre part, il est possible d'étudier l'évolution des propriétés neutroniques du combustible en combinant l'oscillation et l'analyse chimique d'échantillons irradiés.

Nous nous proposons de rappeler, dans cette communication, les principaux résultats qui ont été obtenus d'une manière ou d'une autre, grâce aux piles de Marcoule et aux premiers essais dans EDF1, et qui présentent un intérêt significatif pour les études générales sur la filière graphite-uranium naturel. La plupart de ces résultats figurent déjà dans des publications diverses: nous ne citons en bibliographie que les principales. Ils sont l'aboutissement d'un effort collectif des différents départements et services du Commissariat à l'énergie atomique, et en ce qui concerne EDF1, d'Electricité de France.

I. PROPRIÉTÉS NEUTRONIQUES DE RÉSEAUX A FROID

Les premières informations recueillies sur une nouvelle installation sont en général: a) le chargement critique, b) la pression d'air critique pour le chargement complet. Dans G2 et G3, il a été possible de compléter ces données par la mesure de la pression critique pour plusieurs chargements intermédiaires, et par la mesure du laplacien par carte de flux pour plusieurs chargements.

Les réseaux de G2 et G3 étant uniformes, on a pu déduire de cet ensemble d'expériences, à la fois, k_{∞} et l'aire de migration radiale. Les résultats obtenus, comparés aux valeurs calculées, sont rassemblés dans le tableau I.

TABLEAU I

COMPARAISON DES RÉSULTATS OBTENUS AVEC LES VALEURS CALCULÉES

	Pas (mm)	Diamètre du canal (mm)	Barreau (ϕ : mm)	k_{∞}		M_T^2 (cm ²)	
				Exp.	Calculé	Exp.	Calculé
G2	200	70	28	1,0868 $\pm 0,001$	1,0860	757 ± 30	744,5
G3	200	70	28	1,0886 $\pm 0,001$	1,0860	696 ± 30	744,5

L'accord est excellent sur k_{∞} , et il semble que les valeurs calculées de M_T^2 soient à l'intérieur des incertitudes expérimentales. Ce dernier point est particulièrement important, car l'ensemble du programme expérimental

sur lequel reposent nos méthodes de calcul est basé sur les mesures de laplacien et de remplacement progressif effectuées sur de petites piles (MARIUS). Il est nécessaire de connaître M_1^2 pour passer des laplaciens mesurés aux k_∞ , qui sont les quantités intéressantes pour l'établissement d'un formulaire et pour les calculs de pile.

C'est dans G1 qu'ont été effectuées les premières expériences de remplacement progressif. Le réseau de G1 est constitué d'un pas carré de 200 mm, un canal de 70 mm de diamètre et un combustible de 26 mm de diamètre. Les expériences de remplacement ont porté sur des combustibles de 28 et 32 mm de diamètre, et ont mis en évidence que le maximum de laplacien se situait pour un barreau de diamètre voisin de 30 mm, et non de 26 mm comme on le croyait jusqu'alors. Ce résultat devait être immédiatement exploité dans le projet en cours de G2, dans lequel on a utilisé successivement des éléments de 28 puis de 31 mm de diamètre.

Deux ans plus tard, en 1958, les expériences de remplacement furent reprises dans G2. En un mois d'essais, une dizaine de combustibles furent testés, en particulier des tubes dont le plus grand (20×40) devait être pratiquement adopté pour EDF2. Ainsi, près de deux ans avant de disposer de l'empilement critique MARIUS, les neutroniciens avaient pu confirmer que l'emploi de tubes ne serait pas trop défavorable dans les projets de piles de puissance, et mettre au point une méthode de calcul pour ces projets étayée par des expériences. De plus, l'expérience acquise confirmait l'intérêt de la méthode de remplacement progressif, permettant ainsi de décider la construction d'un empilement critique dont l'emploi reposait sur cette méthode.

II. PROPRIÉTÉS NEUTRONIQUES DES RÉSEAUX A CHAUD ET AU COURS DE L'ÉVOLUTION

Le moyen le plus direct d'obtenir des informations sur l'évolution de la réactivité est de la mesurer dans une pile de puissance. En pratique, au fur et à mesure que la réactivité croît pendant les premiers mois d'irradiation d'une charge neuve, on est amené à compenser cette réactivité à l'aide soit d'absorbants, soit de barres de compensation. On peut en déduire la variation de réactivité si on a pu mesurer au préalable l'efficacité des absorbants ou des barres de compensation, ou si on estime pouvoir se fier au calcul.

Un bon exemple est fourni par le début de l'évolution d'une charge de G3, pendant laquelle aucun déchargement de combustible ne s'est produit. La réactivité était compensée dans une limite de 400 pcm environ par des barres de contrôle, puis on chargeait, sans arrêter la pile, suffisamment de canaux de fer pour sortir les barres de contrôle du cœur. On peut alors reconstituer (fig. 1) la courbe d'évolution de la réactivité. On a également porté sur la figure 1, à titre indicatif, la courbe calculée.

L'excellent accord trouvé entre le fonctionnement réel de G3 et les prévisions n'est cependant pas suffisant pour confirmer la validité des hypothèses de calcul; d'une part, parce que l'incertitude sur la courbe expérimentale est très grande et, d'autre part, parce que la courbe calculée dépend de nombreux facteurs et qu'il peut très bien y avoir compensation entre plusieurs erreurs. Il est donc indispensable d'analyser plus en détails les

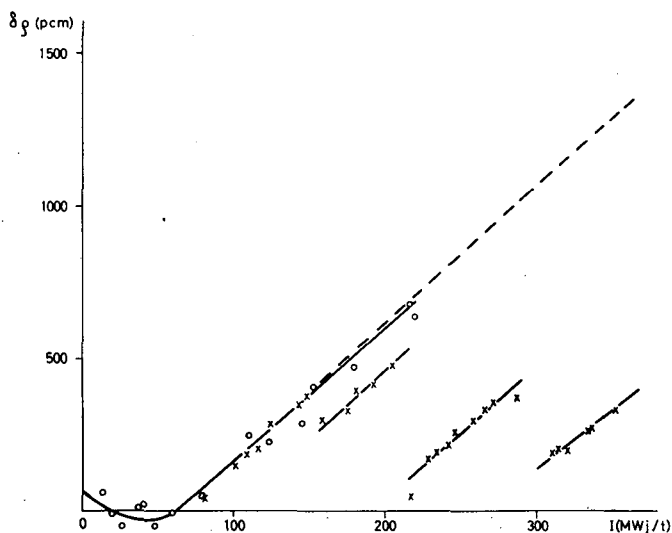


Figure 1

Evolution de la réactivité dans G3.

phénomènes. Les piles de puissance peuvent là encore nous fournir des éléments utiles.

On sait que l'évolution de la réactivité dépend essentiellement: a) des sections efficaces effectives des différents isotopes du plutonium, et au premier chef du ^{239}Pu , b) du facteur de conversion initiale.

D'une façon générale, on cherche à calculer les premières à l'aide d'un modèle de thermalisation qui représente convenablement le spectre de neutrons dans la zone proche épithermique (0,1 à 1 eV), zone très importante tant pour le ^{239}Pu que pour le ^{240}Pu . On cherche alors des méthodes expérimentales fournissant les sections efficaces effectives des isotopes du plutonium dans quelques cas (avec des spectres et des températures différentes) afin de contrôler la validité du modèle utilisé.

Une première méthode consiste à irradier des détecteurs de nature diverse, sensibles plus particulièrement à telle ou telle région du spectre, Pu, U, Lu, Mn, etc. De telles mesures nécessitent pour la plupart des flux intégrés élevés ($\sim 10^{12}$ n/cm²). Si elles sont faites depuis plus d'un an de façon systématique à froid dans l'empilement critique MARIUS, elles ont pu être faites à chaud (40 et 160°C) pour la première fois lors des essais à l'air d'EDF1 au mois d'avril dernier. Signalons également que des mesures de facteur de conversion ont été faites dans EDF1.

Une deuxième méthode consiste à étudier en détail les propriétés d'échantillons de combustibles irradiés dans les piles: a) par oscillation dans des conditions bien déterminées. L'échantillon est comparé à un étalon de même géométrie et le tout se fait au sein d'un réseau dont le spectre est voisin du spectre dans lequel a été irradié l'échantillon; b) par analyse de la composition chimique et isotopique.

L'oscillation se fait suivant une technique qui commence à être classique et qui consiste à mesurer simultanément deux signaux: un signal global pratiquement proportionnel à la différence des facteurs η entre l'échantillon et l'étalon et un signal local proportionnel à la différence des absorptions entre les deux échantillons. On peut donc en déduire l'effet de l'irradiation sur Σ_a et $\nu\Sigma_f$.

L'analyse chimique des mêmes échantillons permet alors une corrélation détaillée et un contrôle des mesures ainsi effectuées. Mais elle peut déjà, seule, apporter des renseignements précieux. On sait que, tant que l'irradiation est assez faible, on peut exprimer la formation des isotopes successifs du plutonium de façon simple en fonction de l'appauvrissement en uranium 235. Si on pose:

$$\chi = \text{Log} \left[\frac{{}^{235}\text{U}(0)}{{}^{235}\text{U}(\tau)} \right]$$

$$\frac{{}^{239}\text{Pu}}{{}^{238}\text{U}} = \frac{\hat{\sigma}_8}{\hat{\sigma}_5} \chi$$

on a:

$$\frac{{}^{240}\text{Pu}}{{}^{238}\text{U}} = \frac{\hat{\sigma}_8}{\hat{\sigma}_5} \frac{\hat{\sigma}_{c9}}{\hat{\sigma}_5} \frac{\chi^2}{2}$$

$$\frac{{}^{241}\text{Pu}}{{}^{238}\text{U}} = \frac{\hat{\sigma}_8}{\hat{\sigma}_5} \frac{\hat{\sigma}_{c9}}{\hat{\sigma}_5} \frac{\hat{\sigma}_0}{\hat{\sigma}_5} \frac{\chi^3}{6}.$$

La mesure directe de la composition isotopique permet donc (moyennant, en pratique, le calcul de quelques corrections d'ordre supérieur) de déduire les rapport $\hat{\sigma}_8/\hat{\sigma}_5$ (c'est-à-dire le facteur de conversion initial au facteur $1+\alpha_5$ près), $\hat{\sigma}_{c9}/\hat{\sigma}_5$ et $\hat{\sigma}_0/\hat{\sigma}_5$.

De telles analyses sont en cours à Saclay depuis bientôt deux ans sur des combustibles provenant de G2 et G3. Les premiers échantillons oscillés l'ont été dans MARIUS pendant l'été 1962. D'autres suivront dans quelques semaines. D'autre part, une quarantaine d'échantillons ont été placés dans EDF1 où ils seront irradiés dans les conditions de spectre et de température bien définies. Ils ont été oscillés dans MARIUS avant d'être mis en pile, et seront oscillés après irradiation dans MARIUS, puis à chaud dans l'empilement critique CÉSAR qui sera mis en service en 1964.

On sait que le bilan neutronique de ce type de pile est assez tendu, mais que l'évolution d'abord positive de la réactivité améliore sensiblement les choses. Suivant la forme précise de la courbe dans sa branche descendante, et suivant l'irradiation que l'on peut espérer du fait des propriétés métallurgiques du combustible, on devra prévoir une réserve de réactivité plus ou moins grande dans l'état initial. Par exemple, si on vise une irradiation moyenne de rejet de 5000 MWj/t au lieu de 4000 MWj/t, on sera conduit à prévoir un cœur plus grand d'environ 10% et une charge d'uranium supérieure de 5 à 7%. Les mêmes précautions seront nécessaires si on vise

une irradiation donnée, par exemple 4000 MWj/t, mais que l'incertitude sur la réactivité à cette irradiation est de l'ordre de 500 pcm.

La connaissance précise des propriétés neutroniques des piles à graphite-uranium naturel apparaît donc comme fondamentale pour les études de projet. Les quelques exemples que nous avons donnés ici montrent clairement qu'il est possible, voire indispensable, d'utiliser les piles de puissance elles-mêmes pour acquérir cette connaissance.

III. ÉTUDE DES FLUX. ABSORBANTS ET BARRES DE CONTRÔLE

Deux modes d'aplatissement radial du flux sont actuellement utilisés dans les piles de puissance à graphite en fonctionnement ou en projet: l'aplatissement obtenu à l'aide d'absorbants (cartouches de fer ou cartouches destinées à la production de radio-éléments) et l'aplatissement obtenu en faisant varier le pas du réseau.

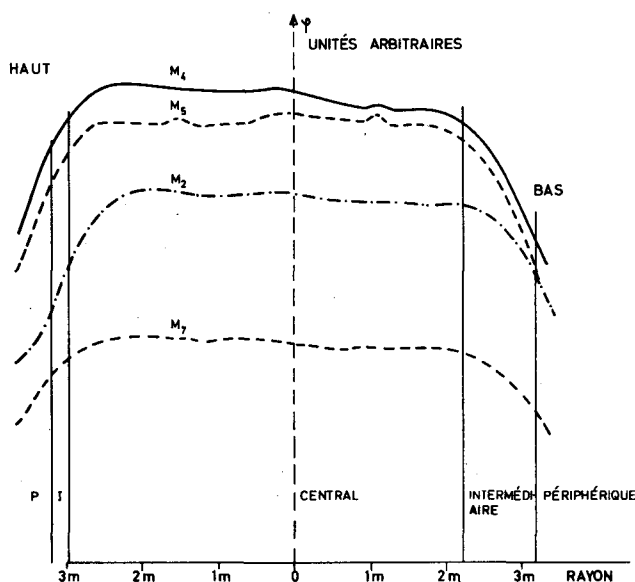


Figure 2

Distribution radiale du flux dans G3 (17.3.61 - 198 MW)

La première méthode est a priori plus simple car elle ne modifie que très peu le spectre et il est relativement aisé d'adapter l'efficacité ou la répartition des absorbants pour obtenir et conserver un aplatissement satisfaisant. Cette méthode est utilisée dans les trois piles de Marcoule avec succès. On constate sur la figure 2 que le flux radial de G3 est constant à quelques pour cent près dans une zone de rayon 220 cm, alors que le rayon du cœur est de 391 cm.

Les neutroniciens ont cependant attiré depuis longtemps l'attention sur le fait qu'il serait plus rentable, du point de vue bilan neutronique, d'absorber les neutrons excédentaires de la zone centrale d'une pile dans les ré-

sonances de l'uranium 238. Ceci peut être obtenu très simplement en resserrant le pas dans la zone centrale.

Cette technique a été utilisée systématiquement dans les piles de Chinon; dans les projets EDF2 et les suivants, le réseau est hexagonal et la variation du pas s'obtient en chargeant tous les canaux de la zone centrale et seulement une fraction des canaux des zones externes (3/4 ou 2/3); dans EDF1, il y a deux zones distinctes de pas différents, la zone centrale carrée qui comprend 256 cellules carrées de pas 196 mm, et la zone périphérique dont le pas est de 224 mm.

La différence de spectre entre les différentes zones provoque des variations du flux dans les zones de transition, et il importait de vérifier que ces variations n'étaient pas trop importantes et correspondaient à peu près aux calculs courants. Les cartes de flux relevées au cours des essais de démarrage d'EDF1 ont confirmé les espoirs placés dans cette méthode d'aplatissement: on compare sur la figure 3 la courbe de flux mesurée à la courbe de flux calculée en théorie à deux groupes. Le calcul n'est jamais à plus de 2 ou 3% du flux mesuré, et l'aplatissement radial obtenu est très satisfaisant pour un coût très faible en réactivité.

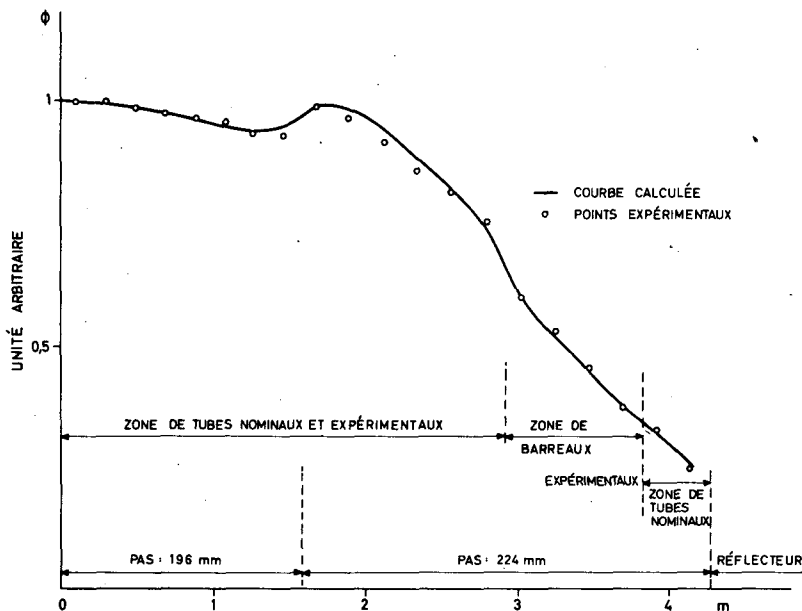


Figure 3

Distribution radiale du flux dans EDF1, sans éléments absorbants.

Il est cependant nécessaire, même dans une pile dont l'aplatissement est obtenu par variation de pas, de prévoir l'adaptation de cet aplatissement en fonction de l'évolution de la réactivité à l'aide d'absorbants. D'autre part, il faut être à même de prévoir les déformations du flux provoquées par les barres de pilotage et les barres de compensation. Les cartes de flux relevées dans EDF1 pour différentes configurations d'absorbants et de

barres de contrôle ont également permis de vérifier les méthodes de calcul utilisées à Saclay: calcul à deux dimensions (X-Y) DAIXY, les canaux d'absorbants étant remplacés par des longueurs d'extrapolation sur des frontières convenablement choisies. Il est donc possible de prévoir, par le calcul, les emplacements satisfaisants pour le pilotage et la compensation dans les projets futurs.

Enfin, on a intérêt dans un projet de pile de puissance à adapter le mieux possible aux besoins le système de contrôle et de sécurité. Les mesures précises d'efficacité de barres de contrôle isolées, et de l'ensemble des barres de contrôle, en fonction de leur enfoncement dans le cœur d'EDF1 * permettront soit de confirmer, soit d'adapter les méthodes de calcul mises au point grâce aux expériences effectuées dans l'empilement critique MARIUS.

IV. COMPORTEMENT CINÉTIQUE DES PILES A URANIUM NATUREL ET GRAPHITE

Les piles à graphite et uranium naturel sont caractérisées par leur stabilité envers des perturbations de courte durée, en raison de leur coefficient de température d'uranium négatif, et leur instabilité envers des perturbations prolongées, en raison du coefficient de température du modérateur qui devient très rapidement positif dès qu'un peu de plutonium s'est formé.

La stabilité «prompte» de ces piles peut jouer un rôle primordial en cas d'accident brutal, soit d'arrêt de soufflage, soit de dépressurisation, et il importe de vérifier dans la mesure du possible les modèles cinétiques utilisés pour l'étude de ces accidents. Des expériences d'arrêt de soufflage ont été effectuées dans les piles G2 et G3, avec ou sans chute de barre de sécurité.

L'instabilité à long terme de ces piles n'a guère d'importance dans les piles de petites dimensions, telles que G2 ou EDF1. Par contre, elle peut se traduire dans les piles dont les dimensions et la puissance spécifique sont élevées par des oscillations spatiales du flux, et même dans certains cas par des instabilités divergentes. Il importe alors de prévoir un système de pilotage particulier pour combattre ce phénomène. C'est ainsi que dans EDF2, la pile est divisée en sept régions (une centrale et six périphériques), et que dans chacune de ces régions on dispose de deux ou trois barres de pilotage spatial dont les mouvements sont commandés par les variations de la température moyenne du gaz de refroidissement.

L'étude de ce système de pilotage assez complexe nécessite une connaissance approfondie du comportement cinétique de la pile et, en particulier, des coefficients d'échange moyens entre le modérateur et le gaz, des variations de température du modérateur et des conséquences de ces variations sur la réactivité de la pile.

* Les étalonnages de barres de contrôle effectués dans les piles de Marcoule n'ont guère permis de vérifier les méthodes de calculs, car les barres sont perpendiculaires aux canaux, et que l'ensemble ne peut être calculé que par des méthodes très grossières.

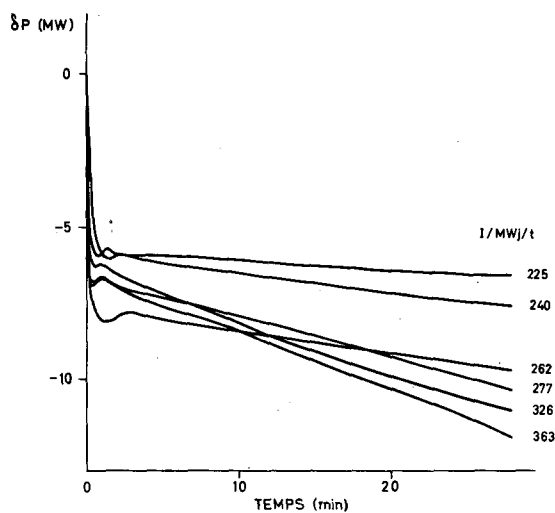


Figure 4

Expérience de perturbation dans G3.

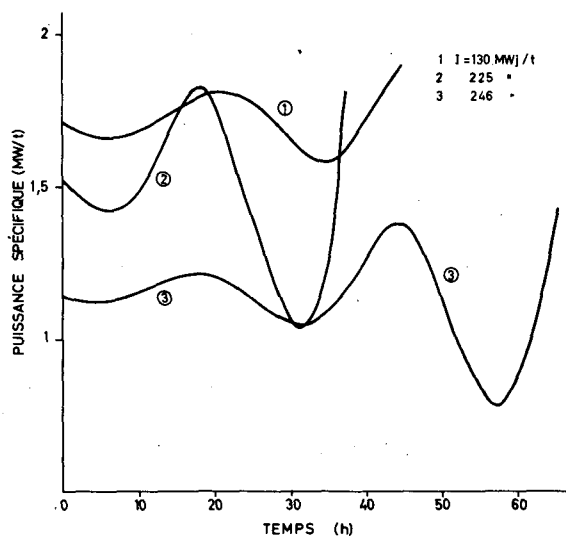


Figure 5

Expérience d'oscillation lente dans G3.

Les expériences de perturbation effectuées dans G3 (fig. 4 et 5) ont permis de tester les modèles cinétiques utilisés dans les programmes de calcul mis au point à Saclay pour l'étude des instabilités spatiales.

CONCLUSION

L'expérience acquise au cours des essais de démarrage et au cours du fonctionnement des premières piles de puissance françaises a joué un rôle très important pour les études de projet de la filière uranium-graphite.

Elle a permis en particulier d'orienter la filière vers l'emploi de tubes de dimensions importantes, de tenir compte de l'évolution de la réactivité sans risquer d'erreurs catastrophiques, d'adapter au mieux les systèmes de contrôle et de pilotage, y compris le pilotage spatial.

On attend encore beaucoup dans l'avenir de l'expérience que pourront fournir les piles de Marcoule et les piles de Chinon, grâce en particulier aux irradiations d'éléments combustibles qui serviront aux analyses isotopiques et aux oscillations.

Cette expérience est cependant acquise dans un domaine assez étroit, et dans des conditions qui ne sont pas toujours parfaitement définies. Il est donc essentiel de la compléter par des expériences systématiques: études de réseaux avec et sans plutonium, mesures d'indices de spectre, oscillations de combustibles contenant du plutonium, le tout à diverses températures. C'est le but assigné précisément aux empilements critiques MARIUS (pour les expériences à froid) et CÉSAR (pour les expériences jusqu'à 500°C).

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DISCUSSION

C. A. PURSEL: As you mentioned burn-ups of 4000-5000 MWd/t, I should like to know exactly what form of fuel is to be used in the EDF reactors.

P. TANGUY: The EDF fuel elements consist of natural uranium in the form of a metal alloy containing a small amount of molybdenum (about 1 wt. %). The tubes, clad in an alloy of magnesium and a small quantity of zirconium, are cooled solely from without.

K. EFFAT: I should like to know whether your experiments were performed under normal operating conditions or whether they required special adjustments of the reactor parameters. To what extent did the experiments interfere with normal operation of the reactors at power?

P. TANGUY: The nuclear tests carried out at the start-up of G2, G3 and EDF 1 did not delay the entry into service of these reactors. The physicists' experiments were conducted during the time available between the mechanical and dynamic tests, and while work on the non-nuclear plant components was being completed.

After start-up, the time allotted to the physicists for nuclear experiments was far more strictly limited. In the case of G2 and G3 certain definite periods were set aside for what were essentially kinetic experiments, during which time it was possible to vary parameters such as power, gas flow, control-rod position, inlet gas temperature and so forth; but always on condition that the total loss of power should not exceed a specific predetermined limit.

Finally, we have taken advantage of a number of shutdown periods to perform control-rod calibrations, requesting the operators to keep the graphite temperature as nearly as possible at a constant level.

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