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BOILING WATER REACTORS

BOILING WATER REACTORS

by

ANDREW W. KRAMER

Argonne National Laboratory



PREPARED UNDER CONTRACT WITH THE
UNITED STATES ATOMIC ENERGY COMMISSION



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PREFACE

This book endeavors to set forth, within the limits of a single volume, the history, the development, and the technology of the boiling water reactor. Although the boiling reactor is among the most recent of the various power reactor concepts, rapidly occurring engineering developments and continued experimental investigations have placed this type of reactor in a favorable position as a source of power from nuclear fission. Among the more important achievements has been the demonstration that the natural circulation boiling water reactor is capable of advantageously high heat density operation. Moreover, recent operation of the Experimental Boiling Water Reactor at Argonne National Laboratory at a power level more than three times the designed thermal output provides what is perhaps the most sudden and dramatic improvement in nuclear power costs seen by the industry.

The purpose of this book is to describe what is known about boiling water reactors for the practicing engineer who is already familiar with the basic principles of reactor technology in general but who has not had the opportunity to study boiling reactor systems in detail. The boiling water reactor is somewhat unique in that its physical simplicity belies the complexities of its operation. Because boiling reactors involve the use of a liquid and vapor phase mixture, with each of the phases having different and varying neutron, thermal, and hydrodynamic characteristics, the theory of this class of reactors is more complex and difficult to understand than that of reactor types using single-phase coolants and moderators. It is only recently that the interrelationships of these varying parameters in boiling reactor operation are beginning to be understood and that reactor designers have learned how to use them in predicting boiling reactor performance. Admittedly, problems still exist which must be solved before boiling reactors can become economically competitive central station power plants, but the favorable operating performance of the boiling cycles to date attest to their attractiveness for central station use.

Since the boiling water reactor is uniquely a contribution to nuclear technology of the Argonne National Laboratory, it is fitting that this book was prepared under the auspices of the Argonne, and with the assistance and cooperation of its many scientists and engineers who conceived the boiling principle and brought it to its present state of development. While much of the material in this volume is contained in previously published laboratory reports and papers in various journals,

the information is scattered; the essential objective of this book has been to collect this diverse material and present it in a single volume. Those interested in more detailed technical data can consult the original sources of information, but they should find the outline of the technology presented here of value in pursuing further study. The Atomic Energy Commission publications listed among the references are available for inspection at the Commission's depository libraries in the United States and abroad and are sold by the Office of Technical Services, U. S. Department of Commerce, Washington 25, D.C.

In presenting this volume, I am conscious of the fact that the state of boiling reactor technology is still such as to permit differences of opinion and judgment. There are varying degrees of enthusiasm among various individuals engaged in the development of boiling reactors concerning the merits of different systems and concepts, and it is not always possible to resolve such conflicting opinions to everyone's satisfaction. Where such conflict of opinion exists I have had to rely on my own knowledge and judgment in presenting what seemed to me to be the most acceptable interpretation in terms of the present state of the art. Throughout, I have sought the most authentic sources of information and while the volume reflects the thinking of the highest and most competent authorities in the field, the presentation must, of necessity, reflect my own judgment.

The presentation, to a large extent, is chronological. The book begins with a brief history of the boiling water reactor concept and a description of the early experiments on boiling processes and rates of heat transfer under boiling conditions. This is followed by a general discussion of basic principles and descriptions of various boiling reactor cycles, including consideration of requirements respecting safety and stability, heat transfer, and radioactive carryover in the steam.

Since the experimental investigations leading to the practical development of the boiling reactor provided significant design data, a detailed description is given of the various BORAX experiments. The early BORAX data was reported in considerable detail at the 1955 Geneva Atomic Conference but it is included here because it forms an essential part of the development of the boiling reactor, and many readers may not have convenient access to the original reports on these tests. Considerable emphasis is given to the recent work with BORAX-IV, involving the development of oxide-type fuel elements, since this type of fuel gives promise of operation at considerably higher temperatures than those used at present.

The physics underlying boiling reactor design is quite involved; consequently the chapter concerned with the design of boiling reactor systems

is one of the most important in the book. This section discusses the effects of steam formation on reactivity, the influence of pressure and temperature, and the interrelationship of the hydrodynamic, thermal, and nuclear processes. Here also are defined such parameters as steam fraction, reactivity worth, steam coefficient of reactivity, and other terminology unique to boiling reactor designs.

Two entire chapters are devoted to the history, development, and operation of EBWR, the Experimental Boiling Water Reactor at Argonne. Here again, because of its intrinsic importance, material is included which is available in other works. A detailed description also is given of the Vallecitos boiling reactor of the General Electric Co. The particular significance of this plant lies in the fact that it serves as a pilot plant for testing components going into the construction of the 180,000-kw Dresden Station which will be the first large commercial boiling water reactor to go into operation. Although Dresden Station is still under construction, the design data are sufficiently crystallized to warrant a separate chapter. Other boiling reactor projects are described insofar as design details are available.

A final chapter is devoted to current and future research development programs on boiling reactors, with particular emphasis on the SPERT project being conducted for the AEC by the Phillips Petroleum Co., at the Idaho Testing Station. The place of boiling water reactors in future nuclear power development is discussed and brief consideration is given to studies involving the development of very large boiling water reactor power plants using heavy-water moderator and coolant, and possibly nuclear type superheaters.

In the preparation of this book I have received invaluable help from a large number of individuals; indeed, without their assistance I could not have succeeded in putting the material together. At the Argonne National Laboratory, through the able assistance and guidance of Dr. Hoylande D. Young, I had at my disposal the entire resources of the Technical Information Division for source material. At Argonne, too, I had the opportunity of personal interviews and discussions with the various members of the EBWR staff and of other departments of the Laboratory. The number of people to whom I am indebted for help is so large that it is not possible to express my appreciation by mentioning all their names but I particularly want to thank J. M. Harrer, Project Manager of EBWR, V. C. Hall, J. A. DeShong, Jr., J. A. Thie, W. R. Wallin, E. A. Wimunc, W. C. Lipinski, P. A. Lottes, and E. A. Bailey, all of Argonne National Laboratory, for their invaluable advice and guidance. Chapter 8, devoted to ALPR (the Argonne Low Power Reactor), is based entirely on the paper by C. R. Braun and his colleagues,

to be presented at the 1958 Geneva Conference. The two chapters on EBWR are based on the comprehensive ANL report issued in May, 1957, and thanks are due to all the many individuals who contributed to that report.

Valuable advice and personal assistance were also give me by officers and staff members of General Nuclear Engineering Corporation at Dunedin, Fla., in particular by Dr. W. H. Zinn, President, and Dr. J. R. Dietrich, J. M. West, and H. V. Lichtenberger, Vice Presidents, all of whom participated at Argonne in the early days of boiling reactor development. In the chapter describing the BORAX experiments specific credit is extended to a number of individuals who participated in the early BORAX work. The fact that Dr. Zinn's name, or the names of West, Lichtenberger and Dietrich are not mentioned in these references is not an oversight. The fact is, it is difficult to find an adequate way of crediting their work. This is particularly true of Dr. Zinn, yet it is generally recognized that his personal contribution to the development went far beyond those which might be considered *in absentia*. Without the on-the-spot work of Zinn and Lichtenberger the BORAX experiments would never have been completely successful. As it was, they found more time for doing things than for writing about them; consequently their work is not adequately reflected in the authority of the BORAX report.

I also wish to express thanks to Samuel Untermeyer of General Electric Company's Vallecitos project, not only for his enthusiastic help in supplying data and information for the description of the Vallecitos reactor but also for his help in acquainting me with the early history of the boiling water concept which he promoted so vigorously from the beginning.

For the details of the Dresden Station credit is due to the Nuclear Division of the Commonwealth Edison Co. of Chicago, in particular to Mr. J. J. Poer, project engineer of Dresden and to H. G. Carson, staff engineer. For the description of the Elk River plant I am indebted to H. Etherington of Nuclear Products—ERCO, Division of ACF Industries. C. B. Graham of Allis Chalmers supplied data on the Northern States Power nuclear plant, and C. C. Wheelchel of the Pacific Gas & Electric Co., material for the Humboldt Bay plant outline.

For review of the various chapters, I am indebted to the members of the Argonne staff already mentioned, and especially to Dr. Stuart McLain for his review of virtually the entire manuscript. His many valuable suggestions and critical advice have added materially to the accuracy and continuity of the book.

Lake Bluff, Illinois
July, 1958

ANDREW W. KRAMER

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CHAPTER 1

INTRODUCTION

Water-cooled, water-moderated nuclear reactors are classified, broadly, into two types: the boiling type, and the nonboiling type. Each of these has its particular advantages and disadvantages and both types are under commercial development in the United States and elsewhere at the present time. The purpose of this book is to review the status of the work on the development of the boiling water reactor in the United States with some reference to foreign developments where information is available.

To enable the reader to view the development of the boiling reactor against its historical background, this opening chapter will be concerned largely with a chronological account of the events which led to the design and construction of the two boiling reactor power plants now in actual operation, namely, the Experimental Boiling Water Reactor at Argonne National Laboratory and the Vallecitos reactor of the General Electric Co. The subsequent chapters will cover these events in greater detail and also will cover boiling water reactor technology to date.

The development of the boiling water reactor is comparatively recent in the field of reactor technology. In the early days of power reactor development, the period, say, between 1948 and 1953, only the nonboiling type of water-cooled reactor received serious consideration. The general attitude toward boiling in a water moderated reactor at that time was not that such operation was not feasible, but rather that the chances for its feasibility appeared poor. It was felt that no water-moderated reactor could be designed to utilize boiling unless experiments first proved it feasible. In those early days the general technology of water cooling of reactors at high temperatures was not developed; consequently the boiling reactor could not be looked upon as an improvement in an existing technology, but rather as a reactor type which would have to be developed more or less from scratch. Those interested in power reactor development had other things on their minds, and in most cases time, money, and, particularly, enriched uranium were not easily available for an experiment to test the feasibility of the boiling water reactor concept.

During the war years, the graphite-moderated plutonium production reactors had been built at Hanford in the State of Washington. These reactors produced enormous quantities of heat which had to be removed. So cold water was pumped through the fuel channels to carry away this heat. From the beginning, the operation of these production reactors was successful but the operating temperature was necessarily low, and no

boiling of the cooling water occurred. Throughout the Manhattan Project it was generally believed that boiling would make a nuclear reactor unmanageable. The prevailing opinion was that delays and disturbances in the boiling process would have effects on the multiplication factor, k , which would be out of step with reactor power, leading to large and uncontrollable oscillations. Today, it is known that such oscillations do occur under special conditions in boiling reactors, but they are not considered necessarily dangerous or unmanageable.

As an illustration of this early attitude in regard to boiling in reactors, the Manhattan District Patent Officer wished to cover all possible types of reactors by patent, so he prepared an application, OSRD-97, on "cooling a reactor by latent heat of evaporation of a liquid," filed in the name of one of the principal scientists associated with the project. When appraised of this action, however, this distinguished scientist asked that the application be withdrawn, so the Patent Officer wrote a letter withdrawing the application, stating in part, ". . . in view of the indication that no way had been worked out for controlling a reactor of the type under consideration, the case will be given inactive status until such time as an operative device can be disclosed."^{1*}

Throughout the following years, water reactor designers avoided boiling in all the reactor projects then underway. During this period a number of water-cooled reactors were built: the Hanford reactors; CP-3 and CP-5 at Argonne National Laboratory; MTR, the Materials Testing Reactor in Idaho; the Savannah River Reactors; NRX, the Canadian reactor at Chalk River, Ontario; and STR, the Submarine Thermal Reactor for the first atomic submarine, *Nautilus*. In all of these designs the water temperatures were maintained below the boiling point even under the most severe operating conditions, and in most cases the metal surfaces were kept below the boiling point so that there was no possibility of local boiling. Considerations of design avoided boiling mainly to avoid loss of flow burnout.

During 1947 work was initiated by the Canadian Research Council at Montreal, using local surface boiling to transfer large heat fluxes from the inside of small electrically heated tubes. Using high water flow rates with the water kept well below the boiling point, heat transfer rates exceeding one million Btu per sq ft per hr were obtained when the tube wall temperatures were raised above the water boiling point. This work was reported by Knowles in the Journal of the National Research Council of Canada. The data stimulated exploratory interest in the United States and in Canada in permitting some local surface boiling in reactors; bulk boiling was still considered impractical.

*See the references at the end of the chapter.

In 1948, just before the AEC power reactor program was centralized at Argonne National Laboratory, the newly formed AEC Advisory Committee on Reactor Safeguards was considering the location of the Clinton High Flux Pile—later built as the Materials Testing Reactor in Idaho. This reactor used aluminum-plate fuel elements cooled by light water. The design had been developed at Oak Ridge. The principal hazard envisaged by the Safeguards Committee in evaluating this design was that the circulating water pump might fail and that the *afterglow* heat would melt the fuel even though it was kept under water. There was also some doubt as to whether steam could be formed fast enough to shut the reactor down if the control system malfunctioned. These apprehensions were quite needless because there was evidence available even at that time to prove that there was little likelihood of the fuel melting from afterglow heat as long as it was kept under water.

The question of the possible melting of the fuel elements was, of course, only one aspect of the situation. The general belief that boiling water reactors would not be stable was certainly supported by the information available at that time. Speculative calculations suggested that reactors might require smooth and delicate control, whereas the very nature of boiling is rough and boisterous; so it was anticipated that the collapse of steam bubbles might initiate sudden and unsafe fluctuations in reactivity and power.

Despite these questions, the idea of the boiling reactor persisted, because the direct generation of steam within the reactor vessel itself is the most straightforward way of removing heat from the core of the reactor. By utilizing the latent heat of evaporation of water it is possible to extract heat at high rates. Also, if steam could be generated within a boiling reactor and fed directly to a turbine, heat exchangers could be eliminated and large expensive pumps would not be needed if good natural circulation of water in the reactor could be obtained.

There was another aspect of boiling water that entered into the thinking of reactor designers: that of safety from a nuclear standpoint. One of the important characteristics of a nuclear reactor is the degree of hazard which it creates in the surrounding area. If the usefulness of reactors, either for research or for power production, is to be exploited effectively, the hazards must be eliminated or minimized, since isolation of a reactor compromises its utility and increases its cost. It is, therefore, important to find means for evaluating the hazards of specific reactors and methods of improving the safety of reactors in general.

The ultimate question in an evaluation of reactor safety is the question of what will happen if the reactor is inadvertently made supercritical and allowed to "run away" without any artificial limitation of its power. For, although safety devices which impose artificial limitations will certainly be

provided in the design of any reactor, the possibility of their failure, as well as the consequence of their finite speed of operation, must be recognized.

In general it can be said that the reactivity of any reactor can be related to its power level once the power has become sufficiently high to cause significant changes in the temperature of the reactor parts. At moderate power levels this dependence can be such as to cause the reactivity either to decrease or to increase with power level, depending on the design of the specific reactor, but eventually at some power level any runaway reactor will become subcritical, through some degree of disassembly of itself if not by other means. The safety question has to do with how violent the energy release becomes before the eventual shutdown is achieved.

The more important ways by which increasing power can cause a reactor to lose reactivity are by expansion of the fuel, by heating and expansion of the moderator, and, if strong resonance absorbers are present, by Doppler broadening of the resonances. In many cases the unknowns in the magnitude of the applicable effects and in their speeds of operation make safety evaluations quite uncertain.

The most straightforward way of evaluating the unknown aspects of the reactor shutdown process is by experiment or, better yet, by observing experimental runaways of actual reactors. As reported in the following chapters of this book, many experiments and observations of the performance of reactors under runaway conditions have been made, particularly on two types of reactors which have favorable power-limitation characteristics, the water-moderated, solid fuel reactor and the water-moderated, homogeneous reactor.

Homogeneous reactors, using a fuel solution of enriched uranium in water, can, in general, be designed to have a negative temperature coefficient of reactivity. The negative coefficient results primarily from thermal expansion of the fuel solution, which decreases not only the density of the moderator but that of the fuel as well and causes it to expand into pressure reservoirs. The coefficient is quite large for small reactors with high neutron leakage. Furthermore, since the heat of fission is liberated directly in the fuel solution, the action of the negative coefficient is very rapid, and insofar as such reactors can limit their power by temperature coefficient alone, they can be expected to be effectively protected against destructive runaways.

Water-moderated, solid fuel, heterogeneous reactors may derive a certain degree of power limitation from the Doppler coefficient if they contain a large fraction of U^{238} . Beyond this, power limitation comes from the moderator temperature coefficient and increased neutron leakage from the core due to expulsion of moderator from the reactor core by the formation of steam at the hot surfaces of the fuel elements.

PRELIMINARY INVESTIGATIONS

The early experiments conducted at Argonne National Laboratory showed that the expulsion of moderator by steam can be very rapid.^{2,3} During the early part of 1949 an experiment was set up to establish the amount of heat that an MTR (Materials Testing Reactor) fuel assembly could dissipate when immersed in a bath of water. Since each assembly in the reactor generated about 1200 kw during operation, a demonstration, involving the removal of about 80 kw of heat by boiling, would serve to dispel fears of melting due to *afterglow* fission product heat. The aluminum fuel element was divided by a saw cut so that low voltage electric current could pass down one side and up the other side to provide heating. The fuel element was tested in a barrel of water. Power was supplied by a 150-kva, low voltage surplus transformer.

In the course of the experiment it was found that the available heating capacity of 90 kw was insufficient to overheat the fuel element. The results of this first crude experiment were encouraging and it was decided that further work should be done along these lines.

This first experiment was primarily concerned with heat transfer. Another question required an answer and this had to do with safety of a boiling reactor. In connection with the design of CP-5, the Argonne Heavy-water Research Reactor, it was necessary to demonstrate that steam could be formed fast enough to shut the reactor down before the fuel plates would melt, even if the controls failed to shut down the reactor. So another series of experiments was started.

In this work, water-filled metal tubes from 1 to 4 cm in diameter and from 10 to 30 cm in length were suddenly heated by the momentary passage of low-voltage alternating current, as shown in Fig. 1-1. The current was furnished by the low-voltage winding of a transformer. At first a simple switch was used to apply the burst of current through a small Variac transformer to a water-filled tube which formed part of the single turn secondary of the Variac. Surprisingly, water was expelled from the tube by steam formation within milliseconds after the current was applied.

Later, a demonstration was arranged in which the water as it was expelled from the tube actuated a switch to turn off the power. With this automatic shut-off arrangement the tube never melted, but if the switch was not opened, it would melt within a fraction of a second. In later experiments, the duration of the transformer primary current surge was more accurately controlled by means of an ignitron tube, as shown in Fig. 1-1.

Sufficient power was available to heat the tube walls to over 500 F within 0.01 sec. Tests were performed both with water at the boiling point and with water subcooled below the boiling point. A typical test

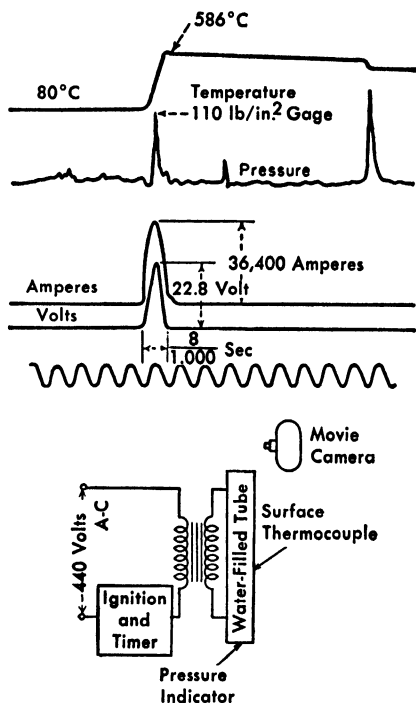


FIG. 1-1. Typical record of water expulsion test. Expulsion from a $\frac{5}{16}$ in. inside diameter 0.015-in. wall copper tube, 6 in. long, started in 0.003 sec.

record of one of these experiments is shown in Fig. 1-1. Steam formed within the tube started to expel water within 0.003 sec after the power was applied.

Thus, it appeared that this process was fast enough to control a reactor. Originally the experimenters had thought that boiling might be able to control slow-acting heavy-water reactors, but the speed with which steam was formed in these experiments convinced them that other types of reactors, notably light-water moderated reactors, could also be controlled. In fact it was difficult to conceive of any realistic thermal runaway so rapid that it would outrun the bubble formation mechanism.

This knowledge was encouraging because time delays in bubble formation were of utmost importance in reactor self regulation. Most thermal reactors have prompt-neutron lifetimes ranging from 10^{-5} to 10^{-3} sec. If a typical reactor has a neutron lifetime of 10^{-4} sec, then a sudden increase in reactivity to 1 per cent beyond prompt criticality would cause the reactor power to increase by a factor of 2.7 in 0.01 sec. To quench such reactivity excursions before the fuel plates melt, steam bubbles must

appear within a few hundredths of a second. It was for this reason that the experiments started in 1948 were important.

Early in 1950 a proposal was submitted by Argonne for an experiment to be conducted at an isolated location, such as Frenchman Flats in Nevada. The purpose of this experiment was to seek answers to the following questions:

1. Would steam formation actually limit the extent of a reactor power excursion, so as to act as a dependable second safety system?
2. Would boiling control an actual reactor, and would such a reactor produce useful amounts of power?

The proposal stipulated the following specifications:

1. The reactor should use light water as a moderator, using available MTR fuel elements.
2. The tests should be conducted at an isolated spot with the control room $\frac{1}{2}$ mile from the reactor site.
3. Transients, involving attempted destructive tests, should come first, so the reactor would be proved safe before steady operation was attempted.
4. All equipment should be simple and portable. A recommendation was made that the control room, reactor, and water supply system each be mounted on separate trailers. As described later, actually the control room was on a separate trailer but the reactor was installed in a hole in the desert.
5. Control might be achieved without rods by raising or lowering the water level in the reactor. (This was tried and worked very well but control rods were also provided in the actual experiments.)
6. Calculations were submitted to show that either a light- or heavy-water boiling reactor was feasible, provided the tests did not uncover unexpected difficulties.

This, then, was the original proposal for what later came to be known as the BORAX experiments.* This proposal received considerable detailed attention within the Argonne National Laboratory, particularly from members of the Reactor Engineering Division. In the course of discussions on the plan, calculations were presented showing that boiling water reactors would be impractical if Stokes' Law assumptions for rate of steam bubble rise through water were applicable. If Stokes' Law was valid in

* The basic proposal was made by S. Untermyer and is referred to in ANL-4774 and 4831. Numerous working memoranda were consulted in drawing up the plan of the experiment and acknowledgment is made to W. H. Jens, N. L. Krisberg, P. A. Lottes, R. R. Rohde, W. P. Staker, F. W. Thalgott, M. Treshow, S. Untermyer, and J. M. West for assistance in the project.

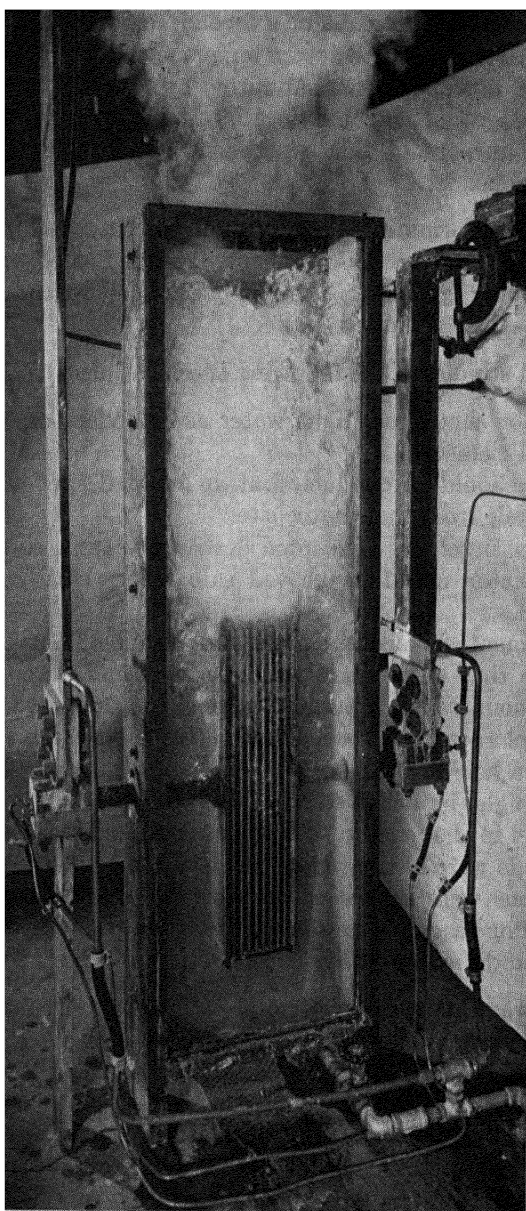


FIG. 1-2. First proof that usable amounts of power could be obtained through the use of the boiling principle was obtained with this laboratory apparatus. Electrically heated plates produced steam velocities greater than predicted theoretically.

this case, it was shown that the assumptions would lead to steam slip velocities lower than 3 ft per sec. Thus, the performance of boiling reactors would be unsatisfactory.

Others who had been involved in these early experiments from the beginning questioned whether Stokes' Law was applicable in the case under consideration, so another series of experiments was started to determine how much power could be removed by natural circulation under boiling conditions. Tests with the MTR fuel elements were difficult to understand if Stokes' Law was applicable. In the MTR element heating experiments, the steam flow area was about $\frac{1}{30}$ sq ft while 90 kw of heat energy was applied, corresponding to steam production at the rate of 2 cu ft per sec. Thus, even if there were no water in the channel, the steam velocity leaving the fuel would have to be at least 30 ft per sec, which seemed incompatible with a Stokes' Law model. A typical set-up of one of these experimental arrangements is shown in the photograph, Fig. 1-2.

The success of the preliminary experiments on heat transfer and steam formation led to the decision to proceed with large-scale experiments with actual nuclear reactors in accordance with the proposal made to the AEC. These were named BORAX from Boiling Reactor Experiments, and it was decided to construct the reactor at the AEC National Reactor Testing Site in Idaho instead of at Frenchman Flats as originally suggested.^{4,5}

In the meantime it was believed that more data were needed for the design of practical boiling reactor power plants in the event that the BORAX tests were favorable. So further tests were made. Using a direct-current power supply, steam was generated in an electrically heated element made by winding a thin plate into a spiral. The steam content of the boiling mixture in the assembly was measured using a Geiger tube to determine the attenuation of gamma rays from a cobalt source. Cobalt, however, proved too energetic for this purpose and poor accuracy was obtained, but there were strong indications that the single bubble, Stokes' Law model was inadequate. Next, the much larger electrically heated device shown in Fig. 1-2 was designed and constructed. In this case a much larger rectified power supply was used. The heated assembly was an accordion type element made from a single sheet of stainless steel. In this case the void fraction was measured by means of a sensitive densitometer with x-rays obtained from a thulium-170 source. This radiation passed through the boiling section and was detected with a sodium iodide crystal and photomultiplier, the latter being connected to a fast response recorder. The recorder could then be used to measure either the steam void fraction or its variation within the heated section. Glass tank walls were provided for visual observation, and operation was limited to atmospheric conditions,

Encouraging information from this test, in which steam velocities as high as 70 ft per sec were measured, led to the construction of a larger test, designed to operate at pressures up to 150 psi. This unit had a 5-ft heated assembly with plates spaced $\frac{1}{2}$ in. apart. It had previously been found that plate spacing had little effect on void content, as measured at atmospheric pressure in the first test unit. In the larger unit, power release up to 300 kw could be dissipated. This unit included a condenser to remove and condense the steam, and improved control equipment. This work is summarized in an article published in July 1954.²

With the improvement noted at high-pressure operation, steps were taken to construct a still higher power test apparatus, and an alternating-current unit was designed using the 150-kva transformer as the power source. This unit with a heated length approximately 4 ft long, could operate at pressures up to 600 psi.

These first experiments at Argonne National Laboratory indicated that boiling reactors might prove feasible insofar as heat transfer, steam formation, and steam velocities were concerned. Data were still needed concerning the effect of steam voids on reactivity and this, of course, involved the question of safety in boiling reactors. Would the formation of steam voids in the core of the reactor be effective in shutting down the reactor? It was primarily to answer this last question that the BORAX experiments were conducted at the AEC Reactor Testing Station in Idaho.

THE BORAX TESTS

To date, the BORAX experiments can be divided roughly into four groups, in terms of the differences in the reactor arrangements. These experiments are described in detail in Chapter 3 and only the essential features of the four groups will be considered here.

The first of these experiments, known as BORAX-I, was conceived as a means of determining whether, or not the formation of steam in the reactor would reduce the reactivity and so cause the system to become self-regulating. During the summer of 1953, BORAX-I not only demonstrated that a boiling reactor has a high degree of inherent safety in its ability to shut itself down in a power excursion, but also that it can operate stably under certain conditions. BORAX-I was a nonpressurized reactor and tests were run with water at various temperatures.

After additional tests in the summer of 1954, BORAX-I was destroyed in a violent subcooled power excursion by the sudden formation of steam in the reactor vessel.⁶ Shortly thereafter a new reactor known as BORAX-II was constructed nearby, utilizing the same control equipment and many

of the actual parts of BORAX-I which had been salvaged after the explosion. This new reactor was capable of operation at 300 psi and, with twice the BORAX-I core volume, delivered about 6 Mw.

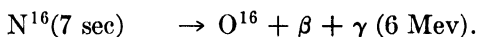
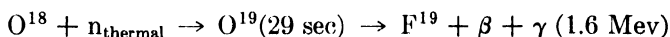
In 1955, BORAX-III was installed. This was essentially a new core in the same pressure vessel used for BORAX-II, and at the same time a turbine and generator were added to provide a complete power generating system.

The first three BORAX reactors used enriched uranium-aluminum fuel elements. In 1956, BORAX-IV was designed using oxide type fuel elements and a new operational program was undertaken, as described in Chapter 3.

BORAX-III was the first direct cycle boiling water reactor power plant to be built. Its heat rating was about 15 Mw and it delivered steam to a turbine-generator unit of 3500 kw electrical capacity. During operation, this plant produced up to 16 Mw of heat in the form of 300 psig steam. At 16 Mw the average power density was 28 kw per liter of core and the amount of reactivity controlled by steam voids was about 2 per cent. Under these conditions the reactor was stable but oscillations in power level were of objectional magnitude, so the reactor was usually run at about 12 Mw. The average density then was 22 kw per liter and the reactivity controlled by steam voids was about 1.5 per cent. Under these conditions short period fluctuations in power level did not exceed 4 per cent. Because the total heat transfer surface was large, the average flux was only 40,000 Btu per sq ft per hr.

One of the uncertainties in the boiling reactor system was the amount of radioactive carryover that would be present in the steam passing to the turbine; indeed, this aspect of the boiling reactor caused considerable concern. The BORAX experiments dispelled much of the apprehension on this point because it was found that the radioactivity carried over was even smaller than had been anticipated.

Almost all of the radioactivity in the coolant of a water-moderated and cooled reactor, assuming that the water is pure, is due to short-lived oxygen decay. There are two principal activities in oxygen:



The second reaction predominates, but neither reaction is very probable, so that the actual activity encountered in the coolant of a water-cooled reactor is not serious.

The fact that the coolant in the boiling reactor leaves the reactor in the form of steam is a distinct advantage. The gammas produced during the N^{16} decay are extremely hard, so that large thicknesses of dense shielding are required to reduce the radioactivity to a tolerable level. The velocity of steam can be many times higher than the velocity of water, since its density is much lower. Thus the amount of coolant contained in the piping external to the reactor is much lower when the coolant is handled in the form of steam than when water is used in the form of liquid.

Another factor of importance is that a boiling reactor is basically a distillation system, and the concentration of non-volatile impurities in the steam is very much less than in the reactor water. Actually the amount of radioactivity in the steam above the reactor water level is some 10,000 times less than in the water itself. The great bulk of radioactivity is concentrated within the shielded reactor tank.

With the availability of the heat transfer data from the electrically heated tests conducted at Argonne, and following the gratifying success of the early BORAX work, a proposal was developed for a larger boiling reactor plant to be built at Argonne, near Chicago. During the winter of 1951, this proposal was presented to the AEC General Advisory Committee in Washington, D. C., as a basis for an Argonne Laboratory Reactor Construction program. Eventually, in 1954, this early move led to Argonne's proposal to build a 20 Mw Experimental Boiling Water Reactor power plant as part of the Atomic Energy Commission's five-year program for the development of power reactors. On the premise that this type of reactor had attractive inherent safety characteristics, it was suggested in a preliminary hazard report⁷ that this reactor should be constructed at the Lemont site of the Laboratory. After reviewing this preliminary evaluation, the Advisory Committee on Reactor Safeguards concluded that "the operation of this reactor at the Argonne site should not present undue hazards if adequate containment is provided." In the early proposal this project was known as the Boiling Experimental Reactor (BER) but it was later renamed the Experimental Boiling Water Reactor (EBWR). Construction of EBWR was started in 1954 and the plant began operation late in 1956. The plant was designed by the Argonne National Laboratory staff working in collaboration with the engineering firm of Sargent & Lundy of Chicago, and Allis-Chalmers Mfg. Co.

EBWR was the first demonstration boiling reactor power plant to be completed. The success of the BORAX-I, II, and III experiments, however, had not gone unnoticed, particularly since they had been fully described at the first Geneva Conference on the Peaceful Uses of Atomic Energy in 1955, and several other organizations had become interested in boiling reactors.

RECENT BOILING REACTOR DEVELOPMENT

Early in 1955, General Electric Company signed a contract with Commonwealth Edison Company of Chicago, Ill., and the Nuclear Power Group, Inc., to build the 180,000-kw Dresden Nuclear Station near Chicago. This station, the largest all-nuclear power station yet scheduled in the United States will be a boiling water reactor plant, but unlike EBWR, which is a direct cycle plant, Dresden Station will use a dual-cycle boiling reactor, this being partly direct and partly indirect, bridging the gap between boiling and pressurized reactors. Construction on this station was started in the spring of 1957 and it is scheduled for completion early in 1960.

General Electric officials believed that a small development plant was needed to assure success of the full-scale Dresden reactor, and in the spring of 1955 a proposal was made for the construction of a 2500-kw (electrical) reactor to be used to develop operational data for application to the Dresden design. Shortly thereafter, the size of the proposed reactor was increased to 5000 kw and the Bechtel Corporation of San Francisco was chosen as engineer-constructor for the project. Bechtel also is the engineer-contractor for the Dresden Station. This developmental boiling water reactor, which is a part of General Electric's Vallecitos Atomic Laboratory near Pleasanton, California, will serve initially as a pilot plant for the Dresden Station. It constitutes a major part of the over-all experimental facility.

Shortly after the decision was made to build this reactor, the Pacific Gas & Electric Co., a member company of the Nuclear Power Group, agreed to participate in the developmental reactor project by supplying a turbine-generator which would use the steam produced by the reactor. The utility agreed to contribute \$572,000 toward the turbine-generator installation and to operate the generating facility as a part of its regular electric network. By December, 1956, major buildings of the reactor installation were in place, and by the end of the winter construction work on the interior of the reactor containment vessel was completed and installation of the equipment started. Construction was completed by mid-June 1957, and after a thorough inspection by the Atomic Energy Commission, a license to carry on "critical experiments" was issued by the Commission in July 1957. On October 24, less than 17 months after groundbreaking, the reactor and the power generating installation went "on line" at full power, and 5000 kw of nuclear energy generated electricity began flowing into the Pacific Gas & Electric system. Thus, Vallecitos became the second boiling water reactor power plant to go into commercial operation as a supplier of electric power.

TABLE 1-1
REACTOR CHARACTERISTICS

Reactor	Maximum pressure, psig	Fuel material	Fuel conductivity	Operator	Sponsor
BORAX-I	130	Al + U ²³⁵	high	Argonne	USAEC
BORAX-II	300	Al + U ²³⁵	high	Argonne	USAEC
BORAX-III	300	Al + U ²³⁵	high	Argonne	USAEC
BORAX-IV	300	ThO ₂ + U ²³⁵	low	Argonne	USAEC
SPERT-I	0	Al + U ²³⁵	high	Phillips Petroleum	USAEC
EBWR	600	U-Zr-Nb (1.44% enriched)	inter-mediate	Argonne	USAEC
Vallecitos	1000	Stainless steel + U ²³⁵	high	General Electric	General Electric and Pacific Gas and Electric

Vallecitos differs from EBWR not only in the fact that it is arranged for dual-cycle operation, if desired, but also in that it operates at a steam pressure of 1000 psi, whereas EBWR is designed for operation at 600 psi.

Commonwealth Edison Company's Dresden Station, when completed in 1960, will probably be the first really large-scale boiling reactor plant in the world, and will be the largest all-nuclear plant in the United States. It is located in Grundy County, Illinois, west of the confluence of the Kankakee and Des Plaines Rivers, approximately 50 miles southwest of Chicago. Like Vallecitos, Dresden will operate at a steam pressure of 1000 psi. The dual cycle has as its objective removal of additional heat without exceeding the maximum allowable steam volume in the core. This is accomplished by removing heat from the saturated recirculating water and then pumping the cooled water back into the reactor. The heat so abstracted from the recirculating water is delivered to an intermediate stage of the turbine in the form of steam at a pressure approximately half of that leaving the reactor.

The boiling reactor concept was a relatively late comer in the field of reactor technology, but the results of the experimental work that have been done on this type of reactor have given it an important place in the field of reactor development. From the beginning, its inherent simplicity made it of great interest to central station designers, and its importance is reflected in the fact that out of the slightly more than a million kilowatts of nuclear power to be installed in the United States through 1962, one third of this capacity will be in boiling reactors. This commercial accept-

ance of the boiling reactor is significant because the early experiments with boiling reactors reported the existence of power oscillations of large magnitude and these reports gave rise to considerable concern, since they might prove dangerous or uncontrollable.

It is typical in any development that attention is focused initially on those features which are most likely to be limiting. As these features become clear, development spreads from the initial concentration in a few areas to become a balanced attack on all technological fronts. It is unfortunate that the initial concentrations of thought and effort on the possible limitations of a system are likely to be prejudicial against it during its early life, and it is only by detailed understanding that acceptance is achieved ahead of widespread application.

To some extent this course of evolution characterized the early development of boiling reactors. Of the aspects of boiling reactors initially thought to be limiting, possible instability associated with boiling in a reactor was the least well understood phenomenon and, therefore, received the greatest initial attention.

In almost all the boiling water reactors operated to date, the reactors have been intentionally operated at increasing power densities until uncontrollable oscillations have occurred. The fact that very satisfactory operation was observed at less severe operating conditions was not always newsworthy. The operating performance of EBWR, over a period of a year and a half has shown that under normal conditions this boiling reactor is completely stable. Moreover, since it began operation in December 1956 at a power level of 20 Mw (heat), its power level has been increased, first to 50 Mw, and more recently to over 60 Mw with no indication of instability. The Vallecitos plant also has been operated at considerably higher power levels than its initial rating without instability.

To date there have been seven boiling reactors operated as well as two others in which bulk boiling has been permitted to occur. These, as already mentioned, fall mostly into the general category of fully enriched, small boiling reactors built primarily for experimental purposes. These include the four BORAX reactors and an experimental reactor operated by Phillips Petroleum known as SPERT (Special Power Excursion Reactor Test). The most recent two (EBWR and Vallecitos) have operated at higher pressures and were built for larger development plant purposes. A summary of the characteristics of the boiling reactors operated to date is given in Table 1-1.

Several other experimental boiling reactor projects are being undertaken. For example, a small 300-kw electrical direct-cycle plant for use in remote areas by the military services is being constructed in Idaho by Argonne National Laboratory. This is known as ALPR, the Argonne Low Power

Reactor. It is a natural circulation, 300 psig, saturated steam boiling reactor and will supply 200 kw, net, electrical and 400 kw, net, thermal for space heating in anticipated isolated stations.

This low power reactor is of interest for several reasons. It is the first reactor to be built in the United States for heating and is the first attempt to design a reactor with a core to last three years before replacement. Certain features of construction also are of interest; the reactor together with all associated equipment is designed to be installed above grade level, since excavation in the frozen north is difficult. The details of this reactor are presented in Chapter 8.

Another experimental facility which has been involved in boiling reactor studies is the SPERT reactor facility already mentioned operated by Phillips Petroleum Co. at the Idaho test site for the Atomic Energy Commission. The SPERT project has been established as a part of the AEC reactor safety program for the purpose of conducting studies on the broad class of heterogeneous, water-moderated, enriched fuel reactors. Thus, its purpose is not only the investigation of boiling reactor safety but also pressurized reactor safety. Considerable work, however, has been done with SPERT on non-pressurized water-moderated reactors.

Studies are also in progress under an Advanced Water Reactor Program at Argonne on two conceptual designs of heavy-water boiling reactors. The first is on a 1035 Mw (thermal) forced circulation, natural uranium fueled boiling reactor, while the second involves a 1000 Mw (thermal) natural circulation, superheating reactor fueled with enriched uranium. These studies have as their object the design of very large nuclear plants for central stations. The second study is of particular interest because of the superheating aspect. Preliminary studies indicate a core composed of 330 fuel assemblies in an 8-inch lattice contained in a 16-foot diameter vessel. The core is divided into two zones: (a) a central flattened, superheating zone which tentatively consists of 42 fuel assemblies; (b) a boiling zone which comprises the remaining 288 assemblies. In the central zone the steam will be superheated to 850 F which produces a cycle efficiency of 32.5 per cent.

In addition to these experimental boiling water projects a number of other commercial boiling reactor projects in the United States are in course of development. Table 1-2 is a list of the reactors which are either in operation or are scheduled to go into operation before 1963. This table shows that of the 17 reactors listed, six are boiling water reactors with a total electrical capacity of 328,000 kw. This is approximately one-third of the total of 1,071,500 kw.

Four of these plants have been mentioned briefly but detailed descriptions of all these reactors are given in later chapters. Particular emphasis is given to EBWR in Chapters 5 and 6 because of its basic importance in

the boiling reactor program and because of its substantial operating record since it went into service in December 1956.

Three other boiling reactors for commercial electric power generation are scheduled to go into service between 1960 and 1962. Northern States Power Co. is building a 66,000-kw boiling reactor plant near Sioux Falls, North Dakota. Allis-Chalmers Manufacturing Co. is building the reactor for this plant. Another projected plant not far from Minneapolis is that being developed by Nuclear Products—Erco, a Division of ACF Industries for the Rural Cooperative Power Association at Elk River, Minnesota. This plant is of interest in that it will be basically a two-circuit steam generating unit, using steam produced in the boiling reactor to generate steam for the turbine in an external heat exchanger. Saturated steam will

TABLE 1-2

U. S. NUCLEAR POWER REACTORS PLANNED FOR OPERATION BEFORE 1963

Project	Reactor type	Power, electric kw	Startup
Experimental Boiling Water Reactor—ANL	Boiling water	5,000	1956
Sodium Reactor Experiment—NAA—SCE	Sodium graphite	6,000	1957
General Electric & Pacific Gas & Elec.	Boiling water	5,000	1957
Shippingport Pressure Water Reactor	Pressurized water	60,000	1957
WAPD—Duquesne			
Homogeneous Reactor Experiment—ORNL	Aqueous homogeneous	1,000	1957
Experimental Breeder Reactor No. 2—ANL	Sodium fast breeder	20,000	1960
Rural Coop. Power Association	Boiling water	22,000	1960
Commonwealth Edison—G.E.	Boiling water	180,000	1960
Consolidated Edison Co.—B. & W.	Pressurized water	275,000	1960
Power Reactor Development Co.—APDA	Sodium fast breeder	100,000	1960
Yankee Atomic Electric Co.—West.	Pressurized water	134,000	1960
City of Piqua	Organic moderated	12,500	1961
Consumers Public Power District	Sodium graphite	75,000	1962
Northern States Power Co.	Boiling water	66,000	1962
East Central—Florida	Heavy water, gas-cooled	50,000	1963
Pacific Gas and Electric	Boiling water	50,000	1962
Chugach Electrical Association, Inc.	Sodium, heavy water	10,000	1962
Total		1,071,500	

be generated in the reactor at a pressure of 875 psig. This will pass through a primary heat exchanger where the steam condenses to give up its heat to a secondary coolant for the indirect production of steam. This steam, then, will be superheated in a coal fired superheater and delivered to the turbine throttle at 600 psig, 825 F. The reactor will produce 58.2 Mw of heat and the separately-fired superheater will produce an additional 14.8 Mw of heat. The turbine generator will be rated at 22,000 kw.

Finally, the Pacific Gas and Electric Company which has been associated with the General Electric Co. in the Vallecitos project is making plans for a 50,000-kw electrical, boiling reactor plant to be constructed at its existing Humbolt Bay Station near Eureka, California. Although this plant will be rated 50,000 kw, 60,000 kw is expected. The Humbolt Bay reactor will employ completely natural circulation and will have the largest steam flow using the single cycle principle of any reactor design so far contemplated for electric generating purposes. This station represents an enlarged Vallecitos rather than a scaled-down Dresden. The fact that this nuclear plant will be installed in an existing plant, where a 50,000-kw conventional outdoor unit is now in service and a second similar unit is under construction, will effect an important reduction in capital cost.

Further details of all of these reactors are given in the later chapters of this book. The earlier chapters are devoted to a discussion of the basic principles and the design of boiling reactors, including an entire chapter (3) on the BORAX experiments. These are of fundamental importance in the development of this type of reactor since these experiments together with the successful operation of EBWR demonstrated the feasibility of the boiling process for power production.

While much has been learned from the BORAX experiments, EBWR, and Vallecitos, considerable further work remains to be done; problems still exist which must be solved before boiling reactor plants can become economically competitive with conventionally fired central station plants. Some of these problems also apply to other types of reactors. The question of lifetime of fuel in the reactor, for example, is a problem in all enriched uranium reactors.

The unstable operation known as "chugging" which was found in the operation of the BORAX reactors is now of less concern to the designers and operators than it was when first encountered. By means of an analytical procedure using transfer function data, it has been found possible to predict reactor instability with a high degree of accuracy. In the case of EBWR this was carried out for power levels well in excess of the known areas. Although EBWR began operation at a thermal rating of 20 Mw, in recent months it has been operated at power levels as high as 61.7 Mw with the same 114 element core loading. In the analytical studies, figures showed that the EBWR would remain stable to at least 65 Mw (t) with

this core loading. Plant mechanical ratings have prevented extended operation above 61.7 Mw (t) at which the reactor still exhibited operable stability. Thus, one outcome of these tests has been to demonstrate a method whereby boiling reactor stability can be evaluated safely in a quantitative manner.

One of the factors which influences stability is the time constant of heat transfer associated with the fuel element. Recent work with BORAX-IV with the use of oxide type fuels indicate that a very large increase in this heat transfer time constant will certainly improve stability but this is true only if it is found that this heat transfer constant becomes a single dominant feedback delay. A very long time constant would reduce the inherent self-limiting ability during a transient.

Other problems that arise as a result of direct-cycle operation include possible turbine contamination as a result of deposition of activated corrosion products or fission products resulting from possible fuel element failures, excess corrosion caused by high oxygen concentration in the steam arising from water dissociation in the core, leakage through turbine seals which is primarily, though not entirely, applicable to heavy water operation, and several other less important but related difficulties such as condenser design and leakage, feed pump design, feed water regulation, etc.

It is the purpose of this book to consider all these various problems in detail and to describe what is known about them and the possible methods which may be used in their further solution. Thus far, the boiling reactors which have been in operation, though designed with a high degree of conservatism, have indicated an approach which seems to yield greater economy than any of the other present reactor power plants.

Considerable attention is devoted to the description and operation of EBWR in Chapters 5 and 6 because this was the first reactor of this type to go into operation.

The Vallecitos experimental plant of the General Electric Co. is also covered in considerable detail, not only because it also is a boiling reactor in actual operation but because it was built largely as a pilot plant for the Dresden Station of the Commonwealth Edison Co. which, as previously mentioned, will be the first large commercial boiling reactor power station in the world.

REFERENCES

1. OSRD Case S-97, Cooling a Reactor, by the Latent Heat of Vaporization of a Liquid.
2. SAMUEL UNTERMYER, Direct Steam Generation of Power, *Nucleonics*, **12**(7), p. 43 (July, 1954).
3. P. A. LOTTES, Boiling Studies at Argonne Relative to Boiling Reactors. Proc. Conf. on Nuclear Engineering, April, 1955, University of California, Los Angeles, California Book Co., Ltd., Berkeley.
4. J. R. DIETRICH, D. C. LAYMAN, and O. A. SCHULZE, The Proposed Boiling Reactor Experiment, ANL-4921, November, 1952.
5. J. R. DIETRICH and D. C. LAYMAN, Transient and Steady State Characteristics of a Boiling Reactor, The BORAX Experiments, 1953, ANL-5211, February, 1954.
6. J. R. DIETRICH et al., The Experimental Determinations of the Self-Regulation and Safety of Operating Water-Moderated Reactors. Paper P/481, Proc. of International Conference in Geneva, August, 1955, Vol. 13, p. 88. New York: United Nations, 1956.
7. J. M. WEST, G. A. ANDERSON, J. R. DIETRICH, J. M. HARRER, A. S. JAMESON, and SAMUEL UNTERMYER, Preliminary Hazards Summary Report on the Experimental Boiling Reactor (BER) ANL-5261, May, 1954.
8. J. A. THIE, Boiling-Water Reactor Instability. *Nucleonics*, **16**(3), pp. 102-111 (March, 1958).

CHAPTER 2

GENERAL PRINCIPLES OF BOILING WATER REACTORS

TYPES OF BOILING WATER REACTOR SYSTEMS

Boiling water reactor power plants are usually classified according to the degree of interconnection between the reactor and the steam turbine. Those in which steam passes directly from the reactor to the turbine are called Direct-Cycle plants. If a heat exchanger is interposed between the reactor and the turbine so that the steam from the reactor is condensed to generate secondary steam for operating the turbine, the plant is said to be of the Indirect-Cycle type. Various combinations of these two basic types may be used, with part or all of the reactor steam going directly to the turbine, and with additional steam being generated in heat exchangers. Such combinations are referred to as Dual-Cycle systems. Each of the basic types of boiling reactors may operate with either natural convection or forced circulation. The fraction of reactor heat utilized in producing steam also may vary over a wide range, depending upon the coolant temperature and flow rate. The distinction between boiling reactors and pressurized water reactors depends upon whether any bulk cooling occurs in the core, not upon the amount of boiling.

Boiling reactors which produce superheated steam have been proposed but so far none of this variety has been built. Various schemes for producing steam, or additional steam by flashing reactor water, have also been considered. It is to be noted that steam could also be obtained from pressurized water reactors by flashing. If no boiling occurs in the reactor core, such a reactor is properly called a pressurized water reactor.

Most of the work on boiling reactors in the United States has been on heterogeneous types, but work on homogeneous boiling reactors has been done at Oak Ridge and also by the Russians. Russian authors presented a paper at the 1955 Geneva Conference which indicated an active interest in homogeneous reactors with thorium fuel and heavy water moderator.¹ There is considerable interest in heterogeneous boiling reactors using heavy water, and various conceptual designs of boiling heavy water reactors have been proposed. EBWR, the Argonne Experimental Boiling Water Reactor, although operating with light water at present, is designed for future conversion to the use of heavy water.

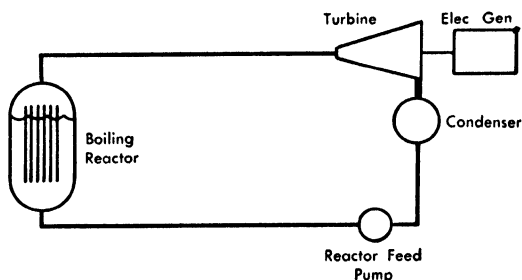


FIG. 2-1. Simplified diagram of the direct boiling cycle.

DIRECT-CYCLE BOILING REACTORS

As shown by Fig. 2-1, the direct-cycle boiling reactor system consists simply of a reactor in which steam is generated, a turbine and condenser and a feedwater pump. Of course, in an actual system, water-treating equipment, waste cleanup and control equipment common to other types of water-cooled reactors would have to be included. Since the steam is generated directly in the reactor, an external steam generator (heat exchanger) is not required. As a result, the same temperature can be attained with a lower average fuel temperature, or a higher steam temperature is possible for a specified fuel temperature. Because no excess pressure is required to prevent boiling of the water in the reactor, as is necessary in water-moderated power reactors of other types, the pressure vessel can be designed for a lower pressure. In the PWR, for example, the pressure in the vessel containing the core is 2000 psi but the pressure of the saturated steam leaving the steam drum is only around 600 psi. In a boiling reactor, the same operating steam pressure can be achieved with a reactor-vessel pressure of a little more than 600 psi. In the direct cycle only the reactor feed pumps are necessary, and since the size of these units is considerably smaller than the coolant circulating pumps in pressurized water systems or in boiling cycles using forced circulation, the pumping requirements of natural circulation boiling reactors are comparatively small.

Since the steam produced in the boiling reactor is used directly, many of the usual thermal barriers within the heat exchanger and reactor are eliminated. Consequently, higher efficiencies should be possible for the same fuel temperature, while tolerable efficiencies should be obtained with reduced temperatures and pressures. This is shown in Figs. 2-1 and 2-2.

If and when reactor technology permits higher reactor temperatures and pressures, there will be corresponding improvements in boiling reactor

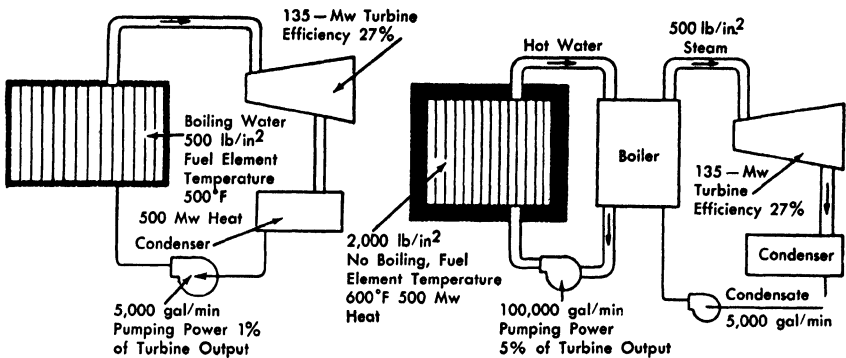


FIG. 2-2. Comparison of direct boiling cycle with cycle using pressurized water coolant and heat exchanger, both systems supplying same turbine requirements.

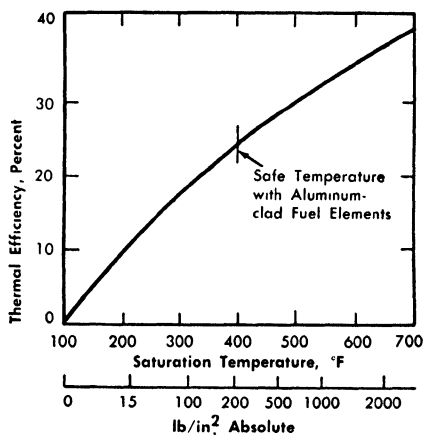


FIG. 2-3. Temperature and pressure limitation on thermal efficiency of direct-cycle boiling reactor.

performance. If steam could be generated within the reactor at the critical pressure, at 3305 psi, a theoretical thermal efficiency of 40 per cent could be obtained, but, of course, that would imply a solution of the moisture content problem which exists with saturated steam. Figure 2-3 indicates the extent of improvement possible in the efficiency of boiling reactors. Until the moisture problem has been solved, however, it is unlikely that these high efficiencies can be obtained.

One of the advantages of the direct-cycle boiling reactor is that it is so simple that it can be constructed in smaller sizes than had at first been deemed practical for power reactors. It seems likely that low-power,

relatively inexpensive boiling plants are feasible because many of the expensive components are eliminated, but until it has been demonstrated it cannot be claimed that such small plants will be competitive with fossil fuel fired plants.

The first direct-cycle boiling reactor plant was the BORAX-III, mentioned in Chapter 1. This was followed by the construction of EBWR. A small 300-kw direct-cycle plant has been built at the AEC Reactor Testing Station in Idaho to serve as a prototype of plants designed for use in remote sites for military installations, and the nuclear plant being built by Allis Chalmers Manufacturing Company for the Northern States Power Company will also be a direct-cycle boiling reactor. As planned at present, this controlled recirculation boiling reactor will have a total core power rating of 203,000 kw (thermal). Of this, 164,000 kw will be in the boiling region and the other 39,000 kw will be in a superheater region of the core. This proposed nuclear superheater constitutes a unique feature of this plant. Earlier designs had contemplated the use of a separately fired, oil or coal superheater but the latest plans call for nuclear superheating.²

General Electric Company has also signed contracts to provide direct-cycle boiling reactor power plants in Latin America.

DUAL-CYCLE REACTOR SYSTEMS

Dual-cycle boiling water reactors are so named because of the two methods of heat extraction used to generate the steam fed to the turbine. The first cycle is a direct cycle in which the steam generated in the reactor passes directly to a steam turbine. The second cycle is an indirect cycle: steam is produced in a steam generator or heat exchanger by subcooling a portion of the saturated water from the reactor.

Thus far, no dual-cycle power reactors have been placed in operation and the analysis of dual-cycle operation presented here is of necessity based on theory rather than experience. While the dual cycle appears to have certain advantages over direct natural circulation systems, it would be unwise to accept them as definite until firm experimental data are available. The two direct-cycle systems now in operation (EBWR and Vallecitos) are both natural circulation systems, and there is no direct experimental information available as to the power increases which might be attained in such reactors with forced circulation.

The dual-cycle system was devised to overcome some of the early apparent limitations of the direct-cycle boiling reactor. In a direct boiling system using natural circulation, as the moderator boils, its density fluctuates and this affects neutron moderation and leakage. Thus, the power

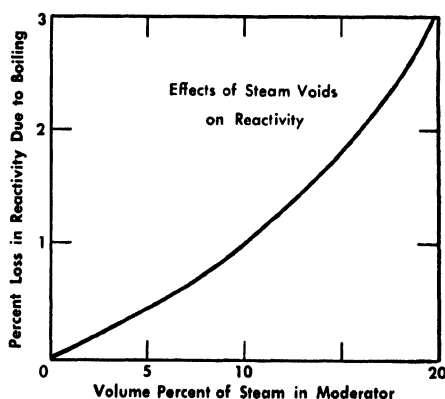


FIG. 2-4. Estimated effect of distributed steam voids on reactivity with core temperature of 486 F.

level will fluctuate and the permissible output tends to be limited by the maximum steam void fraction with which smooth operation can be achieved, and it was thought unfeasible to control more than 3 per cent reactivity with steam voids. Figure 2-4 shows that this corresponds to about 20 per cent steam voids averaged through the core of a direct boiling reactor.

In a natural circulation boiling reactor a sudden increase in load increases steam flow from the reactor, reduces the pressure, core water flashes to steam, and the moderating effect thus lost, causes the flux power to fall. Control rod adjustment is required to increase reactor power to meet the demand. Rod movement, however, takes time and under severe variable load conditions of operation the boiling reactor is less desirable than subcooled types with a negative power coefficient of reactivity.

The initial BORAX tests, as described in the next chapter, the purpose of which was to investigate the safety characteristics, showed that a given boiling reactor can be operated stably up to some power level characteristic of the reactor, and that at higher power levels operation becomes unstable. Early experiments tended to lead to this hypothesis: evidences of instability begin to appear when the amount of reactivity compensated by the steam is something like 1.5 per cent k_{eff} and the instability usually occurs by the time this reactivity was as great as 2.5 per cent. As a corollary, it may be said that, other things being equal, boiling reactors which have a small power coefficient of reactivity will operate stably at higher power densities than will reactors with larger power coefficients of reactivity. This is more fully explained in Chapter 4.

For a small reactor where the length of the steam travel through the core might be as low as 2 ft, a reasonable power density per unit can be

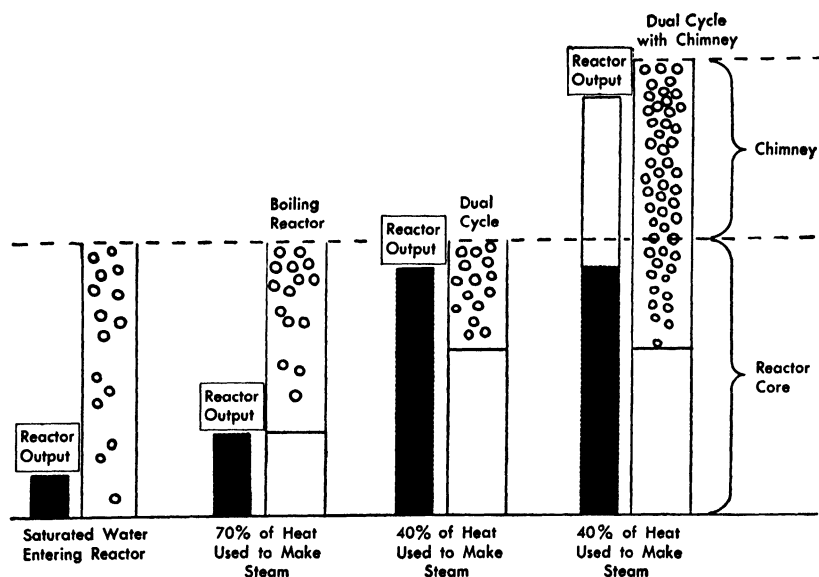


FIG. 2-5. How subcooling improves reactor output.

achieved without exceeding a 20 per cent void fraction. Where the steam path is lengthened to 8 or 10 ft, as in large reactors, the density that could be produced under similar conditions in a simple boiling reactor is reduced so low that unreasonable reactor sizes result.

An analysis of how a natural circulation boiling water reactor's output and load demand characteristics are limited by the way steam bubbles choke coolant flow and reduce reactivity points the way to the improvements possible with a dual-cycle system.

Figure 2-5 shows what happens in a steam-producing channel in a boiling reactor. Each of the reactor channels shown has the same steam void fraction, indicated by the same number of bubbles rising through the water. In the first instance, the water enters the reactor at boiling temperature, steam bubbles form immediately, and their concentration increases as the water rises through the channel, until the bubbles escape from the top of the reactor. All the output of the reactor is devoted to making steam.

In the next instance, the feed water has been subcooled. One-third of the reactor's heat output raises the water temperature to the boiling point, while the remaining two-thirds generate steam. Thus, the water travels about one-third of the way through the core before boiling starts. Since the steam velocity through the reactor is the same in each case, the steam generation rate is established by the number of bubbles leaving the chan-

nel in a unit of time. Since more bubbles are formed near the top of the core in the second case, the steam generation rate is greater. Furthermore, an additional amount of heat is transmitted to subcooled reactor water without net steam generation.

The third example shows the effect of further subcooling, so that two-thirds of the heat is used to raise the temperature of the cooling water to the boiling point. The combined heat output to produce steam and to heat water is further increased.

When the incoming water to the reactor is highly subcooled, the section of the channel that induces natural circulation becomes shorter. This suggests the possibility of promoting circulation by tubes, known as chimneys, in the water space above the fuel. The final diagram is typical of the dual-cycle reactor design; it shows the effect of chimneys two-thirds as high as the reactor. These chimneys augment the natural circulation lift, increasing the water velocity through the system illustrated in the first diagram, if the same steam void fraction is maintained. This illustrates how the power output of a boiling reactor can be multiplied if a supply of subcooled water is available.

This method of reasoning leads to the dual-cycle design^{3,4} shown in Fig. 2-6. In this illustration, it is assumed that the reactor generates steam at 600 psi as shown by the steam line leading directly from the reactor to the turbine throttle valve. This steam passes through the turbine, is condensed and by means of the pumps is returned to the bottom of the reactor. Saturated water from the reactor flows through the flash tank or secondary steam generator where some energy is removed to produce steam at 350 psi. This secondary steam is admitted to a lower stage of the turbine. The cooled water from the secondary steam generator is circulated back to the reactor by means of the circulating pump. This provides the subcooled water for the reactor when mixed with the condensate from the condenser.

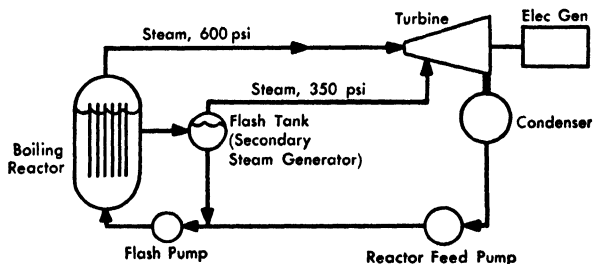


FIG. 2-6. Simplified diagram of the Dual-Cycle Boiling Reactor.

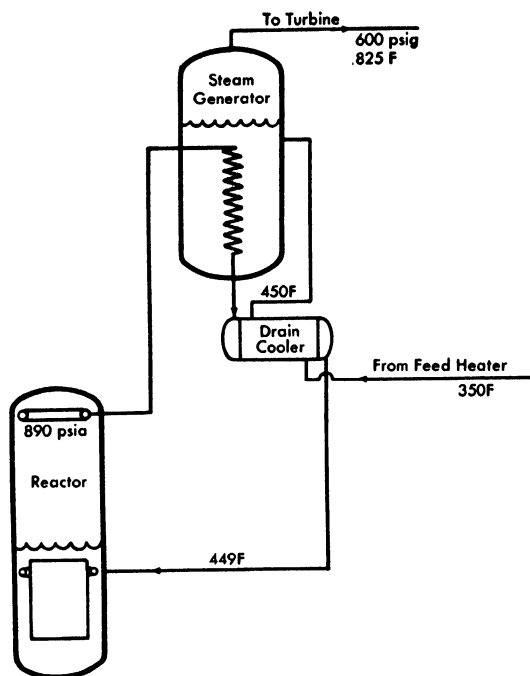


FIG. 2-7. Simplified flow diagram of the Indirect Boiling Reactor System.

The dual cycle endeavors to overcome the two main limitations of the simple boiling cycle, low power output for a given size and poor stability under load variations. It preserves a large amount of natural circulation and much of the basic equipment simplicity of the direct cycle. There is some compromise of the thermal efficiency gains theoretically obtainable in the simple boiling cycle, but this cycle maintains a thermal efficiency advantage over liquid water cooled reactors. The major safety advantages associated with boiling are as effective in the dual cycle as in the simple boiling reactor.

Control of load variations with the dual-cycle system is attained in the following way. Most of the turbine governor adjustment for load variations takes place in the 350 psi admission. Consequently, the subcooling in the reactor increases with load. When there is a demand for more power, the turbine acts to admit more steam from the secondary steam generator while the flow of steam into the 600 psi (high pressure) stage is not immediately affected. As the steam demand from the secondary steam generator increases, the pressure drops and so more water flows into the tank. The flash pump draws off more water from the tank to hold a constant level, and the amount of subcooled water entering the

reactor increases. Consequently the number of steam bubbles within the core is reduced, and the reactor output rises to meet the load demand. Thus, this stability during load changes is obtained without sacrifice of inherent safety characteristics of boiling reactors.

The first dual-cycle reactor to be constructed is the Dresden Station of the Commonwealth Edison Company now being built 47 miles southwest of Chicago. It is a light-water moderated, forced-circulation, boiling reactor having a heat power of 627 Mw and an electrical output of 180,000 kw. This station is scheduled to go into operation in 1960.⁵ A full description of Dresden Station is given in Chapter 9.

The Vallecitos boiling reactor of the General Electric Company, although operating as a direct-cycle plant, is arranged so that dual-cycle operation can be simulated. Included in the pumping loop, provided for forced circulation, is a steam generator which may be cut into the loop by the manipulation of appropriate valves. The steam generator may then be used to simulate dual-cycle operation. In this type of operation, the secondary steam will be exhausted to the atmosphere rather than being used in a turbine. Dual-cycle operation may be obtained with either forced or natural circulation inside the reactor by inserting or removing baffles.

INDIRECT-CYCLE BOILING REACTORS

The indirect boiling cycle evolves from the desire to reduce maintenance and design problems by eliminating radioactive carryover from the reactor to the major pieces of equipment in the system such as the turbine and superheater, if the latter is included. This is accomplished by using two distinct loops linked thermally through a condensing-evaporating heat exchanger. Steam generated in the reactor is condensed on the primary side of the heat exchanger by giving up its heat to generate lower pressure steam on the secondary side. The entire secondary system may be of conventional design and be located external to the containment shell. A simplified flow diagram of the proposed Elk River Reactor, which is an example of the indirect cycle, is shown in Fig. 2-7. The indirect-cycle boiling reactor has available all of the potentialities for increased power through subcooling that are available for the dual cycle.

The heat capacity of the heat exchanger acts as a damper and introduces a time lag which protects the reactor from sudden transients and provides time for the control system to take the necessary corrective action. Because of this time lag, the load response characteristics of the indirect cycle are superior to the direct cycle since the reactor has time to adjust to load change rather than opposing it. This comes about

through the effect of subcooling on reactivity. For an increased load demand, secondary pressure and temperature will attempt to drop, resulting in increased heat transfer across the heat exchanger. There is consequently greater subcooling of the reactor return water which, in turn, means higher reactivity and power. The time delay allows adjustment of control rods to prevent a change of primary pressure, which would affect the power output in the opposite direction. A drop in load results in the reverse sequence.

The added expense of heat exchangers plus piping for two loops instead of one is a disadvantage of the indirect cycle. In addition, due to the temperature drop required across the heat exchanger, the thermal efficiency for a given reactor pressure is less for the indirect cycle than for the direct cycle. Stated another way, for a given turbine throttle pressure, the reactor pressure for the indirect cycle must be greater.

The evolution of noncondensable gases from the reactor into the high-pressure primary system also presents a difficulty. The limit of concentration of gases in steam is set by the tolerable level of oxygen in water to avoid excessive corrosion of zirconium cladding on the fuel elements, when zirconium is used. Since the solubility of oxygen in water is a function of its partial pressure in vapor, the concentration of oxygen in vapor is limited. If this concentration limit is low, then the gases cannot be vented without also removing a great amount of steam and heat. Unfortunately, good data are not available on any of the more important aspects of this problem. The concentration limit of oxygen in water to prevent Zircaloy corrosion, the rate of mass transfer of oxygen into solution in the condenser, and the evolution rate of oxygen in the core as a function of its content in the feed water are all unknown. A calculation may be performed, however, from the limited data available. Using an oxygen evolution rate of 19 cc per liter of condensate, estimated from EBWR and BORAX-III operational data for low concentration levels of oxygen in water, assuming a maximum permissible bleed-off rate of steam equal to 5 per cent of the total flow, assuming an instantaneous solution rate of oxygen in water, and applying Henry's Law extrapolated to high temperatures, a concentration of oxygen in reactor water of 1.15 cc per liter is calculated. This is considerably higher than the 0.01 cc per liter which has caused corrosion of carbon steel in conventional high temperature and pressure boilers. The carbon steel, of course, is more susceptible to corrosion than Zircaloy.

At the present time, reactor experience with this problem is limited. EBWR is operating with oxygen levels of about 0.01 cc per liter, but the effect of higher concentrations is not known. One method of decreasing the concentration would be the addition of hydrogen into the system to

enhance recombination of oxygen in the core. However, this would introduce a serious explosion hazard.

As already mentioned, the proposed Elk River plant of the Rural Cooperative Power Association near Minneapolis will use this indirect cycle. The reactor is a closed cycle, boiling water reactor, cooled and moderated by light water. The primary loop will have no pumps or valves. This is accomplished by raising the level of the evaporator condenser above the water level in the reactor in order to provide the required driving head. Provision is being made to introduce forced circulation for increased power, if required. The Elk River plant is more fully described in Chapter 10.

BOILING HOMOGENEOUS REACTOR

As indicated earlier in this chapter, a considerable amount of work has been done at Oak Ridge National Laboratory on the development of a boiling homogeneous reactor. As early as May 1951, following the construction of HRE-I, the group at Oak Ridge focused their attention on the possibility of removing heat from a homogeneous reactor by boiling the fuel solution, rather than by circulating it, in recognition of the advantages of a boiling reactor. Experiments on bulk boiling at atmospheric pressures in a 1-ft diameter cylindrical tank indicated that power densities up to 5 kw per liter might be achieved. It soon became apparent, however, that high-pressure, power-density measurements would be required, and the design of a boiling reactor experiment (BRE) called the "Teapot" was indicated. To answer the question of nuclear stability of such a reactor, a combined group from the Oak Ridge National Laboratory and Los Alamos operated the SUPO under boiling conditions in October, 1951. Although the reactor operated at a total power level of 6 kw, solution power densities of only 0.5 kw per liter were obtained, owing to boiling in the cooling coils. Later experiments conducted with no boiling in the cooling coils indicated that this reactor could be operated safely as a boiling solution reactor at its maximum power density of 4 kw per liter.^{7,8}

This removed one of the important obstacles to the construction of an experimental boiling reactor, and in January, 1952, the Oak Ridge National Laboratory made a proposal to construct the Boiling Reactor Experiment (BRE). In January 1953, however, the problem of maintaining sufficient oxidizing conditions to prevent reduction and precipitation of the uranium in a boiling uranyl sulfate solution became apparent, and construction of the reactor was deferred pending the outcome of solution-stability experiments. These experiments, completed in October 1953, indicated that at oxygen concentrations likely to be encountered in

a boiling reactor (6 to 7 ppm), reduction of uranium would occur in solution in contact with stainless steel. Since the metallurgy of titanium or zirconium was not sufficiently advanced to construct a reactor using these alternative metals, it was decided to abandon the BRE itself and continue experimental work on the problems of solution stability, steam separation, and power densities at high pressure.¹¹

Work on steam separations and experimental measurements of the movement of air and steam through heated solution at high pressures were carried out by the Babcock and Wilcox Company.¹²

These results and theoretical calculations of the power removal from boiling reactors^{6,13} provide a basis for estimating the obtainable power density of such reactors under varying conditions.

The various applications of boiling as a method of heat removal from homogeneous solution or slurry reactor include the use of a two-region reactor with a boiling core and a boiling blanket. The problem of maintaining a sufficiently oxidizing solution in a boiling uranyl sulfate- D_2O reactor, although serious in a stainless steel system, can be eliminated if all surfaces in contact with the solution are made of titanium and oxygen supplied continuously. Solutions containing 10 g of uranium per liter have been successfully boiled at 325 C in titanium-lined pipe.

Continued interest in boiling homogeneous reactors has led to a number of studies of large-scale reactors of this type. The use of boiling as a method of removal of power from the ThO_2 slurry blanket of large two-region homogeneous reactors does not appear to present any major difficulties.⁹ The apparent advantages are that no circulating pump would be required to handle slurry, containment of the highly active slurry in the reactor vessel, and the possibility of operating the blanket at the core pressure and using the blanket power for heating the secondary steam.¹⁰

The Russians have also been working on this development of a boiling homogeneous reactor and presented a paper on this subject at the 1955 Geneva Conference.¹

This paper described a study made to evaluate the performance of two types of homogeneous boiling reactors, one without a reflector, and the other with a reflector providing for fuel regeneration. These reactor concepts had as their objective the elimination of some of the disadvantages inherent with homogeneous reactors using the fuel slurry. Homogeneous reactors working with solutions of uranium compounds in a moderator demand the use of absolutely reliable pumps for pumping the active mixture through the active zone and heat exchanger; the repair of these pumps must be carried out by remote control, which greatly complicates their operation. Further, the power obtained from a kilo-

gram of fissionable material in such reactors is not high, since a considerable part of all the uranium fills up the circulation circuit and heat exchanger and does not participate in the reaction. Finally, when the heat exchanger utilizes a heat transfer medium which has been activated by neutrons, radioactive steam gets into the turbines which is a great inconvenience.

The boiling homogeneous reactor represents a scheme which to a considerable extent eliminates these difficulties. Such a reactor makes unnecessary the pumps for pumping the working medium; next, the heat in such a reactor is removed directly from the reaction zone by steam, which reduces the total amount of uranium, since the active mixture, in this case, is not pumped through the heat exchanger. Finally, secondary steam entering the turbine is not active.

In the Russian studies, experiments were made regarding the time behavior of suspensions as a function of temperature. An increase in the speed of settling was observed when the suspension of uranium oxide was boiled for several hundreds of hours at atmospheric pressure. Holding the suspension in water at a temperature of 210 F (without boiling) for 20 days did not bring about a considerable change in the speed of settling.

Further experiments were conducted to investigate the precipitation of the uranium oxide powder on stainless steel in contact with the suspension. When stainless steel was in contact with the nonboiling suspension for several hundreds of hours at 210 F, precipitation of the powder onto the surface of the steel was not observed. The experiments were conducted with the suspensions motionless and also when the suspension was continuously being stirred.

In experiments with a boiling suspension conducted over a period of 20 to 25 days at atmospheric pressure, only an insignificant precipitation and adhesion of the powder to the plates of stainless steel placed in the suspension was observed. After a given layer of powder has accumulated on the surface of the stainless steel, further adhesion ceases. When the plate of stainless steel was situated on the interface between suspension and steam, a thick layer of powder adhered to the plate.

The results obtained from the experiments imply the possibility of using a suspension of uranium oxide and thoria for a boiling nuclear reactor.

COMPARISON OF BOILING REACTOR CYCLES

From a practical physical standpoint no nuclear reactor power cycle is more attractive than the simple direct boiling cycle. Its simplicity

both in the equipment used and in the nature of the equipment gives it the lowest possible investment cost. The absence of thermal barriers permit higher efficiencies for the same fuel temperature, while, as indicated previously, tolerable thermal efficiencies can be obtained with lower reactor pressures and temperatures. The simplicity of the direct cycle makes it attractive for small power plants in remote areas where high efficiency is of less importance than simplicity and freedom from maintenance. In fact aluminum cladding is sufficiently corrosion resistant for use in boiling reactors at temperatures that permit reasonable thermal efficiencies.

The dual-cycle approach has as an objective the removal of as much energy as possible by boiling in the upper part of the reactor without exceeding the limit of reactivity for good stability. With the dual system the reactor response to load is correct. The pressure in the reactor is not changed radically by the turbine regulating valve movement. Some load changes are accommodated by actuation of the secondary admission valves by the turbine governor. Decreasing the rate of secondary steam production results in higher temperature water returning to the reactor. The decrease in subcooling causes the reactor flux power to drop. For an increase in load, the reverse procedure prevails. This self-regulating control is obtained by allowing the pressure in the secondary system to vary with the load.

As would be expected, the heat rate of the dual cycle is contingent upon the primary steam pressure and the percentage of power which is removed as primary steam. The size of the largest dual-cycle boiling reactor that can be provided appears to be limited only by the dimensions of the pressure vessel that can be transported to the construction site.

The attractive features of the indirect cycle lie in the fact that there is no radioactive carryover from the reactor into the turbine and associated piping and that it can be coupled into an existing power station. The load response characteristics are improved over those of the direct cycle because of the time lag inherent in the heat exchanger. These attractive features, however, are offset by the added equipment required and perhaps a slightly lower thermal efficiency obtained, but it may prove to be the most attractive for many industrial purposes. One decided advantage of the indirect cycle reactor is that a conventional fuel fired superheater can be used with it.

REQUIREMENTS OF BOILING REACTORS

From the standpoint of stability and safety under boiling conditions, reactors can be divided into two classes:

- (a) Self-regulating reactors in which an increase in reactor power causes an immediate decrease in reactivity. The power is thus limited at all times by the rod setting made by the controls or the operator.
- (b) Auto-catalytic reactors in which an increase in reactor power causes an increase in reactivity excess, resulting in further increase in reactor power. This condition means that, if the reactor operation were not otherwise limited, the reactor would go out of the control of the operator.

All the factors entering into the critical equations may be affected by the boiling and bubble formation. Ordinarily, the largest effects of boiling are increases in: (1) resonance absorption due to expulsion of the moderator; (2) thermal utilization by expulsion of neutron-absorbing coolant; and (3) neutron leakage by expulsion of moderator and scatterer.

Change in thermal utilization is generally negligible for heavy water reactors using natural uranium, while change in resonance absorption is trivial for highly enriched reactors. Leakage is less important in large reactors. Changes in resonance absorption and leakage tend to make reactors self regulating, while the expulsion of a neutron-absorbing coolant, of itself, would make a reactor auto-catalytic.

Either a light- or heavy-water cooled reactor can be designed to have a negative effect on reactivity excess, but it is difficult to determine in advance just how the reactivity will vary under certain conditions. In some reactors the reactivity decreases more or less linearly with coolant expulsion whereas for others the decrease is non-linear, and it is difficult for the designer to choose which type of behavior his reactor will exhibit when it goes into operation. Figures 2-8 and 2-9 show the effect of light-water cooling on reactivity of natural uranium reactors.

A reasonable degree of self-regulation can, however, be provided without seriously compromising other reactor design criteria. Thus practical boiling reactors can be designed so that steam formation will reduce reactivity and hold the power in check. Self-regulating boiling reactors can be constructed using various combinations of fuel, moderator, and coolant, some of which are listed in Table 2-1.

FACTORS INFLUENCING STABILITY AND CONTROL

The possibility of instability in the boiling reactor has always been recognized; in fact, it was the anticipation of such instability that delayed serious work on boiling reactors until recent years. Ever since, the first BORAX tests designers have been keenly aware of the inherent difficulties in predicting the kinetic behavior of boiling systems. If reactivity is

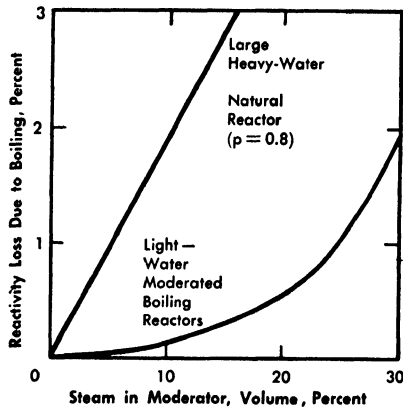


FIG. 2-8. Effect of distributed steam voids on two types of boiling reactors (illustrative curves).

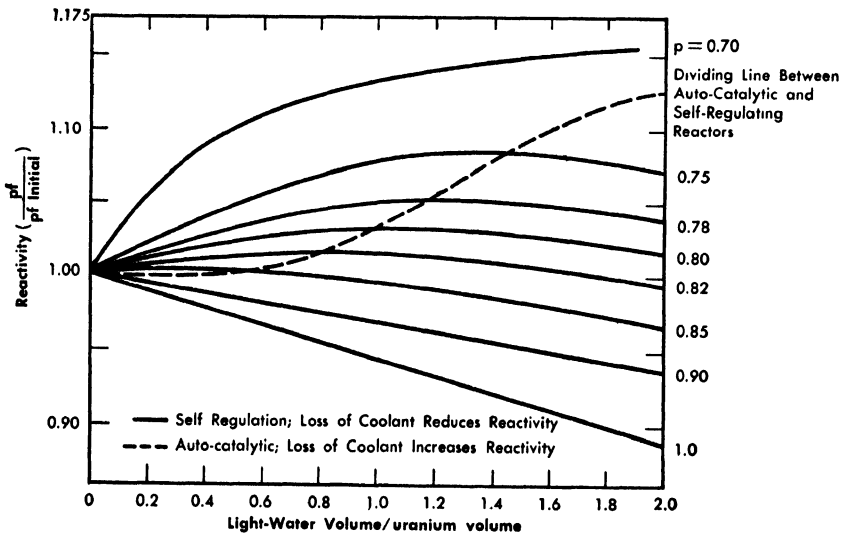


FIG. 2-9. Effect of light-water cooling on reactivity of natural uranium reactor. Moderator is graphite or heavy water.

added to a boiling reactor, oscillatory instabilities can develop, culminating in "chugging." During chugging, coolant is expelled by the formation of steam, reducing reactivity so that the output falls and water drops back into the reactor, increasing reactivity resulting in a power increase, forming steam which again expels the water. While these chugging surges may appear very violent, they are actually limited in severity by the rate at which reactivity can be added by the flow of water back into the reactor core.

TABLE 2-1
SOME POSSIBLE MATERIAL COMBINATIONS FOR SELF-REGULATING
BOILING REACTORS

Fuel	Moderator	Coolant	Self-regulation requirements
Natural uranium	D ₂ O	D ₂ O	None
Natural uranium	D ₂ O	H ₂ O	Limit on amount of coolant. See Fig. 2-9
Natural uranium	Graphite	H ₂ O	Limit on amount of coolant. See Fig. 2-9
Natural uranium	Graphite	D ₂ O	None
Partially enriched uranium	H ₂ O	H ₂ O	Limit on water-uranium ratio
Natural uranium-enriched control rods	H ₂ O	H ₂ O	Limit on water-uranium ratio
Fully enriched	H ₂ O	H ₂ O	Must be small

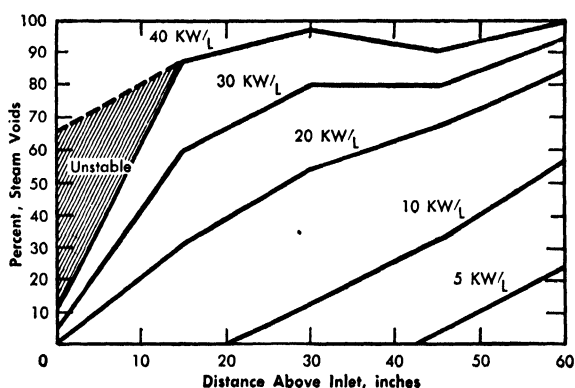


FIG. 2-10. Hydrodynamic instability in heated vertical channel. $D_H = 0.5$ in., 25 psia.

Flow instabilities without nuclear coupling may also lead to a situation similar to chugging. For instance, during non-nuclear heat transfer rate tests at Argonne National Laboratory an instability was observed when boiling extended to the lower entrance of the heated element. This is shown on the chart of Fig. 2-10. Under these conditions, steam was intermittently expelled from the lower end of the heated channel. This might simulate the case where excessive reactivity was tied up in voids.

It is important to know the exact conditions which will lead to chugging since the power of a boiling reactor may be limited by nuclear sta-

bility. In the BORAX experiments chugging often occurred when over 2 per cent of the reactivity was controlled by steam bubbles.

Boiling reactors are designed so that steam bubbles reduce reactivity. When the control rods are withdrawn the reactor output will increase until steam is formed fast enough within the reactor to offset the added reactivity. Consequently, such a reactor will produce a fixed steam flow against a fixed outlet pressure without continuous control adjustment. However, if the outlet pressure is reduced in response to an increased load demand, the reactor will deliver an increased steam flow and may not respond to compensate for this load change—if the change is large.

As has been indicated, the use of the dual cycle affords load matching without rod movement over a wide range. Single-cycle boiling reactors cannot compensate for load demand changes when negative reactivity effects of the steam bubbles are greater than the positive effect of the water temperature change. Stability is improved when operating with low steam void content at high pressures, as shown in Fig. 2-11. This curve indicates the effect of temperature and pressure on stability, but these effects are much greater in the case of low pressure reactors where neutron temperature becomes an important factor tending to increase the temperature coefficient.

Under these conditions, the effects due to change in water density may overshadow those due to steam, and the reactor will then exhibit inherent load matching ability.

With respect to the instabilities that have been observed in some of the experiments on boiling reactors, it should be pointed out that the possibility of instability is not confined to certain reactor types. Neither is an unstable characteristic to be expected frequently in the design of new reactors. When the possibility of instability does exist, it is something to concern the designer, but its significance for reactor safety can be defined only in relation to other dynamic characteristics of the reactor.

In the case of the boiling reactor, the reactor is stable so long as the power level is kept below that at which large uncontrolled reactivity fluctuations can occur. Confidence in the safety of the boiling reactor is also based to a considerable extent on the experimental observations that instability, when it does occur, is preceded by characteristic signs which give ample warning, and by the fact that the boiling reactor has a high degree of tolerance for power excursions. It must be remembered, however, that this tolerance for excursions is very much a function of the specific reactor design and cannot be assumed to be the same for reactors with low conductivity fuel plates as for the early BORAX reactors which had high conductivity fuel plates.

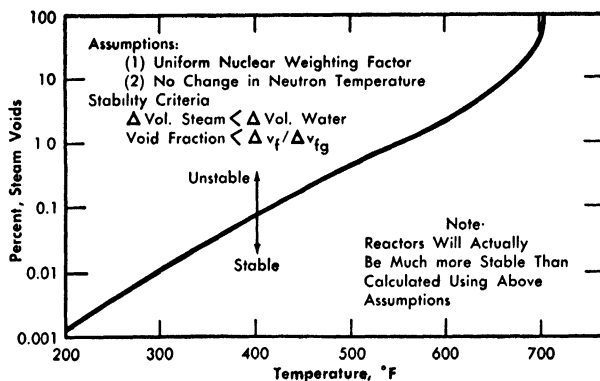


FIG. 2-11. Power demand stability in single-cycle boiling reactor.

In this connection, the BORAX-IV experiments described in the following chapter, which are being conducted by Argonne National Laboratory with fuel elements of thorium-uranium oxide, are of great interest.

Finally, it should be pointed out that the stability limits for boiling reactors which have been observed to date appear to be characteristic of the natural circulation case, and that there is no experimental evidence as to what the limitations will be for forced circulation. It would be surprising if the forced circulation boiling reactor did not have somewhat broader regions of stability than natural circulation designs.

With regard to forced circulation, whether or not forced circulation is used with any boiling reactor system is largely a question of economics. If the desired power densities can be attained for a given power output, it would be uneconomical to use forced circulation since it requires considerable additional investment. However, since the power output of the natural circulation direct-cycle reactor is limited it may be necessary to use forced circulation and subcooling if higher power densities are desired.

STEAM SLIP

When steam and water flow concurrently through a passage, they travel at different velocities, and the amount by which the steam velocity exceeds the water velocity is known as the steam slip. In early Argonne tests it was found that steam slip was strongly dependent on temperature or pressure and it was found that slip increased as the water velocity increased. Since slip rates of over 60 ft per sec were measured at atmospheric pressure, it was apparent that some sort of channeling phenomenon was involved (Fig. 2-12). The dependence of steam slip on pressure and water velocity can be calculated if it is assumed that the steam flows through

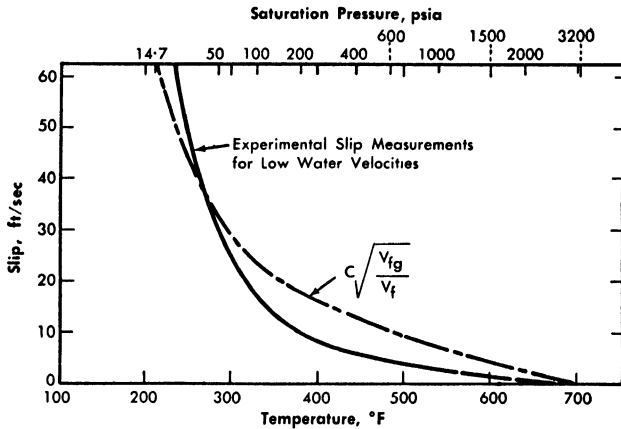


FIG. 2-12. Effect of coolant temperature or pressure on steam slip.

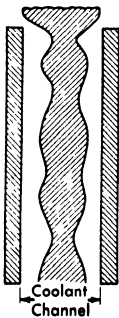


FIG. 2-13. Steam slip model.

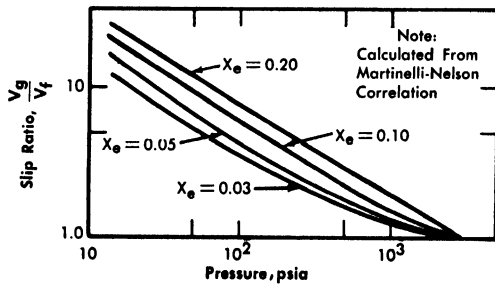


FIG. 2-14. Slip ratio vs. pressure.

channels passing through the water in the coolant passage and that the hydraulic characteristics of the channels do not change. Figure 2-13 portrays a model of steam slip. There is, however, difference of opinion among various investigators concerning the dependence of steam slip on water velocity. Some authorities regard void fraction as the significant variable; higher void fractions, they contend, lead both to higher slip and to higher water velocity.

In Fig. 2-12 the measured variation of slip values with low water velocity is compared with calculated values. The qualitative correlation seems to explain the strong dependence of steam slip on pressure. This model is also consistent with the observed increase in slip velocity.

An "order of magnitude" estimate of the effective size of the steam channels suggests that they are of the order of one-hundredth of an inch thick. Another approach to the prediction of steam slip developed by

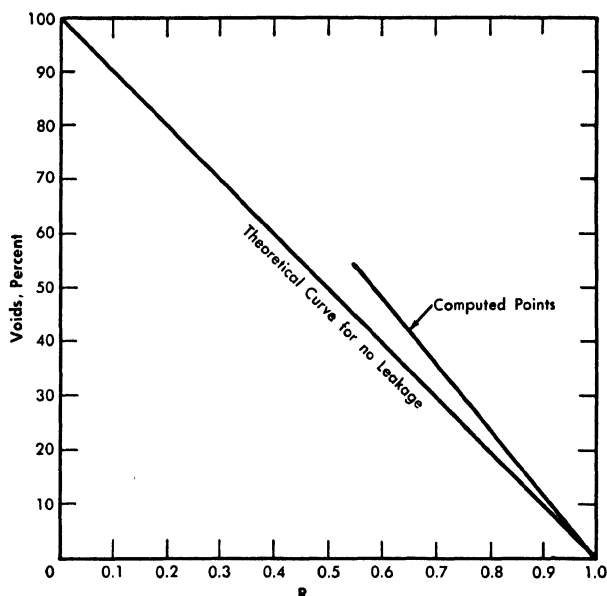


FIG. 2-15. Per cent voids as a function of flux ratio, R .

C. A. Purcell in 1953 is derived from the Martinelli-Nelson correlation for turbulent two-phase flow (Fig. 2-14).

STEAM VOID MEASUREMENT

Calculation methods, based on heat transfer and physics, yield information both about the steam distribution in a reactor and the reactivity controlled by voids. Previously it had been possible only to check such calculations by comparison of observed and computed reactivity, but now a procedure is available which will measure the detailed steam bubble distribution. This method depends upon the moderating properties of the coolant. If the moderator were not present and if the reactor could still be made critical, then all the neutrons would be above thermal energy. The ratio of fast to thermal neutrons present at any position in the reactor depends on the amount of moderation displaced by steam bubbles. Thus "cadmium ratio" measurements at various locations with and without boiling can be made using foils or miniature neutron ion chambers.

The ratio ϕ_f/ϕ_{th} for nonboiling to ϕ_{th}/ϕ_f for boiling is related to calculated void content as shown on the chart, Fig. 2-15. In these expressions ϕ_f and ϕ_{th} represent fast neutron flux and thermal neutron flux respectively. This ratio is 1 at points where boiling does not occur and falls to zero in

places where all the moderator has been removed. This method will correctly indicate conditions in any local regions that are large compared to the slowing down distance that is about 10 cm in light-water reactors.

RADIOACTIVE CARRYOVER

Since the steam generated in the boiling reactor is passed directly to more or less standard power generating machinery, condensed and recycled to the reactor vessel, it is obvious that the carryover of radioactivity constitutes an important factor, not only with respect to safety but also with regard to maintenance. The reactor water contains dissolved and suspended impurities, together with water dissociation products, which are activated by the neutron flux of the reactor. Thus, the steam leaving the boiling interface carries droplets of water and volatile substances which are highly radioactive. The problem is to reduce this radioactive carryover to the minimum possible value.

Fortunately the process of boiling itself provides a natural separating process. In rising in the reactor free space, entrainment is decreased as a result of settling or impingement on the reactor walls and internals. To provide further separation, the primary steam can be made to pass through mechanical separators which, in the case of the experimental reactors in operation, have proved very effective. The steam leaving the separating equipment enters the turbine where deposition of remaining entrained droplets and adsorption of volatile substances can occur. Erosion and corrosion of the equipment contributes additional impurities to the steam. The turbine discharge passes into the condenser where some volatile impurities, both radioactive and inert, are bled via an air removal system. The condensate picks up more impurities as a result of corrosion of the condenser and by in-leakage. The condensate then is returned to the reactor as feed water.

In order to maintain the highest possible purity of the water in the system it is necessary to circulate part of the reactor water through cleanup loops where suspended and dissolved matter is removed. Additional corrosion, erosion, and deposition occur in the reactor vessel and on the reactor core surfaces. Thus occurs a complex transport of erosion and corrosion products, some of which are activated in the reactor. The neutron activation products which find their way out of the reactor into the external machinery and cause turbine contamination are naturally a source of considerable concern. Excessive entrainment of radioactive substances into the power machinery and their subsequent retention could lead to significant problems in shielding requirements and maintenance. The factors that should be considered in the prevention of such contamination include: proper choice of construction materials to

reduce the accumulation of corrosion and erosion products in the reactor, properly designed cleanup loops to maintain high purity of reactor water, careful pretreatment and deaeration of system makeup water, and de-entrainment of water droplets and volatile activation products.

These stringent requirements with respect to radioactive carryover in boiling reactors have led to extensive experimental work both with small-scale laboratory apparatus and on the BORAX reactors.¹⁴ The details of this work is covered in the following chapter. In general, the results of this experimental work have been very encouraging, and also, in the actual operation of EBWR and the General Electric Company Vallecitos reactor the amount of radioactive contamination in the external system has been small and produced no maintenance problem to date.

In the case of EBWR after one year of intermittent turbine operation or 2840 hours of operation, an inspection of the turbine was made. It was in excellent condition with no visually detectable sign of erosion or corrosion on any parts exposed to the steam. Only small deposits or discolorations were noted. The blade deposit was 0.1 mg per sq cm. Radiation surveys were taken 10 days after shutdown when removing the upper half of the turbine casing and the maximum levels found were at the steam nozzle inlets. There the activity was 5 mr per hr at a distance of 2 in. At the throttle valve inlet the activity was 2 mr per hr at 2 in. and at the first row of blades it was only 0.2 mr per hr at 2 in.

In concluding this introductory discussion of the general principles of boiling water reactors, it should be obvious that while much has been learned about the practical design and operation of this class of reactor, many problems remain to be solved before they can be accepted as completely reliable and competitive commercial power stations. It should not be assumed that these problems necessarily impose limitations to the ultimate utility of boiling reactor power plants. They merely reflect the present state of development, one which permits an understanding of the specific problems through experimental investigation but which has not provided sufficient experience to predict the ultimate most effective solution. It would be rather naive to expect that single direct solutions to these various problems will ever be accepted, but there certainly exists a large area for improvement of present designs.

The stability problem which caused considerable concern in the early stages of the BORAX experiments is becoming better understood. Stabilities can now be estimated for a new design through the application of the principle of feedback analysis, as will be explained in subsequent chapters. In general, designers feel that they are aware of most of the major problems concerning boiling reactors, and that the present experimental programs represent feasible approaches to the solutions.

REFERENCES

1. A. I. ALICHANOW, W. K. ZAVOISKY, R. L. SERDUK, B. W. ERSCHLER, and L. J. SUVOROW, A Boiling Homogeneous Reactor for Power. Paper P/624, Proc. of International Conference at Geneva, August, 1955, Vol. 3, p. 169.
2. C. B. GRAHAM, Controlled Recirculation Boiling Reactor. Paper presented at the 1958 Nuclear Congress, Chicago, Ill., March 17-21, 1958.
3. SAMUEL UNTERMYER, Dual Cycle Improves Boiling Water Reactors, *Nuclear News*, July, 1955, p. 34.
4. SAMUEL UNTERMYER, B. R. PRENTICE and A. G. MELLOR, The Dual-Cycle Reactor Plant as a Part of Future Power Systems. Paper presented at American Power Conference, Chicago, Ill., March 30-April 1, 1955.
5. A Design Description of the Dresden Nuclear Power Station. Presented at a joint session of the Nuclear Engineering and Power Divisions, Annual ASME meeting, New York, November, 1956.
6. P. C. ZIMOLA and R. V. BAILEY, Power Removal from Boiling Nuclear Reactors, ORNL-CF-55-7-43, 1955.
7. J. M. STEIN and P. R. KASTEN, Boiling Reactors—A Preliminary Investigation, ORNL-1062, December, 1951.
8. R. J. RICKERT et al., Boiling Homogeneous Reactor for Power and U-233 Production, ORNL-CF-55-8-240, August, 1955.
9. P. C. ZIMOLA, Comments on Boiling Slurry Blankets for Homogeneous Reactors, ORNL-CF-55-9-125, September 27, 1955.
10. P. C. ZIMOLA, Boiling Blankets for TBR-Power Utilization, ORNL-CF-55-3-57, March 9, 1955.
11. R. B. BRIGGS, Cost Comparisons of Boiling Reactors with Circulating Solution Reactors, ORNL-CF-53-3-260, March 30, 1953. Homogeneous Boiler, ORNL-CF-53-10-23.
12. T. A. HUGHES, Babcock & Wilcox Co. Report 5435, February, 1958.
13. L. G. ALEXANDER and S. JAYE, A Parametric Study of Rate of Power Removal from Homogeneous Boiling Reactors, ORNL-CF-57-12-49, 1957.
14. A. J. SHOR, H. T. WARD, D. MILLER, and W. A. RODGER, Radioactive Carry-Over from BORAX-III and Test Systems. *Nuclear Science and Engineering*, April, 1957, p. 126.

CHAPTER 3

EXPERIMENTAL INVESTIGATIONS ON BOILING WATER REACTORS

INTRODUCTION

Reference was made in the two preceding chapters to some of the early experiments at Argonne National Laboratory on steam bubble formation and heat transfer rates by the use of electrically heated elements immersed in water.¹ These tests were necessary in order to determine whether heat transfer rates and steam release velocities were large enough to make boiling reactors feasible. As indicated, these experiments were extremely encouraging so far as these aspects of the problem were concerned. They showed that expulsion of water moderator by steam can be very rapid. Steam velocities of over 60 ft/sec were observed at atmospheric pressure, while velocities of about 10 ft per sec were observed at 260 psi. These first tests by the early workers at Argonne constituted the beginning of an experimental program on boiling reactors that included the BORAX tests, as well as many other experiments that extend to the present day. The first tests were at atmospheric pressure but these were followed by tests, first at 260 psi, and later at 600 psi.

The mechanism of boiling is complicated by many variable factors, such as density, friction, pressure, rate of heat transfer, and in nuclear reactors, by neutron activity. To evaluate the effect of these various factors both singly and in combination in natural circulation boiling, the experimental program was expanded and it is still being pursued, as noted above. This program has developed from the simple studies at atmospheric pressure made by Untermeyer and his associates in the pre-BORAX days to rather complex high-pressure boiling systems now being investigated. Table 3-8 lists, chronologically, the various apparatus and systems used. They will be described later in this chapter.

THE BORAX EXPERIMENTS

The BORAX Experiments have provided a considerable amount of useful information and data, and have contributed significantly to the Power Reactor Development Program of the AEC. In 1952, at the instigation of S. Untermeyer, serious plans were first made for determining, in an experimental reactor, the self-limiting characteristics of water-cooled reactors and the operating characteristics of boiling reactors.² The reac-

tor was first set up near the Experimental Breeder Reactor-I at the Reactor Testing Station in Idaho in the early summer of 1953. During the remainder of the summer and early fall, the reactor was put through a series of *runaway* excursions and periods of steady boiling operation, the latter covering operating pressures up to 130 psig. Quite obviously, it was not possible to determine all the characteristics of boiling in such a short time. Conclusive proof was, however, obtained that steam formation is a most effective, reliable, and rapid power-limiting process, capable of protecting properly designed reactors against reactivity excursions which produce reactor periods even shorter than 5 milliseconds. Steady boiling operation demonstrated that some of the applications of boiling operation were certainly feasible and that the prospect for a true boiler reactor was promising. The detailed results of these experiments are given in the following pages. This group of experiments has been given the code name BORAX, and the experimental reactor used has been known as the BORAX reactor.

BORAX-I

To date there have been four different BORAX experiments or reactors.³⁻⁵ The first, BORAX-I, built in 1952, was built to determine the stability of a reactor using boiling water both as a heat-transfer agent and moderator. It used aluminum-uranium alloy MTR type fuel elements.

As noted above, it demonstrated that a boiling reactor could operate stably but also that it had a high degree of inherent safety in its ability to shut itself down in a power excursion. Figure 3-1 is a vertical section of BORAX-I. The reactor was constructed outdoors and was operated remotely from a control station half a mile away. The reactor tank was contained in a larger shield tank of 10 ft diameter which was sunk part way into the ground and had earth piled up around it for additional shielding. Adjacent to the shield tank there was a pit with concrete walls in which was installed equipment for filling and emptying the reactor and shield tanks, and for preheating the water in the reactor tank. The reactor tank, 4 ft in diameter and about 13 ft high, contained the reactor core, which consisted of an adjustable number of plate-type fuel elements held at the bottom by a supporting grid and at the top by a removable cover grid.

In operation the tank was filled with water to a height of 3 or 4½ ft above the top of the core; this water constituted the reflector, moderator, and coolant. The shield tank was filled with water only when the reactor was shut down. The experiments were made with two different cores designated Reactors I and II, respectively, for the BORAX-I and II experiments (see Table 3-1).

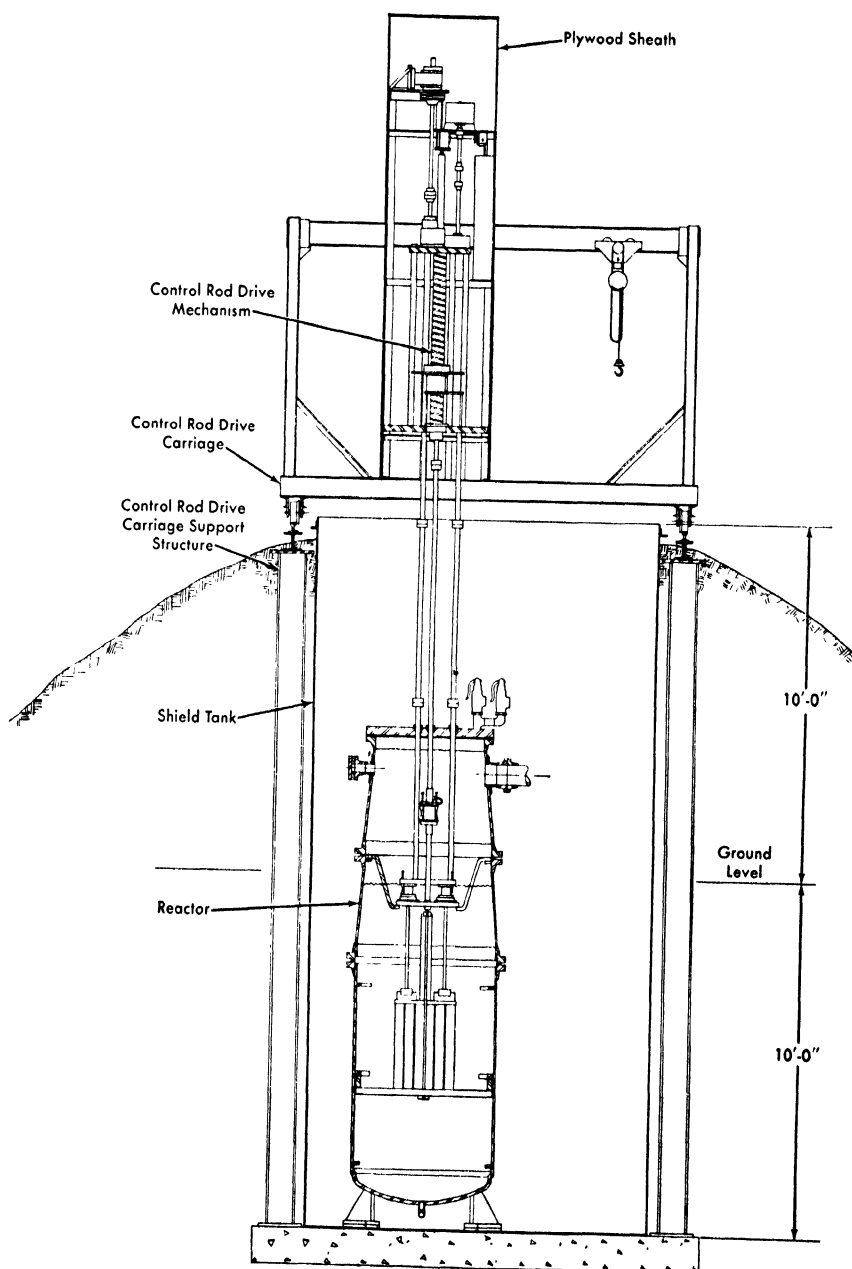


FIG. 3-1. Vertical section, BORAX-I.

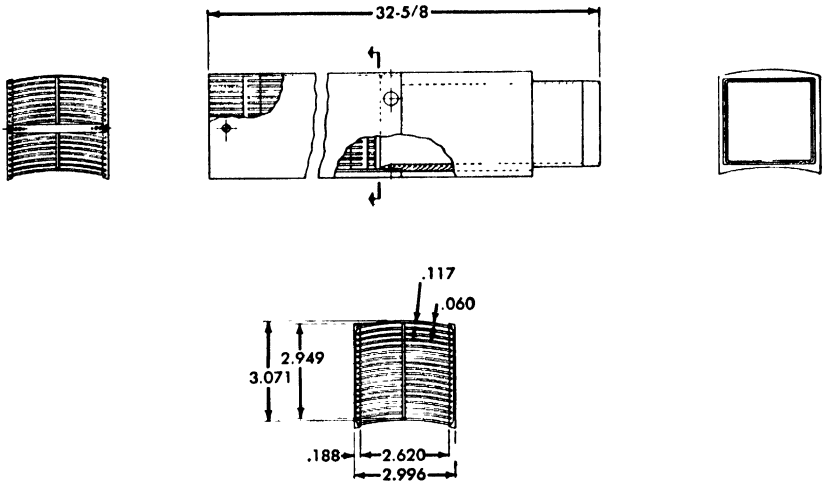
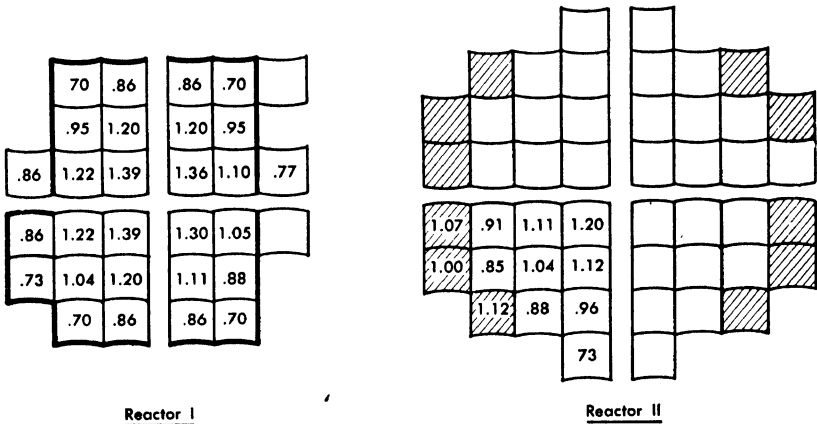


FIG. 3-2. Standard fuel assembly.



Reactor I

Reactor II

Each box represents a fuel element.

The shaded boxes in Reactor II contained fuel plates of higher fuel loading (see Table I).

The numbers in each box indicate the ratio of the total power produced in that element to the average power per element. The measurements on Reactor I were made with a slightly different loading than that diagrammed.

The heavy outline in Reactor I encloses the clean cold critical loading. Reactor II was critical at room temperature with a 6 by 8 element array of the more lightly loaded elements (Table I).

The axial maximum/average power ratio is 1.3 for both cores.

FIG. 3-3. Core loadings for Reactors I and II.

TABLE 3-1
COMPARISON OF BORAX-I AND -II

	Reactor I	Reactor II
Ratio, Al/H ₂ O	0.626	0.49
U ²³⁵ content per fuel element, g	138.6	93.4
		or 157.3
Number of fuel plates per element	18	10
Plate spacing (between center lines), in.	0.177	0.324
Measured reactivity loss with temperature increase, % k_{eff} .*		
80 to 200 F	0.82	0.45
80 to 280 F	1.93	0.76
80 to 420 F		1.57
Calculated loss of reactivity caused by replacement of 10% of core water by steam, at 200 F, % k_{eff}	2.4	1.0
Effective neutron lifetime, sec	6.5×10^{-5}	7.5×10^{-5}

* k_{eff} = effective reactivity.

The reactor contained five control rods which were operated by drive mechanisms located in the rectangular housing above the shield tank. The connection from the mechanism to the rods was through a spring-loaded magnetic coupling. These couplings could be released in unison or individually, allowing the rods to drop freely downward under the acceleration of the springs plus gravity. When released, the center control rod dropped out of the reactor core to apply the excess reactivity used for the experiments. The other four rods, when released, dropped into the reactor core to terminate the experiments. Each rod traversed the length of the core in about 0.2 sec.

The fuel elements were made of 18 curved aluminum-clad aluminum-uranium alloy plates, of 60 mils total thickness, fastened into aluminum side plates to make boxes roughly 3 inches square. Figure 3-2 is a drawing of the fuel element for one of the reactors. The other reactor (Reactor II) used elements of identical outside dimension but each element contained only ten of the fuel plates.

Each reactor was loaded, for any given experiment, with the number of fuel elements which would give a convenient amount of reactivity. Typical loadings for the two reactors are diagrammed in Fig. 3-3. Reactor II contained several elements of higher uranium content around its periphery to flatten the power distribution.

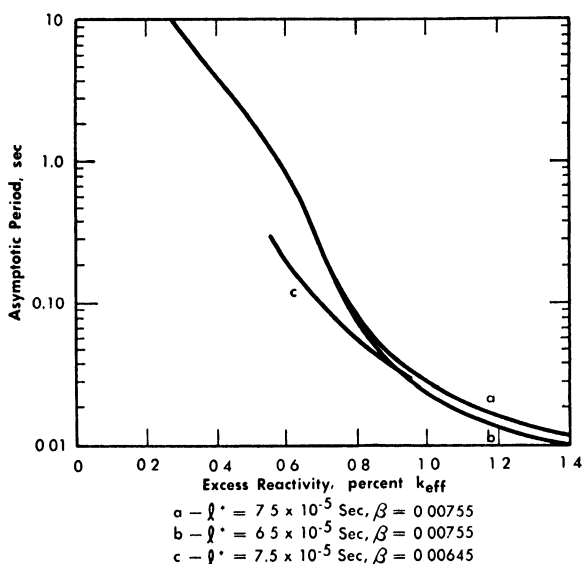


FIG. 3-4. Relation between excess reactivity and reactor period.

The relationship between asymptotic reactor period and excess reactivity for the reactors is given in Fig. 3-4 (curves *a* and *b*). For larger excess reactivities the period (r) is given by

$$r = \frac{1^*}{k_{ex}(1 - \beta) - \beta},$$

where β is the total delayed neutron fraction and 1^* the effective neutron lifetime. Other characteristics of the two reactors are summarized in Table 3-1.

The core of the reactor was divided into quadrants and the control rods were located in the openings so obtained. Each of the four blades (shim rods) was about 7 inches wide and consisted of a cadmium insert, nickel-plated for corrosion resistance in an aluminum supporting structure. The central control rod was a flat blade of aluminum, containing cadmium inserts in the form of 30-mil strips of adjustable width.

An antimony-beryllium neutron source was built into the connecting tube of one of the dummy plugs, at mid-core level. It remained in the reactor throughout operation.

Water flow under natural convection during boiling runs was vertically upward through the reactor core, returning downward through the reflector. During severe transients water was no doubt expelled both upwards

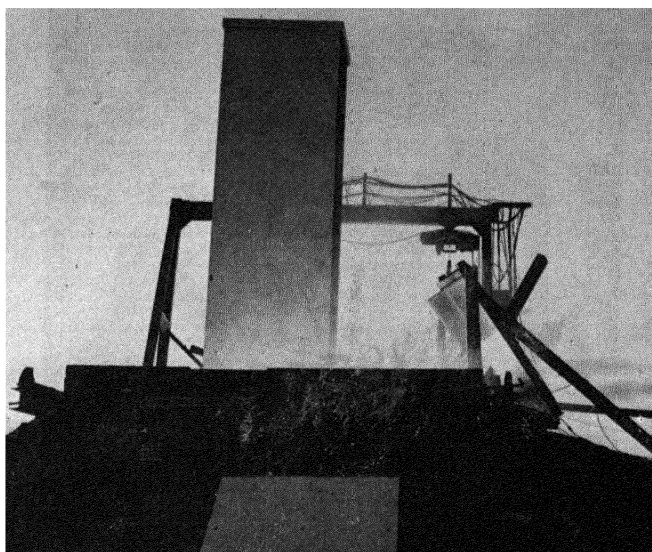


FIG. 3-5. Control rod drive structure. Note mirrors mounted for photographic records of transients.

and downwards from the core. The reactor could be electrically preheated to the desired operating temperature by an external heating system consisting of a circulating pump and immersion heater. Another pump was used to transfer water between the demineralized water storage tank and the reactor. A third pumping system was used to fill and empty the shielding tank.

The control rods were moved and positioned by an external drive mechanism, shown in Figs. 3-5, 3-6, and 3-7, with magnetic attachment to armatures on the control rod extensions. Magnet assemblies were attached to hollow shafts through which the magnet power lines were carried, and these shafts were directly coupled to the drive mechanism proper. Since the drive mechanism pierced the pressure vessel head through packing glands when the reactor was pressurized, it was necessary to effect magnetic disengagement inside the reactor tank. Spring loaded ejection plates were incorporated in the magnet assemblies to increase rod velocity during scram or initiation of transients.

The four blade (shim) rods were ganged to a common ball nut and lead screw drive. This was driven by a conventional gear-motor and speed reducer. The central rod was positioned by a rack and pinion drive, also driven by a conventional gear-motor and speed reducer. Since

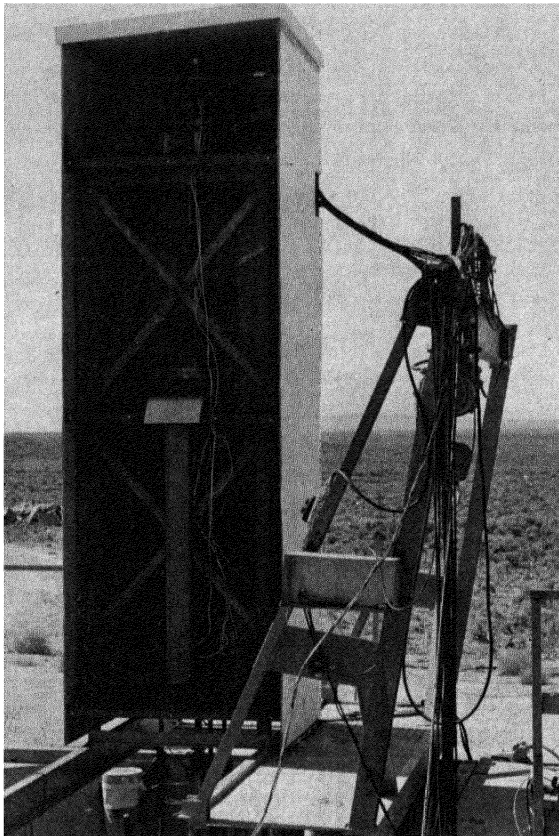


FIG. 3-6. Control rod drive assembly.

it was necessary to provide sufficient force on the descending magnets to compress the ejection springs and promote metal-to-metal contact between the magnet and control rod armatures for engagement, it was desirable to limit the torque by stalling the motors. Voltage control was therefore used to reduce the locked rotor torque to the desired value. Since a failure of the magnetic engagement of the central rod to its drive would instigate a reactor transient, a mechanical safety latch was built into the system which required manual setting, control-air pressure, and instrument power for release. By this method unplanned transients were avoided. A magnetic clutch was placed in the central control rod driving system, and the control rod externally counterweighted. De-energizing this clutch allowed the rod to be rapidly returned to its full upward

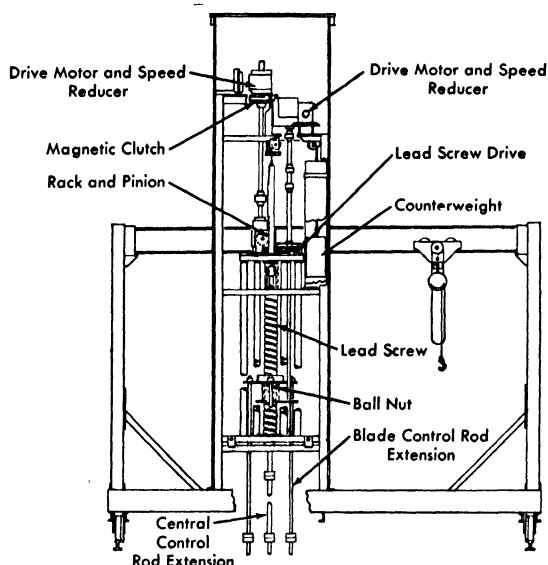


FIG. 3-7. Control rod drive assembly.

(black) position and was of value in rod calibrations, as well as serving as a secondary rapid shutdown.

During operations, all personnel were removed from the reactor area and the reactor was remotely controlled. The shielding requirements, therefore, were based upon long-lived decay radioactivity. Since the duration of the individual runs was of the order of an hour, the buildup of fission products was not excessive, and shielding requirements were satisfied by the use of sand and water.

The reactor was placed in a shield tank 20 ft deep, setting on a concrete slab 10 ft below ground level. Sand and gravel were piled up around the shield tank at about a 45 degree angle. The reactor core was located approximately three feet below ground level, and personnel at ground level were shielded by at least 11 ft of sand and gravel. The arrangement permitted access at all times immediately after termination of a run. At times when it was necessary for personnel to work on the reactor, the shield tank was filled with water, and subtolerance radiation levels were reached with a few hours decay.

The reactor control boards and nuclear indicators necessary for reactor control were located in a trailer just outside the EBR-I (Experimental Breeder Reactor) fence, and one-half mile from the reactor. Electrical switchgear and nuclear signal amplifiers were located in a trailer adjacent to the reactor. These two trailers were interconnected with half-mile

lengths of multi-conductor cables and coaxial cables as required. A third trailer located near the EBR fence contained a photographic dark room and auxiliary equipment. All electric motors except the control rod drives were powered through magnetic contactors in the reactor trailer with 110-v, 3-wire pushbutton controls on the control boards in the control trailer.

Control rod drives and reactor shutdown. Three-phase, 220-v motors were used to power the control rod drive mechanism. Total speed reduction was about 400 to 1 making it necessary to reduce the voltage to these motors so that the locked rotor torque was limited to a reasonable value. This was done by adjustable autotransformers in the power leads.

Magnet current was supplied by full wave bridge rectifiers with voltage control on the a-c side. The change in impedance of the magnet coil when the armature was engaged was utilized to indicate when the control rod was held by the magnet. Contact switches were used to signal, through pilot lights, when the control rod armature was in the dashpot. Selsyns were used to indicate the location of the control rod drive mechanism and, indirectly, through the indication of the control rod armature-magnet engagement, the position of the control rod in relation to the core.

During tests of the runaway type, transients were started by manually activating the release latch (to permit the rod to drop from the magnet) and then manually de-energizing the magnet of the control rod. At a preset time following the central rod ejection a timer automatically de-energized the shim rod magnets to inject the shim rods and terminate the experiment.

Fast reactor shutdown (scram) was effected by de-energizing relays in the magnet current power leads through push buttons located on the control board. Any or all of the four shims could be de-energized manually, with the scram signal de-energizing all four. Scram could be initiated by the transient timer, manually through push buttons located on the control boards in the reactor trailer and, when consistent with the experiment being performed, by the reactor period meter and the linear power meter. Fast reactor shutdown could also be effected by interrupting the shim rod magnet energizing current directly at a telephone jack connection located on the control board. A six-inch dump valve could be opened by means of push buttons on the control board. This valve dumped the water from the reactor into the pump pit constituting a slower but positive shutdown process.

Electric pressure transducers were used to telemeter the water levels in the reactor tank and shield tank to an indicator on the control board. Eight thermocouple circuits were available to transmit water tempera-

TABLE 3-2
PROGRAM OF FUEL ADDITIONS

Addition number	Grid space	Worth of addition, % k_{eff}	Total number assemblies	Core volume, liters
1	13	1.22	27	103
2	18	0.94	28	107
3	24	1.00	29	111
4	12	—	30	115
5	7 and 30	1.29	32	122

tures and some key machinery temperatures. The reactor was filled and emptied by a pump located in the pump pit and controlled from the control board.

The electrical pre-heating system operated by circulating the reactor water through a side loop over the immersion heaters of 150 kw rating.

During pressurized operations, the reactor pressure was telemetered to the control board, and recorded. A limit switch in this recorder also activated a quick valve opening in case of over-pressure.

Criticality. The reactor reached criticality when the core had been built to 26 fuel elements (3.60 kg of U^{235}). With this loading the reactor has 0.23 per cent excess reactivity at 94 F. As the program progressed and more excess reactivity was required, further fuel elements were added in the sequence indicated in Table 3-2. The worth of the added fuel elements is presented in the table for those cases in which it was measured.

The thermal neutron flux distribution was roughly mapped over the volume of the core with gold foils at a low power level (a few watts) and with cobalt foils during steady boiling operation at a power of about 600 kw.

Typical transient behavior. The objective of these experiments was to determine the extent to which the reactor, by the formation of sufficient steam to render the reactor subcritical, would protect itself against the effects of sudden large, externally caused, reactivity increases. The central control rod had a length double that of the core and contained neutron absorbing material (cadmium) in the bottom half of its length. If the reactor was critical with the bottom (absorbing) part of the rod partially inserted in the core, ejection of the rod downward under the force of gravity plus the force of the ejection springs would impose upon the reac-

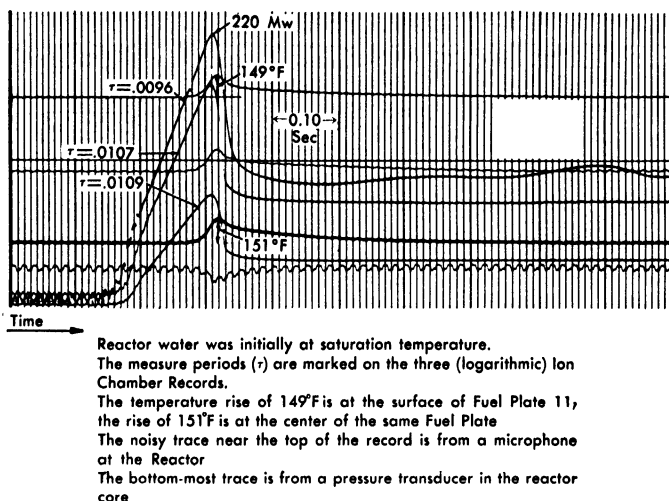


FIG. 3-8. Typical record of power excursion.

tor a sudden reactivity increase equal to the reactivity worth of that portion of the rod which was initially in the core.

The initial power was sufficiently low and the speed of the rod ejection was sufficiently high so that in almost all cases the rod was completely out of the core and the reactor period reached its stable value before the reactor power had risen high enough to produce significant thermal effects. The power was allowed to continue to rise until the formation of steam in the reactor core reduced the reactivity below criticality and caused the power to fall to a low value. After it was evident that the power had been safely limited by the formation of steam, the remaining four control rods were dropped into the reactor to terminate the experiment, referred to as a power excursion. By proper adjustment of the number of fuel elements in the reactor core and of the positions of the four outer control rods, the reactor could be made critical with the center control rod inserted to any desired degree in the core. The magnitude of excess reactivity applied by ejection of the center rod could thus be adjusted at will.

Figure 3-8 is a reproduction of a typical chart from the multichannel magnetic oscillograph which recorded the data on the experiments. In this case the applied excess reactivity was 1.4 per cent k_{eff} and Reactor I was used. The neutron flux (proportional to reactor power) was recorded over about three decades by three different neutron-sensitive ion chambers working through logarithmic amplifiers. The stable reactor period (r) was indicated by the three ion chamber records as 0.0096, 0.0107, and 0.0109 sec, respectively. The temperature of one of the fuel plates, which

was situated at roughly the highest flux position in the core, was recorded by two fast-response thermocouples. Both of the couples were located near the position of maximum neutron flux; one was installed on the surface of the plate, the other at its central plane. There is little difference between the two temperatures, because of the high thermal conductivity of the thin plate.

The ion chambers, which were calibrated in terms of absolute power by thermal methods, indicate that the reactor power reached a maximum value of 220 Mw before the formation of steam checked the rise. Further generation of steam reduced the reactivity below the critical value and caused the power to decrease very rapidly to a value of about 0.2 Mw.

Once the initial power excursion has been checked by boiling in the reactor, the specific power variation depends both quantitatively and qualitatively upon the amount of excess reactivity to which the reactor was initially subjected and upon the bulk temperature of the reactor water. Figure 3-9 contains tracings from excursion records for Reactor I similar to those of Fig. 3-8, but of longer duration. They summarize the typical behavior of the reactor for various amounts of applied excess reactivity when the reactor water is initially at saturation temperature. When the excess reactivity applied is low, corresponding to a reactor period of about 0.03 sec or longer, the reactor power after the initial surge settles down to a relatively steady value in the neighborhood of half a megawatt (top curve Fig. 3-9). For this type of excursion the self-regulating characteristic of the reactor operates rapidly enough to stabilize

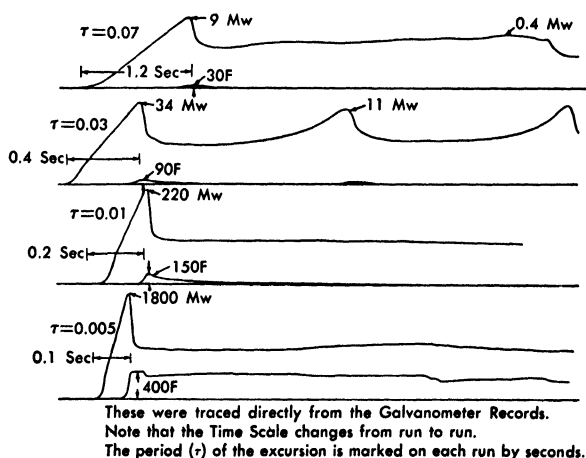


FIG. 3-9. Representative records of excursions at saturated temperature with various excess reactivities.

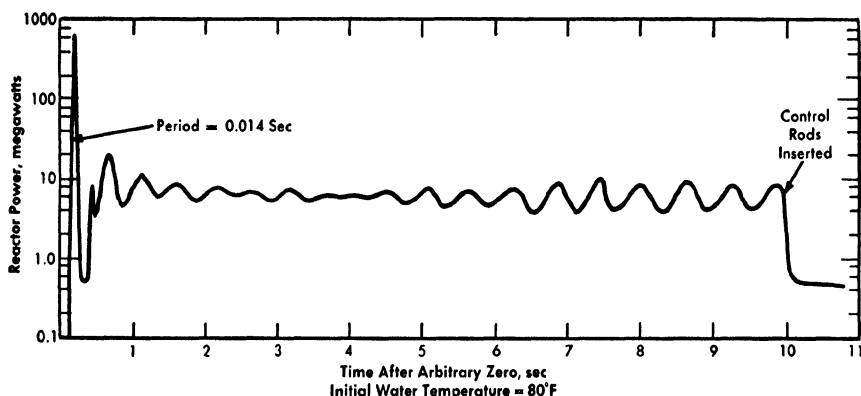


FIG. 3-10. Reactor power variation during 10-sec run following initial excursion of 14-msec period.

the power at a steady value characteristic of the amount of applied excess reactivity. After the power has reached this steady state, further reactivity could, of course, be applied, and the reactor would continue to operate stably in steady boiling at a higher power.

If the excess reactivity which is applied by ejection of the control rod exceeds that corresponding to a period of 0.02 or 0.03 sec, the initial power excursion is followed by a series of qualitatively similar excursions of smaller amplitude which occur at intervals of about 1 sec (second curve Fig. 3-9). The amplitudes of the successive excursions, although they vary in an irregular manner, have no sustained tendency to increase or decrease. This type of operation is referred to as *chugging*.

When the applied excess reactivity was greater than that corresponding to about a 0.01-sec period, the chugging was no longer observed, and the power after the first surge remained at a low value. This permanent shutdown was no doubt the result of expulsion of sufficient water from the reactor tank to partially uncover the reactor core. The occurrence of this behavior in other reactors would, of course, depend upon the specific design of the reactor in question.

When the applied excess reactivity was increased to about 2 per cent k_{eff} to give periods in the 0.005-sec range, the qualitative behavior of the reactor power remained the same, but the fuel plate temperature did not drop immediately after the power surge (bottom curve, Fig. 3-9). The fuel plate temperature remained high for almost a second after the power surge and then decreased by small jumps, as though the plate had been blanketed by steam for some time after the power excursion.

Experiments of this type were not carried to periods shorter than about 0.013 sec when the reactor water was cooler than the saturation tempera-

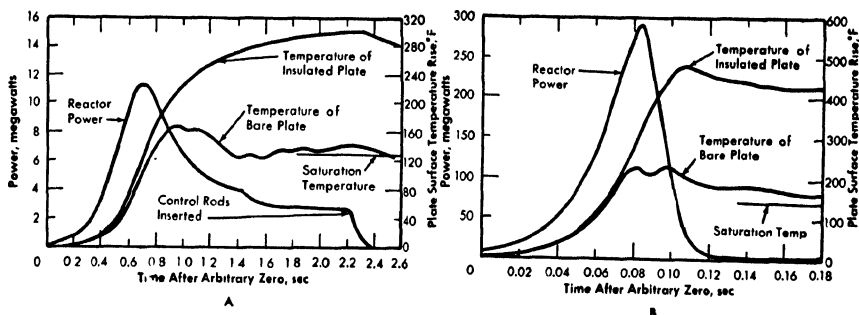


FIG. 3-11. Power and fuel plate surface temperature rise during power excursion.

ture. With this condition, which will hereafter be called the subcooled condition, chugging was never experienced. In all cases after the initial power excursion the reactor power stabilized at some more or less steady value. Figure 3-10 is a record of such an experiment on Reactor I, in which the initial period was 0.014 sec. Although the reactor power oscillated after the excursion, the oscillation amplitude was very much less than that for typical chugging operation.

Some of the details of the nuclear and thermal behavior of the reactor during a power excursion are illustrated by Figs. 3-11(a) and 3-11(b), which apply to the subcooled Reactor I for excursions of two different periods. The curves show the time variation of reactor power, on a linear scale, and the fuel plate surface temperature. In these experiments the fuel plate, with thermocouple attached, was coated with a thermally insulating plastic over a section of its length. The temperature of this section of the plate is also included in the figures. This temperature, except for the effect of a small heat loss through the thermal insulation, is proportional to the total nuclear energy liberated in the plate. The point at which there is a sharp deviation between the temperature of the bare section and that of the insulated section evidently marks the beginning of rapid steam formation at the bare plate surface. Up until this time the temperature of the bare plate, like that of the insulated plate, is roughly proportional to the time integral of the reactor power.

Since no single fuel plate can produce sufficient steam to shut the reactor down, it is evident that the time relationship between peak reactor power and peak temperature of the bare fuel plate, as well as the ratio of maximum bare plate temperature to maximum insulated plate temperature, will depend upon the local value of power density at the plate location relative to the power density elsewhere in the reactor. The plate used for these measurements was located at or near the maximum power density, and the thermocouples were installed near the point of

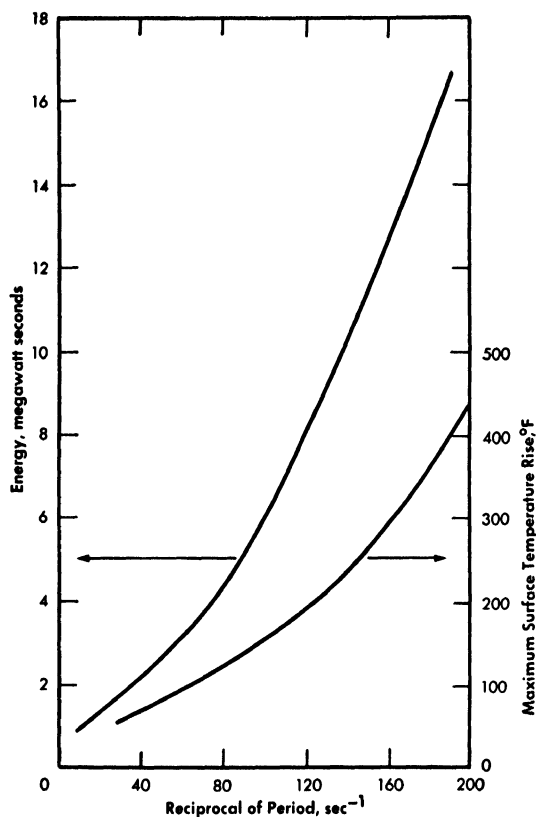


FIG. 3-12. Maximum temperature rise of fuel plate surface and total energy release during power excursion of various exponential periods. Reactor I at saturation temperature and atmospheric pressure.

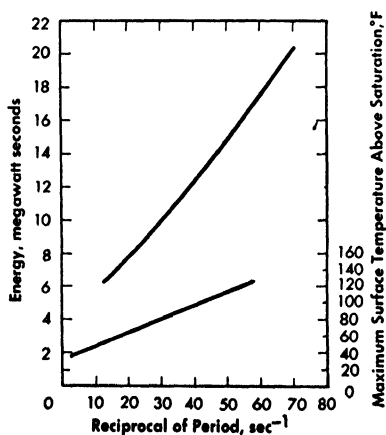


FIG. 3-13. Maximum temperature of fuel plate surface and total energy release during power excursions of various exponential periods. Reactor I at room temperature (approx. 80 F) and atmospheric pressure.

maximum power density in the plate. Nevertheless, Fig. 3-11(a) indicates that for long period excursions other plates were responsible for the first formation of steam, since power began to decrease before plate temperature reached saturation.

Effects of several variables on power, energy, and fuel plate temperature.

Both the total nuclear energy liberation and the maximum fuel plate temperature reached during a power excursion depend upon the amount of excess reactivity involved in the excursion. In the following presentation, the reciprocal of the stable reactor period is used to characterize the excess reactivity. The relation between the two is given in Fig. 3-4. In Fig. 3-12 the total energy* liberated by the power excursion and the maximum fuel plate surface temperature are plotted as functions of the reciprocal period for the case in which the reactor water was at saturation temperature before the excursion began. The shapes of the two curves are quite similar; in fact, the maximum fuel plate temperature rise is roughly proportional to the energy of the excursion for all periods shorter than about 0.03 sec, and the peak temperature corresponds to the temporary storage in the fuel plate of 60 to 70 per cent of the total energy of the excursion.

The energy liberation and maximum fuel plate temperature for the condition in which the reactor was initially at room temperature are given in Fig. 3-13 for Reactor I. The plotted temperature is the maximum above saturation temperature at atmospheric pressure rather than the total temperature rise. The fuel plate temperature is somewhat higher (and the energy release is much higher) for a given reactor period than in the case of saturated reactor water.

For the cases of both saturation temperature and room temperature water the total energy liberated during the power excursion was nearly proportional to the product of the maximum power and the period. These quantities are plotted in Fig. 3-14 for the experiments made with Reactor I. The slope of the line is about two.

In comparing the behavior of the two reactors, I and II, which differed in core size, it is more informative to compare energy density or energy per fuel plate than to compare total energy. In Fig. 3-15 the energy release per fuel plate is compared, as a function of reciprocal period, for

* Throughout the section on solid fuel reactors the term energy is used to refer to the prompt fission energy which is converted to heat in the fuel plates. The total energy release, including delayed emission and energy converted to heat directly in the reactor water, is about 15 per cent higher. A consistent definition is used for reactor power. By the total energy of a power excursion is meant the energy liberation up to the time of the first minimum in reactor power.

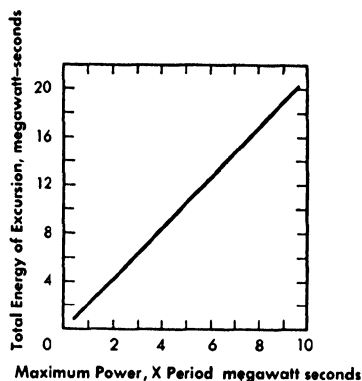


FIG. 3-14. Relation of total energy, maximum power, and period. Reactor I.

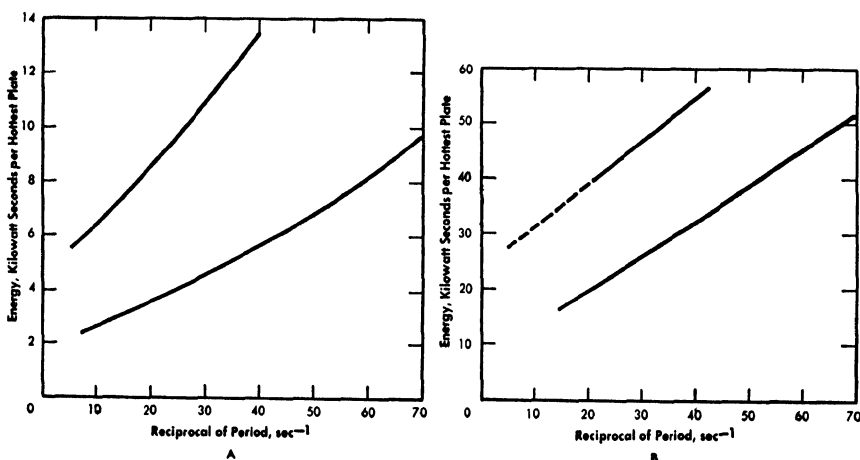


FIG. 3-15. Energy release of hottest fuel plate during power excursions in Reactors I and II. (a) Reactor water at saturation temperature and atmospheric pressure; (b) Reactor water at room temperature and atmospheric pressure.

the two reactors: Fig. 3-15(a) is for the case of saturated reactor water, and Fig. 3-15(b) for the case of room temperature water. The dashed portion of Fig. 3-15(b) is an upper limit only. The plotted energy release is that of the fuel plate in the position of highest neutron flux. In comparing the behavior of the two reactors, reference should be made to Table 3-1. Note that the volume of water associated with each fuel plate is about twice as great for Reactor II as for Reactor I. If the energy release per unit volume of water is compared, the values for Reactor II are only slightly higher than those for Reactor I, despite the fact that the

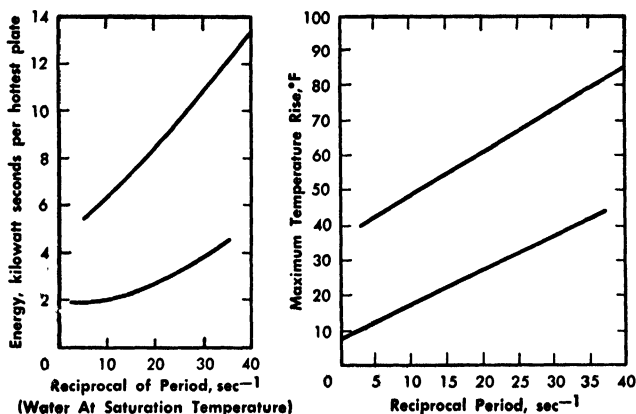


FIG. 3-16. (a) Energy release of hottest fuel plate during power excursions at different pressures; (b) Maximum fuel plate temperature rise during power excursions; comparison of unpressurized and pressurized cases.

steam coefficient of reactivity is more than twice as great for Reactor I as for Reactor II.

In Reactor II the investigation of power transients was extended to reactor pressures as high as 300 psi. The pressurization of the reactor was by the vapor from the reactor water. Consequently, only the saturated condition could be investigated. The power excursions were run with the reactor tank completely closed. In no case did the pressure in the steam space above the reactor water rise by more than about 5 psi as a result of an excursion. The effect of pressurization is to decrease both the energy released in an excursion of given period and the maximum temperature rise of the fuel plates (Fig. 3-16). This was possibly due to the greater change in density of the liquid with temperature at the higher pressure.

Pressures generated during excursions. The energy by which water is ejected from the core to terminate the short period power excursions can only be obtained from an accelerating pressure built up within the reactor core. This pressure is, of course, produced by the generation of steam at the fuel plates. Provision was made for measuring the pressure in the core by means of electrical strain gage pressure transducers. Reliable pressure records were obtained, however, on only a few excursions of relatively long period. Splashing of water (see Fig. 3-17), which occurred with the shorter period excursions, resulted in wetting of the transducer wiring at unanticipated points, and perhaps did some mechanical

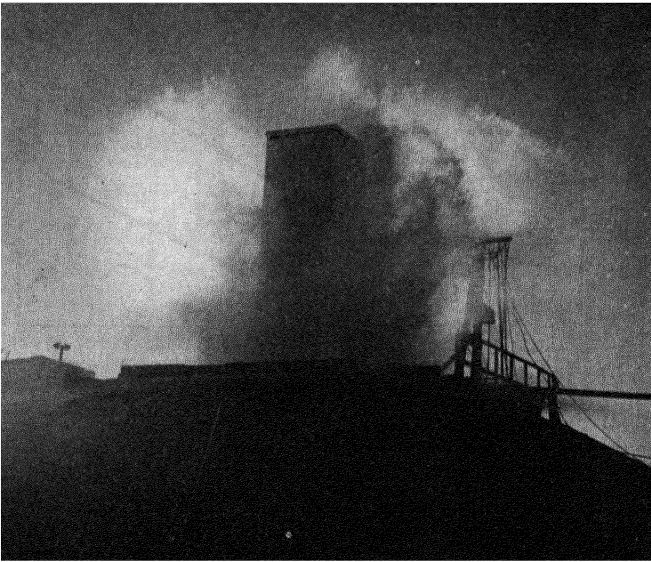


FIG. 3-17. Water expulsion during 5-msec excursion.

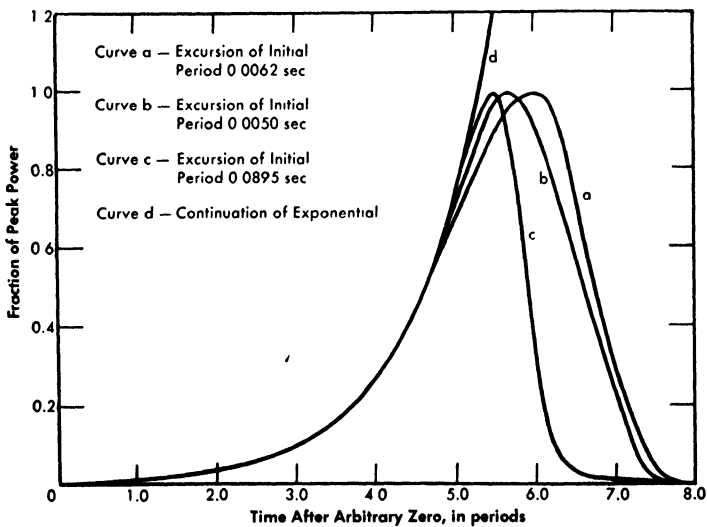


FIG. 3-18. Record of power excursion showing pressure variation. The excursion was made with reactor temperature initially at saturation, with 0.058-sec period. The two pressure traces are from transducers having different sensitivities. The upper temperature trace is from the surface of fuel plate I; the lower trace from the center of plate II.

damage to the transducers themselves, which led to inconsistent results. The limited time available for the experiments did not permit the correction of these difficulties. Figure 3-18 is a record of a power excursion including pressure records from two transducers of different sensitivities. The shape of the pressure curve is consistent with the interpretations previously put upon the ion chamber and thermocouple records; the pressure begins to rise at about the time the reactor power breaks sharply away from the exponential and reaches its maximum value when the power is dropping off sharply.

The pressure records which were obtained at shorter periods recorded a maximum pressure of about 70 psig for the excursion of 0.005-sec period. Within the wide scatter of the records it appears that pressure might have increased linearly with reciprocal period. It is believed, however, that these recorded pressures are considerably too low. In fact, photographic records indicate that virtually all of the water was expelled from the core during the shorter period excursions. Since the ion chamber records indicate that expulsion was completed in, at most, about 4 reactor periods (see Fig. 3-19) it is difficult to see how such rapid water expulsion could be achieved without considerably higher pressures.

The pressures generated during the fast excursions did some mechanical damage to the reactor core. During the sequence of excursions with periods between about 0.01 and 0.005 sec, the outermost plates in many of the fuel assemblies (Fig. 3-2), which are normally concave outward, were permanently reversed in curvature, giving an assembly which was convex on both sides. After these experiments the core was disassembled, and the bent plates were pounded roughly back into shape, but they were

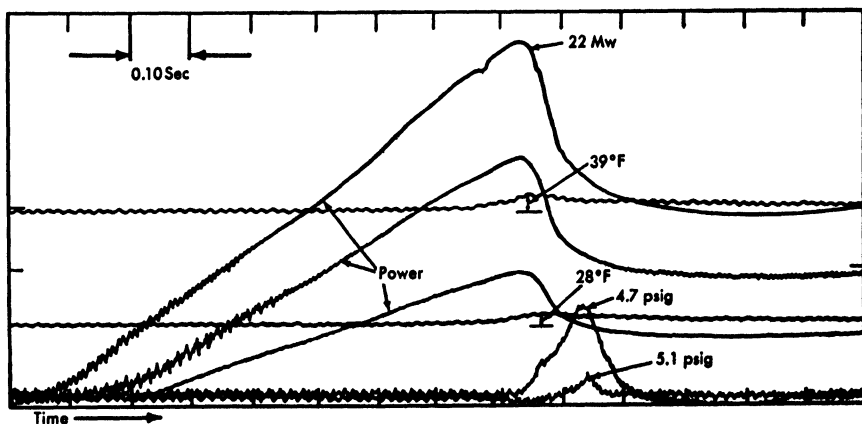


FIG. 3-19. Comparison of three power excursions of different initial periods.

no longer very strong. The subcooled excursion tests were made after this repair of the initial damage. As the subcooled tests with 0.013 sec period progressed, it was evident from slight permanent changes in reactivity that the deformation was again occurring. This deformation proceeded to the point where the innermost fuel assemblies began to be pushed into the control rod channels. At this point the excursion tests were discontinued, the fuel assemblies were again straightened, and the investigation of steady boiling was begun. It was because of this deformation that the subcooled excursions were not run at shorter periods. There is no reason to believe that the subcooled experiments would have deformed the assemblies if they had not been already weakened by the previous bending, since visual and photographic observations showed that the water agitation associated with the subcooled experiments was far less than that which had occurred previously with shorter period excursions at saturation temperature. It should be emphasized that this damage to the reactor core was purely a function of its particular mechanical design and was caused by pressure alone. The necessary differential pressure across the end fuel plate for reversal of its curvature at room temperature has been shown to be between 18 and 25 psi. The total pressures generated in the core during the transient runs were at least several times this value.

Figures 3-20 through 3-23 are enlargements of individual frames of 16 mm motion picture records which were taken of some of the excursions. For the less violent excursions (Figs. 3-22 and 3-23) the camera was located just outside the earth shield and looked down upon the open top of the reactor, where control rod magnets and the surface of the reactor water were visible. The core itself was not visible. Figure 3-5 is a photograph of the reactor showing the location of the mirrors for these movies. For the more violent excursions (Figs. 3-17, 3-20, and 3-21) the motion picture camera was set up at some distance from the reactor to record the expulsion of water and steam. The approximate times indicated on the photographs are measured from the time of release of the central control rod. These excursions were made with the reactor water initially at saturation temperature and are, in fact, two of the excursions whose power records are reproduced in Fig. 3-19. It is interesting to note that in the frames of Fig. 3-20 marked 840 msec and 1180 msec, the pressure inside the control rod housing has lifted the corner of the top of the house an observable amount.

Figure 3-22 depicts a subcooled excursion with a period of 20.2 msec. The flash of light produced by the reactor as it went through its power surge can be seen in the photograph in the upper right hand corner of the page. Most of the agitation of the water surface occurs after this light flash disappears, i.e., after the reactor power has dropped back to a rela-

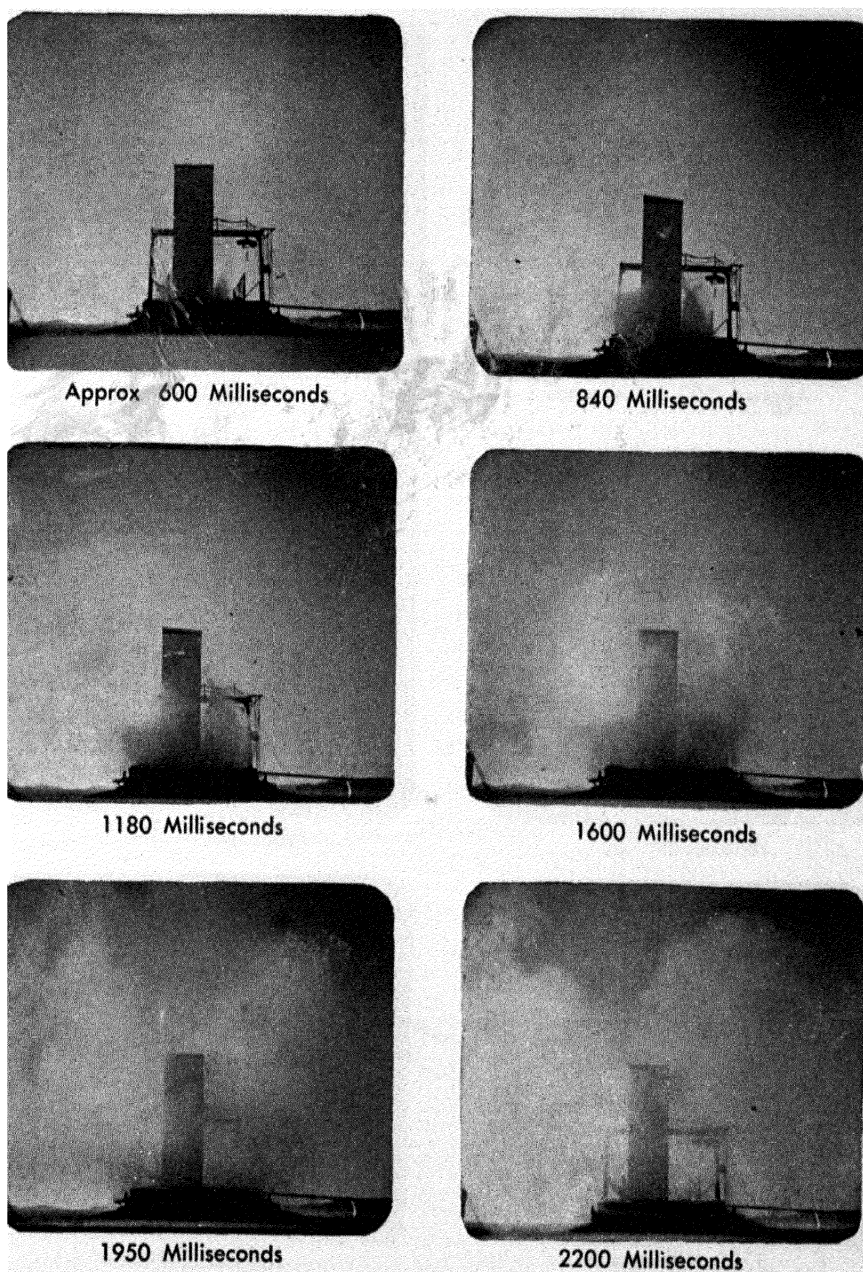


FIG. 3-20. Motion picture records of power excursions. Period, 6.1 msec.

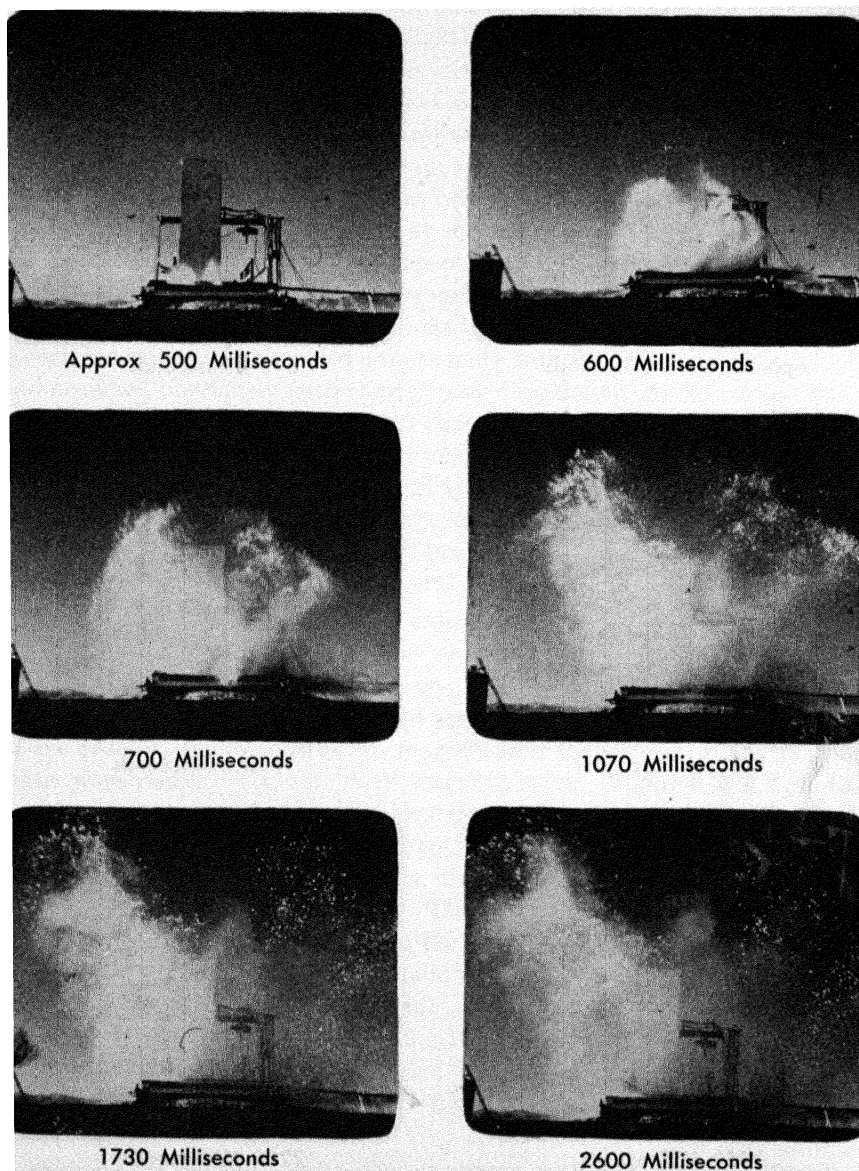


FIG. 3-21. Motion picture records of power excursion. Period, 5 msec.

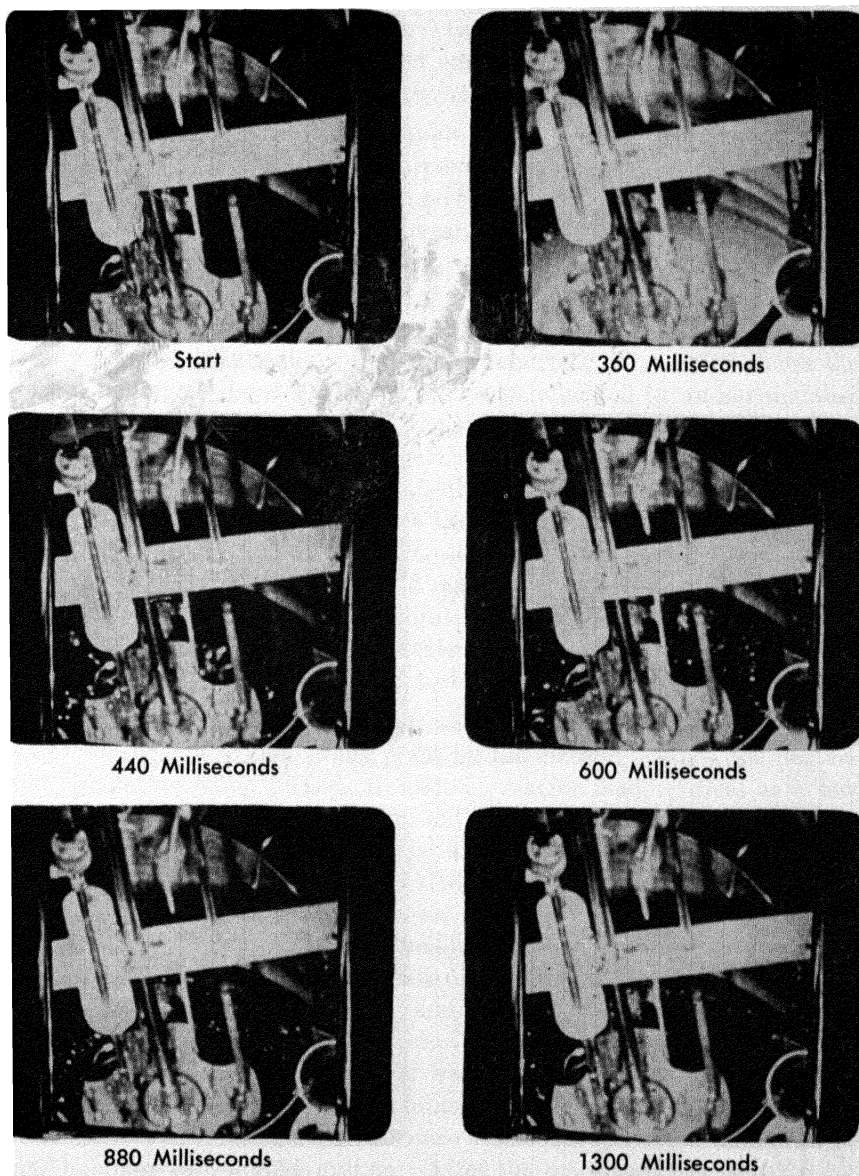


FIG. 3-22. Motion picture records of power excursion. Period, 20.2 msec.

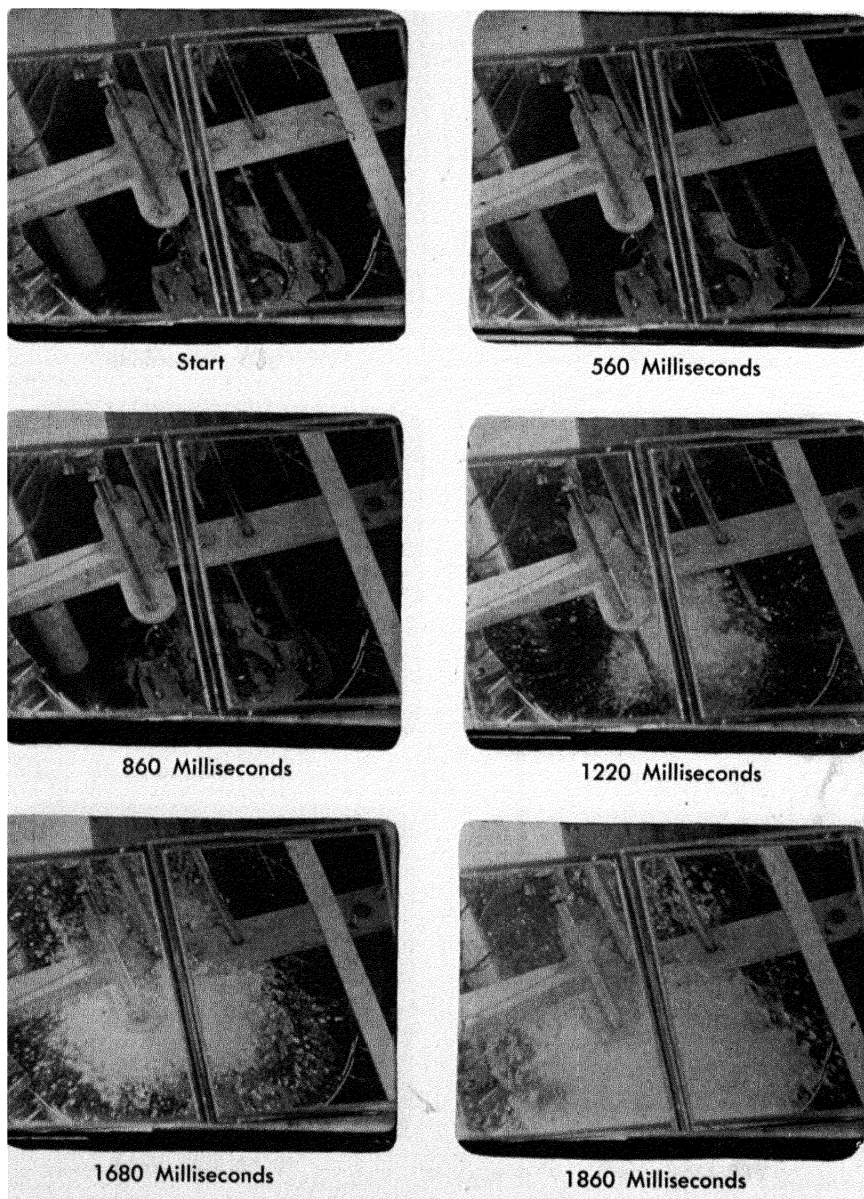


FIG. 3-23. Motion picture records of power excursion. Period, 23 msec.

tively low value. This is no doubt a result of the fact that much of the total steam formed during the excursion is produced by heat stored in the fuel plates after reactor power has decreased. A further striking characteristic is the relatively mild effect of these excursions insofar as agitation of the water surface is concerned.

Figure 3-23 was made with a double mirror rather than the single one used in Fig. 3-22. Figure 3-23 represents an excursion of about the same period as that of Fig. 3-22 but with no subcooling. Agitation of the water surface is considerably more violent, and a little water is spilled over the edge of the tank.

Steady boiling runs. The reactor was operated as a boiling reactor, at steady power, with the lid of the reactor removed. Figure 3-5 shows the reactor boiling at about 700 kw. Power was indicated by an ion chamber in a thimble on the inside wall of the shield tank, and could be recorded simultaneously on a Brown Electronik recorder and a fast Brush recorder. The chamber was calibrated for absolute power by heating the reactor under its own power. Inasmuch as these calibrations were made with a nonboiling reactor, and the neutron leakage to the ion chamber is a function of the water density in the reactor core and, particularly, of the density in the reflector, these calibrations contained some uncertainties. Theoretical corrections for the effects of steam voids in the core were estimated, but an uncertainty of 10 to 15 per cent exists in the calibration.

The usual history of a boiling run was roughly as follows. The reactor water was heated to the boiling point by the electric heaters, the heaters and circulating pump turned off, and the reactor made critical at a low power by withdrawal of the control rods. A control rod was then further withdrawn to put the reactor on a period between 5 and 30 seconds; then power was allowed to rise on this period until the boiling began and steam in the reactor compensated the excess reactivity, causing power to level off of its own accord to some steady value. The power was then increased in steps by incremental withdrawals of the control rod. The reactivity added to the reactor by control rod motion after initial criticality is called the reactivity compensated by steam. When this quantity was relatively low, the power output of the reactor was subject to fluctuations of plus or minus 10 to 20 per cent. These fluctuations did not occur with any noticeable regularity, but had the appearance that one might expect from irregular bubbling in a boiling system. After the reactivity compensated by steam was increased above 1.5 or 1.6 per cent k_{eff} , the variations began to appear as oscillations of definite frequency, the frequency ranging from about 1.2 to 1.8 cycles/sec. These oscillations occurred in pulses containing several complete oscillation periods. The pulses were separated from

one another by periods of irregular variation similar to those seen at the lower reactivities. As the reactivity compensated by steam was increased further, the amplitude of these oscillations increased, the number of oscillations per pulse increased, and the interval between pulses decreased, until finally the oscillation was practically continuous, but with varying amplitude.

In a run after a last increase in reactivity which increased the reactivity compensated by steam to 2.6 per cent, the reactor operated for about half a minute with power oscillations of amplitude which varied from almost zero to a maximum. It then went through a series of oscillations of rapidly increasing amplitude which in the space of a few oscillations took on the characteristics of chugging operation of the type previously described in connection with power excursions. If the ion chamber calibration can be assumed to remain constant during this operation, the average power of the reactor increased by a factor of 2 to 3 over that just prior to the onset of chugging. Maximum powers of 50 to 60 megawatts were reached. It is possible, however, that the water agitation resulting from the chugging changed the average leakage of neutrons from the reactor to the ion chamber and that the ion chamber therefore did not measure actual power. The reactor was shut down by the insertion of the shim rods and the chugging operation caused no damage to the reactor. Actually, more violent chugging operation had been experienced for periods up to 20 seconds in previous runs of the excursion type. The maximum period attained by the chug which reached the highest power was 0.017 sec. This represents an excess reactivity of about 1.2 per cent k_{eff} , only about half the total (2.6%) which had been added and was previously compensated by steam.

Boiling operation at higher pressures. Operation at higher pressures demonstrates some additional characteristics of the boiling reactor beyond those observed at atmospheric pressure. These characteristics have to do primarily with effects of the various system variables upon reactor operation. They become apparent under pressurized operation because, in the first place, a pressure regulator must be added to the system and, in the second place, the heat capacity of the reactor and its water represents a more important heat reservoir as a result of the higher operating temperature. The behavior of the reactor can best be understood if it is remembered that, roughly speaking, the reactor will regulate itself to produce steam at a constant rate so long as the positions of control rods are not changed. Since all the heat produced by the reactor need not necessarily go into the vaporization of water, the reactor will not necessarily regulate itself to a constant power output. Thus, for example, if cooled

makeup water is injected during operation, heat will be required to raise the temperature of the water to the saturation point, and reactor power may be expected to increase if control rod positions are held constant. Again, if during reactor operation the pressure control valve allows reactor pressure to increase, extra energy will be produced by the reactor to heat the reactor water to the correspondingly higher saturation temperature. Variations of these kinds were observed during reactor operation. The pressure regulating valve was used with a wide proportional band, and the pressure maintained by the valve was, therefore, a slowly varying function of steam flow. Cold makeup water was injected, as required, to replace that boiled off. The injection was at a relatively fast rate, higher than the rate of evaporation attained in any of the runs; during any given run, therefore, water was injected either too rapidly or not at all, and neither process could be continued for long without running the reactor water level either too high or too low. Since the heat capacity of the reactor system was high, a relatively long time was required to attain thermal equilibrium. These circumstances, plus the desire to explore the effects of all system variables and to find any that might conceivably lead to instability, resulted in few runs at equilibrium conditions.

As in the case of boiling at atmospheric pressure, in the pressure runs the fluctuations were of two kinds: an irregular fluctuation of small amplitude when reactivity compensated by steam was less than about 2 per cent k_{eff} , and an oscillation of definite frequency and higher amplitude for reactivities above about 2 per cent. In pressurized operation, the amplitude of the irregular fluctuation was quite small, amounting to less than plus or minus 5 per cent in all cases, and usually about plus or minus 2 to 3 per cent. The transition from the irregular to the oscillatory variation usually occurred at a somewhat higher reactivity than the atmospheric pressure (at about 2% Δk_s as compared with about 1.6% Δk_s at atmospheric), and the oscillations when they did occur were smaller and more regular in shape than those at atmospheric pressure. Figure 3-24 is a record of a complete boiling run on which has been plotted the amplitude (ratio of maximum to minimum power) of the fast fluctuation. Note that the oscillations themselves are not plotted. This plot illustrates the way in which the oscillations build up and die out even when other operating conditions of the reactor are held constant.

It will be obvious from the data presented here that these first experiments with a boiling reactor were indeed promising. They proved that operation was steady up to a steam content corresponding to at least 2 per cent k_{eff} . At higher reactivities power irregularities occurred, but the resulting oscillatory boiling represented a stable regime which is maintained up to still higher reactivities. It was realized, however, that it

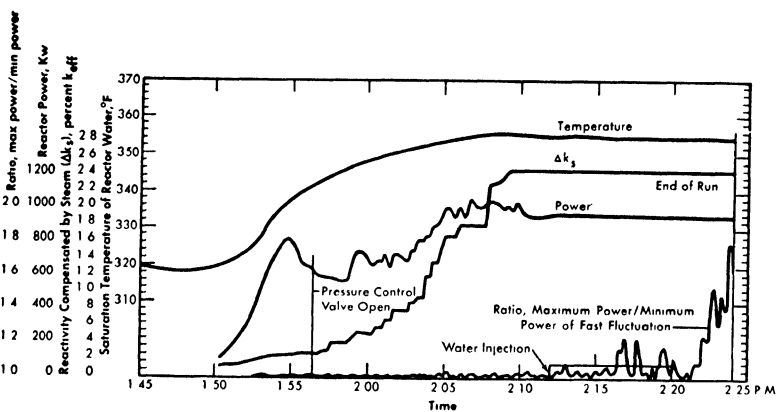


FIG. 3-24. Record of pressurized boiling run with 32 fuel assemblies in core.

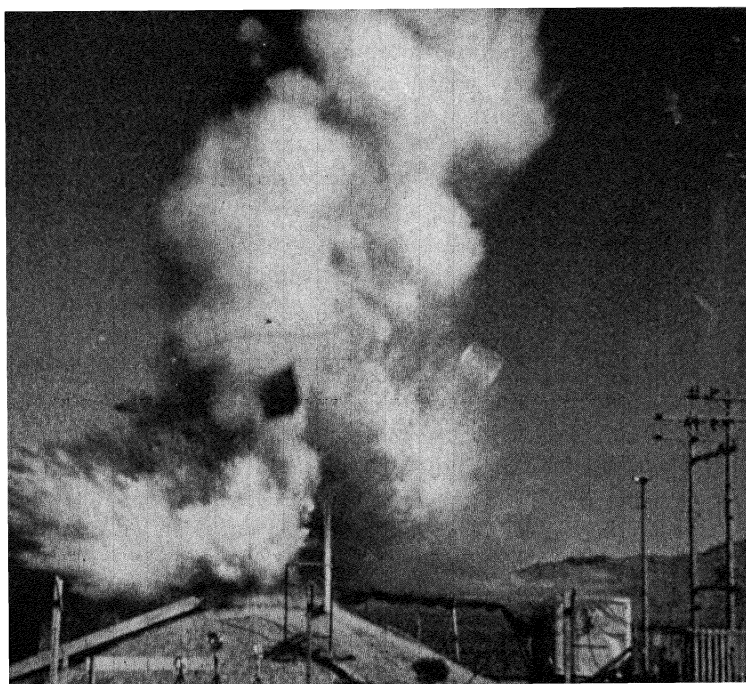


FIG. 3-25. Reactor destroyed by melting of fuel plates after being made supercritical by 3.3% k_{eff} .

could hardly be expected that a boiling reactor of the simple type involved could be operated at much higher powers, and plans were made to extend the experiments with more elaborate equipment and with a larger core. Most important, however, was the fact that the reactor had proved to be self-regulating.

The destructive experiment. In the short period experiments with Reactor I at atmospheric pressure, the steam pressure which built up in forcing the water rapidly from the reactor resulted in permanent deformation of the fuel plates. Because of this effect it was not possible to extend the experiments to periods shorter than about 0.005 sec without damaging the reactor to the point where it became unusable. Despite this mechanical damage the maximum temperatures reached by the fuel plates did not approach the melting temperature. It was decided that the reactor, which by this time had fulfilled its other purposes, should be sacrificed in an experiment which was violent enough to melt the fuel plates. For this purpose a control rod worth 4 per cent k_{eff} was completely ejected from the reactor core. To increase the severity of the experiment it was run with the reactor water at room temperature. Although the ejection of the rod required only about 0.2 sec, the rod was only about 80 per cent out of the core when the reactor power reached its peak value. The minimum period resulting from the ejection was 0.0026 sec.

The power excursion melted most of the fuel plates. The pressure resulting from the molten metal in contact with the reactor water burst the reactor tank and ejected most of the contents of the shield tank into the air (Fig. 3-25). The sound of the explosion at the control station, half a mile away, was comparable to that resulting from the explosion of 1 to 2 pounds of 40 per cent dynamite on the bare ground at the same distance.

The total energy release during the excursion as determined by calibrated cobalt foils in the reactor core was 135 Mw-sec. Other data taken during the excursion are less reliable because of the violent mechanical effects of the explosion. Figure 3-26 is a reproduction of the transient record. The absolute values on the power curve (*A*) may be in error by 30 or 40 per cent, and the shape of the decreasing portion may not be correct. The fuel-plate thermocouple (curve *B*) failed long before peak power was reached. Curve *C* is a temperature record from an insulated fuel plate installed some distance outside the reactor core. It was used for an auxiliary determination of total energy and does not give good transient information, as it was connected to a relatively slow recorder. A pressure transducer, which was installed in the reactor tank adjacent to the reactor core, failed before it recorded a pressure of significant magnitude (curve *D*). Analysis of the mechanical damage of the transducer,

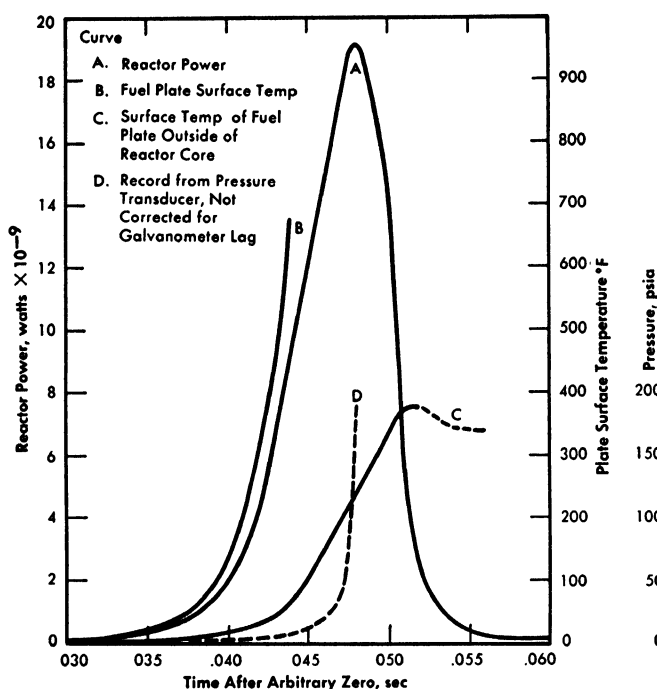


FIG. 3-26. Replot of records from destructive experiment.

however, indicated that the peak pressure was at least as high as 6000 psi and was probably higher than 10,000 psi.

It was evident from examination of the reactor debris that many of the fuel plates had been practically completely melted. Others, evidently those at the edge of the core, had been only partly melted, and portions of them remained fastened to the side plates of the fuel elements. Some of the fragments which had evidently been molten appeared as spongy metallic globules. Other fragments appeared to have been molten inside, while the outside clad remained solid.

Most of the heavy debris fell to the ground near the shield pit. The control rod drive mechanism, which weighed about a ton, fell on the side of the earth shield, after having been thrown about 30 ft into the air. Recognizable fuel plate fragments were thrown as far as 200 ft from the reactor site. Surveys of the total fission-product radioactivity of all the debris indicated that practically all of the fuel originally in the reactor could be accounted for within a radius of 350 ft around the original reactor location. Although these measurements necessarily lacked precision, they showed that no large fraction of the reactor core material left the site in the form of airborne material. At the time of the experiment,

the wind velocity was 8 miles/hr at ground level and 20 miles/hr at 250 ft above ground. Fifteen minutes after the experiment the total beta plus gamma activity level, 3 ft above ground, at a point 0.8 mile directly downwind of the reactor, was 5 mr/hr. At all points farther from the reactor the effects of fall-out were less than this value. Momentarily during the explosion, a gamma dose rate in excess of 400 mr/hr was indicated on a survey meter half a mile from the reactor. This indication decayed rapidly; the total dose received at the half-mile point (cross wind) was less than 10 mr.

Both the observed radiation intensities and the mechanical damage were roughly consistent with the measured nuclear energy release of 135 Mw-sec. Although the explosion was spectacular, its effects were comparable to those which could be caused by a moderate amount of chemical explosive. The destruction of the reactor tank was not surprising, since it was constructed of relatively thin ($\frac{1}{2}$ -in.) steel. Most of the equipment outside the shield tank was either undamaged or repairable, and much of it, including the control rod drive mechanism, was decontaminated, reconditioned, and reused on Reactor II.

There was no evidence that the power-limitation process in the destructive experiment differed qualitatively from that which was effective in the earlier, non-destructive experiments. It was quite evident that the nuclear power release was terminated at an early stage of the explosion; indeed, high-speed motion pictures recorded the light flash emitted by the reactor as it reached high power and showed that it was extinguished before any ejected material appeared above the top of the shield tank. The flash lasted about 0.003 sec. The energy stored in the fuel plates as sensible heat and latent heat of fusion during the relatively short nuclear power burst was, of course, released during the much longer explosion process.

Figure 3-27 is a plot of energy released as a function of reciprocal period for all the power excursions made with cold water in Reactor I. In plotting the curve, a constant energy of 4 Mw-sec has been subtracted from each value. This is the quantity of energy which has been released up to the time when the center temperature of the average fuel plate has reached the atmospheric boiling point of water. The highest point on the curve, which refers to the destructive experiment, does not appear inconsistent with the other values.

Summary of results of BORAX-I tests. While much remained to be learned about boiling reactors, the first BORAX experiments proved that the reactors investigated possessed a high degree of inherent safety and indicated that it is possible to design practical reactors of these types which will be safe against any reactivity accident which is likely to occur

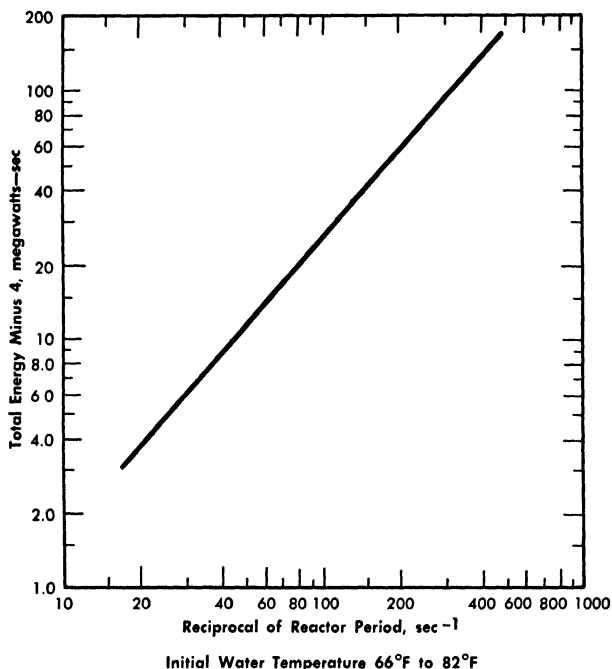


FIG. 3-27. Total energy of excursions minus energy required (4 Mw-sec) to raise temperature of center of average plate to the boiling point.

in practice. The consistent and regular behavior of the reactors during the tests was reassuring. Although such behavior would have been expected in the case of the homogeneous non-boiling reactor, anomalies would not have been surprising in the cases where shutdown depends upon the rapid expansion of moderator and formation of steam. Actually, in the entire series of some 200 experimental runaways, no inconsistencies of behavior were observed which could not be attributed to instrumental errors resulting from rather difficult experimental conditions.

In inferring the behavior of other reactors from those tested in these first experiments, it was realized that the reactors had fuel plates of such high thermal conductance that in all the non-destructive experiments the metal plate represented no important impedance to the transfer of heat to the water. It was quite obvious that the transient behavior would be strongly modified and the effectiveness of steam limitation of power decreased if a departure was made from this condition, as it might well be in other designs. As described later in this chapter fuel elements of quite different design have been used in other BORAX experiments, particularly in the case of BORAX-IV, which uses oxide type fuel elements.

Satisfied that the boiling water concept was feasible, after the destruction of this first reactor in the summer of 1954, Walter H. Zinn and his associates at Argonne made plans to construct a new reactor, one capable of operating at higher pressures and also with a much larger core than that of BORAX-I. In the meantime, however, it was also decided to build a more practical plant for the actual generation of power. This, eventually, became EBWR, the Experimental Boiling Water Reactor, built at the Lemont site of the Argonne National Laboratory near Chicago.

BORAX-II AND III

BORAX-II was constructed during the latter part of 1954 near the site of BORAX-I, utilizing the same control trailer and a great many of the actual parts of BORAX-I.³ Some of the parts had been damaged but could be salvaged. This reactor involved a heavier pressure vessel capable of operating at 300 psi, and with a core twice the volume of BORAX-I, delivered about 6 Mw. BORAX-II operated at various pressures up to 300 psi, while BORAX-III operated only at 300 psi.

This reactor was operated only a short time in the fall of 1954 and early in 1955 because early in 1955 it was decided to install a turbine generator to provide a complete power system, rather than just the reactor alone. A new core was also installed, capable of producing approximately 15 Mw of heat, together with necessary control equipment for continuous operation. This, then, became BORAX-III. Steam generated by the reactor was either vented to the atmosphere or delivered to a 3500-kw turbine generator.

Before this change was made, however, a number of runs were made on BORAX-II. These runs have been analyzed in detail by A. Ulrich.⁶ In this work, ten BORAX-II power excursions were analyzed using oscillograph records. Starting with reactor power and surface temperature of a representative central fuel plate, and in some cases water channel pressure, all as a function of time, the excess reactivities and heat fluxes were calculated as a function of time. Table 3-3 lists the power runs analyzed.

The heat flux into the water from the surface of a centrally located fuel plate was calculated by an energy balance method by subtracting the power absorbed in the plate from the power produced in the plate to give the power absorbed in the water.

TABLE 3-3
POWER RUNS ANALYZED*

Water conditions	Run number	P_i , psig	T_i , °F	Reactor period, sec
Atmospheric subcooled	6	0	79	0.072
	7	0	80	0.043
	7	0	80	0.033
Atmospheric saturated	2	0	203	0.189
	3	0	202	0.097
	1	0	201	0.066
	3	0	196	0.033
300 psig saturated	3	300	417	0.360
	4	300	417	0.081
	6	300	420	0.030

* In this tabulation P_i is the initial gage pressure in the reactor before the excursion, T_i is the initial temperature of the entire reactor before the excursion, and the reactor period is for the excursion before shutdown process starts.

In Fig. 3-28 there is reproduced the Heiland oscillograph trace of a typical power excursion, Run No. 7. The appropriate traces are labeled. The quantities of major interest are shown in Fig. 3-29 for the same run. The power fuel plate, surface temperature rise and heat flux into the water, all for a location near the center of the core, are plotted against time. Also included is a plot of k_{ex} against time. In the figure there may be seen the rapid rise in heat flux from the surface of the fuel plate associated with boiling. It is seen that there is a lag in the decrease of k_{ex} as a result of steam formation; this constitutes the major delay of the system and is due to the inertia of the water.

As shown in Table 3-2, the core in BORAX-II experiments had several concentrations of uranium in the fuel plates. The lowest concentration provided 93.4 g of U^{235} per subassembly; the highest, 157.3 g per subassembly. The spacing of the plates in the subassemblies was the same in the two cores but because of the difference in the number of fuel plates, the ratio of the volume of metal to the volume of water in the two cores was different, 0.49 for BORAX-II and 0.36 for BORAX-III. The power density ratio (item 14 in Table 3-4, p. 90) for BORAX-III was considerably higher, 13 kw/liter per % k , than for BORAX-II, for which the corresponding figure was 8.6 kw/liter per % k .

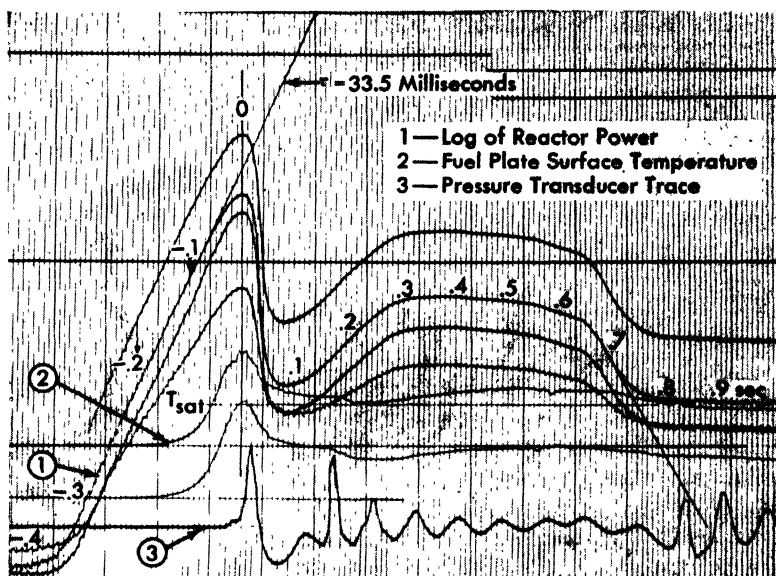
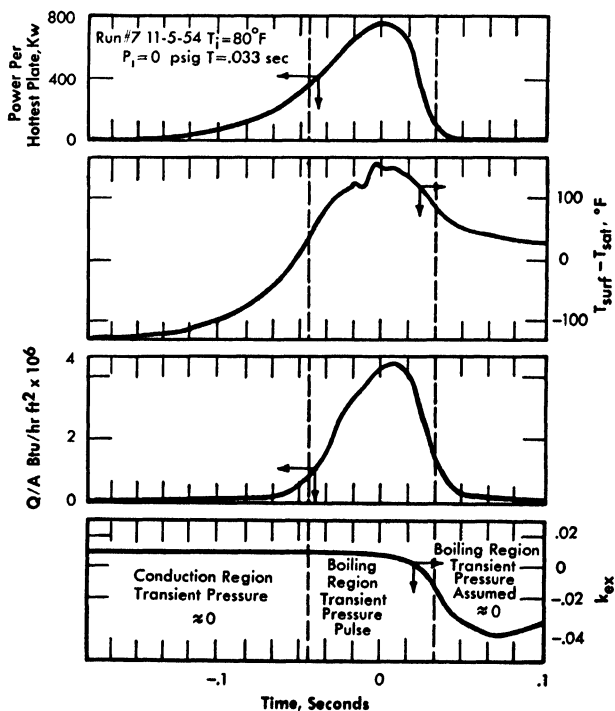


FIG. 3-28. Oscillograph trace for representative run, BORAX-II.

FIG. 3-29. Power, plate temperature, heat flux, k_{ex} for representative run.

The heavily loaded fuel elements are used to adjust the void coefficient of the reactor. Placing the heavily loaded elements at the periphery of the reactor increases the leakage and, therefore, can increase the negative void coefficient.

Reactor core assembly. Since this is a test facility, it was necessary to make arrangements to permit complete change of not only the fuel elements, but of the whole reactor core assembly, including the control rods. The reactor vessel is a stainless steel tank, 15 ft 11¼ in. high and 52½ in. internal diameter, constructed of stainless steel plate, ¾ in. thick, conforming to the code for unfired pressure vessels. The core assembly is carried on a framework which is supported in the reactor vessel from a fixture which is bolted to the reactor vessel at the top, just below the flat lid closure. This construction permits removing the whole core assembly even after the reactor has been operated and core materials have been activated. Figure 3-31 is an illustration of the arrangement of the core assembly in the reactor vessel. Figure 3-32 is a cross section

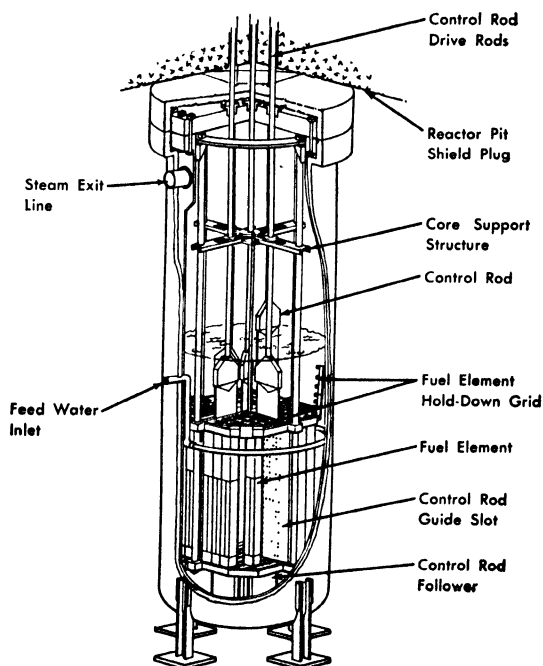


FIG. 3-31. Core assembly in reactor vessel.

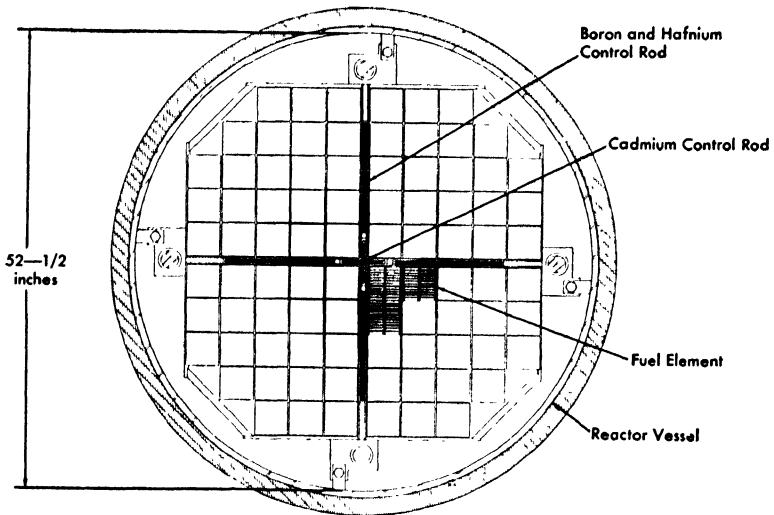


FIG. 3-32. Cross section through reactor core.

through the core, giving the dimensions of the four blade-type control rods and the central cross control rod. Cadmium, boron, and hafnium are the absorbing materials in the control rods.

The control rod drives are situated above the dense concrete shield plug which is rolled over the top of the reactor. In order to change fuel in the reactor, it is necessary to disconnect the control rod drive rods, roll back the shield plug and remove the lid of the reactor vessel. To make this possible, the reactor vessel and the pit in which it stands are flooded with water. The insulation on the outside of the reactor vessel is of a type which does not deteriorate due to immersion in water.

Figure 3-33 is a schematic diagram of a single control rod drive. These drives are so arranged that a control rod at any time can become a rapidly operating shutdown rod. A control rod which is driven by a shaft sealed through the wall of a pressure vessel, such as this, has two severe difficulties. First, the internal pressure acts to lift the control rod out of the core, and, secondly, the friction of the packing gland opposes any motion of the rod. In the present design, the lifting force due to the steam pressure in the reactor vessel is balanced out with a steam cylinder and piston in which the drive rod terminates. Steam for the balancing cylinder is supplied from the reactor vessel.

A large force for moving the control rod is supplied by a pneumatic piston which exerts a force of 800 pounds to drive the rod. This is much larger than any friction force in the packing gland. When the rod is

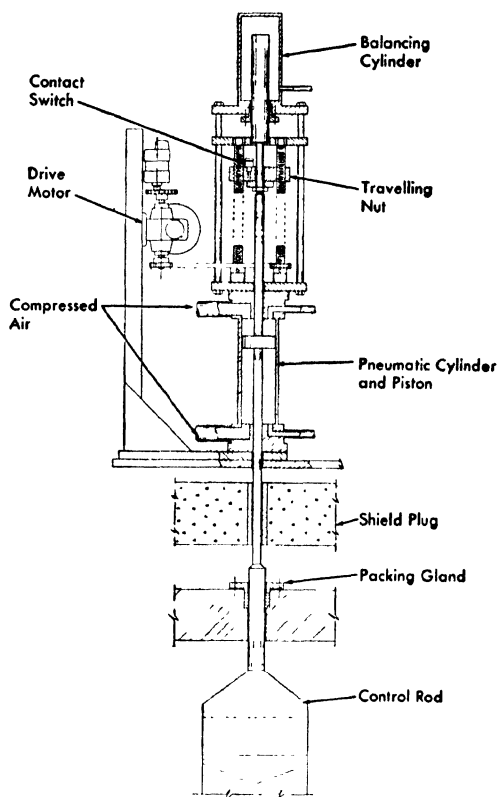


Fig. 3-33. Diagram of control rod drive: (1) control switch, (2) drive motor, (3) compressed air, (4) balancing cylinder, (5) traveling nut, (6) pneumatic cylinder and piston, (7) shield plug, (8) packing gland, and (9) control rod.

in use as a regulating rod, the pneumatic cylinder holds a stop on the drive shaft in contact with a large nut which straddles two lead screws, which are driven in synchronism. Driving the nut up or down imparts the same motion to the control rod. Rapid insertion of the control rod is achieved when the air pressure is switched to the side of the pneumatic cylinder which drives the rod down.

Plant layout. Figure 3-34 is a flow diagram for the whole plant. Three principal circuits are indicated.

The power circuit consists of a feedwater pump which can pump into the reactor either condensate returned from the condenser or freshly de-ionized water from the makeup system. The feedwater is discharged

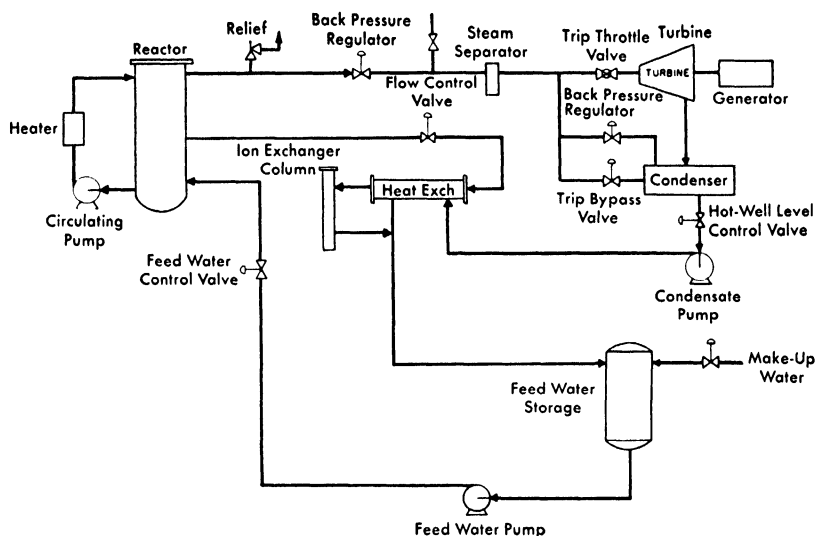


FIG. 3-34. Flow diagram for complete plant.

into the annular downcomer of the reactor. This helps to cool the downcomer water and promotes the natural circulation in the reactor vessel. A large volume in the upper part of the reactor vessel contains only steam. This large volume promotes the separation of steam and water and acts as a cushion, preventing rapid pressure changes. Steam from the reactor vessel is carried by a 6-in. line either to the turbine or to a vent to the atmosphere. Valves suitable for these operations are shown. The condensate from the condenser is pumped to the feedwater supply tank.

The second circuit consists of a pump and electrical heaters. This permits pressurizing the reactor vessel without operation of the reactor. It is useful for testing for leaks and also is used to raise the temperature so that the reactor will not have to be run with the control rods inserted most of their length.

The third circuit is the cleanup circuit and consists of filters and ion exchange columns. In operation these will become quite radioactive; consequently they are installed in the fuel element storage canal so that the water of the canal will provide the necessary shielding.

Instrumentation. Since this facility is also used for transient testing of reactor cores, the control room is situated some distance from the reactor itself. All indication of operating conditions, therefore, must be made by electrical means. The level of the water in the reactor, which

normally is about 4 ft above the active core, is closely controlled. Instruments for recording the level of the water are installed in a standpipe exterior to the reactor vessel and in the shielded pit which contains the pumps and other auxiliary equipment. Level control is by means of a float-type level controller installed at the operating level. A television camera is arranged so that by means of it the level in a sight glass can be read directly. The television camera also can be used to read directly a Bourdon-type pressure gauge which measures the steam pressure in the reactor vessel. In the control room the rate of flow of feedwater as well as the steam flow is recorded.

Ionization chambers recording neutron flux are installed in tubes which extend close to the core, but outside the reactor vessel. Gamma-ray intensity at the top of the reactor vessel is recorded. The neutron intensity is measured by three types of instruments differing in their time constants. For normal operation, an amplifier recorder system having a response time of 0.05 sec is used. In order to observe transient effects, a Brush amplifier recorder with a time constant of 0.002 sec, or a Heiland galvanometer-oscillograph with a time constant of 0.0004 sec, is used for neutron intensity.

Steady-state operation. It has been possible to operate the reactor under conditions which are stable for all pressures from atmospheric to 300 psi. A necessary requirement for reasonably smooth operation is that the amount of reactivity corresponding to the steam void not be excessive. In all operations, however, the power level is subject to rapid variations. These clearly are connected with the turbulence created by the formation of steam. The degree of roughness of the power-time curve is directly related to the excess reactivity which has been subtracted by the formation of the steam voids and replaced by withdrawal of the control rods.

A series of experiments was performed to determine the relationship between the excess reactivity in voids and the power density, and this has been done for a number of pressures. It was observed that for any given pressure the degree of roughness of the power varies with the excess reactivity in steam voids. If the excess reactivity is raised to too high a figure, oscillations in the power become apparent, and further increases in excess reactivity in steam may bring small excursions in the reactor power.

Figure 3-35 shows curves of power density versus excess reactivity in steam for three BORAX reactors (I, II, and III). The core for Reactor III is that illustrated in Fig. 3-32. The core for BORAX-II, although it used the same vessel, was considerably smaller than that for BORAX-III and had a higher (negative) steam coefficient of reactivity. Conse-

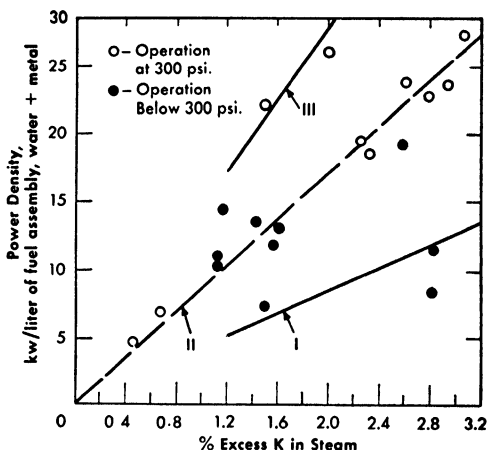
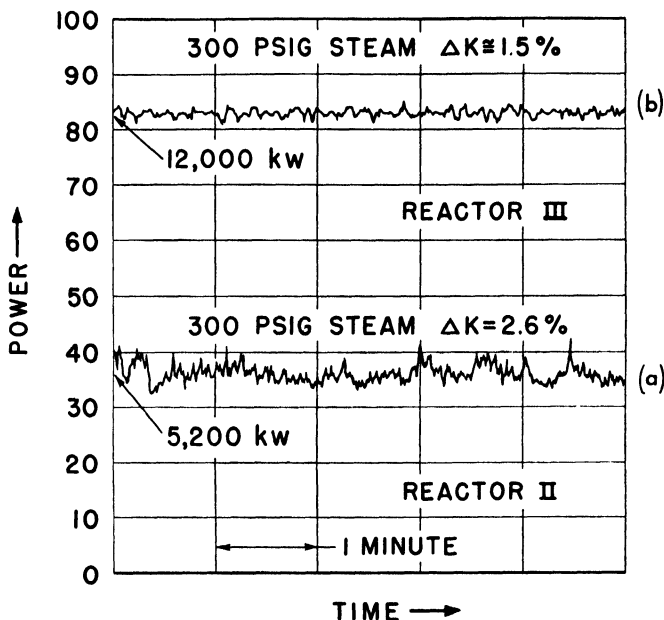


FIG. 3-35. Effect of reactivity on power density in three BORAX reactors.

quently, a given reactivity in Reactor III corresponds to a higher steam content per unit volume than the same reactivity in Reactor II, and the power density of Reactor III is correspondingly higher.

Figures 3-36(a) and 3-36(b) are reproductions of sections of the power traces of Reactors II and III, as obtained by a neutron-sensitive ion chamber and fast recorder. Both reactors were operating at 300 psig and at roughly the same power density, about 22 kw/liter.

The curve for Reactor II in Fig. 3-35 includes points at several operating pressures. There is no great variation with pressure in the power density attainment from a given addition of reactivity, at least in the pressure range investigated. This insensitivity to pressure is qualitatively consistent with the results of laboratory boiling experiments.⁸ Although the energy content per unit volume of the steam increases almost linearly with pressure, the velocity of slip of the steam past the water in the fuel element decreases with pressure. The steadiness of operation of the reactor increases markedly with pressure, however, and it is therefore feasible to operate the reactor with more reactivity in steam voids and consequently at higher power at high pressure. In Fig. 3-35, the highest plotted points for the 75 psi and the 150 psi conditions represent approximately the upper limits of satisfactory operation at these pressures. At 200, 250, and 300 psi, however, the maximum reactivity added was limited by the available amount in the reactor; at these higher pressures the curve could have been extended to somewhat higher power density if additional reactivity had been available. Table 3-4 summarizes the results of power density studies which have been obtained for three



POWER RECORDS FROM TWO BOILING REACTORS

FIG. 3-36. Reproduction of typical power records from two different boiling reactors: (a) Reactor II, (b) Reactor III.

reactor cores which differed in size, plate spacing, and the U^{235} loading of the plates. (Note: This table also includes data on BORAX-IV covered later in this chapter.)

Item 1 in the table shows that the second reactor had the plate spacing increased by 83 per cent with respect to Reactor I. From item 4, it is seen that this increased the ratio of water to metal in the core by the factor 1.26. The effective neutron lifetime is not changed much, as is indicated in item 5.

As in all water reactors, the temperature coefficient is sizable and represents a troublesome problem for control. In item 9, the over-all reactivity loss observed in heating from shutdown to near operating temperature is given. It should not be assumed that the temperature coefficient of reactivity is at all constant over this range of temperature.

In item 10, the calculated void coefficients for the reactors are given. A certain fractional void (less than 10%) was assumed. The corresponding reactivity change then was computed and recorded.

TABLE 3-4
COMPARISON OF BORAX REACTORS

Item	BORAX-I	BORAX-II	BORAX-III	BORAX-IV
1. Plate spacing, inches	0.177	0.324	0.324	0.604
2. Number of fuel plates per subassembly	18	10	24	6
3. Size of subassembly	3 by 3	3 by 3	3 888 by 4.0	3 888 by 4.0
4. Volume ratio of metal to H ₂ O in core	0.63	0.49	0.36	0.52 (includes UO ₂)
5. Effective neutron lifetime, sec	6.5×10^{-5}	7.5×10^{-5}	7×10^{-5}	
6. Volume fraction of water in core	0.614	0.672	0.735	0.656
7. U ²³⁵ per subassembly, g	138 6	93 4	137	283
8. Heat-transfer area per unit volume sq ft/cu ft of core	120	66.7	63 9	34
9. Temperature coefficient of reactivity (27 to 182 C), % k_{eff}	3.5	1 16	0 8	1.6
10. Void coefficient (calculated) % k_{eff} / % steam	-0 24 (207 F)	-0.10	-0.10	-035 (207 F)
11. Power density, kw/liter of core	8 4 (atmospheric)	27 (300 psi)	27 (300 psi)	19.6 (atmospheric)
12. Power density, kw/liter of coolant	10.0 (135 psi)	38.5 (300 psi)	38 (300 psi)	36.0 (300 psi)
13. Reactivity in steam, % k_{eff} in voids	14.7 (atmospheric)	3 1 (300 psi)	2.0 (psi)	30.5 (atmospheric)
14. Power density ratio in kw/liter/ % k to % k_{eff} (void)	17.5 (135 psi)	8.6	13	56.5 (300 psi)
	2.0 (135 psi)			1.25 (atmospheric)
	4.1			31 elements
				5.5 (300 psi)

In item 14, the power density for per cent reactivity subtracted by the voids is given. This quantity is measured by the total power of the operating reactor and by the amount of reactivity added by the calibrated control rods. It would be of great interest to be able to measure directly the actual steam void in the operating reactor.

Laboratory experiments have been done in which the fractional void in a heated pressurized channel has been measured as a function of operating power. The void fraction was measured by the attenuation of γ rays passing through the channel. It can be said that the quantity [power density/% k_{eff} (void)] in the reactor is not in disagreement with the above experiments and the calculated void coefficients.

The void coefficient of reactivity apparently is the principal factor affecting the performance of the several reactor cores so far studied. Obviously from the point of view of obtaining the maximum power density the smallest void coefficient is wanted; considerations of the degree of inherent self-limitation for power excursions will play a large part in the choice of void coefficient for any practical power reactor.

It is interesting to note the power level at which a typical fuel element operates. The power density of 22 kw/liter, which resulted in the power traces of Figs. 3-36(a) and 3-36(b), may be taken as representative. The active section of the fuel element of BORAX-III has a volume of 6.26 liters, and therefore the average fuel element delivers 138 kw. Fully loaded, the reactor contains 87 fuel elements (one fuel element position is reserved for the installation of a start-up source) and hence, at this power density, the total power is 12,000 kw. The surface area of heated fuel plate in an element is 12,000 cm². The average heat flux is therefore 3 cal/cm² sec, and the maximum about 7 cal/cm² sec. It is clear that in these boiling reactors, which employ only natural circulation of the water, the heat rates attained were not high; the limiting factor on power output was the rate of flow of steam out of the reactor core rather than the heat flux from the fuel plates. This characteristic is, of course, favorable from the standpoint of reactor safety; at the same time power densities in the range 20 to 30 kw/liter of core (or 30 to 40 kw/liter of water in the core—see Table 3-4), which have been experimentally attained, are sufficiently large for many power reactor applications.

The actual loading of the reactor is complicated by the fact that, for the experimental program on attainable power densities, core sizes less than the maximum must be used because the extra fuel allowance required for burnup in continuous operation is not wanted. It also is complicated by the fact that the experiments must be performed either without appreciable xenon poison or with saturated xenon poison. Great care in loading has been used to avoid arriving at erroneous conclusions,

because in one experiment the full height of the reactor was used whereas in another the effective height was reduced by operating with the control rods partially inserted.

For continuous operation, where burnup will steadily reduce the available excess reactivity, burnup plates which incorporate a certain amount of boron are affixed to the subassembly of Fig. 3-30. The time interval between successive additions of fuel can be substantially increased by the use of such burnable poison plates.

Stability under various operating conditions. As indicated earlier, reactivity can be added safely to this reactor in amounts which are dependent upon the pressure of operation. Experiments with BORAX-II indicated that at 300 psi with a water channel width of 0.324 in., steady power operation is obtained even with 3.2 per cent reactivity subtracted by the steam voids. Experiments at lower pressures clearly indicate that there is an amount of excess reactivity above which instability sets in. This is not a matter of great concern because, as more and more reactivity is added, the instability manifests itself by oscillations in the power level. The oscillations have a period of about 0.4 sec and as successive amounts of reactivity are taken up by the steam voids, the amplitude grows.

Experiments have been performed in which reactivity has been added at a fixed rate by withdrawal of the control rods. In this case, at the start of the withdrawal of the rods, the power level was 485 kw and the reactor pressure was 300 psi. In approximately 40 sec, 2.5 per cent excess reactivity was added. The power level rose with an approximately 6.5 sec period and leveled off at 5400 kw without any indication of unstable operation. In a second case, an excess reactivity of 2.6 per cent was added in a time of 160 sec. In this case the pressure was 300 psi and the power of the reactor at the beginning of the experiment was only 1.2 kw; in other words the amount of void volume due to steam formation was essentially zero. Reactivity addition was at the rate of 0.016 per cent per second. The reactor power increased with an approximate period of 3.3 sec and the power leveled off at 4300 kw.

There has been considerable conjecture as to the consequences of a sudden closure of the steam exit line from a boiling reactor, so an experiment was performed in which the steam delivery line was suddenly closed. In this case, the reactor was operating at 150 psi and at a power level of 1400 kw. In these circumstances, 2 per cent reactivity has been subtracted by steam voids. Closing the valve initiated an increase in power which then leveled off at a little over 3000 kw. As was expected, the pressure in the reactor vessel rose steadily but no instability was intro-

duced by the sudden closure of the valve in the steam delivery line. This experiment was repeated a number of times and it indicated that there is no particular hazard connected with this circumstance.

Other BORAX-II experiments performed included letting the pressure in the reactor rise until the safety valves opened and here again, this caused no difficulty at all. At a predetermined pressure the safety valves would open, the pressure would fall with a small corresponding fall in the reactor power, and then, when the relief valve closed, the reactor power would readjust itself to approximately its initial value.

BORAX-III SPECIAL TESTS

BORAX water-quality control. The efforts to outline and solve the problems connected with the reactor water represent one of the major efforts of the BORAX-III program. Maintaining high quality water of a proper pH was the first step.⁹

The circuit used for cleanup takes water from the reactor, cools it to 100 F or lower, filters it through a 10 micron felted cartridge, passes it through ion exchanger resin columns, filters it again, and then discharges the water to the feedwater storage tank. Strong anion and strong cation resins are always used.

Early in the operating stages, the reactor water contained boric acid for control purposes. When sufficient buildup of fission products and burnup of U^{235} had taken place, boric acid was no longer necessary, and its final removal from solution was made by means of an anion exchange resin column. The mixed bed resins were used to remove all types of ionic contaminants.

Strong anion resins are satisfactory for completely removing the boric acid. However, the operation of anion plus mixed bed resin columns resulted in the pH increasing to 7-8. This pH is in or approaching the range where the aluminum corrosion rate is intolerable. The water did reach a reasonable level of purity for the specific resistivity reached 500,000 ohms/cm³.

The second attempt at water quality control was through the use of separate anion and cation resin columns. The function of the anion resins was again for the removal of boric acid. The cation column was used to lower the pH down to a range of 5 to 6, which range is closer to optimum operating conditions for the lowest possible aluminum corrosion rate. The reactor water was controlled to a pH level as low as 5.0. At this pH the water quality had decreased so that the specific resistivity was 150,000 ohms/cm³. Operations with a higher flow rate through the anion column resulted in increasing the specific resistivity to 450,000 ohms/cm³ and the pH to a value over 6. A particular effort was made

during this second series of tests to obtain megohm quality water. This was not reached. The columns did perform satisfactorily to allow the desired pH control.

The third series of tests involved cation and mixed bed resin columns. Boric acid was not present in the water during this period. The cation resin column was operated continuously to hold the water in the desired slightly acid state. The mixed bed resin columns were operated periodically to increase the water quality. Attempts to obtain megohm quality water were closer to success in this series of tests. At one time a specific resistivity of 900,000 ohms/cm³ was reached at a pH of 6.6. Normally the reactor water pH was maintained in a range of 5.5 to 6.0, and the specific resistivity was 400,000 to 500,000 ohms/cm³.

The water quality control tests on the BORAX-III system represent a good start at studying and solving the problems. No attempt is made to draw generalized conclusions, since the work is preliminary and was in some aspects governed by the presence of the aluminum in the reactor core. Of the water quality control methods used for BORAX-III, the mixed bed resin and cation resin combination was best. It is probable that a continuous small amount of leakage of raw condenser cooling water into the reactor water system prevented reaching one megohm quality water. However, failure to reach this goal may have no significance. The reactor water rarely had a specific resistance of less than 150,000 ohms/cm³ which is still good quality water.

Water—radioactive carryover-entrainment. It was noticed that the ratio of the reactor water activity to the condensed steam activity (DF factor) increased with increasing time after sampling. This indicated that the foreign material in the steam was not representative of that in the reactor water. Since the DF factor improves with time up to 2 to 3 days, it also shows that the ratio of short- to long-lived activities is greater in the steam than in the reactor water. Figure 3-37 has activity decay curves of reactor water, steam and condensate samples. Figure 3-38 shows the increase in the DF factor with time up to about 30 hours.¹⁰

The carryover of activity from the reactor water into the steam phase did not reach high levels. Decontamination factors were in the range $0.6\text{--}1.6 \times 10^4$. Indications, from the short-term operations, are that short-lived activities do not accumulate in the external circuit. To date there has been no buildup of activity at any point in the external circuit which would make direct maintenance of equipment difficult. For example, the activity at the exhaust end of the turbine twelve hours after final shutdown read 1-1.5 mr/hr through the turbine casing. At shutdown the reading was probably about 20 mr/hr.

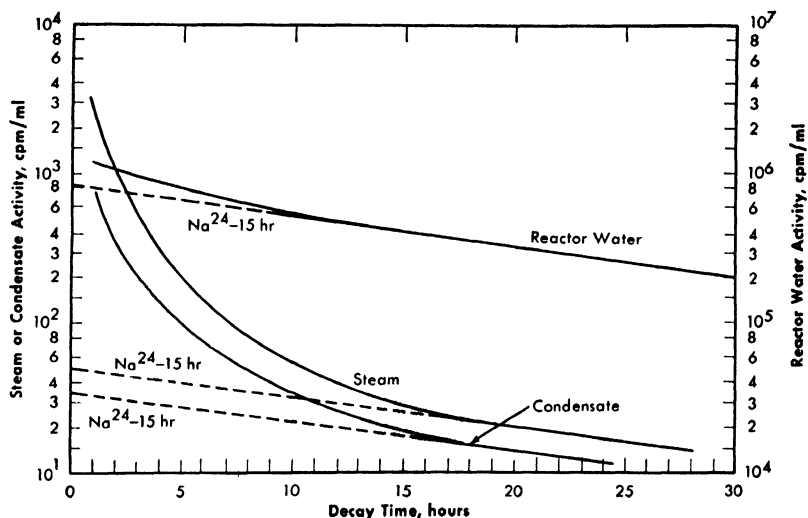


FIG. 3-37. Decay curves for 1 to 30-hr period. BORAX-III samples at 4 Mw.

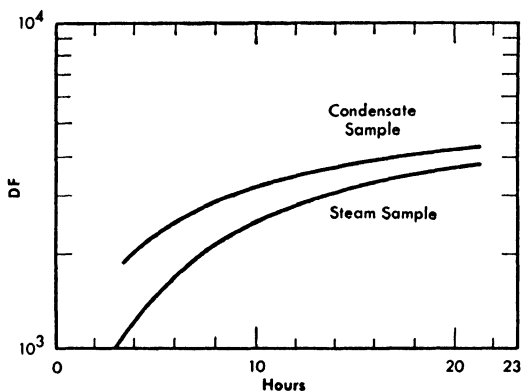


FIG. 3-38. Change in decontamination with time.

The separation of entrained moisture from the steam in the vapor section of the reactor vessel was excellent and the amount of liquid entrained in the steam agreed with experiments performed prior to reactor operation. The steam quality was in excess of 99.9%. Samples of steam condensate from either side of the externally located steam separator showed no differences in activity. The separator did have some drainage which had higher activity than was present in the condensed steam. The net conclusion is that while the separator performed some function, it was obviously a small one.

The studies of the radioactive carryover and entrainment are fully reported by A. J. Shor.¹⁰

The activity in the external circuit varied greatly during operation. The reactor water conditions and the operating temperatures of the condenser markedly affected the activity at various points in the circuit. As the pH increased, the activity measured at the air ejector tended to decrease. A decrease in activity at the air ejector was noted with decreasing condenser temperature. Table 3-5 gives some of the activity levels throughout the system during operation. The water quality is the variable in the three test runs.

In measuring the entrainment on the BORAX-III reactor, the activities carried by the steam and the condensate were sampled and compared to the activities in the reactor water. Figure 3-39 is a sketch of the relative locations of the sampling stations. Samples were taken from the steam line leaving the condenser hotwell pump. Steam samples were removed at isolinetic rates, throttled to about atmospheric pressure, and condensed. Care was taken to prevent knockout of moisture in the sampling process. The reactor water sample was obtained directly from a continuously circulating loop, run through a heat exchanger and throttled to atmospheric pressure. No flashing of the water was permitted.

During the test period the reactor was operated for about three days at each of four different power levels. Initially, samples were taken at the 12-Mw level until a steady condition was achieved. The power was then decreased to 8 Mw and the length of time required to reach a new steady condition was determined by periodically measuring the entrainment. The power was then decreased to 4 Mw for about 2½ days and finally samples were obtained at 14 Mw. The power levels observed correspond to a power density range of about 9 to 32 kw/liter of core volume.

TABLE 3-5
RADIATION SURVEY OF BORAX-III STEAM CYCLE COMPONENTS

Reactor power level, Mw	8	8	8
Water pH	5.7	6.7	8.1
Water, specific resistivity, ohms/cm ³	270,000	490,000	150,000
Survey location			
Steam line-reactor building, mr/hr	430	160	120
Turbine inlet, mr/hr	145	55	30
Turbine outlet, mr/hr	50	20	15
Air ejector, mr/hr	245	210	85
Hotwell bottom, mr/hr	290	100	50

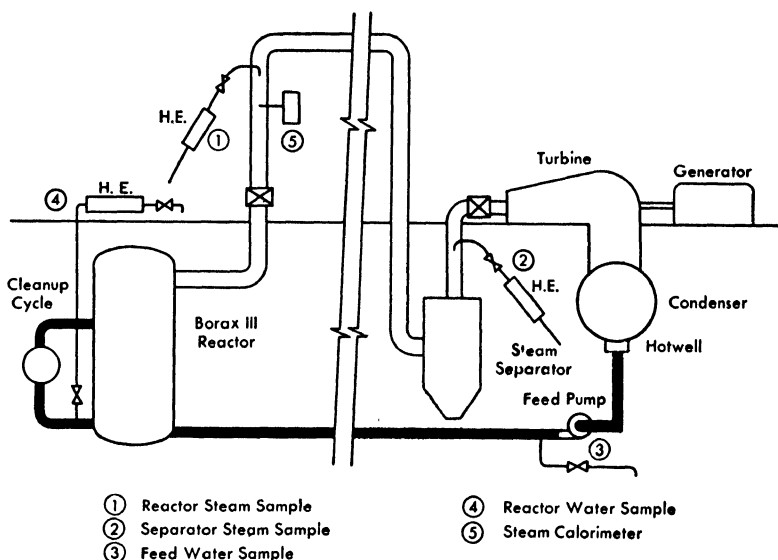


FIG. 3-39. BORAX-III sampling stations.

Samples were removed at approximately 8- to 12-hour intervals. Four milliliter aliquots of overhead samples were counted in a beta-gamma proportional counter.

Sample analysis. The reactor water was continuously deionized and varied in specific resistance between 5×10^5 and 2×10^5 ohm-cm. It was maintained at a pH of 5 to 6 during the tests. The activation products dissolved in the water were derived from natural impurities, from corrosion products of the reactor and steam system, and from dissolved gases present in the water. The activity in the steam and condensate came from reactor water droplets carried as entrainment, and from volatile activation products. These were continually being swept out of the reactor water in the boiling process. In order to differentiate droplet entrainment from the carry-over of volatile activation products, the decay schemes of the reactor water, the steam, and the condensate were studied. The decay curves for the reactor water samples (log of activity vs time) exhibited considerable curvature for about the first 20 hr of decay. However, in the interval of 20- to 30-hr decay, the principal activity appeared to be due to 15-hr Na^{24} . The decay curves for the steam and condensate showed a similar, but not identical, decay scheme as a result of partition of volatiles between the reactor water and condensate. After some 20 hr decay of the steam and condensate samples, the rates of decay were constant with slopes characteristic of Na^{24} .

Therefore, a decay interval between 20 to 30 hr was chosen as that representative of the activity associated with decontamination factors based on droplet entrainment.

Figure 3-37 shows a semilog plot of a set of data obtained at the 4-Mw operating level. Although the steam and condensate decay curves show similar curvature, the marked difference in the 1- to 20-hr decay rates between the reactor water and the condensate is readily apparent. However, after about 20 hr decay all these curves approach the slope of 15-hr Na^{24} . Figure 3-40 shows a semilog plot of a set of data obtained at the 14-Mw operating level. The decay range from 20 to 80 hr is covered. In the interval 20 to 50 hours, the decay rates show half-lives of about 15-hr decay; the slope begins to change so that the steam and condensate appear to decay at a slower rate than the reactor water. The presence of volatile products which are carried out of the reactor water with the steam is considered the principal mechanism for the differing

TABLE 3-6
RADIOACTIVITIES OBSERVED IN REACTOR WATER AND FILTER

Element	Probable formation	Half-life
N^{16}	$\text{N}^{16} (n, \gamma), \text{O}^{16} (n, p)$	7.3 sec
Cl^{38}	$\text{Cl}^{37} (n, \gamma)$	37.3 min
A^{41}	$\text{A}^{40} (n, \gamma)$	110 min
Mn^{56}	$\text{Mn}^{55} (n, \gamma)$	2.6 hr
Cu^{64}	$\text{Cu}^{63} (n, \gamma)$	12.8 hr
Na^{24}	$\text{Na}^{23} (n, \gamma); \text{Al}^{27} (n, \alpha)$	15.0 hr
Cr^{51}	$\text{Cr}^{50} (n, \gamma)$	27 days
Fe^{59}	$\text{Fe}^{58} (n, \gamma)$	46 days
Sr^{89}	$\text{Sr}^{88} (n, \gamma)$	54 days
Ca^{45}	$\text{Ca}^{44} (n, \gamma)$	152 days
Zn^{65}	$\text{Zn}^{64} (n, \gamma)$	250 days
Fission activities observed in reactor water		
Mo^{99}		67 hr
Ba^{140}	$\text{Xe}^{140} \xrightarrow{16 \text{ sec}} \text{Cs}^{140} \xrightarrow{66 \text{ sec}} \text{Ba}^{140}$	12.8 days
Ce^{141}	$\text{Xe}^{141} \xrightarrow{3 \text{ sec}} \text{Cs}^{141} \xrightarrow{\text{S}} \text{Ba}^{141} \xrightarrow{18 \text{ min}} \text{La}^{141} \xrightarrow{3.7 \text{ h}} \text{Ce}^{141}$	32 days
Ru^{103}		41 days
Zr^{95}		65 days
Ce^{144}	$\text{Xe}^{144} \xrightarrow{\text{S}} \text{Cs}^{144} \xrightarrow{\text{S}} \text{Ba}^{144} \xrightarrow{\text{S}} \text{La}^{144} \xrightarrow{\text{S}} \text{Ce}^{144}$	280 days
Ru^{106}		1 year
U^{238}		

decay rates. It is significant from the point of view of operation that at the time of sampling the condensed steam showed a higher activity than the reactor water. The activity in the steam rapidly decayed with about a 7-sec half-life indicative of N^{16} . Radiation survey instruments were used in this test.

Table 3-6 is a list of observed radioactive nuclides in the reactor water. The presence of Ba^{140} , Ce^{141} and Ce^{144} which are daughters of the volatile fission products Xe^{140} , Xe^{141} , and Xe^{144} , respectively, are examples of how longer-lived activities could contribute to the differing decay rates of the overhead steam and the reactor water. The presence of fission products in the reactor water appeared to be the result of very slow leakage through the fuel element cladding. No attempts were made to measure the extent of particulate matter in the reactor water for this series of runs. However, it was apparent that as a result of an effective filtering system, very little curd was present during operation. Samples of reactor water which were centrifuged showed little change in activity in the supernatant liquid.

Experimental results—decontamination factors. Figure 3-41 shows the variation of decontamination factors with time based on the steam and condensate samples. Approximately one day following a change in power and for the duration of the test, the decontamination factors show little fluctuation. A trend can be seen at different power levels. For 14 and 12 Mw the decontamination factors average about 6×10^3 while at the 4- and 8-Mw levels decontamination factors are about 1.5×10^4 . These values correspond to 20- to 30-hr decay so that the decontamination factors are based on Na^{24} activity and is indicative of the droplet carry-over. Referring to the data at 12 Mw, it is noted that the decontamination factors, based on steam samples taken before and after the moisture separation, check very closely.

Results. Figure 3-42 shows a graph of the decontamination factors obtained in the tests as a function of the velocity of the steam in feet per second in the free space above the boiling interface. The vertical lines indicate the range of decontamination factors based on the steam samples. The intersection horizontal lines represent average values for samples permitted to decay for 20 to 30 hr. It is seen from the graph that between the power levels of 4 to 14 Mw and corresponding steam velocities in the reactor free space between 0.3 and 1.0 ft/sec, no sharp break in the curve or abrupt priming of reactor water occurs.

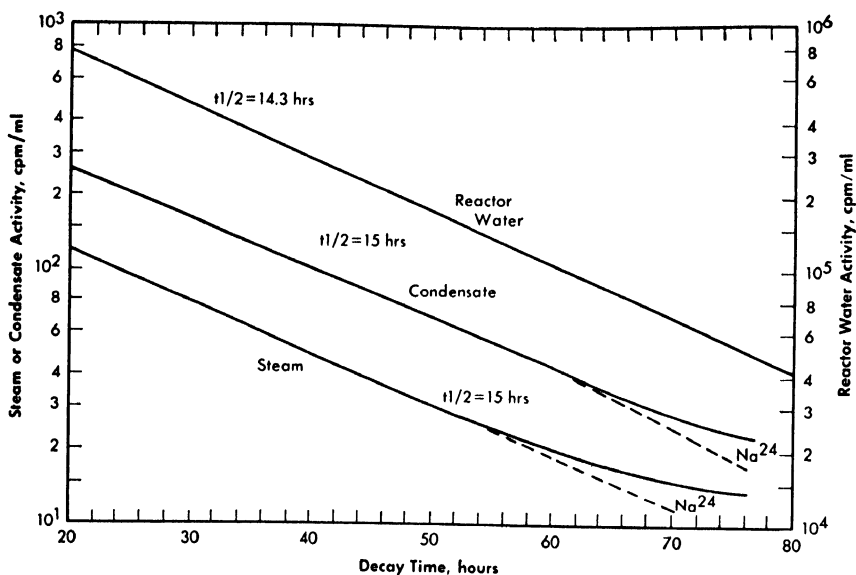


FIG. 3-40. BORAX-III decay curves for 28 to 80-hr period. Samples at 14 Mw

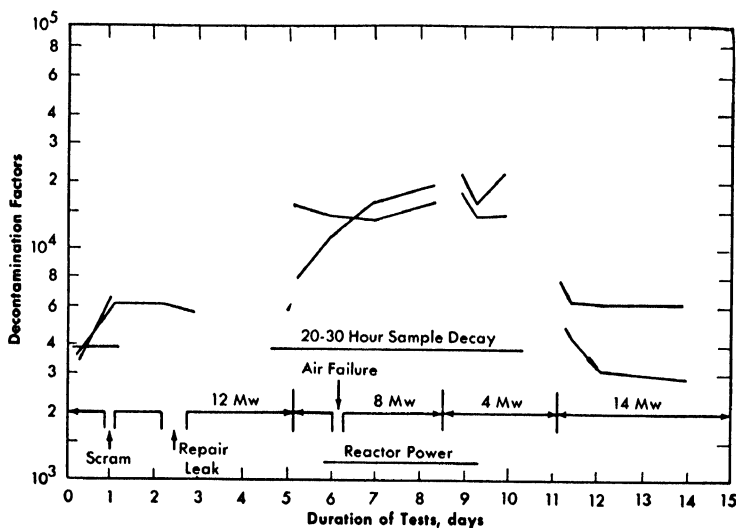


FIG. 3-41. Decontamination factors for steam and condensate. 20 to 30-hr sample decay.

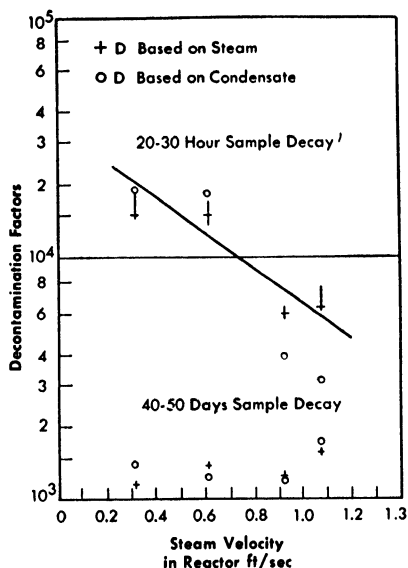


FIG. 3-42. Decontamination factors of steam and condensate in BORAX-III.

A comparison of the decontamination factors based on the condensate, with the points based on the steam, indicates that they are of the same order of magnitude. It appears that, at least for short-lived activities, only a small amount of the entrainment is deposited in the external machinery.

Decontamination factors based on steam and reactor water samples permitted to decay for about 40 days do not exhibit any noticeable trend with power level and fall in the range 1 to 2×10^3 . A mechanism for this behavior based on a partition of activities resulting from relatively long-lived volatile species has been mentioned.

Figure 3-43 shows the data obtained from the BORAX-III test, together with those obtained in the high pressure loop experiments and in the laboratory scale entrainment study apparatus desired. The data taken in the high pressure loop are from the uppermost sampler (about 42 inches from the boiling interface) at pressures from 150 to 600 psia. Although the data scatter, they exhibit a similar trend in the range of steam velocities observed. The three curves are displaced in respect to the range of decontamination factors observed. This displacement may be accounted for in part by wall effects in the experimental apparatus. The diameter of the cylindrical boiler was about 3 inches in the atmospheric runs and about 5 inches in boiling loop experiments. The diameter of the reactor vessel was $4\frac{1}{2}$ feet. Thus, considering the prob-

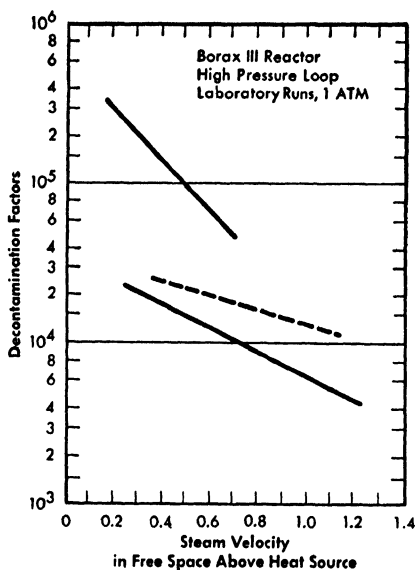


FIG. 3-43. Decontamination factor vs. steam velocity in BORAX-III.

able range and direction of the droplets leaving the boiling interface, it is reasonable to expect that in the smaller vessels a greater proportion of the droplets impinged on the walls.

Water decomposition. During the later stages of BORAX-III considerable effort was made to obtain data on water decomposition. The first data were collected during a period of operation when pH gradually decreased from 6.7 to 5.2 and the specific resistivity decreased from 500,000 to 150,000 ohms/cm³. An anion resin column was then put into service and the water increased in pH to 6.75 and in specific resistivity to 750,000 ohms/cm³. During this period of improving water quality and increasing pH, the noncondensable gases in the steam phase of the cycle decreased from 197 to 72 cc/liter of condensed steam.

It is usually assumed that water quality is the major influence on radiolytic decomposition. However, these data plus data collected on subsequent runs did not correlate with water quality. Figure 3-44 is a plot of the quantity of noncondensable gases in the steam *versus* the pH of the reactor water. These data do not include gases dissolved in the condensed steam samples. It is estimated that the quantity of dissolved gases was a constant of about 6 cc/liter. This would increase the results shown in Fig. 3-44 by less than 10 per cent at the lower end of the curve. The composition of gases was relatively constant throughout all tests, being about 31 per cent O₂, 60 per cent H₂, and 9 per cent N₂.

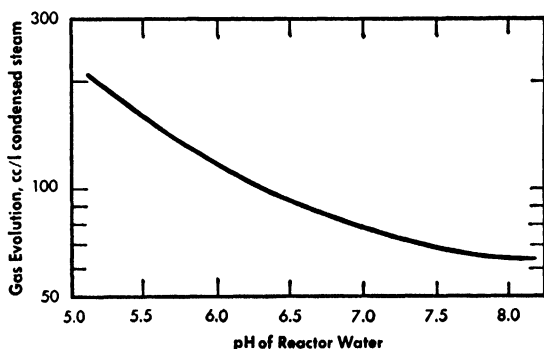


FIG. 3-44. Gas evolution vs. pH of reactor water.

While the data correlate with pH, sufficient evidence is not available to conclude that pH per se was the major contributor to the change of water decomposition. Some fundamental data indicate that certain ions affect the rate of water decomposition. In the BORAX work, the pH was almost always adjusted by removal of ions with ion exchange resins and not by addition of known ions. Thus, the composition of the residual ionic contamination was an unknown.

Figure 3-45 is a plot of pH *versus* the molecules of hydrogen liberated per 100 ev of energy absorbed in the water (G_{H_2}). About 4 per cent of the fission energy was assumed to be absorbed by the water. The value for the liberation rate, if there were no recombination, has been estimated to be 1.0 molecule per 100 ev of energy absorbed in the water. The plot given in Fig. 3-45 indicates that only 50 to 80 per cent of the decomposition gases recombine. It appears possible that the sweeping action due to boiling may cause an appreciable increase in the release rate of decomposition gases over that experienced from static or flowing nonboiling water subject to the same radiation conditions.

Evolution of gases resulting from the radiolytic decomposition of water is of concern mainly because of the effect which the free oxygen can have on the materials of construction used in standard turbines and condensers. The evolution rates shown in Fig. 3-44 represent a range of 25 to 80 ppm of oxygen in the steam. This concentration might cause considerable corrosion within the turbine and condenser. However, virtually no experience has been obtained under operating conditions to either prove or disprove that serious corrosion would exist.

Addition of hydrogen gas to the reactor water was tried as a method for reducing the oxygen content of the steam. Figure 3-45 is a plot of the results of the test. The addition through the feedwater sparger was more effective than addition through the reactor vessel drain line, probably

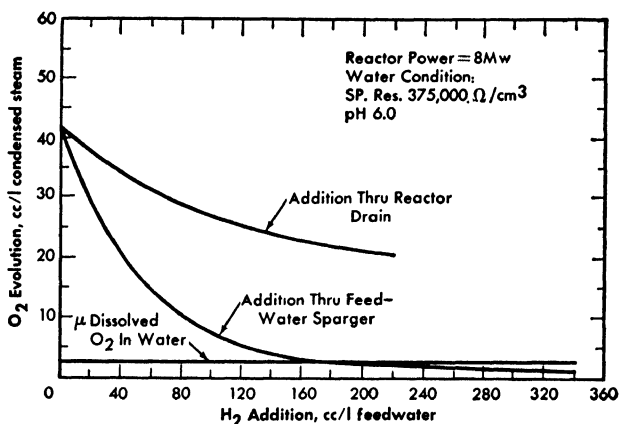


FIG. 3-45. Oxygen evolution vs. hydrogen addition.

because the former position gave greater constant time between the hydrogen and the water for dispersion and solution of the hydrogen.

The addition of hydrogen as a means of decreasing the oxygen content of the steam was the first attempt to date at solving the oxygen problem. It does not represent a recommended procedure as it was used. The quantity of hydrogen involved represents a sizeable quantity of gas to handle in the exhaust system and would probably be uneconomical.

Water-aluminum corrosion. Aluminum clad fuel was used in the BORAX experiments since it was expedient to do so, as indicated earlier in this chapter. The history of the aluminum in contact with the hot boiling water is significant in that a single fuel loading was used over the nine-month test period. Water conditions, particularly from the stand-points of temperature and pH were severe. It is estimated that the fuel was in boiling water at 420 F (215 C) for approximately 1170 hours. It is further estimated that during at least 100 hours the pH water was over 8. In addition to operations under boiling conditions, the fuel was in relatively stagnant water over 300 F (150 C) for 100-200 hours before and after the test runs.

In the later stages of the BORAX-III operation, analyses of the gases from the condenser air ejector indicated the possibility of a break in the fuel plate cladding. Numerous analyses were made of contaminants in the air ejector gas and in the reactor water, and of solids collected on the filters. A⁴¹, Kr⁸⁵, and Xe¹³⁵ were identified in the air ejector gases. In some of the samples the ratios of these gases were such that air leakage into the system does not explain their presence. An analysis of a sample of reactor water identified Sr⁸⁹, Ba¹⁴⁰, La¹⁴⁰, and Mo⁹⁹. The gaseous

fission products or the daughters of gaseous fission products, such as Sr^{89} , Ba^{140} , or La^{140} , might be found outside the fuel because of diffusion through thin sections of cladding metal. Mo^{99} is not in this class and its presence leads one to suspect a cladding failure. In a system having as many materials of construction as did the BORAX-III, the molybdenum could have been picked up as a corrosion product and activated in the reactor core. Recent analysis of material from the cleanup circuit pre-filter showed the presence of the fission products Zr^{95} , Ru^{103} , and Ru^{106} ; uranium was also finally detected. Previous analyses of reactor water had not detected uranium. The alpha count of the reactor water and the filter cartridges was quite low. In addition, a rupture of the cladding metal should release Xe^{133} . This was not detected.

Reactor control. As described earlier, the BORAX-III reactor had four boron-hafnium and one cadmium control rods. At the startup of the tests, the core loading had sufficient excess reactivity to accommodate going from room temperature to full power and saturated xenon. This level of excess reactivity exceeded the reactivity contained in all rods. Therefore it was necessary to shim the reactor by the addition of boric acid into the reactor water. As the reactivity was lost due to temperature effect and the accumulation of xenon, boric acid was removed with strong ion exchange resins. As the reactor operation period was extended, the accumulation of fission products other than xenon and the burnup of U^{235} began to exert an appreciable effect to reduce the excess reactivity. Eventually, the excess reactivity in the core reached a level which was at all times within the limits of the control rods.

The use of boric acid shim control was very successful. The removal of boric acid from the water was accomplished at a satisfactory rate with the strong anion exchange resins. No evidence was found in the operations that significant quantities of boric acid deposit in an insoluble form in the reactor core.

In addition to the boric acid solution control, each BORAX-III fuel subassembly had control in the form of a boron *burnout strip*. The burnout strip, which was spot welded on one outside surface of the subassembly, was 0.020 in. thick, 3.9 in. wide, and 25 in. long. The composition of the strip was 98.4 w/o Al, 0.6 w/o B, and 1.0 w/o Ni.

The burnout strips were used to assure long-term reactor operations by providing a poison that would burn out at a rate partially compensating for the burnup of U^{235} . The advantage of adding the burnable poison in the form and manner used for BORAX-III rather than to incorporate the poison in the fuel alloy is that the burnout strips can be mechanically removed any time this is desirable.

A third use of boron control was used for emergency shutdown. A tank containing a saturated solution of boric acid was located above the reactor and was heated and pressurized within reactor steam. The solution dump line had a steam trap between the scram valve and the reactor to prevent any boric acid solution leakage into the reactor vessel. On a scram signal initiated by the operator, the scram valve opens and the valve in the line to the trap closes, thus allowing the boric acid to drop into the reactor vessel.

BORAX-IV

In 1956, BORAX-IV was designed and its operational program was undertaken.^{11,12} There were two immediate reasons for its existence. The first was to test the suitability of a new type of fuel element developed at the Argonne National Laboratory, utilizing pellets of fuel consisting of a mixture of uranium oxide, UO_2 , and thorium oxide, ThO_2 . These pellets were fired in air. The fuel elements were a mechanical assembly of fuel and cladding. Instead of a metallurgical bond between fuel and cladding, the bond is made merely by filling the void with lead. The second reason was that, since this type of fuel element had a long thermal time constant [approximately 0.9 sec as compared with 0.006 sec for the BORAX-III reactor and 0.8 for the Experimental Boiling Water Reactor (EBWR)], its thermodynamic behavior would be similar to EBWR and hence would give results which would be directly applicable to EBWR. BORAX-I, II, and III had provided virtually all the existing knowledge pertaining to the safety and operating characteristics of boiling reactors. It was realized that the thin aluminum fuel elements of these three reactors would not be typical of boiling power reactors, for it was expected that the latter's fuel elements would have appreciably longer thermal time constants. Other reasons having a bearing on the choice of the BORAX-IV fuel elements were the chemical inertness of an oxide fuel in water, the metallurgical simplicity and cheapness of aluminum tube sheets loaded with oxide pellets, and potential economic advantages associated with thermal conversion of thorium into U^{233} .

Thus, early in 1956, it was decided to design and fabricate a thorium-uranium core for the BORAX facility. As this was just a core change on BORAX-III, the system remains characterized by natural circulation, direct cycle, boiling light water with 2 Mw electrical power capability. The simplicity of the core is demonstrated by the design and fabrication taking place in less than a year. The design basis was conservative, the goal being to assume having a workable reactor suitable for intermittent experimental operation for a year or so. It should, therefore, be empha-

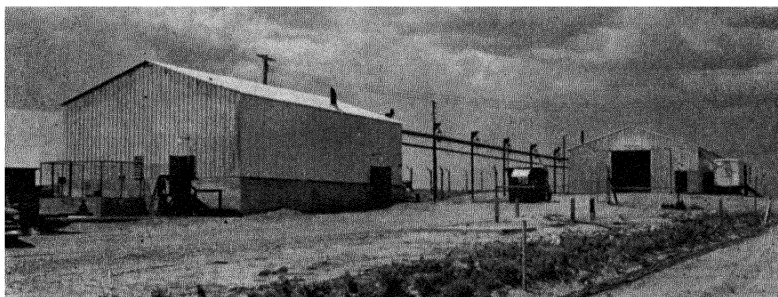


FIG. 3-46 Reactor and turbine buildings, BORAX-III and IV.

sized that this core is not necessarily an optimized prototype for a large reactor.

The experimental program was planned for the reactor with an emphasis toward obtaining more detailed information about the safety and stability of boiling reactors. For this reason a large variety of experiments were performed, with whatever emphasis on internal instrumentation and data collecting the limited budget of this relatively low-cost experiment would permit. *Ringing* and *rod oscillator* experiments were the first of their kind in boiling reactors. Also, incidental to 300 Mw days of experimental operation during 1957, a quantity of U^{233} was generated, the first for electricity-generating nuclear power plants.

Description of facility. The test facility for the BORAX-IV experiments uses the identical plant and components used in BORAX-III. Thus the same reactor building, turbine, generator, reactor pit, pressure vessel, rod drives, instrumentation, control console, and auxiliary equipment were used, the only major change being in the fuel assemblies. Since BORAX-IV test facility is the same used for BORAX-III, the residual radioactivity, due to high power runs in BORAX-III less than a year before, hindered experimental operation of BORAX-IV. In addition to this, general deterioration of components, including control rod drives and instrument cables built originally for only short term use, caused considerable difficulties.

Figure 3-46 is a photograph showing the reactor and turbine buildings. The instrument cables shown in the foreground of this photograph lead to the control trailer located $\frac{1}{2}$ mile away. A drawing of the interior of the reactor building showing the relationship between the reactor pressure vessel, control rod drives, and auxiliary equipment is given in Fig. 3-47. The flow diagram is the same as that shown in Fig. 3-34 for BORAX-III.

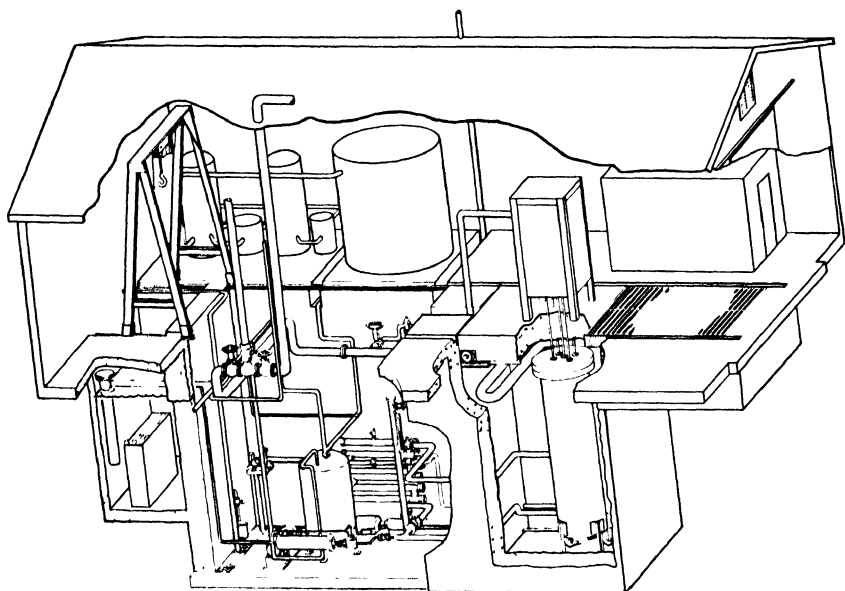


FIG. 3-47. Drawing of interior of building showing location of equipment.

Fuel assembly. The feature which distinguishes BORAX-IV from the other BORAX reactors is the fuel assembly design. This fuel assembly, developed by the Metallurgy Division of Argonne, utilizes pellets of thorium-urania ($\text{ThO}_2\text{-UO}_2$) fired in air. The uranium is enriched in U^{235} to a greater than 90 per cent value. These fuel pellets were inserted into aluminum tube sheets with a mechanical bond made by filling in the spaces with lead. Figure 3-49 is an end view of a completed assembly showing 6 extruded aluminum plates each containing 8 fuel channels. There are 47 active channels with one channel being available for the insertion of a boron-steel rod. The completed fuel assembly with riser and the bottom fitting is shown in Fig. 3-48.

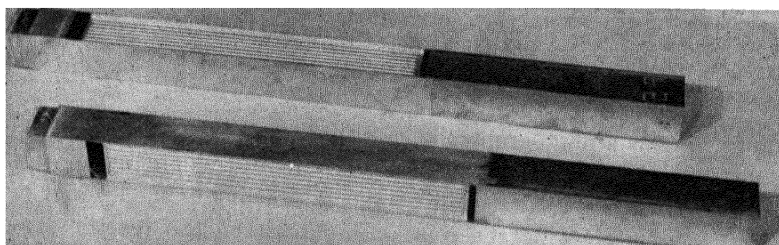


FIG. 3-48. Fuel element.

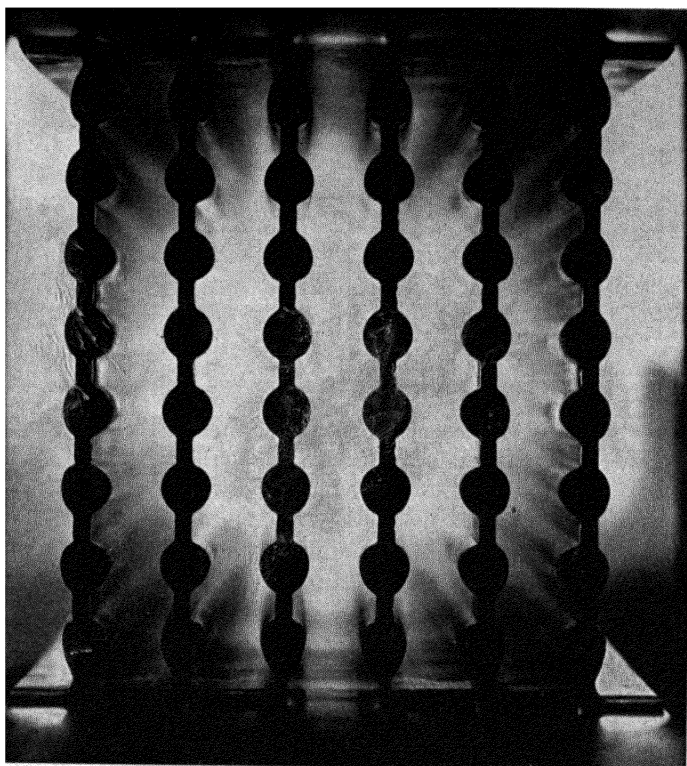


Fig. 3-49. End view of the fuel element, showing the six extruded aluminum plates in the assembly.

Pertinent nominal dimensions of the fuel element and other data are as follows:

Over-all length	56 6 in.
Active fuel section	24 in.
Cross section	3 875 by 3 828 in. (Identical with BORAX-III)
Fuel plate spacing	0 604 in.
Thickness of web between fuel channels	0 0875 in.
Side plate thickness	0 051 in.
OD of fuel channel	0 2975 in.
ID of fuel channel, including ribs	0 256 in.
Number of ribs per channel	6
Height of internal ribs (six in channel)	0.005 in.
Effective cladding thickness	0.016 in.
Effective thickness of lead sealer	0.018 in.
Cladding and structural material, aluminum	1 w/o nickel

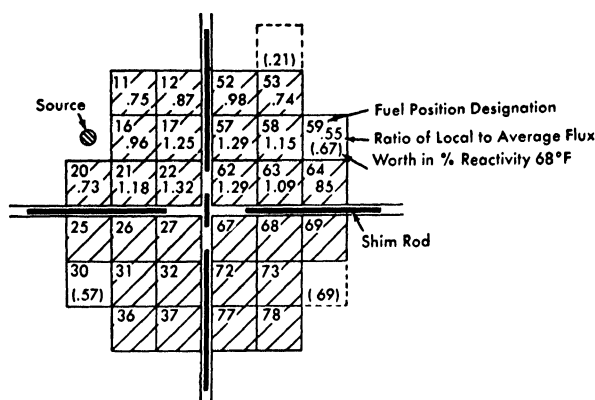


FIG. 3-50. Flux distribution at the vertical center of the fuel elements (30-element core). The cross-hatched area indicates critical loading with all rods withdrawn.

The fuel pellets are 0.230 in OD and vary in length from about $\frac{3}{8}$ to $\frac{3}{4}$ in. The pellet density is about 9.1 g/cc and contains 0.0559 g of U^{235} per gram of ThO_2 . Each assembly contains about 5425 g of ThO_2-UO_2 , or about 283 g of U^{235} . The active portion of the fuel assembly is about 24 in. in length.

As in the case of the other BORAX reactors, the control blades operate in channels which divide the core into quadrants. The grid plate is capable of accepting up to 88 fuel assemblies, each with cell dimensions of 3.888 by 4.000 in. These are held down by a hold-down grid.

During atmospheric tests the reactor vessel lid was off, and steam escaped into the building. There was no shielding over the top of the core. Because of this, particularly during transient experiments, the control was located in a trailer $\frac{1}{2}$ mile away.

Zero power operation. The reactor was first brought to criticality on December 3, 1956. The approach to criticality was made as fuel assemblies were added one at a time to the water-filled reactor pressure vessel. As loading progressed, the counting rates from two BF_3 counters were measured. This procedure was repeated after each fuel assembly addition, with repositioning of the neutron source as required, until criticality was reached. Criticality was obtained when the core had been built to 28 fuel assemblies (7.92 kg of U^{235}) in the pattern diagrammed in Fig. 3-50. With all the shim blades in their maximum out-position, this loading has an asymptotic period of 18 min. As more excess reactivity was required for the later experiments, fuel assemblies were added to the periphery of the core.

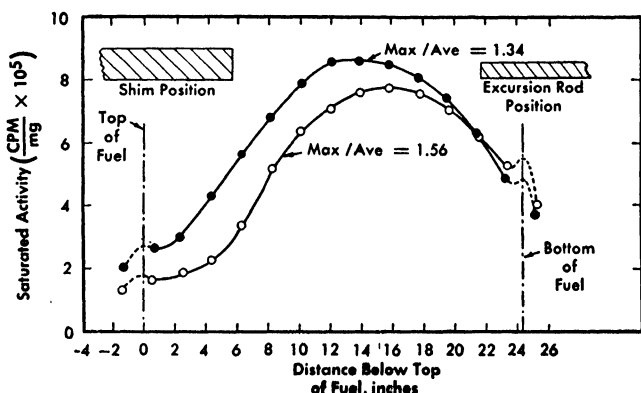


FIG. 3-51. Axial flux distribution at 68 F. Solid circles, fuel position 22. Open circles, fuel position 63. (Shims inserted 5.70 in.)

A differential worth calibration of the four shim blades was performed in the usual method of measuring the asymptotic period after withdrawing the shim blades in a *bank* a short distance from their critical position. The critical position was varied by adjustment of the center blade, by the addition of fuel assemblies to the periphery of the core, or by addition of boric acid to the water in the reactor.

The initial calibration of the shim blades was performed at room temperature (64 F) and at saturation (207 F) in a core consisting of between 29 and 30 fuel assemblies. At reactor pressures above atmospheric, the shim blades were calibrated in a core consisting of 59 fuel assemblies. Each of the center 16 assemblies contained a boron-steel poison rod. The shim blades were calibrated at various temperatures in the range from 100 to 420 F (300 psig).

Flux distribution. The flux distribution at room temperature (68 F) was measured for two core loadings by activations of bare gold foils located at various positions in the core. The gold foils were covered with aluminum foil to prevent contamination by impurities in the reactor water and were attached to foil holders which are suspended between the fuel plates. The initial neutron flux distribution measurements were made for a core consisting of 29 fuel elements as shown in Fig. 3-50. The relative local to average flux in the radial positions for this loading are given in the figure. The neutron flux distribution parallel to the core axis for two fuel positions is indicated in Fig. 3-51 in terms of the saturated foil activity. It is to be noted for this core that the excursion and shim blades are inserted partially in the core. The maximum to average flux for this loading is about 1.8.

The neutron distribution measurements were repeated when the core had built up to 56 fuel assemblies. The central 16 fuel assemblies in this core contained a boron-steel rod, and the water was poisoned by boric acid.

Power calibrations. Power calibrations of the nucleonic instruments were made by various methods depending on the power level. During the initial startup of the reactor, measurements of the power level were made by activation of gold foils and from the response of an absolute fission counter. For the low power calibration of the reactor (i.e., watts), power was taken above the source level and both bare and cadmium covered gold foils were activated for a known period of time. The count rate on the absolute fission counter was also taken. The gold foils were counted on a calibrated counter. The relative average thermal neutron flux between the absolute fission counter and the core was measured by gold foils. From these measurements and the U^{235} content in the core and fission counter, the power level was calculated. The power measurements by these two methods agreed to within 40 per cent.

More reliable power calibrations were made in the kilowatt range by comparison of heating rates between the reactor and electrical heaters. With the reactor shut down, the heating and cooling rates of the reactor water was measured with the electrical heaters (55 kw) turned on and then off. The reactor alone was then taken to about the same power level as the electrical heaters had been, and the same types of measurements were repeated. From the comparison of the total heating and cooling rates from the two power sources, the power calibration of the nucleonic instruments was made. During both measurements, the reactor water was continually circulated through the electric heater circuit and the reactor tank. The power calibration by this method was used for the transient test measurements.

During the steady state boiling operations at atmospheric pressure, the reactor power level was calibrated by the rate of change in the water level with zero feedwater flow, or by the measurement of the feedwater flow required to attain a constant water level.

For the pressurized runs, the reactor power was calibrated by using the flow obtained by the measurement of pressure drop across an orifice delivering steam either to the turbine or to the atmosphere. The feedwater flow required to maintain constant water level served as a check on this method.

Transient behavior. The experimental procedure used in making the transient experiments was essentially the same as in the case of the earlier BORAX experiments. The response of the reactor power and tempera-

tures to stop inputs of reactivity was investigated to determine the self limiting characteristic of the reactor over a range of initial periods from a few mil sec to slightly below 100 msec. It was not intended to completely and exhaustively cover the range of interest for this reactor, but rather that the transient behavior be explored so that operation at steady power could proceed with greater confidence. The transient behavior at atmospheric pressure to stop-inputs of reactivity was investigated with the reactor water initially at saturation (207 F at 5000-ft elevation) and later subcooled (120 to 160 F).

The positive reactivity step was achieved by ejecting a spring-loaded central control rod downward out of the core.

A series of excursions were made with the reactor at boiling temperature. For these excursions there were 30 elements in the core.

Unfortunately, during the first excursion there were no thermocouples available. However, during the later runs thermocouples, both fuel and surface, were installed.

In order to detect, if possible, the approach of damage to the core as the period of the excursion was reduced, special rods containing fuel pellets enriched to two and four times the normal enrichment were inserted in the core. These so-called "bayonets" were visually inspected after any excursion whose period was shorter than that of any previous run. The bayonet was essentially one fuel channel and was inserted in the water channel between two fuel plates. Since the heat generation in a pellet is approximately proportional to the enrichment, the four-times-normal enriched pellets placed in the position of maximum flux would certainly give warning by reason of noticeable damage before any real damage was suffered by the core. In none of these excursions, down to periods as low as 83 msec, was there any detectable external damage.

A number of these bayonets were removed, and the cladding and lead fillers were stripped off. Although some of the pellets were cracked and fell to pieces, there were no evident trends of damage attributable to enrichment, to exposure, or to period. The nearly unirradiated pellets seemed to be no better and no worse than the irradiated pellets. Figure 3-52 depicts a typical excursion from saturation. In this case the period was about 109 msec.

Figure 3-53 indicates the peak powers reached in an excursion vs. the reciprocal period. There was only one excursion with a period less than 100 msec, and that particular excursion indicates that a sharp break in the curve has been reached. Also included are curves for subcooled excursions and comparisons with SPERT-I and BORAX-I.

Figure 3-54 shows the energy released up to the time of peak power. It is interesting to note in the range of periods covered that, as the period

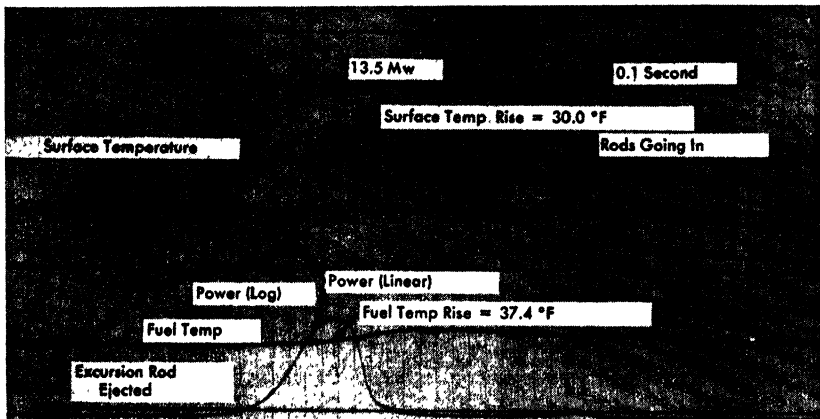


FIG. 3-52. Excursions from saturation (207 F). Run No. 8, Feb. 4, 1957.

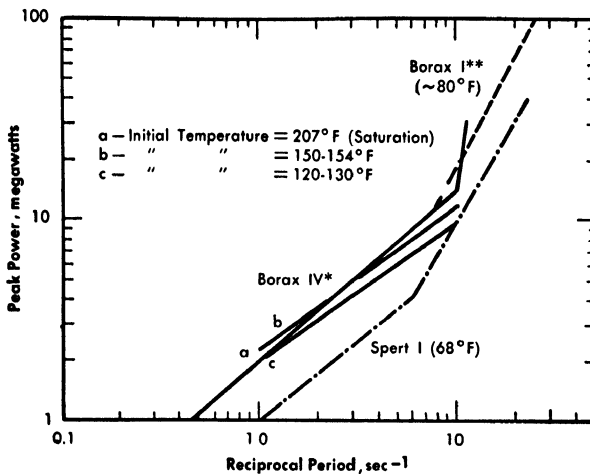


FIG. 3-53. Peak power vs. reciprocal period for BORAX-I, BORAX-IV, and SPERT-I. The single asterisk means that the data include delayed energy, and the double asterisk means prompt power only.

decreased, the peak power increased but the energy decreased. Both peak power and energy are considerably higher than those reported for SPERT-I in this range.

Surface thermocouples were installed on the bayonets by spot welding 24-gauge wire to the cladding. The thermocouples were chromel-alumel. Figure 3-55 shows the surface temperature rise of the thermocouples near the core center during excursions from 207 F.

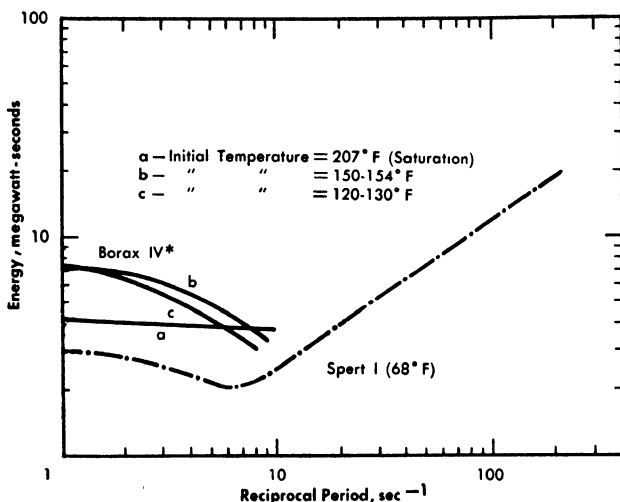


FIG. 3-54. Energy up to time of peak power vs. reciprocal period for BORAX-IV and SPERT-I. The asterisk means that the data include delayed energy.

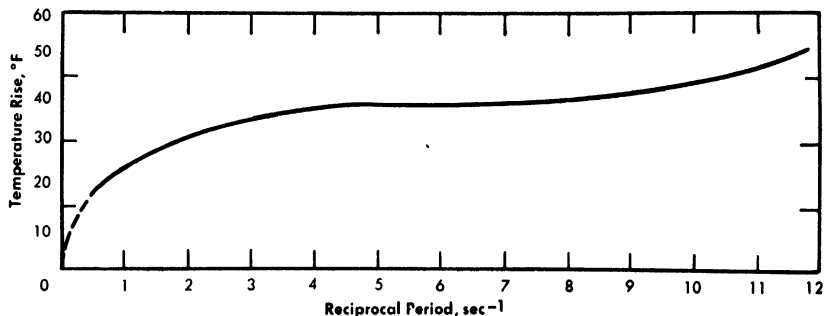


FIG. 3-55. Maximum surface temperature rise at peak power vs. reciprocal period in various excursions from saturation.

Subcooled excursions. It was decided to investigate excursions with the water initially subcooled, although this was not part of the original program. This was suggested since there was reason to suspect that a shutdown mechanism other than boiling was occurring in subcooled excursions in SPERT-I. It was thought that possibly, because of the long time constant of the fuel in BORAX-IV, it could be demonstrated that shutdown would occur before the cladding surface had reached the boiling point of water.

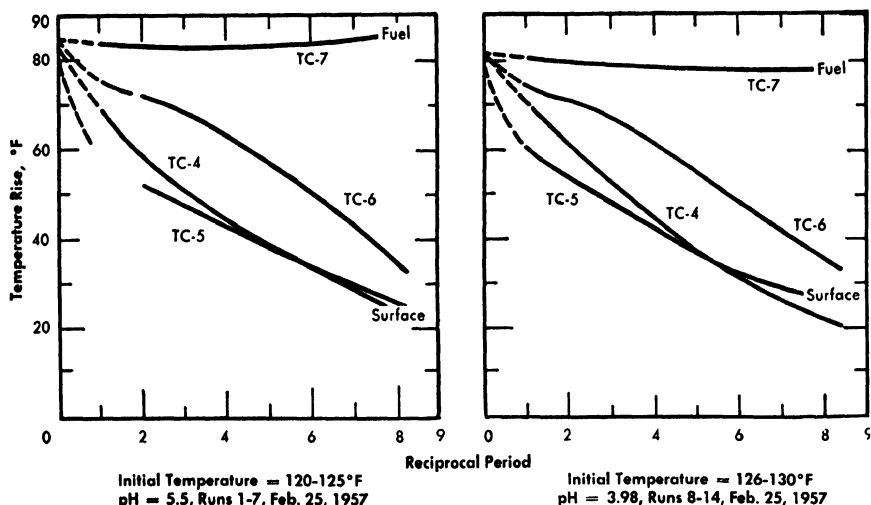


FIG. 3-56. Surface and fuel temperature rise at peak power vs. reciprocal period.

Figure 3-56 indicates centrally located surface-thermocouple rises at peak power for various periods in excursions from 120 to 130 F. It is to be noted that in all cases peak power was reached before the cladding surface reached the boiling point of the coolant.

A fuel thermocouple was installed in an oxide pellet in a bayonet by drilling a hole axially in the pellet, installing the thermocouple, and plugging with a plastic material. Since there is no bond between the thermocouple wire and the fuel material, the thermocouple readings may not be accurate, particularly for the transient tests. However, it was believed that inaccurate readings were better than no readings at all. Figure 3-56 also indicates the fuel-thermocouple readings at peak power.

Figure 3-57 presents the fuel- and surface-thermocouple temperature rises plotted against energy. Since, in this period range, energy increases with period (Fig. 3-54), the temperature difference between fuel and cladding increases as the period decreases, a result which is to be expected. A number of runs were analyzed to determine the reactivity behavior with time. Figure 3-58 indicates the relation of energy and reactivity loss to period. There are several points of interest:

1. Reactivity loss at peak power is proportional to energy release.
2. Reactivity loss at peak power is in turn proportional to surface temperature.
3. The fraction of the input reactivity compensated at peak power decreases as the period decreases.

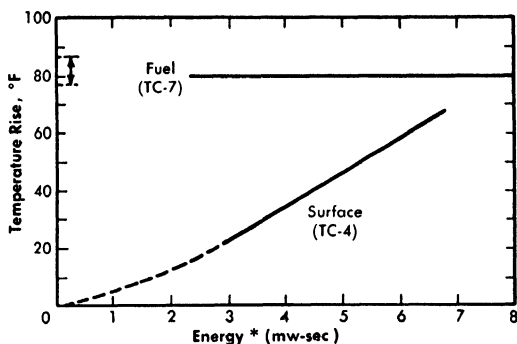


FIG. 3-57. Fuel and surface temperature rise vs. total energy liberated in subcooled excursions. Asterisk means delayed energy included.

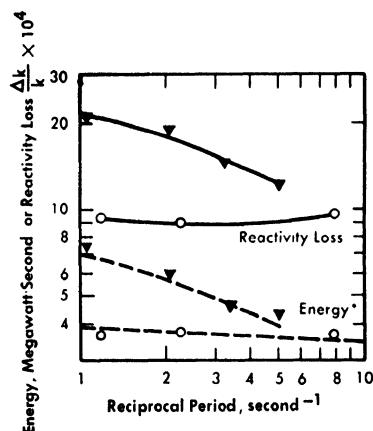


FIG. 3-58. Energy and reactivity loss to peak power vs. reciprocal period. Triangles, subcooled excursions (120 F). Circles, excursions from saturation (207 F). — data for all runs. Asterisk means that delayed energy is not included.

Since it has been demonstrated that surface boiling is not the shutdown mechanism in the subcooled excursions, other explanations for the loss of reactivity have been investigated. Preliminary calculations indicate that the combined effects of temperature coefficient, Doppler effect, and radiolytic gas formation account for only one-third of the reactivity loss. Further analysis is being done. It is possible that the nonuniform nature of the above effects, plus a better estimate of the void coefficient of reactivity, plus a testing of the fuel-thermocouple response, will help to close the gap.

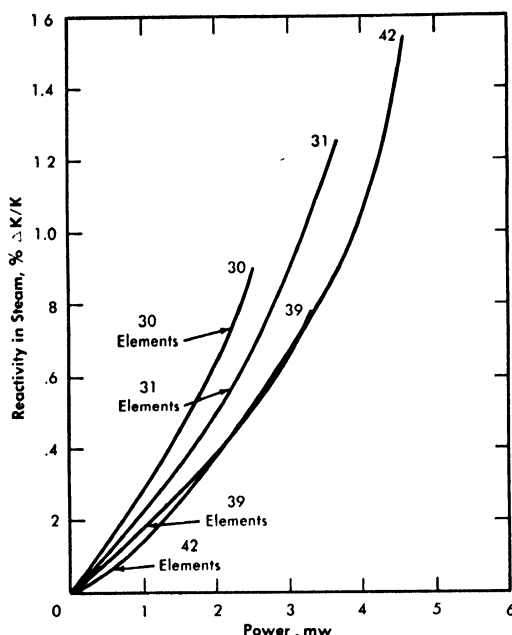


FIG. 3-59. Reactivity in steam vs. power for various core loadings.

Boiling operation—atmospheric. *Steady state.* After the excursions were completed as far as was practicable, the reactor was run at steady boiling at various power levels. Figure 3-59 indicates the reactivity represented by the steam voids vs. power level. The highest power attained was 4.6 Mw. The power level was obtained (1) by timing the rate at which the water was boiled from the core for the lower power runs and (2) by measuring the rate at which feed water was added to maintain a constant water height for the power runs. The core with 39 and 42 elements will be described later.

Figure 3-60 plots the data on Δk in voids vs. power density. Here it is seen that power densities as high as 30 kw/liter of coolant were achieved at atmospheric pressure.

Figure 3-61 indicates the fuel temperature rise above saturation for various power levels. The variation in temperature during each run is indicated by the limits on the data points.

The effect of water height on power density for several runs of different loadings is shown in Fig. 3-62. It appears that water heights greater than about 4 ft above the top of the fuel have little effect on the power output. However, as the water level decreases from 4 to 2 ft (the height of the risers), the power decreases rapidly.

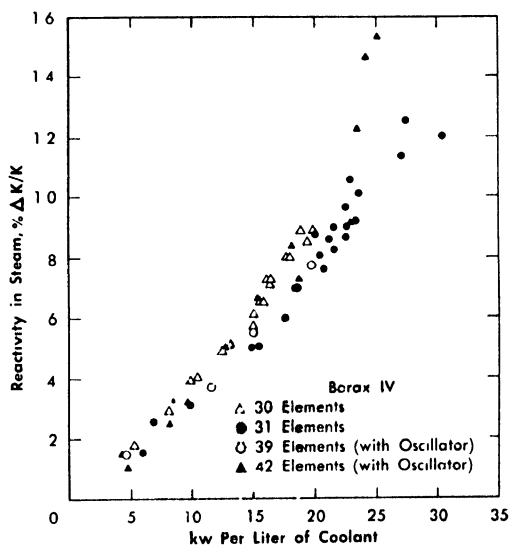


FIG. 3-60. Reactivity in steam vs. power density.

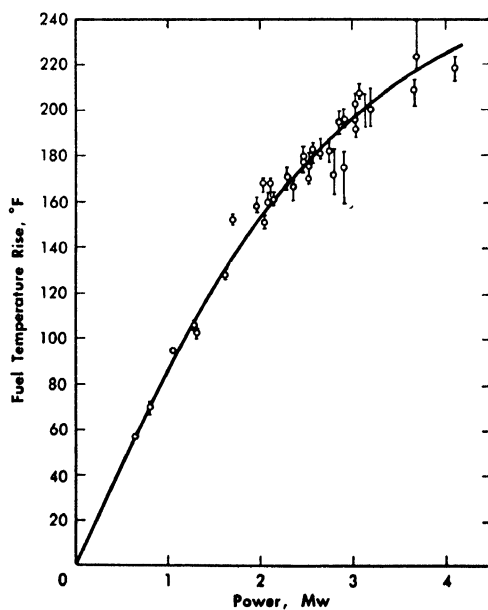


FIG. 3-61. Fuel temperature rise vs. steady-state power level (30 and 31 elements).

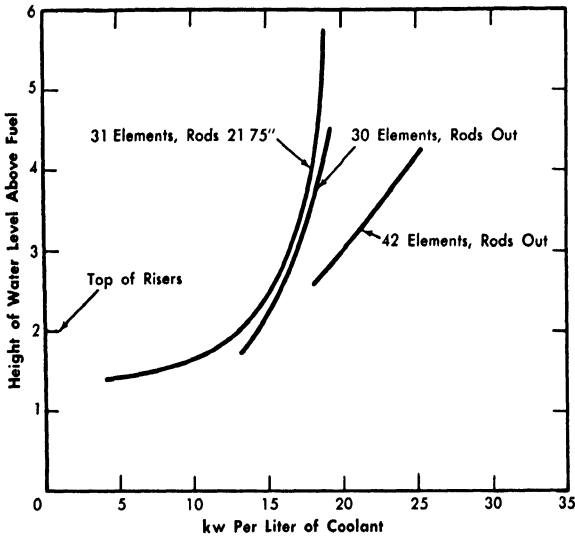


FIG. 3-62. Power density vs. water height.

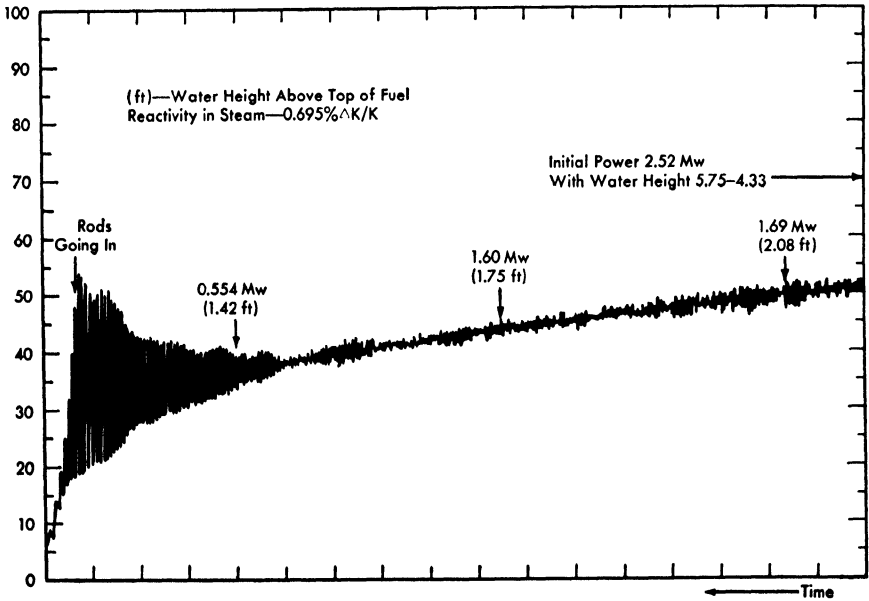


FIG. 3-63. Boiling off the top reflector (Feb. 27, 1957).

Figure 3-63 is a trace of the linear instrument recording of the power behavior as the top reflector was boiled off. When the level fell below the top of the riser, the reactor was performing like a "teakettle," and its operation became quite unstable.

Table 3-4 is a comparison of various characteristics of the four BORAX reactors. Most of the information on BORAX-I, II, and III was reported in the 1955 Geneva Conference.⁵ The table also includes some recent data on BORAX-IV at 300 psi.

Stability. The problem of stability is of great concern in the operation of boiling reactors and it would be of particular benefit to be able to predict the conditions under which a boiling reactor will become unstable. To determine such conditions, transfer function measurements were made using a sinusoidally reciprocating cylindrical control rod. This was a hollow cadmium tube with water inside as well as outside and was capable of a 9-in. stroke. Since it replaced a fuel element, thus introducing a water hole in the reactor, the characteristics of the loadings used in these oscillation experiments were not the same as those of the smaller cores. For the same power density, however, the reactivity in voids was about the same in all atmospheric pressure cores. Figure 3-64 shows a typical core loading for the rod oscillator.

Although reactors had been oscillated before, this was the first time that such experiments were done on boiling reactors. The oscillator experiments are a major step in obtaining an understanding of boiling reactor self-oscillatory tendencies. A sharp resonance was found in the transfer function in the region of the self-oscillatory frequency. Experience with the rod oscillator technique on this reactor, and subsequently on EBWR, has indicated that it may be a practical tool in obtaining information about stability safety at low powers without incurring the risk associated with large self-induced chugs at the unstable point at high powers. When a completely satisfactory theory of instability exists, its constants can be evaluated by low power transfer function measurements, and the theory then applied at higher powers.

The oscillator rod used in these tests consisted of a 2-in.-OD aluminum tube 0.040 in. thick. Attached to the inside of this tube was a cylinder of cadmium 0.032 in. thick and 12 in. long. The assembly was contained in a water-filled $2\frac{1}{16}$ -in.-ID aluminum thimble 0.0625 in. thick and was located as shown. The aluminum tube was connected to a drive capable of reciprocating the cadmium with frequencies from 0.1 to 2 cycles/sec. When the cadmium was in its uppermost position, its top was just at the midpoint of the core. The amount of reactivity oscillated was determined by the stroke. For most of the runs the amount of reactivity was ± 0.05 per cent k_{eff} . This was achieved at 207 F by a stroke of $4\frac{5}{8}$ in.

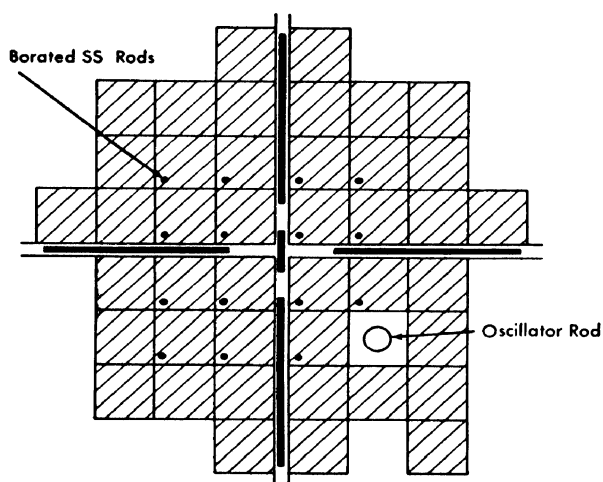


FIG. 3-64. Core loading for oscillator experiments (42 elements).

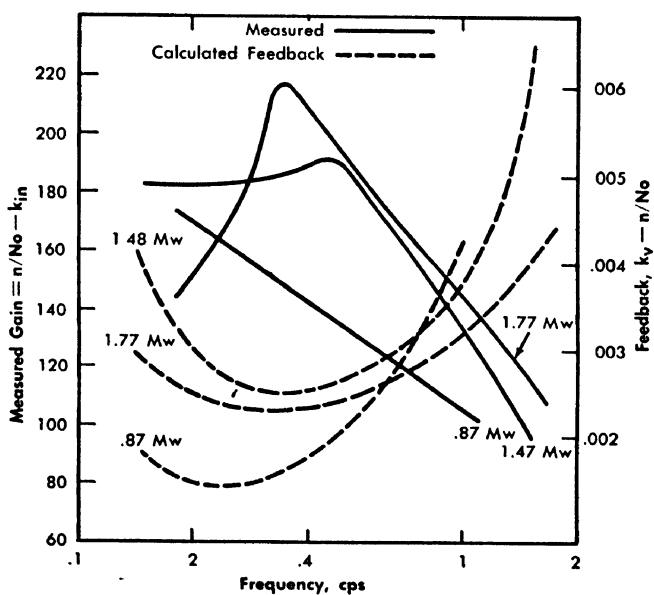


FIG. 3-65. Amplitudes of BORAX-IV transfer functions.

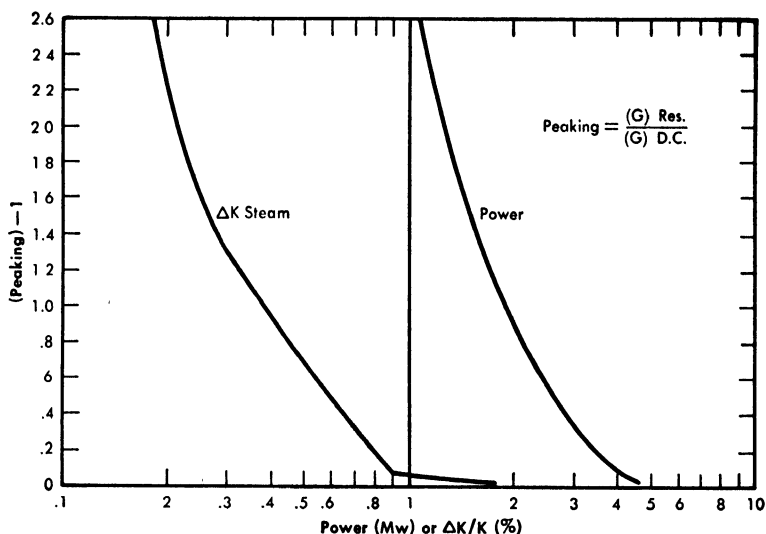


FIG. 3-66. Reciprocal peaking vs. power.

Borated stainless steel rods were placed in the empty fuel channels of 15 of the innermost elements. This was done to make the core size larger on the pressurized runs to follow.

It should be noted that this particular loading was far from symmetrical and, furthermore, had a large water hole due to the oscillator. This probably accounts for the observation that the power density for the 42-element core was less than for the 31-element core although the reactivity in steam was greater (see Fig. 3-60).

Frequency-response curves were obtained as a function of power level. The solid lines in Fig. 3-65 illustrate the response at three different power levels. Although there was considerable scatter (not shown) in the data, it was quite evident that the gain increased sharply near the resonant frequency. Also, the resonant frequency increased with power level. The maximum achieved was about 0.9 cycle/sec with a gain of 1000.

Plotting the reciprocal of the peaking, which is the gain at the resonant frequency divided by the DC gain (i.e., zero frequency gain), vs. power level and Δk in steam as in Fig. 3-66 gives some information about the power level (or Δk in steam) at which the reactor becomes unstable (infinite gain). This curve indicates that 4.7 Mw was the upper limit for the particular core arrangement employed.

Figure 3-67 indicates the power trace as recorded on the Heiland recorder for a run at 4.57 Mw with 1.535 per cent Δk in steam. This was the 42-element core with all rods withdrawn. Although the oscillator

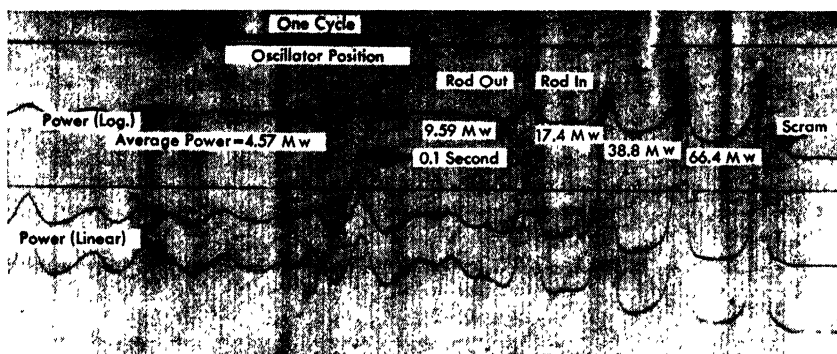


FIG. 3-67. Power trace with oscillator in operation. Water height above fuel is 4.25 ft. Number of fuel assemblies, 42. Reactivity in steam 1.535 per cent $\Delta k/k$.

was in operation, the reactor was oscillating on its own with a frequency of about 1 cycle/sec.

The operation was quite unstable under these conditions, causing a high power scram, and thus bearing out the predicted point of instability.

Operation of BORAX-IV under pressurized conditions (300 psi) has demonstrated that the reactor (containing 59 fuel elements) is quite stable up to 14.4 Mw and with as much as 5.5 per cent Δk in steam voids.

Ringing tests. This type of test was proposed by Bethe for fast reactors but is quite old. In its general form it consists in putting a sudden stop disturbance into a system and measuring the resulting response of the system. The name *ringing* is derived from the popularity of such tests in resonant acoustical systems, in which an audible ring results from a step input.

A number of central control rod ejections at various initial powers were done with the procedure being otherwise similar to that for excursions from essentially zero power. It was found from these tests that the damping of the disturbance decreases as the power increases. This is to be expected because the peak in the transfer function *sharpens* as the power increases.

OXIDE FUEL ELEMENTS FOR BORAX-IV

Since BORAX-IV used an entirely new and different kind of fuel element from those used in other boiling reactors, information about the characteristics and manufacture of the oxide fuel is presented. As indicated earlier, this type of fuel element was developed by the Metallurgy Division of Argonne National Laboratory.¹³

Thorium as a source material for U^{233} is of importance in the field of power reactors. Through the Th- U^{233} cycle, thorium can be converted to fissionable material.

Both thorium and uranium metals are susceptible to corrosion by water, and these metals must be carefully clad with a corrosion-resistant material when used in water-cooled reactors. The oxides, or the common corrosion products, of thorium and uranium are corrosion resistant to water, and these materials are more suitable for nuclear fuels.

Urania (UO_2) and thoria (ThO_2) have long been used as refractory materials for melting metals such as uranium. Ware made of UO_2 must be sintered in a vacuum or protective atmosphere to prevent oxidization of the UO_2 to U_3O_8 , whereas ware made of ThO_2 may be sintered in air since ThO_2 will not form a higher oxide. Both UO_2 and ThO_2 have face-centered cubic or fluoride type structures, and, as shown by Lambertson and others, they will form a complete series of solid solutions. This stability of ThO_2 , as well as its conversion to U^{233} , would indicate the possibility of a reactor fuel composed primarily of ThO_2 with a small amount of enriched UO_2 to cause it to become critical.

Preliminary considerations. The work of Wards and of Hoekstra indicated that U_3O_8 is stable in an oxidizing atmosphere up to about 1150 C. With prolonged heating above this temperature range, U_3O_8 decomposes to a lower oxide and, eventually, to UO_2 . In order to eliminate the use of a protective atmosphere in sintering UO_2 , and in order to avoid the disruptive volume change, during sintering, of UO_2 to U_3O_8 below the decomposition temperature of U_3O_8 , UO_2 - ThO_2 solid-solution compositions were prepared with UO_2 added as an equivalent amount of U_3O_8 .

Mixtures of U_3O_8 and ThO_2 containing up to 30 w/o U_3O_8 were dry pressed into small right cylinders $\frac{1}{4}$ in. in diameter. These compositions were fired to 1700 to 1750 C in air. The resultant ware appeared to be free of cracks or other visible defects and underwent no visible changes when reheated in air to 1400 C for periods as long as 96 hr. The geometric density of these samples ranged from 7.85 g/cc, for compositions containing $2\frac{1}{2}$ w/o UO_2 (added as an equivalent amount of U_3O_8) and $97\frac{1}{2}$ w/o ThO_2 , to 7.00 g/cc, for compositions containing 30 w/o UO_2 (added as an equivalent amount of U_3O_8) and 70 w/o ThO_2 .

X-ray diffraction analyses of samples showed diffraction patterns of the fluoride type which are characteristic of thoria-uranium solid solutions. No extraneous lines were observed even though a careful search was made for any U_3O_8 lines.

Specimens of two compositions were irradiated in both the Materials Testings Reactor (MTR) and the CP-5 reactor. The initial irradiation tests were made on a body having a composition of $97\frac{1}{2}$ w/o ThO_2 - $2\frac{1}{2}$ w/o UO_2 (added as an equivalent amount of enriched U_3O_8). The average apparent density of these specimens was found to be 9.27 g/cc, and the average apparent porosity was found to be 8.13 per cent. Since these were initial test specimens, the geometric density of the specimens was not determined. These pellets were encased in Zircaloy-2 and stainless-steel cans with the annulus filled with gas or NaK. Burnout in the CP-5 tests ranged from 0.02 to 0.05 a/o (total metal atoms). These tests were made in water autoclaves at temperatures of 500 to 520 F and pressures of 680 to 810 psi. Surface heat flux was estimated to be 6×10^4 Btu/(hr) (ft²). Burnout in the MTR ranged from 0.2 to 0.95 a/o (total metal atoms). These samples were subjected to approximately 50 scram cycles. A few of the specimens were irradiated bare or in cans in which a 0.020 in. hole had been drilled. Post-irradiation examination revealed that the CP-5 specimens were intact; however, a few had fragmented. The specimens from MTR appeared to be sound; however, microscopic examination revealed that a few cracks had developed in the samples. Two specimens irradiated bare (but packed in steel wool to prevent mechanical damage) were found to be completely free of visible damage. These specimens were irradiated to a burnout of 0.88 a/o of the total metal atoms. Other specimens irradiated in either Zircaloy-2 or stainless-steel cans were found to have developed some microscopic cracks but were intact.

Irradiation tests were also made on a body having a composition of 90 w/o ThO_2 -10 w/o UO_2 (added as an equivalent amount of enriched U_3O_8). The specimens submitted for irradiation in the CP-5 reactor were found to have an average apparent density of 9.2 g/cc and an average apparent porosity of 5.4 per cent. The specimens submitted for irradiation in the MTR were found to have an average apparent density of 10.06 g/cc and an apparent porosity of 1.3 per cent. These pellets were encased in aluminum-1 w/o nickel cans with the annulus filled with lead or gas. CP-5 burnouts ranged from 0.19 to 0.29 a/o of the total metal atoms. These specimens after irradiation were found to be intact but cracked.

MTR burnout ranged from 0.34 to 1.25 a/o (Table 3-7). Surface heat flux was estimated to be 27×10^4 to 102×10^4 Btu/(hr) (ft²); central metal temperatures were estimated to range from 1060 to 3900 C (melted).

Postirradiation examination revealed that dimensional changes, if any, were slight in the specimens in which the central metal temperature was estimated to be 2000 C or less. Some fragmentation has occurred in these specimens, but in general the condition of the slugs was good. Specimens

TABLE 3-7
EFFECTS OF IRRADIATION ON CERAMIC FUEL SPECIMENS*

Specimen No.	Type of bonding	Burnup, % metal	Fuel temp., °C (center)	Specific power, cal/(sec)(cm ³)	Heat flux, cal/cm ³
CA-1-1	Lead	0.34	1060	125	19.8
CA-1-2	Gas	0.34	1060	125	19.8
CA-2-1	Lead	0.64	2000	240	38.1
CA-2-2	Gas	0.64	2000	240	38.1
CA-3-1	Lead	1.10	3300	400	63.4
CA-3-2	Gas	1.10	3300	400	63.4
CA-4-1	Lead	1.25	3900	475	75.3
CA-4-2	Gas	1.25	3900	475	75.3

*All CA specimens were ThO₂-10 w/o UO₂. Gas bond consisted of welding gas (80% helium, 20% argon). All specimens were clad in aluminum-1% nickel.

irradiated to 1.25 a/o burnout appeared to fail in two ways. In the tests in which the annulus was filled with lead, the failure appeared to be due to gas pressure with subsequent hoop tensile-stress fracture. In the tests in which the annulus was filled with gas, the failure appeared to be due to melting of the cladding.

These preliminary tests indicated that a thorium-uranium solid-solution fuel would be feasible and that the ceramic material could be fabricated with little difficulty. The low thermal conductivity of the ceramic indicated a high central metal temperature, and in order to minimize this, the fuel design was established as 1/4-in. rods. This diameter of ceramic was considered the minimum size that could be easily dry pressed.

Fuel fabrication. Both the thorium and U₃O₈ were received as —325-mesh material. Prior experience had shown that the pressing characteristics of the thorium were improved by calcining overnight at 1000 C, and so all the thorium was calcined before mixing. The U₃O₈ was used as received without further treatment.

The pressing mixture for the BORAX-IV elements consisted of 93.41 w/o ThO₂ and 6.59 w/o U₃O₈. The U₃O₈ used was above 90 per cent enrichment, and for accountability this material was weighed in a dry box. One-gallon porcelain-lined pebble mills were used for mixing. The capacity of these mills was approximately 3000 g. To ease the problem of weighing the U₃O₈, approximately 300 g of the oxide was transferred to a tared bottle. The accurate weight was determined, and the amount of ThO₂ to be added was calculated and weighed in an open hood. The

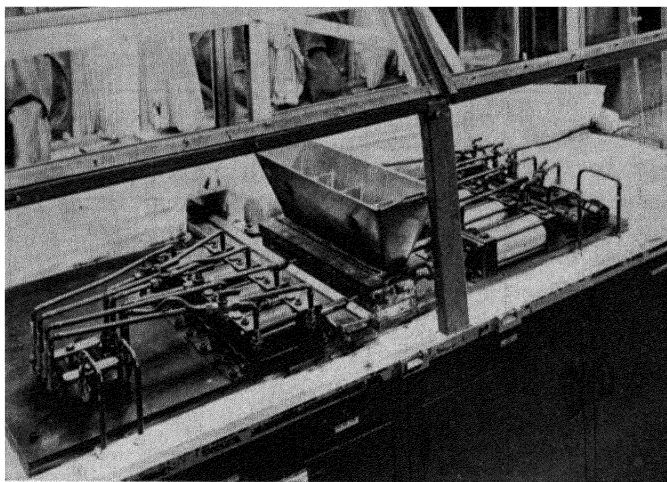


FIG. 3-68. View of press in glove box.

oxides, together with $2\frac{1}{2}$ w/o (of the batch) of polyvinyl alcohol, were charged into the mill in a glove box. Approximately 1 kg of porcelain pebbles was added, and the mill was closed and wrapped in polyethylene plastic sheeting. The mill was removed from the glove box and transferred to a mill rack, where the ThO_2 , UO_2 and polyvinyl alcohol were mixed together for 3 hr. The mill was then removed and returned to the glove box.

The dry oxides were transferred to a small Hobart mixer in a dry box, and the powder was dampened with 12 w/o water containing 0.5 w/o aerosol. The moist mixture was forced through a 16-mesh screen to form granules, and the granulated material was dried at 80°C for 4 hr. This granulation produced a free-flowing product; however, before pressing, 1 w/o of a mixture of 50 w/o kerosene and 50 w/o oleic acid was added as a lubricant. This material was charged to the press hopper shown in Fig. 3-68.

This press was constructed to fit the present glove box in the ceramic laboratory. It is essentially a double-acting press with four cavities and is capable of producing up to 80 pellets per minute. The diameter of the pressed pellets was controlled by the die size, and the length of the pellets was controlled by the free-flowing characteristics of the granulated material. These pellets were found to vary in length from 0.750 to 0.375 in. and in diameter from 0.267 to 0.370 in.

The $\text{ThO}_2\text{-U}_3\text{O}_8$ mixture was pressed at 14,000 psi, and the pressed pellets were randomly loaded into stamina crucibles. These crucibles were heated to 260°C overnight in an electric furnace to remove most of

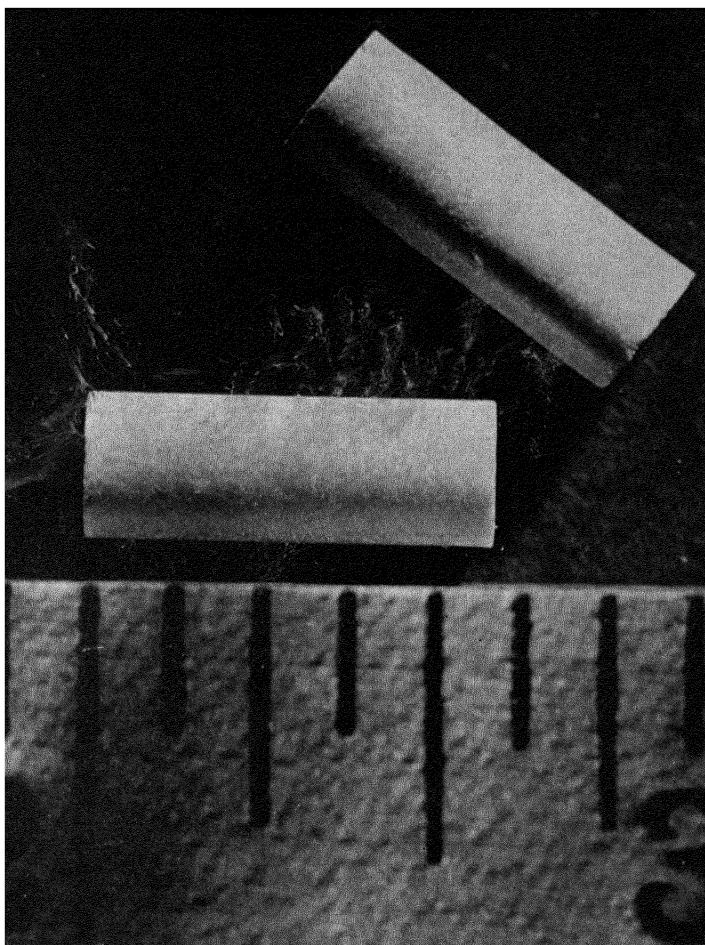


FIG. 3-69. Fired thoria pellets.

the organic binder. After calcination they were transferred to a gas-fired furnace, where they were fired to 1750 C in approximately 16 hr. The peak temperature was held for 2 hr, and the ware was allowed to cool in the furnace to room temperature.

The fired ware was found to have an average geometric density of 8.3 g/cc and an average apparent density of 9.8 g/cc. The diameters of the pellets were found to be 0.229 ± 0.002 in. This tolerance was such that it was not necessary to grind the pellets to size since 0.010 in. of lead was specified to fill the space between the pellet and the inside diameter of the tube. The fired ware in general was of uniform appearance (Fig.

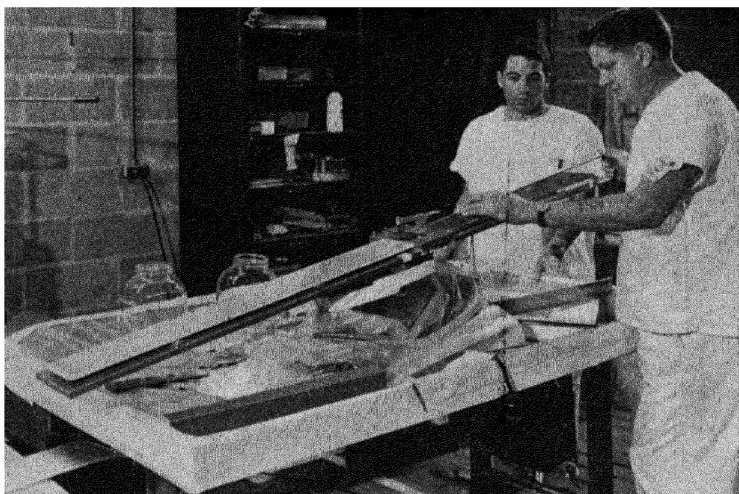


FIG. 3-70. Fixture for loading fuel pellets.

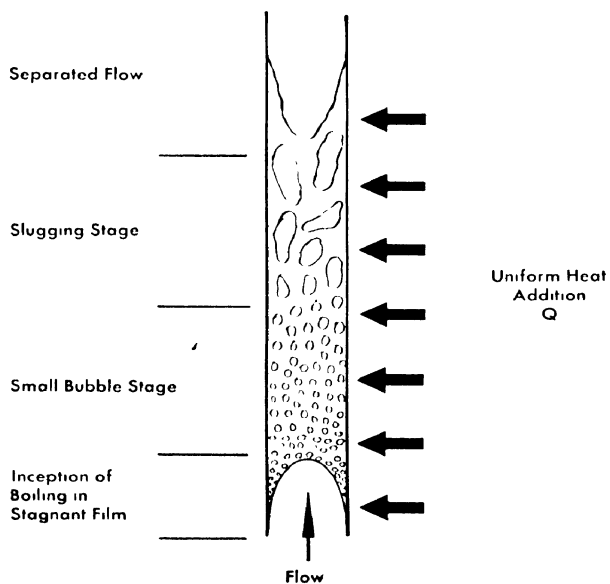


FIG. 3-71. Schematic of flow mechanism for boiling.

3-69), and rejects due to cracks, chips, or other imperfections averaged less than 4 per cent.

The fuel subassembly, as described earlier, consists of six tube plates, eight tubes per tube plate, spot welded into an assembly. The fuel charge consists of 24 in. of thoria-urania pellets lead bonded into place. After providing end closure the plates were tested using a helium mass spectrograph. The thoria-urania pellets were loaded into the tube plates with the aid of the fixture shown in Fig. 3-70. Eight columns of pellets 24 to $24\frac{3}{16}$ in. long were aligned in a grooved steel plate and then pushed into the aluminum tube plate. An aluminum plug was inserted as the last pellet, and a crimp mark was made just above the aluminum plug to prevent the charge from floating out during the subsequent lead filling.

Lead filling. Lead filling was performed by dipping the loaded tube plates into molten lead maintained at 490 to 500 C. The exterior of the tube plate was painted with a mixture of fine alumina suspended in an alcohol-glycerine carrier to limit lead adherence to the external surfaces. Since design requirements called for an unfilled volume above the aluminum pellet, it was necessary to remove the lead rod which was cast in the tube above the fuel charge. This was accomplished by inserting a $\frac{3}{16}$ in. diameter by 12-in. long stainless-steel cable into the tube-plate before lead dipping and by pulling the cable and lead out after the tube-plate had been lead filled.

After removal of the lead from above the aluminum pellet, the fuel plates were cut to size, and final end closure was made as before. Finally, a formed plate, along with the aluminum side plates, was assembled into a fixture and spot welded to the aluminum side plate by welding at four locations along the length. The clamps were removed, and the assembly was completed by welding at intervals of $\frac{7}{8}$ in. along the full length of both edges. Additional plates were added in a similar manner until the full six-plate subassembly was complete.

OTHER EXPERIMENTAL STUDIES ON BOILING REACTORS

Boiling density studies. Before the BORAX experiments and the studies described here, few data were available in the literature for the density of steam and water mixtures during boiling. Since the calculation of water flow with natural circulation boiling in vertical channels requires a knowledge of driving head or density difference between riser and down-comer, a test program was initiated to measure the density of steam water mixtures as a function of pressure and power density for natural circulation boiling systems. Table 3-8 lists chronologically the various apparatus and systems tested.¹⁴⁻¹⁷

Before describing some of these tests, it is of interest to examine the flow mechanism in a boiling system. A schematic diagram of such a flow mechanism is shown in Fig. 3-71. This is not necessarily correct but it will serve to illustrate the problem. In the diagram there is represented a channel with uniform heat addition at the walls. Water enters the channel with its bulk temperature below saturation. As heat is added, the water temperature rises to a point where it reaches saturation temperature. These bubbles will expand and coalesce upon rising. If the channels are sufficiently small, the bubble diameter will rapidly become large compared to that of the channel. When this occurs, a *slugging* mechanism of flow, wherein water is lifted on a conveyor of vapor for an interval of length, should come into effect. It is also possible that a region might be reached where the percentage steam generation is great enough so that the vapor no longer acts as a conveyor, but channels itself so that vapor and liquid flow simultaneously in separate channels having a common surface.

If it could be assumed that the steam and water velocities were identical at every point along the tube, then relationships could easily be developed to show the effect of pressure on density. However, the velocity of the steam is always different from that of the liquid due to buoyancy effects. As indicated earlier, the amount by which the steam velocity exceeds that of the water is known as the steam slip. If this effect were known as a function of the operating parameters, then the density of the boiling system could be predicted.

Before the flow in the system is known, the frictional pressure losses must be known. The pressure drop in the nonboiling region can be predicted by the Fanning Friction Equation.

The calculation of pressure drop in the net-boiling or two-phase region presents more difficulty. Although there has been appreciable work in the field of two-phase flow, no completely satisfactory design method has yet been developed. The most widely used methods are the Martinelli correlations and the friction factor methods.¹⁸ Although they can be used successfully in many cases, they are subject to serious criticism.

The table top apparatus indicated in Table 3-8 was extremely simple and was used to determine the effect of various additives on the natural circulation flow rate during boiling at atmospheric pressure. The results showed very little effect of additives on the flow rate.

The atmospheric boiler, the second item in the table, was used by Untermyer in his early studies at Argonne and is shown in Fig. 2-1. It requires no additional comment.

The next apparatus listed is the 260 psia multi-channel boiler shown diagrammatically in Fig. 3-72. The test section was essentially a direct-current resistance heater composed of seven parallel stainless-steel plates,

TABLE 3-8
NATURAL CIRCULATION BOILING SYSTEMS

Apparatus	System type*	Number of channels and geometry	Maximum power kw/liter	Maximum inlet velocity, ft/sec
Table top visual density study apparatus (atmospheric)	Open	3 annuli, $\frac{1}{16} \times \frac{1}{2}$ ID $\times$ 36 in. $\frac{1}{8} \times \frac{1}{2}$ ID $\times$ 36 in. $\frac{1}{4} \times \frac{1}{2}$ ID $\times$ 36 in.	60 (AC)	2.8
Atmospheric boiler	Open	4 to 20 rectangular, $\frac{1}{8}$ to $\frac{1}{2}$ in. $\times$ 5 $\times$ 9 to 24 in.	110 (DC)	
260-psia multichannel boiler	Closed	6 rectangular, $3\frac{7}{16} \times \frac{1}{2} \times 60$ in.	50 (DC)	3
260-psia single-channel boiler	Closed	Single, rectangular, $3\frac{7}{16} \times \frac{1}{2} \times 60$ in.	60 (DC)	3
600-psia multichannel boiler	Closed	3 rectangular, $\frac{7}{16} \times 4\frac{1}{4} \times 48$ in; 3 rectangular, $\frac{7}{16} \times 4\frac{1}{4} \times 24$ in.	50 (AC)	1 8
600-psia single-channel boiler	Open	Single, rectangular, $\frac{1}{2} \times 2 \times 60$ in.	50 (AC)	3 1

*Closed: contains internal condenser. Open: steam taken from system and makeup water added.

separated by $\frac{1}{2}$ -in. water channels. To insure continuity of flow within each channel from inlet to outlet, a teflon-lined stainless-steel shroud girded the active portion of the test section. The rectangular shroud in contact with the edges of the plates, formed end channels $\frac{1}{4}$ -in. wide, compared to the $\frac{1}{2}$ -in. channels between the plates. An external downcomer in the pressure vessel provided a return path from the test section outlet to the inlet.

Steam generated between the heated plates passed through a steam separator and on to the condenser. Water circulation flow rates were measured by injecting a small amount of cold water into the main stream and measuring the temperature decrease after complete mixing.

A series of wells for the density measurements was installed along the vertical height of the pressure vessel. Each set of wells, one for a gamma-

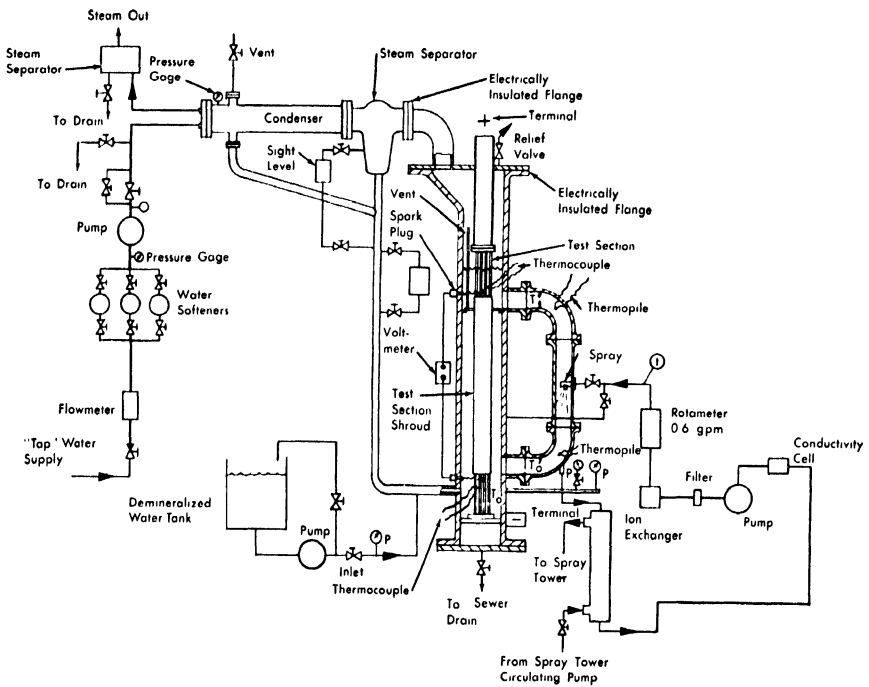


FIG. 3-72. 260-psi multichannel boiler test system.

ray source, and the other an in-line well for a photomultiplier tube, was spaced on 15-in. centers along the vessel. In this system as well as in subsequent systems, the following equations are satisfied:

$$\Sigma \Delta P = 0, \quad (3-1)$$

$$Q_t = W_t[(h_f - h_{in}) + xh_{fg}] = W_g[(h_f - h_0) + h_{fg}], \quad (3-2)$$

$$\rho = \rho_f(1 - \alpha) + \rho_g \alpha, \quad (3-3)$$

$$\frac{V_g}{V_f} = \left(\frac{x}{1-x} \right) \left(\frac{1-\alpha}{\alpha} \right) \left(\frac{\rho_f}{\rho_g} \right), \quad (3-4)$$

where ΔP = pressure drop, psi,
 Q = heat input, Btu/hr,
 W = flow rate, lb (mass)/hr,
 h = enthalpy, Btu/lb (mass),
 h_{fg} = latent heat of vaporization, Btu/lb (mass),
 x = steam flow weight fraction, lb steam/lb mixture,
 V = velocity, ft/sec,
 α = steam volume fraction, ft³ steam/ft³ mixture,
 ρ = density, lb (mass)/ft.

Subscripts: t = total flow or total heat,
 f = saturated liquid,
 in = inlet conditions,
 g = saturated vapor,
 0 = nonboiling section of channel.

Equation (3-1) is a pressure drop balance around the closed loop consisting of riser, downcomer, and cross-over pipes. Equation (3-2) is a heat balance. Equation (3-3) is the definition of local density in terms of steam volume fraction, α , and liquid and steam densities. Equation (3-4) is a mass balance at any plane perpendicular to the direction of flow.

The results of the multichannel tests are shown in Fig. 3-73. These data are for a pressure range of 25 to 260 psia, power range of 5 to 50 kw/liter, velocity range of 0.5 to 3.0 fps and steam volume fraction range of 0 to 0.90.

The scatter in the data of Fig. 3-73 was believed to be primarily due to (1) inability to accurately measure the flow rate with resulting errors in calculated slip ratios; and (2) scatter due to flow rate effect on slip ratio.

Other data from this system are shown in Fig. 3-74. These data gave the initial clue that 2-phase friction pressure drop might be dependent only on steam volume fraction.

At this point in the program, preliminary information was available at pressures up to 260 psia, but the data were such that extrapolation to 600 psia was impossible. The test program was then directed along two avenues of endeavor: (1) further refinements on the 260-psia boiler, and (2) design and construction of a 600 psia multiple channel test facility to extend the pressure range to EBWR conditions.

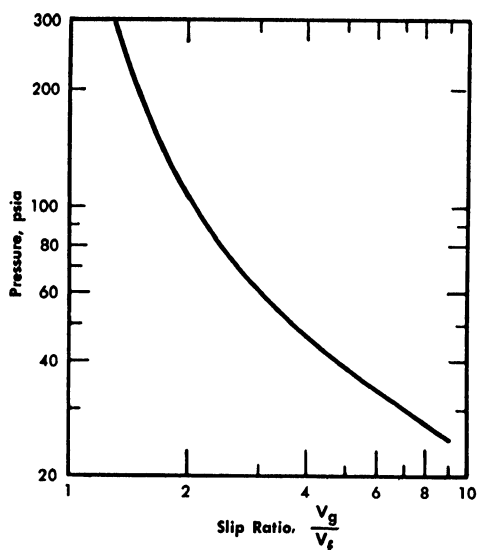


FIG. 3-73. Preliminary data showing effects of pressure on slip ratio.

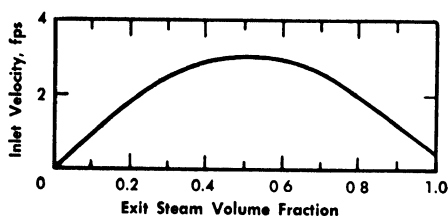


FIG. 3-74. Effect of exit voids and pressure on inlet velocity.

A single channel was substituted for the 7-plate assembly just described and the external piping modified so as to include a pump for extending the tests to forced circulation boiling. The results indicated that slip ratio was dependent upon pressure and mass flow rate.

To extend the pressure range, a 600 psia multichannel facility was constructed. The major improvement of this system over the 260 psia multichannel unit was the ability to make steam void measurements at any location along the channels, instead of measuring voids at three fixed locations.

Finally, in order to investigate more completely the various phases of boiling, the 600-psia single channel systems was designed and constructed in conformance with the following specifications:

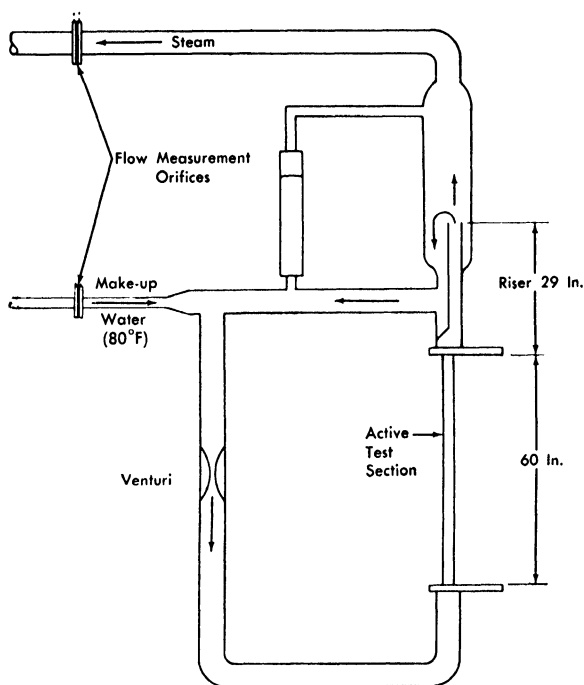


FIG. 3-75. Schematic of Armadillo test loop.

1. Unrestricted external system to permit relatively free natural circulation.
2. Accurate and positive flow rate measurement.
3. A small system which would permit rapid and inexpensive changes in test section, and also permit high power densities with and without forced circulation.

The system (Fig. 3-75) is designed to accommodate single channels up to 5 ft in length and 0.5 in. channel spacing. There is no internal condenser. Steam is discharged from the system and makeup water is added to simulate operation of an average reactor channel. The boiling length at any fixed pressure is constant, independent of the recirculation rate. The nonboiling length to boiling length ratio is equal to the ratio of sensible heat rise to latent heat of vaporization of each pound of water that passes through the channel.

Provisions were made to vary the resistance to flow in the downcomer circuit, and pressure taps are available for determining two-phase pressure drop in the heated channel. The system can handle up to about 400

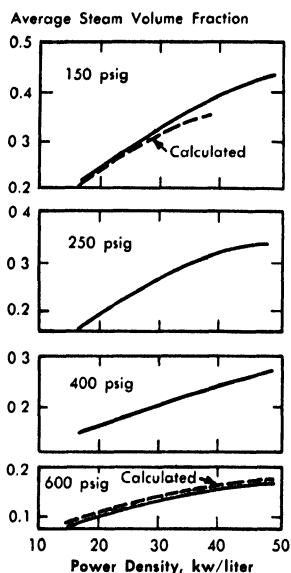


FIG. 3-76. Average steam volume fraction as a function of power density and pressure for $\frac{1}{2} \times 2 \times 60$ in. vertical heated channel.

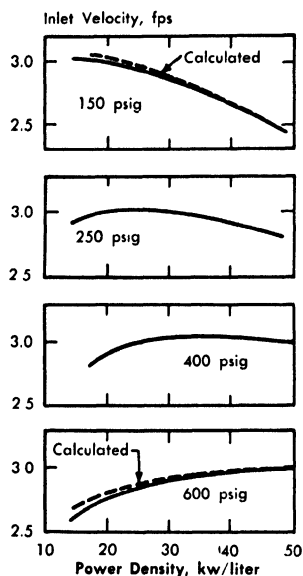


FIG. 3-77. Inlet velocity as a function of power density and pressure for $\frac{1}{2} \times 2 \times 60$ in. vertical heated channel.

kw/liter for $\frac{1}{8}$ in., 4 ft channels. Provisions are also made for forced circulation testing.

This system has a 29-in. chimney above the heated channel to prevent steam carryover into the crossover pipe and into the downcomer. Some of the results of tests made thus far with this system are shown in Figs. 3-76 and 3-77.

Also shown in Figs. 3-76 and 3-77 is a comparison between experimental and calculated performance of the 600-psia single-channel boiler.

LABORATORY TESTS ON RADIOACTIVE CARRYOVER

The problem of entrainment in processes such as steam generation, evaporation, and distillation has been studied for many years, and considerable effort has been expended in the design of de-entrainment equipment. The complexity of the problem can be surmised from the large number of factors which influence this effect. In a boiling reactor the boiling process is of considerable significance. The size distribution of drops above the boiling surface is determined in various degrees by: free or forced convection conditions, changing slip velocities, heat transfer

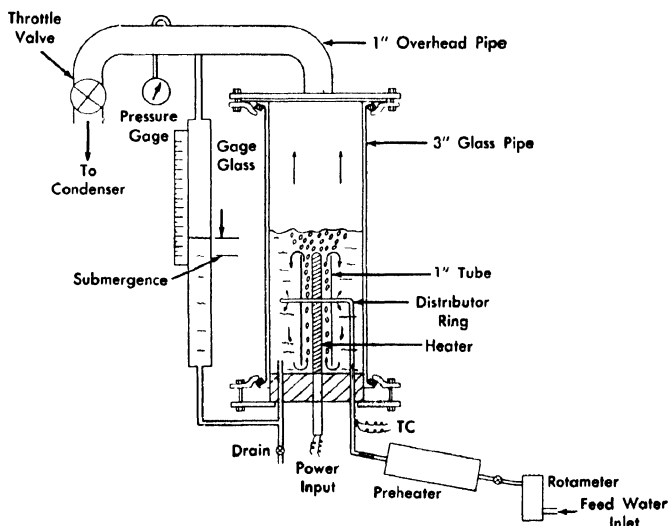


Fig. 3-78. Experimental apparatus for the study of entrainment obtained under natural circulation.

surface roughness and contamination, suspended and dissolved solids in the water and at the vapor-liquid interface, power level, power density, and operating pressure.

In view of the multiplicity of parameters that required study, an experimental approach was used which was based on relating the gross variables, such as steam velocity or power density to the entrainment. The initial experiments were performed on a small-scale 3-in. diameter natural circulation boiler with 1500 watts total heat input operating near atmospheric pressure. A second series of experiments was performed in a natural circulation boiling loop at pressures up to 600 psia at total power inputs up to 100 kw. Finally entrainment tests were made on the BORAX-III reactor, as described earlier.

Small scale experiments. Figure 3-78 shows the small scale natural circulation laboratory equipment used to measure entrainment at pressures near one atmosphere. Although no attempt was made to simulate any actual reactor fuel tube, power densities were of the anticipated order of magnitude.

Steam was generated in a 1-in. diameter vertical tube, 9 in. long, open at both ends. A 0.5-in. diameter quartz heater with 9-in. effective length ran axially along the length of the tube. Steam and water flowed up the annular space between the heater and the tube wall. After leaving the

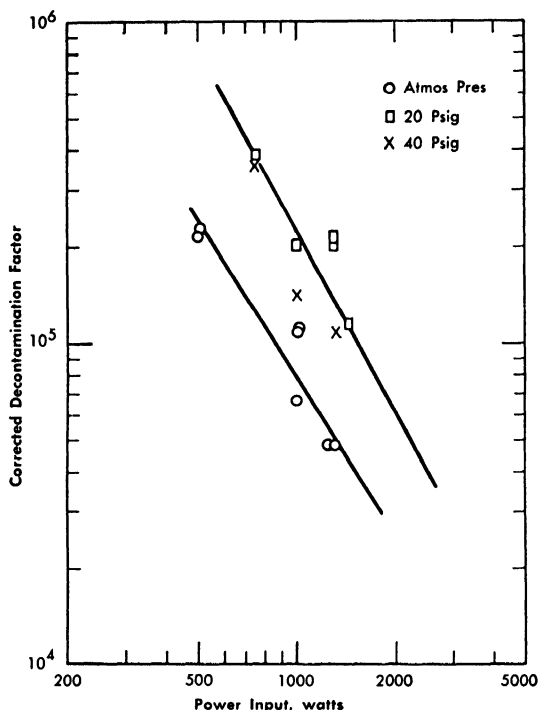


FIG. 3-79. Variation of decontamination factor with power input under natural convection. Equipment used described in Fig. 3-78.

1 in. tube, the steam rose through a layer of water 1 in. in height. Water entered the bottom of the 1 in. tube to replace that ejected at the top by the action of the steam. Thus, a circulation of water was continuously maintained in the annulus. Makeup water was added through a distributing ring centrally located in the main body of the fluid contained in the annular space between the 1 in. and 3 in. tubes. The rate of water addition was adjusted to maintain a constant submergence. Feed water (distilled water of 300,000 ohm-cm resistivity) was preheated to saturation temperature. Before the startup of a run the 3 in. tube was charged with sufficient distilled water containing tracer as to give the desired submergence. The submergence was observed and defined during the run in a sight glass connected to the 3 in. tube. Steam passing up the 3 in. tube was taken overhead through a throttle valve and condensed at atmospheric pressure.

Figure 3-79 shows data obtained at 0, 20, and 40 psig at various power inputs. The entrainment is expressed as decontamination factor, that is, the ratio of the activity concentration as counts per minute in the boiler

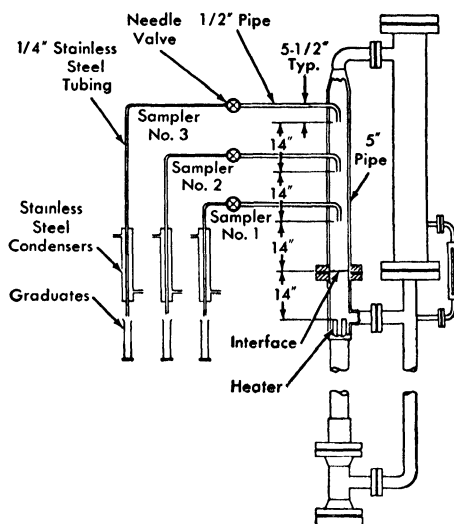


FIG. 3-80. Sampling section 600-psi loop.

liquid to the activity concentration in the condensate. There appears to be a small variation with pressure in this low-pressure region. These preliminary runs on a small scale unit indicate that decontamination factors of the order of 10^4 are possible at steam flow rates and heat fluxes comparable to reactor operation.

TESTS ON HIGH PRESSURE EQUIPMENT

In order to obtain entrainment data in equipment more closely approaching a reactor in scale and at higher pressures, a series of experiments was performed in a high-pressure test loop. Tests were run at pressures from 150 to 600 psia and at power inputs of 50 to 100 kw corresponding to 14 and 28 kw/liter, respectively.

The test apparatus shown in Fig. 3-80 consisted of a closed loop constructed of stainless-steel pipe sections. Water was electrically heated in the boiling section contained in a 5 ft vertical length of 5-in. pipe. Steam passed directly up a 5-in. riser pipe mounted above the boiling section. Three pitot tube-type samplers, facing into the direction of flow, were mounted in the riser section. These were spaced about 14 in. apart with the lowest one 14 in. above the boiling water interface. A condenser was located above the riser, and condensate was recirculated to the boiling section via a return line thus completing the loop. The sampler pitot tubes consisted of a section of $\frac{1}{2}$ -in. stainless steel pipe, 3 in. long, located

at the axis of the riser pipe. The flow of sample vapor was into the tube, through a bend, and then through the wall of the 5-in. pipe to the outside. Thence the sample passed through a condenser. The sampling rate was controlled by a $\frac{1}{2}$ -in. throttling valve.

The water used in these tests had a resistivity of about 300,000 ohm-cm and was spiked with Cs^{137} tracer as CsCl . The concentration of activity in the feed varied from 3.3×10^5 cpm/ml to 4.5×10^3 cpm/ml. The feed (about 15 gallons), once charged, was not removed during the course of the runs. Daily feed samples were taken. The ratio of the counts of Cs^{137} in the feed to the counts in the condensed sample is reported as the decontamination factor. Samples of vapor were drawn off continuously from one or more samplers, condensed, plated, and counted. The sample flow rate was controlled in most cases at a value calculated to give the same linear steam flow rate within the sampler as that which existed in the 5-in. riser pipe. Thus isokinetic sampling was attempted. Other sampling rates were tried in order to evaluate the effect of sampling rate on the measurements. In addition, tests were made to determine the transient behavior of the apparatus. It was found that a $1\frac{1}{2}$ to 2-hr interval between operational changes was sufficient for the system to recover and stabilize. In all, the following operating conditions were investigated:

1. Pressures: 600, 450, 300, 150 psia.
 2. Total power inputs: 100 kw, 50 kw corresponding to power densities of 28 and 14 kw/liter, respectively.
 3. Sampling rates: $\frac{1}{2}$, 1, $1\frac{1}{2}$, 2, $2\frac{1}{2}$, and 3 times isokinetic values.
- Some, but not all, combinations of these variables were observed.

In a number of runs distinct steady-state plateaus were observed after $1\frac{1}{2}$ hr of operation, and these data were considered the most reliable. Some runs were discarded where obvious contamination of the sample lines occurred. The latter runs did not reach steady state during the time of operation. The most severely contaminated conditions occurred toward the end of the series of tests while the best data were obtained at the beginning of the series. It is considered possible that continued daily boiling of the same feed water over several weeks resulted in an increase in dissolved solids with a tendency for the boiling liquid to foam. Contamination was particularly noted in sampler 1 (Fig. 3-80) which was the one nearest the boiling interface and thus most subject to splashing. All decontamination factors obtained were greater than 1×10^4 , nearly all the data being in the range 1×10^4 to 4×10^4 .

In Fig. 3-81, which illustrates the data from a typical series of runs, a high initial value for activity carryover (corresponding to a low

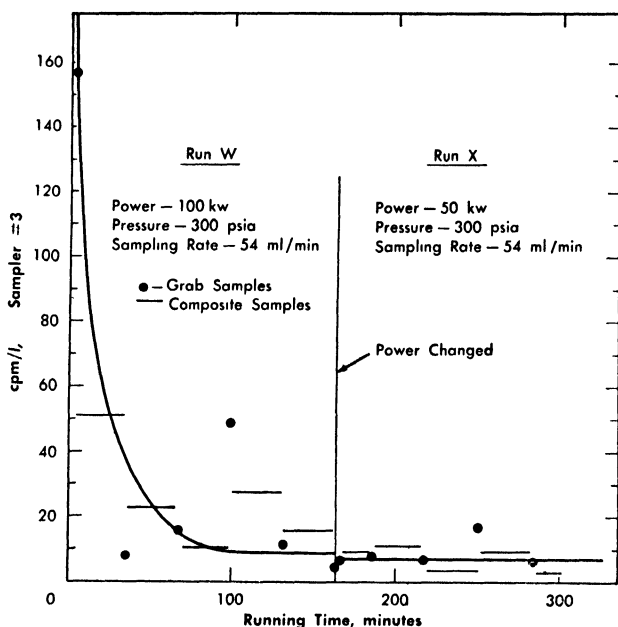


FIG. 3-81. Entrainment at 300 psia.

initial value for decontamination factor) may be noted. This phenomenon is probably similar to that observed in the small scale experiments. It was observed that the decontamination factors showed little sensitivity to pressure in the range 150 to 600 psia. A trend with steam velocity was observed for the set of results obtained at the uppermost steam sampler. This sampler was least susceptible to contamination resulting from random surges or foaming.

REFERENCES

1. SAMUEL UNTERMYER, Direct Steam Generation for Power, *Nucleonics*, 12(7), 43-47 (July, 1954).
2. J. R. DIETRICH, D. C. LAYMAN, and O. A. SCHULZE, The Proposed Boiling Reactor Experiment, ANL-4921, November, 1952.
3. J. R. DIETRICH and D. C. LAYMAN, Transient and Steady State Characteristics of a Boiling Reactor. The BORAX Experiments, 1953. ANL-5211, February, 1954.
4. J. R. DIETRICH et al., Experimental Determinations of the Self-Regulation and Safety of Operation Water-Moderated Reactors. P/481, p. 88, Vol. 13, Proc. Int. Conf. Geneva, 1955. New York: United Nations, 1956.
5. J. R. DIETRICH, H. V. LICHTENBERGER and W. H. ZINN, Design and Operating Experience of a Prototype Boiling Reactor. P/851, p. 56, Vol. 3, Proc. Int. Conf. in Geneva, 1955.
6. A. J. ULRICH, Results of Recent Analysis of BORAX-II Transient Experiments, ANL-5532, April, 1956.
7. D. E. WALKER and R. A. NOLAND, The Fabrication of Fuel Subassemblies for the BORAX-III Reactor, ANL-5559, January, 1958.
8. P. A. LOTTES, Boiling Studies at Argonne Relative to Boiling Reactors. Proc. Conf. Nuclear Engineering, University of California, April, 1955, Los Angeles, California Press, 1955, pp. 1-28.
9. W. H. ZINN et al., Operational Experience with the BORAX Power Plant, *Nuclear Science and Engineering*, 1(5), pp. 420-437 (October, 1956).
10. ARTHUR SHOR, H. T. WARD, DAVID MILLER and W. A. RODGER, Radioactive Carry-Over from BORAX-III and Test Systems, *Nuclear Science and Engineering*, 2(2), pp. 126-142 (April, 1957).
11. B. MAXON, O. SCHULZE, and J. THIE, Reactivity Transients and Steady State Operation of a Thoria-Urania Fueled Boiling Reactor (BORAX-IV), ANL-5733. In press.
12. O. A. SCHULZE, BORAX-IV: Preliminary Report on the Present Series of Experiments with Oxide Fuels, TID-7535, May, 1957, p. 114.
13. J. H. HANDWERK, C. L. HOENIG, and R. C. LIED, Manufacture of the ThO_2 - UO_2 Ceramic Fuel Pellets for BORAX-IV, ANL-5678, August, 1957.
14. P. A. LOTTES, Boiling Density Studies (Appendix B), p. 160, EBWR ANL-5607, May, 1957.
15. W. H. COOK, Boiling Density in Vertical Rectangular Multichannel Sections with Natural Circulation, ANL-5621, November, 1956.
16. J. F. MARCHATERRE, The Effect of Pressure on Boiling Density in Multiple Rectangular Channels, ANL-5522, February, 1956.
17. W. S. FLINN and M. PETRICH, Performance and Potential of Natural Circulation Boiling Reactors, ANL-5720, October, 1957.
18. R. C. MARTINELLI and D. B. NELSON, Prediction of Pressure Drop During Forced Circulation Boiling of Water, *Trans. ASME*, 70, pp. 695-708 (1948).

CHAPTER 4

DESIGN OF BOILING WATER NUCLEAR REACTOR SYSTEMS*

INTRODUCTION

Basic principles. Certain of the basic principles underlying the operation of boiling reactors have been considered in Chapters 2 and 3. In this chapter these will be discussed in greater detail, particularly as they apply to the design of boiling reactor systems.

The designer of conventional steam generators is essentially concerned with producing a design that gives ample circulation in order to insure a large factor of safety against burnout of the heating surfaces. The nuclear reactor designer also has this same concern about heat transfer limitations, but in addition he has to know relatively accurately the density changes associated with the boiling process, since density changes in the moderator of a reactor affect the moderator property of slowing down the generated fast neutrons to useful thermal (velocity) levels. In addition, since moderator density variations affect the *cold to operating* fissionable material requirements, and also have an effect on reactor stability and overload safety shutdown capabilities, an understanding of the variables that control the power and density in boiling reactors, particularly those with natural circulation, is important to the reactor designer.¹

To aid in understanding the nature of the problems associated with natural circulation boiling reactors, the diagram of the simple natural circulation boiling reactor shown in Fig. 4-1 is helpful. For the direct steam cycle shown, the coolant at entry to the core is subcooled due to the addition of the colder return feed water entering the reactor, and bulk boiling does not start until the water traverses some height L_{nb} into the core. This height at which bulk boiling begins is established by the difference in temperature between the operating saturation temperature and the temperature of the return feed water as well as by the axial heat generating pattern. Beyond this height, steam is continuously generated over the boiling height L_b all the way up to the core outlet at a rate depending on the axial heat flux pattern.

The steam-water mixture on leaving the core flows upward through the chimney without further evaporation; separation of the liquid and vapor phases occurs in the overhead mixture with the steam quantity W_s leaving

* Large portions of the material presented here are based on a report prepared for ANL by General Nuclear Engineering Corp., Dunedin, Florida.

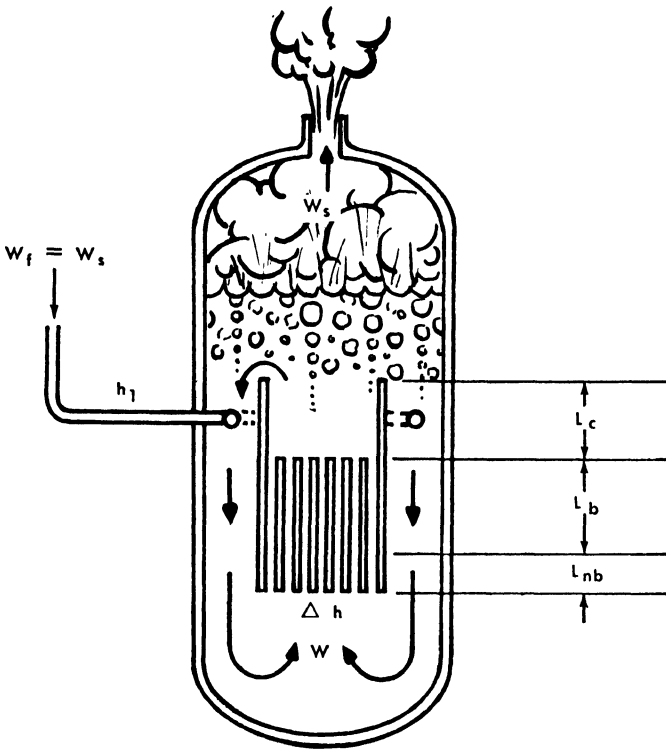


FIG. 4-1. Natural circulation boiling reactor.

the interphase; the liquid quantity $(W - W_s)$ is recirculated back to the core inlet with the return feed to complete the flow cycle. The net effect of the evaporation within the heated channel is to produce a less dense steam-water mixture in the core and chimney. In parallel and in pressure balance with this column is the heavier nonboiling water in the downcomer. The magnitude of the density difference between the two establishes the pressure differential or driving head to circulate the coolant.

With the driving head established by the density differential between the downcomer and the riser, a flow is established such that the sum of the head losses in traversing its flow cycle is equal to the available driving head. Considering the flow dynamics, there is a contraction loss in going from the inlet plenum into the channels, a friction loss in the local boiling and bulk boiling regions of the heated channel in addition to the pressure drop required to accelerate the coolant from inlet to outlet. To these head losses must be added the head required to push the two phase mixture into, through, and out of the chimney, and the separated liquid through the downcomer.

There are essentially two major uncertainties in calculating natural circulation flows. First, to calculate the driving head or the density of a known weight fraction steam in a steam-water mixture, it is necessary to know the relative velocities of the steam and water components of the mixture. For instance, if the steam velocity is very much greater than the water velocity, the volume of the steam in the channel will be very much less than for the case of equal velocities. Thus, for the case of steam slippage relative to the water, the driving head is then less than for the case of equal velocities.

Although the relative velocity phenomenon has been known qualitatively for some time by many experimenters, it was not until recently that it became a measured and determined quantity. Techniques employing radioactive sources have made this possible.²⁻⁴ In this measurement the attenuation of a monoenergetic particle beam passing through the mixture of known weight proportions is used to yield a direct measurement of the volumes occupied by the two mixture components. From these data of weight and volume proportions the relative velocities or the ratio of velocities can be ascertained.

The second uncertainty in predicting the performance of a natural circulation cycle lies in the estimation of the frictional losses occurring during the boiling process. These losses constitute the greater portion of the available head, and considering the fact that the magnitude of the driving head is relatively small, of the order of a few feet, the importance of being able to predict these losses with some accuracy should be evident.

Two equations can be written defining the heat transfer and flow processes occurring in natural circulation. They are shown below. The first equation is obtained by making a heat balance on the heated channel; the second, the flow equation, results from a pressure drop analysis around the flow cycle. In the first equation, the power density-void equation:

κ is the average power density over the heated height L in units of power-unit volume of channel, usually expressed in kilowatts per liter,

V_l is the circulation velocity established by the flow equation,

C_e/C_l is the exit slip ratio, the ratio of the steam velocity to the water velocity at the exit of the heated channel,

α_e is the steam volume fraction at the exit of the heated section,

h_{fg} , ρ_s and ρ_l are the latent heat and the saturated steam and liquid densities, respectively, and

L_{nb} and L_b are the nonboiling and boiling lengths of the channel. (For other symbols, see list at end of this chapter.)

Heat balance:

$$\kappa = \rho_l \frac{V_l}{L} \left(\frac{C_s}{C_l} \right) \left(\frac{\alpha_e}{1 - \alpha_e} \right) \left(\frac{\rho_s}{\rho_l} \cdot h_{fg} \right) \left[1 + \frac{L_{nb}}{L_b} \right].$$

Flow:

$$(\bar{\alpha} L_b + \alpha_e L_c) = \frac{V_l^2}{2g} \left[f_1 \left\{ \frac{L_{nb}}{D_e} + \left(\frac{\Delta P_{tp}}{\Delta P_o} \right)_b \frac{L_b}{D_e} \right\} + f_2 \left(\frac{\Delta P_{tp}}{\Delta P_o} \right)_{nb} \frac{L_c}{D_c} + A_{cc} + C \right].$$

The flow equation, though it appears rather complicated, merely states, without going into detail on all its terms, that the circulation velocity V_l established in natural circulation is dependent on the voids or driving head and the summation of the pressure drops previously mentioned.

A considerable amount of work has been done experimentally in studying the hydraulic aspects of natural circulation systems with a view of determining the factors which influence the variables in the power-density void equation. Some of this experimental work has been described in the preceding chapter in connection with the BORAX tests and other investigations conducted at Argonne National Laboratory.

Before considering other experimental findings on the major variables in the power-density-void equation it is of interest to describe briefly the method used in measuring densities of steam-water mixtures. Figure 4-2 shows a cross section through one of the natural circulation test sections. It consists of 4 parallel $\frac{1}{16}$ -in. thick plates electrically connected in series. Heat generation in all tests is uniform, but plate spacing can be varied. The thulium oxide source and photomultiplier tube are placed in line with the long axis of the channel. Two full channels are scanned by the densitometer. Most of the work done at Argonne has been with radioactive thulium oxide which is a 0.113 Mev gamma emitter with a $2\frac{1}{2}$ -year half life. As already mentioned, with this equipment it has been possible to measure very accurately the void fraction in water-vapor mixtures.

Circulation velocity. The first controlling variable in the power-density equation is the circulation velocity V_l . In all the experimental work it was found that the circulation velocity for a given system is solely a function of voids. Pressures in the range from 25 to 600 psi has no discernible effect on this factor. Typical test results from a multichannel test section are shown in Fig. 4-3. Although the data shown are for a constant boiling length, other boiling tests on a different channel in another test rig have demonstrated that the length of the channel in boiling does not greatly affect the circulation values obtained here at any given exit void fraction. The characteristic curve of circulation velocity versus exit voids

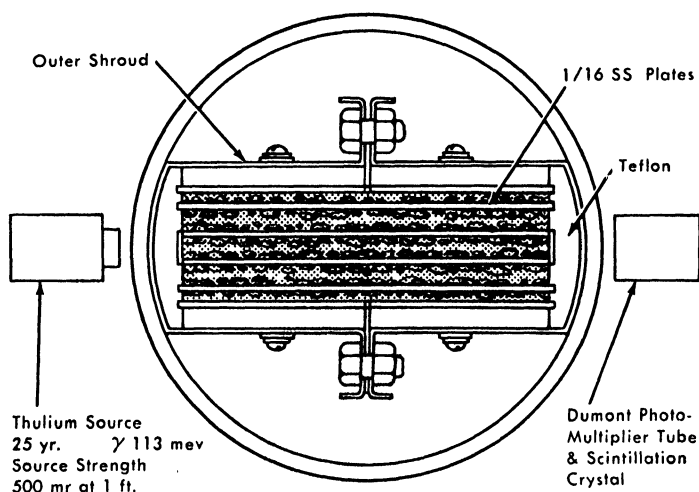


FIG. 4-2. Multichannel test section.

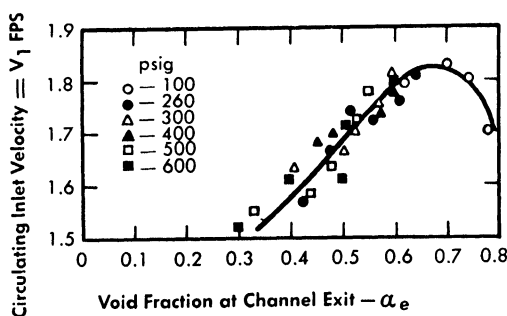


FIG. 4-3. Dependence of natural circulation on steam voids.

applies only to a given heated channel and downcomer system. For other geometries similar curves would result with the flow peak at different levels and at some other void values. It should be noted from this typical curve that in natural circulation there is a maximum self-pumping limit. On the right side of this maximum a continued increase in steam volume fraction in the channel would eventually lead to flow chugging and burnout of the channel.

The experimental finding of the dependence of the circulation velocity on only the boiling channel steam volume fraction indicates that the controlling boiling frictional losses within the channel are also only a function of the steam volume fraction. Pressure drop measurements confirm this dependence as shown in Fig. 4-4. The curve shows that the ratio of the two phase boiling pressure drop to the single phase liquid pressure drop is

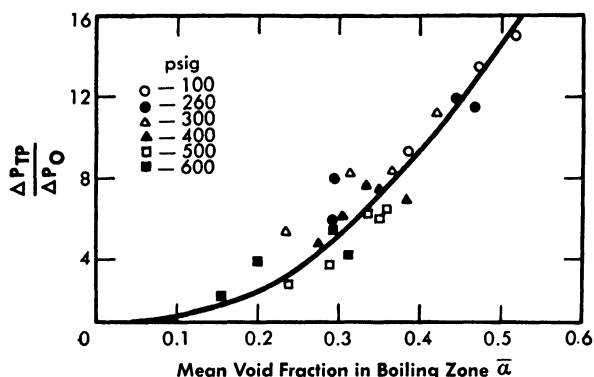


FIG. 4-4. Effect of pressure and steam voids on the two-phase friction factor.

independent of the pressure within the test range of 100 to 600 psig and increases relatively rapidly with the void fraction. These results are at variance with available data in the literature which indicate that the boiling frictional ratio is a function of the weight rate quality and pressure. Additional testing is currently under way.

Slip ratio. The next variable that appears in the power density equation is the slip ratio term C_s/C_l . Data obtained by the experiments indicate that the slip ratio is a function of both pressure and the circulation velocity. At all pressures tested (in the range of 100–600 psi) the slip ratio decreased with an increase in velocity. As the pressure increases the slip ratio decreases and the dependence on velocity is reduced. At velocities greater than 3 ft/sec, it appears that the C_s/C_l ratio is relatively insensitive to velocity. This has been confirmed by forced circulation tests wherein the velocity was varied over the range from 1 to 10 ft/sec. The results of some of these tests on slip ratio are shown in Fig. 4-5.

Considering the fact that the power density for a given void fraction is proportional to the product of the velocity and the slip ratio, it would appear that the maximum attainable power density in natural circulation is limited by the maximum circulation velocity that can be obtained. For power densities greater than this maximum, forced circulation must be relied upon. At velocities greater than 3 ft/sec the attainable power density will just about increase directly with velocity.

Knowing that the circulation velocity is established by the void fraction, that the slip ratio is a function of the circulation velocity and pressure, and that the term $h_{fg}\rho_s/\rho_l$ is linear with pressure, the pressure effect on the power density can be evaluated.

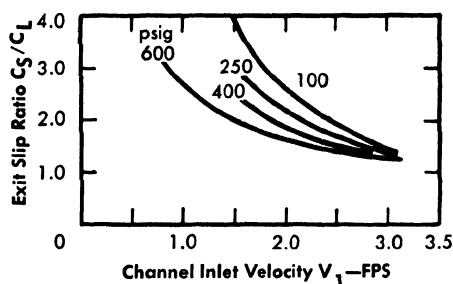


FIG. 4-5. Effect of pressure and velocity on slip ratio.

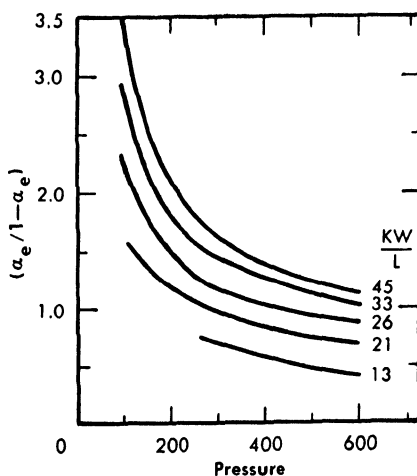


FIG. 4-6. Effect of pressure on steam voids.

To predict the power performance of a natural circulation setup, at the same void fraction, but at some other pressure, we find that the power density is proportional to the product of the absolute pressure saturation liquid density and the slip ratio term C_s/C_l . For example, considering the experimental data shown in Fig. 4-6, for a given void fraction index of 1.0 and at a power density of 21 kw/liter at a pressure of 300 psi, if one considered the direct pressure effect, then for the same voids the power density would increase by a factor of 1.88 to 40 kw/liter in going to 600 psi. But it must be remembered that in going from 300 psi to 600 psi at the same voids or circulation velocity, that the slip ratio decreases. For the case given here the slip ratio decreases by a factor of approximately 30 per cent. The power density that one would then predict in going to 600 psi, would be approximately 30 kw/liter; which agrees very well with the data.

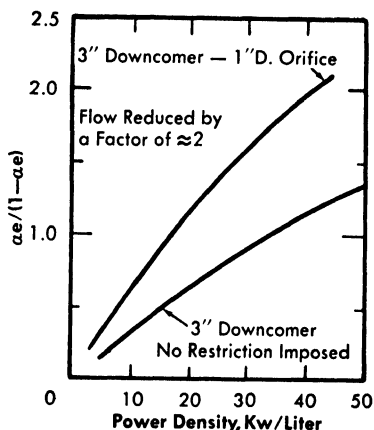


FIG. 4-7. Effect of downcomer resistance on steam voids.

Downcomer resistance is another variable that can affect the power density void relation. As might be expected, extraneous resistances added to the downcomer of a natural circulation loop have the undesirable effect of reducing the circulation velocity which tends to decrease the attainable power density for the same voids. But the effect of the reduced velocity is somewhat offset due to the fact that the slip ratio increases with the decrease in velocity. Then for the condition of equal voids at the same pressure, the effect of downcomer resistances is to give a power density proportional to the product of the velocity and the slip ratio. Experimental tests confirm this. Figure 4-7 shows the steam void-power density characteristics at 600 psi with unrestricted and restricted downcomer. The size of the restriction was such that circulation velocity was reduced by a factor of 2.5 for the same void levels. The slip ratio, in going to the lower velocity, increased approximately 50 per cent. One would then predict for the stated velocity reduction of 2.5 and slip ratio increase of 1.5, that at the lower flow condition, the corresponding power density for the same voids would be reduced by a factor of 1.6. As shown by the curves, at the void fraction index of 1.0, the power density is approximately 34 kw/liter at the higher velocity condition. At the same void fraction, by analysis one would predict the power density for the reduced velocity condition to be approximately 21 kw/liter. The test curve indicates approximately 17-18 kw/liter, which is well within the accuracy of any of the test results.

Effect of height. The last variable in the power density equation is the effect of channel heated height. Most of the data on this factor are based on the results obtained with a 4 ft high heated test section, but some data

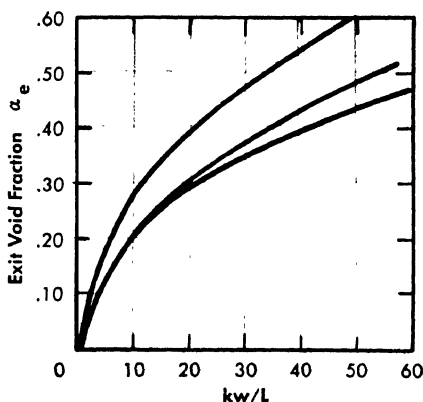


FIG. 4-8. Exit void fraction versus power density. Upper curve is 4-ft test section, lower curves are 2-ft sections.

were obtained on a 2 ft high test section, with the channel cross-section dimensions the same as for the 4 ft high section. The power density void relation for each test section, at the same pressure and inlet subcooling, is shown in Fig. 4-8. Comparing the power densities at the same void fraction, it should be noted that a factor of 2 increase in the heated channel height reduces the power density by a factor of approximately 1.7. Theoretically one would predict for the same voids, assuming no change in slip ratio and circulation velocity, a factor of 2.0 reduction in power density for a height ratio increase of 2.0. Considering the limited test data available and the experimental accuracy, the test data agree very well with what would be predicted, that the power density, for the same voids, varies inversely with heated height.

Figure 4-8 also shows data for a 2 ft high heated test section with two different heights of chimney above the heated sections. The difference in chimney heights is 17 in. From the limited testing that has been done with chimneys, it has been observed that chimneys do produce an increase in driving head greater than the corresponding increase in frictional resistance due to the chimney addition, with the net result that there is a positive increase in the circulation velocity. The increase in power density with the increased chimney height at the same steam void levels is apparent from the curves.

The effect of risers on reactor performance can be evaluated by determining their influence on the coolant flow rate. As the flow rate increases, the ratio of reactor power to steam volume fraction increases, thus improving reactor performance. In restrictive-type riser designs, the greatest benefit is obtained from the first few feet of riser height since the ratio of per cent additional driving head to per cent additional flow resistance is in-

herently low. By enlarging the riser diameter, the ratio of per cent additional driving head to resistance can be considerably increased.³

One of the most interesting features of some recent studies³ is the manner in which pressure affects the performance of boiling reactors. Power density increases at higher pressures, and circulating rates do not inherently drop off with pressure. The basic reasons for this are threefold:

1. On the constant density deviation basis, driving head does not drop off with pressure.
2. Although the two-phase friction factor multiplier does increase with increasing steam volume fraction, this is more than offset by the decrease in friction due to the increased temperature as a result of the pressure increase. The viscosity of the water decreases with temperature increase.
3. At very high pressure there is additional driving head due to the density-temperature relationship of water. At the high power densities of high pressure operation, this additional head can be quite significant.

An interesting performance characteristic of high pressure operation at 2500 psi shows that as power density is increased, the true core density increases. Such interrelationships of power density, coolant density, and inlet subcooling suggest a reactor with low core steam volume fraction operating by natural circulation near the critical pressure, possibly with temperature control.

At higher pressures the boiling lengths inherently become shorter (about 35% of core length at 2500 psi for uniform axial flux distribution); therefore lower steam volume fraction and qualities will exist in the zone near the higher heat flux. This shorter boiling length and its location near the discharge end should be beneficial from the standpoints of burnout and stability.

The possible beneficial aspects of high pressure operation have to be weighed against associated drawbacks. Burnout considerations presently appear to dictate a general lowering of permissible heat fluxes above system pressures of about 1000 psi, but this value should not be regarded as the optimum. Maximum reactor vessel diameters at high pressures further lower attainable total power output. Additional questions such as those concerning water dissociation at high power densities and pressures, and the resultant effects on turbines, piping, fossil fuel-fired super-heaters, etc., have yet to be answered. For high pressure operation it is probable that stainless steel cladding of the fuel elements will be required. Depending upon the particular design, this may significantly affect neutron economy.

Burnout. In order to evaluate the potential of boiling water reactors it is necessary to know where and how burnout occurs. This is still a basic uncertainty in boiling reactor design. However, burnout data available at the present time indicate that economically high power densities at pressures well above 1000 psi are feasible.

Very high pressure burnout bears further study. The properties of steam and water at pressures slightly less than the critical pressure are such that a sudden phase change in this region may not lead to a burnout or excessive temperature condition due to channel blockage as can happen at lower pressure.]

Stability. With regard to stability, there is quite general agreement that hydrodynamics has an important bearing on stability considerations. A trend of increased stability with pressure has been shown by reactor and thermal-hydrodynamic experiments up to 600 psi.

Qualitatively, a natural circulation loop such as that shown in Fig. 4-1, can be considered as a physical system having inertia, resistance to flow, a pressure differential due to density head, and energy storage in water and steam. Disturbances in heat input, steam flow rate, or feed-water input upset the balance of these individual parameters, and a realignment must occur to satisfy the new conditions. At certain combinations of power level and system pressure for a given flow loop configuration, the damping within the system will decrease until sustained oscillations occur in flow and pressure and the system is completely unstable, no nuclear contribution being necessary to excite this action. This does not imply that nuclear effects may not also contribute to the action. Nuclear effects are, of course, important contributors in a nuclear reactor.

While much work remains to be done on the hydrodynamic aspects of boiling reactors, certain conclusions can be drawn from the experimental work done to date. These are as follows:

1. Forced circulation may not always be necessary to achieve high flow rates.
2. High pressures offer promise of high power density operation.
3. Power density is likely to be a primary limiting feature of low pressure design, while burnout heat flux is likely to be the primary limiting consideration of higher pressure design. With the recent use of oxide fuels, however, power density is beginning to seem less important; with oxide fuels the temperature drop in the fuel slug, rather than stability considerations, seems to become the limiting factor.
4. There is considerable need for further analytical and experimental development of significant parameters and correlations.

NUCLEAR ASPECTS OF BOILING WATER REACTORS

The formation of steam in a boiling reactor causes some of the water moderator to be displaced from the core. Such a change in the moderator density will, in general, cause a change in the reactivity of the reactor, though it is possible by suitable reactor design to minimize this change. In order that the reactor should not be autocatalytic in its behavior it is necessary that the reactor be so designed that under any possible operation conditions a reduction in moderator content (i.e., an increase in steam content) will *not* increase the reactivity. In achieving this characteristic, it usually happens that the reactor is given a relatively strong negative dependence of reactivity on steam content. An alternate statement of this characteristic is that the reactor has a relatively strong negative power coefficient of reactivity. The result is that the reactor is strongly self-regulating with respect to power. That is to say, under given operating conditions, the position of the control rods in a given reactor determines the power level, rather than the rate of change in power level as is the usual case. The negative power coefficient also places a limit on the power level which can result from a given increase in reactivity, and thus gives an important safety factor against the possibility of a reactor runaway. Since the negative power coefficient is not instantaneous in its action, but rather is subject to time lags resulting from the finite rate of heat transfer to the water and from the inertia of the reactor water, the inherent self-regulating property of the reactor is one which, like any feedback control, can lead to instability under certain circumstances. This possibility of instability is heightened by the fact that the self-regulating system which is inherent in the boiling characteristic is a very *noisy* system because of the irregular fluctuations in the behavior of steam bubbles.

Since the reactivity change associated with a given change in steam content of the reactor is an important aspect of the reactor behavior, it is necessary to define certain terms which will be needed in a quantitative discussion of this relationship, see Fig. 4-9.

(a) *The steam fraction* is just the ratio of the volume or weight of steam in the core to the volume or weight of steam plus liquid in the core; or in other words, it is the fraction of the coolant channel volume occupied by steam. The steam volume fraction is usually designated by the symbol α . The steam weight fraction is indicated by X .

(b) *The reactivity worth of the steam* is the reactivity of the reactor when it has steam in it (whatever amount of steam one is interested in) minus the reactivity when there is no steam in the reactor.

(c) *The steam coefficient of reactivity* is simply the ratio of these two quantities, and this coefficient provides a quantitative measure of the

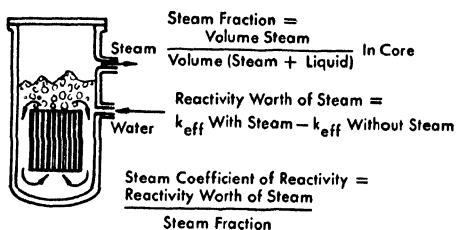


FIG. 4-9. Diagram explaining terms used in discussing nuclear properties of a reactor.

effect of steam on reactivity. There is no implication that the steam coefficient of reactivity is necessarily independent of the steam fraction; the concept is nevertheless a useful one.

This last quantity, the steam coefficient of reactivity, could be either positive or negative in sign. If it was positive, it would mean that as steam forms in the reactor, the reactor becomes more reactive, the power goes up, more steam is formed, the power goes still higher—in other words, the reactor would be autocatalytic and would run away.

The first thing one must be certain of in designing a boiling reactor is that at the design condition the steam coefficient of reactivity is negative. For example, it may be desirable that this be true at room temperature.

To show how this can be achieved a simple enriched uranium reactor such as the BORAX reactor may be taken as an example. Roughly, such reactors can be considered as being cylindrical cores made of aluminum plate containing enriched uranium, and submerged in water. Consider what would happen as the diameters of such cores are changed, keeping the height roughly constant, as was done in the series of three first BORAX reactors, previously described.

In Fig. 4-10 is shown a cylinder to represent the core of constant height—65½ centimeters. The question is, what happens as the diameter is made larger. It is more informative to plot, not the diameter, but the cross-sectional area of the core, as is done in the drawing.

The steam coefficient of reactivity is plotted as the ordinate. Note that these are negative numbers increasing downward. As shown, a small diameter core has a large negative coefficient of reactivity. A large diameter core has a small negative coefficient, and presumably if the core is made still larger, the coefficient will become positive.

This is easy to understand in terms of the simple reactor equation shown in the diagram. Here k_{eff} , the effective multiplication factor of the reactor, is shown to be the product of k_{∞} times one minus the fraction of neutrons which leak out of the reactor. k_{∞} is the infinite multiplication constant. Expanding this equation in terms of the two-group reactor

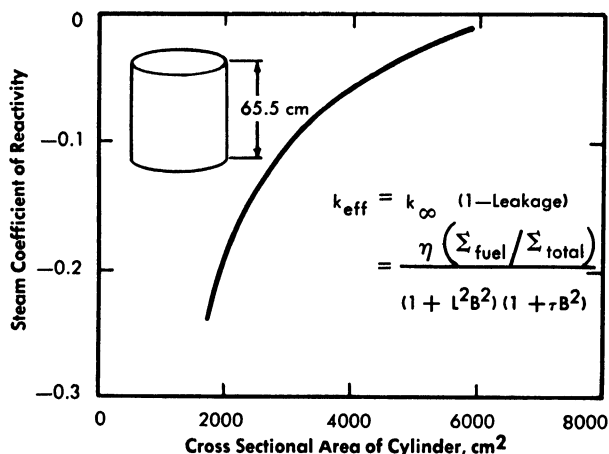


Fig. 4-10. Variation of steam coefficient of reactivity with cross-sectional area of cylinder.

equation* it is found that the expression for k_{∞} is simply n , the number of neutrons emitted per neutron absorbed in fissionable material, times the thermal utilization, which is just the ratio of the macroscopic absorption cross section of fuel (or fission) to the macroscopic absorption cross section of all material in the core.

The term $(1 - \text{leakage})$, on the two-group picture is simply the reciprocal of this product $(1 + L^2 B^2)(1 + \tau B^2)$, where L^2 is the diffusion area, τ is the slowing down area, B^2 is the geometric buckling. This is a measure of the curvature (or buckling) of the neutron flux at any point in the critical reactor. The buckling is large for small cores and small for large cores. Consequently, the result of steam forming in the reactor core is to push liquid water out of the core, causing a loss in absorption. This tends to make k_{∞} increase and the reactivity of the reactor (k_{eff}) to increase. On the other hand, $L^2 \tau$ also increases, tending to make the denominator of the fraction increase and to decrease k_{eff} .

In a small reactor, in which the buckling is large, the denominator will predominate strongly. With a large reactor the buckling is small and the numerator of the fraction becomes more important; eventually if the reactor were so large that there was effectively no leakage, the steam coefficient of reactivity would inevitably, for such a reactor, become positive. The coefficient is made negative by suitable choice of reactor shape and size.

* In the two-group treatment, neutrons of thermal energy constitute one group while all those of higher energy are in the other group. For an explanation of the group-diffusion method as applied to reactor calculations see Glasstone and Edlund, *The Elements of Nuclear Reactor Calculations*, D. Van Nostrand Co., Inc., p. 227.

Having attained the negative steam coefficient, two desirable operating characteristics are obtained. The first is that the power is just the opposite of autocatalytic; it is self-limiting. Thus, if by some unusual chance a large amount of excess reactivity were applied to the reactor, the power would increase moderately and then enough steam would be formed to remove this excess reactivity. Most other types of reactors, under such circumstances, would experience a power rise exponentially with time until the reactor was damaged. As explained in the preceding chapter, the BORAX experiments demonstrated this characteristic of boiling reactors in a dramatic fashion.

The second desirable operating characteristic resulting from the negative steam coefficient is the self-regulation of the reactor power. The negative steam coefficient of reactivity implies a negative power coefficient of reactivity also. If the control rod of the reactor is kept in a fixed position, the reactor will try to maintain a constant amount of steam in its core. This means that the power of the reactor will have to stay constant. This is called self-regulation. However, a reactor behaves like any other automatic control system, and it is known that whenever a feedback control system involves inertia and associated time lags, if the gain of the feedback control system is increased sufficiently, the system will become unstable. The gain of the system is the product of the ratio of power to reactivity, of voids to power, and of reactivity to voids.

For the boiling reactor, the open loop gain is proportional to the steam coefficient of reactivity; thus, if this coefficient is permitted to become too large, an oscillating condition may result. Figure 4-11 shows records of such oscillations, with some sections of the chart speeded up to show the approximately sinusoidal form of the oscillations. It has a frequency of about two cycles per second. The period is of the same order of magnitude as the time constant describing the motion of water in the reactor core under the force of gravity.

This oscillatory operation is by itself not inherently undesirable, nor necessarily dangerous. It should, however, be considered a warning that the reactor system has reached an operating condition which may lead to the chugging type operation. This would be unsatisfactory in a power reactor. For this reason it is desirable to design the reactor in a way which avoids these oscillatory tendencies. This can be done in many ways. For example, the steam coefficient of reactivity can be made small, or the steam content of the core can be kept small. The latter method, however, means a low power density in the core or a low performance reactor.

Figure 3-35 shows the gains that have been made in the power densities obtained from the BORAX reactors as the reactors were increased in size and as their steam coefficients of reactivity consequently were decreased.

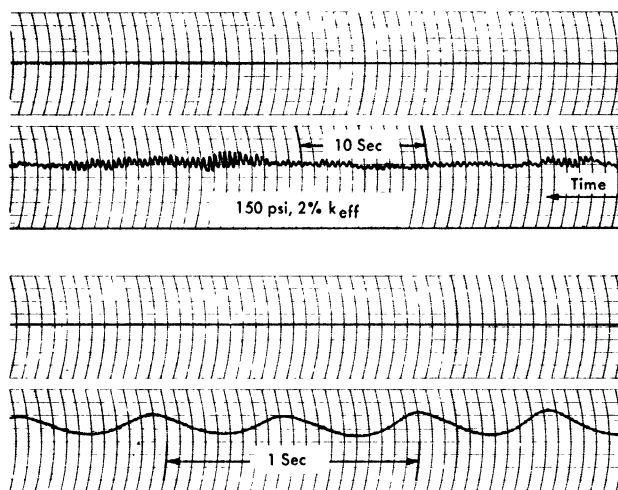


FIG. 4-11. Chart from BORAX experiments showing oscillations in reactor power.

Experience has shown that oscillations in reactor power would not be likely when the reactivity worth of the steam (designated in the chart as excess k in the steam) amounts to something like 2 per cent. In the small BORAX-I reactor this value of k , a power density of about 8 kw/liter was obtained. In BORAX-II, which had a lower steam coefficient, the power density went up to 17 kw/liter, and finally, in BORAX-III, it was up to almost 30 kw/liter.

Whether this limiting value of 2 per cent excess k can be accepted as applicable generally is open to question. Recent successes in exceeding 2 per cent k by factors of 2 and 3 show that this value does not provide a realistic limit on the capabilities of a boiling reactor. In the early days of boiling reactor technology, 2 per cent seemed to be a limiting factor, but now actual practice has gone well beyond that supposed limit.

These theoretical considerations and the BORAX experiments have made it possible to approach the physics design of a boiling reactor with a high degree of confidence. First it is necessary to have a negative steam coefficient of reactivity as indicated before. Secondly, this coefficient should be as small as consistent with safety considerations.

It may be of interest to point out how k excess was kept small in the case of the EBWR. A cross section of the core of this reactor is shown in Fig. 5-12. The outer circle represents the reactor vessel. The core, which is submerged in water, is made of a number of fuel elements, some of which are made of plates of natural uranium; others are plates of enriched uranium containing zirconium. These are necessary in order to obtain sufficient reactivity. The average diameter of the core is roughly

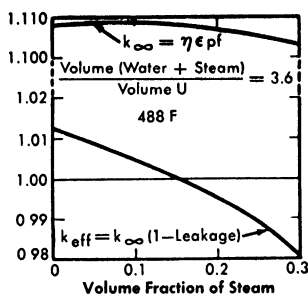


FIG. 4-12. Variation of k_{eff} with steam volume fraction.

50 in. which is considerably larger than any of the BORAX reactors. On the basis of the considerations discussed before, this reactor would tend towards a positive coefficient of reactivity. This reactor has a diluted core containing U^{238} with an average enrichment near 1.41 per cent of U^{235} .

Now (Fig. 4-12), the expression for k_{∞} , is more involved. It is equal to η times ϵ , the fast effect, times p , the resonance escape probability, times the same term as for BORAX, the thermal utilization f . Disregarding the other terms for the moment, it will be of interest to look at p , the resonance escape probability.

Resonance escape probability depends upon the ratio of the number of neutrons at resonance energies to the number of neutrons at thermal energy. If the water content of the reactor core is reduced, the moderation of neutrons is reduced and the number of resonance neutrons relative to the number of thermal neutrons is increased. In other words, the resonance escape probability is increased.

By proper choice of the ratio of water to uranium in the reactor, the effects of the two terms, p , which tends to cause reactivity to decrease as steam is formed, and f , which tends to cause reactivity to increase as steam is formed, can just about be cancelled out. So k_{∞} is almost a constant as more and more steam is put into the reactor. This means that any enhanced leakage which may occur as the result of steam is available to decrease the reactivity and result in a negative steam coefficient.

The steam coefficient for this reactor is just the slope of this line at the operating point, which is 488 F and a steam content of about 15 per cent by volume. These values were obtained in a parameter study for the EBWR type reactor—not for EBWR itself. The value of this coefficient is about 0.14 per cent k /per cent steam. This it is believed is a satisfactory value though it would be better if it were a bit smaller.

Brief mention should be made here for the transfer function measurements made in an analytical procedure to predict the stability of EBWR. These are described in Chapter 6 in greater detail. These transfer function

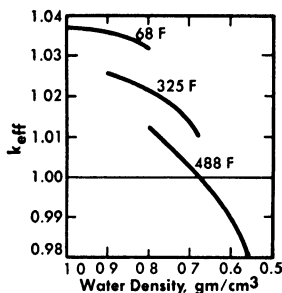


FIG. 4-13. Water density versus k_{eff} .

studies were carried out for power levels well in excess of the known areas. The program was divided into four parts: First extensive power response measurements were made with a reactivity oscillator at powers up to the original power rating of the EBWR reactor, 20 Mw. Then an expression for the power-to-reactivity feedback transfer function was chosen that gave the best fit to the data. The transfer function was then extrapolated to higher powers and was examined for stability in the region of 40 Mw thermal. Finally, after minor preparations for the larger heat removal, the reactor was taken up to the proposed new power level. In the process, a good theoretical model of the reactivity feedback loops and a method for predicting stability of boiling-water reactors at higher power levels were developed.

Figure 4-13 shows why it cannot be smaller. A reactor cannot be considered just at its operating condition. It is also necessary to have a negative steam coefficient at room temperature. The chart in Fig. 4-13 is slightly different from the one in Fig. 4-12 in that it plots not the steam content of the water, but the water density (with the scale running backwards). So, actually, as long as the density is being decreased by the formation of steam, the coordinates of this curve are the same as those of the previous one. However, it is not the presence of the steam, but the water temperature as well, that affects the average value of the water density.

It is found that all these characteristic curves for k_{eff} as a function of water density do not have a constant slope but a slope which increases in a negative fashion as the water density becomes lower and lower. It is for this reason that room temperature was chosen as the design point. The coefficient was made slightly negative at this point. It is almost zero. Then it becomes necessary to accept whatever value occurs at the operating condition and, as previously mentioned, this is believed to be satisfactory. There is reason to believe that it will be satisfactory for any boiling reactor that is designed according to these principles.

SPECIAL PHYSICS CHARACTERISTICS OF BOILING REACTORS

Many of these reactor physics problems, which are unique to the boiling reactor as has been shown, have to do with the attainment of a steam coefficient of reactivity of proper sign and magnitude. Aside from such problems, boiling reactors involve special physics concerning the effects of steam distribution on neutron distribution and on power distribution. Both classes of problems vary between reactors which are highly enriched and those which use natural or slightly enriched fuel, and between reactors which are moderated by light water and those which are moderated by heavy water. The physics of these classes of reactors will be considered in the following sections.

HIGHLY ENRICHED BOILING REACTORS

It is instructive to consider the steam coefficient of reactivity for the idealized case of a bare, highly enriched, water-moderated reactor. The effective multiplication constant k_{eff} can be written in the one group approximation as:

$$k_{\text{eff}} = \frac{k}{1 + M^2 B^2}, \quad (4-1)$$

where k is the infinite multiplication constant, M^2 is the migration area, and B^2 is the geometric buckling. A change in the water content of the reactor will change k because of the change in neutron absorption and will change M^2 because of the changes in moderating and scattering effectiveness. The buckling would change by a very small amount, which will here be considered negligible, because of the change in extrapolation length at the reactor boundary. Differentiation of (4-1) with respect to k and M^2 yields

$$\frac{dk_{\text{eff}}}{k_{\text{eff}}} = \frac{dk}{k} - \frac{M^2 B^2}{1 + M^2 B^2} \frac{dM^2}{M^2}. \quad (4-2)$$

The change in k when liquid water is displaced by steam is caused entirely by the change in neutron absorption by water (if the reactor may be considered homogeneous and if the steam is assumed to be produced uniformly throughout the reactor), and may be expressed in terms of the change in macroscopic absorption cross section of water, $\Sigma_{\text{H}_2\text{O}}$:

$$\frac{dk_{\text{eff}}}{k_{\text{eff}}} = \frac{-f_{\text{H}_2\text{O}} d\Sigma_{\text{H}_2\text{O}}}{\Sigma_{\text{H}_2\text{O}}} - \frac{M^2 B^2}{1 + M^2 B^2} \frac{dM^2}{M^2}, \quad (4-3)$$

where $f_{\text{H}_2\text{O}}$ is the fraction of the total absorptions which are absorptions by water.

It appears quite evident, that if the composition of the reactor has been chosen, the sign and magnitude of the change in k_{eff} for a given change in steam fraction (the steam coefficient of reactivity) can be adjusted only by adjusting the geometric buckling, B^2 . The criterion that the steam coefficient be negative requires that the final term of the equation (4-3) be larger in absolute magnitude than the immediately preceding term. In other words, at least one dimension of the fully enriched reactor must be sufficiently small that the increased neutron leakage due to water displacement by steam will be more important, reactivity-wise, than the accompanying decrease in parasitic absorption.

In the case of a highly enriched reactor using aluminum fuel plates (such as the BORAX reactors), a typical value for the fraction of absorptions by water ($f_{\text{H}_2\text{O}}$) might be about 0.3. If one considers that the variation expressed by equation (4-3) is caused by a small change in water density, $d\rho$, the fraction $d\Sigma_{\text{H}_2\text{O}}/\Sigma_{\text{H}_2\text{O}}$ is just equal to $d\rho/\rho$. The variation of M^2 with water density cannot be expressed exactly by a simple relationship. If L^2 , the slowing down area in the core, were negligibly small, and if the neutron scattering in the reactor core were due to the water alone, M^2 would vary in inverse proportion to ρ^2 . Although these conditions are not fulfilled in practice, it is sufficiently accurate for this general discussion to assume that M^2 varies as $1/\rho^n$, where n is slightly less than 2. (For the BORAX-III reactor the relationship is reasonably accurate with $n = 1.7$.) The expression for the variation of M^2 with ρ then becomes

$$\frac{dM^2}{M^2} = -n \frac{d\rho}{\rho}. \quad (4-4)$$

The expression for the variation of reactivity with water density, ρ , then becomes

$$\frac{dk_{\text{eff}}}{k_{\text{eff}}} = \left[-f_{\text{H}_2\text{O}} + n \frac{M^2 B^2}{1 + M^2 B^2} \right] \frac{d\rho}{\rho}. \quad (4-5)$$

The requirement that the steam coefficient of reactivity be *negative* is just equivalent to the requirement that the water density coefficient of reactivity be *positive*, or that

$$\frac{M^2 B^2}{1 + M^2 B^2} > \frac{f_{\text{H}_2\text{O}}}{n}. \quad (4-6)$$

For a reactor of the BORAX-III composition, at room temperature, $M^2 = 55 \text{ cm}^2$, and the criterion would be

$$\frac{55B^2}{1 + 55B^2} > \frac{0.3}{1.7}; \quad B^2 > 3.9 \times 10^{-3}. \quad (4-7)$$

Thus, if the reactor were made in the shape of a cylinder of minimum critical volume, the limiting value of the radius would be $R = 2.945/\sqrt{3.9 \times 10^{-3}} = 47$ cm. Reactors of larger radii would exhibit positive steam coefficients of reactivity at room temperature.

Although the requirement of negative steam coefficient of reactivity imposes a limitation on the minimum value of buckling which can be used for a highly enriched H_2O moderated boiling reactor, it does not place a limitation on the permissible volume of the reactor, since the reactor shape can be varied to attain the necessary value of buckling for any given volume. In the larger BORAX reactors, for example, the length-diameter ratio of the reactor core was made considerably less than that which results in minimum critical volume in order to preserve the buckling at a suitably high value.

Variation of steam coefficient with void fraction. It has been mentioned that the steam coefficient of reactivity does not in general remain constant as the steam fraction changes. As equation (4-3) suggests, the steam coefficient will be most positive or most nearly positive when the maximum water density exists in the reactor. This would usually be the case when the reactor is at room temperature and contained no steam voids. It might be questioned whether a negative value of steam coefficient is essential when the bulk temperature of the reactor water is below the boiling point. Transient tests on reactors containing cold water have indicated, however, that steam can probably be formed at the fuel plate surfaces before the bulk of the water is heated to the boiling point if the reactor power is rising very rapidly as it would under certain hypothetical reactor accidents. It has therefore been the practice for most reactors in the past to be designed for a negative steam coefficient of reactivity even at room temperatures.

The data shown in Fig. 4-14 were prepared on the assumption that the bare, highly enriched reactor considered in this discussion has been built with a geometric buckling of 3.9×10^{-3} , giving it a steam coefficient of just zero at room temperature and when the steam fraction is negligible. The fractional absorption by water, f_{H_2O} , and $n[M^2B^2/(1 + M^2B^2)]$ have been plotted as functions of these two curves, which gives the water density, on the assumption that $n = 1.7$. The difference of these two curves, which gives the water density coefficient of reactivity, $(dk_{eff}/k_{eff})/(d\rho/\rho)$, has also been plotted.

The change in water density in the chart may be considered to be the result of either increased water temperature, or of the presence of steam, or both. The relation between steam coefficient of reactivity and density coefficient of reactivity may be derived from the relationship

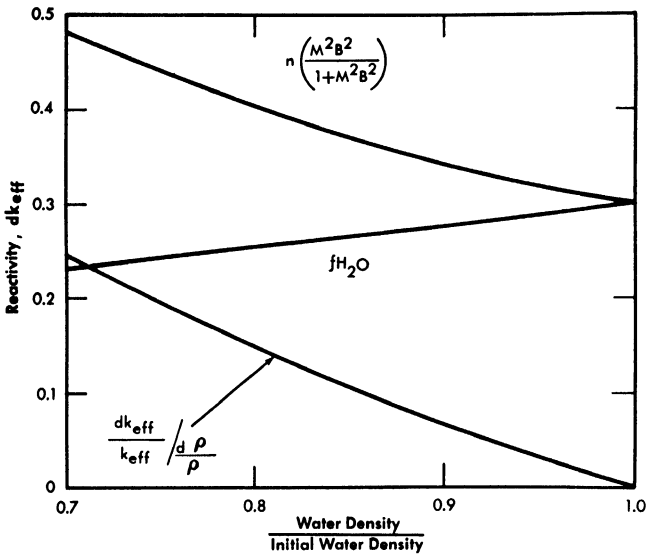


FIG. 4-14. Change in reactivity in a bare, highly enriched reactor with change in water density.

$\rho = \rho_0(1 - \alpha)$, where α is the steam fraction. Then $d\rho/\rho = -d\alpha/(1 - \alpha)$, and Fig. 4-14 defines the steam coefficient of reactivity $(dk_{\text{eff}}/k_{\text{eff}})/d\alpha$, as well as the density coefficient.

The increase in the net coefficient with decreasing water density is striking. It is evident that a reactor which is designed to have a very slightly negative steam coefficient of reactivity at room temperature may have quite a substantial negative coefficient at some useful operating temperature between, say, 400 and 500 F, and that the coefficient will become even more strongly negative as the reactor is brought into boiling operation. This characteristic dependence of steam coefficient on water density holds qualitatively for all H_2O moderated reactors, whether they use highly enriched or slightly enriched fuel. The magnitude of the variation, however, depends upon the reactor design. Because of the usual rather rapid increase of the coefficient as water density is decreased, the requirement that the coefficient be negative at room temperature places rather severe limitations upon the degree to which the desirable condition of low negative coefficient under power operating conditions can be realized. It may be that the requirement for a negative temperature coefficient at room temperature may be relaxed in the future in respect to large power reactors.

Effect of reflector. A practical boiling reactor will normally be surrounded by a reflector. If the reactor is moderated by water the reflector will usually be composed also of water. The reflector saving will not usually remain constant as the average density of the water in the core changes, but rather will increase with decreasing density of the core water. In other words, the effect of the reflector is to reduce the magnitude of the change in neutron leakage which is associated with a decrease in moderator density in the core. The change in the reflector savings represents a change in the geometric buckling of the core, and its effect on the change in effective multiplication constant associated with a given change in water density can be expressed by extending the relation of equation (4-2) to include the effect of a change in geometric buckling as in the following equation:⁶

$$\frac{dk_{\text{eff}}}{k_{\text{eff}}} = \frac{dk\alpha}{k\alpha} - \left[\frac{M^2 B^2}{1 + M^2 B^2} \right] \left[\frac{dM^2}{M^2} + \frac{dB^2}{B^2} \right]. \quad (4-8)$$

The change in geometric buckling associated with a given change in average core water density or a given change in average void fraction can be evaluated if the reflector saving is known as a function of water density or void fraction.

The calculated reflector savings for reactors having the general composition characteristic of the BORAX reactors (but not identical with any of the BORAX reactors) for a number of water temperatures and core void fractions are given in Table 4-1. For a reactor such as this, of reasonably large size and having an infinitely thick water reflector, the reflector saving is primarily a function of the slowing down area of the core and does not vary rapidly as other characteristics of the core are changed. To illustrate the point that the reflector saving is primarily a function of L_f^2 , the slowing down area, independently of whether the variation in L_f^2 results from a change in water temperature or a change in void fraction, the calculated reflector savings of Table 4-1 have been plotted as a function of L_f^2 in Fig. 4-15.

Since the variation in reflector saving is such as to constitute a positive contribution to the steam coefficient of reactivity, the effect is to increase the limiting value of the geometric buckling which marks the boundary between positive and negative steam coefficients of reactivity for highly enriched reactors. Thus the dimensions of a reflected reactor must be smaller in order to guarantee a negative steam coefficient of reactivity, not only because the presence of the reflector decreases the buckling, but also because the change in reflector saving with the steam content constitutes a positive contribution to the steam coefficient.

TABLE 4-1

RADIAL REFLECTOR SAVINGS FOR A BORAX TYPE REACTOR
AS CALCULATED BY THE TWO-GROUP APPROXIMATION*

Temperature, °F	Void fraction	L_f^2 , cm ²	Reflector savings, cm
68	0 0	50 6	7 60
68	0 1	60 8	8 41
200	0 0	53 8	7 89
200	0 2	79 4	9 77
200	0 4	127 7	12 81
420	0.0	67 7	9 14
420	0 2	98 8	11 37
420	0 4	158 0	14 95

* The reactor core is a uniform mixture of H₂O, aluminum, and U²³⁵. Volume H₂O - (void/volume aluminum) = 0.42. Reflector is pure H₂O, infinitely thick, at same temperature as the core.

Effect of spatial distribution of steam. The leakage of neutrons from the reactor, which gives rise to the negative component of the steam coefficient of reactivity, is determined in the two-group approximation by the thermal absorption cross section, the slowing down cross section, the thermal diffusion constant, and the fast diffusion constant. If the spatial dependence of these effects is examined by perturbation theory, it is apparent that the effects of the thermal absorption cross section and the slowing down cross section are proportional to the products of real and adjoint fluxes (or the differences of such products), whereas the effects of the diffusion constant are proportional to products of gradients of the real and adjoint fluxes. The statistical weight for a change in infinite multiplication constant k , which constitutes the positive contribution to the steam coefficient of reactivity, is proportional to the product of real and adjoint fluxes. The relative magnitudes of the positive and negative components of the steam coefficient therefore depend upon the position at which the steam is formed in the reactor, the local value of the negative component of the coefficient being smaller in proportion to the positive coefficient at the center of the reactor, where the fluxes and adjoint fluxes have their maximum values and where the gradients are zero. It is therefore possible in a large, highly enriched reactor to have a positive steam coefficient of reactivity associated with steam which is formed at the center of the reactor, and a negative coefficient associated with steam formed closer to the edges of the reactor. It is something of an over-

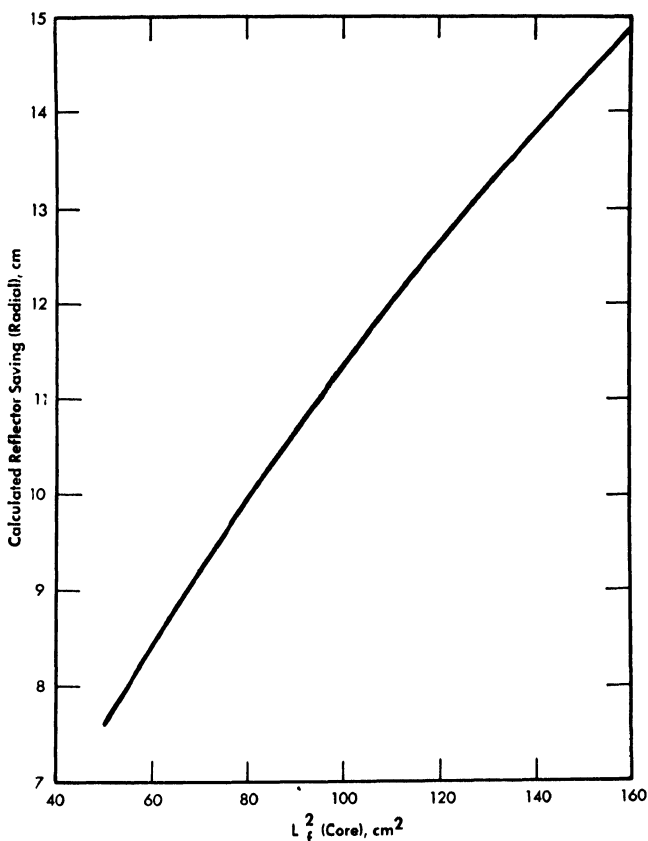


FIG. 4-15. Calculated reflector savings for reactor of BORAX type, plotted as a function of slowing-down area in core.

simplification to speak of an average steam coefficient of reactivity unless it is a spatial average weighted in proportion to the spatial distribution of steam content. Further, if a reactor which has its highest power density at the center has also a positive steam coefficient at the center, the result of a rapid power excursion which starts from the nonboiling condition (e.g., as the result of a startup accident) will be to form steam first in the region where the coefficient is positive. Thus the early effect of steam formation in such a case would be autocatalytic, but the effect would be reversed after the power increased sufficiently to cause boiling throughout a large fraction of the reactor core. Such an effect was observed experimentally in transient tests on the BORAX-II reactor. The total integrated positive reactivity effect which resulted from the early formation of steam near the reactor center was sufficiently small that it did not constitute a severe limitation to the self-regulating property of the reactor.

The experimental effects of the steam coefficient on power density were described in Chapter 3 in the BORAX experiments. The calculated steam coefficients of reactivity under operating conditions are given in Table 3-4, along with other differences which characterized the various reactors. In Fig. 3-35 the power density obtained in each of the first three BORAX reactors is plotted as a function of the reactivity worth of the steam. The three curves are not strictly comparable because all points were not taken at the same operating pressure. The proposition that higher power densities are obtained in reactors with lower steam coefficients of reactivity is borne out in a general way by these data. It is hardly to be expected, however, that all the characteristics of boiling reactors can be explained in terms of the steam coefficient of reactivity alone.

Effect of control rods. So far in this consideration of variation of steam coefficient of reactivity with temperature and with steam fraction, no attention has been given to the method by which criticality is maintained in a reactor as these changes take place. The normal practice is, of course, to maintain criticality by the proper positioning of absorbing control rods. It may be said in general that the absorption rate of control rods relative to the absorption rate in the reactor core as a whole increases as water density is reduced and as the diffusion length in the core is thereby lengthened. Furthermore, in most types of control-rod installations the geometric buckling of the reactor is substantially greater when the rods are inserted or partially inserted than when they are completely withdrawn. This effect can be quite pronounced under some types of rod installation and programming. For example, if all control rods are removed together as a bank, their effect, when they are partially inserted, is effectively to shorten the reactor core and thereby to increase its geometric buckling. Both of these control-rod effects, the increase in absorption rate with decrease in water density and the increase in geometric buckling caused by the rods, contribute components of negative sign to the steam coefficient of reactivity. Consequently if a high degree of excess reactivity is contained in the reactor at operating conditions (for purposes of fuel burnup, for example) the negative steam coefficient may be substantially larger than that predicted by a calculation which does not take the control rods into account. Unfortunately this extra negative contribution to the void coefficient by the control rods cannot ordinarily be taken advantage of in meeting the condition that the cold reactor have a negative steam coefficient of reactivity. It is almost always possible to visualize cases in which the cold reactor, because of fuel burnup and transient xenon poison, could require complete withdrawal of all control rods for criticality.

H₂O BOILING REACTORS OF LOW ENRICHMENT

The steam coefficient of reactivity for a boiling reactor using natural or slightly enriched uranium is made up of those components which entered the coefficient in the highly enriched case plus three new components, all of which are associated with the infinite multiplication factor, k . A decrease in average water density in such a reactor will cause an increase in the fast fission factor, ϵ , a decrease in the disadvantage factor which enters the determination of the thermal utilization, and an increase in resonance absorption. Of these three changes, the change in ϵ and the change in thermal disadvantage factor make positive contributions to reactivity, and the change in resonance absorption makes a negative contribution. Of the three effects, the increase of resonance absorption is by far the most important in most cases, and in fact is the one which must be counted on for the major fraction of the negative component of the steam coefficient of reactivity in those practical power reactors where considerations of neutron economy make it unattractive to achieve the negative coefficient by increasing the buckling (i.e., by increasing the neutron loss by leakage).

The fast fission factor. Only that component of the fast fission factor which arises from the mutual interaction of fuel elements with one another is affected by the water density in the reactor core. The effect is usually small, but it may become important for reactors which use a high ratio of uranium to water and fuel elements of small size.

The resonance escape probability. Although the resonance escape probability, p , depends upon the disposition of fuel in the lattice as well as on the fuel-water ratio, the effects of changes in water density (or of steam fraction) can be evaluated, if those changes are not too large, by assuming

$$p = \exp\left(-\frac{1}{A\rho}\right), \quad (4-9)$$

where A is a constant depending upon the fuel composition and its disposition in the lattice, and ρ is the water density. Then

$$\frac{dp}{p} = \left(-\ln p\right) \frac{d\rho}{\rho}. \quad (4-10)$$

Thus, it is possible to make the steam coefficient of reactivity as strongly negative as desired by increasing the resonance absorption, that is, by decreasing p . The term p would normally be decreased by decreasing the ratio of water to uranium in the lattice.

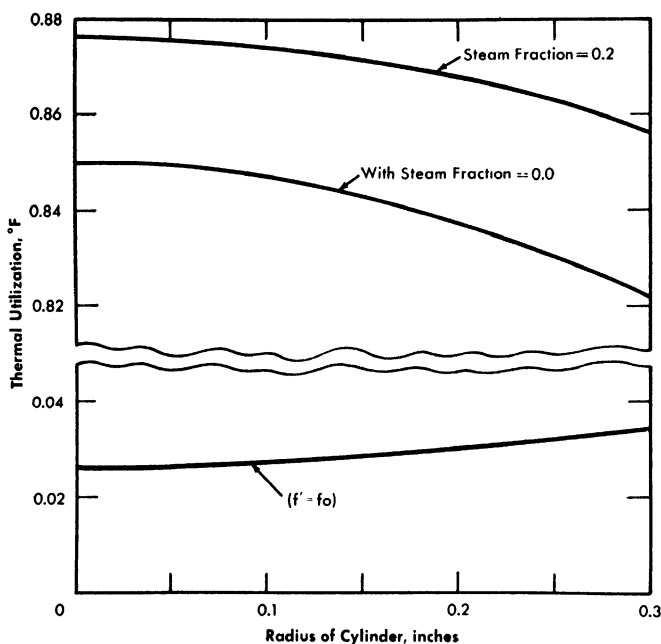


FIG. 4-16. Calculated thermal utilization for lattice of UO_2 cylinders in H_2O at room temperature.

Effects of disadvantage factors. The approximate expressions for the components of the steam coefficients of reactivity due to thermal neutron absorption by the water and resonance escape probability as contained in Eqs. (4-5) and (4-10) respectively, ignore the effects of thermal and resonance disadvantage factors. Figures 4-16 and 4-17 have been prepared to give a rough idea of the effects of these disadvantages factors. Figure 4-16 shows the thermal utilization as calculated by simple diffusion theory for a lattice of uranium oxide cylinders in H_2O with various radii of the cylinders. The calculations apply to room temperature conditions but treat two cases of steam fraction: a steam fraction of zero and a steam fraction of 0.2. As one goes from the case of uniform mixture of UO_2 and water to the case in which the UO_2 cylinders have a radius of 0.3 in., the difference between the thermal utilization for zero steam fraction and that for a steam fraction of 0.2 increases by approximately 37 per cent. This accentuation of the effect of steam fraction on thermal utilization results from the fact that when the fuel is present in sizable lumps a decrease in average water density not only reduces the amount of water present for neutron absorption, but decreases the thermal disadvantage factor between the fuel and the water as well.

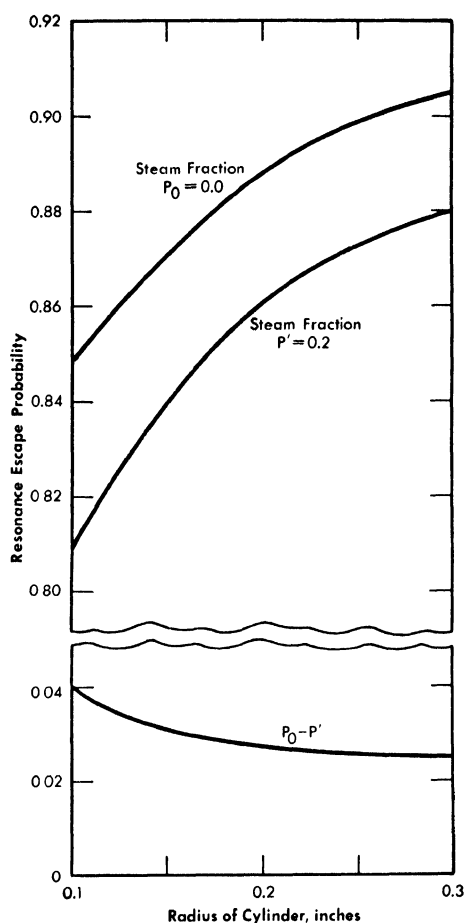


FIG. 4-17. Calculated resonance escape probability for lattice of UO_2 cylinders in H_2O at room temperature.

In Fig. 4-17 the calculated value for resonance escape probability are given for the same conditions. Here the effect of rod size is the opposite of that observed in the case of the thermal utilization, the larger rod sizes leading to smaller differences between the resonance escape probability for full density water and water containing 20 per cent steam by volume. It is to be noted that in both the case of the thermal utilization and that of the resonance escape, the effect of increasing the size of the fuel lumps is to make the net effect of thermal utilization and resonance escape probability on steam coefficient of reactivity somewhat more positive.

Calculation of steam coefficient of reactivity. The discussion of steam coefficient of reactivity can be summarized by writing an equation for the reactivity change $dk_{\text{eff}}/k_{\text{eff}}$ resulting from a change in steam fraction, as a sum of several components:

$$\frac{dk_{\text{eff}}}{k_{\text{eff}}} = \frac{d\epsilon}{\epsilon} + \frac{dp}{p} + \frac{df}{f} - \frac{M^2 B^2}{1 + M^2 B^2} \left[\frac{dM^2}{M^2} + \frac{dB^2}{B^2} \right], \quad (4-11)$$

where it is understood that the derivative in each case is to be taken with respect to the steam fraction, α , to give the steam coefficient of reactivity, $(dk_{\text{eff}}/k_{\text{eff}})/d\alpha$. If the two-group approximation (rather than the one-group) is used, the last term of equation 11 expands to

$$- \left[\frac{L^2 B^2}{1 + L^2 B^2} \frac{dL^2}{L^2} + \frac{L_f^2 B^2}{1 + L_f^2 B^2} \frac{dL_f^2}{L_f^2} + \left(\frac{L^2 B^2}{1 + L^2 B^2} + \frac{L_f^2 B^2}{1 + L_f^2 B^2} \right) \frac{dB^2}{B^2} \right], \quad (4-12)$$

where L^2 is the thermal diffusion area and L_f^2 is the slowing down area.

The terms dp/p and df/f can be approximated roughly by the simple expressions

$$\frac{(dp/p)}{d\alpha} = \frac{\ln p}{1 - \alpha}, \quad (4-13)$$

(note that since $p < 1$, this term is inherently negative) and

$$\frac{(df/f)}{d\alpha} = \frac{f_{\text{H}_2\text{O}}}{1 - \alpha}, \quad (4-14)$$

where $f_{\text{H}_2\text{O}}$ is the fraction of thermal neutron absorption which are absorptions by H_2O .

In many specific cases simple approximations can be arrived at for the variations of the other terms of (4-11) or (4-12) with steam fraction, but simple general expressions do not exist. Often such approximations are adequate for roughing out the design of a boiling reactor, but even the best and most detailed calculations are none too good for establishing a final design. The usual procedure for converging on the design point is to pick that operating condition which is to define the steam coefficient of reactivity, and to calculate the reactivity at that point for the two conditions, that of zero steam fraction and that of some more convenient finite and relatively

small steam fraction. A steam fraction of 0.1 is often convenient for the second calculation. Needless to say, an accurate theoretical treatment of the reactor physics is more important in computing steam coefficient of reactivity than in computing reactivity alone. Many of the recipes which have given reasonably satisfactory results for reactivity in the past are inadequate for calculating the steam coefficient of reactivity, and in fact it can be said that proved methods for calculation of the coefficient of reactivity have not been established up to the present time./

Calculation of steam coefficient of reactivity in EBWR. It is instructive to consider the several components of the steam coefficient of reactivity and their variation between room temperature and operating temperature as calculated for EBWR. The reactor has an average volume ratio of water to uranium metal of 3.5, and an average enrichment of 1.4 per cent U^{235} in U^{238} . The calculated components of the steam coefficient are given in Table 4-2. These values were obtained in a study of an EBWR type reactor, not the actual EBWR. It should be noted that the coefficient at the operating temperature of 486 F is larger by almost a factor of 5 than at room temperature.

The EBWR is a relatively small reactor. In reactors of sizes attractive for economic power generation, it may be expected that the contribution of neutron leakage to the negative steam coefficient will be considerably smaller than in EBWR, and in fact may do little more than compensate for the positive contribution of the fast fission factor. Roughly it may be said that one will achieve the desired coefficient of reactivity by balancing the effect of resonance escape probability against the effect of the thermal utilization.

Limitations on attainable values of steam coefficient of reactivity. Table 4-2 demonstrates that the negative coefficient of reactivity will increase with increasing temperature and increasing steam content for a slightly enriched reactor, just as for the fully enriched reactor. The basic reason for this increase is the same as that illustrated by Fig. 4-14. The net coefficient represents a relatively small difference between relatively large components which do not remain constant as temperature and steam content change. Since it is desirable to minimize this change in coefficient, it would be desirable to minimize the magnitude of the major components of the coefficient. Unfortunately once the fuel is chosen, there are no longer sufficient design variables available to allow one to adjust the size of the components of the coefficient. The controlling component of the steam coefficient is that which is associated with the thermal utilization. To decrease this component one must decrease the fractional absorption of neutrons in the water. In principle this could be done either by increasing the

TABLE 4-2
EFFECT OF STEAM IN EBWR*

Item	68 F			486 F		
	Steam fraction = 0	Steam fraction = 0.10	Fractional change Steam fraction	Steam fraction = 0	Steam fraction = 0.15	Fractional change Steam fraction
η	1.535	1.535		1.535	1.535	
ϵ	1.037	1.039	+0.019	1.042	1.042	+0.019
P	0.870	0.859	-0.128	0.840	0.819	-0.17
f	0.787	0.800	+0.162	0.813	0.828	+0.12
k	1.090	1.096	+0.055	1.093	1.089	-0.024
τ , cm ²	34.75	39.74		47.09	56.96	
L^2 , cm ²	3.02	3.02		6.53	7.49	
Refl. saving, cm	7.02	7.54		9.50	10.62	
B_g^2 , 10 ⁴ cm ⁻²	16.66	16.42		15.55	15.08	
M^2 , cm ²	37.77	43.06		53.62	64.45	
$1 + M^2 B_g^2$	1.0630	1.0707	+0.071	1.0834	1.0972	+0.084
k_{eff}	1.0252	1.0229	-0.022	1.0082	0.992	-0.107

* Calculated by J. A. The, Argonne National Laboratory.

enrichment of the fuel or by increasing the ratio of the fuel to water in the reactor. However, for a given ratio of water to fuel, the enrichment is specified by the reactivity requirement, unless one is willing to waste neutrons deliberately in parasitic absorbers. The ratio of fuel to water is fixed, in turn, by the requirement that the net steam coefficient be negative at the lowest conceivable operating temperature. Thus, one usually designs for a very low negative value of the coefficient at room temperature and accepts whatever coefficient results at the normal operating temperature. If one does consider variations in the fuel elements, it is evident that the higher the macroscopic thermal cross section of the fuel the smaller will be the variations in steam coefficient with temperature and with steam fraction. As used here the term "fuel" is intended to include fissionable and fertile material and cladding, and in fact any material which is not associated either with the water or with control rods. It is usually not economic, however, to increase this cross section simply by the addition of parasitic absorber, unless additional benefits are derived from the absorber. Fuels which incorporate fertile material with high macroscopic thermal absorption cross section, however, are more attractive with respect to steam coefficient than are fuels which have lower fertile cross sections. Thus uranium metal is more attractive in this respect than uranium oxide.

Limitations on conversion ratio. In most reactors the conversion ratio may be increased, within the fundamental limits set by the value of η for the fissionable material, by increasing the enrichment and at the same time changing the lattice in such a way as to increase the resonance absorption in fertile material; η , of course, represents the number of fission neutrons per neutron absorbed by the fuel. Thus, one decreases the fractional absorption of neutrons by parasitic materials and increases the fractional absorption by fertile materials, and pays for this improvement by using fuel of higher enrichment. In the boiling reactor the extent to which this adjustment of neutron economy may be exploited is limited by the requirement that the steam coefficient of reactivity be kept as low as possible. If one arbitrarily increases the resonance absorption far above that which is dictated by this consideration, the coefficient may become so large that the power output of the reactor is severely limited. From the standpoint of achieving high conversion ratio with an attractively low steam coefficient, a fertile material having a high ratio of thermal absorption cross section to resonance integral is most attractive. Thus, for example, higher conversion ratio may be obtained using thorium oxide as the fertile material than can be obtained when one uses uranium oxide as the fertile material.

Effect of fuel type on reactor physics. As already indicated, the use of fuel having high macroscopic thermal neutron cross section for fertile ma-

terial favors the attainment of higher conversion ratio and minimization of the variation in steam coefficient of reactivity. From these standpoints uranium metal is preferable to uranium oxide; and thorium oxide, because of the high microscopic thermal absorption cross section of thorium, is also preferable to uranium oxide. The difference between uranium and thorium oxides as fuels is illustrated by the data in Tables 4-3 and 4-4, which summarize the results of some preliminary calculations on a boiling H_2O reactor fueled alternatively by UO_2 and by $\text{ThO}_2\text{-U}^{235}\text{O}_2$ mixture. The components of the reactivity calculation were somewhat different from those which have been used in the previous discussion, as a three-group approximation was used for the calculation. The basic physical characteristics of the reactors are tabulated in Table 4-3, and the components of the steam coefficient of reactivity are given in Table 4-4 for two different temperatures, 68 and 350 F. The total steam coefficient of reactivity is also given at the temperature 533 F, the contemplated operating temperature. It is to be noted that although the total steam coefficient is nearly the same for the two reactors at room temperature, the major components of the coefficient are quite considerably larger in the UO_2 fueled reactor. The result is that at the operating temperature of 533 F, the negative steam coefficient for the UO_2 case is almost twice that for the ThO_2 case. The initial conversion ratio was computed to be 0.72 in the ThO_2 case and 0.69 in the UO_2 case. These figures, of course, are true for the assumed cases; they are not necessarily representative.

Effect of spatial distribution of steam on flux and power distribution.

Since the boiling reactor is operated with a negative steam coefficient of reactivity, it may be expected that if the spatial distribution of steam in the reactor is not uniform, the neutron flux will tend to be relatively lower in regions of high steam content than it would be if the steam distribution were uniform. Considering the distribution of steam and flux in the radial direction, it may be said that if no other attempts are made to flatten the neutron flux, the characteristic flux shape for the reactor, with the highest value of flux and therefore the highest value of power at the reactor center, will result in a comparable distribution of steam fraction, with the highest fraction at the center of the reactor. The presence of this extra steam near the center will in turn reduce the flux peak, and the flux and steam distributions will arrive at some equilibrium distribution, in which the neutron flux and power density are somewhat flatter than in a reactor of uniform composition. Figure 4-18 shows an estimate of the steam distribution radially in EBWR, and the calculated resulting fast and thermal flux distributions. It will be noted that the radial maximum to average ratio with no steam in the core is computed to be 1.71.

In the axial direction, the steam content of the reactor core increases

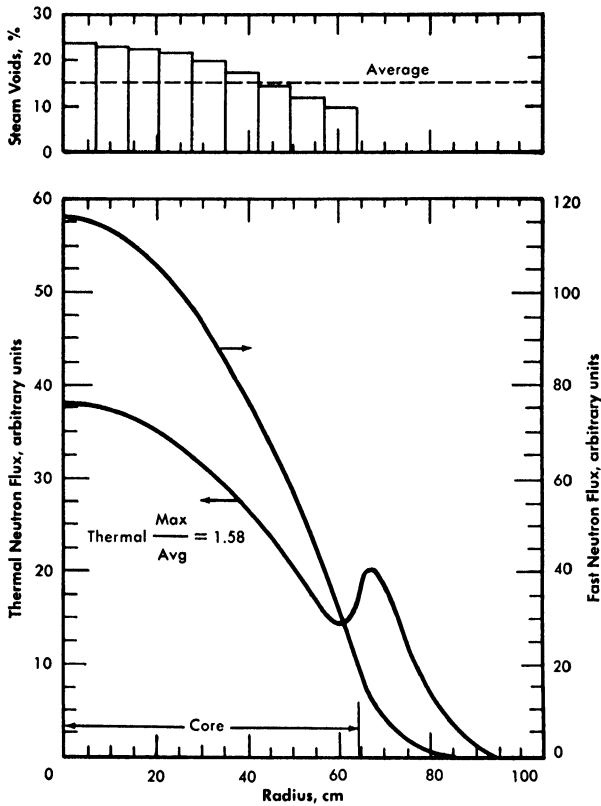


FIG. 4-18. Radial neutron flux and void distribution in EBWR.

continuously with greater height in the core, since all steam must issue from the top of the core. This steam distribution tends to cause the flux and power levels near the top of the core to be lower relative to those at comparable positions near the bottom of the core. Figure 4-19 shows the computed axial flux distribution in EBWR. The departure from symmetry about the mid-plane of the core is not serious in this case. It is possible that for very large steam fractions the effect may become great enough to affect seriously the axial maximum to average power ratio. This situation is more likely to occur in large reactors, because for a given steam velocity and a given power density at a given pressure, the steam fraction at the exit of the reactor core will be larger for large reactors than for small reactors.

D₂O moderated boiling reactors. At present so far as is known only one D₂O moderated boiling reactor has been constructed. This is the Halden reactor in Norway which was expected to become critical in the summer of

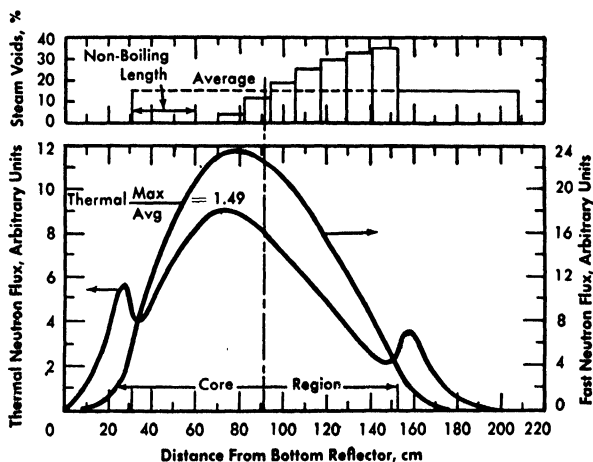


FIG. 4-19. Axial flow and void distribution.

1958. Little actual data are available on this installation except that it is designed for a thermal capacity of 20 Mw. This project is noteworthy not only in that it will be the first D_2O moderated boiling reactor but also in that it will be used to furnish steam for heating in an industrial establishment.

Since the absorption cross section of D_2O is small, there is little probability of encountering a positive steam coefficient of reactivity in the design of the D_2O reactor. The variation in steam coefficient of reactivity with temperature and with steam fraction will be much smaller than in the H_2O case. The contribution of neutron leakage to the negative component of the steam coefficient may be large unless the reactor is made quite large, because of the large migration area in typical D_2O moderated reactors. In principle the contribution of the resonance escape probability to the steam coefficient can be made as small as is desired by making the resonance escape probability correspondingly large. Other considerations, however, notably the desire for high conversion ratio and high power density, will limit the extent to which one is willing to increase the resonance escape probability. It is believed that the most attractive type of boiling D_2O reactor will be one in which the fuel elements are disposed in clusters, with isolated moderator between the clusters, and with forced circulation of the coolant-moderator among the elements in the clusters. With such an arrangement the required average steam fraction for a given power density may be smaller than in the case of the H_2O moderated reactor. Furthermore, the displacement of water by steam within a fuel element cluster will have a smaller effect on resonance escape than would the equivalent

TABLE 4-4
COMPONENTS OF STEAM COEFFICIENT OF REACTIVITY FOR REACTORS FUELED WITH UO_2 AND $\text{ThO}_2\text{-U}^{235}\text{O}_2$

	ThO ₂			UO ₂		
	68 F	350 F	533 F	68 F	350 F	533 F
$\ast \left(\frac{dk}{dv} \right)_{\text{total}}$	-1.14×10^{-2}	-7.18×10^{-2}	-9.3×10^{-2}	-1.79×10^{-2}	-9.02×10^{-2}	-17.3×10^{-2}
$\left(\frac{\partial k}{\partial v} \right)$	$+0.40 \times 10^{-2}$	$+0.47 \times 10^{-2}$		$+1.58 \times 10^{-2}$	$+1.87 \times 10^{-2}$	
$\left(\frac{\partial k}{\partial v} \right)_1$	-4.51×10^{-2}	-6.25×10^{-2}		-4.42×10^{-2}	-6.13×10^{-2}	
$\left(\frac{\partial k}{\partial v} \right)_2$	-2.63×10^{-2}	-3.86×10^{-2}		-2.46×10^{-2}	-3.54×10^{-2}	
$\left(\frac{\partial k}{\partial v} \right)_{L^2}$	-0.27×10^{-2}	-0.51×10^{-2}		-0.42×10^{-2}	-0.79×10^{-2}	
$\left(\frac{\partial k}{\partial v} \right)_p$	-9.81×10^{-2}	-12.24×10^{-2}		-17.84×10^{-2}	-22.12×10^{-2}	
$\left(\frac{\partial k}{\partial v} \right)_a$	$+15.36 \times 10^{-2}$	$+14.80 \times 10^{-2}$		$+21.45 \times 10^{-2}$	$+21.27 \times 10^{-2}$	
$\left(\frac{\partial k}{\partial v} \right)_{B^2}$	$+0.32 \times 10^{-2}$	$+0.41 \times 10^{-2}$		$+0.32 \times 10^{-2}$	$+0.42 \times 10^{-2}$	

$\ast \frac{dk}{dv} = \frac{\text{fractional change in } k_{\text{eff}}}{\text{steam fraction producing change in } k_{\text{eff}}}$

displacement over the entire volume of the reactor core. When water is displaced within the fuel cluster, the resulting decrease in slowing down power is to some extent compensated by an increase in the resonance self-shielding of the fertile material.

HEAT REMOVAL FROM BOILING WATER REACTORS

The thermal power of any boiling water reactor, including those using natural and forced circulation and cycles utilizing the steam directly or indirectly for electric power generation, can be expressed in terms of the total coolant flow rate, the subcooling of the coolant at the core inlet and the mass flow rate of steam leaving the core. The heat balance relating reactor thermal power and those parameters is given by equation (4-15) as follows:

$$Q_t = (W_s + W_c)(X_e h_{fg} + \Delta h_r). \quad (4-15)$$

The right-hand multiplier of equation (4-15) denotes that the total thermal power is distributed between sensible heat gain of the coolant up to the boiling temperature and evaporation at constant temperature. Physically, within the fuel element flow channels of a boiling reactor, a corresponding proportion of the core height is in the nonboiling region, and the remainder of the height is populated with steam bubbles, the bubble density depending upon the thermal (heat) flux.

The fractions of the core height wherein nonboiling and boiling occur are, for the case of the thermal heat flux being constant over the core height, proportional to the corresponding power fractions expended in raising the coolant temperature and the production of steam respectively. Although the axial heat generation within the core of a boiling reactor is a cosine distribution, and radially the heat flux varies as a Bessel function, for this analysis the reactor may be represented as if the average heat flux were uniform for all heat transmitting surfaces of the core. Later, it will be shown that the error in assuming the heat flux constant, instead of the actual heat distribution pattern, yields a relatively small difference in reactor power from a given core density.

Expressing the reactor thermal power in terms of the active core height and boiling length, equation (4-15) can then be converted to

$$Q_t = (W_s + W_c) X_e h_{fg} \left(\frac{L_t}{L_b} \right). \quad (4-16)$$

Since the power level of thermal water-moderated reactors is a function of the moderator density, a boiling water reactor differs from a pressurized water reactor in that density changes are due not only to moderator

temperature change but also to the formation of the less dense steam phase in the bulk of the moderator. In a boiling reactor, a relatively small change in moderator density occurs as the subcooled water passes upward through the fuel element channels and is heated, with some local or surface boiling, to saturation temperature. The reduction in density due to an increase in water temperature is relatively small for the existing and proposed boiling reactors as compared to the operating and proposed pressurized water reactors. The major reduction in moderator density occurs above that height in the channels of a boiling reactor when sufficient heat is transmitted to raise the temperature to saturation conditions. From that core height where bulk boiling starts to the core outlet, there is a continuous reduction in density. It is the effect of this major reduction in density in the boiling zone on neutron balance which gives the boiling reactor the desirable feature of self-power regulation and the inherent ability to terminate power excursions.

Equation (4-16) can be converted to a more applicable form involving the coolant moderator core voids by substituting the relationship between steam weight fraction and steam volume fraction and the relationship between the exit steam volume fraction and the average value for the core, excluding the lesser heated water within the control rod flow passages:

$$X_e = \frac{[\alpha_e/(1 - \alpha_e)] (C_s/C_l)_e (\rho_s/\rho_l)}{1 + [\alpha_e/(1 - \alpha_e)] (C_s/C_l)_e (\rho_s/\rho_l)} \quad (4-17)$$

$$\alpha_e = \frac{\bar{\alpha}_c (L_t/L_b)}{Y} \quad (4-18)$$

The result of the substitution of equation (4-17) and equation (4-18) in equation (4-16) yields a general power equation for boiling reactors:

$$Q_t = (W_s + W_c) \left(\frac{C_s}{C_l} \right)_e \left(\frac{\rho_s h_{fg}}{\rho_l} \right)' \left(\frac{L_t}{L_b} \right) \left[\frac{\frac{\bar{\alpha}_c (L_t/L_b)}{(Y - \bar{\alpha}_c (L_t/L_b))}}{1 + \left(\frac{C_s}{C_l} \right)_e \left(\frac{\rho_s}{\rho_l} \right) \left\{ \frac{\bar{\alpha}_c (L_t/L_b)}{[Y - \bar{\alpha}_c (L_t/L_b)]} \right\}} \right] \quad (4-19)$$

Equation (4-19) may be simplified for small values of $(C_s/C_l)_e (\rho_s/\rho_l)$ and low steaming rates such that the term

$$\left(\frac{C_s}{C_l} \right)_e \left(\frac{\rho_s}{\rho_l} \right) \left\{ \frac{\bar{\alpha}_c (L_t/L_b)}{Y - \bar{\alpha}_c (L_t/L_b)} \right\} \approx 0.$$

For this case, the power equation then becomes:

$$Q_t = (W_s + W_c) \left(\frac{C_s}{C_l} \right) \left(\frac{\rho_s h_{fg}}{\rho_l} \right) \left(\frac{L_t}{L_b} \right)^2 \left[\frac{\bar{\alpha}_c}{Y - \bar{\alpha}_c(L_t/L_b)} \right]. \quad (4-20)$$

To evaluate how the power level of a boiling reactor is affected by the control variables indicated in equation (4-19) and (4-20), it is first necessary to establish the inter-relationship between the parameters themselves and then their individual and combined effect on the power level. These include slip ratio, core void fraction, pressure, subcooling, flow, and natural circulation.¹

(a) *Slip ratio* (C_s/C_l). It has been well demonstrated in both static and upward flowing 2-phase mixtures that the less dense component, due to buoyancy, moves upward at a faster rate than the more dense phase. This slippage of the light phase relative to the other mixture component affects and controls the volume occupied by the light phase and, in turn, establishes the density of the mixture in accordance with equation (4-21):

$$\rho = (1 - \alpha) \rho_l + \rho_s \alpha = \rho_l \left[1 - \alpha \left(1 - \frac{\rho_s}{\rho_l} \right) \right]. \quad (4-21)$$

The factor which controls the volume proportion of the steam in a 2-phase boiling mixture for a given weight proportion of steam in the mixture is the slip ratio (C_s/C_l) or the relative velocity ($C_s - C_l$). The definition of (C_s/C_l), by revising equation (4-17), is given by

$$\left(\frac{C_s}{C_l} \right) = \left(\frac{X}{1 - X} \right) \left(\frac{1 - \alpha}{\alpha} \right) \left(\frac{\rho_l}{\rho_s} \right). \quad (4-22)$$

The experiments conducted by Argonne National Laboratory described briefly in Chapter 3, simulating reactor heat release by electrical resistance heating of single and multiple elements, with various sizes and lengths of flow passages and with both natural and forced circulation at pressures up to 600 psi and power densities to 100 kw/liter, have indicated the following empirical results pertaining to the slip ratio (C_s/C_l):

1. The variation of C_s/C_l locally along the channel heated height is a function of the local weight proportion of steam in the mixture. Confirmation tests on non-heated risers where the steam quality is constant have shown that the slip ratio is constant along the riser height. Typical experimental results of the (C_s/C_l) variations along a test channel height are shown for the case of essentially saturated inlet water conditions at 600 psig in Fig. 4-20.

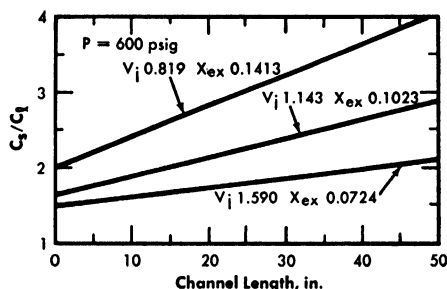


FIG. 4-20. Change in phase velocity ratio with channel position for saturated inlet temperatures.

2. The slip ratio at the exit of the heated channels was found to be mainly dependent on the system pressure and the velocity of the coolant at the inlet to the test section with a slight dependency on power density; the ratio decreased with increased velocity and increased pressure.

The bulk of the data obtained by Argonne was with natural circulation, and therefore the range of the inlet velocity parameters was restricted to approximately 3.0 ft/sec, maximum. Experimental data, limited to relatively low pressures of 150 and 250 psig, were also obtained with forced circulation. Inlet water velocities up to 11 ft/sec were attained in these tests. The experimental variation of the slip ratio with velocity obtained in these tests shows a decrease in slip ratio with increased velocity, with the rate of decrease in slip ratio being much less than that for the low velocity regions of natural circulation.

3. Argonne tests also indicated that the slip ratio at the channel exit could be correlated over a wide range of subcooling and velocity with the parametric grouping $X_e(L_t/L_b)$.

As to how the reactor power depends upon the slip ratio, from equation (4-19) it appears, for all other conditions being constant, that the power level is directly proportional to the exit slip ratio. An analytical solution of equation (4-19) for the case of constant flow forced circulation subcooled cycle at 900 psig and core void fraction constant at 0.20, shown in Fig. 4-21, indicates that for subcooling conditions, the total power level gain is much less than that due to an increase in the slip ratio although the power in the boiling height is directly proportional to an increase in the slip ratio.

(b) *Core void fraction.* The average steam void fraction within the core of a boiling reactor, neglecting the water region associated with the control rods, has previously been shown by equation (4-18) to be a function of the integrated void fraction within the boiling height and the proportion of the active core height within which the steam is formed. For a specific core exit density or steam volume fraction, the mean void fraction within

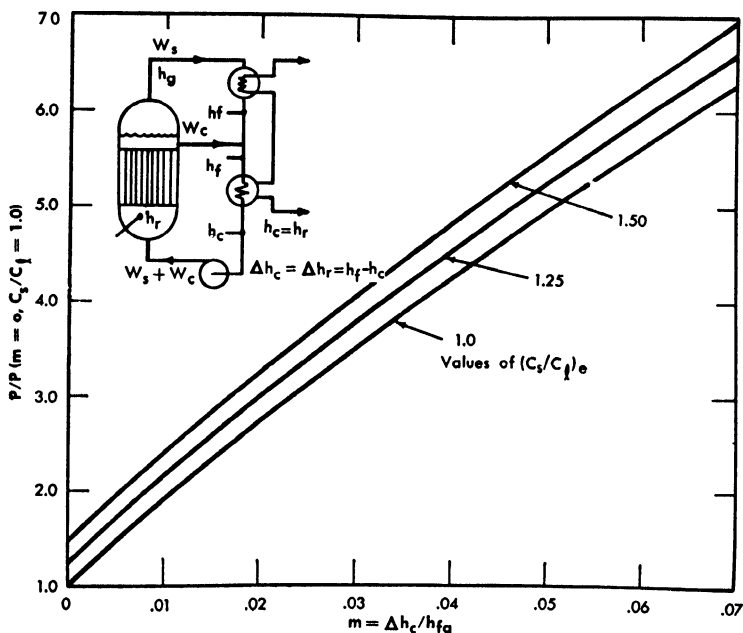


FIG. 4-21. Typical effect of exit slip ratio (C_s/C_l) on reactor power for a forced circulation constant flow cycle ($P = 900$ psia, mean core voids $\bar{\alpha}_c = 0.20$).

the boiling height actually depends on the variation of the heat flux along the heated channel length.

In a reactor with the fissionable material distributed uniformly in the fuel elements throughout the core, the power distribution axially is usually represented by a cutoff cosine distribution. For light-water moderated reactors, the error in assuming the power generation initiates and terminates at the inlet and exit of the fuel zone and the maximum heat flux is at the midheight of the fuel zone is relatively small. For a given pressure and mean core void fraction, the major difference between uniform and symmetrical cosine heat distribution is the effect on the height of the channel in boiling and the value of the steam volume proportion leaving the channel. Figure 4-22 illustrates these major differences. It should be noted that for external cycle conditions which result in a relatively large fraction of the core used for increasing the sensible heat of the coolant, i.e., for small values of Q_b/Q_t , boiling starts much sooner with a cosine heat flux pattern in comparison to the case where the heat flux is constant over the heated length. Although the boiling zone is more extensive with the cosine type heat flux, the volume occupied by the steam at the core exit, for a constant core void fraction, is correspondingly at a lower level than with a uniform

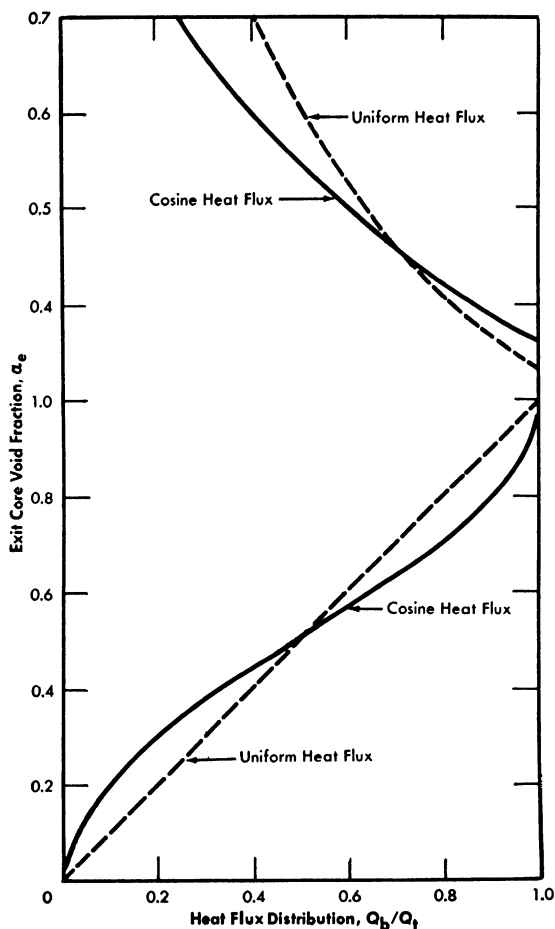


FIG. 4-22. Effect of heat flux distribution on boiling length fraction and exit void fraction for $P = 600$ psia ($C_s/C_l = 1.0$, $\bar{\alpha}_e = 0.20$).

heat flux. Although the experimental data available have shown that the slip ratio increases along the channel heated height, the error involved in assuming the slip ratio constant for calculating the relationship between $\bar{\alpha}_b$ and α_e is relatively small for all practical purposes. The slip ratio has a relatively small effect on the density ratio at moderate pressures around 500 psi, but as the pressure increases, the dependence of the mean density ratio on the slip ratio becomes more important.

A boiling reactor is capable of a wide variation in power level at constant moderator density and control rod setting. With fixed setting of the control rods, a reactor operating at constant pressure will seek some power

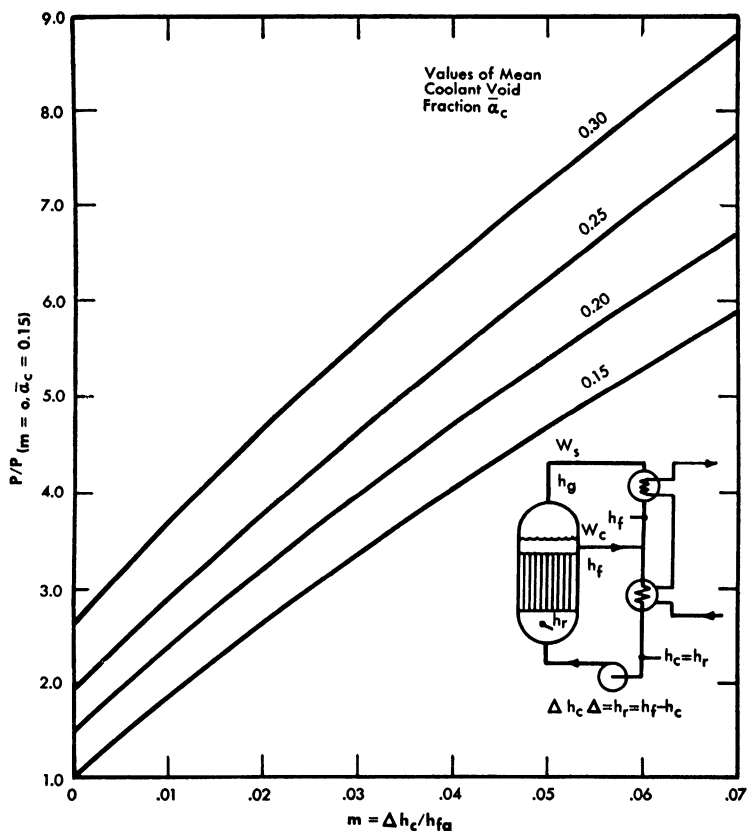


FIG. 4-23. Typical effect of mean core voids $\bar{\alpha}_c$ on reactor power for a forced circulation constant flow cycle ($P = 900$ psia, $C_s/C_l = 1.5$).

level to maintain the moderator density or volume of steam within the core constant, whenever the external conditions affect the flow through the core or the subcooling at the core entrance. Control rod withdrawal from the core will increase the reactor power to some level such that increased neutron leakage, resonance absorption, etc., resulting from higher steam population within the core compensates the additional reactivity.

The power gain that results from permitting the void fraction to increase, such as by rod withdrawal, keeping other conditions constant, is a function of the change in the core void fraction. This is illustrated by Fig. 4-23, which shows, for a typical forced circulation cycle and various average core void fractions, the theoretical power increase as a function of the subcooling. It should be noted for the case of boiling over the entire core height, zero inlet subcooling ($m = 0$), the power gain is not exactly proportional to

the increase in void fraction. For instance, increasing the mean core void fraction from 0.15 to 0.30, a factor of 2 increase, theoretically, from analysis of equation (4-19), results in a power increase of approximately 2.6.

As the subcooling is increased, an increase in the void fraction for constant subcooling results in a power gain, but less than that obtained with saturated inlet conditions. For instance, for a subcooling equivalent to 2 per cent of the latent heat h_{fg} , increasing the core voids from 0.15 to 0.30 results in a power gain factor of 1.75, compared to the factor of 2.6 noted above.

(c) *Pressure.* The effect of operating pressure on the power level of a boiling water reactor depends on the pressure effect on the parameters $(C_s/C_l)_e(\rho_s h_{fg}/\rho_l)$ and (L_t/L_b) in equations (4-19) and (4-20). The experimental results correlating pressure and slip ratio, with the slip ratio decreasing as the flow and pressure are increased, have already been discussed.

The major properties of steam that vary with pressure are the temperature, liquid and steam density, latent heat, surface tension, and viscosity. Pressure has the effect of reducing the size of the bubbles in the steam-water mixture; thus it is evident that by increasing the operating pressure of a reactor, a larger power output can be obtained for a given core void fraction. Associated with an increase in the pressure is the reduction in the quantity of heat required to form unit weight of steam; the combination of reduced specific volume and latent heat of steam results in a heat of vaporization per unit volume of steam which increases with pressure. Considering, also, the change of liquid density with pressure and neglecting the other properties such as surface tension and viscosity, the grouping $(\rho_s h_{fg}/\rho_l)$ of the physical properties of the fluid is found to be linear with pressure.

In addition to the decrease in latent heat of the steam with increased pressure, there is an increase in the saturation temperature with increasing pressure. The combined result of these two effects is to alter the distribution of steam within the core height as the pressure is changed. Figure 4-24 illustrates typical theoretical results of the effect of pressure on the distribution of the boiling zone for constant flow, constant core void fraction, natural and forced circulation cycles. The curves show that for constant subcooling the fraction of the core height in the boiling zone is increased by increasing the pressure.

The combined effect of the pressure-sensitive variables $(C_s/C_l)_e(\rho_s h_{fg}/\rho_l)$ and (L_t/L_b) on reactor power depends on the subcooling of the reactor water at the core inlet. For saturated water inlet conditions, the reactor power level varies directly as the product $(C_s/C_l)_e(\rho_s h_{fg}/\rho_l)$; doubling the pressure will result in a power increase slightly less than the pressure gain; because of the decrease in the slip ratio with increased pressure. It is assumed in this analysis that the reactor temperature coefficient of re-

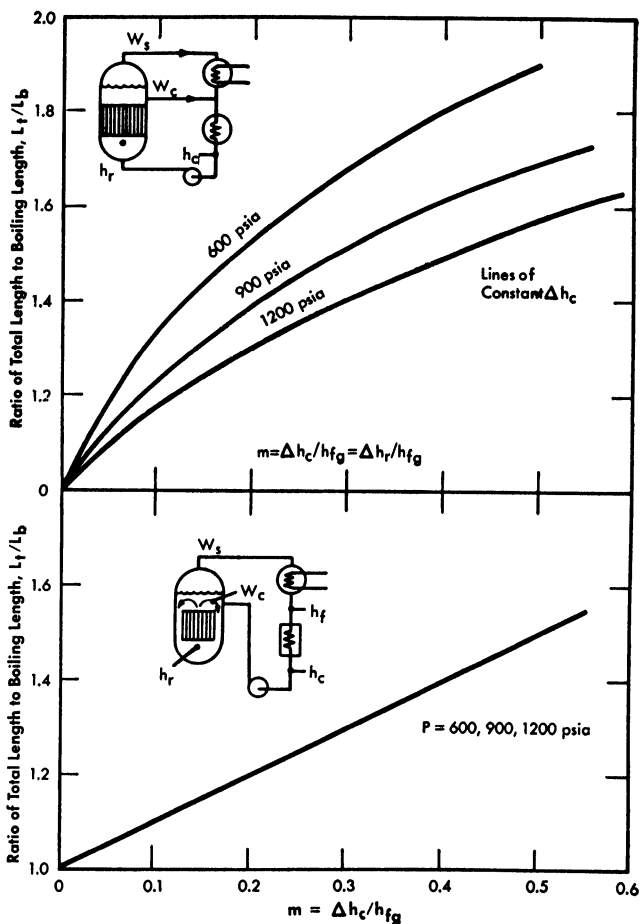


FIG. 4-24. Effect of pressure on the core boiling length for constant core void fraction (0.20) for natural and forced circulation cycles with uniform heat generation.

activity does not offset the gain due to a pressure increase. In an operating reactor, an increase in pressure for the same rod setting will result in an increase in moderator temperature, and the actual power gain will thus be reduced due to the temperature rise of the moderator.

For the case of subcooled water at the core inlet, a pressure increase will not produce the same power gain as that resulting with the boiling over the full core height. The power gain is reduced due to the increase in the L_t/L_b ratio which results with pressure increase.

(d) *Subcooling.* A method of increasing reactor power level for the same volume of steam in the core is by reducing the temperature in the water

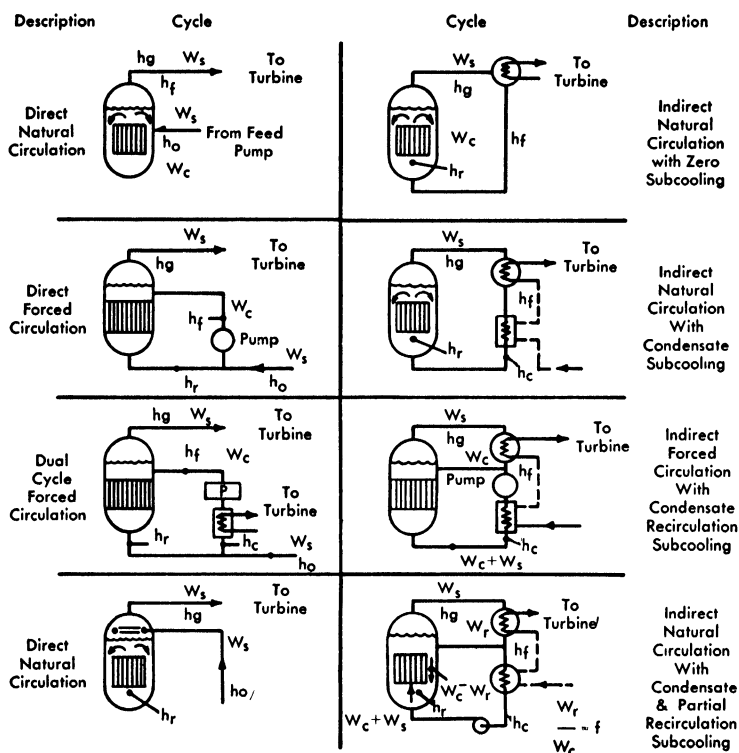


FIG. 4-25. Boiling reactor cycles.

to the core. By changing the differential between the saturation temperature and the actual inlet temperature, the distribution between the non-boiling and 2-phase heating processes in the core is altered. An increase in the subcooling pushes the point at which boiling is initiated higher up in the core and results in a higher steaming rate within a shorter boiling height to maintain the same core density. Figure 4-26 illustrates for natural and forced circulation cycles, at various pressure levels, how the boiling height is reduced as the subcooling fraction is increased.

The subcooling at the core inlet depends on the reactor cycle, i.e., whether natural or forced circulation is used or whether the generated steam is used directly or indirectly through heat exchangers in the power conversion cycle and also on the power level. Some of the many reactors in use or being proposed are shown in Fig. 4-25. The cycles shown in these diagrams indicate that the subcooling at the core inlet can be controlled by (a) cooling of the steam condensate, (b) absorbing sensible heat from the recirculated water, and (c) combination of both of these.

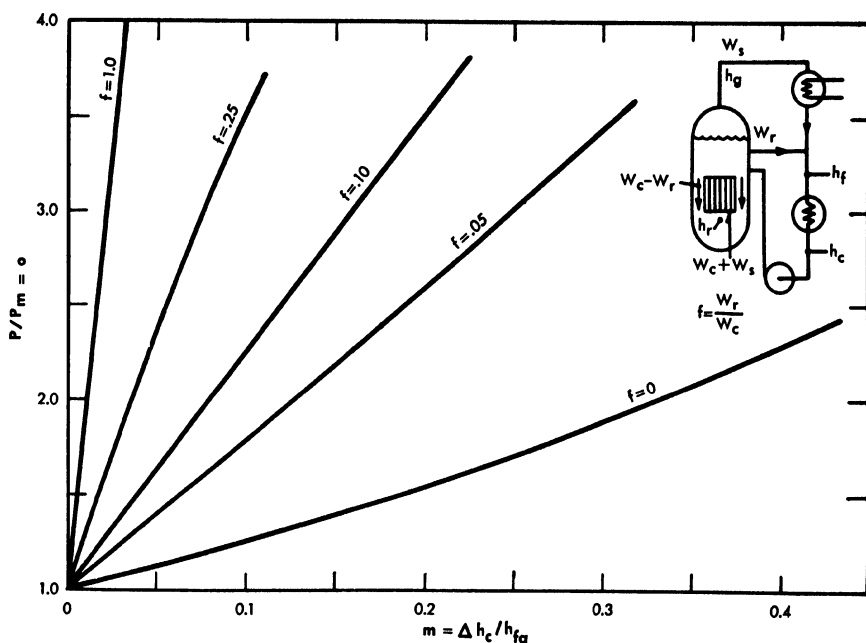


FIG. 4-26. Effect of reactor power level of subcooling portions of recirculation flow in a constant flow natural circulation cycle ($P = 600$ psia, $\bar{\alpha}_c = 0.20$).

For the present operating power reactors, the BORAX-IV and the EBWR (direct cycle, natural circulation reactor systems), the feedwater return to the core is maintained at a relatively low temperature established by the turbine condenser vacuum. Although such a non-regenerative cycle is less efficient than the regenerative cycle using steam extraction and feedwater heating, for the same core density the reactor power capability is greater than for a regenerative cycle which effectively reduces the subcooling at the core inlet. For the 300 psig BORAX-IV and the 600 psig EBWR cycles, nonboiling occurs in these non-regenerative cycles over approximately 25 and 40 per cent respectively, of the core height.

The forced circulation dual cycle for the Dresden boiling reactor is one utilizing regenerative heating of the feedwater. The subcooling of the water at the core inlet to this reactor results in nonboiling over approximately 50 per cent of the core height and is obtained by cooling all of the reactor water that circulates through the core to a level determined by the final temperature from the last regenerative heater. This case differs from the EBWR, in which a small quantity of the total flow through the core is subcooled a large amount (370 F) by work extraction in the turbine and condensing. Typical gains in power level resulting from subcooling

either the condensate or the total flow are shown for natural and forced circulation reactor cycles in Fig. 4-21 for various values of slip ratio, and in Fig. 4-23 for various mean core voids. Figure 4-26 illustrates how the power level can be increased with a natural circulation cycle by subcooling various fractions of the water recirculated.

As was previously noted, subcooling essentially changes the boiling height fractions (L_b/L_t). It can be inferred from equation (4-20) that for all other conditions being the same and changing the subcooling only, the power level will vary as $(L_t/L_b)^2$. Using the more exact equation (4-19), the effect of a change in the boiling length (L_b) will result in a power increase greater than that indicated by equation (4-20). For 600 psia, the power varies approximately as $(L_t/L_b)^{2.5}$. At higher pressures, the gain is slightly reduced.

(c) *Flow*. In addition to the pressure level and subcooling, the other sole means of increasing the power level of a reactor at a given void fraction is by increasing the flow through the fuel element flow channels. For forced circulation cycles, this can be done by increasing the pumping flow rate. With a natural circulation power reactor, the power level can be increased by incorporating in the core design a chimney or riser above the core. As to the effect of increasing the flow rate ($W_s + W_c$) on the reactor power, the experimental evidence available indicates that for all other conditions constant the power varies directly as the product $(W_s + W_c)(C_s/C_l)_c$. This indicates, since the experimental data show that (C_s/C_l) decreases with increased flow, that the power gain is slightly less than the fractional increase in flow rate.

(f) *Natural circulation*. In a reactor using natural circulation of the coolant, the driving head promoting the flow is created by the difference in density between the liquid phase in the downcomer and the 2-phase mixture in the core coolant channels. The flow established is such that the sum of all the head losses in traversing the flow cycle is equal to this driving head. The components of the total head loss usually are the following:

1. Contraction loss in going from an inlet plenum into the discrete coolant channels.
2. The friction loss in the subcooled length of the flow channel.
3. The friction loss in the evaporation zone.
4. The pressure drop required to accelerate the flow from inlet to outlet.
5. An expansion loss in passage from the coolant channels to an upper plenum or chimney.
6. The isothermal friction drop in passing through the non-heated chimney or riser above the core.
7. The turning loss in directing the separated water back and down through the downcomer.

The relationship between the available driving head and the parasitic head losses may be expressed in equation form as follows:

$$\left(1 - \frac{\rho_s}{\rho_l}\right) \left[L_t \bar{\alpha}_c + L_c \bar{\alpha}_e \right] = \frac{V_l^2}{2g} \left[K_c + f_1 \left\{ \frac{L_{nb}}{D_{ec}} + \bar{R} \frac{L_b}{D_{ec}} \right\} + f_2 \bar{R} \frac{L_c}{D_{er}} + 2r\rho_l + K_e + K \right]. \quad (4-23)$$

NOMENCLATURE AND SYMBOLS

A	= Cross-sectional area for flow, ft ²
C_s/C_l	= Local steam to water velocity ratio
D_{ec}	= Flow equivalent diameter of core flow passages, ft
D_{er}	= Flow equivalent of riser, ft
f	= Thermal utilization
f_1	= Isothermal liquid friction factor for core zone
f_2	= Isothermal liquid friction factor for riser section
h_l	= Saturated liquid enthalpy, Btu/lb
h_g	= Saturated steam enthalpy, Btu/lb
h_{fg}	= Heat of vaporization of steam corresponding to reactor pressure
K	= Loss coefficient for turning the saturated water into, through, and from the downcomer
K_c	= Single-phase contraction loss coefficient based on channel velocity
K_e	= Two-phase expansion loss coefficient based on channel velocity
k or k_∞	= Infinite multiplication factor
k_{eff}	= Effective multiplication factor
L_b	= Height of core in boiling, ft
L_c	= Height of nonheated riser above core, ft
L_t	= Core height, ft
L_{nb}	= Height of core nonboiling = $L_t - L_b$, ft
$\frac{L_b}{L_t}$	= $\frac{\text{Thermal power in boiling}}{\text{Total reactor thermal power}}$

m	= Subcooling fraction = $\Delta h_c/h_{fg}$
P	= $Q_t/(W_s + W_c)$, Btu/lb
Q_t	= Total reactor thermal power, Btu/hr
r	= Acceleration pressure drop multiplier, ft ³ /lb
R	= Ratio of 2-phase isothermal pressure drop to that if only the liquid phase were flowing
$\bar{R}$	= Ratio of 2-phase boiling pressure drop with heat addition to that if only the liquid phase were flowing
V_l	= Average inlet velocity through fuel element flow passages = $(W_s + W_c)/\rho_l A (3600)$, ft/sec
W_c	= Water flow rate at core exit, lb/hr
W_s	= Steam flow rate at core exit, lb/hr
$W_s + W_c$	= Total water flow rate entering core, lb/hr
X_e	= Ratio of steam flow to total flow at core exit = $W_s/(W_s + W_c)$
Y	= $\bar{\alpha}_b/\alpha_e$ and is a function of pressure, slip ratio (C_s/C_l), exit void fraction α_e , and variation of heat flux with length
α_e	= Steam volume fraction at core exit
$\bar{\alpha}_b$	= Mean steam void fraction in the boiling height
$\bar{\alpha}_c$	= Steam void fraction within core boundary but excluding coolant within control rod flow passages
Δh_c	= Subcooling, Btu/lb condensate or Btu/lb of total flow depending on cycle
Δh_r	= Subcooling of water at core inlet, Btu/lb of total flow
ρ_l	= Saturated water density, lb/ft ³
ρ_s	= Saturated steam density, lb/ft ³

REFERENCES

1. A. S. JAMESON, Natural Circulation Boiling Density Tests, TID-7506, Pt. I, p. 19, November, 1955.
2. W. H. COOK, Boiling Density in Vertical Rectangular Multichannel Sections with Natural Circulation, ANL-5621, November, 1956.
3. W. S. FLINN and M. PETRICH, Performance and Potential of Natural Circulation Boiling Reactors, ANL-5720, October, 1957.
4. J. F. MARCHATERRE, The Effect of Pressure on Boiling Density in Multiple Rectangular Channels, ANL-5522, February, 1956.
5. W. S. FLINN and P. A. LOTTES, A Method of Analysis of Natural Circulation Boiling Systems, *Nuclear Science & Engineering*, 1(6), p. 461 (December, 1956).
6. SAMUEL UNTERMYER, Direct Steam Generation for Power, *Nucleonics*, 12(7), 43 (July, 1954).
7. J. R. DIETRICH, The Physics of Advanced Reactors, *British Journal of Applied Physics*, Supplement No. 5, pp. 89-823, 1956.
8. H. P. ISKENDERIAN, 20-MW D₂O-Moderated Experimental Boiling Water Reactor Design Studies, ANL-5685, February, 1957.

CHAPTER 5

THE EXPERIMENTAL BOILING WATER REACTOR (EBWR)

The early success of the BORAX experiments indicated a need for a larger boiling water reactor, and during the winter of 1952–1953 a proposal was made for the construction of a pilot power plant reactor at Argonne National Laboratory near Chicago. This proposal was presented to the General Advisory Committee of the United States Atomic Energy Commission as the basis for an Argonne Laboratory reactor construction program, and ultimately the project became a part of the U. S. Atomic Energy Commission's first five-year power reactor development program. Construction of the plant began in 1954. The reactor became critical on December 1, 1956, and reached its rated power of 20 Mw, thermal, on December 29, 1956.¹⁻³

The core of the Experimental Boiling Water Reactor, EBWR, was designed and built by Argonne National Laboratory. The conceptual design for the remainder of the nuclear system was developed by Argonne and detailed by Sargent & Lundy. Allis-Chalmers Mfg. Co. designed and built the power equipment. While the reactor plant was designed as a 5000-kw electrical plant, the principal goal was not the generation of electricity; its purpose was to provide an experimental plant in which information for future use in designing large central station units could be obtained. The EBWR design, therefore, does not represent an optimization of factors from the standpoint of producing electricity at minimum cost from a plant of 5000-kw output. Instead, an unusual degree of flexibility was incorporated in order to facilitate investigation of the basic parameters in the boiling water reactor concept.

Argonne followed the unorthodox procedure of doing the critical experiments for the reactor in the reactor itself after it had been built. In these critical experiments Styrofoam was used to simulate voids. EBWR began operation at a power rating of 20,000 kw, thermal, but the heat output of the reactor has since been raised by a factor of 3 or over 60,000 kw thermal.

The plant regularly supplies a substantial part of the Laboratory's power needs during the day and can take over the full load at night. In this service, the reactor power plant operates in parallel with the local utility network and uses standard power electrical switching and control equipment. The EBWR has operated satisfactorily in respect to require-

ments of load and reliability expected of a commercial power station.

The plant includes such safety features as an all-welded steel outer containment shell, 119 ft high and 80 ft diameter, a 15,000 gal tank of water for emergency use suspended at the top of the containment shell, two emergency boron solution injection systems, and provision for complete remote operation from outside the containment shell.

A considerable amount of flexibility has been incorporated in EBWR's design. For example, the reference 4-ft diameter core can be increased to 5 ft if desired. Fuel elements of different dimensions are provided to permit changing the uranium-to-water ratio in the core and the study of the influence of void coefficient of reactivity on reactor stability. Two thicknesses of uranium fuel plates allow the critical size and power distribution pattern in the core to be varied. Although the reactor is designed for operation with light water moderator and natural circulation cooling, future conversion to forced circulation cooling was provided in the design. If necessary, the entire core can be removed and replaced by one of radically different design. It will be possible to substitute D_2O or H_2O and install a core more suitable for a D_2O moderator after the desired operating experience is obtained with H_2O . With this in mind, the entire water and steam circuit is designed to be essentially leakproof. Expediency in completing the light-water system now in use called for adding only those zero leakage seals which had to be an integral part of the major equipment. Specifically, these are the turbine shaft seals, the steam-control bypass and pressure relief valves, and the feedwater pump seals. On small valves, packings were used. Special packings will be installed on these valves prior to use with D_2O . The principle used in designing the completely sealed system was to add a dry-air buffer seal to various shaft seals so that the net leakage of dry moisture-free air was inward to a take-off chamber, from which a combination of air and moisture could be taken to a recovery system.⁴

As explained in previous chapters, one of the concerns of the boiling water reactor concept has been that the reactor is fundamentally unstable to changes in steam demand. Increase in steam demand decreases the reactivity of the reactor while a sudden decrease increases the reactivity. This occurs because an increase in steam demand results in a pressure drop and an increase in steam voids, thus reducing the reactivity.

The designers of EBWR got around this problem by installing a special bypass throttle in a steam line that goes directly to the main condenser, bypassing the turbine generator. The bypass throttle mechanism maintains a constant reactor system steam pressure by automatically adjusting the bypass flow rate to compensate for changes in the turbine steam demand. In practice the steam bypass device is used to hold the steam

pressure constant until the reactor power level can be adjusted to the new load condition. The performance of this bypass control system in EBWR has been quite satisfactory.

Although technical feasibility has been established, the economic feasibility of the EBWR steam bypass control system is not yet clear. Since the bypassed steam represents wasted energy, this method of control will always reduce the overall efficiency of the reactor plant. The question as to whether or not this loss of efficiency will prove to be serious remains to be answered. The EBWR engineers, however, are optimistic about this point, believing that under ordinary operating conditions it will prove to be feasible to run the plant with the steam bypass valve completely closed. They believe the bypass will be mostly used as a convenient safety valve for relieving pressure buildup under abnormal circumstances.

Another reason for building EBWR was to test the feasibility of using the primary-circuit steam in a direct turbine cycle. The possibility that radioactive contamination of the turbine as well as associated piping and auxiliaries might cause drastic maintenance problems has been regarded as one of the major concerns of the direct-cycle concept. Because of the experience gained in the BORAX experiments in determining the radioactive carryover, the EBWR staff is not greatly concerned about this potentiality. After some 6 months' operation the turbine was examined and showed no signs of contamination. Although there is some chance that contamination will build up over a much longer period of operation, the general belief is that there will probably be no major difficulty.

The construction and operation of EBWR has already answered many questions of feasibility for boiling-water reactors and for reactor power plants in general, but others remain to be answered and the primary purpose of EBWR is to find these answers. The achievement of the EBWR engineers is first of all that they have successfully transformed the complete reactor power plant, safeguards and all, from lines on paper into functioning hardware. They have then shown that this hardware can actually be run to produce power with a reliability that would satisfy commercial demands. Further, they have succeeded in reducing the operation and maintenance of the plant to a set of routine procedures such as may be found in a conventional power plant. In the operation of this nuclear power plant, technicians without college backgrounds have been trained as operators and have been given the responsibility of routine operation and maintenance.

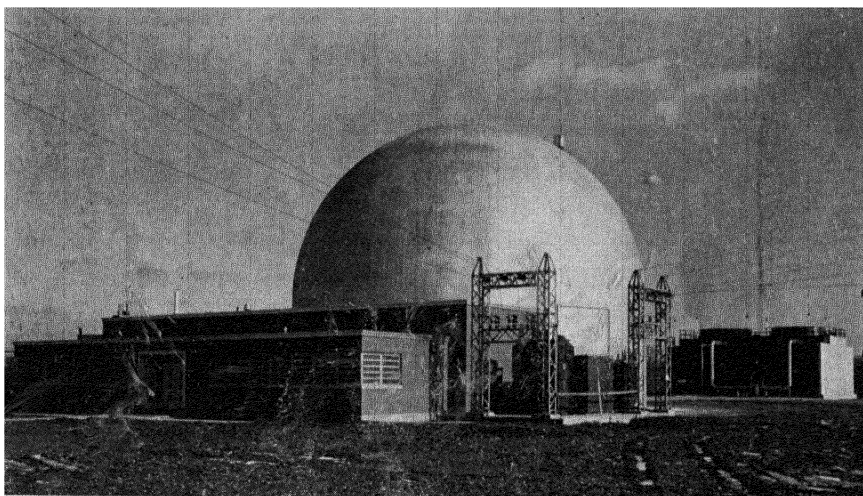


FIG. 5-1. View of structures.

GENERAL FEATURES OF EBWR

The Experimental Boiling Water Reactor is located near the center of the total exclusion area of Argonne National Laboratory, about 20 miles southwest of Chicago, near Lemont, Illinois. The closest community is the village of Lemont with a population of about 2500.⁵

Figure 5-1 is a photograph of the plant, showing the containment shell with the service building and the electrical substation in the foreground. The cooling tower can be seen in the rear at the right. A better idea of the layout can be obtained from Fig. 5-2, which is a cut-away view showing the entire plant in perspective. Access to the containment vessel from the service building is through an air lock to the main floor. The reactor, the turbine generator unit together with the main condenser, and all auxiliary steam and water equipment as well as the storage well for fuel are all located within the containment shell. Only the control room is outside the containment shell.

Steam and water system. In the EBWR power plant saturated steam at 600 psig pressure having a temperature of 489 F is generated within the core and is passed from the reactor vessel through a steam dryer directly to a turbine as shown in the flow diagram, Fig. 5-3. This simplified diagram shows the principal features of the plant. Starting at the reactor vessel, the water ring for feedwater injection is just at the top of, and completely surrounds, the core so that the reflector or downcomer

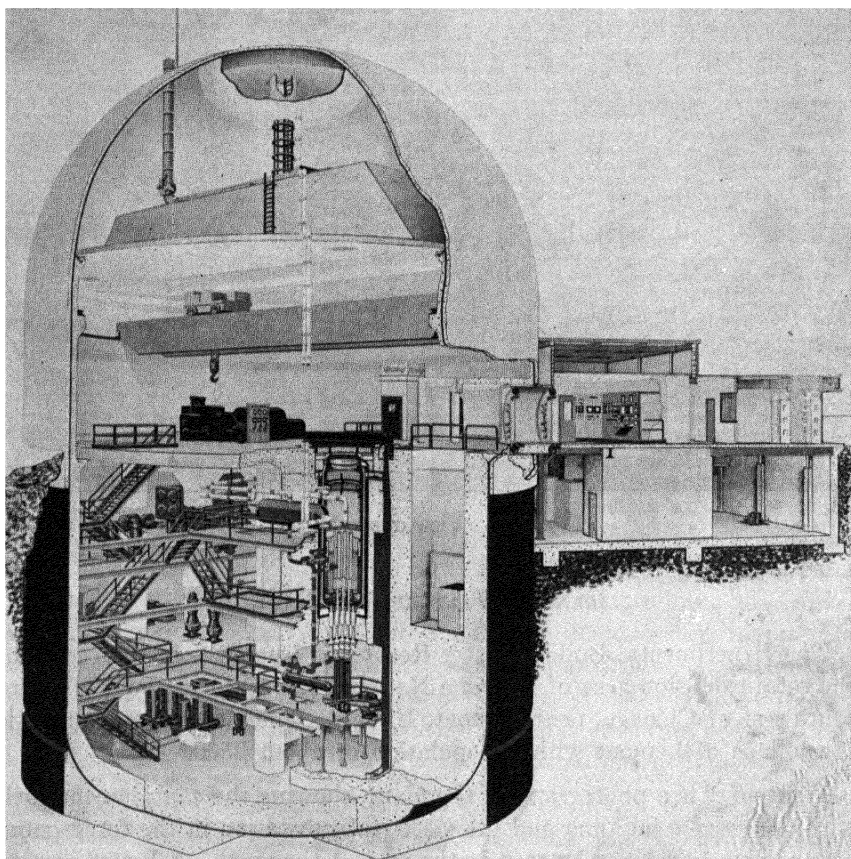


FIG. 5-2. Plant layout in perspective.

is uniformly cooled by the feedwater. As the feedwater mixes with the hotter water in the downcomer, the reflector temperature is reduced, and the water flows downward around the outside of the core and upward through the fuel sections. Approximately 5.2 degrees of subcooling for the water is obtained at the fuel inlet. The average amount of water recirculating in the core at 20 Mw is calculated to be about 7000 gpm. The flow of feedwater through the feedwater lines is 120 gpm. The total weight of water in the reactor vessel is measured by the height of the water in the water column at the left. All instrumentation for water level is attached to this water column, which is outside the biological shield. This provides for maintenance of the measuring instruments and control floats.

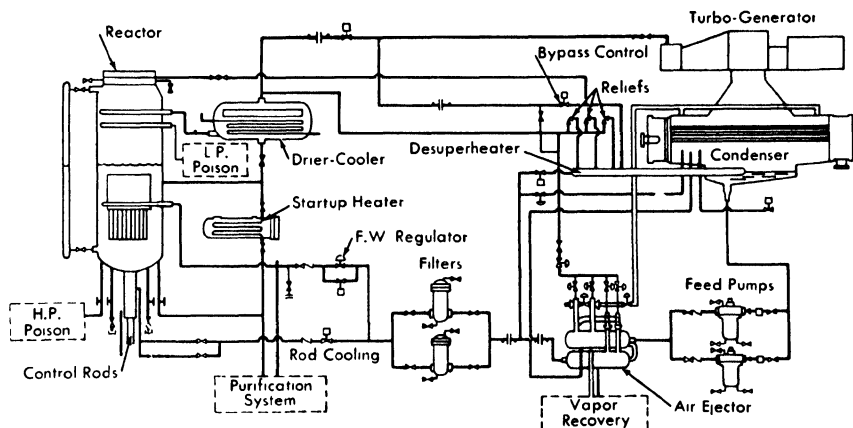


FIG. 5-3. Flow diagram.

A steam dome is provided above the core through which the velocity of the steam is low. This assists in drying the steam as well as reducing the amount of active particulate carryover into the external piping.

Steam is taken out of the reactor vessel through the steam ring at the top of the steam drum and passes through the dryer, which is outside the biological shield. Any water accumulated in the dryer is trapped and flows to the condenser. It does not return to the reactor vessel directly, except when the drum is used as an emergency cooler. Water (15,000 gal) is stored in a tank at the top of the building shell. This water can be allowed to circulate downward through this cooler and back up to the tank by natural convection in the event that all other cooling is lost. This system is capable of removing the shutdown heat of the core.

It will be noted from the flow diagram that the system includes a startup heater. In this heater 190-psig process steam from the laboratory system is used to heat the reactor water to 325 F and 100 psig. This condition is interlocked with rod controls, and the increased temperature before startup makes the reactor startup safer.

In the event that the rod drives are incapable of controlling the reactor, high pressure boric acid can be injected into the system through the feedwater ring. If the water were to leak out of the reactor vessel, the core would be exposed to the possibility of melting. For this reason a low pressure boron system has been provided which can spray water over the elements and cool them to prevent melting.

The control rod drives in this reactor are located below the core. This means that the absorbing sections of hafnium or boron steel must be pushed upward out of the core by a follower having low neutron absorption. Zircaloy-2 was used for this purpose.

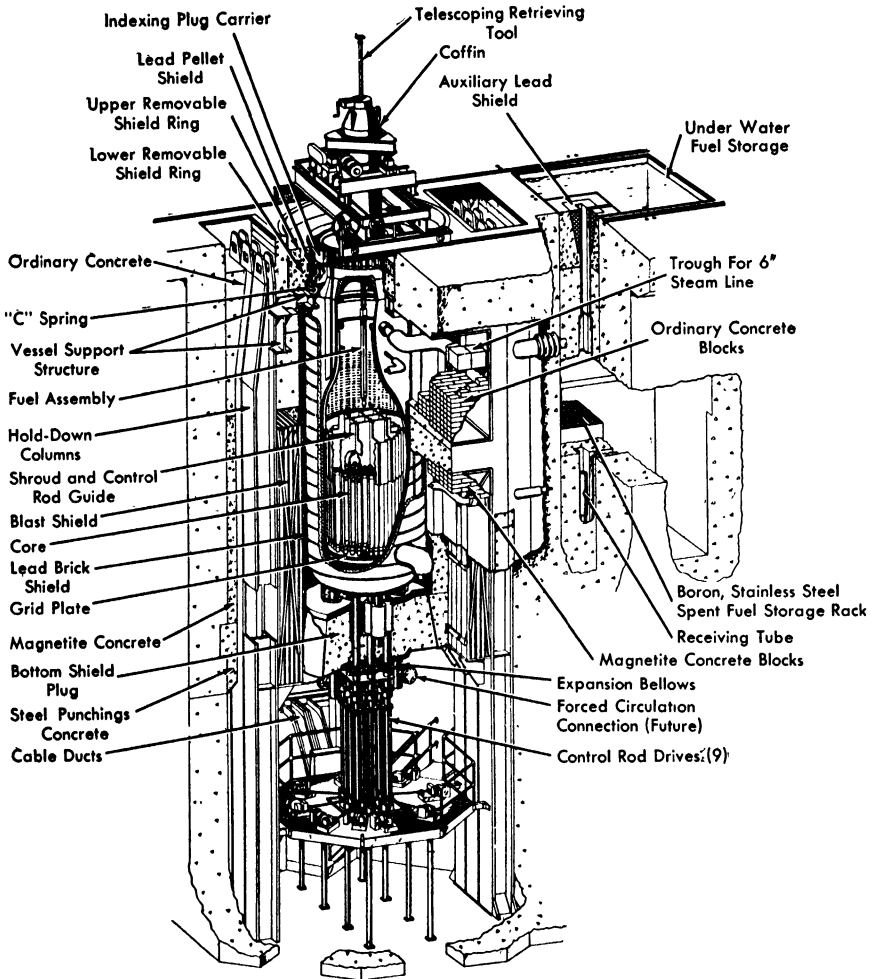


FIG. 5-4. Cutaway pictorial of EBWR reactor and components.

Top closure for the reactor vessel is double gasketed, and a bleed-off between the gaskets is connected to the condenser to reduce the leakage to zero at the second or outer gasket.

From the dryer the steam proceeds through the main steam line to the turbine generator, is condensed, and the condensate accumulates in the hot well. The water is pumped from the hot well and raised to 860 psig by an 8-stage centrifugal pump, there being no hot-well pump or surge tank ahead of the feedwater pump. This was done to reduce the number

of leakage points in the system and was pointed more toward the eventual use of heavy water than toward the present light-water system. There probably would be no reason for using a vacuum-to-pressure pump in a light-water system. As indicated in the diagram, two feedwater pumps are installed, one serving as a spare.

The steam dryer is equipped with cooling coils and is so placed as to permit its use as an emergency cooler. At the dryer, the steam pressure is held constant by the bypass system described earlier in this chapter. This system drains the steam directly to the condenser, its principal function being to automatically make the steam load equal to the reactor thermal power regardless of the amount of steam taken by the turbine. The inlet valves to the turbine are mechanically linked to this bypass control valve so that, as the turbine valves open, the bypass control valve closes.

For overpressure relief a second line comes from the steam dryer-emergency cooler and passes through three relief valves directly to the condenser. Two of these valves are pressure-relief or safety valves. The third is a steam-powered back-pressure regulator with a nominal 50-psi proportional band.

If the reactor pressure reaches 625 psig, the back-pressure regulator tends to hold the pressure constant. If it fails to hold the pressure below 700 psig, a second relief valve would open. The opening of any of the 3 relief valves automatically shuts down the reactor.

Vacuum in the condenser is maintained by steam air ejection, and the ejector condensers are cooled by feedwater. To ensure that the feedwater is passing through the air-ejector coolers at a rate of 75 gpm, water is recirculated to the condenser when the flow to the reactor is below this amount. This water can also be recirculated to the desuperheater but this method has not been used.

Gases from the air ejector at present are removed by way of the stack and do not go to the vapor recovery system. All the water returning to the reactor passes through cotton-fiber filters, and the flow is regulated by an electrically positioned valve, where the pressure is reduced from 850 psig to the 600 psig in the reactor. The rod drives operate through thimbles at the bottom of the vessel and, through seals, are connected to the control rod proper. It was anticipated that, if the leakage through the labyrinth drive seals was high, the thimbles would approach the reactor water temperature. For this reason some feedwater was to be injected into the thimbles to keep them cool; however, the flow through the seals has been so low that thimble radiation maintains its temperature at less than 150 F, and the rod cooling system has not been used too often.

A bypass water purification system, consisting of ion exchangers and

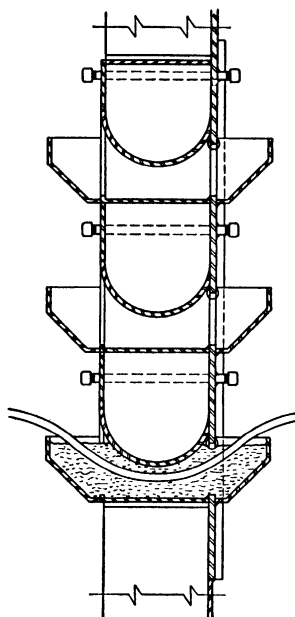


FIG. 5-5. Power plant shell penetrations.

filters, is provided to maintain the desired water purity. This, of course, is highly essential in a boiling water reactor system. This purifying system removes water (by pumping) from the bottom of the reactor vessel and returns it through the feedwater distribution ring. In order to prevent discharge of active liquids up the stack and to avoid loss of working fluid when the plant is converted at a later date to the use of D_2O as coolant-moderator, a vapor recovery system collects any water vapor carried over by the air ejectors and any leakage through turbine, pump, and valve seals, and returns the water to the condenser.

Reactor core. With reference to Fig. 5-4, the core is made up of fuel assemblies arranged in a 4-in. square pattern within a core shroud structure and seated in a support plate which is bolted through legs at the bottom of the reactor vessel. The fuel assemblies each contain 6 plate-type elements of uranium-5-w/o zirconium-1.5-w/o niobium, clad with Zircaloy-2. The reference core, about 4-ft in diameter, consists of 106 such assemblies containing uranium enriched with 1.44 per cent U^{235} , and 8 assemblies containing natural uranium, evenly distributed. Half the assemblies of each enrichment contain plates slightly thinner than

those of the other half in order to permit experimental variation of metal-water ratio.

The core support plate and shroud structure are designed to accommodate up to 148 fuel assemblies in order to allow experimental variation of the core diameter up to 5 ft. Consequently, additional enriched (1.44 per cent) assemblies of the two different plate thicknesses are provided for these tests. During tests with core diameters smaller than the maximum, the outer fuel positions are occupied by dummy assemblies. The core shroud structure serves the triple purpose of positioning the upper ends of the fuel assemblies, providing additional riser height above the assemblies to enhance natural circulation, and acting as a guide for the 5 hafnium and 4 boron-stainless steel control rods moving within the core.

Control rod drive mechanisms, located below the reactor, are of the external lead screw-and-nut type employing a linear labyrinth pressure breakdown seal. Auxiliary emergency shutdown control is provided in the form of a high-pressure and a low-pressure boric acid injection system. In the high-pressure system, 160 gal of saturated boric acid solution are maintained under 1600 psig air pressure in a tank and are injected into the feedwater line in the event of failure of certain control rods to fully insert themselves. The low-pressure system is used only when the reactor pressure is reduced to a value less than 20 psig, and water level is lowered to a point 6 inches above the active portion of the fuel assemblies. Boric acid is introduced into the top of the vessel at a distribution ring equipped with spray nozzles arranged to insure impingement of the solution on all exposed parts of the core for the purpose of removing the decay heat. Here the boric acid is present only to maintain subcriticality with water addition.

Reactor vessel. The EBWR reactor vessel is 7 ft in inside diameter and 23 ft in internal height, and is fabricated of SA-212 grade B carbon steel with a Type 304 stainless steel lining. The vessel wall is protected from excessive heat generation by a boron-stainless steel thermal shield. The vessel is supported at its upper end by lugs resting on C-shaped compression springs which, in turn, rest upon a Carilloy T-1 steel framework attached to columns which extend downward and are embedded in the concrete basement floor. The springs and framework are designed to absorb dynamic loadings which might result from the accidental dropping of heavy objects such as the vessel cover plate on top of the vessel.

The reactor is located below the main floor in an octagonal concrete shell extending to the basement floor as shown in Fig. 5-2. The vessel is surrounded with a 3-in. thickness of stainless steel wool insulation,

3 in. of airspace, and a carbon steel cylindrical tank which serves as an inner form for shielding. The tank is equipped with cooling coils on its outer side to remove heat generated within the shielding and transferred from the reactor to the shielding, and is covered with lead brick and with a plaster containing ferroboration as part of the fine aggregate. The remainder of the space between the plaster and the wall of the octagonal shell is filled with either magnetite concrete with steel punchings added, magnetite concrete, or ordinary concrete, depending upon the radiation intensity at the point concerned. The total thickness of radial shielding is 8 ft 3½ in.

Shielding below the reactor vessel is essentially magnetite steel punchings concrete. Penetrations for the nine control rod drive nozzles and four forced circulation discharge nozzles are stepped to prevent direct radiation streaming, and additional layers of lead and ferroboration are provided in the critical areas. At the top of the reactor, shielding is provided by a stainless steel indexing plug assembly within the vessel, the top cover of the vessel, and a portable water tank which also affects the top seal of the air envelope surrounding the reactor. Spaces surrounding the vessel support lugs and top closure are filled with specially shaped magnetite-steel punchings concrete blocks, steel plates, and a container filled with lead pellets.

With further reference to Fig. 5-2, removal of spent fuel is accomplished by a lead shielded transfer coffin which travels between the reactor and the adjacent fuel storage pit on a positioning carriage. After removal of top shielding component and the vessel cover, a plug carrier assembly containing bearings and drive gears is placed on top of the vessel, attached to the internal indexing plugs, and positioned over the assembly to be removed. The coffin is located above the discharge openings in the plugs, the telescoping fuel grapple is lowered and engaged, and the assembly is lifted into the coffin. The coffin is moved to a position over the discharge chute in the fuel storage pit, and the assembly is lowered into the receiving tube and released. Subsequent movement of the assembly to a position in the boron-stainless steel storage rack for cooling, as well as ultimate removal of end fittings and shipment to a processing plant, is accomplished within the pit by use of special tools.

The entire power plant as described thus far is housed within a cylindrical steel tank having an ellipsoidal bottom head and a hemispherical top head, designed to withstand an internal pressure of 15 psig. The purpose of the shell is to contain any radioactive materials which might be released as the result of accidental rupture of any part of the system during operation. In order to prevent excessive pressure buildup in the building because of steam release, a spray system has been provided to

spray water into all parts of the building and condense any steam released. This system is supplied by the 15,000 gal emergency water storage tank located just inside the top of the building dome, as already mentioned. Were all the water in the reactor system turned to steam, the pressure of the shell would rise to about 9 psig. Spraying the 15,000 gal of water into the building would reduce the pressure to nearly atmospheric in about 15 minutes. Lowering the pressure reduces the leakage of radioactive steam from the shell to a very low value.

The shell is not maintained under any pressure during operation; instead, the air in the building is changed through intake and exhaust ducts, the exhaust duct being visible in the attic, Fig. 5-2. Both the intake and exhaust ducts are closed with dampers; in the event of a release of radioactivity, these dampers would close and automatically shut down the plant.

The building is designed to have a leakage of less than 1000 cu ft per day when the internal pressure is 15 psig. Two vacuum breakers prevent a negative internal pressure. The principal problems encountered in constructing the shell enclosure were those of obtaining a low leakage rate. The guaranteed minimum leakage rate was to be 1000 ft³/day at 15 psig. The containment shell was built and tested prior to the installation of any concrete or equipment. The shell openings for condenser circulating water, air ducts, mechanical operators for boron injection valves, freight door, air locks (of which there are two; the main air lock and the smaller emergency air lock), and electrical cable penetrations were omitted from the initial test.

The original test on the shell alone was quite successful, but when the building was finally completed for operation, obtaining a second successful leakage test was quite another problem. Even though some precautions were used to make what appeared to be tight joints and secure gasketing, the principal penetrations of the shell, listed above, leaked very badly.

It was not expedient to continue pressurizing the building just to find leaks, consequently a three-day test was run in which all the leaks which could be found were inventoried. Then the various devices, such as the air lock doors and the vent and intake dampers, were isolated into units so they could be pressurized individually and tested for tightness. This arrangement probably should have been part of the original specifications because considerable time could have been saved if individual unit tests had been provided for instead of depending upon field fabrication of test methods.

A major leakage problem existed at the cable troughs through which the 300 multiconductor cables penetrated the shell for control and power

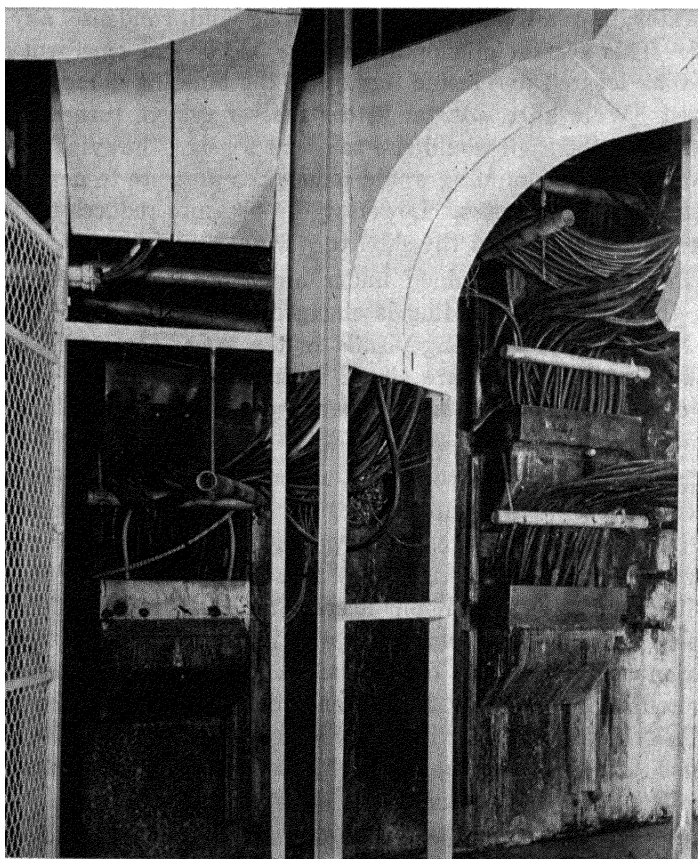


FIG. 5-6. Built-up cable troughs.

purposes. The precaution of testing each cable was taken to insure that there was no leakage along the cable itself. If there was leakage, the ends were closed with solder lugs, and a solid wrapping of tape was used to make the cable itself leaktight. All stranded conductors were treated in this manner. However, where the troughs penetrated the shell proper, a real problem existed. The cross section of this system is shown in Fig. 5-5. The shell is to the right; the trough itself is 2 ft wide and presents a 2 ft by 1 ft area for cable egress. The total length of the trough was about 6 ft, penetrating the 2 ft concrete liner support inside the lower shell region. The design called for filling the troughs with Chico cement. Chico cement is not itself leakproof; nor does it seal tightly to the cable or steel. The EBWR engineers were aware of this problem and intended to fill the trough at the inside with Novoid com-

pound, which under pressure from the building would seal the Chico by penetrating any tiny openings along the cable or side of the trough.

This idea probably would have worked, except for one minor difficulty. The trough seams were not seal-welded shut. Instead, a sample seam weld was run. Therefore, leakage simply bypassed the Novoid compound. This leakage was extremely high, and makeshift methods were tried to caulk the cables on the outside of the shell. This was necessary since the troughs were then completely encased in concrete and could no longer be welded shut. Several weeks of experimentation led to building up the troughs about 8 in. at the outside (the right of the drawing) and finally filling this with a two-part epoxy resin, which sealed tightly against the cables and troughs and effectively caulked the opening. The final result is shown in Fig. 5-6, where resupporting of the cables is seen and the added sections of the troughs are visible. The upper part of the trough in this view is filled with about 4 in. of epoxy resin. A successful test was made which showed that the leakage was actually less than the required minimum.

Special provisions have been made to prevent puncturing the steel building by missiles which might result from such accidents as reactor, turbine or pipe ruptures. The entire shell is lined with concrete except for the upper hemispherical dome. The latter is protected by a steel plate and concrete ceiling over the main floor. In addition, the reactor structural arrangement includes features to prevent a catastrophe in the event of an explosive zirconium-water reaction within the core. The shock pressures developed would cause the bottom head of the reactor to be blown, together with the bottom shielding, into the basement, so that the vessel itself would tend to act as a projectile. The large upward force would be restrained by three heavy beams located across the top of the reactor and pinned by wedges to 6 columns whose bases are firmly anchored to the large mass of concrete forming the basement floor of the building. Laminated blast shields of compressible material and steel are interposed between the reactor and the columns (in the vicinity of the core) to protect the latter from side shear.

The service building, adjacent to the power plant building, houses the plant control and electrical switchgear, a laboratory, service, office, reception lobby, and sanitary facilities. The entire control of the plant is effected from the control room in the service building, so that the presence of personnel in the power plant building during reactor operation may be limited to occasional maintenance and inspection tours.

The electrical system of EBWR is not particularly different from that of a conventional steam-electric plant. The turbine generator is a standard 5000-kw unit with direct connected exciter generating at 4160 volts,

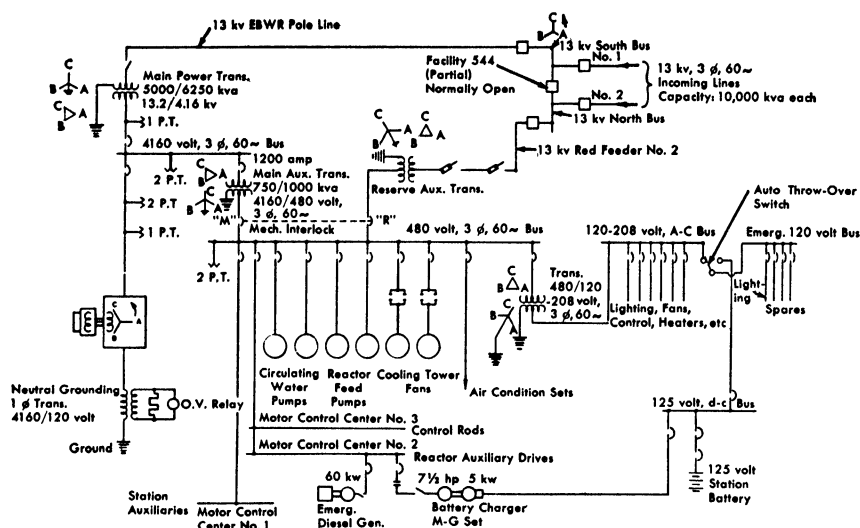


FIG. 5-7. Simplified line diagram of EBWR electrical system.

3-phase, 60 cycles, arranged to operate in parallel with the utility power system. The generator leads are connected to the 4160-v bus of an outdoor unit substation through a 1200-amp air circuit breaker. The 13.2-kv side of the main transformer is connected to the north bus section of the Laboratory central switching station by means of the gang operated disconnect switches in the EBWR outdoor substation, the overhead 13.2-kv EBWR pole line, and an oil circuit breaker in the north bus section of the central switching facility. Figure 5-7 is a simplified line diagram of the EBWR electrical system.

Costs. In starting the EBWR project, the persons heading the group responsible for its design were cognizant of the interest that would be created by the successful operation of this reactor, and it was recognized that the costs were an important part of the project.⁶ As a consequence, a special effort was made to have labor and materials entered into the proper accounts. In considering costs, it is important to emphasize that the "E" in EBWR stands for "Experimental." The reactor was designed for variable fuel loading. In addition, the system was designed for the containment of heavy water, not merely light water. Provision was made for the inclusion of forced circulation in future operations. The reactor must be considered a pioneer unit. This fact reflects the unusual situations in that many items which operations now prove to be over-designed were, at the time of designing, quite necessary to cover uncer-

TABLE 5-1
EBWR RESEARCH AND DEVELOPMENT COSTS

	Total	Reactor system	Power generation system	Fuel elements and materials	Heat-transfer studies
Technical man-years					
Staff	54.5	21.9	7.9	17.1	7.6
Technicians	40.6	9.0	4.2	17.8	9.6
Salaries	\$ 726,900	\$245,700	\$ 95,600	\$ 262,700	\$122,900
Special materials (zirconium, niobium, and boron stainless steel)	136,100			133,100	3,000
Materials and supplies (all other)	132,200	11,900	6,900	99,000	14,400
Irradiation services	91,900			91,900	
Uranium processing	14,500			14,500	
Computer services	20,100	15,100		5,000	
Chemical analyses	33,000	400	200	32,400	
Hot laboratory use	26,600			26,600	
Subcontracted develop- ment and design	31,000			30,800	200
Shop fabrication	296,600	94,300	2,000	175,700	24,600
Engineering and drafting (ANL shops)	44,600	30,100	2,900	8,800	2,800
Services (all other)	82,900	20,800	8,500	45,500	8,100
Allocated adminis- trative and service costs	1,402,300	551,500	191,700	471,400	187,700
	\$3,038,700	\$969,800	\$307,800	\$1,397,400	\$363,700

tainties. Finally, although EBWR was designed for a gross electrical output with natural circulation, operating experience has shown that the system is capable of a much greater electrical output, since the reactor has operated at thermal outputs as high as 60 Mw—3 times the design thermal output.

The accompanying cost figures show all the costs charged to the EBWR project with one exception in the construction costs, as will be explained. Table 5-1 covers the research and development costs assigned specifically

TABLE 5-2
CONSTRUCTION COSTS

A. Buildings

1. Office and control building

a. General construction	\$139,500	
b. Heating, ventilation, air conditioning, plumbing, lighting, electricity, inspection	39,500	\$ 179,000

2. Power plants

a. Excavating and backfill	\$ 87,000	
b. Steel shell	288,000	
c. Shell coatings and thermal insulation	54,100	
d. Concrete inside shell	177,400	
e. Inside steel	123,500	
f. Utilities, air conditioning, ducts, painting, etc.	147,400	
g. Miscellaneous equipment	89,200	\$ 966,600

B. Reactor system

a. Foundations	\$106,800	
b. Vessel external structure	163,700	
c. Vessel	202,200	
d. Vessel internal structure	147,800	
e. Shielding, total internal and external	397,600	
f. Control and instrumentation	139,100	
g. Control rods and drives	181,600	
h. Boric acid system	58,000	
i. Fuel handling and storage equipment	67,700	
j. Feed water system	146,800	
k. Water purification	60,200	
l. Steam system	133,900	
m. Miscellaneous equipment	145,500	\$1,950,900

C. Turbogenerator

a. Foundations	\$ 71,200	
b. Turbogenerator	408,400	
c. Condensers and water lines	126,600	
d. Air-dryer system	20,100	
e. Control	16,000	\$ 642,300

TABLE 5-2 (continued)

D. Electrical		
a.	Motor controls, switch gear	\$122,800
b.	Emergency power	8,500
c.	Transformers and bus	164,000
d.	Miscellaneous	67,700
		\$ 363,000
E. Outside facilities		
a.	Roads, walks, parking, site preparation	\$ 21,600
b.	Sewers	16,900
c.	Water lines	10,600
d.	Cooling tower, basin, water lines, water treatment	56,100
e.	Steam	35,300
		\$ 140,500
F. Architect-engineering		\$ 385,800
	Grand total	\$4,628,100

to the EBWR. The column headed "Reactor System" covers work rather specific to the reactor proper. The column headed "Power Generation System" covers overall system planning and development. The remaining two columns are self-explanatory. In evaluation of these accounts, it is impossible to draw sharp lines of demarcation between the research and development costs and the construction costs. Some of the costs in the reactor system and power generation system columns are in this uncertainty region between the two major cost areas. This point is mentioned again in connection with Table 5-2. The total of the research and development costs that are specifically for the EBWR are shown at \$3.04 million.

The next set of costs (Table 5-2) covers the items more properly defining the worth of the EBWR facility as it now stands. The construction cost heading has the one set of deletions from the EBWR accounts. This presentation covers the EBWR as it is now operating; consequently, where items were clearly defined as being solely either for forced circulation or for heavy water, these costs were eliminated. Many items, however, had their costs strongly influenced by the possible operation with heavy water or forced circulation, but no attempt was made

to make some percentage reduction in these costs. They were left in their entirety. The total subtractions amounted to \$220,000.

The first items covered in Table 5-2 are the two buildings on the site. The first is an orthodox type building having brick walls with concrete backing. The building has a conventional flat roof and a full basement, and the resulting cost is quite standard. The power plant building, on the other hand, is the steel shell building. Perhaps the single item of most interest is 2(b). This covers the total cost of the erected steel shell, including the pressure-tight doors and, in fact, covers the cost of significant repairs to the doors. The total building costs, including the standard utilities, is \$1.15 million.

Under the subheading B, the reactor costs cover the reactor vessel and all associated equipment up to the turbine. The vessel external structure covers the reactor supports and the heavy hold-down members over the top of the reactor. The internal structure covers such items as the fuel supports and control-rod guides. The thermal shield cost is covered in the next item, as are all other forms of shielding which are external to the reactor vessel. The remaining items are self-explanatory as to content. The miscellaneous items may seem somewhat large and it results from lumping all shop purchase orders under \$1000 into the miscellaneous item. The total cost of the reactor, as shown, is \$1.95 million.

Subitem C includes the basic cost of the turbogenerator plus the amount required to dismantle, nickel plate the inside of the turbine casing, and reassemble the machine. Much of the cost of the system to supply dry air to the shaft seals was left out because this cost was specific to heavy-water operation.

Thus, the total cost of EBWR as a light-water, natural-circulation reactor operating as a 4500-kw net generating facility is \$4.63 million.

The fuel fabricating costs in Table 5-3 cover the total time and materials for fabricating fuel, excepting, as noted, that the cost of uranium from the production channels is not included. The first part of this table covers the cost of fabricating 920 fuel-containing plates and assembling 888 of them into 148 fuel assemblies. Of the 148 subassemblies, 114 of them are in the reactor. The \$793,000 is the prorated share of the \$1.08 million which is attributed to the 114 subassemblies at a unit cost of \$6960 per assembly. In making up the fuel fabrication costs, no credits were given for scraps containing special materials such as zirconium and niobium.

The important EBWR costs are covered in these three tables (Tables 5-1 to 5-3). These costs apply only to construction and design, and fuel fabrication. Power production costs are covered in the section on Operating Performance in Chapter 6.

TABLE 5-3
FUEL FABRICATION COSTS

Technical man-years	
Staff	7.6*
Technicians	8 *
Salaries	\$ 122,400*
Materials	391,200
Shop fabrication	260,200
Other direct costs	127,700
Allocated administrative and service costs	179,800
Total fabrication costs	\$1,081,300†
Cost of EBWR fuel, 114 assemblies at \$6957	\$ 793,100‡
Significant costs included above	
Materials	
Zircaloy-2 and zirconium	\$ 315,100
Niobium	49,000
Shop fabrication	
Scalp forged Zircaloy-2 slabs	\$ 5,100
Weld Zircaloy-2 cladding billet components	23,700
Fabricate steel jackets for billets	52,200
Machine Zircaloy-2 cladding billet components	49,500
Machine fuel plates	14,200
Weld plates into fuel assemblies	9,100
End fittings	11,500
Other direct costs	
Equipment and instrument costs	\$ 53,100
Chemical analyses	51,900
Forging Zircaloy-2 ingots for cladding	11,200

* Includes foundry operations, testing, and supervision.

† Includes cost of fabricating 920 plates and assembling 888 of these into 148 assemblies. Only 114 assemblies are used in EBWR to achieve designed power of 20 Mw (heat).

‡ Excludes cost of uranium from production channels.

OPERATIONAL CHARACTERISTICS

A. Reactor. Basically, the reactor of the EBWR plant has the same function as the boiler of a conventional fuel-fired power plant; its purpose is to generate steam. Feedwater at a temperature of around 110 F is pumped into the reactor vessel, and by the addition of heat from the fission process in the fuel elements, the water temperature is increased to saturation (489 F, at 600 psig). Evaporation occurs and an equal quantity of saturated steam leaves the reactor vessel to perform useful work in the turbine-driven generator. At 20 Mw core power, the corresponding steam rate is 60,000 lb of steam/hour. With the reactor purification system in operation, this value is reduced, since about 150 kw of heat is rejected by the purification system.

The amount of steam formed within the core is limited to a given fraction of the core height. At 20-Mw heat load, 35 per cent of the heat is expended in raising the cold feedwater to saturation condition (489 F) and the remaining 65 per cent is used to form steam.

With uniform heating, that is, heat addition uniform over the core height and radius, these power percentages would correspond to the percentage of the total core height in nonboiling and boiling regions respectively. But the actual generation of heat within the core varies both radially and longitudinally; the radial power density distribution is a zero order Bessel function and the longitudinal power density variation is a cosine distribution. These normal distributions are perturbed slightly by the effect of nonuniform steam volume distribution on buckling. The net effect of the cosine power density distribution is that the fraction of the core height where nonboiling occurs is slightly higher than for the uniform heat generation case noted above. With the assumed axial cosine flux and radial power density distribution, the average fraction of the core height over the entire core in the nonboiling and boiling regions is 0.40 and 0.60, respectively. Figure 5-8 shows the variation of the power distribution and boiling length fraction with radius of core, in addition to the radial variation of the mean actual steam void fraction.

After mixing with the return flow from the ion exchangers of the reactor purification system, the feedwater, at a temperature of 130 F, enters the reactor vessel through a drilled torus ring in the reflector downcomer at a height corresponding to the top of the active core. This method of feedwater injection, through 33 holes (each $\frac{3}{8}$ -in. diameter) has been selected to minimize the carryover of steam into the downcomer to give uniform distribution across the downcomer. Laboratory tests have shown that the flow distribution through the injection ports is maintained down to a load or flow rate corresponding to 10 per cent of rated power.

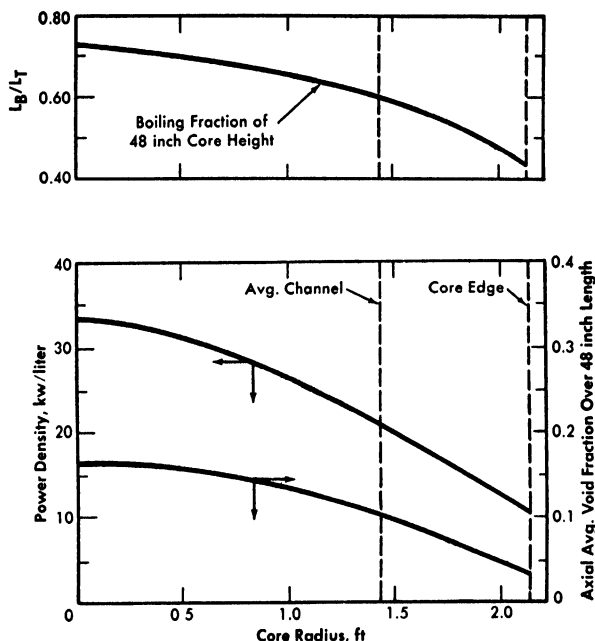


FIG. 5-8. Radial variation of core coolant power density, boiling length and steam void fraction.

The mixing of the cold return feedwater flow with the recirculated saturated water flow in the downcomer induced by natural circulation results in the subcooling of the water at the core inlet grid. Since the flow quantities that establish the subcooling are the feedwater flow, which equals the steam flow, and the total recirculation flow, the degree of subcooling will vary directly as the average quality of the steam leaving the core.

$$\Delta h_{\text{sub}} = \text{subcooling (Btu/lb)} = x\Delta h,$$

where x = core exit steam quality, i.e., steam flow/total water flow,

$$\frac{1-x}{x} = \text{recirculation rate} = \frac{\text{water flow recirculated}}{\text{steam flow}},$$

Δh = difference in liquid enthalpies of the saturated water in the downcomer and the entering feedwater.

At the 20 Mw design condition, the coolant enters the core 5.2 F below saturation value; this is due to the mixing of 60,000 lb/hr of the 130 F

feedwater with the 3.8×10^6 lb/hr recirculated saturated water flowing in the downcomer, the annulus between the thermal shield and the core structure.

Since the core exit steam quality increases with the power output because the recirculation flow rate does not increase as rapidly as the power, the subcooling also varies with the reactor power level as shown in Fig. 5-9. Also shown are the variations of the steam voids and exit quality with steam flow.

Consider the coolant as it flows through any of the fuel assemblies. The water is below saturation when it enters the core. After flowing a

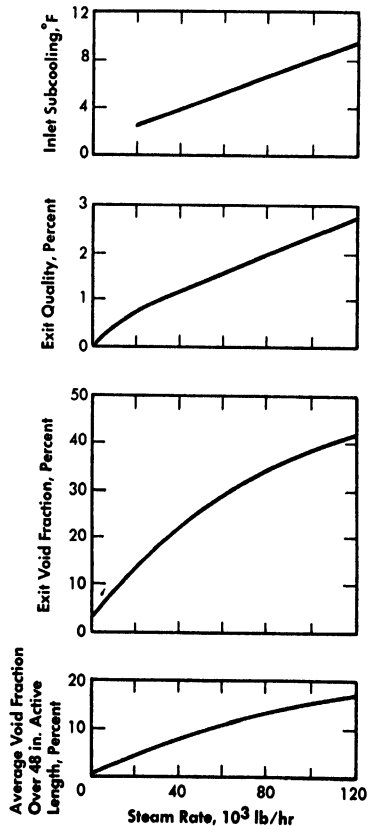


FIG. 5-9. Calculated performance curves for EBWR with 600 psig steam and 110 F makeup water.

specified height up into the core such that the total heat picked up by the coolant is equal to the sensible heat to bring the quantity of coolant to saturation, bulk boiling of the coolant begins. Actually, during passage of the subcooled fluid through the core, boiling on the surface occurs. This local or surface boiling brings about a small density change in the coolant. From this point on, the steam in the channel increases as a function of the power density distribution up to the active core outlet. In the chimney, i.e., the nonheated section above the active core, the steam-water mixture flows isothermally. On leaving the chimney, the steam separates from the water component and leaves the interphase with the steam from all the other fuel assemblies. The separated water returns in the downcomer, at a velocity of 0.9 fps to complete the cycle.

In order to promote the flow through the core, a pressure or driving head is required. This driving head is established purely by the density differential which exists between the steam-water mixture in the core fuel channels and chimney and all the liquid coolant in the downcomer.

If $\bar{\alpha}_c$ and $\bar{\alpha}_{cH}$ are the mean steam voids in the core (from inlet grid to top of active core), and in the chimney (from top of active core to outlet), respectively, and H_c and H_{cH} are the corresponding heights, then the driving head to create the circulation is

$$\text{Driving head (feet)} = [H_c \bar{\alpha}_c + H_{cH} \bar{\alpha}_{cH}] \left(1 - \frac{\rho_s}{\rho_l} \right)$$

This can be simplified, by omitting the term $(1 - \rho_s/\rho_l)$ which for 600 psig is 0.97. At rated load, the driving head with natural circulation is approximately 1.1 ft of water.

The driving head is the pressure differential required to force the flow through the sum total of the resistances in the flow path. The pressure drops which occur in passage through the inlet grid, through the core chimney and out of the chimney are:

1. Inlet contraction loss in going from bottom plenum through inlet grid.
2. Expansion loss on passage from circular inlet connection adapter to square plenum of fuel subassembly.
3. Contraction loss in getting flow from fuel assembly plenum into fuel channel passage.
4. Nonboiling or surface boiling friction from inlet grid to point of bulk boiling.
5. Two-phase boiling friction loss from inception of bulk boiling to top of active core section.

6. Acceleration pressure drop accompanying the change in density from inlet to outlet of core.
7. Expansion loss in passage from fuel channels to chimney section.
8. Nonheating two-phase pressure drop in chimney.

To the above pressure drops must be added the frictional resistance in the downcomer. Other extraneous losses occur, such as the resistance to radial flow from the chimney over to the downcomer; in the design calculations this incalculable loss has been neglected.

The calculation of the flux density consists, basically, of assuming a core stream exit quality. From the slip ratio (the ratio of the steam velocity to the liquid velocity at the same section), the average voids can be calculated for a given exit void, and the pressure drops enumerated above can be evaluated so that a circulation flow can then be calculated. With the circulation velocity and subcooling established, the corresponding power density for the assumed conditions are then fixed. This trial and error procedure will eventually yield the voids and velocity such that the power density calculated agrees with the power density for which the conditions are desired.

In Table 5-4, the voids and density have been based solely on the steam formation. The gaseous products of decomposition of the water within the core due to the irradiation of the water have negligible effects on the voids. For instance, for the case of 100 cc of total O_2 and H_2 at STP per liter of condensate flow, within the core the corresponding gas voids at the core outlet would be approximately

$$(100) \left(\frac{15}{615} \right) \left(\frac{949}{560} \right) \left(\frac{1}{50} \right) \frac{\text{cc of gas}}{\text{liter of circulating coolant}}$$

or less than 10^{-4} void fraction.

At the 600 psig design 20-Mw rating of the EBWR core, the natural circulation performance conditions with uniform fuel distribution and 112 subassemblies are listed in Table 5-4.

The steam rises through the pressure vessel to the "dry pipe" at a velocity of 0.39 fps. Although the ion exchange system will keep the total impurities in the water to a low value (~ 1 ppm), some irradiated impurities will be carried over, in water droplet form, to the steam system. Laboratory tests using soluble radioactive cesium in a boiling water solution at 600 psig and with steam velocities at the magnitude mentioned indicated that the ratio of the activity of the steam per pound to that of the reactor water will be approximately 10^{-4} for the long-lived activities

in the water. Tests conducted on the BORAX-III reactor confirm this de-entrainment or decontamination factor of 10^{-4} .

Additional protection against water carryover during any possible surge or flush operation is provided within the reactor vessel by the 6-inch dry pipe at the top of the vessel. Moisture reduction is obtained by reversing the direction of the steam flow (steam entry holes are at the top of the circular dry pipe). Since the density ratio of the water to that of the steam at 600 psig is 38.5, the directional change, de-entrainment results by the separating force of centrifugal action on the water particles.

The main radioactive components in the reactor and also in the steam will be due to the neutron capture of O^{16} to form N^{16} [$O^{16}(n, p) N^{16}$]. The N^{16} has a half-life of 7.4 sec with 80 per cent of the decays emitting a 6.2-Mev gamma ray.

It is estimated that the activity 1 foot away from the 6-inch steamline between the reactor shield and the steam dryer-emergency cooler is of the order of 100 mr/hr from the N^{16} gamma emission. The corresponding neutron flux at this same location and distance, due to the neutron emission from the decay of the isotope N^{17} (4.1-sec half-life) in the steam, is approximately $55 \text{ n}/(\text{cm}^2)(\text{sec})$.

Provision has been made for sampling the steam leaving the reactor to determine: (a) the long-lived activity and analysis of the carryover products; (b) the quantity of the decomposition gases evolved; and (c) the resistivity and pH of the condensate. In addition to this experimental sample station other stations are located throughout the plant as follows:

1. Inlet to steam dryer-emergency cooler.
2. Effluent from steam dryer-emergency cooler.
3. Water to feed water filters.
4. Effluent from feedwater filters.
5. Reactor water to ion exchange prefilter.
6. Reactor water from ion exchanger.
7. Air ejector exhaust.

B. Steam plant operational characteristics. Before passage of the steam to either the turbine or through the flow control bypass valve, further drying of the steam is effected in the plate scrubber, all stainless steel steam-dryer emergency cooler. The moisture separated from the steam is discharged through a steam trap to the condenser.

During normal operation the steam-dryer emergency cooler will be relatively active from N^{16} gamma radiation: approximately 150 mr/hr at 2 in. from the surface of the insulation. Normally, access to the steam-dryer emergency cooler area will be restricted to maintenance personnel.

TABLE 5-4

EBWR HEAT TRANSFER AND FLOW CHARACTERISTICS

Reactor power	20,000 kw
Pressure	600 psig
Steam flow	60,000 lb/hr
Total coolant heat transfer volume (consists of fuel channel, surrounding water within guide structure but does not include the control rod guide channels)	955 liters
Coolant volume contained within fuel plate channels	790 liter
Average power density based on total coolant volume	21.0 kw/liter
Average power density based on coolant contained within heated channels	25.3 kw/liter
Water volume in average fuel element to water volume in thick plate assembly	1.07
Maximum flux radially to average radial flux	1.58
Local maximum power in thick enriched assembly to average fuel assembly power	1.12
Flux in water channel near control rod to average flux in assembly	1.30
Power density in hottest channel radially	62.5 kw/liter
Maximum axial flux to average axial	1.49
Total heat transfer surface, based on fuel area	1,500 ft ²
Core average heat flux	45,500 Btu/(hr)(ft ²)
Average surface temperature	497 F
Avg. fuel centerline temperature enriched thick plate	540 F
Avg. fuel centerline temperature enriched thin plate	529 F
Avg. fuel centerline temperature natural thick plate	526 F
Avg. fuel centerline temperature natural thin plate	513 F
Max. heat flux in thick enriched fuel plate near control rod	155,000 Btu/(hr)(ft ²)
Max. fuel centerline temp. in enriched thick plate	660 F
Average core height nonboiling ratio	0.40
Average core height boiling	2.4 ft
Mean core exit quality based on total coolant circulated	1.56 per cent
Average voids at core exit	28.5 per cent
Average voids in core height (48 inches)	10.5 per cent
Average voids in chimney	16 per cent
Subcooling at core inlet	5.2 F
Estimated burnout heat flux	750,000 Btu/(hr)(ft ²)
Total recirculation flow ratio	63.0/1.0
Steam to water velocity ratio	1.5/1.0
Downcomer flow area	23.0 ft ²
Velocity in downcomer	0.9 fps
Coolant flow area in fuel zone	8.43 ft ²
Coolant inlet velocity in fuel cluster	2.5 fps

No provision has been made for shielding the area adjacent to the dryer drum; however, should circumstances deem it necessary, a lead brick wall can be constructed on the existing platform.

Normally, during operation, the built-in stainless steel emergency cooler within the dryer shell is inoperative. On loss of dc power, or on loss of air pressure, a solenoid-operated pilot air-diaphragm valve in the supply line from an overhead storage tank (the 15,000 gal tank previously referred to) opens to admit water into the cooling coils, and the 4-inch valve below the cooler opens. The steam formed on passage through the coils is circulated back by natural convection up to the top of the storage tank above the water surface. The cooler is designed to remove 1000 kw (heat) following an emergency reactor scram at 600 psig. To permit servicing the cooler supply water automatic valve, a shutoff valve is provided upstream of the control valve, and provision is also made for draining the cooler element if desired.

Water in the overhead storage tank is maintained at a fixed level by an automatic pneumatic control system. A float-type liquid level controller mounted in the overhead storage controls the position of a pneumatically operated valve in the Laboratory water supply line. On too low a level in the tank, the valve automatically opens. Should there be a failure of the compressed air supply, which will be announced in the control room by the alarm system, the valve will open to its maximum position. Before the tank level exceeds the overflow line to the retention tanks, a high-level alarm will sound to advise the operator, during this condition, to close the shutoff valve upstream of the control valve.

Since this water supply is vital for enforced emergency shutdown and during a casualty, such as a ruptured steam line when the building spray system is required, the piping, valving, and the tank are protected from missiles. The 2-inch pipe is encased in the concrete walls, valves are behind the missile shield wall, and the tank is protected from flying objects by the concrete and steel lined first floor ceiling.

If the operator desires to use the cooler, the previously mentioned control valve may be operated directly from the control room. The position of the valve, whether opened or closed, is indicated by telltale lamps mounted with the valve switch. In addition, a flow alarm switch is provided in the flow circuit to annunciate the cut-in of the emergency system. Prior to cut-in of the unit the reactor is isolated from the steam system by closure of the motor-operated main steam stop valve; failure to do so will allow the transfer of the reactor water over to the steam cycle.

The performance of the dryer during operation will be established by test of the effluent steam. A steam sampling station is located in the discharge line immediately above the dryer. A similar unit is located

upstream. By comparison of the long-lived activities in the steam condensate before and after the dryer, the effectiveness for dissolved and insoluble solid removal can be checked.

A liquid level float switch is mounted on the shell of the steam dryer and is normally used only during the period of reactor shutdown when the reactor vessel is flooded so as to provide a water shield above the core for protection to personnel engaged in fuel element handling or core inspection operations. The float switch annunciates that the water level is approximately $9\frac{1}{2}$ ft above the reactor core. The shield flooding operation, a relatively slow process at a rate of 10 to 20 gpm, will be done with the reactor cover plate in position. The function of the alarm is to advise the operator to close the water supply valve so as not to flood the lines leading to the relief and main steam stop valve.

The total steam flow from the reactor is metered by a stainless steel nozzle in the 6-in. branch pipe leading to the turbine and bypass flow control valve. This instrument gives a measure of the reactor power for a given inlet feedwater temperature and steam pressure. A recording flowmeter on the main control panel maintains a continuous record of power or flow performance. Reactor power in kw, at 600 psig and condenser vacuum of 27.5 in. Hg, can be calculated from the steam flow (lb/hr) by multiplying by the factor 0.33.

In addition to the flow-recording function, the flow nozzle is an element of the feedwater control system. In this system, the feedwater regulating valve functions to maintain an equality between the steam and feedwater flow for a range of reactor level control.

The steam plant can be isolated from the reactor by the closure of the main steam stop valve. This floor stand, motor-operated, 6-in. gate valve, mounted on the main floor of the plant, can be operated either electrically from the control room or by hand at the valve. Limit switches, which actuate at either the full-closed or full-open position, energize telltale pilot lights on the control room operating panel to indicate the extreme position of the stop valve. Normally, when the plant is operating the stop valve is in the full-open position. Its main use will be during reactor shutdown periods. When it is desired to maintain the reactor vessel at pressure, the main stop valve is closed.

The reactor is intended to be normally operated at a constant power or steam flow rate. To minimize the regulation of the reactor power to suit the electrical demand, a system of dumping the excess power, i.e., the difference in the reactor power over the electric demand load in the form of steam, is utilized in the EBWR system. The accomplishment of this function is the purpose of the steam bypass valve, which in the flow circuit bypasses the turbine and delivers the steam load to the condenser.

The steam bypass valve is single seated and can be set for either automatic control or manual hand, local or remote electrical positioning. For most of its service it will be set for automatic operation. The automatic control system which sets the valve position is hydraulic. With a fixed steam flow from the reactor, an increase in turbine load will result in the closing of the bypass to divert the flow to the turbine. It should be stressed here that if the load demand increase is greater than the bypass flow, the sole recourse, to maintain the power output, is then the repositioning of the control rods. For a decrease in the demand load, the excess steam is automatically transferred to the condenser by further opening of the bypass valve.

For the case of an increased reactor power setting, for a fixed electrical load, the effect is to increase the system pressure. The increase in pressure that ensues, is sensed by the pressure control element of the automatic system, which tends to regulate the set pressure by relieving the excess through the opening of the bypass valve.

For the case of actual overspeeding of the turbine generator, either due to an accidental loss of load, or an intentional turbine trip to periodically check out the turbine safety control, the bypass valve control system is so designed that the closure of the turbine trip throttle valve results in the opening of the bypass valve to dump the steam load.

The control system is operative over the variable pressure range of 100 to 600 psig. The valve is designed to pass 100,000 lb/hr of steam at 300 psig.

The flow diverted through the bypass system is measured by an in-line flow nozzle in the 6-in. line upstream of the valve. The flow nozzle has been sized for the condition of full reactor flow bypass; the actual recorder in the control room is capable of metering up to 100,000 lb/hr. For bypass flows of the order of 5 to 10 per cent, this recorder will not be very effective for the measurement of flow. The addition of a differential head transducer with the lower maximum differential head for the same signal to the recorder can be made at a later date if required.

The discharge from the bypass valve passes through the desuperheater and then to the condenser. The desuperheater is a 14-in. diameter carbon steel pipe, containing two water spray nozzles, which extends across the condenser below the tubes and above the water level in the divided hot well. The steam leaves the desuperheated pipe through a series of 2 sets of drilled holes along the horizontal centerline of the pipe. The entering steam, upstream of the spray nozzles, is at the same enthalpy as the high-pressure steam leaving the reactor. At condenser vacuum conditions of 27.5 in. Hg, the steam temperature is approximately 320 F (110 F superheat).

Under normal plant operating conditions the desuperheater spray nozzles are inoperative. Since the superheated steam enters below the condenser tubes, in the hot well, the condensate water shower from above desuperheats the steam to prevent the high temperature from reaching the shell and tube surfaces. If the condenser is operated with only one-half the tubes in service which might be required if the tubes on one bank leak badly, it is essential that the desuperheater spray nozzle be operated. Without the desuperheater spray in service and with one-half the tube bank drained, the entry of the superheated steam on the drained half tubes would create an expansion differential between the operating half of the "cold" tubes, and the inoperative half of the "hot" tubes. The resultant stresses on the tubes could loosen the tube sheet rolled joints and thus cause a greater leakage.

To prevent this from happening, the control of the desuperheater spray nozzle valve is automatic in operation. The closure of the inlet and outlet motor operated valves to each half of the condenser actuates limit switches at the closed position of these valves which then energize the desuperheater spray solenoid valve. Power to the solenoid opens the valve to admit feedwater to the spray nozzles. In addition to this automatic control feature, manual electric control is also provided at the discretion of the operator.

The relief and safety valves also discharge into the desuperheater pipe. These valves provide protection to the reactor vessel in case of combined accidental closure of the turbine trip throttle valve and bypass valve. The relief of the pressure within the reactor vessel is in three stages.

The relief valve (Fig. 5-10) is a self-actuated (steam) internal pilot piston operated unit sized to pass 100,000 lb/hr at 350 psig inlet. The opening set pressure can be varied by the handwheel control unit on the superstructure of the valve. The adjustable pressure range with this manual control is 350 to 650 psig with the proportional band approximately 10 per cent of the set pressure, e.g., if the set pressure is 610 psig, the pressure at which the valve opens, the valve will be wide open at 675 psig. At this operating pressure, the valve will then pass approximately 200,000 lb/hr, or approximately three times reactor power steam generation. Failure of the mechanical loading compression spring will result in the valve going to the wide open position.

In addition to the self-actuated relief valve, two standard safety valves also provide protection against overpressure. One is set to pop open at 725 psig; this valve will dump approximately 65,000 lb/hr of steam. The valve will reseal at approximately 695 psig. The second safety valve will open if the reactor pressure exceeds 750 psig; its discharge capacity at this pressure is also approximately 65,000 lb/hr. Each valve is provided with

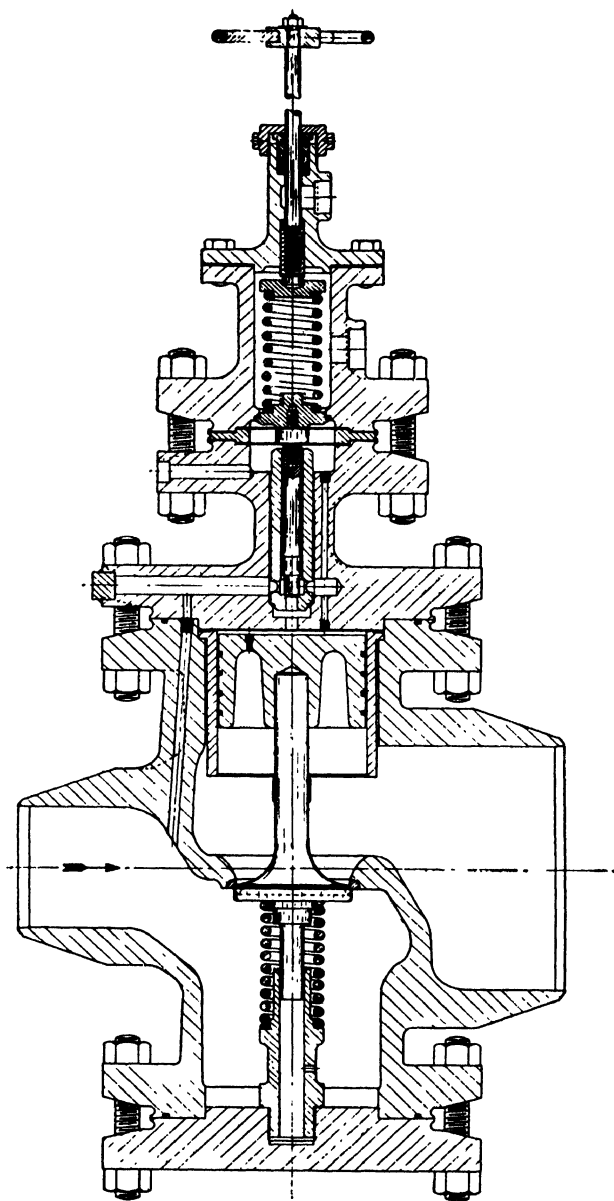


FIG. 5-10. Pressure relief valve.

a packed lifting lever to permit periodic checking of the mechanical operation of the valve. When the safety valves open, the reactor will scram due to the interlocking of the safety valve limit switches with the rod control circuit.

The two safety valves, the overpressure relief valve and the forementioned motor-operated stop and bypass valves are all welded in the steam system piping. The turbine trip-throttle valve is similarly integral with the piping. Welding has been adopted to minimize steam leakage into the plant area, and to eliminate contamination from gasket material corrosion. To reduce the possibility of leakage from the valve bonnets, a pressure-type seal is used for the main steam stop valve and a seal weld bonnet closure on the relief valves and bypass valve. The safety valve bonnet joint is a gasketed-bolted connection.

The turbine trip-throttle valve does not have a standard bonnet closure. Access to the valve internals is from the underside of the valve plug through a bolted and gasketed blind flange. The sealing of this flange and gasket against outleakage to the surroundings is one of the many turbine components that is connected to the special seal and recovery system which is described in a later section.

Different methods of sealing the valve stems of the forementioned valves in the steam system are utilized. The main steam stop valve has an overstandard depth packing gland with packing above and below a lantern ring; thus leakage along the valve stem is collected at one point. This type of construction permits collecting the leakage if the total leakage from the plant is too high, especially during future reactor operation with heavy water.

The lifting gear stem of the safety valves is connected to the condenser vacuum side; therefore inleakage of atmospheric air is of prime concern. For this application the sealing is accomplished with O-rings.

Since there is a relatively long pipe run between the takeoff point, at the outlet of the steam dryer-emergency cooler to the safety and relief valves located at the condenser desuperheater, provision must be made to keep this line relatively free from collected water. The condensate from the relief line is discharged through a steam trap to the condenser. In addition to this flow, there will always be a fixed leakage through the relief valve. The latter was done intentionally to keep the piston and cylinder of the valve at uniform temperature to minimize distortions and resulting differential expansions between these close-fitting members.

Thus far, description of the components in the steam system has been limited to only one path for steam flow when in normal operation, i.e., the bypass path, and when abnormal conditions exist, the pressure relief and safety path.

The EBWR plant has been designed to demonstrate the practicability and also the problems arising from the generation of electric power with radioactive light or heavy water steam. The turbine of the EBWR plant therefore, has design features which are radically different from the conventional turbine of the same pressure, temperature, and power rating.

Personnel manipulations around the turbine are normally confined to only the startup and shutdown operations. During the startup warming period the turbine turning gear must be placed into service. This is done after the auxiliary oil pump is started from the control room by starting the turning gear motor and operating the engaging mechanism at the turbine. Also, the trip reset must be restored to the run position before the trip-throttle valve can be opened. Both these operations and check of the overspeed trip cutout and automatic startup of the auxiliary oil pump are performed locally at the turbine during the warmup and rolling periods.

Standard practice is used to start the turbine. With the reactor at operating pressure and at reduced power level sufficient to maintain the heat losses and the air ejector steam load, the turbine trip-throttle valve may be opened wide. Changes in steam flow are then made by regulation of the motor-driven speed changer up to idling speed. At approximately 3000 rpm, the turbine governor control will maintain the turbine speed constant for a given speed changer setting.

Supervisory instruments with alarm features are provided to ensure proper operation of the unit. Vibration recorders indicate any turbine rotor unbalance. Bearing oil temperatures are measured continuously; an audible alarm is actuated if the temperatures rise above normal. The oil pressure to the turbine-generator bearings and the governor oil pressure are both indicated in the control room. Failure to maintain the proper oil pressures with the shaft-driven oil pump when the turbine is running is immediately annunciated; automatically the electric driven auxiliary oil pump is started to take over the oil delivery function.

The operator is also advised by the warning system of either too high or too low a level in the oil reservoir, too high an exhaust temperature, and leakage of the generator air cooler. Thus the plant, as evidenced by the turbine controls, has been designed for minimum attendance in the plant proper and for remote control from the control room.

In the operation of the turbine trip-throttle valve, two methods of load control are possible. If the turbine load were suddenly removed as would occur with a fault in the electrical distribution system, the overspeed trip device would trigger open the bypass valve so that the reactor load is maintained by steam dumping. In addition, the actuation of the overspeed trip can also scram the reactor by the electric interruption of power

to the rod circuits by the opening of a limit switch when the trip valve fully closes. Either type of control, i.e., reactor scram or the continuation of reactor operation on closure of the trip-throttle valve, is possible, depending on the selection by the operator.

At the turbine design 5000 kw rating the exhaust entering the condenser contains approximately 17 per cent moisture (including the quantity of water collected in the moisture extraction stages). At this steam rate, 62,750 lb/hr, with zero flow through the bypass valve, the corresponding reactor power is 20.7 Mw. The condenser thus dumps 15.7 of the 20.6 Mw of power generated.

The condenser is designed to dump the above heat load, plus 6000 lb/hr of superheated bypass steam to give a total condenser heat load of 17.7 Mw at a pressure of $2\frac{1}{2}$ in. Hg (abs.) with 13,650 gpm of 85 F circulating water. The corresponding reactor heat load for the condenser design duty is 22.7 Mw. If the turbine is completely bypassed and the reactor steam load is dumped to the condenser, an increase of approximately 30 per cent in the heat load is imposed on the condenser, and for the same coolant water conditions there will be a slight reduction in the condenser vacuum.

To minimize the entry of contaminants from the higher pressure circulating water system into the evacuated steam space, double tube sheet construction has been used. The condenser tubes are rolled in the twin tube sheets at each end; the space between the tube sheets is maintained at vacuum conditions. The bottom of the tube sheet space is piped to a 1200-gal vacuum tank located at a lower elevation. A leaky rolled tube joint adjacent to the inlet or outlet water box will leak into the evacuated space between tube sheets and eventually drain to the inner tube sheet compartment of the 1200-gal vacuum tank. In the design of the latter, any leakage must be detected locally by viewing the sight glass on the tank. At a later date, for application with heavy water, a high-level alarm will be added to advise the operator remotely of this undesirable leakage condition.

Evidence of a tube failure can be detected by two methods. The in-leakage of the high-conductivity circulating water into the feedwater system will show an increase in the conductance of the feedwater. Two conductivity cells, one in each of the twin suction pipes to the feed pumps, are installed for this purpose. An alarm integral with the conductivity recorder in the control room announces the abnormal feedwater condition to the operator. Since the condenser has a divided hot well, an increase in the feedwater conductance in one-half of the condenser indicates to the operator which portion of the condenser contains the leaking tubes. Any inleakage of circulating water from a ruptured tube will result in transport of soluble impurities into the reactor. The net effect of their

entrance into the system will be an increase in the radiation level throughout the plant. Assuming no other source of contamination, by following both feedwater conductivity and system activity the leak integrity of the condenser can be checked continuously.

Upon detection of circulating water leakage into the system by the forementioned conductivity recorders, the operator may decide to secure the reactor and make the necessary repairs. To minimize the entrance of a quantity of circulating water into the steam space and hot well of the condenser, provision is made for relatively fast drainage of the circulating water from the tubes and inlet and outlet water boxes. Closure of the inlet and outlet motor-operated valves to one-half of the condenser restricts water circulation through this half. When these valves are closed, it is only then possible (due to electrical interlocking) to open the water box motor-operated 6-in. drain valves located at the turbine inlet end of the condenser. With the circulating water inlet end water box drain valves open, the circulating water in the outlet end water box drains through the pitched tubes to the inlet end water box and thence, through the drain valves, to the 1200-gal vacuum tank. From the latter the collected circulating water can be disposed of by gravity drainage to the plant retention tanks. The prime purpose of the quick dump system is for operation with heavy water. It has been incorporated in the EBWR system to minimize the contamination of the system heavy water with the circulating light water.

To prevent accidental flooding of the vacuum tank, the operation of the 20-inch condenser inlet and outlet valves are electrically interlocked with the 6-in. drain valves. In addition to the interlock sequence previously mentioned, the inlet and outlet valves cannot be opened unless the drain valves are closed. To ensure tight closure of the drain valves, they are the Saunders patent diaphragm type; a neoprene diaphragm provides the seal on the valve seat. The leakage that does occur through the drain valve with approximately 3 atmospheres differential across the valve can be locally detected by the sight glass indication on the vacuum tank. A remote warning high-level alarm device will be added at a later date.

Special attention has been given in the design of the EBWR condenser to prevent the leakage of atmospheric air into the system. This design feature is again a requirement for heavy water usage. All joints of the condenser are welded; the turbine exhaust to condenser neck extension is a weld joint; the inner tube sheet of each end is welded to the shell; manholes are sealed by welding and all piping connections to the condenser are also welded. The only joint on the condenser not welded is the 20-psig rupture disk which is bolted and gasketed to the extension neck.

The condenser elevation and hot well storage capacity have been selected such that the "cold" reactor water level and the condenser water level are the same so as to keep a minimum water inventory when the plant is operating with heavy water and with the reactor and steam plant shut down, the fuel plates are maintained submerged without the necessity of controlling level with the feed pumps. With the reactor cold water level at elevation 715 ft 1 in., the corresponding capacity within the hot well is approximately 250 gallons. Normally the elevation in the condenser hot well remains at this value, but in the reactor, as soon as operating conditions of pressure, temperature and steam volume are attained, the "hot" level in the reactor vessel is 716 ft 11 in. which gives 4 ft 5 in. of water above the active portion of the fuel elements.

If the reactor level is too high at startup, water can be transferred from the reactor to the condenser by opening, locally at the condenser, the two startup equalizing valves. The flow from the reactor to the condenser is through the reactor purification system. The reactor water is passed through the mixed-bed ion exchange to ensure that irradiated corrosion products are not directly transferred to the condensate system. To make the transfer from reactor to condenser, it is necessary to have one or both of the reactor purification pumps in operation. By observation of the televised view of the reactor level column sight glass, the operator can secure both pumps and valves when the level is at the correct value noted above. Failure to stop the transfer will result in an alarm when the reactor level drops 2 in. below cold normal. An additional alarm will sound if the process is continued by the actuation of the high level float switch on the condenser hot well. The condenser high-level alarm is set to annunciate when the hot well level approaches within 2 inches of the desuperheater pipe; this is approximately 8 inches above normal hot well level. If these warnings are neglected, the automatic injection of the low-pressure boric acid spray system will be initiated, due to the combination of low-water level, 6 inches above the top of the core, and low pressure, less than 20 psig, in the reactor vessel.

In view of the leak proof design of the plant piping and components, the demineralized makeup water storage capacity is only 200 gal. The demineralized, but nondeaerated, makeup is added directly to the condenser. This operation is carried out remotely from the control room by opening the motor-operated makeup valve. Deaeration is accomplished within the condenser and can be efficiently carried out over a wide load range due to the location of the desuperheater pipe directly above the hot well surface and the bypass valve steam flow provision. By the intimate contact of the steam leaving the desuperheater pipe, the water within the hot well will always be saturated. The condenser manufacturer guaran-

tees the dissolved oxygen in the feed water leaving the condenser will not exceed 0.01 cc/liter of condensate, and that an oxygen concentration of 0.005 cc/liter can be readily attainable.

The insoluble gases that enter the condenser due to (a) atmospheric air leakage, (b) the water decomposition gases, and (c) dissolved air in the makeup water, are removed from the condenser by twin, two-stage, steam jet air ejectors. Evacuation is initiated by reactor pressure in excess of 100 psig with the startup heater in service. A pneumatic, pilot-operated pressure-reducing regulating valve maintains the steam pressure at the ejector nozzles at a fixed pressure of 100 psig.

The ejectors can be started only from the control room. All valves in this system, the four steam valves at each of the twin set of ejector nozzles and the first and second stage air inlet valves, are pneumatically operated.

A sequence-type control is used for the operation of the ejector system valving. Control air pressure is first admitted to the air-operated steam valves supplying the first and second stage ejectors. After the steam valves are wide open it is then possible (by electrical interlocking) to open, by air pressure again, the air inlet valves to the first and second stage of the corresponding ejectors. With all valves open the air-removal system is in operation.

Normally, one of the twin set of ejectors is used to maintain condenser vacuum. Both sets are cut into service during "hogging," i.e., the initial evacuation during turbine startup, when high air-removal rates are encountered. During this phase the 2-in., air-operated, bypass valve across the ejector air discharge flow meter is opened automatically, by electrical interlock, to protect against overrange of the flow meter. The discharge of air and other noncondensable gases, not condensed in the catalytic recombiner, is continuously recorded in the control room to keep the operator advised of any abnormal air leakage. The effluent gases can be discharged to either the vapor recovery system or directly to the stack and outside surroundings.

Loss of vacuum will result if there is either a loss of air pressure to the air-operated steam and air valves, or loss of power to the solenoid pilots controlling these valves; such loss will result in the isolation of the ejectors and, therefore, the elimination of the air-removal equipment from the system. The decrease in vacuum and its rate will depend on the rate of formation of water decomposition gases, and leakage.

The loss of vacuum either by the above cause or by the more serious casualty of loss stoppage of the circulating water will result in, first, the annunciation of the loss of vacuum condition and eventually the initiation of the reactor scram. The alarm will sound when the condenser absolute pressure reaches 5 in. Hg; normal pressure is to be $2\frac{1}{2}$ in. Hg (abs).

An increase in pressure up to 20 in. Hg (abs) will cause the reactor rod trip circuit to be de-energized and initiate the insertion of the control rods to stop the generation of steam. This automatic stoppage of condenser pressure buildup is far below the design maximum pressure for the unit; the condenser shell is designed to withstand an internal pressure of approximately 20 psig; this is the pressure at which the rupture disk mounted on the condenser extension neck will open to protect the condenser.

Insertion of the control rods into the reactor core does not imply that the heat generation process terminates, as it does with conventional plants when there is a fuel stoppage. For the nuclear type plant, heat is continuously generated, but at a much lower rate over an extended period. After an elapsed period of 7 hours after a reactor scram, the heat generation rate is still of the order of 1 per cent of design rating. This heat must be removed, otherwise the reactor pressure will continuously increase by steam formation and, if continued evaporation takes place without maintaining the water level, it is possible to expose the fuel elements and bring them to melting or clad failure temperature.

Since the condenser circulating water system is the heat sink for the plant and because it is of utmost importance to remove the decay heat after reactor shutdown, the condenser circulating water system has been designed to permit reactor heat dissipation for the condition of power failure to the circulating water pump motors. The centerline of the condenser is located at an elevation $10\frac{1}{2}$ ft below the water surface of the cooling tower basin. With heat addition to the circulating water from higher temperature condensing steam on the condenser shell side, and with the return circulating water lines continuously rising towards the basin, the difference in density between the cold water from the basin and the hotter water in the vertical return would promote natural circulation. The tubes in the condenser have been purposely inclined, 2 in. in 20 ft, to assist in promoting natural circulation. Due to the fact that the circulating water pumps, connected to the water outlet end of the condenser, are located at a lower elevation than the condenser (dictated by plant arrangement), the effectiveness of the intended natural circulation loop is somewhat impaired. Once circulation is started, it will maintain itself; the uncertainty with the present circulating water circuit is inducing the heated fluid to flow down through the pump loop and up to the riser return to the tower.

Normally the pumps discharge the circulating water to the top of the cooling tower, but for the emergency natural circulation provision, this line is bypassed by the opening of the 8-in., motor-operated, bypass valve connected directly to the tower basin.

The condensate leaving the hot well is pumped back into the reactor by the reactor feed pumps. Two are provided; alternately one acts as the main unit and the other the spare. These multistage pumps are protected from operating dry by a low level switch which is actuated when the hot well storage capacity is less than 100 gallons. This circuit interruption stops the pumps to protect them from possible damage resulting from cavitation. Manufacturers' tests have demonstrated that the pumps may be run with a net positive suction head much less than the normal submergence of approximately 17 ft.

With the discharge valves closed, protection against overheating is provided by the continuous leakage of 5 gpm through the inboard breakdown balance bushing to the condenser. In addition to this high-pressure leakage, from the high-pressure breakdown bushing through the low-pressure (condenser) leakoff connection, there is also the intermediate pressure breakdown bushing. The pressure differential across the intermediate bushing is from 100 psig to condenser vacuum pressure; the leakage through this bushing to the condenser is less than 5 gpm. The 100-psig intermediate pressure, from the first stage impeller volute, is the water pressure against which the mechanical seal has to operate to prevent water leakage.

The seal housing of the reactor feed pumps is designed to contain two types of seals, a mechanical seal and the throttling labyrinth type. The former maintains the outleakage to negligible values when performing properly, and the latter, with dry air and vacuum leakoffs maintained by the recovery system, collects any leakage from the former and returns it to the recovery system. The labyrinth seals also act as a backup seal in case of mechanical seal failure.

The discharge from the feed pumps, at a pressure of approximately 900 psig corresponding to a feedwater flow of approximately 120 gpm for 20-Mw reactor power, is the coolant for the condensation of the steam load of the ejectors. The feedwater flows in series through the two-pass intercooler and the single-pass aftercooler.

The minimum flow through the ejectors to condense the steam load required by the ejectors (325 lb/hr) is approximately 70 gpm. An automatically controlled recirculation system ensures that a minimum of 70 gpm flows through the ejectors. A flow nozzle, in the 3-inch feedwater line, in conjunction with pneumatic differential pressure transducer and pneumatic relay, constitutes the major control element which regulates the position of the pneumatically powered recirculation valve. The feedwater recirculated is returned to the condenser.

At power levels corresponding to feedwater flow rates less than 70 gpm, the recirculation system bypasses an amount such that the total of the

feedwater flow to the reactor and the amount recirculated equals 70 gpm. As the reactor load and correspondingly, the feedwater flow to the reactor is increased, the amount recirculated to the condenser is accordingly automatically decreased until at a reactor power of 11,700 kw, equivalent to 70 gpm feedwater flow, the recirculation is zero.

This air-operated system is initiated locally; annunciation of loss of air pressure to the recirculation valve diaphragm can warn the operator that the system is not functioning properly. On loss of air pressure, the recirculation valve will open wide, but adequate prior warning is given by the annunciator system.

The total feedwater supplied to the reactor through the feedwater ring and the nine control rod thimbles is measured by a flow nozzle in the 3-in. feedwater line downstream of the circulation flow nozzle. Control of the flow rate to the reactor is such that a constant, but adjustable, water level in the reactor is maintained. Provision has been made to substitute condenser water level control for reactor water level control if the former is considered desirable. The substitute arrangement would maintain the hot well (usually at a relatively constant temperature or density) at a fixed value which, in turn, would allow the reactor level to vary with temperature.

Before the feedwater enters the reactor, it passes through disposable cotton fiber filters to minimize entrainment of particulate matter into the reactor vessel. Provision is made during startup to bypass the feedwater regulator. A 2-in. motor-operated valve, in parallel with the feedwater regulator, can be positioned from the control room to regulate the flow and therefore level in the reactor vessel. The valve will be placed in service only for operation of low flows or during abnormal operation of the feedwater regulator. Corrosion of the EBWR system is to be expected, the amount and type depending on the oxygen liberated by radiolytic decomposition and the materials in contact with the fluid. For reasons of economy, corrosion resistant materials have not been used completely throughout the steam/feedwater cycle. In the steam phase, stainless steel type-304 is used for the in-shield piping between the reactor vessel and the all-stainless steam dryer-emergency cooler. Beyond the dryer the piping to the turbine, bypass valve and relief valves is nickel plated. Although the major portion of the turbine surfaces has been nickel plated, the first stage curtis nozzle block and turbine rotor are of conventional, less corrosion resistant materials.

As for the condenser, some corrosion of the carbon steel shell, tube sheets, steady plates, and piping from the hot well to the suction of the feed pumps can be expected at a greater rate than the aluminum condenser tubing. Beyond the feed pumps constructed of 12 per cent chrome

steel, the remainder of the feedwater system back to the reactor is stainless steel.

Such corrosion products that do result tend to settle in low-velocity regions. With frequent shutdown and startup of the experimental unit, the full-flow filters can prevent slugs of corrosion debris, and any foreign matter not eliminated by initial plant cleaning, from entering the pressure vessel.

Two filters are provided; normally one is in service, the other as the standby. The time of replacement or transfer to the clean standby unit can be established by the operator by the progressive change in the pressure drop across the unit. Local pressure gages before and after the filter will indicate fouling of the cartridge elements.

Water sampling taps have been provided upstream and downstream of the filters as a means of checking filter performance by the reduction in gross activity on passage through the filter. In addition to the provision for sampling and analysis of the feedwater, a continuous monitor of the activity level in the filter area will be maintained. The continuous removal of corrosion products that do enter the reactor vessel is the function of the reactor purification system. This all stainless steel system removes water from the bottom of the vessel, through 1 of the 4 future 6-in. recirculation lines, and after cooling in 2 stages, delivers it to the ion exchange columns after which it is pumped back with the feed water into the reactor vessel.

Two duplicate systems are incorporated and the components are sized for two functions: (1) purification, and (2) shutdown cooling. The ion exchanger, and cotton filter units before and after the columns, and the regenerative cooler are sized for the normal purification flow of 10 gpm. The canned rotor pump, secondary cooler and bypass piping around the ion exchangers and regenerative cooler are sized for the shutdown cooling flow rate of 50 gpm. Shutdown cooling, in this case, is different from the process of disposal of reactor heat at high temperatures and pressures. The high flow rate condition has a limited application during operations when the reactor top cover flange is removed for core inspection and/or fuel replacement operations. The function of the ion exchange system pumps and cooler for this condition is to maintain the temperature of the reactor water at relatively low level (100 F), to minimize surface evaporation and thermal convection currents so as not to impede inspection and fuel handling operations. During this phase the reactor water is well above normal operating level, to provide adequate shielding for the personnel engaged in these operations.

The flow rate of the reactor water to the ion exchange system can be regulated from the control room by the positioning of the motor-operated

flow valves in each of the two systems. The flow, measured locally at the ion exchangers, is set to maintain the lowest conductivity of the effluent leaving the ion exchange columns. Both the influent and effluent conductivities are recorded continuously; too high a conductivity of the water to the resin columns advises the operator of abnormal performance. Similarly, too high a fluid temperature (120 F) to the resin beds is also annunciated.

The components and piping of the reactor purification system will be one of the greatest sources of radioactivity in the plant, since there will result an accumulation and concentration of irradiated corrosion products. For this reason both the ion exchange prefilter and the ion exchange column containers are integrally lead shielded to provide radiation protection when in service and to serve as coffins for shipment to disposal areas. Shielding of the entire area may be required depending on the activity level. For this situation, a concrete block shadow shield wall can be installed to surround the system. Radiation monitors are mounted on each ion exchange column and prefilter and are incorporated to warn the operator of intolerable radiation levels.

The ion exchange system may be in operation, prior to reactor startup, to clean up the reactor water before any heating. But when the startup heater, which utilizes auxiliary steam to heat the reactor to 325 F (which corresponds to a reactor pressure of 100 psig), is put into service the reactor purification system can be stopped since heat is rejected by the secondary cooler to the circulating water.

In addition to the normal function of maintaining a high purity of the reactor water, the ion exchange system will be utilized to remove the boric acid from the reactor vessel, introduced by operation of either the high-pressure or low-pressure boric acid injection systems, described elsewhere in this book.

DESCRIPTION OF EBWR COMPONENTS

A. Reactor. 1. *Fuel assemblies.* The primary fuel element selected as a reference for the first core of EBWR is a plate in which a core of U-5 w/o Zr-1.5 w/o Nb is clad with Zircaloy-2. The cladding is metallurgically bonded on all sides to the core material which is heat treated to give maximum radiation stability. The combination was chosen because it appeared to offer the best solution to the early design requirements which could be arrived at from available materials.

Two early design specifications for fuel elements for the reactor were: (1) A long lifetime (at least 10 years) to reach the one per cent burnup of core atoms considered necessary to make an economically feasible

power-producing reactor.^{7,8} (2) The required strength, irradiation stability, and integrity must be attained with a minimum of absorbing material in the core, cladding, and associated structural material.

To meet these requirements it appeared that fuel should be largely uranium with the minimum amount of alloying elements for strength, corrosion resistance, and irradiation stability, and the minimum degree of enrichment for reactivity.

Since efficient power reactors require high water temperatures (486 F was specified for this reactor), which greatly accelerate the corrosion of uranium, a corrosion resistant cladding was considered necessary. The requirements for high-temperature corrosion resistance combined with low neutron cross section are satisfactorily met by Zircaloy-2, selected as the reference cladding material.

Since the satisfactory application of protective cladding with 100 per cent efficiency appeared impossible, a corrosion resistant core alloy was originally specified, and a great deal of effort was expended in developing a corrosion resistant core material. When it was subsequently found that heat treatments of the reference alloy for corrosion resistance and irradiation stability were incompatible, the heat treatment for irradiation stability was selected, and numerical specifications on corrosion resistance were dropped. The decision to heat treat for irradiation stability left all the responsibility for corrosion protection to the cladding, and every possible effort has been made to ensure cladding integrity.

The fuel assemblies for the EBWR core are rectangular, box-like structures, 77 $\frac{5}{8}$ in. long and 3 $\frac{3}{4}$ in. square. The components of each assembly, as shown in Fig. 5-11, are the lower locating end fitting, the fuel plates, and the top fuel handling fitting.

The locating end fitting is machined out of a stainless steel tube 3 $\frac{7}{8}$ in. OD by 3 in. ID. Pilot fingers are provided on the bottom end to guide the assembly into the support plate location holes. Two chamfered seating shoulders are machined on the periphery of the tube to ensure verticality of the assembly when in position in the support plate. A recess is turned on the upper surface of the fitting to provide a seating area for the side plate and to eliminate any projection which might *hang up* the assembly during insertion.

The fuel element for the reactor is a Zircaloy-2 clad, ternary uranium alloy plate (Fig. 5-11) manufactured in two thicknesses and in two enrichments. The reason for the different types of fuel elements is to provide a means of adjusting the criticality of the reactor with cores of varying diameters. One plate is 54 in. long, 3 $\frac{5}{8}$ in. wide and 0.212 in. thick. The other plate has the same length and width but is 0.280 in. thick. The two enrichments are natural uranium in one case, and 1.44

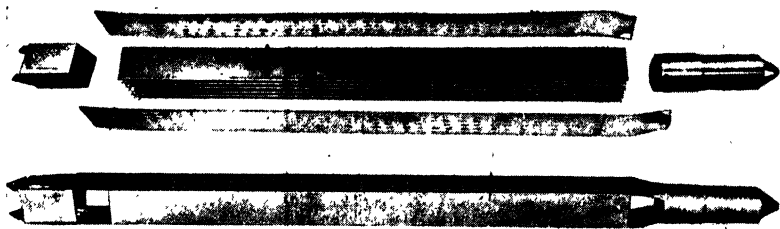


Fig. 5-11. Fuel assembly.

per cent U^{235} enrichment in the other. The reactor is presently loaded with 336 enriched thin plates, 300 enriched thick plates, 12 natural thin plates, and 36 natural thick plates. The arrangement of these plates, in assemblies, is shown in Fig. 5-12. The fissile alloy is clad with Zircaloy-2 sheet by a hot rolling process. The nominal cladding thicknesses are 0.020 in. over the plate faces, 0.125 in. over the edges. The plate edges and ends are machined to size after rolling; the faces remained in the as-rolled condition.

The side plates, which measure $61\frac{1}{2}$ in. long, $3\frac{3}{4}$ in. wide, and $\frac{3}{8}$ in. thick, are fabricated of Zircaloy-2 sheets. The lower end of each plate is formed to effect a smooth transition with the locating end fitting. The purpose of the perforations is to reduce the tensile strength of the side plate without making it so thin that welding to the fuel plates would be difficult. This arrangement permits differential thermal expansion of the fuel plates to be absorbed by the assembly with minimum restraint to the fuel plate.

Slots are provided in the side plates, between fuel plates, which reduce the side plate shear area between plates to less than the total weld area. This permits the side plates to shear before the welds in the event of differential growth between fuel plates. The top fuel handling fitting is made of $\frac{1}{16}$ -in. thick stainless steel sheet, formed into a box $3\frac{3}{4}$ in. square. The sheet is machined to shape, die punched, press brake formed, and welded, using standard production shop practices. The holes in the top ears provide lifting points for the button lugs of the fuel-handling tool. The flaps extending from the side of the box form springs, which space the assemblies uniformly in the reactor and provide some degree of later stability. By compressing all springs in a 9 fuel assembly cell, a clearance of $\frac{3}{4}$ in. in two directions can be realized for withdrawing distorted fuel elements after irradiation.

Edges of the fuel plates are welded to the side plates by large, pluglike, spot welds spaced 3 in. apart along the full length of the fuel plate. Three closely spaced welds are made at either end of the fuel plate.

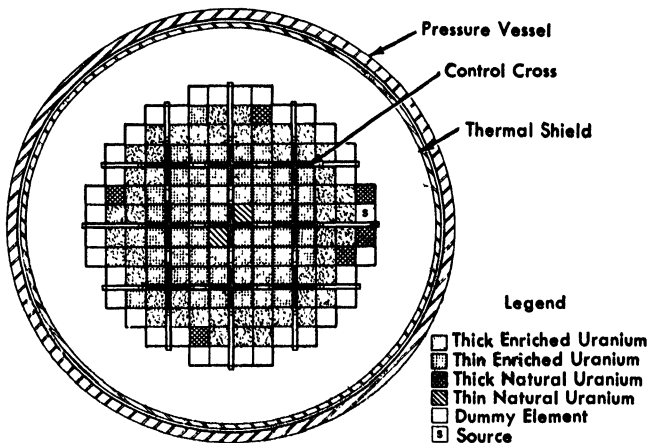


FIG. 5-12. Fuel loading of EBWR.

These more massive weldments are the major load-bearing welds in the event of longitudinal fuel plate elongation; the 3-in. spaced welds act mainly as restraint to lateral fuel plate motion.

The lower locating fitting and top fuel handling fitting are made of stainless steel for reasons of economy. Both components are outside the active section of the core; consequently, the more neutron-absorbent but less expensive material can be used. Riveting is used to fasten these components to the Zircaloy-2 side plates since the two materials cannot be joined by welding. Dummy fuel assemblies are used to fill the outer fuel positions when a core less than 5 ft in diameter is used. The dummies consist of the lower locating end fitting, the body, the hold-down assembly, and the top fuel handling fitting. The top fuel handling fitting is the same as the fitting described previously for the fuel assemblies. The bottom of the dummy ($3\frac{3}{4}$ in. square by $64\frac{3}{4}$ in. long) is fabricated of aluminum-1 weight-per-cent nickel alloy sheet ($\frac{1}{16}$ in. thick). The sheet is press brake formed and spot welded to make a square tube. One side plate is extended to form a tab which runs almost the full length of the body. This tab bears against the side of an adjacent assembly and will serve to reduce flow leakage between assemblies during proposed reactor operation with forced circulation. Aluminum-nickel alloy is used in lieu of zirconium for reasons of economy. The alloy has exhibited satisfactory corrosion resistance in laboratory tests; however, if corrosion is excessive, the elements can be removed and replaced.

Operation with forced circulation also requires the use of a hold-down assembly to withstand the expulsion force of the pump pressure.

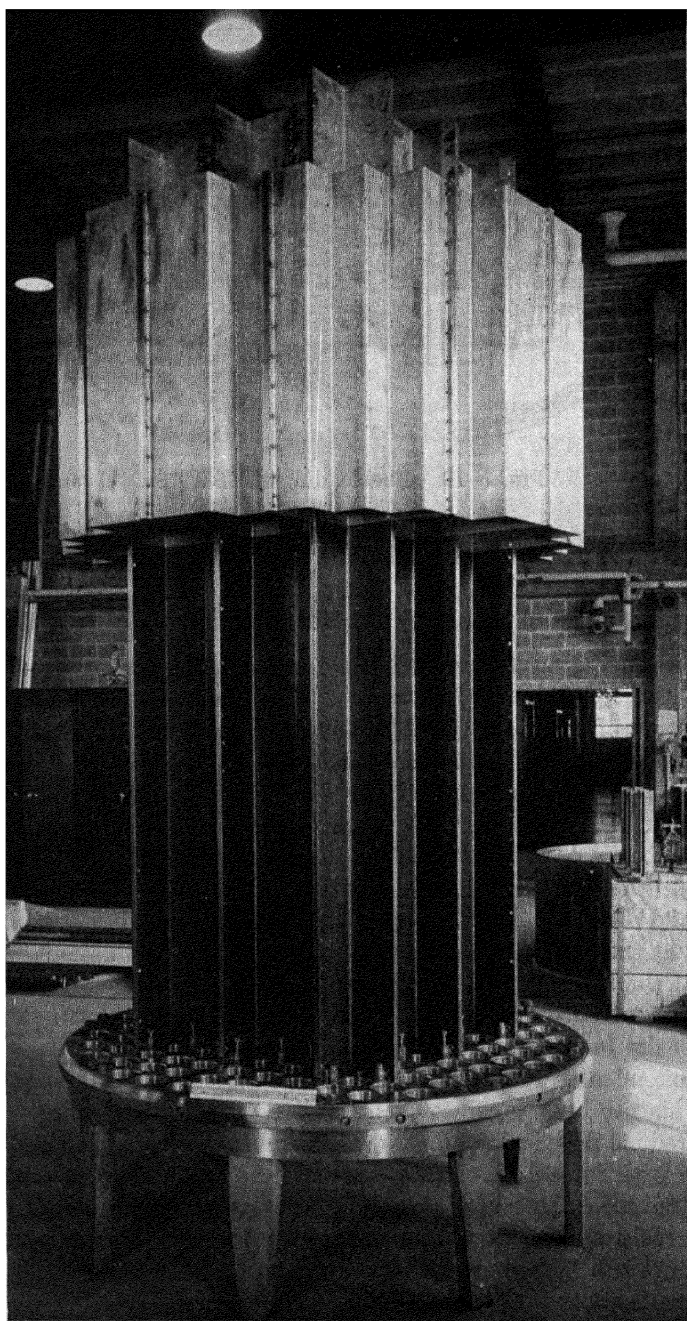


FIG. 5-13. EBWR core structure.

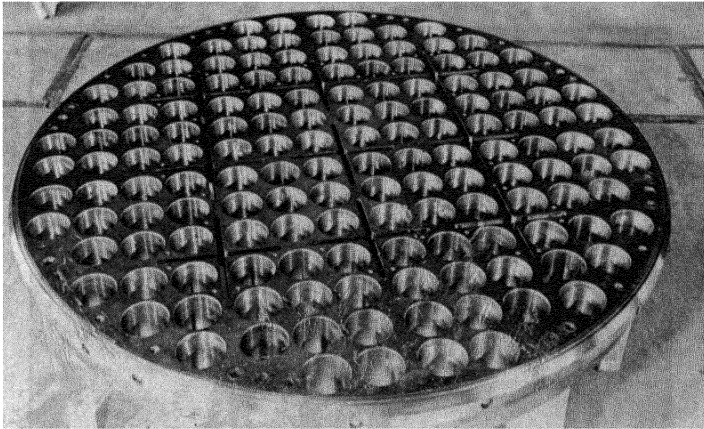


Fig. 5-14. Core support plate.

2. *Core structure.* The EBWR core structure consists of the control-rod guide and top shroud structure, the core support plate to which it is attached, and the core hold-down assembly which bolts to the lower head of the pressure vessel. These are shown in Fig. 5-13. The manufacture of the core support plate and core hold-down assembly presented no unusual problems either in a materials sense or mechanically. The core support plate, shown in Fig. 5-14, was fabricated from a type 304 stainless-steel forging. It was produced by a commercial supplier using standard heat-treatment and machining procedures. The vertical penetrations were made within the required tolerances on a horizontal boring mill. The plate is 60 inches in outside diameter and $5\frac{3}{4}$ inches thick. The periphery of the plate is fitted with a 1-in. thick, 2-in. high, stainless steel ring to provide a sealing surface for a future forced circulation shroud. In addition to the 148 fuel location holes, $3\frac{5}{8}$ in. in diameter, the core support plate contains 9 cross-shaped control rod slots, $\frac{9}{16}$ in. wide by $10\frac{1}{2}$ in. across tips. The verticality of these penetrations is rigidly held to ensure adequate clearance at the top of the rod guide structure for the control rods and fuel assemblies. The fuel location holes contain two seating chambers, one at the top surface of the plate and the other $5\frac{1}{4}$ in. below this surface. These are spaced in such a manner that they impart a lifting action to the fuel assembly when the top is displaced sideways, an action which insures a good fit between the locating end fittings of the fuel assemblies and the location holes in the plate. At the same time it provides for more sidewise motion at the top of the fuel assemblies which may simplify the removal of damaged fuel assemblies, should this occur.

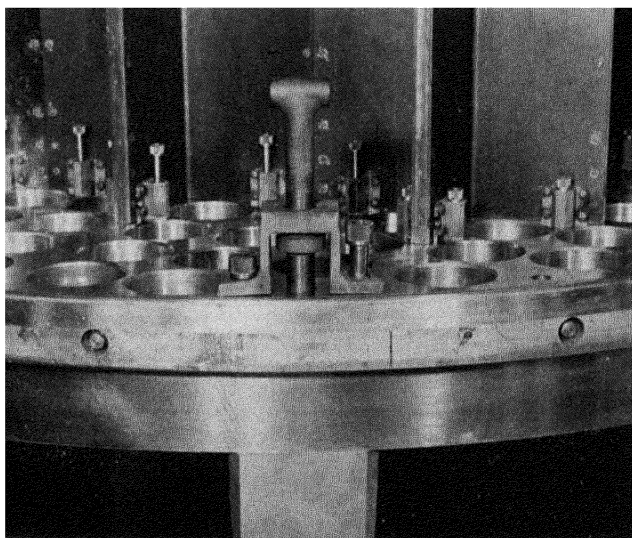


FIG. 5-15. Core hold-down assembly.

The core support plate also contains 40 tapped $\frac{7}{16}$ in. diameter holes for bolting the rod guide structures to the plate. Finally, 8 tapped holes, $\frac{1}{2}$ in. in diameter, are provided for bolting the forced circulation shroud to the support plate. The plate constitutes the main assembly point of the core structural components since the plate is fastened to the rod guide structure and supports this assembly as well as all of the fuel assemblies. It also serves to attach the entire core structure assembly to the lower head of the pressure vessel through the hold-down assembly.

Figure 5-15 shows the core hold-down assembly. It also was manufactured by a commercial supplier. This item performs a dual function in that it also serves as the attachment point for the core-structure lifting assembly. The hold-down assembly consists of 8 support legs, U-brackets, and hold-down T-bolts. The assembly is attached to the support plate, near its periphery, by 8 sets of holes which penetrate the plate. Each set provides a bolting point for one support leg, U-bracket and T-bolt. Two holes of the set allow bolts to be passed through the U-bracket wings, through the plate and into tapped holes located in the top surface of the support legs, thus joining all three parts securely together. The center holes of the set match similar holes drilled through the cross bars of the U-brackets and lengthwise through the support legs. This allows the T-bolts to pierce the entire assembly.

A thread is turned on the lower end of the T-bolts which mates with a tapped hole in the support leg seating pad welded to the lower head of the pressure vessel. The upper end of the shaft has a ring just below the cross bar of the T whose diameter is larger than the diameters of the penetration holes in the U-brackets and support legs. In this manner the vertical motion of the bolt is restricted between the top surface of the support plate and the underside of the restraining U-bracket. This design permits the T-bolt to perform in the dual capacity of a lifting bar, suitable for use in handling the support plate and as a hold-down bolt to secure the core assembly to the lower head of the reactor vessel. The first function is effected by the top shoulder of the ring bearing against the underside of the U-bracket when a lifting force is applied to the T-bolt. The second function is fulfilled when the lower shoulder is brought to bear against the support plate, as the T-bolt is threaded into the seating pad, clamping the entire core assembly firmly to the vessel.

The control rod guide and top shroud structure, shown in Fig. 5-13, by virtue of the uniqueness of its function and the materials of construction, presented a number of problems from a fabrication standpoint.

The wide top portion, which lies outside the active zone of the core, is type-304 stainless steel. The lower portion, 60 inches high, which supports the top portion, is in the active zone of the core. This is made of Zircaloy-2 because of its low neutron-absorption characteristic.

A structure built up of $\frac{1}{8}$ -in. thick flat plate angles was chosen because it seemed to provide the most economical solution to the problem of fuel support and control-rod guiding as shown in Fig. 5-16. Zircaloy-2, as well as stainless steel was commercially available in flat plate of sufficient width and length to produce the required shapes. Furthermore, it was felt that cold-press brake forming could yield these shapes with the necessary accuracy to meet all the tolerance requirements. After forming, simple machining operations could be used to trim the angles to final size. Suitable control-rod passages could be provided by attaching the legs of the plate angles to spacers of bar stock extending the full height of the core. Bar stock was also commercially available, in both materials, in sizes requiring minimum machining.

Fabrication of the stainless steel portions of the structures presented no particular problem. However, certain of the properties of Zircaloy-2 required some special handling during the manufacture of this portion of the structure. Specifically these properties were the lack of ductility at room temperature and the loss of corrosion resistance at high temperatures as a result of oxygen and nitrogen absorption. The lack of ductility forced the abandonment of the cold-forming operation in favor of bending at 1000 F, the temperature at which the required bend radius ($\frac{3}{16}$ in.)

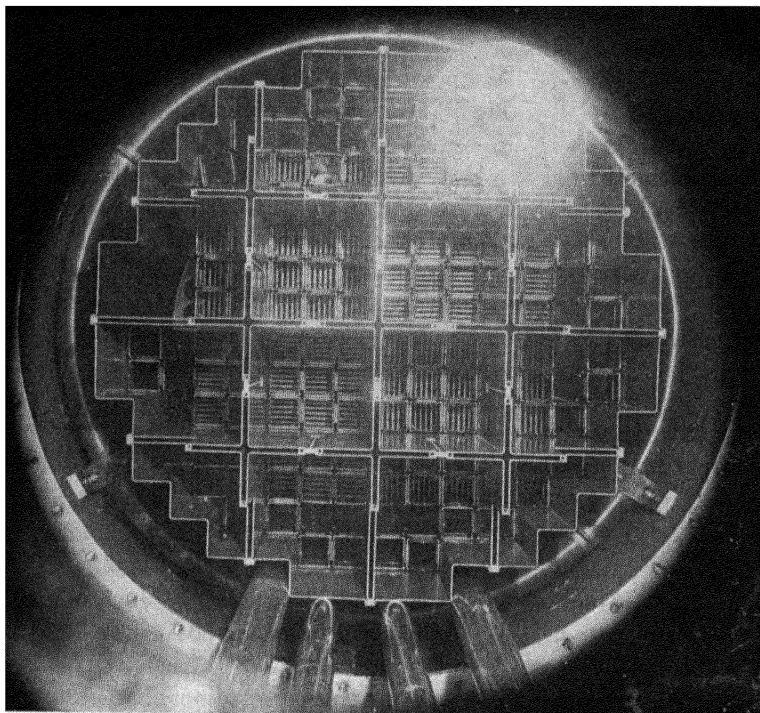


FIG. 5-16. View of flat plate core structure.

could be obtained without cracking at the bend line. The consequence of this heating, if done in air, is a loss of corrosion resistance caused by gas absorption occurring during the heating, forming, and cooling operations.

The solution to this composite problem was to make a sandwich of two sheets of $\frac{1}{64}$ -in. thick plain carbon steel containing between them the Zircaloy-2 sheet to be formed. The edges of the two steel sheets were then crimped along the full periphery, forming a loose envelope. A small tube was inserted between the sheets at one corner, and connected to a supply of argon gas. The flow of gas was initiated prior to heating (which was done in an electrically heated Inconel muffle furnace), and maintained through the full cycle of heating, forming, and cooling. After reaching room temperature, the steel sheets were stripped from the formed plate, which was then given a light pickle in dilute hydrofluoric-nitric acid solution. This procedure yielded plate angles of the required size, shape, and corrosion resistance at very little additional cost over normal stainless steel fabrication.

The use of angle shapes produces an easily assembled structure divided into 12-inch square cells, each of which has a capacity of 9 fuel assemblies. The cells are formed by riveting the mating legs of the angles to spacer bars located at the midpoints of each side of the cell. The opposing angles of the adjacent cells are riveted to the opposite faces of the spacer bars tying all of the cells together and forming the control rod channels. Riveting is used in lieu of welding to eliminate the post-assembly clean-up necessary to corrosion resistant Zircaloy-2. The outer angles are also equipped with stainless steel lugs along their lower edges to provide a means of bolting the entire rod guide and top shroud structure to the support plate.

The top shroud which is 34 in. high encircles the upper portion of the rod guide structure, beginning at a point 60 in. from the bottom of the structure. The shroud is press brake formed from type-304 stainless steel, $\frac{1}{8}$ in. thick, and attached to the structure by riveting the sheet to wing extensions of one leg of certain of the outer rod guide angles. The shroud surrounds the tops of the outer ring of fuel assemblies and prevents the downcomer flow from entering these assemblies. It also provides flow channeling to prevent steam and water mixtures from entering the downcomer too soon after leaving the fuel assemblies. This could result in excessive steam entrainment in the downcomer water and a possible lowering of core inlet water density. Some "chimney effect" benefit to natural circulation is expected.

To reduce leakage during forced circulation operation (a possible 30 per cent of available pump capacity), the control rod channels are provided with caps. These caps are formed by welding $\frac{1}{8}$ in. thick stainless steel sheet angles to spacer bar extensions. The caps are bolted to the guide structure by long-shank bolts which pass through vertical holes in the spacer bar extensions and into tapped holes in the top ends of the guide structure spacer bars.

The core shroud and support plate are inserted and removed from the reactor vessel with the lifting ring and tubes shown in Fig. 5-17. Space limitations between the pressure vessel wall and the shroud structure necessitate assembling the shroud to the grid plate outside the reactor. The whole assembly must be lowered into place as one piece and attached to the machined pads in the bottom head by eight stainless steel captive T-bolts, 1 in. in diameter.

Each of the eight T-bolts is turned by means of a 21-ft long bayonet-type extension wrench. The T-bolts have pilot pins extending upward from the T to act as guides for locating the wrench and lifting rods. The lifting rods are then lowered individually by locked-on extension rods and rotated on the T-bolts to the lifting position. The spider ring is

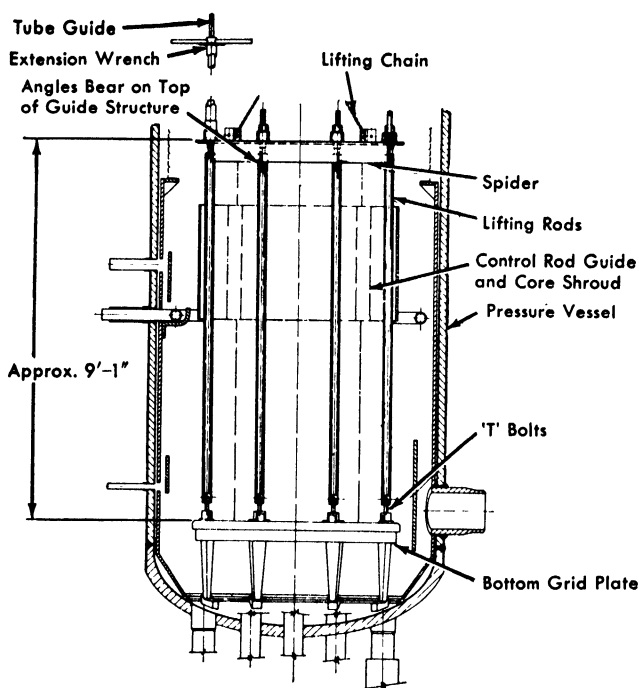


FIG. 5-17. EBWR shroud and grid plate lifting ring.

lowered into place over the rods until the captive nuts rest against the tops of the lifting rod threads. The long hollow tube extension hex-nut wrench is lowered, in turn, over each extension rod until the captive nuts are engaged. The extension rod is held to prevent turning of the lifting rod and T-bolt, and the nut is tightened.

With the nuts tight, the spider ring angles bear on the top surface of the shroud which is placed in compression, and the lifting rods in tension to provide a rigid unit for handling. The extension rods are detached from the lifting rods and removed, after which the unit is lifted from the vessel by means of the overhead bridge crane and placed in the storage pit. Insertion of the initial or a replacement assembly follows a reverse procedure. All personnel except those required to operate the crane from behind the shield are vacated from the room when an active assembly is removed from the vessel.

3. *Control rods and drive mechanisms.* Two types of material are used in the EBWR control rods. Five units of the 9 are fabricated from hafnium and Zircaloy-2 as shown in Fig. 5-18, and 4 units are made from 2 per cent boron-type 304 stainless steel and Zircaloy-2 as shown in Fig. 5-19.

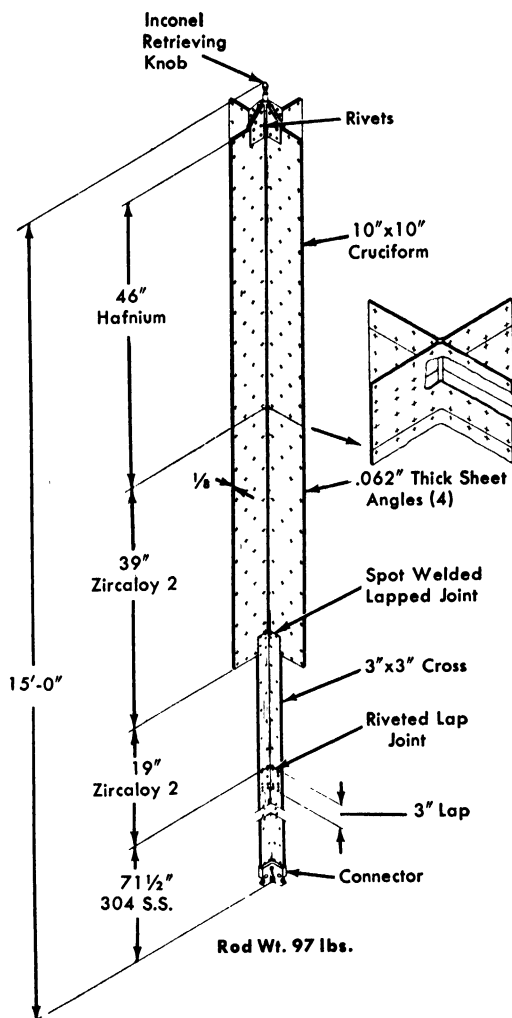


FIG. 5-18. Hafnium-Zircaloy 2 control rod.

Sheet metal spot-welded construction has been utilized as much as possible in both types of rods in keeping with the general philosophy of core construction. The hafnium-Zircaloy-2 control rods are made up from $\frac{1}{8}$ inch thick sheet metal bent into angles and spot welded together to form a cross-shaped section, 10 by 10 by $\frac{1}{8}$ -in. thick. The end fittings are machined from stainless steel and Inconel weldments.

Due to the brittle nature of the Zircaloy-2 sheet it must be enclosed in a sheet metal jacket for hot forming. The jacket must be evacuated

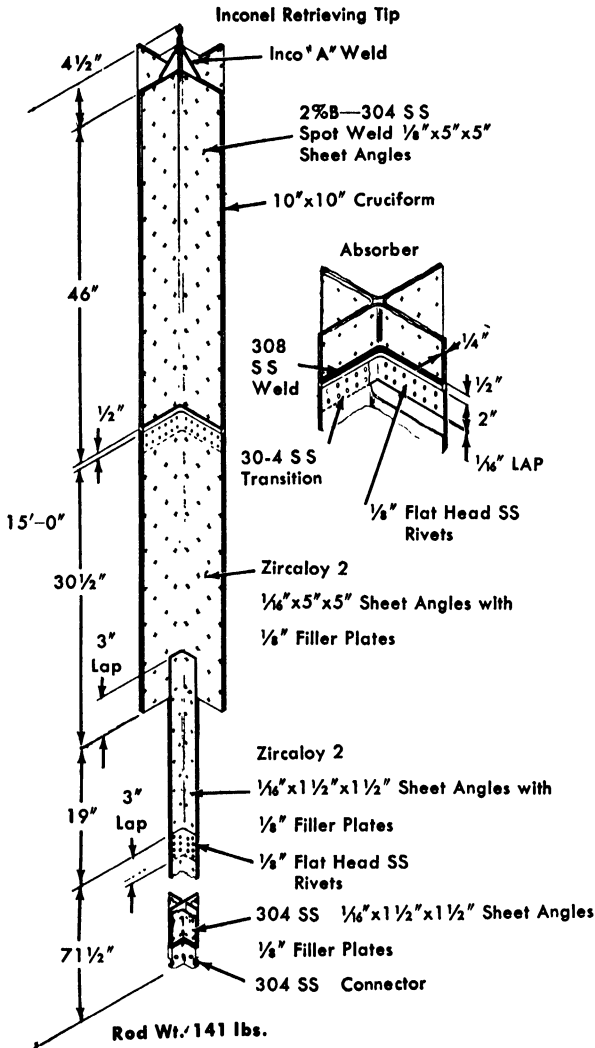


FIG. 5-19. Boron-stainless steel control rod.

and flushed with argon gas before heating to approximately 1000 F for bending. The formed sheet metal angles are then grit blasted before being spot-welded together. Although the lap-type joint employed to join the hafnium absorber section to the Zircaloy-2 follower section results in only half thickness absorber for 3 in., it is not considered to be detrimental to performance. This type construction effects considerable economy over a butt weld since the latter requires very careful preparation.

It has been necessary to can the hafnium sheet for forming in exactly the same procedure as required for the Zircaloy. The reason for this action is discussed later under the subject of costs.

Performance tests conducted on a hafnium-Zircaloy-2 control rod for a period of 6 months at 486 F in the control rod drive test facility revealed no significant defects in design or material. The Zircaloy-2 follower section developed a slight buckle near the bottom end which resulted in a slight amount of abrasion on the Zircaloy-2 rod and guide section. The wear amounted to approximately 0.008 in. on the control rod surface over a period of 5000 cycles with a scram at each cycle. There was no evidence of fretting corrosion or increased drag on the control rod drive. The stainless steel rivet heads, which join the lower stainless steel section of the control rod to the Zircaloy-2 portion, showed some superficial corrosion which is attributed to fabrication technique.

The calculated effectiveness and burnup of 1, 2, and 3 w/o boron-stainless steel and hafnium control rods are listed in Table 5-5. The burnup is based upon the assumption that the absorber material is in an average core flux of 1.5×10^{13} . In normal operation, the control rods probably will be pushed almost all the way out of the core.

It appears that a 2 per cent boron-stainless steel rod, one-fourth inch thick, will have somewhat greater blackness than a thermally black control rod and is nearly as effective as a hafnium rod one-eighth inch thick.

TABLE 5-5
CALCULATED BURNUP AND EFFECTIVENESS OF HAFNIUM AND BORON-
STAINLESS STEEL CONTROL RODS
(Average reactor flux: 1.5×10^{13})

Rod composition	Thickness, inches	Effectiveness, %*		Per cent burnup in 1 year
		Initial	After 1 year burnup	
1 per cent B-SS	$\frac{1}{4}$	88	80	28.5
2 per cent B-SS†	$\frac{1}{4}$	109	104	14.5
3 per cent B-SS	$\frac{1}{4}$	117	114	9.8
Hafnium†	$\frac{1}{8}$	108	108	negligible

* Values are referred to a black thermal rod. Values >100 per cent are due to absorption of epithermal flux.

† Material used in EBWR control rods.

The 2 per cent boron-type 304 stainless steel control rods are fabricated from $\frac{1}{8}$ -in. thick sheet metal in the absorber section only. A welded type-304 stainless steel transition piece connects the absorber and follower sections. Type-304 is used because there is slight tendency for 2 per cent boron-stainless steel to become increasingly notch sensitive under irradiation, and this portion of the rod is exposed to severe neutron irradiation. The follower is made up of $\frac{1}{16}$ -in. thick Zircaloy-2 angles with $\frac{1}{8}$ in. Zircaloy-2 filler plates to equal the total thickness ($\frac{1}{4}$ -in.) of the absorber section. End fittings are similar to those used on the hafnium rod. Spot-welded assembly has proved feasible and economical.

Since the neutron irradiation effects on 2 per cent boron-stainless steel are not fully established, some concern has been expressed about the use of this material for control rods. Hafnium rods can be substituted, if necessary; however, during normal operation the rods are completely withdrawn; therefore, they will not be subjected to severe fluxes.

A cost comparison between hafnium and 2 per cent boron-stainless steel control rods, each with Zircaloy-2 followers, is given in Table 5-6. The figures are based on procurement of materials and fabrication by Argonne.

The hafnium-Zircaloy control rod suffers costwise because of the low yield (52 per cent) of finished hafnium sheet from the iodide crystal bar. The sheet metal design employed results in a fabrication scrap loss of less than 1 per cent. The low yield is due strictly to the ineptness of the commercial vendor in melting and rolling.

A quantity of 454 lb of hafnium crystal bar was supplied to the processor for converting to cold finished sheets, 10×0.062 in. A total of 233 lb, in 18 usable sheets, was returned. Of this amount, only 5 pieces were of the specified size. The remainder varied in length from 44 to 36 in.; therefore it was necessary to piece together odd lengths to achieve final dimensions. In addition the material was embrittled by contamination; therefore, it was necessary to jacket each sheet for hot forming in an argon atmosphere. Fortunately, contaminated hafnium still has satisfactory corrosion resistant properties, and jacketing of the hafnium was not a problem, aside from the added cost, since this technique is required for hot bending Zircaloy-2.

The excessive hardness and brittleness of this lot of hafnium is best illustrated by the Rockwell B hardness reading which was 102, as compared to a Rockwell B hardness of 85 to 88 for normal purity material. Cold bends of less than 15 degrees resulted in cracking, whereas good quality hafnium may easily be bent 90 degrees.

It is indeed ironic that 77 per cent of the cost of the boron-stainless steel control rod is in Zircaloy-2, which is required only for the purpose

TABLE 5-6
COST COMPARISON BETWEEN HAFNIUM AND 2 PER CENT BORON-
STAINLESS STEEL CONTROL RODS

Hafnium-Zircaloy-2 rod (thickness: $\frac{1}{8}$ in.; weight: 97 lb):			
Material:			
Hafnium (4 sheets: $\frac{1}{16} \times 10 \times 46$ in.)	52.4 lb @	\$154.00	\$8,050.00
Zircaloy-2	26.4 lb @	33.00	870.00
Type-304 SS	19 lb @	1.00	19.00
Inconel	3 lb @	1.30	3.90
Total materials cost			\$8,942.90
Labor:			
Zircaloy-2 canning and forming	32 hr @	\$5.50	\$176.00
Hafnium canning and forming	20 hr @	5.50	110.00
End fittings	40 hr @	5.50	220.00
Final assembly	8 hr @	5.50	44.00
Total labor cost			550.00
Total cost			\$9,492.90
2 per cent boron-stainless steel-Zircaloy-2 rod (thickness: $\frac{1}{4}$ in.; weight: 141 lb):			
Material:			
2 per cent boron-type 304 (4 sheets: $\frac{1}{8} \times 10 \times 46$ in.)	66 lb @	\$ 3.40	\$ 224.00
Zircaloy-2	52 lb @	33.00	1,716.00
Type-304 SS	38 lb @	1.00	38.00
Inconel	3 lb @	1.30	3.90
Total materials cost			\$1,981.90
Labor:			
Zircaloy-2 canning and forming	32 hr @	\$ 5.50	\$176.00
2 per cent boron-type 304 forming	1 hr @	5.50	5.50
End fittings	40 hr @	5.50	220.00
Final assembly	10 hr @	5.50	55.00
Total labor cost			456.50
Total cost			\$2,438.40

of filling the water hole of the guide with a low cross-section material when the absorber section is withdrawn.

Control rod drive mechanism. The control rod drive is designed to meet the following requirements:

- | | |
|--|--|
| 1. Operating reactor pressure | 600 psig |
| 2. Operating temperature of seal | 110 F |
| 3. Control rod spacing | 12 $\frac{3}{4}$ by 12 $\frac{3}{4}$ in. |
| 4. Number of control rods | 9 |
| 5. Control-rod travel | 48 in. |
| 6. Control-rod withdrawal rate | (max) 28 in/min |
| 7. Scram speed at 200 psi | 42 in. in 0.5 sec |
| 8. Rod speed in reactivity: | |
| Min | $1 \times 10^{-5} k$ per sec |
| Max | $4 \times 10^{-5} k$ per sec |
| 9. Positionability | ± 0.025 in. |
| 10. Deceleration distance | 6 in. |
| 11. Deceleration force (constant) | 1700 lb |
| 12. Maximum leakage rate per rod | 750 cc/min |
| 13. Complete "fail-safe" characteristics | |
| 14. Removal of each control rod and drive unit from reactor without draining reactor vessel. | |

Location of the control rod drives below the reactor core offers a number of advantages but there are also some disadvantages. The advantages may be listed as follows:

1. The shielding problem over the top flange is simplified and entrance into the vessel is simplified.
2. The top surface of the core structure is free of obstructions during removal and replacement of fuel elements.
3. The control rod drive extension is always flooded and conveniently cooled by the addition of a small amount of feed water to replace the seal leakage loss.
4. The control rod has both gravity and reactor pressure differential across seal, aiding the scram for a normal fail safe condition.

To balance against these advantages, the following disadvantages must be considered:

1. Additional building depth is required to accommodate the control rod drive room.
2. Shielding must be provided between the bottom head of the reactor and the control rod drives.

3. There is a natural tendency for corrosion products to settle into the control rod thimbles and on into the control rod drives.
4. The absorber section of the control rod is driven up out of the core during operation; hence a low neutron absorption material (Zircaloy-2) must be used for a follower section to fill the "water hole" of the guide channel. Where boron-stainless steel control rods are used, the Zircaloy-2 follower section accounts for 75 per cent of the total cost.
5. In the case where forced circulation is used, "caps" must be provided over the control rod channel tops to prevent the bypass of some 25 per cent of the normal circulation flow.

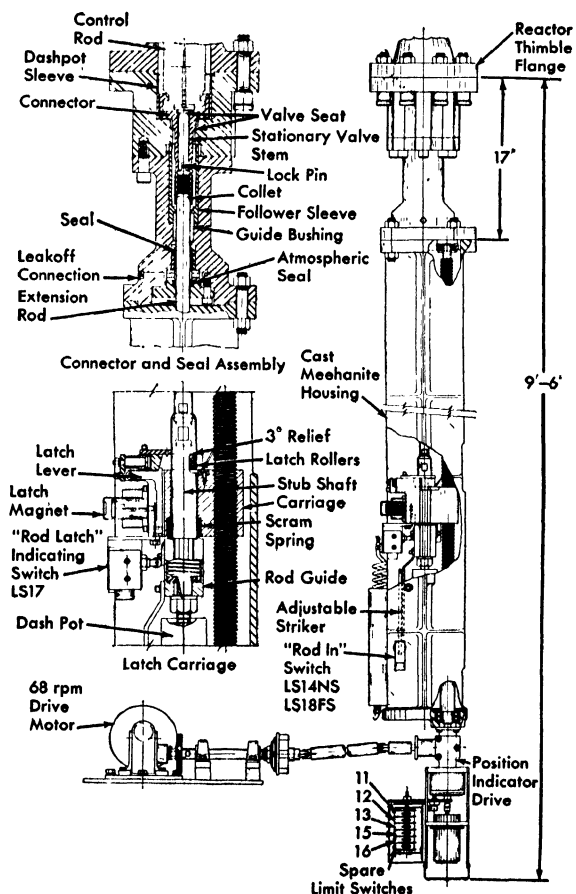


FIG. 5-20. Control rod drive mechanism.

The main components of the drive unit as shown in Fig. 5-20 are the connector and seal assembly, main housing and roller latch carriage with lead screw and extension rod, position indicator assembly, and the gear motor and shaft. The unit is a reciprocating type with all operating parts external to the reactor vessel. The control rod locks into the connector by means of a cross-shaped bayonet type joint with the load being taken on a cross slot in the top of the stationary valve stem. The valve faces on the connector and the stationary valve stem are inlaid Haynes Stellite No. 6, ground and lapped into their respective seats. This arrangement provides the shutoff valve for the reactor when the mechanism is removed for maintenance.

Because of sludge in the guide bushing, the control-rod drives have been removed from the reactor several times for servicing since the initial start-up in December 1956. The shutoff valve feature of the connector has proved that the drives may be serviced without draining the reactor vessel. In each instance the water that leaked out while the drives were removed (approximately 1 hr per drive) was less than 10 gal.

An additional function of the connector is that it serves as the piston for the internal dashpot.

The seal assembly consists of a nylon guide bushing and a five-element labyrinth type pressure-breakdown seal. The nylon guide bushing ensures the alignment of the extension rod through the seal elements. The seal has 5 stationary rings and 5 floating rings. Leakage is collected in the lantern ring and returned to the reactor vapor-recovery system. A garter type shaft seal assures that no leakage is lost to the atmosphere.

Originally a Stellite No. 3 guide bushing was used with the seal assembly, but, in the full-scale test operation, it was found that scoring of the extension rod and floating seal rings occurred due to chrome flakes being scraped from the extension rod. The substitution of the nylon guide bushing reduced that problem to a minimum. However, during reactor operation it was found that the diametral clearance between the extension rod and the nylon guide bushing had to be increased from 0.001 to 0.007 in. A considerable amount of sludge (primarily iron oxide) was trapped in the small-clearance annulus, and the resulting wedging action caused the rods to scram unsatisfactorily when the reactor was at atmospheric pressure.

The substitution of a nylon Type FM-10001 guide bushing proved a satisfactory solution to this problem. In addition, the nylon acts as a very good wiper to trap oxide particles which fall from above. Radiation damage of the nylon poses no serious problem inasmuch as there is approximately 6 ft of water and 3 ft of steel shielding between the bushing and the core. However, in order to determine how much irradiation the nylon could stand, a bushing was exposed for 48 hr to a thermal flux of 10^{12}

n/(cm²) (sec) in the Argonne Research Reactor (CP-5). A slight change in color and dimension was noted but the physical properties were unaffected. Subsequent exposure to a hard gamma irradiation dosage of 2.5×10^8 roentgens resulted in a measured growth of 0.0015 in/in on the inside and outside diameters and length. In addition, the surface had a slight tackiness which disappeared in 5 days. The bushing was then placed in a seal housing and functioned in the same manner as an unirradiated bushing. A pessimistic calculation of the amount of radiation likely to strike the bushing under reactor operation indicates a minimum lifetime of 10 years for the nylon. The normal clearance between the bushing and the extension rod may vary between 0.0015 and 0.0035 in. Internal pressure-equalizing grooves machined into the bore of the bushing, act as debris catchers to the benefit of the seal and rod.

Seal leakage is replaced by the introduction of 110 F water, taken from the feed system through a connection on each thimble just above the reactor flange. Enough water is introduced to allow approximately 1000 cc/min to flow upward into the reactor to keep the thimble at a lower temperature.

The extension rod (1.000 ± 0.0001 in. diameter $\times$ 60 in. long), operating through the seal, is the connecting link between the control rod and the drive mechanism. The top end is rigidly connected to the connector (Figs. 5-18 and 5-19). The bottom is notched and attaches to the carriage by means of a roller latch. The roller latch, or scram latch, is a solenoid operated device used to rapidly disconnect the extension rod from the carriage and allow the control rod to drop (scram). Reactor pressure is the primary control-rod expellant, but a spring attached to the extension rod overcomes the inertia at low pressures. The spring is most effective in accelerating the scram at pressures under 300 psi; maximum rod velocity is achieved in 4 in. A typical scram time curve is shown in Fig. 5-21. The average scram time for the 9 operating drives is 0.35 sec for 42 in. of travel at 600 psig.

The bronze latch carriage is a platform for the roller latch, the latch magnet, and the latch-indicating switch. It is driven by a 1¼-in. Acme thread tool-steel lead screw, and moves up and down between grooves machined into the side walls of the cast Meehanite housing. The lead screw is, in turn, coupled by means of a bevel-gear arrangement to the drive shaft and gear motor. These gear motors are ⅓-hp, 440-volt, 3-phase, 60-cycle, 68 rpm U. S. Electric gear motors with electric brake. The brake provides the operator with excellent "no coast" control and permits very accurate positioning of the control rods.

Position indication is achieved through a synchrotransmitter and receiver, with the transmitter being driven by the lead screw through a

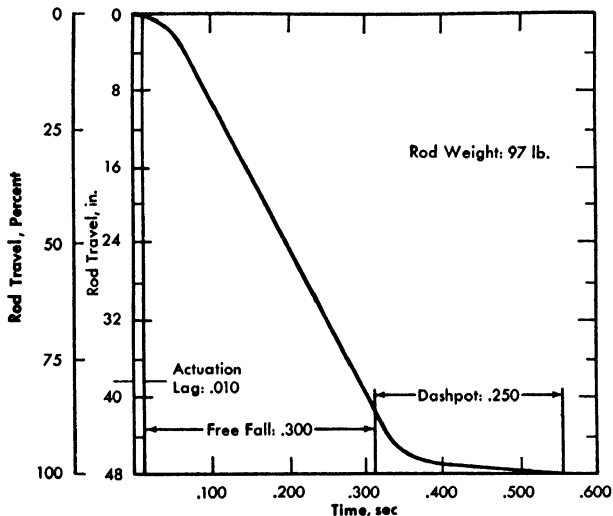


FIG. 5-21. EBWR control rod scram time.

50:1 high precision reduction gear box. An adjustable cam-operated microswitch assembly is also driven off the 50:1 gear box. These switches control pilot lights and act as limits for rod sequence positioning. The original micro-limit-switch system was not too successful, primarily because of its inaccessibility as installed. Part of the steel framing blocked the operator when trying to adjust the movable cams; therefore the lower limit was repositioned to be actuated directly by the carriage, and the others have been disconnected.

The external dash pot, shown in Fig. 5-22, is located in the bottom of the housing. It consists of a square carbon steel body with a bronze perforated hollow piston, a return spring, and an O-ring sealed top retainer. SAE 20 grade hydraulic oil is used for the buffing fluid. The dash pot is designed to decelerate the scrammed control rod over a 6-inch stroke at a constant force of 1700 lb. The latter value is the allowable column load for a suddenly applied loan on a 1-in. diameter shaft of 60 in. unsupported length, with a safety factor of 3 at an impact velocity of 13 fps. The piston of the dash pot is hollow with properly spaced orifice holes drilled through the wall to achieve the desired results. Although the dash pot as designed has never failed to work satisfactorily during performance tests, a failure of the piston to return to the up position would have serious consequences. Therefore, the internal dash pot above the seal has been included. The internal dash pot may have a shorter deceleration length since no column load is imposed upon the extension rod.

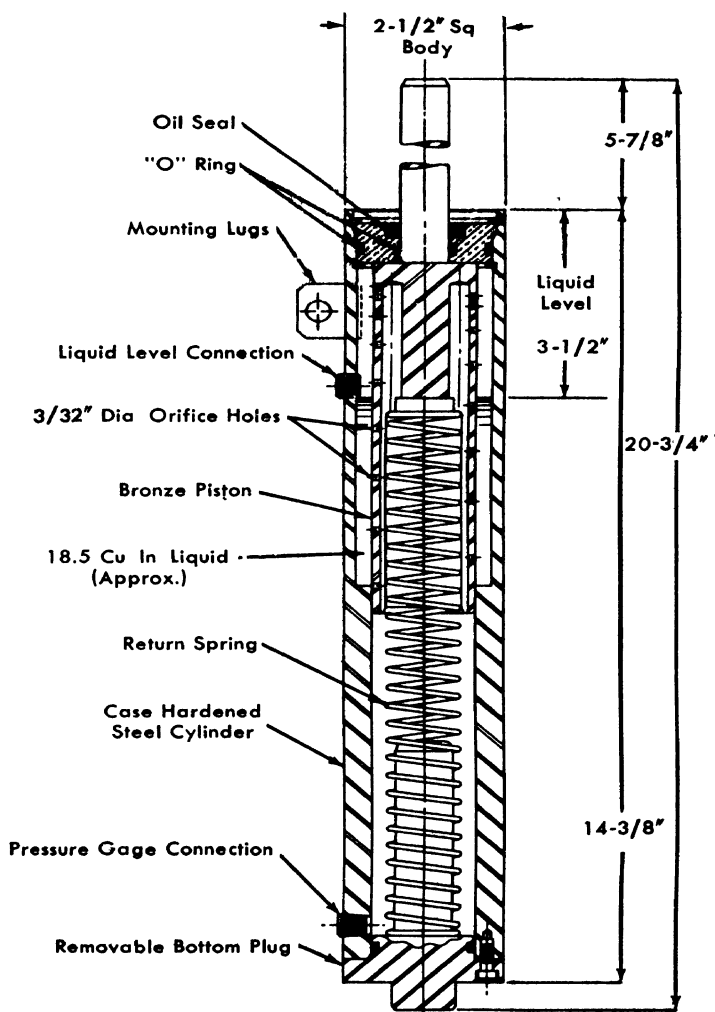


FIG. 5-22. EBWR external dashpot.

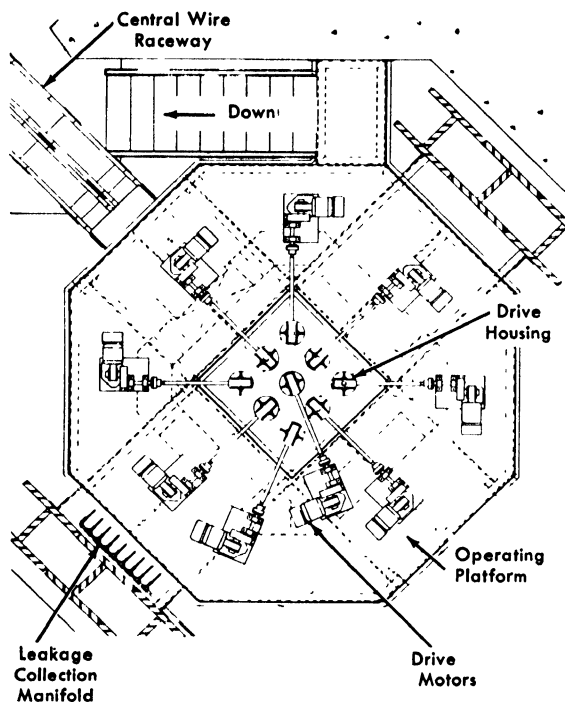


FIG. 5-23. Orientation plan of control rods.

Location of the drive motors with reference to the control rods is shown in Fig. 5-23.

A cost summary for the 9 control rod drives and all related equipment, such as motor drive, position indicator, spare parts, switches, and wiring to the main junction box is given as follows:

Labor (9964 hr at \$5.50)	\$54,802.00
Raw material	10,932.99
Purchased parts	8,189.59
Total for 9 drives	\$73,924.58
Average cost of each drive, complete	\$ 8,213.84

4. *Pressure vessel.*⁹ The reactor pressure vessel, illustrated in Fig. 5-24, incorporates provisions for the study of natural and forced circulation and for both light and heavy water. Conventional steels, SA212 Gr B plate, and SA105 Gr II forgings were clad with either SS304 sheet or weld metal deposited from SS308 L electrodes. The design meets the general require-

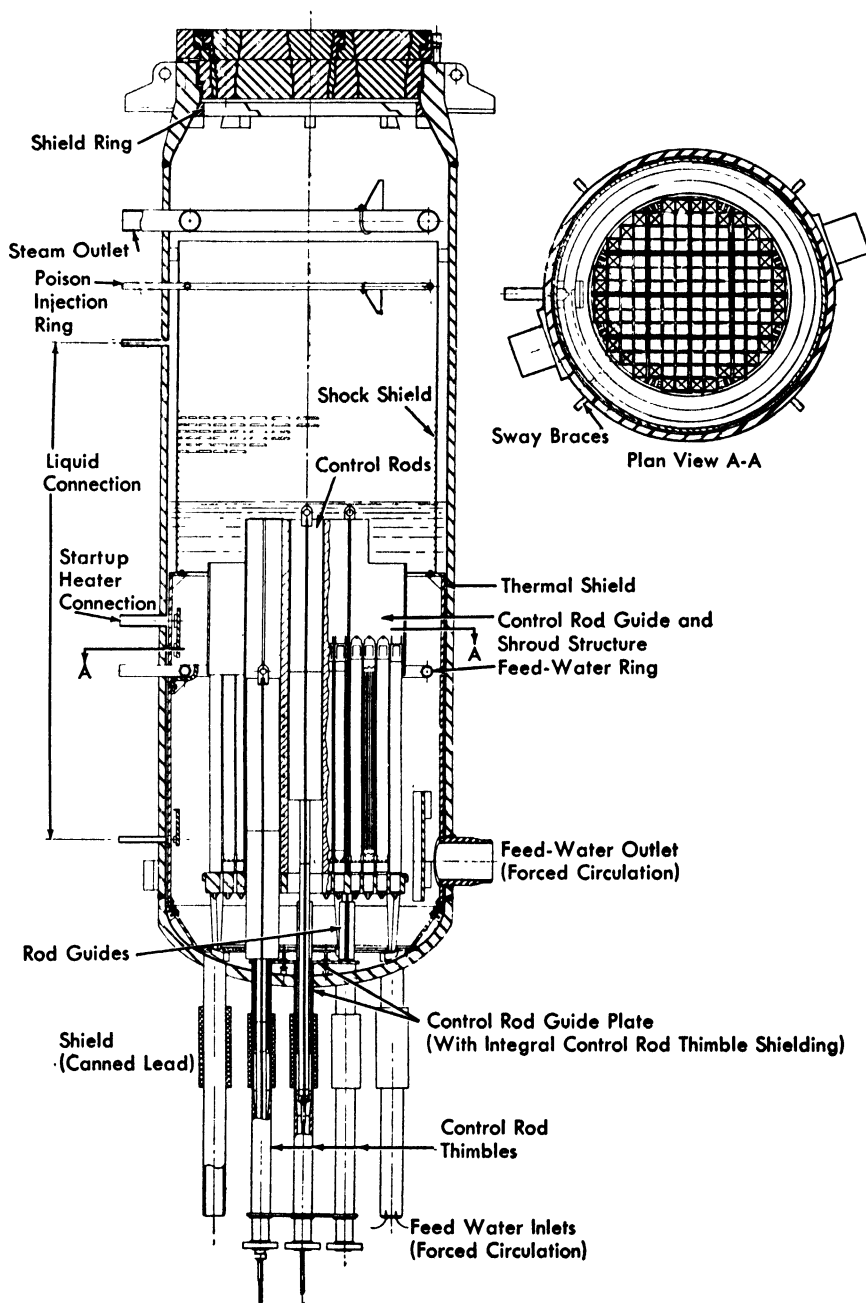


FIG. 5-24. EBWR pressure vessel.

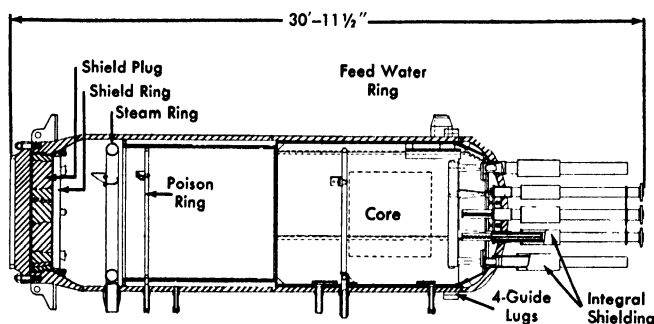


FIG. 5-25. Structure of EBWR pressure vessel.

ments of Section I, Power Boilers, of the ASME Boiler and Pressure Vessel Code. All nozzles permitting the introduction of cold fluids into the interior of the vessel were equipped with thermal sleeves. All pressure and nozzle welds were radiographed, and the helium leak test established their final soundness. The vessel is insulated with commercial, fine grade stainless steel type-430 wool packed to a density of 5 lb/ft.³

As shown in Fig. 5-26, the vessel consists of an upper ring forging, a 2-course rolled plate shell and spun plate lower head. A flat cover plate is held against the 2 sealing gaskets by 44 alloy bolt studs screwed into the ring forging. The nominal inside diameter is 84 in.; the length is 30 ft 11½ in. and the dry weight, including removable internals and cover plate, is approximately 60 tons.

The vessel is designed for the following conditions:

Maximum design pressure	800 psig
Operating pressure	600 psig
Maximum design metal temperature	650 F
Operating metal temperature	500 F*
Minimum weld joint efficiency	95 per cent
Radiography (includes all pressure strength welds)	100 per cent
Minimum shop hydrostatic test pressure	1200 psi
Minimum shop helium test pressure	800 psi
Maximum permissible helium leakage at 800 psi (closure flange excepted)	1 3 cc/hr
Minimum field hydrostatic test pressure	1200 psi (after erection)
ASME Code Inspection and Stamping	Required

* Nuclear heat generation in the vessel wall raises the metal temperature above the saturation temperature (489 F) of steam at 600 psig.

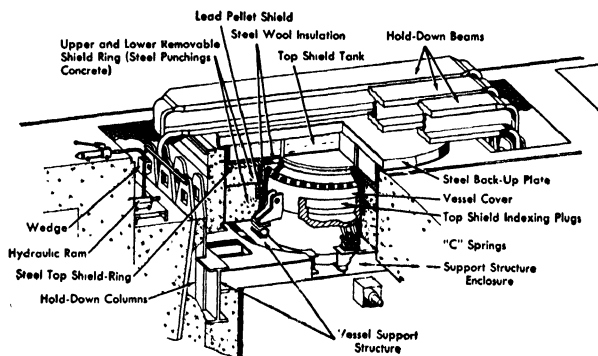


FIG. 5-26. Upper vessel structure and shielding arrangement.

The structural features of the pressure vessel are more clearly shown in Fig. 5-25. Eight support brackets, paired symmetrically and radially about the principal axes of the vessel, are welded to the ring forging for mounting the vessel. Vertical sway braces (9 in. long by $4\frac{1}{2}$ in. wide and $1\frac{1}{2}$ in. thick) are welded to the shell just above the circumferential weld between the shell and the lower head. These are placed symmetrically around the circumference and are oriented 30 degrees clockwise from the principal vessel axes. Their purpose is to prevent "rocking" of the vessel in operation. Radial restraint is effected by a parallel plate yoke mechanism welded to and extending inward from the lead shield form. Vertical travel is unimpeded to permit thermal expansion of the vessel.

The maximum displacement provided for is 3 in. At maximum displacement, radiation shielding around the control rod thimbles will "bottom" on the steps of the heavy concrete biological shielding beneath the pressure vessel. The advent of loadings to produce a 3-in. deflection would result in major structural damage to the vessel and support structure.

Thermocouple blocks, containing 2 chromel-alumel thermocouples per block, were added to the exterior of the lower head, shell and ring forging at approximately 2-ft intervals. The thermocouples are designed to indicate the extent and distribution of the internal heat generation in the steel wall, and the startup heating rates.

Inside the vessel, the top shield plugs are supported from the ledge of the ring forging. Continuous contact of the plug system with the ledge is prevented by 1-in. square bars welded to the radiation blocking step of the plug to provide a controlled port for the admission of steam to the upper portion of the ring forging, cover plate, and the plugs. The shield plug assembly is utilized for refueling and other operations with heavy water reactor modifications which preclude the use of water for shielding.

In addition to nozzle attachments and distribution rings, the shell of the reactor vessel houses (a) an upper segmented and bolted 4-piece shock shield which is bolted to the upper ring of the thermal shield. The primary function of this shock shield is to prevent accidental chilling of the hot steel walls by the admission of cold water through the poison line distribution ring. (b) A lower segmented, and bolted, 10-piece thermal shield supported on lugs welded to the lower head. Thermal shielding is not required in the light-water version since thermal stresses are at very low levels. The more open heavy water cores and the much higher neutron and gamma fluxes require an inexpensive neutron absorber and steel, both of which are combined in the boron-stainless alloy. (c) A type-304 stainless steel, circular grid or core support plate (60-inch diameter by $5\frac{3}{4}$ in. thick). The grid plate is bolted to 8 legs which, in turn, are secured to the lower head as described in the section under Core Structure.

In the lower head a stainless steel control rod guide plate (1-in. thick) is bolted to threaded blocks welded to the carbon steel head. Like the grid plate, the control and guide plate is held down by captive T-bolts to facilitate future remote removal. The plate bears on a common machined plane, including all 9 control rod thimbles.

The stainless steel caps of the forced circulation flow distributors are surmounted with 1-in. thick caps of stainless steel containing 1 per cent boron by weight to reduce the thermal neutron streaming into the control room.

Eight blocks of stainless steel are welded to the carbon steel shell for the attachment of the grid support legs. A continuous stainless angle, formed by machining, is welded to the lower head to form a seating ring for a future 60 in. OD plenum chamber for forced circulation operation and encompasses all the internal connections to the lower head.

Ten upper anchoring bars and a like number of guide blocks are welded to the knuckle section of the lower head for attachment of the 5-piece conical segmented thermal shield to the lower head. This shield is fabricated of rolled stainless steel plate containing 1 per cent boron by weight.

The closure consists of a flat cover plate (6 ft $8\frac{7}{8}$ in. OD by $8\frac{13}{16}$ in. thick) bolted to the forged cylindro-conical ring section of the pressure vessel. Two concentric gaskets, each with a $\frac{5}{8}$ -inch face width, are used to maintain a pressure tight joint. Inleakage of air (and water vapor) across the face of the inner gasket (which may develop in service conditions) is collected in an annulus between the two gaskets. Gas from an external source may be admitted to the inter-gasket leakoff annulus to verify the soundness of each of the gaskets. The gaskets are compressed by 44 equally spaced bolts nominally $2\frac{1}{2}$ in. in diameter, on a 6 ft $3\frac{1}{2}$ in. bolt circle diameter.

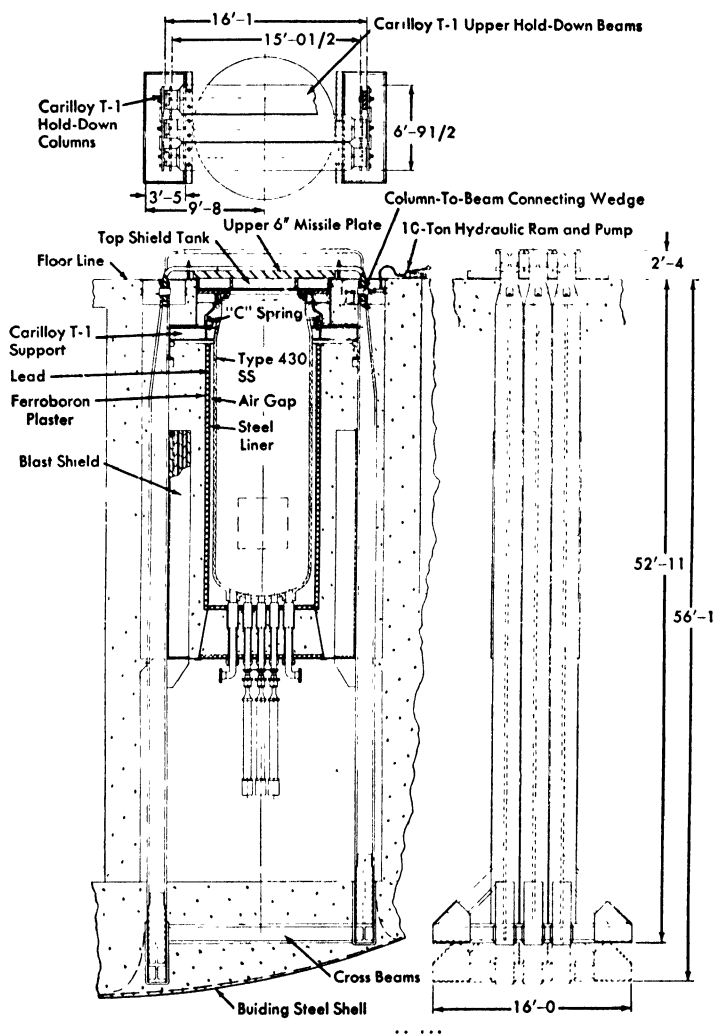


FIG. 5-27. EBWR pressure vessel hold-down structure.

The cover plate is fabricated from a carbon steel forging, the steam side being clad with stainless steel weld metal deposited by the submerged arc process for corrosion resistance and is $\frac{1}{16}$ in. in thickness after machining. A machined flat surface 72½ in. OD by 68 in. OD by $\frac{1}{8}$ in. in thickness, also deposited by the submerged arc process, constitutes the male flange facing. The upper periphery of the cover plate was machined from the $8\frac{13}{16}$ in. thickness to $4\frac{13}{32}$ in. to form an unobstructed plane

for the installation of the neutron shield tank-missile plate system interposed between the cover plate and the yoke hold-down beams.

Like the cover plate, the ring forging is clad (after machining) with $\frac{3}{16}$ in. thick stainless steel weld metal deposit for corrosion resistance.

The gaskets are stainless steel type-304, spiral wound asbestos-filled.

5. *Support, hold-down and shielding structure.* The vertical location of the support structure with respect to the EBWR building floor is dictated by the water level requirements of the turbine and condenser. The top of the pressure vessel closure flange was fixed at 24 in. below the main floor level. This establishes the criterion for a number of possible impact loadings on the vessel and its support structure.^{10 12}

Since the power plant building shell is most vulnerable to missiles launched in a vertical direction, the first concern was to hold down the top portion of the pressure vessel. This was accomplished by placing three parallel horizontal hold-down beams across the top of the pressure vessel (Fig. 5-27). A biological shield was also required above the pressure vessel. A convenient design was evolved using a steel ring and a structural steel backup plate to transmit the upward force from the pressure vessel cover through the ring and the plate to the bottom flanges of the hold-down beams. The backup plate serves as a biological shield and missile shield and distributes the upward force over a wide area on each of the three hold-down beams. The space below the plate within and around the steel ring is enclosed with lighter structural steel plate and filled to a depth of $11\frac{1}{2}$ in. with water to provide a hydrogenous biological shield.

In order to anchor the ends of the horizontal hold-down beams, the two sets of three parallel columns extend downward into the floor of the control rod room below the reactor. The concrete floor is not affected by the shock waves from a possible explosion. The shock wave energy would be dissipated by passage through the various mediums from the reactor core and the air in the control rod room.

To develop the required anchoring force, the columns are buried as deep as possible in the concrete control rod room floor. Reinforcing rods are welded to the anchor beams so that failure of the concrete would be in shear along planes 45 degrees from vertical beginning at the edges of the lower anchor beam flanges. Failure strength in these planes exceeds that of the six columns.

From the earliest calculations it was apparent that the conventional structural steel design was inappropriate. The calculated section sizes (with the conventional safety factor of 3) were extremely heavy and bulky. Accordingly, conventional design criteria were abandoned, and the use of high strength quenched and tempered alloy steel (U. S. Steel Corp.

Carilloy T-1) was adopted. The sections required are much larger than any conventional rolled sections listed and they were fabricated from plate.

The explosion mechanics for a core-water reaction (considered as violent as TNT) were calculated by the Armour Research Foundation. The results indicated that unprotected columns could not withstand the outward shock wave pressures. After a series of high-pressure crushing tests on various materials a suitable blast shield was adopted. This blast shield consists of sandwich layers of redwood and steel, and celotex and steel, placed against the inside face of each set of three columns in a region where the shock wave intensity is most damaging. The crushing of the redwood and celotex will absorb the shock energy and reduce the bending load on the columns.

Shielding. The governing shielding calculations are for the future case where D_2O is used as a reflector and the reactor is operating at a power level of 40 Mw.¹⁰

The approximate composition of the shield as actually constructed on the core centerlines is given in Table 5-7.

TABLE 5-7
SHIELD THICKNESS ON CORE CENTERLINES
(In inches)

	Radial	Axial (up)	Axial (down)
DO ₂	12 (min)	24	12
Iron		1*	4†
D ₂ O		29	25
Steam		126	...
Steel (1 per cent B)	1		...
Steel	2½	21½	4
Insulation and air	6	3	6
Steel	¾		...
Lead	3		3
Ferrobaboron plaster	½		1
Heavy concrete	61¼		48
Ordinary concrete	34		...
H ₂ O		11	...
Iron		6	...

* One per cent B. † Equivalent.

Radial components. Moving radially outward from the core center, the first shielding medium is a 12-in. thick annular ring of D_2O which sur-

rounds the core. Next a 1-in. thick plate of 18-8 stainless steel containing 1 per cent boron is rolled into a cylinder, with an outside diameter slightly smaller than the inside diameter of the pressure vessel. This cylinder was fabricated in sections and then assembled in the pressure vessel. The 2½ in. of steel next listed in Table 5-7 represents the wall of the pressure vessel, which is carbon steel internally clad with type-304 stainless steel. Immediately outside the pressure vessel is a 6-in. thick annular space. Pads of type-430 stainless steel wool (3 in. thick) are strapped to the pressure vessel with stainless steel bands and wire mesh to provide a thermal insulating layer. Stainless was chosen to resist corrosion in the event of a water-spill over the top of the vessel. Nitric acid attack must also be considered in the high flux areas. The remaining 3 in. of space contains air, or other gas, which can be drawn out by a blower and vented to the atmosphere.

A ¾-in. thick steel shell rolled into a cylinder is next placed all around the reactor and is welded to the bottom shield plug which will be described later. Lead bricks (3 in. thick) formed to the contour of the cylinder, or inner shield form, are stacked up on the outside of the inner shield form to a height slightly above the core. Two-inch thick bricks are used in the upper portion of this circular lead wall. Notched bricks are used in every third ring to form an annular space for a double cooling coil circuit between the lead and the steel inner shield form. After each cooling coil circuit has been hydrostatic and flow tested, molten lead is poured around the copper cooling coils to fill the annular spaces.

Studs are driven into the outside of the lead wall for attachment of metal lath and to indicate the depth of a coating of ferrobore plaster (½ in. thick) which is next applied. Heavy magnetite concrete (220 lb/ft³) is used to fill the lower part of the space between the ferrobore plaster and an octagonal wall (2 ft 10 in. thick) made of ordinary concrete which surrounds the entire reactor. Ordinary concrete is used to fill the upper part of this annular space. Below the reactor, magnetite and punchings concrete (290 lb/ft³) is used beginning at an elevation 6 in. below the bottom of the pressure vessel and extending 4 feet downward to the ¾-in. steel plate ceiling of the control rod room.

Axial upward. From the center of the core upward the radiation passes through 24 in. of D₂O and then a perforated thermal shield fabricated from 1 per cent boron-stainless steel plate, 1 in. thick. The boron-stainless reduces thermal neutron leakage and minimizes the effect of D₂O surface fluctuations on reactivity. Twenty-nine inches of D₂O top reflector are next followed by 126 in. of steam. The 21½ inches of steel represents a 12-in. thick shield plug which is mounted inside the pressure vessel at the top, the 9-in. thick cover of the pressure vessel, and the ½-in. thick bot-

tom of the water-filled shield tank. Three inches of type-430 stainless steel wool insulation are placed on top of the pressure vessel cover. A ring welded to the cover supports a top shield tank which contains 11 in. of water. The top of this shield tank is a 6-in. thick plate of A-7 structural steel.

Axial downward. From the center of the core the radiation must travel downward through D_2O to the grid plate, through more D_2O below this, and then through the 4-in. thick bottom head. The 6-in. space below the bottom of the vessel is filled with bulk type-430 stainless steel wool. The air in the steel wool can be vented through the same system which serves the annular space surrounding the pressure vessel.

A bottom shield plug form is made from $\frac{3}{4}$ -in. thick steel plate. The diameter of the plug is approximately 10 ft at the bottom and tapers to the diameter of the inner shield form at the top. Stepped sleeves are provided to form holes through the plug for the 9 control rod drive nozzles, and the 4 recirculation inlet nozzles. Asbestos rope insulation is wrapped around the 4 recirculation nozzles to a thickness of $\frac{1}{2}$ in. Copper cooling coil circuits are provided in the shield around the recirculation nozzles. The top layer of the bottom shield plug is 3 in. of lead with embedded cooling coils, covering a 1-in. layer of ferroboration plaster. Extending downward, approximately 4 ft, the plug is filled with magnetite and punchings concrete (density: 290 lb/ft³). All nozzles at the bottom of the shield plug are encircled with lead (2 in. thick).

At points up the side of the reactor vessel, where pipes must penetrate the shielding and attach to the vessel, large stepped rectangular access holes, or tunnels, are left in the bulk shielding by inserting permanent forms made up of $\frac{1}{4}$ -in. thick carbon steel plate prior to pouring the bulk shielding. These permanent forms are welded at their inner end to the $\frac{3}{4}$ -in. thick inner shield form. Pipes are brought out from the vessel through holes in the inner shield form, and after going through a triple offset to eliminate direct paths for radiation streaming, the pipes pass through the outer concrete retaining wall through larger pipe sleeves set in the concrete.

6. *Fuel-handling system.*¹¹ The problems of fuel handling in a light-water core are normally somewhat difficult because of the compact nature of the fuel lattice necessary in this type of reactor. This basic problem is made even more difficult in the case of EBWR by the comparatively long distances over which the fuel must be moved (because of the vessel height) and by the weight of the fuel assemblies themselves. As a consequence of these complexities, it was necessary to design an easily adjusted, positionable, biological shield at the vessel closure flange through which the fuel could be safely removed, and a positively locked telescopic

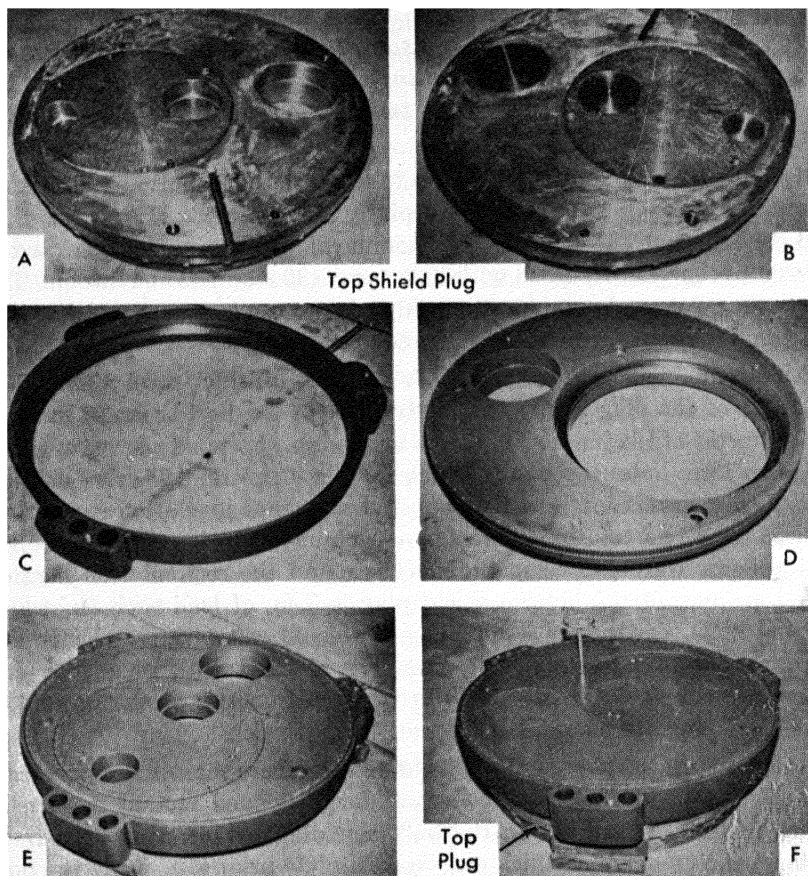


FIG. 5-28. Top shield and carrier assembly.

fuel-handling tool which could be operated with reasonable headroom requirements and without the danger of dropping a fuel assembly. The damage to the core structure and fuel assemblies as a consequence of a 150-lb fuel assembly falling through a distance of 15 ft need not be emphasized here. The remaining items of equipment comprising the fuel handling system are the coffin and coffin carriage used to transfer the fuel from the reactor vessel to the storage well, the fuel-handling tool, and the fuel storage well.

The solution to the problem of biological shielding at the vessel closure flange is the composite steel shield-plug assembly shown in Fig. 5-28. It is composed of an internal set of type-304 stainless-steel plugs, shown in views A and B, and an external carrier assembly, shown in the view E. The carrier assembly consists of a set of plain carbon steel plugs shown

in views D and E and the carrier ring shown in view C. Both the eccentric and main carrier plugs are mounted upon ball-bearing thrust washers and are equipped with peripheral ring gears to permit easy rotational positioning of the access ports.

The internal set is mounted on supports located on the inner surface of the upper ring forging of the pressure vessel. They remain in the vessel during reactor operation, forming part of the upper reactor biological shield as well as providing protection against pressure-vessel-cover neutron activation.

During the fuel handling operation the carrier assembly is placed upon the closure flange of the pressure vessel. The internal plugs are raised from their supports and clamped to the undersurface of the carrier assembly plugs by jack bolts located in the carrier plugs. The eccentric plugs and the access ports are aligned prior to the clamping operation, permitting the entire complex to be operated as a unit that provides unhampered access to the vessel interior.

This system of independently rotatable ported plugs, one mounted eccentrically within the other, enables the operator to easily and quickly locate a given port over any desired point in the core without compromising personnel radiation protection.

The irradiated fuel is moved from the vessel to the storage well in the fuel transfer coffin shown in Fig. 5-29. The coffin is supported by three screw jacks mounted on the carriage trolley, which is, in turn, mounted on rails running the length of the carriage bridge. The carriage bridge is mounted on rails which extend between, and span, the reactor and the storage well. Vertical coffin motion is obtained by the 3-part gear-head motor-driven screw jacks. The trolley is equipped with a motor-driven lead screw to permit lateral motion. The bridge is motor driven between the well and the reactor vessel. This freedom of motion in three directions permits the coffin to be positioned over any fuel location within the core. Normal procedure during fuel handling operation is to place the unloading port in the carrier plate over the desired fuel position. The coffin is then located directly over the unloading port and lowered on to the carrier-plate surface. This method eliminates any radiation beams during the transfer process.

The coffin provides approximately 13 in. of shielding radially about the fuel zone tube and is fitted with removable shielding plugs at both ends of the tube. Two sets of cooling coils are wrapped helically about the fuel zone tube, within the lead, to remove fuel-assembly decay heat during the transfer operation. Should this prove inadequate, the lower tube plug or door is equipped with a watertight seal allowing complete flooding of the fuel zone tube. Suitable fill lines and drains are provided for this purpose.

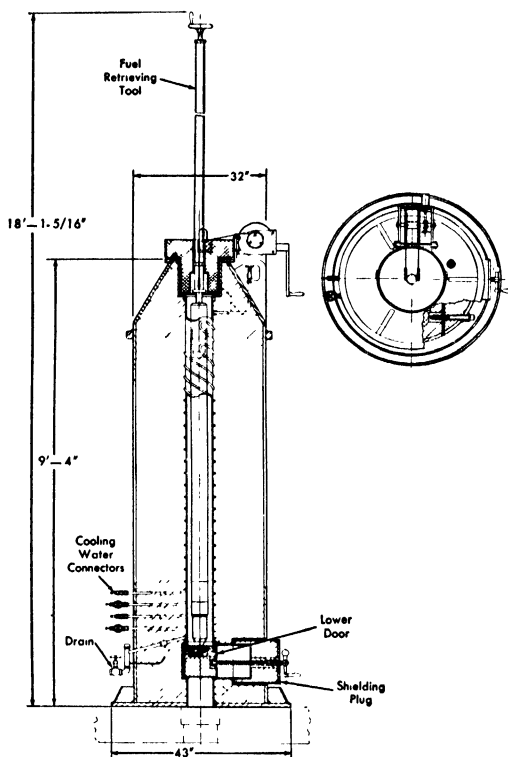


FIG. 5-29. Fuel transfer coffin.

As shown in Fig. 5-30, the coffin also contains the telescoping fuel-handling tool. This is stored within the coffin, and operates through the fuel zone tube. This tool is used only during irradiated-fuel transfer operations. A somewhat simplified version is used for loading cold clean fuel.

The fuel assembly and the tool are locked together by the two latch pins contained in the tool head shown in Fig. 5-30. The pins are pushed into or removed from the latch pinholes of the fuel-assembly top fitting by the operator rod. Raising and lowering of the tool and fuel assembly are done by means of a pair of cables secured to a winch located at the top of the coffin. The tool also has an operator lock which is actuated by the lifting action of the cables. Once this lock has been engaged it cannot be disengaged (to permit operator rod motion) until the telescoping operator rod has been fully extended and the weight of the fuel assembly has been removed from the tool. In this manner the accidental delatching of a fuel assembly during the fuel handling operation and its attendant damage to the reactor core are eliminated.

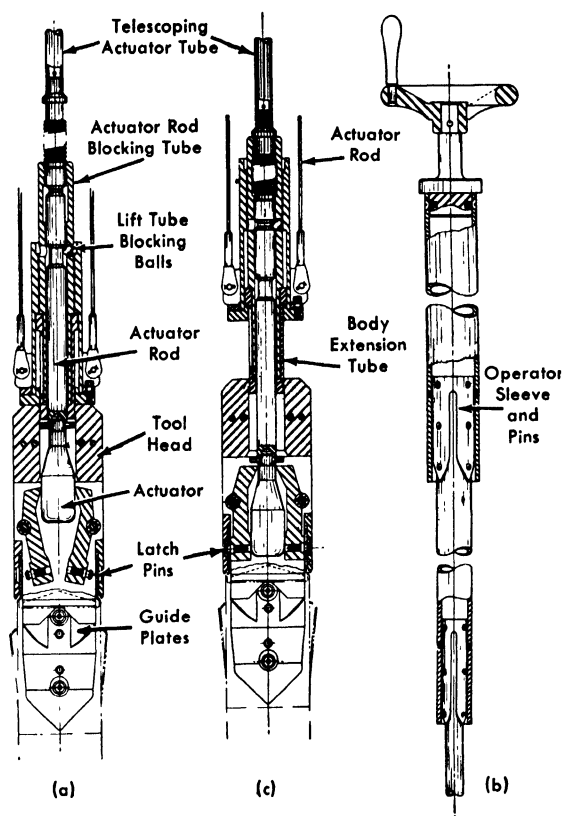


FIG. 5-30. Fuel assembly retrieving tool.

The fuel storage well, shown in Fig. 5-31, is a water-filled pit 25 ft deep, 10 ft square at the top, and 8 ft wide at the bottom. The well contains a lead and magnetite concrete unloading chute at one corner. The purpose of this chute is to provide adequate personnel biological shielding during the transfer of irradiated fuel from the coffin to the well. The well also contains a boron-stainless steel fuel storage rack located at the bottom and along the reactor side of the well.

The storage rack is a rectangular grouping of 4-inch square boxes formed by interlocking $\frac{1}{8}$ -in. thick sheets of stainless steel and boron-stainless steel. While this storage method provides a simple solution to the shielding problem, a criticality problem is posed in exchange. The storage rack, fully loaded with irradiated fuel and covered with water, can form a slab reactor approximately 5 ft wide by 4 ft deep by 4 ft high. Hence, a sufficient quantity of absorber must be built into the rack to maintain a k_{eff} less than 1.

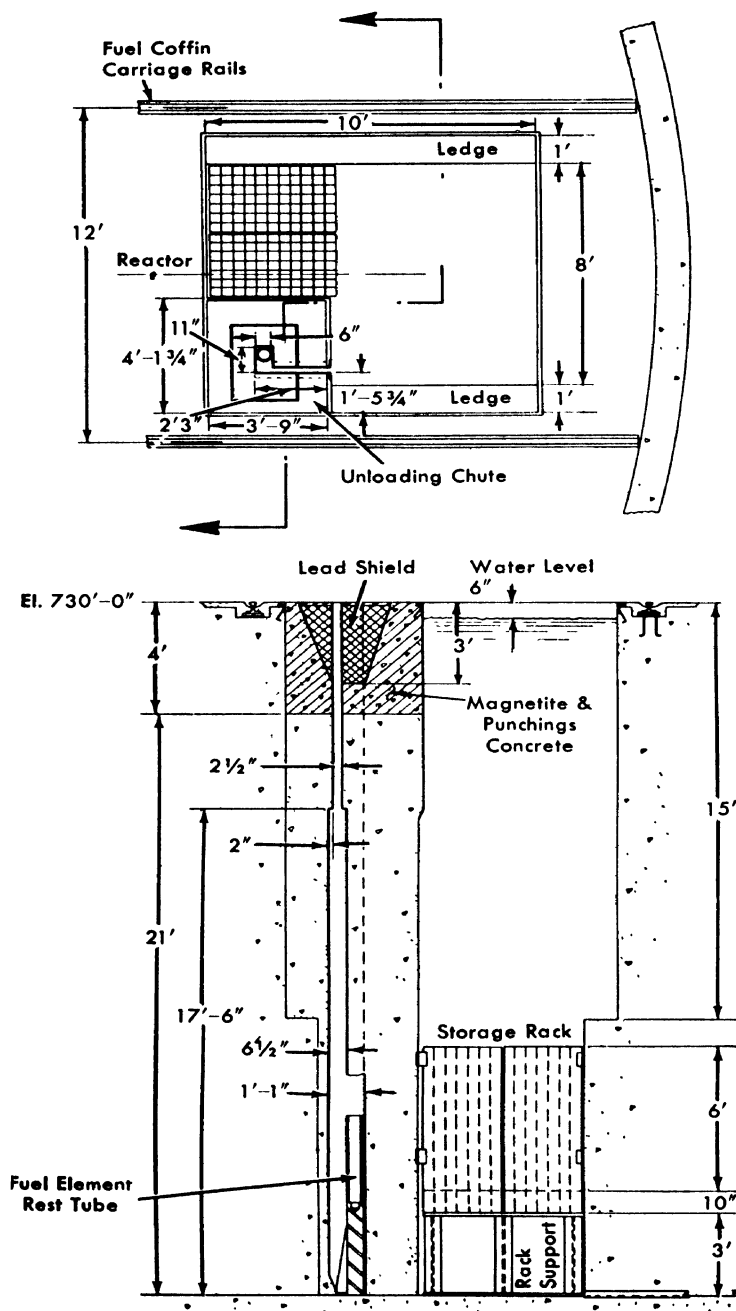


FIG. 5-31. Fuel storage pit.

Calculations of absorber worth, in the described configuration, indicate that as much as 29.1 per cent reactivity could be controlled by using 1 w/o boron-stainless steel sheet to form 2 sides of each of the 4-in. square boxes, and type-304 stainless steel to form the remaining two sides. The amount of reactivity which might require control would be approximately 23 per cent. This value is based on a full storage rack loading of 148 fuel assemblies containing 6 plates each of 0.271-in. thick, 1.4 per cent enriched type. The value also includes a realistic leakage, based on the slab dimensions of the storage rack, and a maximum possible error in the fuel reactivity calculations of 7 per cent. Some degree of pessimism is also included, since no credit has been taken for burnup or for the xenon and samarium buildup during irradiation.

The worth of the boron-stainless steel was determined by comparing its absorption to that of the water in the fuel assembly, and multiplying the ratio of reactivity controlled by the water in the fuel assembly (15 per cent—for previous calculations).

$$\rho = \frac{\Delta k_{\text{eff}}}{k_{\text{eff}}} = \frac{0.15 V_2 \Sigma_{a_2} d}{V_w \Sigma_{a_w}}, \quad (5-1)$$

where V_2 = volume of boron-stainless per assembly (24 in³),

Σ_{a_2} = macroscopic absorption cross section of the boron-stainless (3.056 cm⁻¹),

V_w = volume of water per assembly (485.85 in³),

Σ_{a_w} = macroscopic absorption cross section of the water (0.11959 cm⁻¹),

$d = \frac{\bar{\phi}_{\text{boron}}}{\phi_{\text{assembly}}} = \text{disadvantage factor for the boron-stainless.}$

The value of d was obtained in two steps. First, a value for the ratio of the average flux in the boron-stainless to its surface flux was obtained from existing transport theory solutions. This value (d_2) was found to be about 0.4.

Then, the ratio of the flux on the surface of the boron to the average flux in the fuel assembly was calculated from a solution of the diffusion equation with appropriate boundary conditions. The equations used for this step are shown below.

$$D_1 \frac{\partial^2 \phi_1}{\partial X^2} - \Sigma_{a_1} \phi_1 + q = 0, \quad (5-2)$$

$$\frac{\partial \phi_1}{\partial X} = 0, \quad X = 0, \quad (5-3)$$

$$-D_1 \frac{\partial \phi_1}{\partial X} = d_2 \phi_1(a) \Sigma_{a_2} t_2, \quad X = a, \quad (5-4)$$

where D_1 = diffusion coefficient for the fuel assembly (0.23953 cm),

ϕ_1 = flux in fuel assembly,

Σ_{a_1} = macroscopic absorption cross section of the fuel assembly (0.12597 cm⁻¹),

q = thermal neutron source in the fuel (neutrons/cc),

t_2 = half thickness of boron-stainless,

$d_2 \phi_1(a) \Sigma_{a_2} t_2$ = total number of neutrons absorbed in the boron-stainless per cm² of area,

X = distance from center of fuel cell,

a = half thickness of fuel cell.

The above system was solved assuming the fuel cell to be an infinite slab 4 in. thick with boron-stainless steel on the faces. The disadvantage factor d_1 was then determined as

$$d_1 = \frac{\phi_1(a)}{\bar{\phi}_1}$$

The product of d_1 and d_2 is the desired over-all disadvantage factor, d , and was found to be 0.22.

The resulting value for ρ was 25.5 per cent.

In the interest of minimizing the size and weight of the fuel shipping coffin, the locating end fittings and top handling fittings will be removed from the irradiated assemblies in the storage pit and only the fuel plates and adjacent side plates will be handled in the shipping coffin. Removal of fittings is accomplished by shearing the side plates just above and below the fuel plates with a manually operated shear. The shear consists of a pair of chrome-plated tool steel blades mounted at the lower end of two 3-in. aluminum pipes, 22 ft long. The pipes are sealed at the ends, and the resulting buoyancy causes the device to float at approximately the correct level for use. The blades move horizontally and are provided with a double angle to prevent the side plate from slipping out during shearing operation. Figure 5-32 presents details of the shear, and Fig. 5-33 shows a fuel element in position to have the bottom end fitting cut off.

After a suitable decay period the fuel assemblies will be removed from the storage pit and transported to a chemical processing plant in a shipping coffin.

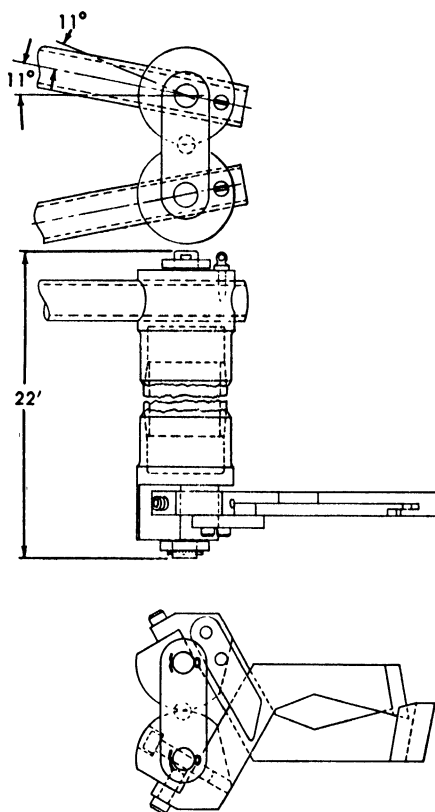


FIG. 5-32. Fuel element end fitting shear.

The entire fuel-handling system was fabricated using standard engineering and machine shop practice. Indeed, some of the clearances and tolerances were so loose as to qualify them as so-called "agricultural" fits.

7. *Instrumentation.* The major difference between the instrumentation for EBWR and the instrumentation of a conventional power plant is in the addition of the nuclear instrumentation and the gastight seal feature requiring all instrumentation to be done electrically. In this description, therefore, the nuclear instrumentation will be emphasized.¹³

There are 9 channels of nuclear instruments on EBWR. Three of the channels are used during startup, and the remaining 6 are used during full-power operation. Figure 5-34 shows the ranges covered by the various instruments used to measure the neutron density of the reactor. The middle column indicating neutron flux at the instrument location is based on a chamber sensitivity of 10^{-14} amp/(n)(cm²)(sec). The starting

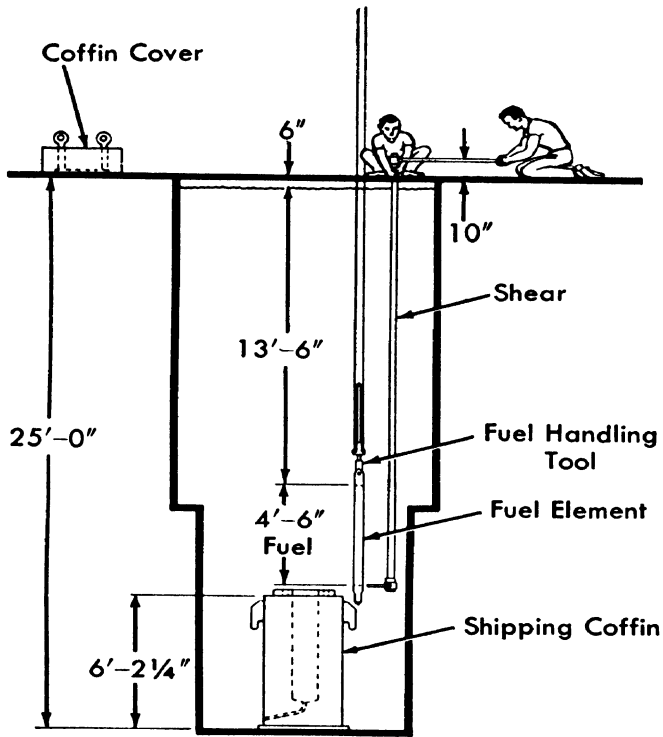


Fig. 5-33. Spent fuel element cutting operation.

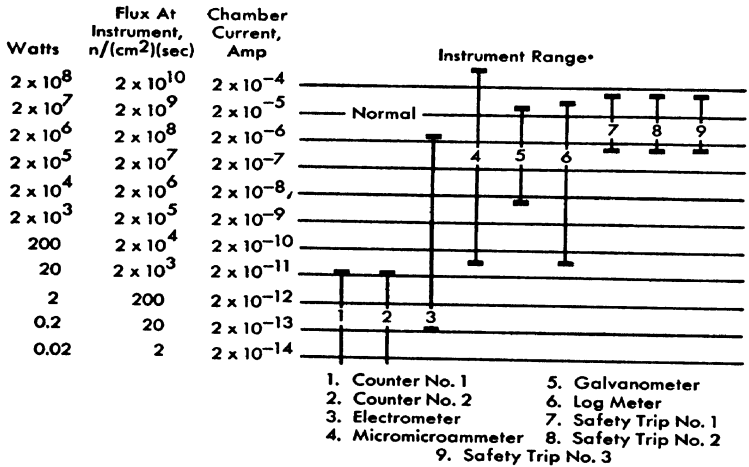


Fig. 5-34. Ranges of nuclear instrumentation.

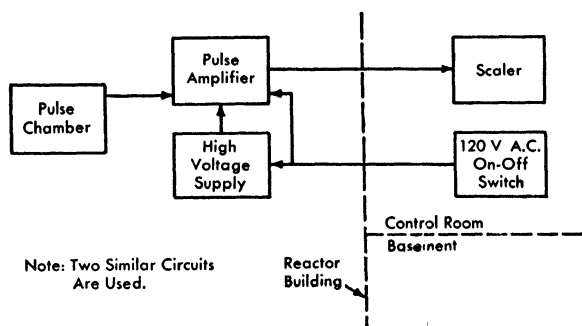


FIG. 5-35. Fission counter chain.

source level after extended shutdown is usually about 2 watts. Channels 1 and 2 operate from pulse chambers. Counting channels 1 and 2 and the electrometer are effective from source level to intermediate power, and the rest are effective from intermediate to full power. Equipment which requires a short cable run between the detector and the input circuit is installed adjacent to the detector. The balance of the amplification equipment is located in the service building basement. Coaxial cables between the reactor and the service building are terminated at the building shell in pressurized feed-through connectors used to maintain the gastight feature of the building. The feed-through connectors are mounted in bakelite to prevent a common ground point for cable shields.

The counting chains, Fig. 5-35, are used during reactor startup to read multiplication of neutrons at source level. They are designed to accommodate either BF_3 or fission pulse chambers as sensing elements. The preamplifiers are now located close to the instrument holes containing the pulse chambers to minimize the lead length between chamber and amplifier. The output of the preamplifier is registered on scalars at the control board. A new sensing unit is being developed which combines the pulse chamber and preamplifier into a single unit. The combined unit will improve the counter performance by reducing the signal-to-noise ratio and providing better neutron-to-gamma discrimination.

The electrometer is a sensitive current-measuring instrument with a vibrating capacitance modulator which is used to measure the output current of a compensated ionization-current chamber. The compensated chamber is useful at low current levels to discriminate against the fission-product gamma background.

Figure 5-36 shows the electrometer chain. The electrometer has a minimum sensitivity of 3×10^{-7} amp full scale and a maximum sensitivity of 3×10^{-13} full scale. The amplifier is located at the reactor, and range changing is done from the control console through a remote selector switch.

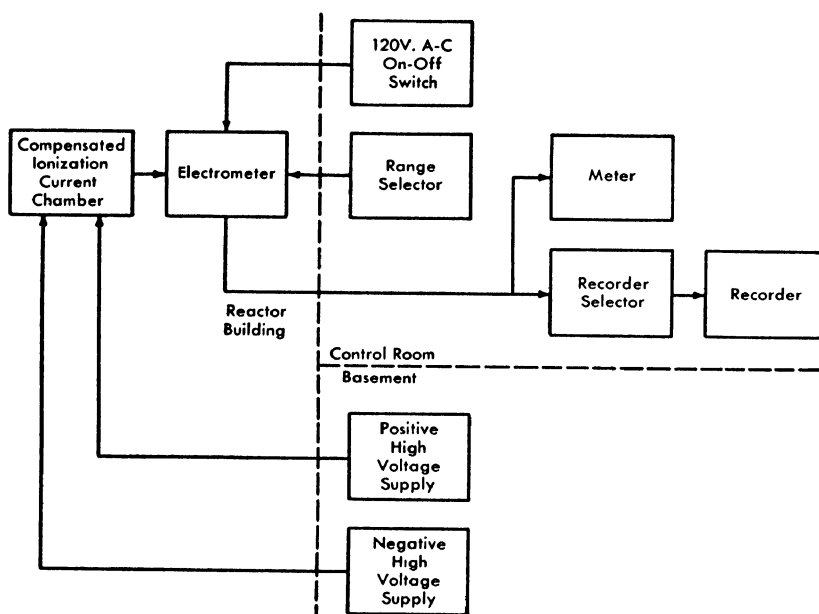


FIG. 5-36. Electrometer chain.

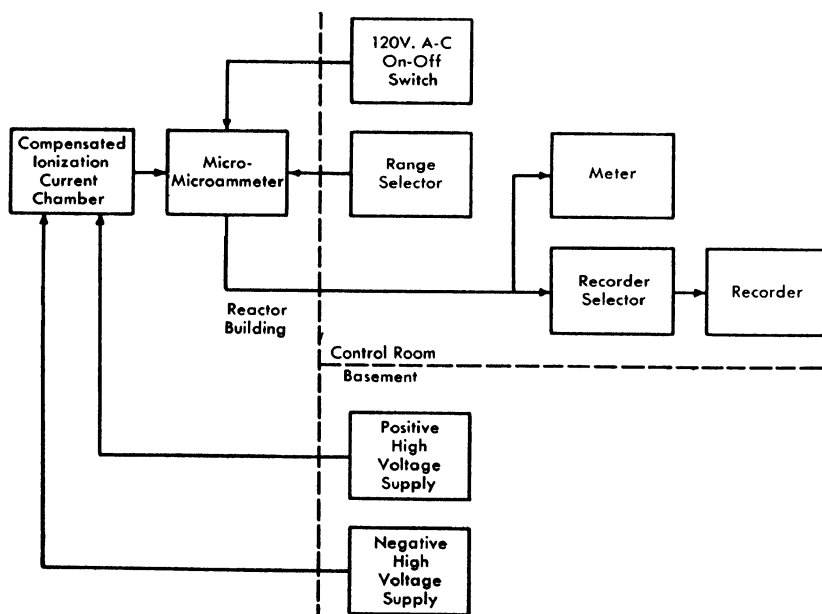


FIG. 5-37. Micromicrometer chain.

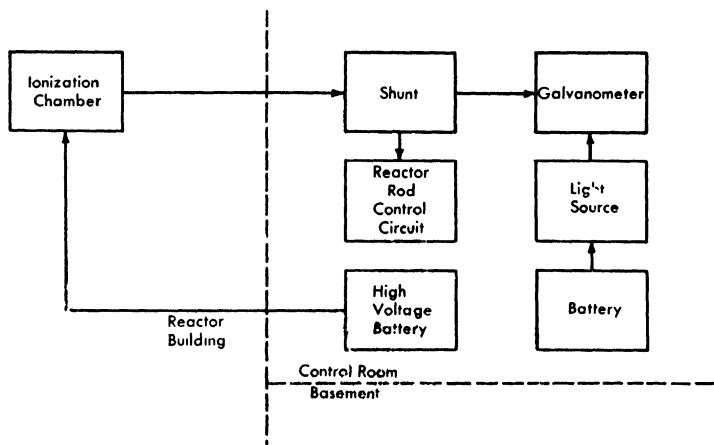


FIG. 5-38. Galvanometer chain.

Special noise-free coaxial cable with a semiconductive coating over the dielectric is used to connect the chamber to the amplifier. Amplifier output is displayed on a meter at the reactor console and can be connected to a linear recorder through a selector switch.

The micromicroammeter is a modulated chopper type d-c amplifier with a variable input resistor which measures the current from a compensated ionization-current chamber. Figure 5-37, shows the micromicroammeter chain. This instrument has a maximum sensitivity of 10^{-9} amp full scale and a maximum current reading of 5×10^{-4} amp. Amplifier output is displayed on a meter at the reactor console and is connected to the linear recorder through a selector switch.

The linear flux recorder, which is a strip-chart instrument, is used to record the output of the electrometer at low power level and the output of the micromicroammeter from the intermediate to full power level. The instrument is calibrated linearly from 0 to 100, has a 3-sec pen speed, and has chart speeds of 2 inches/min and 12 in/hr.

A light-beam galvanometer, shown in Fig. 5-38, is used to measure the current from the uncompensated ionization-current chamber. Batteries are used for both the ionization-chamber high-voltage supply and the galvanometer light source. The galvanometer has a sensitivity of 3×10^{-10} amp/mm, and the addition of a shunt results in a range from 150 kw to 50 Mw full scale.

The logarithmic amplifier shown in Fig. 5-39 measures the current output from a compensated ionization-current chamber over the range of 10^{-11} to 10^{-3} amp. The period amplifier will indicate periods as short

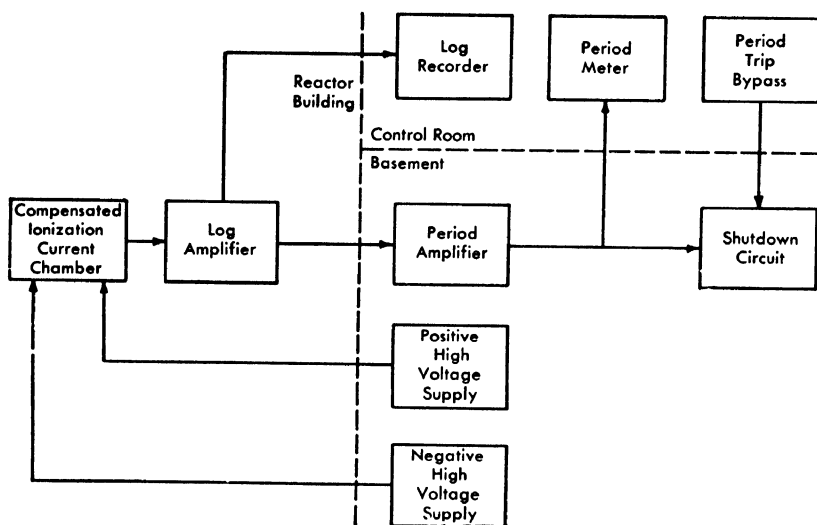


FIG. 5-39. Log period chain.

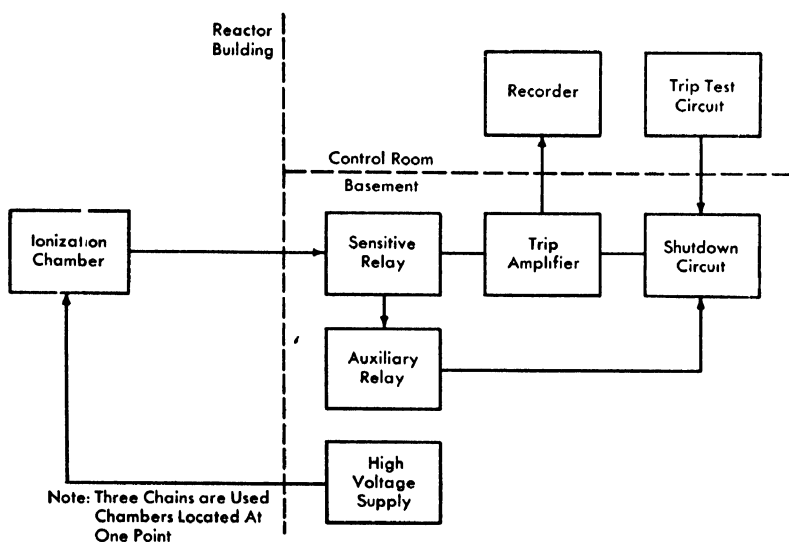


FIG. 5-40. Safety trip chain.

as 3 seconds and is set to trip on a 5-sec period. The log-amplifier is located at the reactor to minimize the cable capacity and reduce the effect of spurious noise on the period meter. The log-period chain is most useful during startup when the reactor is subcooled and can maintain a period. A switch is provided to bypass the period trip circuit when the reactor is boiling.

The output of the log amplifier is recorded on a strip-chart recorder with a logarithmic scale calibration of 10^{-4} to 3×10^2 . The recorder is calibrated for 10^2 to correspond to 20 Mw. It has a 3-sec pen speed and chart speeds of 2 in/min and 12 in/hr.

Figure 5-40 shows the safety trip chains. Three identical chains are used to shut down the reactor under abnormal conditions of high flux. The ionization chamber current is used to actuate a sensitive relay and to trip an electronic amplifier. The electronic amplifiers are set to trip at 120 per cent of normal full power, and the sensitive relays are set to trip at 125 per cent of normal full power. Tripping of any two of the three amplifiers is required to shut down the reactor. If a single amplifier failure occurs, a warning signal is provided at the reactor console, and the amplifier can be repaired with the reactor in operation. Actuation of any of the three sensitive meter-movement relays shuts down the reactor.

Three miniature recorders are used to record the output of the three safety-trip amplifiers. The recorders have a 3-in. chart, 3-sec pen speed, and a chart speed of 1 in/hr.

Radiation monitoring system. The radiation monitoring system was designed to achieve the following objectives:

- a. Register any changes in radiation levels which might indicate abnormal plant operating conditions or contamination from fission products.
- b. Indicate general conditions of personnel safety.
- c. Initiate reactor shutdown and closure of building air inlet and exhaust systems when there is an excessive amount of radiation present in the building exhaust air.

Gamma radiation has been selected as the best measure of over-all activity because it is usually the strongest type present in a gross activity. It can be conveniently measured by a simple rugged detector, the ion chamber, and shielding problems do not seriously affect the measurement. Relative gamma activity of liquid or gas flowing in a steel pipe is more easily measured outside of the pipe than beta and alpha activity. This is important because radioactivity measurements in a nuclear steam power plant involve contained gases and liquids.

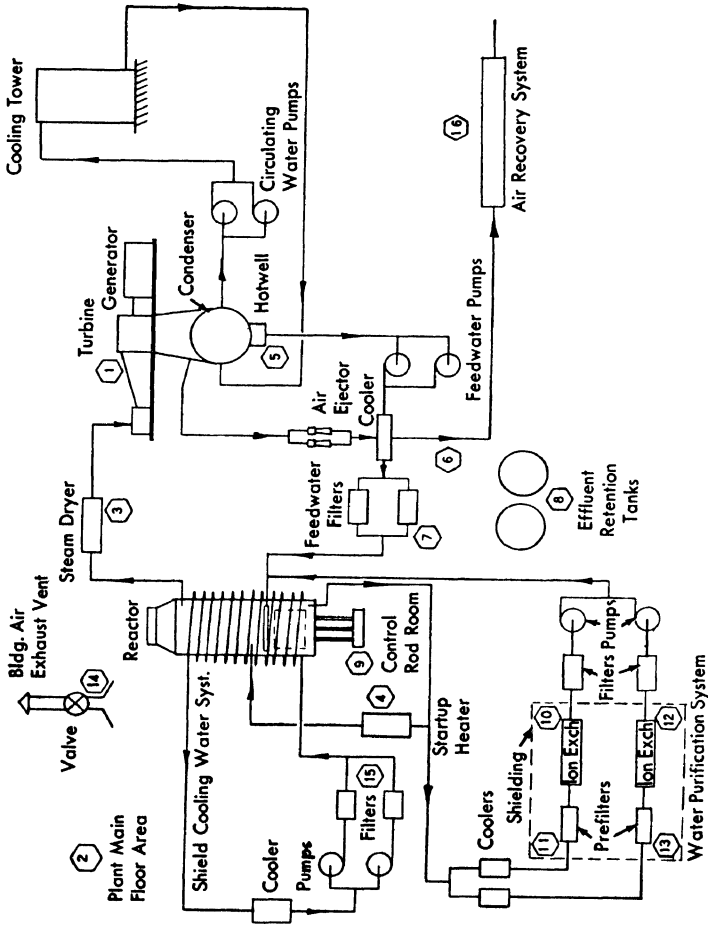


FIG. 5-41 EBWR radiation monitoring channels.

The monitoring system consists of 14 gamma monitor channels and one special channel for detecting fission products in the condenser exhaust air. Figure 5-41 is a plant schematic showing the monitor locations. Each channel has a portable amplifier unit located near the plant equipment or operation being monitored. The detector for each gamma channel is a gamma-sensitive ionization chamber connected to the amplifier by a few feet of coaxial cable. All gamma amplifier circuits are of the direct current, feedback type input with range switching over four decades. The panel meter indicates true radiation level at the detector location. All channels operate a common annunciator alarm light indirectly through an adjustable trip contact on the recorder slide wire mechanism. This trip circuit is operated by any channel whose recorder level signal exceeds the preset value. Examination of the chart identifies the channel giving an alarm.

Plant parameters. All instrumentation of plant parameters is done by electrical transmission. Remote controlled, closed circuit television is used to provide a direct indication of reactor water level from a sight glass and main line steam pressure from a bourdon tube-type gage. Remote controls are provided at the viewer.

Plant temperatures are measured with either iron-constantan thermocouples or resistance thermometer detectors as indicated on each instrument in the control room boards. Reactor vessel temperatures are measured with chromel-alumel thermocouples.

The multipoint temperature recorders have reference junctions located inside the plant shell. Regular iron-constantan thermocouple lead wire is used between the reference junction box terminals and the thermocouple, and one pair of bonded rubber covered wire per thermocouple is used to connect the recorder to the reference junction. One pair of special bonded iron-constantan lead wires are used to connect the recorder to the reference junction thermocouple. In this way the instrument is compensated for the temperature inside the plant.

For the measurement of pressure, the pressure transducers are of the force-balance type. The pressure-sensitive element is either a bourdon tube or bellows, depending on the pressure range to be measured. Movement of the pressure-sensitive element displaces a balanced beam and changes the current in an electric oscillator. The change in oscillator current restores the beam to its balanced condition. The current output of the oscillator is proportional to pressure and varies from 0.5 to 5.0 ma. The pressure indicators operate in reverse fashion, the indication being proportional to current input. Long-life vacuum tubes are used in both the transmitters and receiver amplifiers.

Level is measured by a differential pressure cell coupled to a beam balance unit.

Water and steam flow are measured by applying the differential pressure across a flow nozzle to a differential pressure cell. Air flow is measured by using the beam balance unit to determine the position of a plug inside a tapered flow tube.

Conductivity is determined by means of an eight-point conductivity recorder which has a sensitivity of 0 to 5 $\mu\text{ohms}/\text{cm}^3$.

Except for the control-rod drives which use synchro repeater systems, all position indication is done with potentiometers attached to the valve stem or device being monitored. The indicators are powered with 125 volts DC. A 1000 ohm/volt meter measures the output voltage of the potentiometer.

8. *Control systems.* The power controls of EBWR to be considered here will be limited to those having a major influence on the plant dynamics. This group of controls may be conveniently divided into four categories which are: (1) reactor power-output controls, (2) steam-system controls, (3) level and feedwater controls, and (4) generator controls. These are not listed in the order of their importance but rather in the order of their appearance in the power cycle. The generator controls are conventional and will not be considered here. The other three control groups are all interactive to an extent not ordinarily found in conventional power plants. This interaction results from the fact that the steam voids in the core are drastically influenced by both steam load and feedwater injection. The steam voids, in turn, strongly affect reactivity and reactor power output or firing rate. For these reasons, manual and automatic controls for the steam and water systems are provided at the operating center as well as on the panel boards. The reactor power-output controls are also located at the operating center, thus making it possible to exercise complete plant control from one point where the interaction of all three systems can be observed.

(a) *Reactor power output control.* Five hafnium and four boron stainless-steel neutron-absorbing rods are used to control power output.¹⁴ Their position is indicated at the operating center on dials that include "in" limit, "out" limit, and latch lights to inform the operator of rod conditions at startup or following a scram. Operation of the rods is supervised by means of a set of interlocks which require plant parameters to be at safe values in order to start and operate the reactor. The interlocks, together with the operating values, are as follows: (1) Water temperature, <325 F. (2) Reactor pressure, >670 psig. (3) Turbine condenser pressure, >4.5 psia. (4) Reactor water level, >6 ft above core. (5) Reactor water level, <2½ ft above core. (6) Pressure relief valves open. (7) Turbine throttle valve closed. (8) Reactor power level, >26 Mw.

Additional startup interlocks require that the power galvanometer shunt

be put in the most sensitive position, that the period instrument interlock not be bypassed, and that all rods be fully inserted. Interlock bypass and test-key switches are used to facilitate startup. It is necessary, for example, to bypass "turbine trip throttle" and "condenser pressure high" interlocks on cold starts since the reactor is at power when these interlocks clear.

The rods may be raised only one at a time to increase reactivity. Each shim rod is selected for operation by use of a central switch, although the center rod has its own set of controls to permit interchange of shim-rod positions. All 9 rods may be lowered simultaneously for fast power reduction without a scram. This is useful for rod calibration by the increment method. Individual drop of rods is provided to check out their insertion time prior to reactor operation.

The five hafnium rods along with the four boron steel rods are operated as a bank. The reactor is critical with the rods withdrawn 12 to 17 in., corresponding to operating pressures of about 100 to 600 psi with low xenon concentration. A typical rod power calibration is shown in Fig. 5-42. The slight curvature can be accounted for by the nonlinearity of reactivity variation with rod position. Withdrawal of about 5.5 in. with a 5-rod bank will permit zero to full power operation with low xenon. Compensation for equilibrium xenon requires about 5 in. additional withdrawal for criticality and about 6.3 in. to go from zero to full power.

Reactor power maneuverability becomes evident when the rod withdrawal rate of 28 in./min is considered. It would be mechanically possible to change power over the range shown in Fig. 5-42 in about 1½ min. In the actual case the power can overshoot owing to feedwater mixing delay, and about 5 to 7 min is ordinarily used in the maneuver to be sure to reach equilibrium rod position without high flux trip-out.

(b) *Steam system control.* Large transient variations in the steam void volume of a boiling water-reactor core must be avoided because such variations cause large changes in reactor output. EBWR is designed for normal operation at constant pressure and constant steam-flow load irrespective of steam demand by the turbine.¹⁵ This is effected by the steam bypass system described earlier in this chapter under "Operational Characteristics," and shown schematically in Fig. 5-43. The conditions for equilibrium are shown in the equation:

$$K_1(P - P_0) = BP + TA = K_2(RS)$$

where K_1 and K_2 are constants. The difference between reactor pressure P and a manually set pressure reference P_0 is used as a proportional control to determine the total flow. A signal TA is provided from the turbine admission valve to decrease the bypass steam BP in direct proportion to

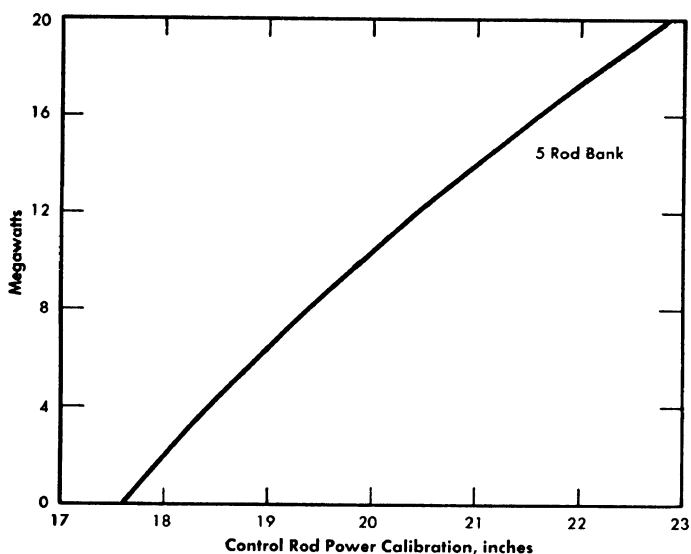


FIG. 5-42. Control rod power calibration.

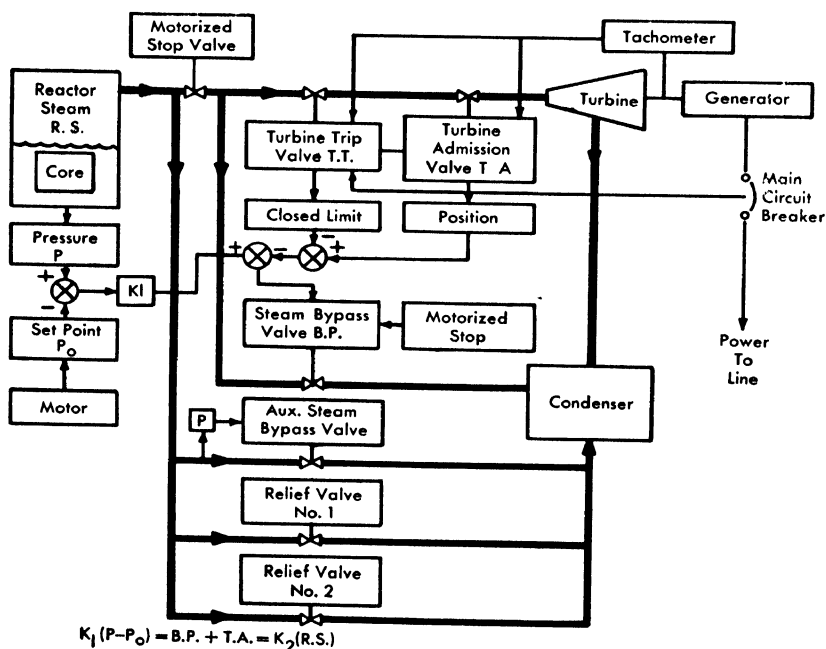


FIG. 5-43. EBWR steam system.

the amount of steam taken by the turbine. Also, a trip is provided so that if the turbine-generator is dropped off the line, the steam load, ranging up to 20-Mw output, is automatically diverted through the bypass system with less than a 5 per cent pressure excursion. In this way a constant load is maintained on the reactor to improve stability. If the steam load ($BP + TA$) differs from the steaming rate RS , the pressure P will change and adjust the bypass load until equilibrium between load and steaming rate is reestablished.

The operating center steam controls provide for automatic control by means of P_0 adjustment and manual control by direct positioning of the valve with an electric motor. The reactor pressure P and the reference setting P_0 are indicated on meters just above the control rod switches. Figure 5-44 is a photograph of the valve and its hydraulic regulator. The valve port is designed to produce steam flow in direct proportion to valve position. This was also done with the turbine admission valve so that flow interchange could be accomplished without changing the system gains and stability. A hydraulic piston is used to position the bypass valve during automatic operation. The valve is made to fall closed by means of a large spring because a reactor blowdown would temperature-stress the pressure vessel and produce other undesirable effects. A steam-pressure-modulated bypass relief valve is provided to take care of any overpressure caused by a closure of the bypass control valve. Figure 5-45 shows the valve hydraulic controller interior with the important variables labeled. Figure 5-46 shows the hydraulic transducer which generates the TA signal at the turbine. A schematic diagram of the bypass control system is shown in Fig. 5-47.

The valve drive piston is operated by hydraulic oil at a pressure of 1000 psi. The piston is controlled by three jet-pipe hydraulic amplifiers, using oil at 100 psi from a pressure reducer. Each amplifier is a jet-pipe pivoted on a perpendicular axis, so that it directs a stream of oil between two holes in a face plate. When the jet-pipe is displaced toward either of the holes, a differential pressure between the holes results, with the higher pressure at the hole to which the jet-pipe is directed. Small mechanical directing forces on the jet-pipe can achieve a pressure differential of up to 50 psi.

(1) A change in governor valve position moves a cam which transmits a force (through a spring) to jet-pipe A . This moves the pipe off center and changes a hydraulic pressure signal. The pressure change produces a force in opposition to the original force by means of a bellows and linkage. When the two forces are equal, the jet-pipe is returned almost to center. The hydraulic pressure signal produced is proportional to the position of the turbine governor valve, and it is piped from the turbine unit to the bypass controller.

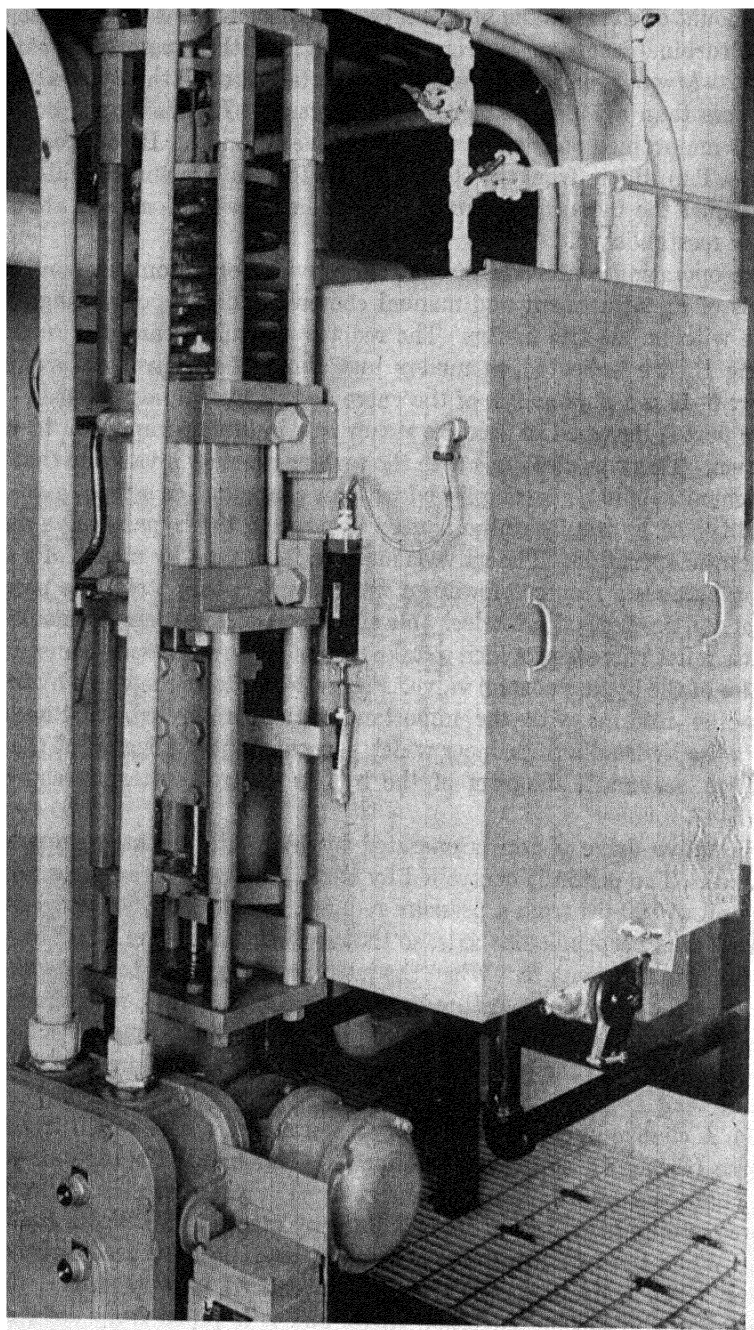


FIG. 5-44. Steam bypass regulating valve controller.

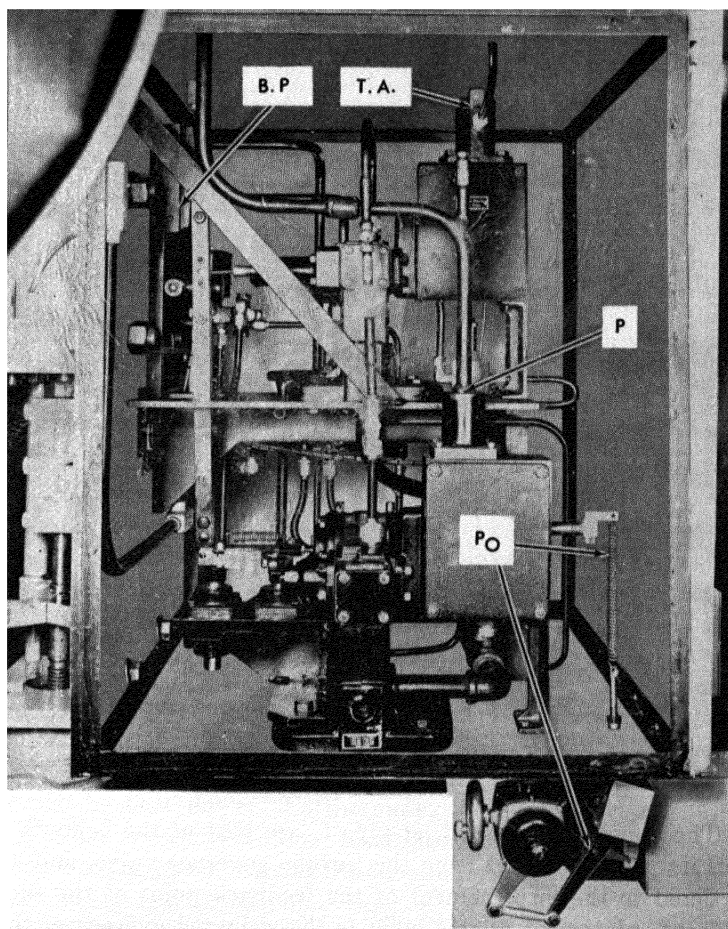


FIG. 5-45. Interior of the bypass valve controller.

(2) Reactor pressure is transmitted to a bellows assembly in the controller through a water line connected to the reactor sight gage standpipe. The difference between the bellows force and the force exerted by the pressure reference spring, which is set by a motor operated from the control room, moves jet-pipe *B* off its center position until the force applied by the auxiliary piston through the feedback linkage recenters it. The auxiliary piston position is thus proportional to the difference between reactor pressure and the pressure reference. The proportional band may be set by varying the feedback linkage.

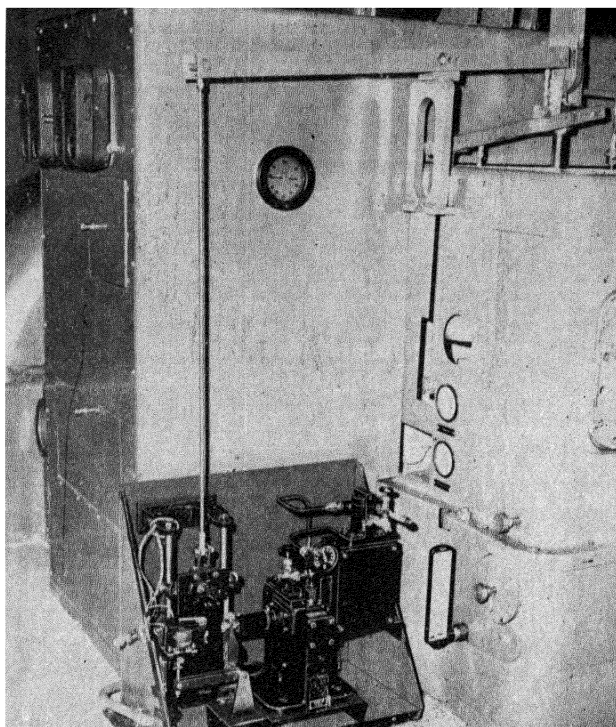


FIG. 5-46. Turbine admission valve position transducer.

(3) The opposing forces on jet-pipe *C* are that of the bellows, which receives its pressure signal from the turbine governor valve, and a force proportional to the displacement of the feedback point of the summing bar. The displacement of this point is the weighted average of the displacements of the end points of the bar. If jet-pipe *C* is off center, the differential pressure will force the pilot valve to move until the differential is zero. With the pilot valve off center, the high-pressure oil supply will drive the bypass valve piston to reposition the feedback point of the summing bar which forces jet-pipe *C*, and consequently the pilot valve, back to center position. The bypass valve position is therefore equal to a constant times the reactor pressure error minus the turbine governor valve position. Both the bypass and the turbine governor valves have linear flow versus valve position characteristics to maintain constant system gain.

(c) *Water level and feedwater control.* Feedwater control is important to boiling-reactor operation for two reasons.¹⁶ One is that the head

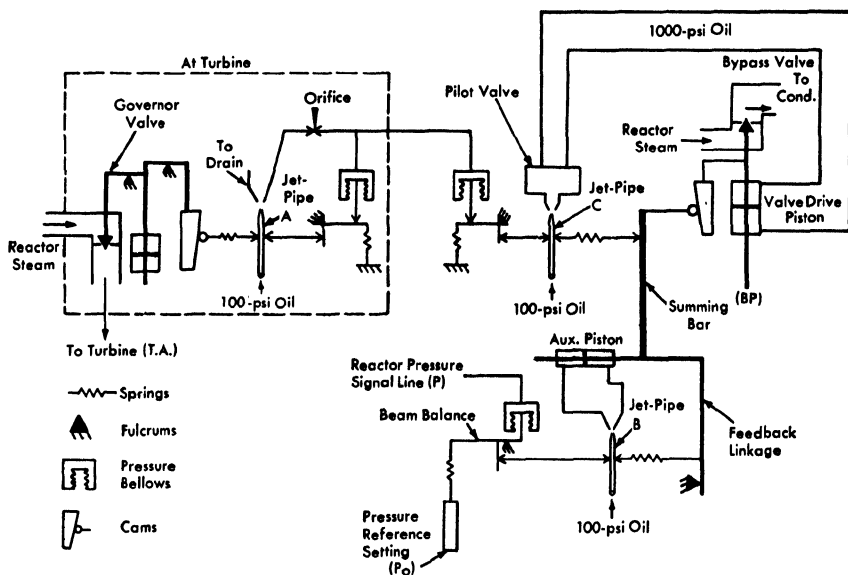


FIG. 5-47. Schematic diagram of the bypass valve control system.

of water over the core has an effect on power stability, and the other is that transient changes in temperature of the water entering the core have a strong effect on reactor power output. Feedwater flow, together with steam flow, determines the reactor water level or head, and the controller must keep the level between the interlock limits described earlier or the reactor will be shut down. Furthermore, the controller must function smoothly since intermittent flow produces intermittent changes in the temperature of the water entering the core and, in turn, causes intermittent variations in the reactor output.

The EBWR feedwater controller is shown in schematic form in Fig. 5-48. It is basically a standard electric slide-wire computer servo which controls the throttling valve in the feedwater line to the reactor. It has alternate 2- or 3-element control as shown at *A* and *B*, respectively. Early operation was confined to 2-element proportional control without automatic reset action as at *A*. The reference water level L_0 , minus the reactor level L , sets the feedwater flow W . Wide proportional bands were required in the beginning because of an oversize throttling valve. This meant that L_0 had to be manually reset with each power level change. At the present time the correct throttling valve is in use together with full 3-element control which is described by the equation at position *B* in the figure. Under equilibrium conditions the error E is zero, water flow W is equal to steam flow S , and reactor water level L is equal to the

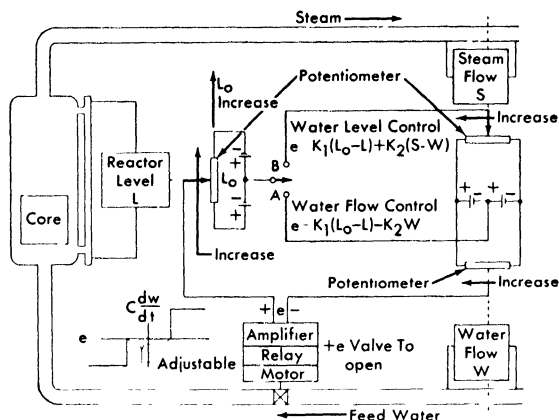


FIG. 5-48. EBWR feedwater control system.

set point L_0 . A relay amplifier with adjustable gain is used to amplify the error E and activate an electric motor which positions the throttling valve. The system dead zone or minimum error is inversely proportional to the gain of the amplifier. Therefore the gain is set as high as over-all stability permits. Both integral and derivative controls are provided to improve stability and permit higher gain settings.

The operating-center feedwater controls consist of manual and automatic push buttons together with a manual motor control button. Feedwater valve position is displayed immediately above the controls.

One factor concerning the feedwater system in EBWR that differs materially from that in a conventional steam plant is that the reactor power output increases if the feedwater flow is increased or if the feedwater temperature is decreased, either of which lowers the core inlet water temperature. The feedwater is injected at some distance from the core inlet so that a considerable delay is experienced before the feedwater-influence is displayed in the reactor output. The effects on reactor power of step injections of feedwater are shown in Fig. 5-49. On the right is shown a $1\frac{1}{2}$ times power increase resulting from a feedwater injection at low reactor pressure. Following this is an injection of about the same amount at much higher pressure, resulting in a power increase of about $2\frac{1}{2}$ times. The time delay until equilibrium is reestablished is about one minute.

Reactor power varies inversely with steam voids for constant control-rod setting and fast changes in voids. Therefore a sudden increase in steam load by means of the bypass valve, for example, will result in a rapid decrease in reactor power as shown by the third transient which resulted from improper use of the P_0 setting device. An almost instant-

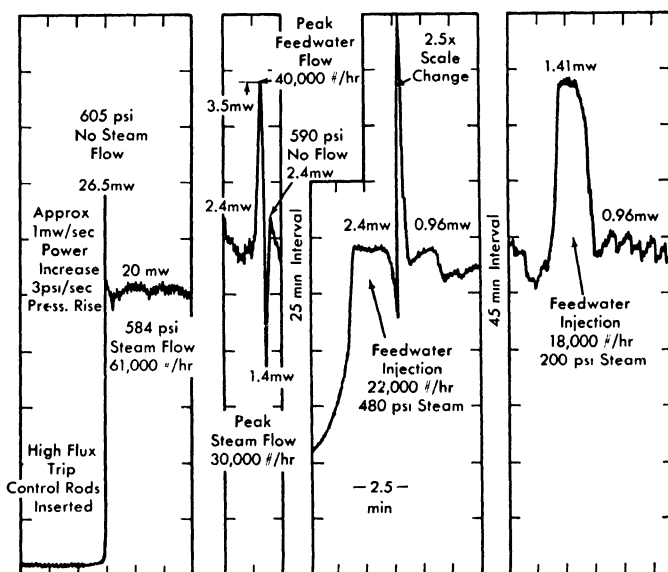


FIG. 5-49. Transient feedwater and steam-flow effects on reactor power.

neous demand for a pressure drop of 20 psi was made by means of the P_0 control. The bypass responded quickly with very large flow which began at a lower pressure. A short time later the feedwater controller responded to increase the steam flow by increasing feedwater flow. As soon as the increase in feedwater was felt at the core entrance, a rapid power excursion upward occurred. At this point the blowdown was completed and steam flow was cut off, cutting off the feedwater flow. Reactor power then returned to the level existing before the P_0 adjustment was made.

An example of what happens to reactor power output when steam load is suddenly lost while running at full power is demonstrated in the transient at the left in Fig. 5-49. A malfunction of the "turbine trip throttle" interlock resulted in a momentary closure of the bypass valve with reduction in total steam load to nearly zero flow. The resulting decrease in steam voids with increasing pressure caused the reactor power to rise quickly to a value which actuated the "high flux level" interlock and shut down the reactor.

None of these examples represents desirable operation. They are intended merely to emphasize the large interactions between plant and reactor which are not encountered in conventionally fired plants and which must be considered to secure stable over-all plant operation.

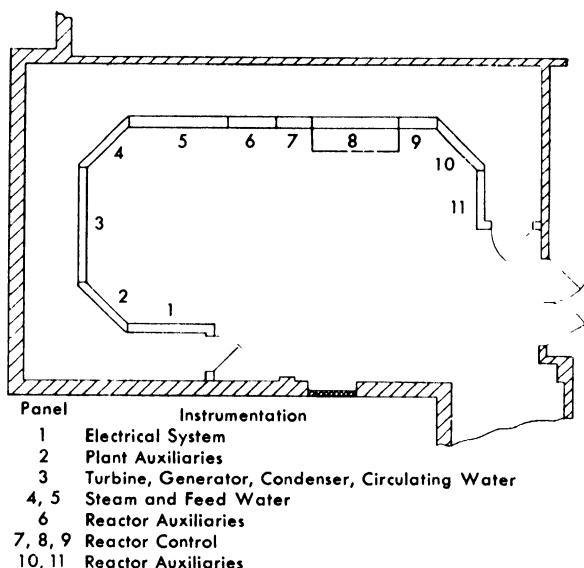


FIG. 5-50. Plan view of EBWR control room.

To determine the performance of the steam bypass control system, two series of frequency response measurements were made.¹⁷ The first obtained bypass valve position response to reactor pressure variation or $BP/(P - P_0)$. The second measured bypass valve position response to turbine admission valve variation or (BP/TA) .

The reactor pressure measurement required a pressure signal generator capable of producing a nearly sinusoidal pressure variation with peak-to-peak magnitudes of about 15 psi at average pressures of several hundred psi with a frequency range of from 0.1 to 3 cps. A pneumatic device was designed and constructed which operated a fast response pressure transducer through a bellows. The transducer was of the variable capacitance-diaphragm type and its output was fed to an amplifier. In this way the pressure signal could be measured. A differential transformer was used to measure the bypass valve position. The rectified differential transformer output and the pressure signal were graphed on a fast two-channel oscillograph.

Measurement of the bypass valve response to sinusoidal variation of the turbine governor valve was accomplished by disconnecting the cam drive rod from the governor linkage and connecting it to a variable speed reducer with a crank pin. The crank pin permitted sinusoidal excitation of the cam in the frequency range of interest.

These pre-power response measurements proved very worthwhile, not only in determining response time, but also in detecting improper operation of the system; rubbing pressure bellows, binding push rods, and the like were found and corrected. At the completion of the testing, the system performed properly and with adequate response time.

(d) *Control room arrangement.* The entire EBWR facility—the steam plant as well as the reactor—is operated by remote control. This requires an intricate system of instruments, alarm signals and control mechanisms, with all communication element located in the control room outside the power plant containment structure.

From his station at the console panel in the control room, the operator maintains constant vigilance of the meters and signals that show conditions throughout the plant. Figure 5-50 shows the general arrangement of the panels. The most important position is the reactor console. From this point is maintained a constant control over reactor operation and steam system performance essential to reactor operation. All plant and reactor alarm or annunciation is located here and from this position the operator may keep constant watch over the primary measurement of reactor steam pressure and water level by means of the closed-circuit television system.

For safe, steady operation, many of the variables are equipped with automatic controls but adjustments for the numerous instruments and controls may be made manually at the reactor console. In the process of starting up the reactor, a complicated interlock system requires that 15 conditions must be safe before the reactor can be started up. Similarly, if the limit set on any one of these conditions is exceeded when the reactor is operating, it will be automatically shut down. When all 15 conditions are in order, the operator has full command of the control rod operation and can control the reactor power within the limits of the safety interlock system.

Adjustments to the remainder of the panel board instrumentation are made at the panel for the particular equipment involved. This includes the metering, recording, and the control stations for the electrical system switching and detailed function of the steam system.

Components of the steam plant. The general features of the steam plant have been described under "Operational Characteristics" earlier in this chapter, and shown schematically in the flow diagram, Fig. 5-3. It involves the reactor itself, the dryer-cooler, the turbo-generator and condenser together with the steam bypass system, the feed pumps, the vapor recovery system, filters, feedwater regulator, the startup heater and the feedwater purification system.¹⁵

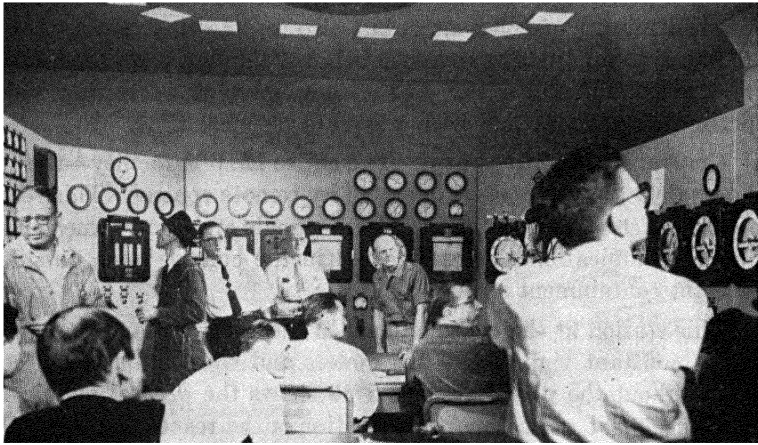


FIG. 5-51. A tense moment during the approach to full-power generation.

The steam dryer-emergency cooler, shown in Fig. 5-52, has as its prime function drying of the steam leaving the reactor as well as minimizing radioactive carryover. Steam flow through this unit will be 70,000 lb/hr at 600 psig for normal full load, with the steam effluent having less than 0.1 per cent moisture by weight and a maximum solids content of 0.001 ppm. The pressure drop through the vessel, from inlet to outlet nozzles, will not be more than 5 psi.

During an emergency, this unit can provide shutdown cooling of the reactor decay heat. This condition can also be brought about by a loss of instrument air or power. Normal shutdown cooling is provided by the pumps and coolers in the Reactor Purification System or by the main condenser or both. When this emergency condition arises, an air-operated valve with a solenoid pilot will automatically open the 2-inch line between the emergency cooler and the 15,000 gal overhead water storage tank as previously described. The cooler is capable of condensing 4530 lb of steam per hour at 600 psig.

The drying section of the unit consists of a large number of parallel corrugated plates (scrubbers) which entrain most of the moisture and particulate matter. The shell is made of stainless steel and all internals and nozzles are also stainless steel. Short carbon steel transition lengths are welded to the steam outlet, cooling water inlet and outlet nozzles. In addition both coolant nozzles have thermal sleeves. The steam piping from the dryer to the turbine and condenser is carbon steel, type A-106, Grade B, plates on the inside with 0.005 in. of nickel. Between the reactor vessel and the dryer, 6-in., schedule-40, type-304 piping is used.

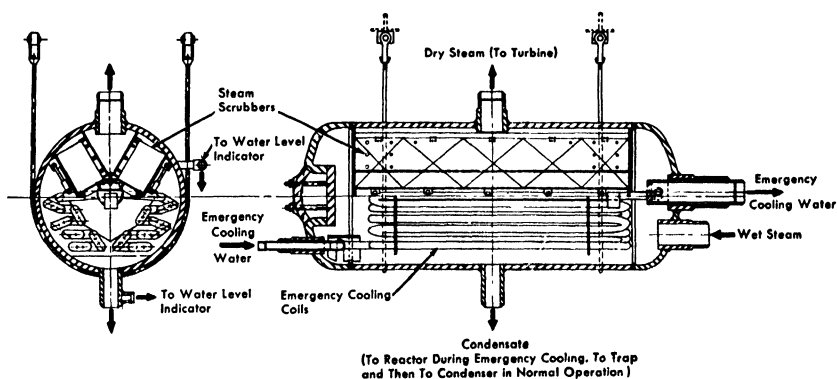


FIG. 5-52. General arrangement of steam dryer emergency cooler.

The turbine-generator unit comprises a 5000-kw, multistage turbine direct-connected to a 6250-kva, 3-phase, 60-cycle generator with direct-connected exciter. The turbine is designed for a throttle pressure of 560 psig and a temperature of 482 F. Features of this turbine that differ from a standard type turbine of the same rating are the sealing system, the special moisture-removal deflectors, and the Kanigen plating.

A labyrinth seal at each end of the casing, through which the turbine shaft extends, prevents (1) the escape of steam into the room and (2) air in-leakage. These seals consist of a series of spring-backed baffle rings which form a series of radial restrictions between the shaft and casing. In addition to conventional seals, the turbine shaft, the governing-valve lift rods, and the trip-throttle-valve stem are provided with a special sealing arrangement. It consists of a dry air barrier, a vacuum leak-off, and a steam seal.

To reduce erosion in the exhaust rows of the turbine due to the high moisture content, the last three rows incorporate special moisture deflecting lips on the diaphragm faces of the last two rows and interstage drains. The specially formed lips protect the casing from erosion and also conduct the trapped moisture down to the various interstage drains. In addition, the leading edge of each bucket in the last row is protected with a layer of stellite.

Laboratory findings on the BORAX-III reactor indicated a high water dissociation rate on the order of 75 cc of gas per liter of condensate at 300 psig. (EBWR operating at 600 psig generates 55 cc of gas per liter of condensate.) In order to check possible excessive corrosion anticipated in EBWR due to BORAX-III findings, the turbine and the 6-in. main steam piping were Kanigen processed. This process deposits a nickel and nickel

phosphide coating of a fairly uniform thickness over the prepared surfaces; a thickness of 0.005 in. was placed on these components.

Components of the turbine which were plated included the upper and lower halves of the turbine casing, the trip throttle valve, all the diaphragms, the Curtis admission shield, and the high- and low-pressure gland seal. Surfaces which were not to be plated were masked with a special compound. No effort was made to coat the turbine rotor assembly because it contained (1) too many dissimilar metals and (2) voids between blade roots and between blade tops and bands, neither of these being conducive to good plating.

The main condenser is suspended and welded to the turbine exhaust casing to conform to an all welded system. In view of the seriousness of a circulating-water leak into the steam cycle, double tube sheets were used to secure each end of the condenser tubes. This double tube sheet space is maintained at condenser vacuum, and to date there has been no leakage. Aluminum has been used for the condenser tubes with the idea of keeping the radioactivity level in the reactor circuit low. Tubes are rolled in the inner tube sheets as well as rolled and flared in the outer tube sheets.

Differential expansion between tubes and shell is compensated for in the expansion joint built into the condenser shell. Regions in the condenser where the tubes are directly exposed to water particles from the turbine exhaust are protected by stainless-steel shields snapped on over the vulnerable sections of the tubing.

The condenser hot well and the water boxes have been longitudinally divided to permit plant operation even in the event of a tube rupture. Increased conductivity in the condensate manifold would warn the operator of a tube failure, and the operator could isolate that half of the condenser that developed the leak.

A desuperheater built into the condenser is capable of passing full reactor steam flow without endangering the condenser, with normal circulating-water flow available. However, if the circulating-water pumps should fail, the two sprays located in the desuperheater pipe are placed in operation to prevent condenser damage.

Circulating-water piping between the condenser and the cooling tower is so arranged that, if a power failure should occur, natural circulation of the water would take place to provide sufficient cooling.

Reactor feed pumps. To eliminate pump shaft seals wherever possible, the usual condensate pumps have been incorporated in the feedwater pump design. The first stage of each pump serves as the condensate pump; it also supplies liquid to the hydraulic piston type bearing in the suction piece of the pump and to the mechanical seal.

Special seals at the upper end of the pump shaft and at the flanged joints of the outer tank assembly are designed to limit the loss of fluid to the atmosphere and prevent contamination of the reactor feedwater with atmospheric moisture.

The pump shaft seal consists of three separate sections: a labyrinth type seal, a rotary mechanical seal, and a serrated bushing. An annulus located below the labyrinth seal is connected to the vent condenser vacuum, whereas the annulus above the bushing is connected to the first-stage discharge pressure (approximately 100 psig). Between the bushing and the labyrinth seal is a rotary mechanical seal consisting of a stellite surface (spring loaded and water lubricated) operated against a carbon-ring face. Dry air at 6 in. water pressure is supplied to the center of the labyrinth seal, and a vacuum of 6 in. water pressure is applied between the labyrinth and the rotary seal and is piped to the liquid recovery system.

Feedwater bled from the first stage of the operating pump is fed to the mechanical seal chamber of both pumps so that atmospheric air cannot be drawn through the seal of the nonoperating pump causing contamination and loss of suction on the operating unit.

The air ejectors for EBWR are of the twin, two-stage, condensing type with a capacity of 75 lb/hr each. Interiors of the cooler shells, tube sheets, etc., were plated with 0.003 in. of nickel to reduce the amount of corrosion within the inter and after coolers. The amount of noncondensable gases from air leakage is $\frac{1}{2}$ cfm. In addition the air ejectors must also evacuate the hydrogen and oxygen evolved from water dissociation in the reactor, and this amounts to 63 lb/day at 20 Mw. Aside from the nickel plating there is nothing special about the air ejectors.

All the condensate leaving the condenser is filtered before being pumped back into the reactor. Two "Fulflo" filters of 180 gpm capacity (each) are installed. These are in parallel but only one unit will be in operation at a time. The filters are not shielded inasmuch as calculations indicated that a very small amount of particulate matter would find its way into the primary steam system. To date the maximum level of activity found on the filters has not exceeded 10 mr per hr.

Removal of the "hot" filter cartridges is facilitated by having the cartridges in a removable cage assembly. The entire assembly can be lifted from the vessel and placed in a container for removal to either the cave building for replacement of filter cartridges or to the burial ground if activity level is high.

The general features of the steam bypass control system have been covered earlier in this chapter. The primary function of this system is to maintain steam pressure in the main steam line at any selected value between 100 and 650 psig. Since the principal cause of pressure varia-

tions is a change in steam flow to the turbine (resulting from generator load fluctuations), the bypass control system is essentially a steam dumping device. It is designed to maintain a constant rate of flow from the reactor until such a time as the power level of the reactor can be adjusted to a new value corresponding to the new load. In order to maintain constant steam pressure, steam not required by the turbine is piped to the main condenser desuperheater.

The reactor startup heater is used to preheat the reactor water to a temperature of 325 F prior to the withdrawal of the control rods. The heater consists of a heat exchanger with a heat transfer area of 84.6 ft². Laboratory steam at 190 psi (378 F) is used as a source of heat in this heater. Taking into account the weight of metal in the reactor vessel, shielding structural core, etc., and the weight of the water in the reactor, it was calculated that the total heat required to bring the reactor water up to temperature was 40,600 Btu/F. The approximate time required to heat from room temperature to 212 F is 1.4 hr; from 212 to 325 F, 1.9 hr, or a total elapsed time of approximately 3.3 hr.

While the reactor is in operation the Laboratory supply steam and condensate return lines to the cooling tower are isolated from the startup heater to eliminate any possibility of radioactive contamination outside the gastight building in case of a leak in the heating unit.

Feedwater purification. Two mixed-bed ion exchange units (10 gpm, 6000-gal capacity each) supply demineralized water (not deaerated) having a specific resistance of 5 to 6 megohms (0.2 to 0.16 μ mhos conductance) and a pH of 7.0. The total solids will be below 0.2 ppm. The effluent pipes each contain a "Fulflo" filter unit to protect against carryover of resin fines or extraneous material into the reactor water system. The conductivity of the water from the operating unit is recorded in the control room in order to determine when to switch to the reserve unit.

The ion exchange units are located in the service building basement. Both units are of conventional construction, having $\frac{1}{8}$ -in. rubber-lined, $\frac{3}{16}$ -in. steel wall tanks (24 in. OD $\times$ 9 ft straight side). The effluent will flow through a filter unit (cotton-thread cartridges in a nickel-plated brass body) to the 200-gal, type-304 stainless steel storage tank (36 in. OD by 6 ft high) located on the turbine floor of the reactor building. Low and high water level in the storage tank will be indicated in the control room. Because of the expected low makeup water requirement, the 200-gal storage tank will be ample. All demineralized water piping is stainless steel.

Safety systems. The reactor power plant is designed for fail-safe operation. The reactor will scram on loss of power. The turbine trip

valve and the bypass valve would close to prevent blowdown of the reactor.¹⁸ The only loss of reactor water would be to the steam jet air ejectors which would lower the water level at a rate of 2 in/hr. The ejectors would be shut down manually if the power failure were to be of long duration, forcing a complete plant shutdown.

An emergency motor generator set in the service building will supply sufficient power (60 kw) to operate the following equipment:

1. Control room instruments.
2. Instrument air compressor.
3. Cooling-water pumps.
4. Recovery system.
5. Turbine-oil pump.
6. Shield-cooling pump.
7. Motor-operated valves.

The solenoid-operated valves, valve position indicators, and the controls for the feedwater pumps and circulating-water pump can be operated from a battery-powered DC bus in the event of failure of the AC power line.

In most cases the essential pumps are provided in duplicate, one operating unit, the other a standby. In the event of failure of one pump, warning devices notify the operators so that the standby pump can be impressed into service. In some cases if the operator did not act in time the reactor would be automatically scrammed. For example, if the operating feedwater pump were to fail, the introduction of feedwater would be discontinued and the reactor power level would decrease due to increasing voids in the core. However, the reactor water would continue to be evaporated and would accumulate in the condenser hotwell. At 60,000 lb/hr steam flow, the reactor water level would lower at the rate of 6.25 in/min. The condenser hotwell level would rise at the rate of 3.75 in/min. The flow warning alarm would indicate a malfunction and the operator at this time would have several minutes to determine the cause of the loss of flow before the reactor automatically scrammed.

In the case of the condenser circulating water pumps, which are both normally in use, the failure of one pump would result in a slight increase in back pressure with a correspondingly higher temperature in the hotwell. Introduction of water at a higher temperature into the reactor by the feedwater pump would result in more steam in the core per unit of power, and the power level would drop. If both pumps failed, there would be a complete loss of vacuum in the condenser. As the back pressure increased, the operator would be alerted by an alarm signal in the control room. In the event this circuit failed to function, the reactor would scram at a back pressure of 20 in. Hg Abs.

All instruments which require compressed air for operation are designed to fail-safe either open or closed, as the system dictates. Instruments which are capable of inserting the control rods upon receiving proper signal will also insert the rods when the instruments themselves are not functioning properly. Similarly, alarm circuits are designed to give alarms when malfunctioning of the circuit develops.

The steam pressure control system fails closed to prevent excessive drop in pressure or loss of reactor water. Should the reactor pressure increase to 625 psig, the steam-powered regulating valve would begin to dump the excess steam to the condenser through the desuperheater. The regulating valve would be full open at a pressure of 680 psig with a capacity of approximately 80,000 lb/hr. The reactor would be scrammed automatically upon opening of the regulating valve.

With nine safety rods, the combined strength at operating conditions is greater than the maximum amount of k to be controlled; failure of 1 or even 2 safety rods will not create a dangerous condition. The reactor will never be operated with less than the number of safety rods required to control all available excess k , plus 2 rods. In emergency the rods may be augmented by the borated water injection system.

Safeguards are also provided against excessive changes in power demand by the electrical system, high water level and low water level in the reactor, water leaks in the external steam system, excessive pressure in the reactor, rupture of the steam piping, etc.

Of particular interest are the boric acid injection systems, two of which are installed: one, a low-pressure system, and the other a high-pressure system. The low-pressure system is provided for the automatic spraying of water on the core to remove decay heat from the fuel elements in the event of simultaneous low pressure (20 psig) and low water level (6 in. above core top) in the reactor vessel. In order to remove the remote possibility that the water addition to the core would cause criticality, provisions have been made to add boric acid to the spray water. A continuous boric acid dissolution method was selected in order to eliminate the necessity of storing a large supply of boric acid solution. The required concentration of boric acid, established to overcome $12\frac{1}{2}$ per cent reactivity, is 2.0 g of H_3BO_3 per liter of water.

The boric acid solution will be injected at the rate of 26 gpm into the reactor vessel through a spray ring approximately 6 ft above the top of the reactor core. There are 24 stainless steel nozzles preset to give proper distribution of sprayed water over the fuel elements.

The boric acid crystal storage tank is located in the basement of the service building. The tank is constructed from 12-in., stainless steel pipe, 10 ft long with pipe caps on either end. There are 8 equally spaced nozzles to spray Laboratory water over the boric acid crystal bed.

The high-pressure boric acid injection system is a standby emergency system to reduce the reactor core reactivity below criticality. The boric acid solution is injected if, after a scram signal, 1 pair of each of 4 pairs of control rods (1 and 2, 3 and 4, 5 and 6, 7 and 8) do not fully enter the core. A 2-second delay is allowed for the complete insertion of control rods after a scram signal. The amount of boric acid solution injected is sufficient to overcome $12\frac{1}{2}$ per cent reactivity. This amounts to 49.5 lb of H_3BO_3 for the EBWR core.

The high-pressure boric acid tank location is inside the reinforced concrete missile shield on the main floor. The injection is through the feed water line. The connecting lines are $2\frac{1}{2}$ -in. stainless pipe (approximately 75 ft long). There are two $2\frac{1}{2}$ -in. solenoid trip valves and a check valve in the line at the exit of the high-pressure tank. The trip valves may be actuated automatically or manually by a mechanical trip lever located inside the service building. The equivalent length of straight pipe to account for pressure drops in the valves, elbows, etc., is approximately 225 ft.

In order to inject the solution into the pressurized reactor, the high-pressure tank has an air volume above the liquid maintained at 1600 psig by a compressor. During the injection this air will expand as the liquid level drops, reducing the over-all head pressure.

Air drying and fluid recovery system. The air drying and fluid recovery system shown schematically in Fig. 5-53 is designed to (1) minimize leakage and to recover most outleakage of radioactive gases and vapors and return this leakage to the steam system; and (2) minimize contamination and loss of D_2O during the future use of heavy water at the EBWR working fluid. The system consists of two sections, one for drying atmospheric makeup air and the other for separating and recovering the working fluid and sealing air.¹⁵

The components to be sealed are listed in Table 5-8.

The turbine is equipped with a shaft seal which minimizes leakage of radioactive steam to the surrounding atmosphere. The seal consists of a labyrinth containing three annuli surrounding the shaft at the high-pressure and low-pressure ends. The outer annulus will be supplied with dry air at +6 in. H_2O pressure; the inner will be supplied with system steam at 2 psig, and the center annulus will exhaust to a vacuum of -6 in. H_2O which will collect leakage from the adjoining grooves and return it to the recovery system. A small recess cut circumferentially into the upper horizontal joint flange face is evacuated by the recovery system (-6 in. H_2O) to collect leakage. The interior of the housing flange is held at +6 in. H_2O pressure as in the case of the shaft seals.

1. Air & Possible Vapor From Horizontal Turbine Casing Seal

2. Air & Vapor From Turbine Seals & Pressure Control Valve

3. Air & Vapor From Air Ejector After-Cooler

4. Air From Reactor Feed Pumps

5. Condensate From Air Ejector After-Cooler

6. Water From Control Rod Thimble Seals

7. Water From Reactor Feed Pumps

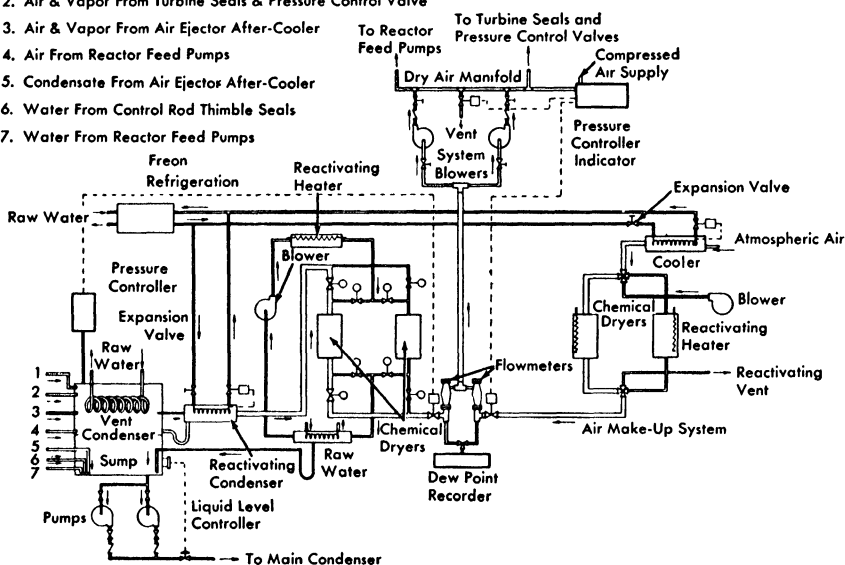


FIG. 5-53. Air drying and fluid recovery system.

In addition to a mechanical seal, the reactor feed pumps are equipped with a labyrinth fluid recovery seal.

With the exception of the few special valves, the steam-powered regulating valve, the turbine throttle valve, and the steam bypass valve, all valves in the steam power plant are of conventional globe or gate design, with welded or pressure sealed bonnets slightly modified for deep packing glands, lantern rings and bleedoff opening. Teflon or teflon-impregnated asbestos packing is used in most of the valves. The bleedoff openings will not be used unless excessive leakage dictates the necessity for collection or an alarm system.

The system will deliver air at a controlled pressure of 6 in. H_2O above atmosphere. It will take an air-vapor mixture from the seals at a controlled vacuum of 6 in. H_2O below atmosphere. These pressures are controlled automatically. When the total air at the suction of the working recovery section exceeds the amount of air necessary to maintain the set system pressure, an automatic relief valve will prevent overpressure.

A vent condenser consisting of a water-cooled primary section is capable of condensing and cooling at 250 F water vapor and air mixture (150 lb/hr of vapor and 150 lb/hr of air). A secondary section cooled by mechanical refrigeration further cools and dehumidifies to a saturated

TABLE 5-8
ESTIMATED GLAND SEAL FLOWS

Item	Air required to seal, lb/hr	Air to recovery system, lb/hr	Steam to recovery system, lb/hr
IIP turbine gland	36.4	18 2	32.7
LP turbine gland	36.4	18 2	32.7
Turbine lift rods (2)	1.2	1 0	2 6
Reactor feed pumps (2)	30.0	15 0	
Pressure control valves:			
Bypass valve	0 6	0 4	0 8
Steam-powered regulating valve	0 5	0 4	0 7
Throttle valve	1 6	1 0	7 0
Total	106 7	54 2	76 5

temperature of 40 F. The vent condenser is equipped with a system sump capacity of 50 gal to which all condensate from the "fluid recovery" portion of the system will drain.

A sump will return the condensate to the main condenser during startup and shutdown, at which time the main condenser is not operating under vacuum conditions. During normal operation the sump pump can be turned off in order to take advantage of the free flow of the working fluid due to the pressure differential between the sump tank and the main condenser. A selector switch starts either of the two sump pumps in the system. The normal level in the sump tank is controlled by a Fisher Governor, Level-Trol column type liquid level controller. A high level Magnetrol float switch will engage both sump pumps should the level rise above normal in the event of failure of the operating pump.

Lectrodryers (chemical air dryers) are able to perform the following:

Makeup air section. Drying 150 lb/hr of 40 F saturated atmospheric air to an average moisture content of less than 0.5 grain of water vapor per pound of dry air on a continuous basis. This unit is capable of handling 150 lb/hr of 90 F saturated air for several hours.

Fluid recovery section. Drying a 40 F saturated vapor and air mixture of 250 lb/hr to an average moisture content of less than 0.5 grain of water vapor per pound of dry air on a continuous basis.

During normal operations it is estimated that the flow rate will be 55 lb/hr. During hogging the unit will handle 150 lb/hr. This unit is capable of handling 250 lb/hr of saturated air at 90 F for several hours.

Reactor shield cooling. The reactor shield cooling coils (most coils are $\frac{1}{2}$ -in. ID copper tubing) are embedded in the lead bricks surrounding the inner shield tank. The horizontal loops are spaced 12 in. apart. The cooling coils are comprised of 13 circuits, with 13 alternate or spare circuits installed adjacent to the main loops.

The amount of heat which must be removed from the shielding has been calculated to be approximately 30 kw or 100,000 Btu/hr using D_2O as the working fluid. Water flow to remove this heat with a 10 degree temperature rise is approximately 20 gpm.

The cooling circuit is a closed water loop with a pump, heat exchanger, and expansion tank. Two filters are provided, one on the discharge of each pump to ensure that no particulate matter will circulate in the shield cooling circuit. Demineralized water is used in the circuit.

Condenser circulating water and cooling tower. The circulating water system serves to condense the low-pressure steam from the 5000-kw turbine and supply cooling water for other auxiliary systems.

The system has a capacity of 113,000 gallons and circulates at 14,250 gpm. Of this 14,250 gpm, 13,650 gpm flows through the condenser and 600 gpm is used in the auxiliary systems. The condenser water is pumped through the system by 2 pumps each rated at 6825 gpm at a 50-ft head. One 600-gpm circulating-water pump is used to pump water to the auxiliaries while a second pump is used as a standby unit.

Heat removed by the circulating water is dissipated to the atmosphere by a Marley double-flow cooling tower. The tower is designed to cool the water in the summer months from 95 to 85 F. The cooling tower basin has a capacity of 100,000 gallons.

Laboratory water is used in the system. However, the water is continuously treated with sulfuric acid to maintain a pH value of 6.5, and a polyphosphate compound to maintain a phosphate residual of 30 ppm. This treatment is used to reduce the scaling tendency of the concentrated water and acts as a corrosion inhibitor by formation of a molecular iron phosphate film on iron surfaces.

The EBWR electrical system. The electrical system of EBWR is not essentially different from that of a conventional plant of 5000-kw capacity. A simplified line diagram of the system is shown in Fig. 5-7. The output leads of the generator are connected to the 4160-volt, 3-phase,

60-cycle bus of the outdoor unit substation switchgear through a 1200 amp, air circuit breaker. The 4160 side of the 5000/6250-kva main power transformer is connected to the 4160-volt bus of the same outdoor unit substation, also through a 1200 amp breaker. The 13.2-kv side of the main transformer is connected to the north bus section of the Laboratory Central Switching Station, Facility 544, by means of the gang-operated disconnect switches in the EBWR outdoor substation, the overhead 13.2-kv EBWR pole line, and an oil circuit breaker in the north bus section of Facility 544. A 1200-amp air circuit breaker is included in the EBWR unit substation to feed the 1000-kva main auxiliary power transformer.

As explained earlier in this chapter, the EBWR power plant is arranged for parallel operation with the Utility Power System. When the plant is shut down, power for the Laboratory is supplied by the Utility System. To provide emergency power supply for the reactor control rod drives and such auxiliaries as the reactivity blowers and heaters, the make-up air Freon compressor, recovery system Freon compressor, cooling water pumps, shield cooling pumps, etc., an automatic starting diesel-generating set (60 kw, 480-volt, 3-phase, 60-cycle) is incorporated in the design of the system. The control is arranged for the initiation of automatic starting of the diesel-generator set following loss of voltage on Motor Control Center No. 2 after the time required for the restoration of voltage on the 480-volt switchgear bus. The automatic starting of the diesel generating set begins after approximately four seconds of sustained loss of voltage, and restores voltage to Motor Control Centers Nos. 2 and 3 within approximately 15 sec.

REFERENCES

1. The EBWR—Experimental Boiling Water Reactor, ANL-5607, May, 1957.
2. J. M. HARRER, EBWR Design, TID-7506, Part I, p. 67, November, 1955.
3. J. M. HARRER, EBWR—General Description, TID-7535, p. 1, May, 1957.
4. EBWR Operating Experience, *Nucleonics*, **15**(7), 53-64 (July, 1957).
5. J. M. WEST, J. R. DIETRICH, A. S. JAMESON, G. A. ANDERSON, J. M. HARRER, and H. F. BRUSH, Hazard Summary Report on the Experimental Boiling Water Reactor (EBWR), ANL-5781, November, 1957.
6. L. E. LINK, Economic Aspects of EBWR, TID-7535, p. 12, May, 1957.
7. R. E. MACHEREY, C. H. BEAN, N. J. CARSON, JR., and J. R. LINDGREN, Manufacture of Fuel Plates for the Experimental Boiling Water Reactor, ANL-5629, June, 1954.
8. C. H. BEAN, R. E. MACHEREY, and J. R. LINDGREN, Roll Cladding Uranium-Zirconium and Uranium-Zirconium-Niobium Alloys with Zircaloy-2 for Plate Type Fuel Elements, ANL-5628, February, 1958.
9. N. BALAI, EBWR Pressure-vessel Design Rules Reviewed, Installation History, and Preliminary Performance, TID-7535, p. 18, May, 1957.
10. M. GROTENHUIS and J. W. BUTLER, Experimental Boiling Water Reactor (EBWR) Shield Design, ANL-5544, August, 1956.
11. C. K. SOPPET, Design and Fabrication of the EBWR Core Structure and Fuel Handling System, TID-7535, p. 27, May, 1957.
12. W. J. KANN, EBWR Control-rod Drive Mechanism, TID-7535, p. 50, May, 1957.
13. WALTER C. LIPINSKI, Instrumentation of the EBWR Power Plant, TID-7535, p. 69, May, 1957.
14. JAMES A. DESHONG, JR., EBWR Power Controls, TID-7535, p. 54, May, 1957.
15. E. A. WIMUNC, EBWR Plant Equipment, TID-7535, p. 81, May, 1957.
16. R. E. BAILEY, EBWR Water Treatment, Decomposition, and Decontamination Factor, TID-7535, p. 98, May, 1957.
17. J. A. DESHONG, JR., and E. S. BECKJORD, Frequency Response Measurements of the EBWR Automatic Steam By-pass Valve Control System, ANL-5726, June, 1957.
18. S. A. BERNSEN, Safety Provisions for the EBWR, TID-7535, p. 103, May, 1957.

CHAPTER 6

EBWR OPERATING EXPERIENCE

As mentioned in Chapter 5, the designers of the EBWR followed the rather unorthodox procedure of doing critical experiments in the reactor itself after it had been built. The core design was based entirely on the results of two-group neutron-diffusion calculations. The fuel comprised two different thicknesses and two enrichments, and it was fabricated to provide enough flexibility to cover the estimated inaccuracies in the physics calculations.

Although this procedure of performing the critical experiments in the reactor itself was successful, the designers feel that they would not follow it again since it has some serious disadvantages. First, since the critical experiments tie up a very expensive facility, there is the pressure of time. Only three weeks were allocated for these investigations in the EBWR. During this time the investigators worked in double shifts but still did not obtain as much information as they needed. Second, the extra investment in fuel required to cover the inaccuracies in the physics calculations would help to pay for more precise determinations by ordinary critical experiments. Plans are to back up the future heavy-water core design with standard kinds of critical experiments before it is built.

PREPOWER GENERATION PHYSICS EXPERIMENTS

EBWR construction was completed during October and November of 1956. Reactor criticality was first achieved on December 1 of that year, and the following three weeks were spent in carrying out low-power experiments.¹

Since critical experiments involving slightly enriched uranium plates and light water moderator had not been run, because the pressurized full-power loading was to be determined, a considerable effort was devoted to physics experiments. The phases involved in the program were as follows:

- (A) Approach to criticality.
- (B) Addition of reactivity by expanding the core size and trading elements.
- (C) Measuring void coefficients.
- (D) Measuring worth of boric acid solutions.

- (E) Adjustment of the load to that needed for pressurized operation.
- (F) Measuring the temperature coefficient.

Control rod calibrations were required in virtually all phases.

As explained in greater detail below, in the first approach to criticality the core was loaded with a uniform distribution of the four different types of fuel elements. Critical mass was reached between 81 and 87 elements. The 81-element loading had a very high multiplication, probably greater than 10,000. The next loading of 87 elements completed a symmetrical pattern; exact criticality, with no rods in the reactor, was not measured.

After the first critical loading, it was evident that more enrichment would be needed, and as the loading was increased to 112 assemblies the enriched-to-natural ratio was made higher. Rods were calibrated, keeping the nine rods in a bank. Position accuracy and repeatability was 0.01 in. After a 112-element loading was reached, enrichment was increased by exchanging enriched for natural elements, until ~ 6 per cent reactivity above cold critical was available.

A boiling water reactor requires considerable excess reactivity, i.e., additional fuel to compensate for changes in reactivity occasioned by boiling voids, temperature coefficient, and xenon buildup. The importance of these effects in EBWR led to a decision that a way should be found to measure the reactivity required by the first two effects in the course of the cold critical experiments. The xenon buildup, of course, could only be measured at full power and therefore was not included. In the case of the temperature coefficient, the reactivity change is largely due to the variation in moderator density with temperature. (Doppler, fuel expansion effects, increased neutron energies, changed cross sections, etc., also enter into reactivity.) For the boiling condition, the effective moderator density is inversely proportional to the amount of steam present. Therefore, a dimensionally stable method had to be found to vary the moderator density in the fuel assembly water channel in a way that conformed to the temperature and boiling characteristics of the reactor.

Several methods were considered; among these were: bubbling of gas through the water channels, controlled introduction of gas into water-filled tubes inserted in the fuel assembly water channels, and, finally, the use of some low-density expanded solid. The first method was inexpensive, but had the major disadvantage of poor control, which constitutes not only a hazard to reactor operation, but also makes it difficult, if not impossible, to distribute the voids in the proper way. The second method overcomes the objections to the first and would probably lead to high experimental accuracy. However, the fabrication of a tube arrangement for each water channel becomes expensive if 560 such channels must be filled.

The third method, that of using a low-density expanded solid, is really a variation of method two, with no means for circulating the gas within the solid. It also appeared to be an expensive method, because of fabrication problems. However, expanded polystyrene or Styrofoam had been used for float controls and its use was suggested as a possibility. The problems and costs of fabricating this material were investigated and found to be reasonable, and it was selected for the experiment. Further details concerning the use of this material will be found on page 318.

A. Approach to criticality. Four types of fuel assemblies were available, as shown in Table 6-1. Any given fuel assembly contains six identical plates in a box occupying a 48-in. (fuel height) by 4 by 4-in. cell. Thus the various configurations of these fuel assemblies can be used to: (1) change the critical size of the core; (2) change the power distribution in the core; or (3) change the amount of reactivity corresponding to a given volume of steam in the core.

Nineteen subcritical loadings were made, ranging from loading No. 1 (3T + 3H + 3ET + 2EH) to loading No. 19 (14T + 14H + 25ET + 28EH). The procedure for each loading was:

1. Slowly insert one element at a time into the water-filled core, observing counting rates and ion chamber current; control rods are partially withdrawn at this time.

TABLE 6-1
FUEL ELEMENT STRUCTURE

Type	Thin natural	Thick natural	Thin enriched	Thick enriched
Symbol	T	H	ET	EH
Enrichment of U^{235} , %	0.72	0.72	1.44	1.44
Meat thickness, in.	0.174	0.239	0.174	0.239
Plate thickness, in.	0.214	0.279	0.214	0.279
Average water channel thickness, in.	0.453	0.388	0.453	0.388
$\left[\frac{H_2O \text{ volume}}{(U^{235} + U^{238}) \text{ volume}} \right]^*$	4.42	2.76	4.42	2.76
Nominal grams of U^{235} per 6-plate element	291.1	413.0	582.2	826.0
Average grams of U^{235} per 6-plate element found in loading No. 19	293.5	401.3	585.4	813.1

* Volume ratio is a reactor average and includes water in control rod channels.

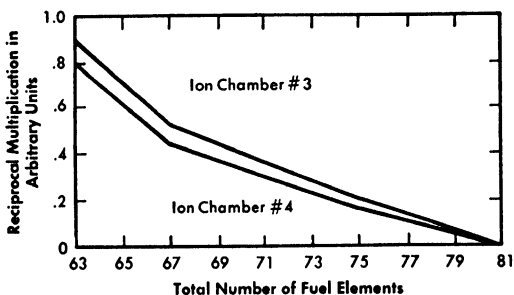


FIG. 6-1. Approach to criticality.

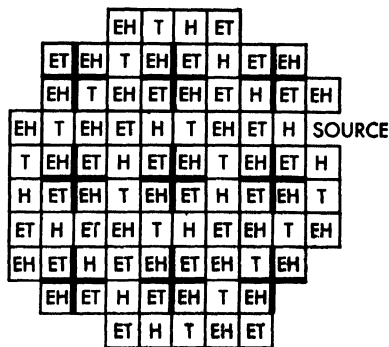
2. Take several counting rates at the completion of the loading with all control rods out.
3. Move the neutron source and counter locations when necessary and obtain counting rates in the new positions. The neutron source was a strong antimony-beryllium source irradiated in another reactor. It was installed in a dummy element positioned near the center.
4. Plot the reciprocal counting rate or current against the number of elements, and decide the next loading from this curve.

All loadings were designed to approximate a uniform mixture of the elements in the proportion 18T:18H:38ET:38EH, mainly because calculations existed using these numbers.

Figure 6-1 shows a curve obtained for the last few loadings. The curvature existing when subcritical by many elements is in the expected direction because of the decrease in reactivity worth of one element as the reactor radius increases.

Figure 6-2 shows the last subcritical loading. With all control rods out, the reactor approached its equilibrium multiplication with a time constant of approximately 270 sec, which indicated a reactivity of -0.03 per cent, essentially a critical reactor.

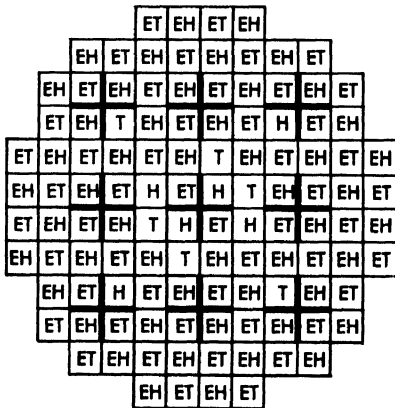
A comparison of the loading (14T + 14H + 25ET + 28EH) with that previously predicted to be critical by 2-group theory (6.83T + 6.83H + 14.43ET + 14.43EH) gives a measure of the error in the theory to a reactivity basis. The difference in the number of elements corresponds to 3.78 per cent, while the difference in ratios of elements corresponds to -0.42 per cent. Hence, a 2-group theory *overestimated* the reactivity by 3.39 per cent. Since this is within a previously estimated root-mean-square error of the theory (± 3.67 per cent) based on cross-section errors, etc., the agreement between theory and experiment is satisfactory. (The criterion for choosing the enrichment and numbers of element was that the theory could have been optimistic by twice its root-mean-square error,



Composition: 28 Thick Enriched, EH
 25 Thin Enriched, ET
 14 Thick Natural, H
 14 Thin Natural, T
 81

Total Wt. of U^{235} , 47.13 kg

FIG. 6-2. Last subcritical loading (No. 19).

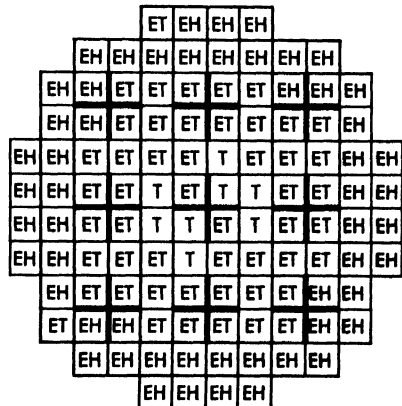


Composition: 50 Thick Enriched, EH
 50 Thick Enriched, ET
 6 Thick Natural, H
 6 Thin Natural, T

112

Total Wt. of U^{235} , 74.1 kg

FIG. 6-3. Core loading No. 26.



Composition: 54 Thick Enriched, EH
 50 Thin Enriched, ET
 8 Thin Natural, T

112

Total Wt. of U^{235} , 75.5 kg

FIG. 6-4. Core loading No. 33.

and still sufficient reactivity would have been available.) If the same 2-group methods used on EBWR are applied to a lattice of 0.06-in. diameter, 1.143 per cent enriched uranium rods with a water to uranium volume ratio of 4:1, the theory underestimates the reactivity by 1.33 per cent.

B. Reactivity addition. Core loading No. 26, shown in Fig. 6-3, with 112 elements, was built up in several steps from loading No. 19. Measurement of the void coefficient at this stage indicated a negative value sufficient to permit a shift of the thin elements to the center. This is desirable from the standpoint of power generation, in which a low void coefficient (and, possibly, the absence of a thick-thin hot spot factor) may aid maximum power. The resultant loading (No. 33) is shown in Fig. 6-4.

As pointed out previously, all critical measurements and rod calibrations were done with all nine control rods level. This is the simplest configuration and also simplifies the calibration procedure.

C. Void coefficient measurement. Void coefficient measurements were performed by inserting perforated Styrofoam sheets in the water channels of the fuel elements in core loadings No. 26 and No. 33.²

Styrofoam is a white, multicellular structure with a density of about 0.03 g/cm³. Measurements of the rate of water absorption for the material gave values far greater than that for polystyrene, which is very low. This is explained by the presence of ruptured cells at or near the surface which slowly fill with water. As shown in Fig. 6-5, the rate of absorption of water is fairly rapid for the first two days, slowing up considerably at the end of that time. Also, the initial value of the displacement fraction is less than one, because many of the surface cells fill with water immediately. The original or starting value of displacement fraction is a function of the ratio of volume to surface area, being larger as the ratio is increased.

The mechanical properties of the material in the strip form were found suitable for strips greater than about 1/4 in. thick. For smaller thicknesses, the loss in rigidity, as well as strength, leads to considerable breakage. The machinability by sawing or sanding is good, although care must be used to avoid crushing. It should be pointed out that the melting point limits the usefulness of this material to low-temperature experiments. The low-melting point characteristic was used to good advantage in the manufacturing operations which will be described later.

The nuclear properties of Styrofoam were investigated, since the void material would occupy a region of high importance in the reactor (fuel assembly water channels). A spectrochemical analysis was made to

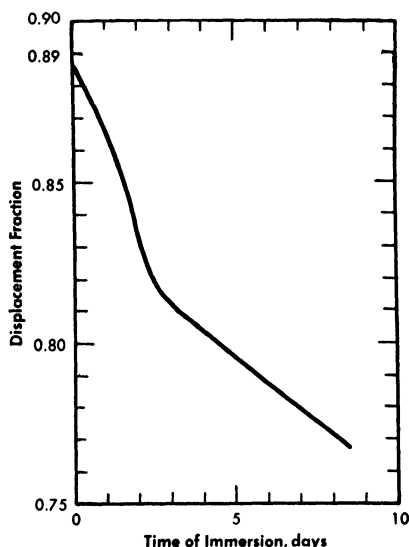


Fig. 6-5. Displacement fraction for Styrofoam strips.

determine if any materials with objectionable nuclear cross sections were present. The results were negative. The nuclear purity of the material was further verified by examining its effect on the multiplication of a subcritical Argonaut assembly. Irradiation of the material in the Argonne Research Reactor CP-5 for 15 min at a flux of $\sim 2 \times 10^{12}$ n/(cm²)(sec) produced no significant change in the water absorption or mechanical properties, and only short-lived activation occurred.

The void volume calculations were done in two parts. The first dealt with temperature effect. The amount of void required to represent the decrease in density of the EBWR moderator between 60 and 475 F (respectively, the critical experiment temperature and the saturation temperature for the operating pressure of 600 psia) is 20 per cent of the total volume of water in the core. This void was distributed within the fuel assembly water channels only. A symmetrical loading of alternate thick and thin fuel assemblies was assumed. Further, it was assumed that the control rods were withdrawn at least 80 or 90 per cent, which gives little or no rod displacement by water. One cell of nine fuel assemblies, bounded at each of its four corners by a control rod, was used to calculate the total water volume in the core.

The void value was allocated between thick and thin fuel assembly water channels on a percentage basis, channels with more volume being assigned more void. This was deviated from only slightly to permit the use of the same type of Styrofoam strip in both the thick and thin fuel assemblies by loading one less strip in the thick assembly. Allowance

TABLE 6-2
SUMMARY OF CALCULATED VOIDS

	Voids per fuel assembly type, in. ³			
	KN	NN	KE	NE
Temperature Coefficient	118 0	140.0	118.0	140.0
Boiling Voids, Zone No. 1	50 3	36.9	57.5	57 5
Boiling Voids, Zone No. 2	29 2	20 0	33 9	33 9

TABLE 6-3
SUMMARY OF AVERAGE MEASURED VOIDS

	Voids per fuel assembly, in. ³			
	KN	NN	KE	NE
Temperature Coefficient	109 0	130 0	109 0	130 0
Boiling Voids, Zone No. 1	45 8	33 3	48 4	51 3
Boiling Voids, Zone No. 2	19 0	16 3	40 1	40 3

was made for the density of the Styrofoam by increasing the void by 3 per cent. No attempt was made to simulate conditions external to the circumference of the core loading.

The second phase of the void calculations dealt with boiling voids. Several assumptions were necessary. For example, the rods were assumed to be near the "all-out" position so that boiling could be represented for a full-length core. Uniform steam distribution was assumed along a boiling length equal to the upper 60 per cent of each channel.

Figure 6-6 shows the radial variation of power density modified for the 60 per cent boiling length. The curves correspond to the flux distribution obtained when the core is loaded with a uniform mixture of the four types of fuel assemblies.

It was decided that a better simulation of voids would be obtained by dividing the core into two zones and assigning an average power density to each zone. As shown in Fig. 6-6, Zone No. 1 extends from the center of the core out to a radius of 1.1 ft and includes the central 6 by 6 square of 36 elements, with the exception of the four elements at each corner of the square. The latter elements are included in Zone No. 2, which represents the balance of the assemblies external to Zone No. 1.

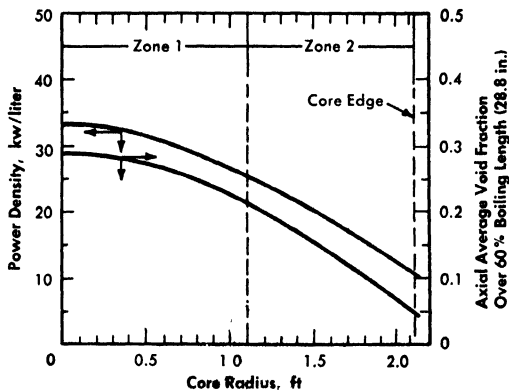


Fig. 6-6. Radial variation of power density and steam void fraction.

The average power densities aligned were; 30 kw per liter for Zone No. 1, and 18 kw per liter for Zone No. 2. The average power density for each of the four types of fuel assemblies was then determined and the data used to calculate the void volumes (Table 6-2). The voids measured with average sized Styrofoam strips are listed in Table 6-3.

It should be pointed out that the boiling void and temperature coefficient effects were combined in a second set of strips rather than adding a set of boiling void strips to the set of temperature coefficient strips used in the first part of the experiment. This was necessitated by the relatively large voids required in the boiling sections of the channel and the requirement that the voids be discontinuous, if possible, to help reduce neutron streaming. The effect was to reduce the accuracy of boiling voids since they were the differences of two large numbers. A further contribution to the differences shown in the tables was the necessity for limiting the number of types of strips by using multipurpose standardized strips in different combinations.

The mechanical design of the Styrofoam strips was governed largely by the dimensions of the fuel assembly water channels and the desire to distribute the voids along the length of the channel to minimize neutron streaming. With one exception, the strips were 54 in. long (the same length as the fuel plates) and 3 in. wide. It was later found necessary to cut the strips for the two outermost channels to $2\frac{1}{4}$ in. wide in order to clear identification straps welded in the inside corners of the fuel assembly loading brackets. A thickness of 0.281 in. was selected for all but two strips after trial calculations. The two exceptions were made 0.343 in. thick to accommodate large boiling voids. The above dimensions gave the strips reasonable strength so that they could be handled

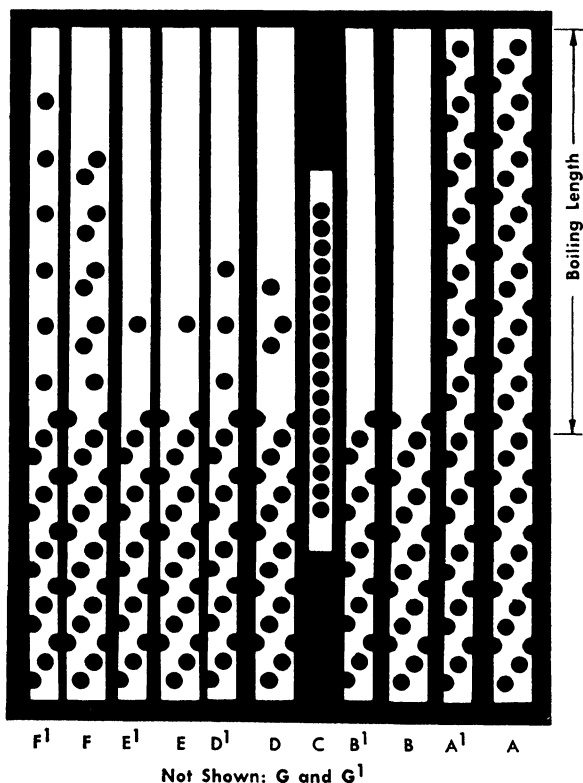


FIG. 6-7. Styrofoam void strips

with little breakage. Good channel clearances were obtained and the volume to surface ratio was such as to give a good displacement fraction.

The temperature coefficient strip was designed with 1.315-in. diameter holes to permit hole offset to stop neutron streaming, to make the number of holes feasible, and to keep the structure as strong as possible. The void volume was compromised slightly to permit use of the same strips in both thick and thin plate fuel assemblies. This was accomplished by loading all five channels in the thin, and only four channels in the thick, fuel assemblies. The loading was independent of enrichment; hence, only two types of strip were required for the entire temperature coefficient core loading. These were designated A and A' and are shown in Fig. 6-7, which displays all the types manufactured.

The boiling voids were included with temperature coefficient in a second set of strips. The hole size of 1.315 in. was retained and the holes were grouped in such a way that the same punches and punch groupings as were used in the temperature coefficient strips could be used again with

TABLE 6-4
SUMMARY OF FUEL ASSEMBLY LOADINGS

Type of void	Fuel assembly type			
	KN	NN	KE	NE
Temperature coefficient	3A, 1A'	3A, 2A'	3A, 1A'	3A, 2A'
Boiling plus temperature coefficient				
Zone No. 1	3G, 1B', 1C	3B, 2B'	3G, 1G', 1C	3G, 2G'
Zone No. 2	3F, 1F', 1C	3F, 2F'	3E, 1E', 1C	3E, 2E'

as little alteration as possible. Since the boiling voids were confined to the upper 60 per cent of the channel, the lower 40 per cent is temperature coefficient only and looks just like the A strips. To permit a reduction of types of strips, use was made of one short strip (designated C in Fig. 6-7), which equalled the 60 per cent boiling length. This permitted adjustment of total boiling void per fuel assembly without altering the temperature coefficient in the lower 40 per cent of length. Table 6-4 is a summary of the types of strips used in loading each fuel assembly, with one strip used per water channel.

One very important aspect of the design was to provide a safe loading scheme which would guarantee that the strips could not shift from the proper loaded position. It was essential that the strips not be allowed to escape upward out of the fuel assembly, since their buoyance provides very large acceleration which would lead to rapid reactivity increase. The solution to this problem was to insert into the fuel assembly loading bracket a thin stainless steel sheet bent almost double, so that its spring action held it securely in place, thus effectively blocking off the end of the assembly. The sheet was wrapped with tape to prevent scratching of the fuel assembly, which could lead to corrosion. Downward movement of the strips was prevented by the use of a specially designed pin. Similar to a safety pin, it was thrust through the top edges of all strips in an assembly and then locked in place (see Fig. 6-8).

The fabrication of the Styrofoam required the development and application of semi-production techniques to produce the required number of strips uniformly and inexpensively. The Styrofoam was obtained from the manufacturer in the form of $\frac{5}{16} \times 3 \times 54$ in. and $\frac{3}{8} \times 3 \times 54$ in. strips. The manufacturer's sawing tolerance was $\pm \frac{1}{32}$ in., which is equivalent to a void error of ± 10 per cent, or more. Therefore the thickness was increased by $\frac{1}{32}$ -in. to permit finishing to the correct size.

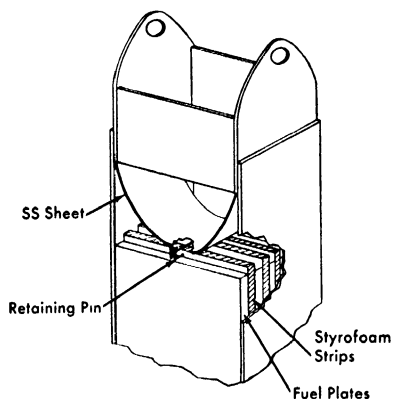


FIG. 6-8. Method of containing Styrofoam strips within fuel assembly.

A method of sanding was developed for this purpose. It consisted basically of a disc sander with a special fixture which channelled the strip against the sanding wheel. The sanding forces exerted by the wheel drew the strip through, thus eliminating the need for a feeding mechanism.

Requirements for strips of width and length other than indicated above were met by sawing to a size on a large band saw using a special fixture.

The most difficult problem to solve was that of producing void distribution through the length of the channel. The use of distributed holes in the strips was made feasible by the development of a method to produce large numbers of holes rapidly. Basically the method uses a hot cup, similar to a cookie cutter, which melts through the Styrofoam very rapidly when pressed against it. The cups were heated by soldering irons. The heating of the irons was controlled to give uniform hole production. A total of 48 cups were mounted on a framework which pivoted in such a way that all the cups were pressed into the Styrofoam simultaneously (Fig. 6-9). The maximum production was about 10 strips per minute.

THE ZERO POWER EXPERIMENT

The EBWR void experiment was conducted in three parts:

1. Temperature voids were gradually added to core loading No. 26 starting at the center and working radially outward. The purposes were to provide data on the temperature void coefficient as a function of radial position and to ensure that the void coefficient was negative at all points.

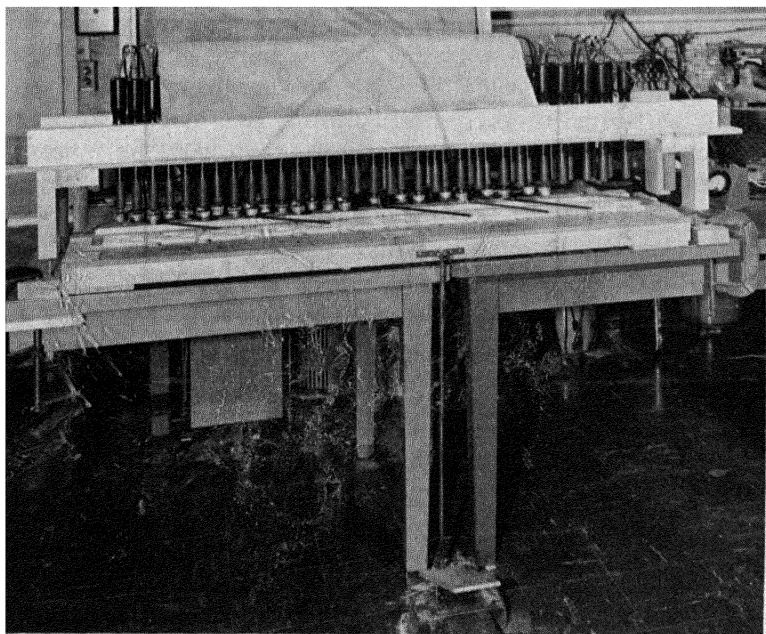


FIG. 6-9. Multihole press.

2. Temperature voids were used with core loading No. 33 (Fig. 6-4) to obtain reactivity as a function of void volume. The data were then used to calculate the temperature-void coefficient.
3. Boiling voids were used with core loading No. 33 to measure the boiling void coefficient. Boric acid was also used to enable a rod withdrawal consistent with the assumption that boiling takes place only in the upper 60 per cent of the water channel.

Figures 6-10 through 6-12 present the data from experiments (1), (2), and (3), respectively. The abscissa in all figures is the average void volume expressed as a per cent of the water channel volume, without regard to axial distribution. Uniform radial distribution of voids is assumed in Figs. 6-11 and 6-12. The void coefficients shown were computed by the relationship:

$$\text{Void coefficient} = \frac{\Delta k}{\Delta V},$$

where Δk = reactivity change in per cent caused by ΔV ,
 ΔV = change of void volume in the fuel assembly water channels expressed as a per cent of the total no-void volume of the channels.

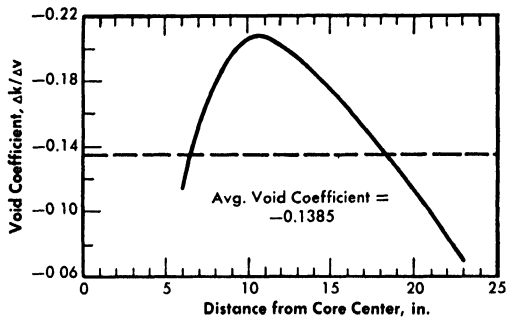


FIG. 6-10. Radial variation of void coefficient, core loading No. 26. Control rods in upper 67% of core.

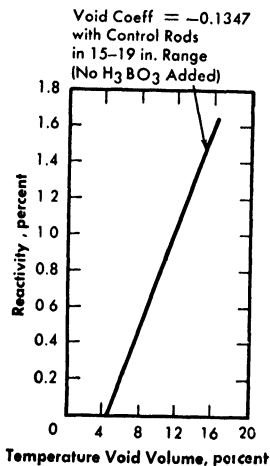


FIG. 6-11. Temperature void addition to core loading No. 33.

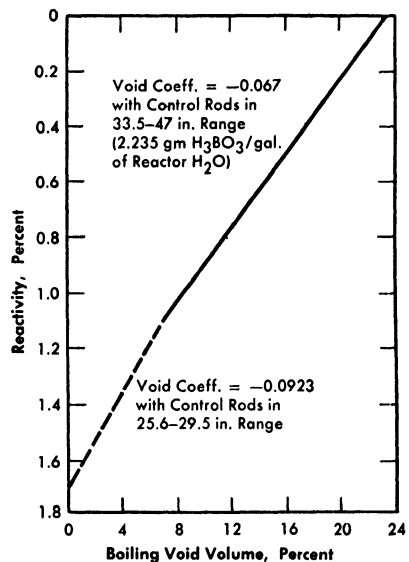


FIG. 6-12. Boiling void removal from core loading No. 33.

The varying slope in Fig. 6-12 shows that control rod insertion affects the results. This would be expected since it was necessary to size the voids in accordance with an assigned core height. The height, of course, will vary with control rod position, and thus error will be introduced. The data should be most accurate for the largest rod withdrawal.

The usefulness of this method of using Styrofoam to produce voids for zero power representation of hot reactor conditions was proved by the success attained in predicting actual performance. An important consideration is that it was an economic solution though it sacrificed

some flexibility. More flexibility of void placement would be attained, for example, by the use of a more expensive, water-filled tube system that could be emptied and filled at will. Such a system might prove more desirable where long run experiments are to be carried out. The experimental accuracy of the Styrofoam method could be improved by developing a method of sealing the surfaces to prevent the absorption of water. This not only would eliminate the need for correcting data, but would permit re-use of the material in succeeding experiments.

Credit for the success in developing and applying this method belongs to a group of members of Argonne's technical staff, including: J. A. Goleb, J. M. Harrer, J. A. DeShong, Jr., A. S. Jameson, D. H. Lennox, P. A. Lottes, J. A. Thie, J. M. West, and, in particular, M. C. Glass, who carried out the manufacturing and testing, contributing many ingenious ideas.

D. Measurement of boric acid worth. As evidenced by Figs. 6-10 through 6-12, the void coefficient values are sensitive to the control rod configuration. The fact that loading No. 33 is not significantly less negative than No. 26 (Fig. 6-3), as would be expected, is not understood, although it may be due to its extra void content.

By adding boric acid to loading No. 33 and removing the control rods a more useful number (-0.067) for the removal of "boiling voids" is obtained. A preliminary comparison using a crude 2-group, one-region approximation to this loading has the theory giving value of $d\rho/dV$ very near zero, and also erring on the reactivity value by 3.39 per cent. Using a modified theory²¹, however, good agreement on reactivity and void content is obtained.

The use of boric acid serves the dual purpose of allowing the critical position of the rods to be varied in a given loading, as well as to obtain information on thermal utilization. To ensure thorough mixing, the acid is first dissolved in hot water, then poured into the reactor water, the latter being agitated by an air-bubble stream.

Again using a crude 2-group, one-region approximation to loading No. 33, Table 6-5 shows the good agreement with experiment, in this case, fortuitously good.

TABLE 6-5

REACTIVITY AND VOID COEFFICIENT OF LOADING NO. 33
(with 2.235 g of H_3BO_3 per gal and 23.75 per cent voids)

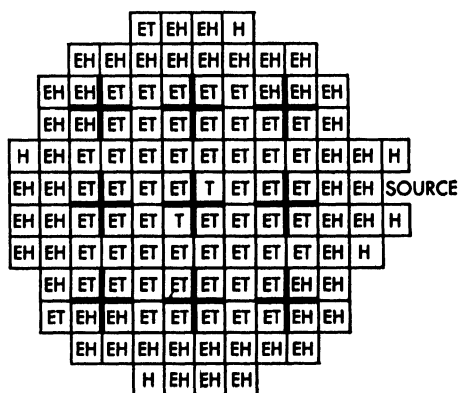
	Experimental	Theoretical
Reactivity, ρ	0	0.00382
Void coefficient $d\rho/dV$	-0.067	-0.0604

TABLE 6-6

FLUX AND POWER DISTRIBUTION AT ROOM TEMPERATURE IN LOADING No. 46

Axial max to average power density, control rods in upper 71% of fuel	2.17
Axial buckling from the flux	$0.002 \text{ cm}^2 \pm 20\%$
Reflector savings of control rod zone	$28.0 \pm 8 \text{ cm}$
Radial max to average power density (ignoring hot spot factors)	1.66
Radial power distributions	
Thin naturals	1.6%
Thin enriched	65.0%
Thick naturals	1.1%
Thick enriched	33.0%

E. Loading for pressurized power. A comparison of the borated and Styrofoam-loaded core No. 33 with the xenon, samarium, and voids calculated to be present at 600 psi and equilibrium full power showed the need for additional reactivity. The final loading, (No. 46) shown in Fig. 6-13, gave an additional 2.06 per cent reactivity, which was more than adequate.



Composition: 50 Thick Enriched, EH
 56 Thin Enriched, ET
 6 Thick Natural, H
 2 Thin Natural, T

114

Total Wt. of U²³⁵, 76.4 kg

FIG. 6-13. Loading No. 46.

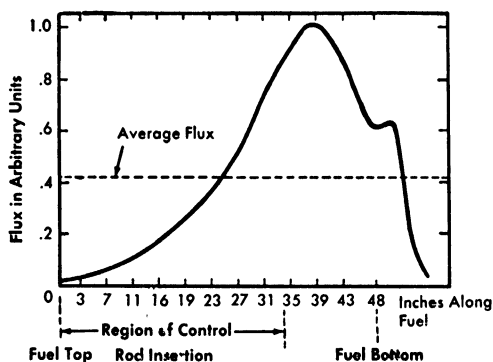


FIG. 6-14. Axial flux distribution two or more inches from a control rod in loading No. 46 at room temperature.

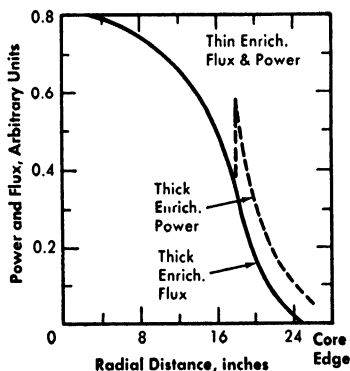


FIG. 6-15. Radial flux and power distribution in enriched elements of loading No. 46.

The logic behind the configuration of loading No. 46 is as follows:

Thin elements in the center give a less negative void coefficient and possibly more stable power. Natural elements in the center improve the radial maximum to average power density; whether this improves power stability is speculative. Defective thick enriched elements were replaced by thick naturals at the core edge. The choice of 114 elements is arbitrary, since presumably any number in the range from roughly 100 to 148 would give comparable power density performance.

Gold foils and wires were used to measure the flux distribution at room temperature in loading No. 46. The results are plotted in Figs. 6-14 and 6-15.

Using the relation

$$\rho = 1 - \exp(-\Sigma_i \rho_i) \quad (6-1)$$

for obtaining large reactivities, it is found from Fig. 6-16 that the control rods are holding down 7.13 per cent at cold clean criticality. Extrapolation yields an additional 6.82 per cent in shutdown strength. A more accurate measure of the shutdown strength is obtained by determining maximum buckling of the upper rodded region from the axial flux plot (Fig. 6-14), as if it were an exponential assembly. This yields a shutdown strength of 5.27 per cent.

F. Temperature coefficient measurement. The startup heat exchanger which utilizes auxiliary steam was used to heat the reactor water up to

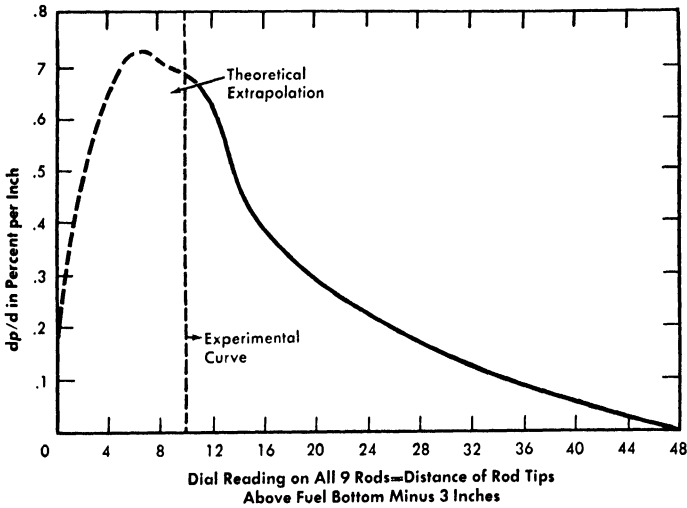


FIG. 6-16. Differential worth of all 9 control rods as a bank at room temperature in loading No. 46.

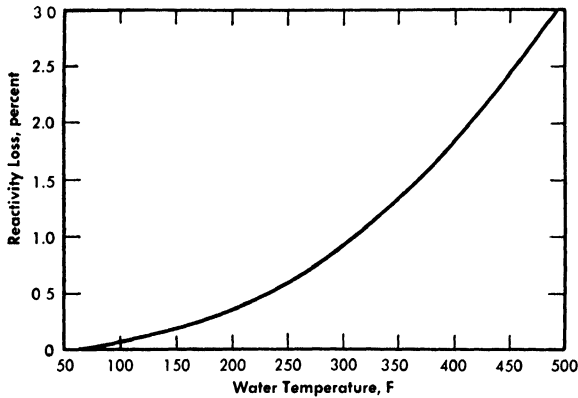


FIG. 6-17. Integrated temperature coefficient for loading No. 46 (unpoisoned).

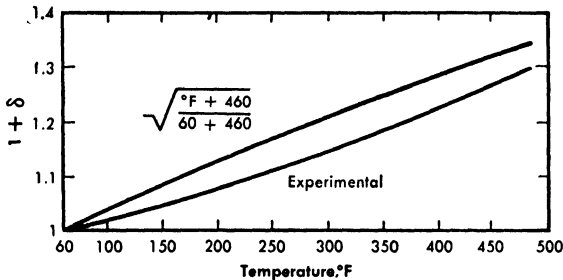


FIG. 6-18. Effect of temperature on differential rod worth.

325 F (reactor pressure = 100 psig). Higher pressures were obtained by nuclear heat. Figure 6-17 shows the integrated temperature coefficient obtained from frequent critical measurements and control rod calibrations performed during the heating process. A preliminary comparison with the void coefficient shows that the effect of water density probably dominates the contributions to temperature coefficient.

Figure 6-18 shows the effect of temperature on differential rod worth, using the factor:

$$1 + \delta = \frac{\text{differential rod worth at some temperature}}{\text{room temperature differential rod worth at the same rod setting}} \quad (6-2)$$

It is estimated that experimental errors lead to a 20 per cent uncertainty in δ .

Since the temperature coefficient is a function of rod position and the rod worth is a function of temperature, there is no unique value for the integrated temperature coefficient. For example, three acceptable definitions are:

$$\int_{X(68)}^{X(489)} \left(\frac{d\rho}{dX} \right)_{T=68} dX = 2.47 \text{ per cent}, \quad (6-3)$$

$$\int_{X(68)}^{X(489)} \left(\frac{d\rho}{dX} \right)_{T=68} [1 + \delta(T)] dX = 2.93 \text{ per cent (from Fig. 6-17)}, \quad (6-4)$$

$$[1 + \delta(489)] \int_{X(68)}^{X(489)} \left(\frac{d\rho}{dX} \right)_{T=68} dX \\ = 1.3 \times 2.47 \text{ per cent} = 3.21 \text{ per cent}, \quad (6-5)$$

where $X(68)$ and $X(489)$ are the critical positions at room temperature and 600 psig respectively.

These values, which pertain to a reactor whose control rod bank is in the 11 in. to 15 in. position, are, of course, appreciably larger than the corresponding value for the reactor with all rods out. The latter is $7.13 - 6.46 = 0.67\%$, but is very inaccurate due to uncertainties in the value of δ .

INITIAL STARTUP OF EBWR

Before the reactor vessel was closed, prior to the first power generation experiments, foil irradiation at about 100 watts was used to determine flux distribution and to calibrate the ion chambers against average flux power in the core. After the system was closed up, the approximate power was determined by reading ion chamber current. Boiling effects were first noted at about 200 kw of estimated flux power. These were evidenced by variations of about 2 per cent in chamber current. The reactor was not preheated in the first power run, and the startup temperature was about 80 F.

Several hours were consumed in reaching the rated 600 psig. Power was held at values below an estimated 1 Mw throughout this time. This turned out to be a fortunate choice, since calibrations of heating rate against chamber current showed that the flux power was probably 2 or 3 times higher than estimated. This error was attributed to reflector water-density changes.

At pressures below about 150 psig, a rather regular pattern of cyclic flux variation occurred, the cycles having a magnitude of about 20 per cent of the total power and a period of about 15 to 20 sec. (This is shown in Fig. 6-19.) Here the starting pressure is 130 psi. Control rods were pulled out of the core until a 30 sec period was obtained, and then the control rods were left in a fixed position. Reactor power increased exponentially starting from about 2 watts. As the temperature increases, the period increases due to the negative temperature coefficient. Boiling takes place, and as indicated on the chart, a peak power of 640 kw was obtained.

Steam lines were next vented and drained. The effect of each such operation was readily apparent on the flux recorder. For this reason, close contact by phone was maintained with the operators temporarily working in the containment shell. The communication system was a vital plant component throughout this stage of operation. As each line was opened, the flux would fall, then tend to return to the original value as equilibrium was re-established. Disturbances were some 10 to 20 per cent in magnitude and 10 to 15 sec in duration.

As a rule of thumb, no rods were dropped unless the flux rose some 20 per cent above the starting equilibrium value. When this condition was exceeded, one rod was dropped to cut back power. The rod was then raised again and equilibrium power re-established. As the operating group became aware of the effect of each plant operation, the startup procedure was improved.

Experience to date seems to indicate that it is probably safe enough to start the reactor with room temperature water. On the other hand, for routine operation, preheating to a higher pressure gives a much more predictable set of startup observations, probably because circulation is established more quickly and the 15 sec cyclic variations are lower in magnitude. Compare Fig. 6-19 with 6-20.

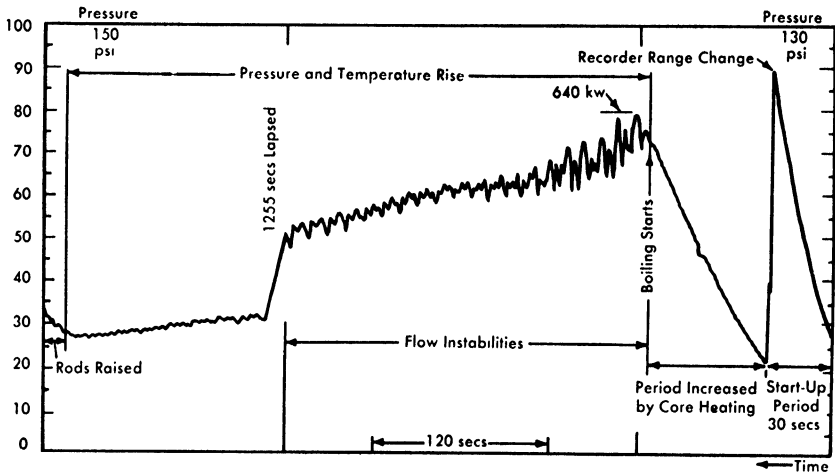


FIG. 6-19. Linear flux chart, low-pressure startup.

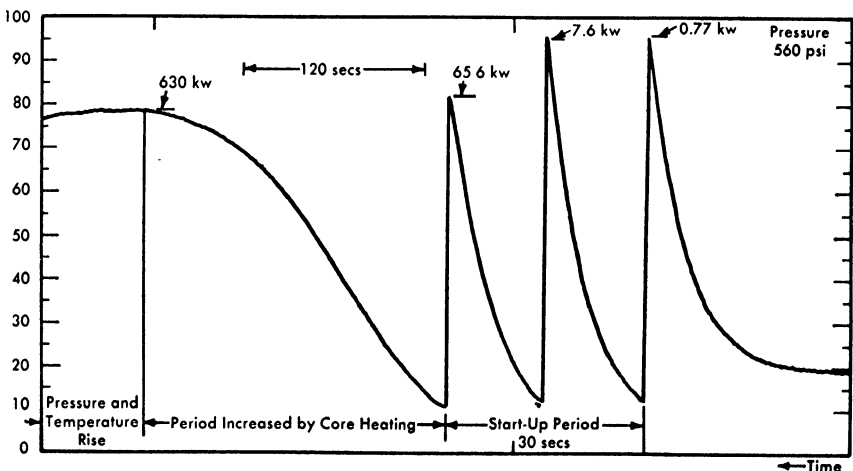


FIG. 6-20. Linear flux chart, high-pressure startup.

An experiment was run in which the reactor water level was raised about one foot over the upper core shroud. As heating progressed at pressures below 100 psig, the water was allowed to boil off. Some increase in magnitude of the cyclic variation in flux was observed as the level of water fell below the upper shroud. It decreased when the level was subsequently raised. This test seemed to indicate that water surging over the shroud and irregular downcomer water flow account for the cyclic variations. When properly recognized, the effect on flux of the inception of boiling is not disturbing and probably will not be a major consideration in future operation.

The entire steam plant had been checked out prior to boiling the reactor, using 190 psig steam from the Laboratory supply. This work went on while the core loading was being tested. This process served, among other things, as a cleaning process for steam and water systems. The condensate was run to the drains.

The importance of this step should not be minimized, because a shakedown period is necessary in any power plant. In the boiling-reactor plant, it would be difficult, if not hazardous, to go through this same process with radioactive steam.

The water, steam, and electrical portions of the plant were run in this way, with the generator putting out about 750 kw for a period of several hours. After this checkout, the final connections for steam between the reactor and the steam drum were made, and the feedwater system was connected to the reactor. The reactor vessel was scrubbed and washed with high purity water.

APPROACH TO FULL POWER

The reactor power was raised until about 10,000 lb/hr of 600 psig steam was passing directly to the desuperheater in the condenser through the steam bypass valve. Feedwater flow and steam flow were balanced by manual control. Both were then put on automatic control, the controlled parameters, rate of change of both pressure and water level, being held near zero.

The functioning of these automatic control systems had been thoroughly checked by preoperational tests and were determined to be reliable.¹⁸ Most important was the fact that they were thoroughly understood by the operating crew. Since these controls allowed focusing attention on the more important parameters, such as flux variation, the automatic systems were used immediately.

Power was raised in steps of about 10,000 lb/hr until the steam flow reached 60,000 lb/hr. A check on rod motion showed that the target

value of about 2 per cent reactivity in voids probably would be met. A high-frequency response recorder was used to watch for possible "chugs" or a sudden large fall or rise in flux signal. This supplemented the operational flux recorder, which had a full-scale pen travel of about 4 sec.

When the steam-flow meter indicated a flow of 60,000 lb steam/hr, the reactor should have been at a power level of 20 Mw. There was, however, no correlation between water-flow reading and steam flow. Steam samples were taken to check for moisture, and it was found that the steam was very high in moisture content. Apparently the separated moisture in the steam dryer was not returning to the reactor by gravity as expected. Instead, due to the low pressure in the steam-dryer drum (resulting from the steam flow through the pipe in the reactor shield), the water was flowing upward into the dryer and mixing with the steam.

To correct this condition the return line was closed by installing a valve, and a steam-water trap was installed at the drum with a blowdown to the condenser. A design error was also found in the steam-flow orifice, so that the instrument had to be readjusted. After these changes, the plant was restarted and all meter readings were found to check. Full power of 60,000 lb/hr was reached, the turbine was warmed up, and the generator synchronized and loaded. Adjustments were found necessary to the generator protective relays, and the generator governor was reworked to give better speed control.

OPERATION AT FULL POWER

Following the above experiments, a test was started to run the plant at full power. Overflux trips were set to about 120 per cent power. The steam powered, back-pressure regulator was set to open at about 625 psig. Overpressure shutdown interlock was set at 625 psig. The run was continued for seven days, during which time several system safety tests were run.

Even though at full power and 600 psig the flux variation was less than 4 per cent, there was some chance that a sudden uncontrolled surge in the feedwater could send the reactor into unstable operation. To test this possibility, the feedwater was run to zero, then up to 90,000 lb/hr in about 30 sec, using the full speed of the control valve. The flux followed the variation, falling to about 65 per cent of power at zero feed and rising to about 120 per cent power as the feed rate was raised. The time delay between the start of a change in feedwater flow and the beginning of the resulting flux change was about 10 seconds. This test established the fact that unstable operation would not result from mismanagement of feedwater.

A similar test was made with steam flow in which the flow was lowered to about 10,000 lb/hr and raised to 80,000 lb/hr. Here the flux responded by rising rapidly as the steam flow was cut down and falling rapidly as the steam flow was raised. No time delay was observed, but the instrumentation was not adequate to produce quantitative results. Again, it was demonstrated that no unstable operation could be effected in this manner.

Setting the overflux trips to 120 per cent power provided adequate protection against steam or water-control malfunction. One should, of course, observe further that, if the trips are set at this overflux value in a central-station power plant, the failure of the feed water or steam-flow control could result in an overflux shutdown. Partial or selective reactivity reduction to improve plant reliability should be considered for such plants.

EBWR OPERATIONAL DATA AND EXPERIENCE

An item of principal concern when the reactor was first placed in operation was the effectiveness of the reactor vessel closure at the operating pressure of 600 psig. The inside diameter of the closure is 68 in., and double gasketing was provided to seal the flat head to the vessel. Flexitalic gaskets were used, having a zinc plated carbon steel core with a teflon filler. Leakage measurements were made by monitoring the moisture collected between the two concentric gaskets.³⁻⁵

During the cold hydrostatic test, a leakage of 184 cm³/hr at 1040 psig was measured. No leakage was ever detected during operation at power. After 14 thermal cycles from room temperature to the operating temperature of 478 F, the cold hydraulic test was repeated. Leakage then was 1700 cm³/hr at 1200 psig. Closure bolt stress has been reduced to about 20 per cent of the original value after the thermal cycles had accounted for this leakage at room temperature.

The reactor vessel was opened twice for installation of experimental equipment since the initial startup; once after six months of operation, and again after twelve months of operation. In each instance new gaskets were installed. While the long range performance of the closure, therefore, must remain speculative, it is clear that no major difficulty is in prospect.

System maintenance. The operating cycle for the first year was interrupted several times to perform condenser cleaning, valve inspection, pump-motor repair, and water purification system maintenance. Where spare units were available the plant was not shut down, but replacement was made with the plant operating.

Figure 6-21 presents a record of the operation of the plant from December 1956 to December 1957. The first months of operation included many training runs in which the first permanent staff was instructed by the design engineers and specialized personnel who engaged in the startup and commissioning of the equipment. By June 1957, the initial or shake-down period had been completed, and during the shutdown extending into August, experimental equipment to be used in the reactor stability and system response testing was installed. These tests, called "transfer function measurements," extended into October. These tests are described later in this chapter.

During the June-to-August period the fuel handling equipment was checked out and fuel elements were lifted for inspection under water. Deposits were found to be mainly iron oxide with some aluminum and nickel oxides. No defects were discovered and the elements were put back in service.

To remove the fuel elements, the structure over the reactor is first taken away and the vertical coffin and core indexing top plug are moved into position at the top of the reactor vessel. The time required for cooling the reactor vessel before removing the top without inflicting undue mechanical stresses produces the longest delay. The normal time consumed was 48 hours because there was no incentive for rapid completion of the opening operation. A time of 24 hr is sufficient to complete the job if cooling is maintained at the rate of about 50 F/hr and temperature differentials across the vessel are kept below 100 F. Cooling of the upper vessel is done by mixing cold water with the water in the vessel and filling the upper steam space. Heat is removed by operating the water cleanup system coolers. Convection cooling and water mixing must be carefully checked and controlled by reading the vessel temperature along the length. The reactor vessel top closure is removed when the temperature of the upper vessel is about 100 to 125 F.

Normal water purification is maintained during the cooling operation, and when the top is removed, the core is clearly visible and fuel loading can proceed promptly. As the fuel element is raised above the water and passes through the 2 ft of air above the water, before entering the coffin, the activity at the top of the vessel reaches 5 r/hr. Operation of the mechanism from above the coffin prevents this activity from affecting persons performing the lifting. In any event, as long as no mechanical difficulty is encountered, the exposure would be only for a fraction of a minute and the total radiation dose would be very low. The maximum activity measured in contact with the coffin surface when it contains an element is about 5 mr/hr. The 1 ft of lead in the coffin annulus is adequate for shielding.

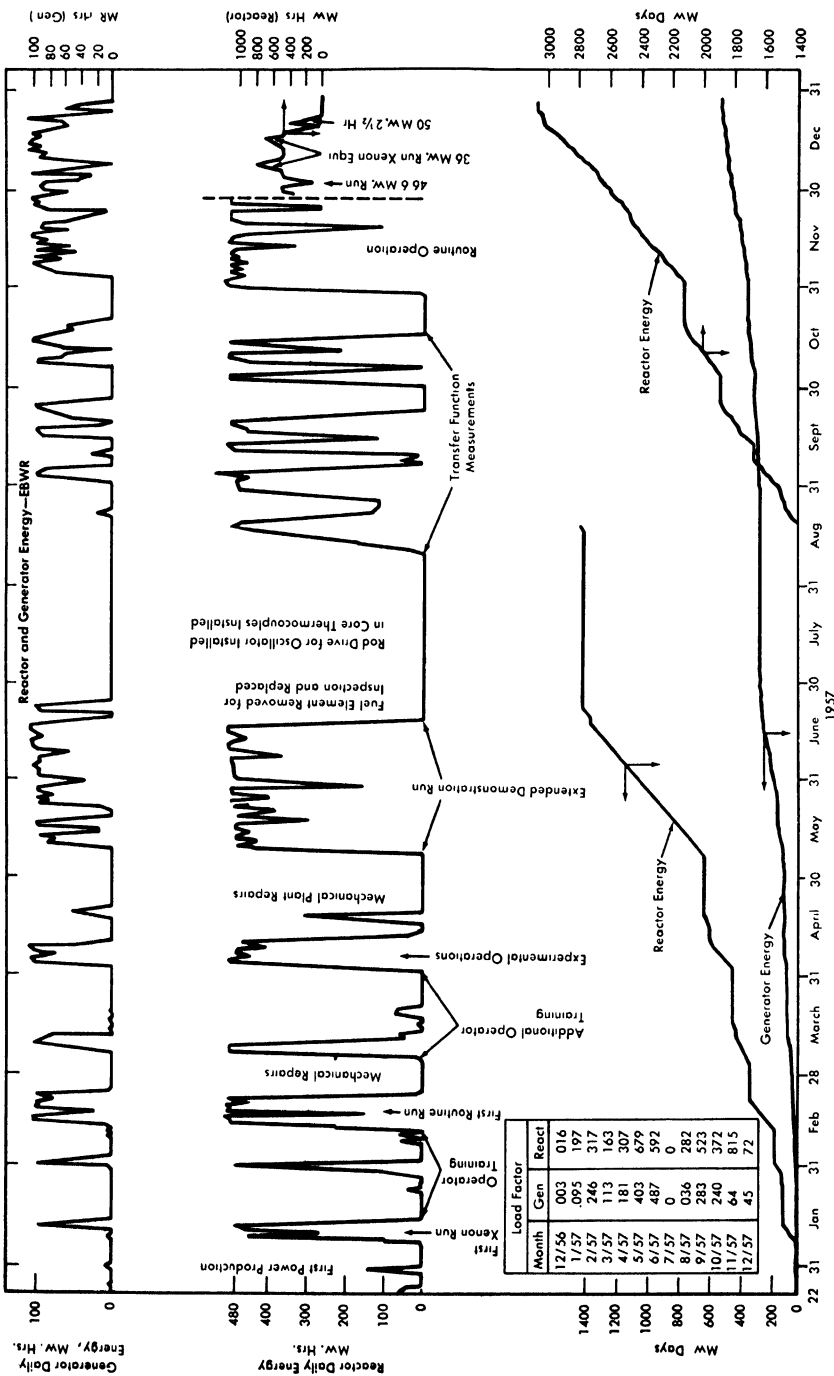


Fig. 6-21. EBWR operational record.

Moving the coffin over the fuel storage well is routine, as is the lowering of the fuel into the well. The complete fuel handling cycle can be completed in 45 min for one element. Extensive loading could be so routine that this time is shortened to about 30 min per element.

Reclosure and hydrostatic testing of the vessel required about 16 hr. The complete operation cycle from shutdown to startup, therefore, requires between 40 and 48 hr plus the time required to change the fuel.

FUEL-ELEMENT FAILURE AND RADIOACTIVITY MONITORING

An important phase in the early operation of EBWR was the concern about fuel-element failure. In any type of heterogeneous reactor, the failure of any fuel element necessitates the removal of that element. When all the fuel is in a single container, the problem of locating and removing the defective element is paramount.

In the case of a boiling water reactor, fission product gases will be released to the steam turbine system when failure of a fuel element occurs, and so it becomes necessary to establish a method of detecting these fission product gases at once. The investigation of the radioactivity levels in the EBWR system was the direct result of the need for a fast and highly sensitive fission product monitor.⁶ It had to be sensitive enough to detect minor cladding failures such as small fissures.

Examination of existing fission product monitoring systems showed that they were lacking in both sensitivity and response time. The water monitors using beta, gamma, and delayed neutron detectors all dilute the released fission products in large quantities of water, and the radioactivity in a particular volume is low.

The use of the activity of the fission product gases was very appealing from the very beginning of the investigations. These gases are among the first fission products to escape into the reactor water. They are driven off by the boiling action and carried over into the condenser. They are collected and concentrated by the air ejector and then exhausted. A large percentage of fission products are gases and they are concentrated by the air ejector.

A comparison of the beta and gamma ray spectra of the fission product gases indicated that the gamma rays from $\text{Xe}^{135\text{m}}$ and Xe^{138} offered the best means of detection. Table 6-7 shows the activities considered.

An analysis of the gamma ray spectrum at the air ejector exhaust had to be made. The first experimental equipment consisted of a holdup tank in the exhaust line with a well for a scintillation counter; a scintillation counter consisting of a 1-in. by 1-in. sodium iodide crystal, NaI (Th) and a Dumont 6292 phototube; and a 256 channel pulse height analyzer. The activity was higher at this point than the maximum acceptance rate of

TABLE 6-7
GASEOUS FISSION PRODUCTS SUITABLE FOR LEAK DETECTION

Isotope	Energies Mev	Half-life	Yield, %
Xe: ^{135m}	0.52	15 m	6.7
Xe: ¹³³	0.42 & 0.51	17 m	6.4
Kr: ^{85m}	0.150	4.4 hr	1.3
Kr: ⁸⁸	0.191 & 0.240	2.8 hr	4.0
Kr: ⁸⁹ to	none	3.2 m	4.6
Rb: ⁸⁹	1.05 Mev	15 m	4.6

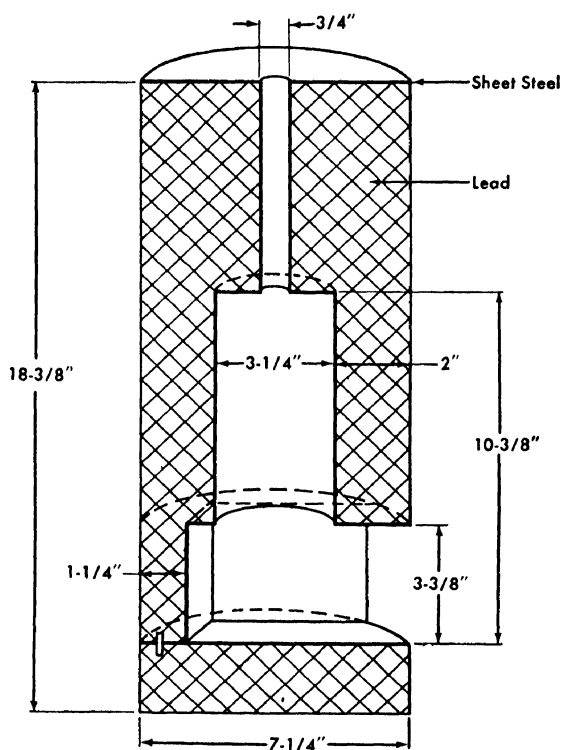


FIG. 6-22. Shield for fission product monitor.

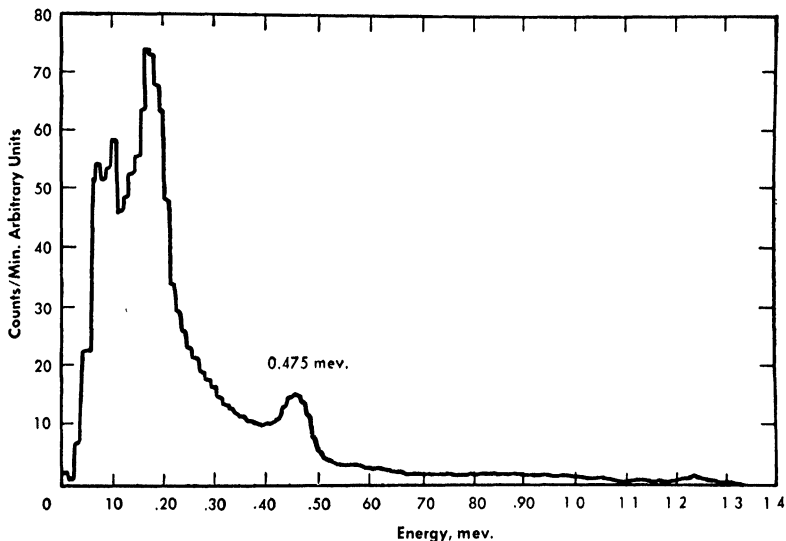


FIG. 6-23. Observed gamma spectrum of off gases.

the analyzer. Therefore, a shield was designed as shown in Fig. 6-22. The collimator was directed towards the air ejector tanks. This shield reduced the activity to the desired level for the analyzer. A comparison was made between the 1-in. by 1-in. crystal and a 2-in. by 2-in. crystal, and it was found that the 1-in. by 1-in. crystal is adequate for monitoring.

The general characteristics of the observed spectrum are shown in Fig. 6-23. The broad Compton plateau from the N^{16} activity, the broad peak at 0.475 Mev, and the large x-ray peak are characteristic of the activity at the air ejector tanks.

The peak at 0.475 Mev was exactly where the Xe^{235m} and Xe^{138} activities were expected. Because of this peak an immediate investigation into the activities present was made. The major activities observed were Cs^{138} , Xe^{135} , and Xe^{138} . This analysis definitely established the presence of fission product gases in the air ejector exhaust. Subsequent analysis established the presence of Kr^{85m} , Kr^{87} , Rb^{89} , and the probable presence of Kr^{88} , I^{132} , I^{133} , and I^{135} . An estimate of the quantities evolved in the three major activities is shown in Table 6-8.

In conjunction with the gas analysis, the activity of the reactor water was analyzed. The major activities are Na^{24} , Co^{60} , and Fe^{59} plus various amounts of other activation products. Fission product activities could not be detected against this background. An analysis of the anions and cations in the reactor water showed that most of the activity was associated with the cations. The major exception being a peak at 0.33 Mev

TABLE 6-8
MAJOR ACTIVITIES IN AIR EJECTOR EXHAUST

Cs^{138}	284 $\mu\text{c}/\text{min}$
$\text{Xe}^{135\text{m}}$	18 $\mu\text{c}/\text{min}$
$\text{Kr}^{85\text{m}}$	6.6 $\mu\text{c}/\text{min}$

which is still unidentified. A radiochemical analysis of the water by E. P. Steinberg for Mo^{99} was positive. It gave a yield which would result from 4×10^{13} atoms of fissioning uranium being exposed to the water. Analyses of the activities present in the filters and ion exchange resins of the water purification system gave the same activities that were present in the reactor water.

The possible sources of the fission products considered are diffusion through the cladding, diffusion through small fissures in the cladding, and surface contamination of the cladding. Calculation of the diffusion rate of Xe^{135} through the cladding gave a yield of approximately 7×10^{-6} atom/(cm^2)(sec). The observed rate of production is 5×10^{12} atoms/(cm^2)(sec). Thus the observed rate is greater than the calculated by a factor of 10^8 . This rules out diffusion as having a major role in the release of fission products.

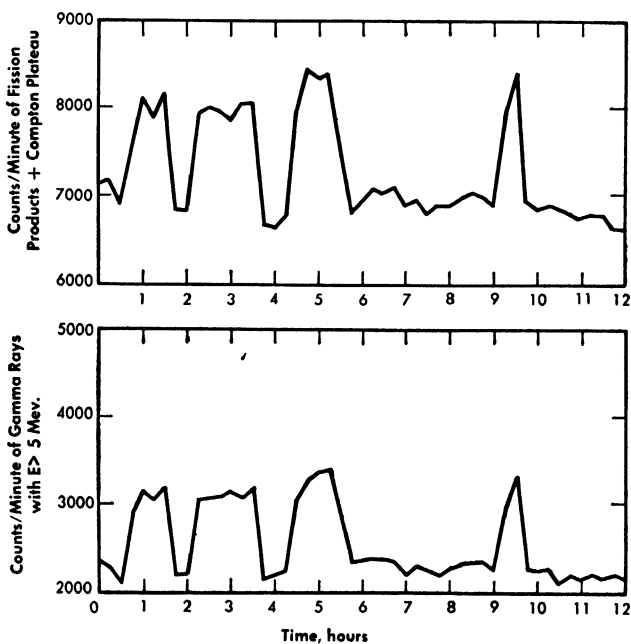


FIG. 6-24. Analysis of gamma-ray energies of off gases.

The observed activity and production of the fission gases has not changed over extended operation of the reactor. Since even small fissures corroded rapidly when the elements were tested, the observed activity and production rate should have increased. As there has been no measurable increase, it was concluded that release through fissures was not likely.

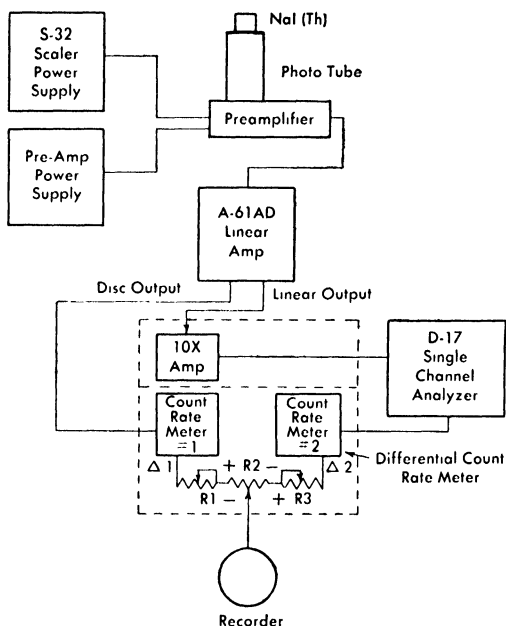
It can be shown that the amount of fuel required to give the observed activity is between 30 and 40 mg. If this is evenly distributed over the core a concentration of approximately 3.5×10^{-8} g/cm² is reached. Since the activity has not changed with continued operation of the reactor and no fuel has been found in the reactor water, it is believed that the contamination must be bonded to the cladding of the fuel plates.

The two most likely sources of contamination are the autoclave in which the elements were tested before use and the existence of small amounts of uranium in Zircaloy-2. The autoclave was contaminated during the testing of elements. It was decontaminated afterwards but even after the cleanup there probably was enough fuel present to give the required contamination of approximately 35 mg on fuel element surfaces. As for contamination of the zirconium, it is estimated that there may be as much as one part per million of uranium in the zirconium. This is much greater than the calculated concentration.

Several criteria were established for the successful design and operation of a fission product monitor. First, the monitor must be unaffected by small fluctuations in the N¹⁶ activity. The Compton scattered gamma rays from the N¹⁶ contribute many counts in the region of interest. N¹⁶ is the major gaseous activity present. This activity is influenced by power level, air ejector flow and steam flow to the turbine. The monitor should be independent of small transient changes in these parameters. Second, the electronic components must have the utmost stability and reliability. Finally, a recorded signal and an alarm system must be provided.

To determine the final design of the monitor, two parallel channels were set up—one to read N¹⁶ activity, and the other to read the fission product activity plus the N¹⁶ Compton. These channels consisted of the same detector feeding signals to two single channel analyzers in parallel. One analyzer was set to read the integrated count rate above 5 Mev. This is essentially N¹⁶ activity. The other analyzer was set for a range of 0.42 to 0.52 Mev. This includes the photo-peaks of both Xe¹³⁸ and Xe^{135m}. The output of the two analyzers was plotted for extended periods of reactor operation. Figure 6-24 shows how the fluctuations in the fission product range follow those of the N¹⁶ activity.

The final design of the monitor is shown in Fig. 6-25. The discrimina-



Fission Product Monitor For EBWR

FIG. 6-25. Final design of fission product monitor for EBWR.

tor of the A61AD amplifier is set at 5 Mev. The gain of the amplifier is set so that the maximum output corresponds to a pulse from a 7-Mev gamma. This gives a linear output of zero to seven Mev. This output is amplified by a gain of ten order of magnitude. This signal goes to the single channel analyzer which is set to pass an energy band from 0.42 to 0.52 Mev. The output of the discriminator and the analyzer go to two count rate meters whose output may be used to take either a difference or a ratio. The output of the count rate meters drives a recorder which has a variable set point alarm. This system is presently undergoing tests. By allowing some statistical fluctuations, a sensitivity of 1 per cent change in activity can be detected by the differential system, and a 5 per cent change when using a ratio system.

By means of this monitor, it has been found possible to detect very small quantities of fission products in the reactor system. As already mentioned, analysis has shown that $284 \mu\text{c}/\text{min}$ of Cs^{138} , $18 \mu\text{c}/\text{min}$ of $\text{Xe}^{135\text{m}}$, and $6.6 \mu\text{c}/\text{min}$ of $\text{Kr}^{85\text{m}}$ are released at the air ejector. These quantities have been nearly constant for an entire year of operation, and the early assumption that no fuel defect had occurred was verified. It is evident that because of this easy means of detection, an early defect in

fuel clad can be found. If a severe break occurs, the plant activity will rise to several r per hour. Since no defect has occurred to date, the full effect of such a release cannot be evaluated, but it is certain that operation with a major release of fission products will be intolerable.

In the case of metal fuels, oxidation of the fuel will occur in the case of clad failure. A general inspection of the core under water will very likely reveal the location of a swelling. Test units which were defected in test autoclaves were easily recognized because of the plate distortion, and it is expected that this effect will be enhanced if a clad failure occurs during reactor operation.

Removal of the element and cleanup of the system will have to be experienced to recount the difficulties. A major portion of the fission products will remain in the reactor vessel. Cleanup of the water by dilution with clean water would be the first step and disposal of the drainage would be difficult. Circulation of the water through ion-exchange beds will remove the dissolved material, but disposal of the beds will present a problem. These problems are common to all reactors of this class and not unique for EBWR.

RADIATION LEVELS IN THE PLANT

The radiation levels in the EBWR plant during the period it has been in operation are lower than the most optimistic predictions.⁷ The only shielded equipment in the power plant is the reactor water ion-exchange cleanup resin and prefilter, which accumulate long-lived activity. Provisions were made, however, for sufficient floor loading around all major equipment to permit installation of concrete or lead shielding, should it become necessary.

The primary radiation from equipment during operation is N^{16} with a $7\frac{1}{2}$ -sec half-life. The N^{16} is activated from an (n, p) reaction on oxygen in the water. The N^{16} decays, giving off a 6 Mev gamma radiation. Figure 6-26 shows the radiation levels.

Fifteen ionization chambers are located on the major equipment, and the surface activities are continuously recorded in the control room. The general background activity in the plant is less than 5 mr/hr. To date none of the operators has recorded measurable exposures to radiation during plant operation or in routine maintenance.

Radioactivity after shutdown also has not presented any unusual problems. Deposits have been observed on the condenser tubes, and analyses of those deposits have indicated iron, aluminum, and nickel as the major constituents. The activity inside the condenser steam dome after shutdown is only 0.1 mr/hr and presents no problem in obtaining

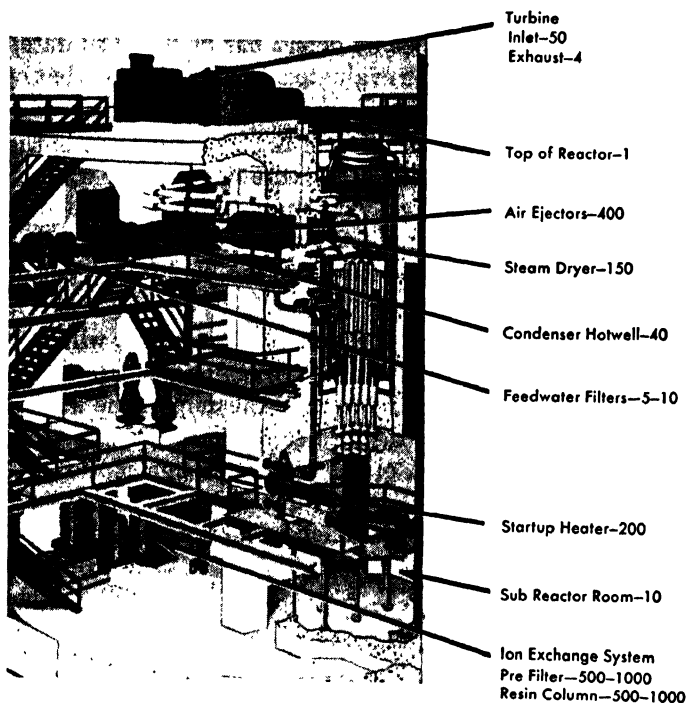


FIG. 6-26. EBWR radiation levels (20-Mw reactor output).

samples for analysis of the deposits. Typical activities observed after shutdown are given in Table 6-9.⁴

The leakage from the EBWR steam system is of primary interest for future D_2O operation. The design specifications (from the economic standpoint) limit the D_2O loss to 1 lb/day and H_2O in-leakage to 0.5 lb/day. As explained in the preceding chapter, the fluid recovery system was installed to supply dry air and to recover the gland steam from the labyrinth type seals on the turbine. The recovery system also acts as a sump for any other leakage, such as the control rods and gasket leak-offs. To date very little effort has been spent to minimize the small leaks in the system. Measurements were made to establish a starting point for determining system tightness. The total measured loss from the system was approximately 88 lb/day, i.e., 20 lb through system leaks and 68 lb through the air ejector.

There are approximately 200 valves in the steam and feed-water system which are subject to leakage. The leakage was measured on 43 of these valves; 24 proved to be leaking measurable amounts. As expected, the larger the valve and the higher the temperature, the greater the leakage.

TABLE 6-9
EBWR RADIOACTIVITY LEVELS AFTER SHUTDOWN

Source of radiation	Hard and soft, mr/hr
Inside surface of equipment	
Steam dryer	0.5
Main steam pipe	0.2
Turbine trip valve	0.2
Condenser	
Steam dome	0.2
Hot well	2.0
Radioactivities encountered while servicing equipment	
Feedwater filters	5
Control rod bushings	2
Ion exchange system	
Top of resin bed	1000
Prefilters	500
Afterfilters	500

The average leakage rate as measured was 0.1 lb/day per valve, or a total leakage of 20 lb/day. The valves that are leaking will be re-packed and again checked for leaks. This leakage must be reduced to less than 0.5 lb/day to meet the over-all system requirements.

DISSOCIATION OF WATER

Laboratory findings on the BORAX-III reactor indicated a high water dissociation rate; on the order of 75 cm³ of gas per liter of condensate at 300 psig reactor pressure.⁸ EBWR operating at 600 psig generates 55 cm³/liter of condensate. This dissociated water accounts for the major part of the system loss through the air ejector. At 20-Mw operation this loss amounts to 1 ft³ of hydrogen per minute, or a total system loss of 62.5 lb of water per day. This system loss from the dissociation of water will be recovered in a hydrogen recombiner using a catalyst consisting of 0.3 per cent platinum on alumina pellets. Tested efficiencies of this recombiner exceed 99 per cent, thus giving promise of reducing the losses from dissociated water to less than 0.5 lb/day. EBWR water decomposition data are shown in Table 6-10.

Another method of reducing the losses from the dissociated gases is to recombine the hydrogen and oxygen in the steam phase, prior to entering

TABLE 6-10—EBWR WATER-DECOMPOSITION DATA*

Date	Time	Power level, Mw	pH	Reactor water quality conductivity		Gas yield, cc/liter ^(a)	Condensed-steam gas analysis, %			Air-ejector gas flow, cfm ^(b)	Air-ejector gas analysis, %		
				determined, ohms/cc	recorded, μmhos		O ₂	H ₂	inert gas		O ₂	H ₂	inert gas
12-23-56 ^(c)	0115	20	7.0	180,000		50	22.7	47.0	30.3 ^(d)		25.2	32.8	42.0
12-23-56 ^(c)	2015	20	7.0	180,000		69	20.4	44.5	35.1 ^(d)		22.6	27.2(?)	50.2
12-29-56 ^(c)	2210	20	5.2	90,000		53	30.2	55.9	13.9		29.4	56.1	16.1
1-11-57 ^(e)	1630	20	5.0(?)	410,000		47.5	29.7	68.6	1.7				
1-30-57	1715	20	7.05	900,000		66	31.0	63.4	5.6				
2-11-57	1917	20	7.0	760,000		52.5	30.0	66.2	3.8				
2-11-57	2330	20	7.0	600,000		46	29.2	62.4	8.4				
2-12-57	1300	20	7.15	640,000		45.7	29.3	69.2	1.5				
2-12-57	2020	20	7.2		0.8	47.8	30.2	67.3	2.5				
2-13-57	1030	20	7.1		0.9	58.5	31.6	66.6	1.8				
2-13-57	1920	20	7.08		0.9	55	30.0	66.0	4.0				
2-14-57	1200	20	7.1		1.0	57.8	31.1	66.5	1.8				
2-14-57	2110	20	7.02		1.0	63.3	32.5	67.5	0.0	1.2			
2-18-57	1115	20	7.3		0.8	59.8	31.7	67.0	<1.3	1.0			
2-20-57	1000	20	6.9		0.9	58.0	31.7	66.2	2.0	0.92			
Average of samples of 1-11-57 through 2-20-57						55.1	30.7	66.4	2.9		29.7	50.5	19.7

^(a) Not corrected for gases dissolved in the condensed steam.^(b) Not corrected for gas density; values for air at 80 F in cfm.^(c) This sample was steam bypassed to the condenser.^(d) Values high due to transfer of sample or actual nitrogen in system.^(e) This sample was steam going to the turbine.^(f) Boric acid in the reactor water was 0.748 g/liter for this sample.^(g) Only 8.3 ppm boric acid was in the reactor water for this sample.* Data from R. E. Bailey's paper, *EBWR Water Treatment, Decomposition and Decontamination Factor*, TID-7535.

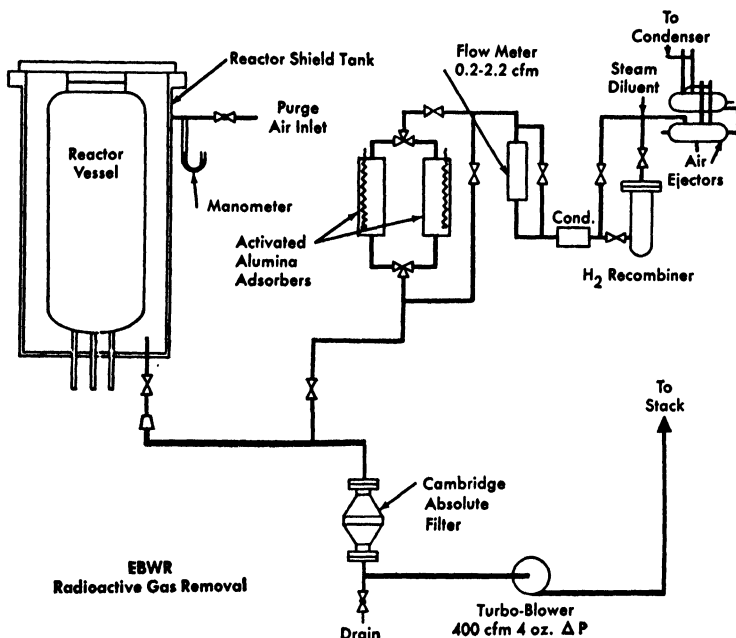


FIG. 6-27. EBWR radioactive gas removal.

the turbine. This is especially desirable from the corrosion standpoint, and the feasibility of this method is being studied in the Laboratory.

In addition to the losses just described, the entrained moisture in the air ejector gas accounts for a loss of 5.5 lb/day. When the activated alumina absorber, shown in Fig. 6-27 is installed, these losses will be reduced to less than 0.001 lb/day.

As a result of the BORAX-III findings, in order to check possible corrosion anticipated, the EBWR turbine and the 6-in. main steam piping between the dryer and the condenser were Kanigen processed as already described in the preceding chapter. The importance of this precaution has not been evaluated, but the plating has apparently remained in excellent condition judging from an inspection of the turbine inlet valve after six months of operation.

An experiment to determine the decomposition as a function of the power level was performed in the early part of 1957, the first half year of operation. This experiment involved the determination of (1) cubic centimeters of gas per liter as a function of steam flow and (2) cubic centimeters of gas per second as a function of power level. In the former case there is a definite increase in the decomposition with power level. The range is 26.1 cc of gas per liter at 5 Mw and 40 cc of gas per liter at 20 Mw. Dissolved gases were not considered here at all. In the latter

case, there was a definite exponential increase in the total free-gas evolution rate with increasing power. If the total oxygen produced per second is observed as a function of power, a slightly exponential function is noted upon which BORAX-III decomposition data at pH 7.0 falls high. (See Table 6-11.) It has been decided that in the future decomposition will be reported as oxygen produced per second since it is easy to follow this dissolved gas.

A dissolved-hydrogen analysis on the reactor water indicated a concentration of 5.2 to 5.9 ppb (parts per billion), with a dissolved-oxygen concentration of 221 ppb. A dissolved-oxygen analysis of the water withdrawn from a sample point on the third from the last stage of the turbine indicates a concentration of 5 to 12 ppb in the water leaving this stage of the turbine at 20 Mw. This, however, was only a preliminary figure and should be regarded as such.

DECONTAMINATION FACTOR

The decontamination factor between any two points of a system has been measured by observing the ratio of the quantity of a given activity in one of the systems over the other. The average decontamination factor between the water and the steam after the dryer is 5.9×10^3 (4.7 to 8.1×10^3) at 20 Mw. The decontamination factor between the reactor water and the condensate is somewhat lower than this (the average being 3.5×10^3) since all drainage is dumped to the condenser. The average increase in the decontamination factor across the dryer is a factor of 1.6. One set of decontamination-factor determinations was made across the cleanup loop. A decontamination factor of 1.35 was observed across the prefilter

TABLE 6-11
WATER DECOMPOSITION AS A FUNCTION OF POWER LEVEL

Date	Power level, Mw	Press., psig	Steam flow, lb/hr	Decomposition,* cc of gas/liter of condensate	Decomposition rate, cc of O ₂ /sec	Dissolved O ₂ in condensate, ppm
2-20-57	20	600	60,000	55.1	157	3.5
5-19-57	20	600	60,000	40	148	12.0
5-19-57	15	600	46,000	35.8	113	12.0
5-19-57	10	600	30,000	32.0	69	12.0
5-19-57	5	600	13,000	26.1	29	12.0
BORAX-III	8	300	30,000	77	102	~3.5

* The composition of gas is given in Table 6-10 for the 600-psig operation; the composition of the BORAX-III gas was 32 per cent oxygen, 60 per cent hydrogen, and 8 per cent inert gas.

and 1.2×10^2 was observed across the ion-exchange column. These are both with regard to the reactor.

In obtaining the decontamination-factor data, both an open-end Geiger tube counter and a proportional gamma counter were used. The decay curves of the reactor water, steam, and condensate have different initial slopes and do not attain the same slope until after about 20 hr of decay.

The carryover of boric acid in the steam was determined during operation with 353 and 748 ppm boric acid in the reactor water. At 20 Mw the ratio of boric acid in the reactor water to that in the steam was found to be 18.

WATER PURIFICATION

The reactor water in the EBWR system is maintained at neutral pH and very high quality, approximately 5 microhms. The makeup water is passed through a mixed bed ion-exchange purification system and then from a storage tank into the deaerating condenser. This condenser is able to keep the dissolved oxygen content of the reactor feedwater down to 0 to 5 ppb. The conductivity of the water leaving the well is 0.31 to 0.33 $\mu\text{mho/cc}$. Using an analysis from 1-in.² of filter area (2- μ size) from unit 2 of the feedwater full-flow filters after 230 hr of operation and multiplying it by the total filter area, an indication as to the quantity of insolubles in this system was obtained. The results are as follows:

Element	Quantity, g (factor of 2)
Fe	34.6
Ni	15.7
Cr	0.3
Mn	0.8
Mo	0.3

The cleanup loop for the reactor water is operated under reactor pressure. It consists of a prefilter (2- μ size), a mixed bed, and an afterfilter (2- μ size). The mixed bed consists of 6 ft³ of highly basic anion resin and 3 ft³ of cation resin. There are two complete separate loops in the system. The efficiency of these loops has been good, but there is an indication that the regenerative cooler tends to become fouled with "curd" or sludge as operation continues.

The loops were put to test when the reactor was loaded with boric acid and other dissolved solids during early 20-Mw operation. Here the water

quality was improved from 26.3 μ mhos to 2.44 μ mhos in about 25 hr of operation of unit 1. This particular unit was in use most of the time up to May 5, 1957, when the resin was removed and the column was filled with new mixed resin. Since that time unit 2 has been in operation. The quantity of water put through the first loading of unit 1 amounted to approximately half a million gallons. During this time 20 lb of boric acid had been removed from the system. An analysis of the reactor and the effluent from the bed on April 26, 1957, showed a boric acid concentration of 7.4 ppm in each. This possibly indicates a lower limit in the removal of 20 lb of boric acid with the system. The old resin, which was contained in a stainless-steel basket, was easily removed. The radiation levels were 1 r/hr hard at 6 in. (at the top), 500 mr hard at 2 in. (1 ft down), and 100 mr hard at 2 in. (2 ft down). The bed was able to decay for 14 days before removal of the resin.

At present the reactor water quality is 0.45 μ mho with an oxygen concentration of 200 to 300 ppb and less than 0.1 ppb dissolved iron. Insoluble iron, however, has been readily found in the system.

The prefilters have done an excellent job in removing insolubles from the reactor water. Both the prefilters and afterfilters have been changed twice since operation began. The afterfilters had to be changed because their activity level increased to 2.0 r at 2 in. This increase could have been caused by the agglomeration of colloidal particles in passing through the bed and their subsequent deposit on the afterfilter.

A preliminary examination of the material deposited on the prefilter showed Mn⁵⁶, Fe⁵⁹ and Co⁶⁰ to be the major sources of activity. Ni⁵⁹, Cr⁵¹, Zr⁹⁵, Ce¹⁴⁴, and Zn⁶⁵ were also found to be present. Uranium was found in small quantities, i.e., 71 μ g of uranium per square inch of filter surface. An analysis of an afterfilter is presently being made.

In an analysis of reactor water, 99 per cent of the gamma activity emitted came from the following: Mn⁵⁶, Sr⁹¹ plus Y⁹¹, and Na²⁴. The Mn⁵⁶ is formed from an (n, p) reaction on Fe⁵⁶; the Na²⁴, from an (n, α) reaction on aluminum; and the Sr⁹¹, from an (n, α) reaction on Zr⁹⁴. Mo⁹⁹ was found in the water as noted by Hall right after initial startup. This means that uranium could possibly be present. The source of this estimated uranium has been guessed to be from UO₂ attached to the plates during fabrication and/or corrosion testing, or to be from an actual contamination in the Zircaloy itself.

A method for determining the dissolved and undissolved solids in the water has been initiated. The residues obtained will be analyzed chemically and radiologically.

Pressure drops (taken at atmospheric pressure) across unit 2 before and after new filters were installed were as follows:

	Clean	Dirty
Pressure drop across prefilter, psi	1.6	8.9
Pressure drop across ion exchange, psi	6.2	4.7
Pressure drop across afterfilter, psi	2.5	1.7
Total psi	10.3	15.3
Flow rate, gpm	6.8	5.7

The retention tanks in the reactor building have been emptied innumerable times since startup of the reactor. All samples taken prior to discharge have been below the maximum permissible level, and hence all waste water has been discharged directly to the Laboratory sewer system. If by chance they should have to be cleaned up, a portable plug-in mixed bed is available. This bed is also used to keep the water in the fuel storage pit clean.

Some trouble has been experienced with the platinized full-flow conductivity cells used in the water system. It is presently believed that perhaps polished stainless steel would be more suitable for full-flow operation.

CONTROL ROD MAINTENANCE

Prior to each startup, a pre-operational rod drop test is required. Each rod must drop 48 inches within one second when the reactor pressure is at approximately 100 psig, and if this condition is not met corrective steps are necessary. To date this corrective step simply involved the replacement of one of the guide bushings. To make this replacement the reactor is cooled to a point where the pressure will be at atmospheric condition. Since the control rod has a backseating arrangement, there is no need for draining the reactor, and the bushing can be readily replaced. From all indications, the sticking which caused the excessive drop time was due to "curd" collecting on the bushing along with some deformation of the nylon guide bushing. One of the steps taken to possibly alleviate this condition has been to use teflon within a stainless-steel cylinder as the guide bushing and also to shorten the bushing.⁹

As explained in the preceding chapter, a pressure breakdown seal around the control rod extension drive breaks down the reactor pressure of 600 psig to atmosphere with practically no leakage. A leakoff connection does recover what leakage may occur and returns it to the vapor recovery system. Excessive leakage past the breakdown seal will raise the temperature around the bushing and this leakage is replaced by introducing reactor feed water (110 F) at a controlled rate through a connec-

tion on each thimble just above the reactor flange. During reactor operation the thimbles normally do not require cooling.

On the rare occasions that the thimbles require cooling, care must be exercised so that excessive cooling water will not place the reactor vessel under thermal stress about the thimbles. With a certain amount of care this presents no problem, for the temperature drop in the lower portion of the vessel can be observed in the control room, and the control of the thimble cooling water is also maintained in the control room.

On several occasions it was necessary to remove the control rod connector because of damaged threads. This piece contains the piston portion of the dash pot as well as the back seating arrangement which permits removal of the control rod drive mechanism without draining the reactor. In order to remove the connector, an ice plug had to be made in the thimble just above the reactor thimble flange (Fig. 5-24). Space limitations allowed but a shallow pan 2 in. deep to be placed around the thimble. After approximately three hours of freezing with a dry ice and alcohol mixture in the pan, an ice plug of about 8 in. in length was formed in the thimble. The control rod connector was then removed, repaired, and replaced, without losing any water.

STEAM-DRYER EMERGENCY COOLER

Two methods of determining the quality of the steam entering and leaving the steam-dryer emergency-cooler have been used, these being the throttling calorimeter and the radiation decontamination factor method. In both methods the quality of the steam entering the dryer was greater than 99 per cent at 20 Mw, and therefore the dryer was unnecessary. However, measurements made later of power above 30 Mw showed it to be both effective and necessary.

After operating the reactor for 1370 Mwd of heat power, the dryer was opened for examination. The radiation level was 5-6 mr/hr and was primarily gamma activity. The walls of the vessel were, in general, clean and had a brownish appearance but no deposit sample could be obtained. A sample of reddish-brown deposit was obtained from the surface of the cooling coils and this was primarily iron oxide. Spectrographic analysis of metals present as oxides indicated Al-3 per cent, Ni-1 per cent, Cr-1.5 per cent and others 0.1 per cent or less. However, since the dryer has a built-in emergency cooler, its presence in the system is necessary. Since the dryer and cooler are made of type-304 stainless steel, it was necessary to replace the 15,000 gal of tap water in the overhead storage tank with demineralized water. This was done to eliminate the chlorides in the water that could have a detrimental effect on the coils in the form of chloride stress corrosion.

TURBINE, CONDENSER, AND FEED PUMPS

Because of the experimental nature of the EBWR plant a number of startups and shutdowns are encountered. This places undue wear on some components that would otherwise normally be unaffected. A specific instance was the trip throttle valve on the turbine. Cam-type switch mechanisms were installed to replace the direct acting in-line type which were shearing the pins on the switches.

With regard to the condenser, most of the radioactivity (about 99 per cent) in the reactor water has a half-life of less than a day so that all the major activity (due to droplet carryover) has decayed after a day, and the level of radiation in the condenser has never exceeded 2 mr/hr after a day's decay.

Manholes have been opened and personnel have crawled into the condenser to obtain deposit samples and to closely examine the tubes and turbine exhaust casing. So far there has been no erosion on the external surfaces of any of the condenser tubes as determined macroscopically. This included the tubes exposed directly to the steam that is dumped through the desuperheater which is located directly below the condenser tubes. Sample deposits scraped off the stainless steel snap-on shields over the vulnerable tube areas are similar to the steam dryer samples.

The condenser hotwell and the water boxes are divided longitudinally to permit the continuation of plant operation in the event of a tube rupture. A conductivity cell is located just below each half of the hotwell so that increased conductivity in the respective outlet would warn the operator of a tube rupture so that the operator could isolate that particular half of the condenser. A test simulating a failure of a tube was made and the results indicated that this arrangement would perform satisfactorily.

Two feed pumps are installed in the reactor cycle; one is kept for standby while the other is in use. The upper bearings of both units failed and had to be replaced with bearings having bronze cages, whereas the original bearings had steel cages. Both failures occurred after each pump had operated intermittently for approximately 1000 hr. Several speculative reasons for their failure are (a) improper grease, (b) improper design or construction, and (c) excessive thrust on the bearings. Such failures could occur in any type of plant or system, but the point to be stressed here is that one of the motors was removed from the pump during operation and was removed from the plant after the first shutdown. The other motor was overhauled, re-aligned, and placed in service during plant operation. In both instances the standby pump was placed in service without shutting down the plant when the reactor bearings failed.

Two high-pressure full-flow condensate filter vessels are used in the feedwater line; one unit is a standby. Each unit contains 75 cotton fiber filter cartridges capable of removing particles of 2 micron size. After operating the reactor 400 hours at 20 Mw of heat, the pressure across the filters was 30 psi. This included three shutdowns in which the reactor was allowed to cool and also having the vacuum broken on the condenser. Inasmuch as the condenser shell is constructed of steel, breaking the vacuum on the condenser can almost be correlated with the pressure drop on the filter due to the collection of iron oxide. The 30 ppm of oxygen in the steam does not appear to contribute as much iron oxide during operation as is experienced by breaking the vacuum on the condenser.

Changing filters has proved to be no problem so far as radiation is concerned, for it amounts to about 5 mr per hr. The changing is usually accomplished during a shutdown but could just as well be made during plant operation.

RADIOACTIVE GAS DISPOSAL

Figure 6-27 is a schematic diagram of the gas handling facility for EBWR. All potentially radioactive gases pass through a Cambridge absolute filter and are blown out the building stack where they are diluted with the building ventilation air circulating at 3000 cfm.⁴

The total volume of air surrounding the reactor vessel is 350 ft³. The activation of argon in the air could potentially create an undesirable activity level within the plant. This chamber, therefore, is maintained at a negative pressure of 2 in. water and the exhaust air discharged outside the building. The leakage rate into the chamber at the operating pressure is 2½.

The air ejector gases are discharged through a flow meter directly to the filter and out the stack. The composition of the gases is 50 per cent hydrogen, 30 per cent oxygen, and 20 per cent inert. These gases are diluted to a hydrogen concentration of less than 1 per cent below the filter housing.

The air discharged from the plant is monitored for radioactive gas content. Gas samples are counted at regular intervals and the gas leaving the EBWR stack is below minimum permissible level for continuous exposure.

OPERATION OF EBWR AT HIGHER POWER LEVELS

From the time of the first BORAX experiments, the stability of boiling reactors has been vigorously investigated. Originally the agreement was

advanced that the formation of bubbles would create a void in the core, producing negative reactivity to offset that introduced by the control rods. The statistical nature of the violent boiling process would cause random variations in the negative reactivity which would be reflected in reactor neutron density and cause a ragged power output not suitable for use in a turbine.

The process forms a feedback loop,¹⁰ which can be seen in Fig. 6-28. The input is the rod reactivity control. From this, subtract the reactivity in voids, assuming a negative void coefficient of reactivity. The net amount of reactivity affects reactor power through the zero power reactor dynamic characteristic. As the power changes, the voids will change and the loop is completed by the effect of voids on reactivity. From this loop, it is immediately evident that a feedback system exists for boiling reactors which is conditionally stable.

In such a feedback system the steady-state change in power is proportional to the change in rod reactivity, if the feedback system is dynamically stable. On the other hand, if the feedback system is dynamically unstable, then loss of control will occur, and the rods will not be able to regulate the power output. The question which had to be answered was: which of these conditions would exist in practice? Analysis and syntheses of the components of the system led to many different conclusions, and it was not until BORAX-I was built and actually run at a steady power output that the feasibility of a boiling reactor was established. It was found that if the amount of reactivity in voids was above 2 per cent, the system tended to change from a condition of dynamic stability to dynamic instability. This limitation has been overcome in later designs.³

The second, or unstable condition, was christened "chugging" during the early experiments because the flux in the reactor dropped quickly to a low value as the core water was expelled. Then, as the water reentered, the core power rose again in a short period. The effect seen on the flux recordings is a series of rises and falls, with no apparent tendency to decrease in magnitude. Further details of these effects are described under the BORAX experiments.

Early EBWR operating experience at the design power rating of 20 Mw heat output indicated that the design was conservative and that the plant might have considerable power in reserve. For this reason, in the case of EBWR, a program of stability investigation was undertaken as the first experimental effort, the object being to determine the maximum stable operating power with the first core loading. This aim has been realized, together with the development of a method of extrapolation for the prediction of boiling reactor stability at higher power. In addition, a good theoretical model of the boiling water reactor has been developed, and

values obtained for the various power coefficients and time constants represented in the model. The project was carried out in four major phases which follow in chronological order.

1. Experimental determination of reactor power transfer function. A convenient way of expressing reactor kinetics is by means of a transfer function.¹¹ As used here, the term transfer function is defined as the amplitude and phase relation of the sinusoidal variation in flux or power output with respect to the sinusoidal reactivity input driving function. It is a very useful function for stability studies of feedback systems.^{12,13} Furthermore, the frequency response technique of measurement determines the function precisely enough to permit the separation of variables in a complex system, such as that representing the boiling reactor.

The first phase of the program was to measure the reactor power transfer function for many different values of power parameters, all within the limits prescribed by the safeguard report.¹⁴

The reactor power transfer function, P_1 , is simply the zero power function modified by power-to-reactivity feedback. The relation is

$$P_1 = \frac{G}{1 + GH_1} = \frac{\text{flux output}}{\text{reactivity input}}, \quad (6-6)$$

where G is the zero power function and H_1 is the power-to-reactivity feedback function for the specified reactor power parameters. This is shown schematically in Fig. 6-28. Several techniques are available for determining quantitative stability for the mode of operation which P_1 represents.^{12,13,15}

P_1 could be calculated and stability predicted if H_1 were known. However, H_1 is difficult to determine from thermodynamic data, and specific measurements of the open loop are not feasible. Another method is to measure P_1 and calculate H_1 , using the relation

$$H_1' = P_1^{-1} - G^{-1}, \quad (6-7)$$

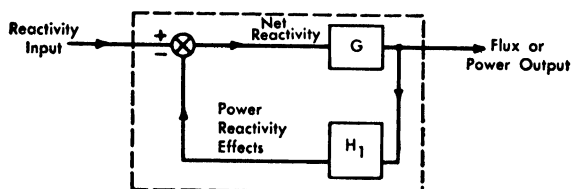


FIG. 6-28. Schematic of power-to-reactivity feedback.

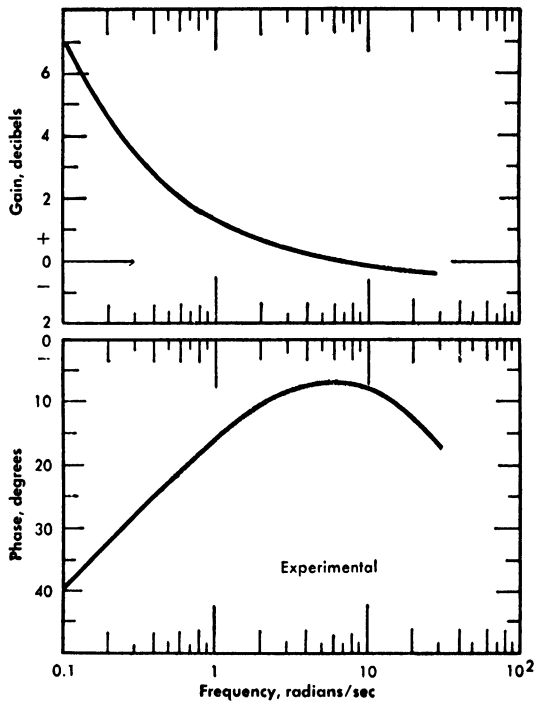


FIG. 6-29. Zero power frequency response.

which is simply a rearrangement of Eq. (6-6). The latter technique was used in the work reported here, and the first experiment was made to measure the G function (that is, P at zero power). The experimental points are shown on Fig. 6-29 where the solid line represents the theoretical curve. The "gain" ordinate represents $20 \log_{10}$ of the ratio obtained by dividing the flux or power output peak amplitude, N_p (expressed as a function of the average flux value, N_0), by the ratio of the input reactivity peak amplitude, Δk_p , to the total delayed neutron fraction, β_T . Mathematically, this may be shown as

$$\text{Gain} = 20 \log_{10} \frac{N_p/N_0}{\Delta k_p/\beta_T} \text{ db.} \quad (6-8)$$

The phase ordinate is simply the phase of N_p with respect to Δk_p , measured in degrees.

A parallel effort during this phase of the program was the devising of a theoretical model to assist in explaining the experimental results and

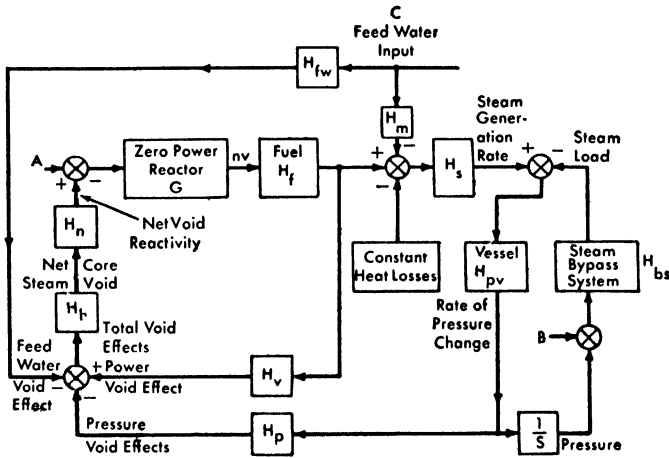
to guide further measurement.^{16,17} A simplified power-to-reactivity feedback diagram for the EBWR model is shown in Fig. 6-30. Each H in the diagram is a transfer function representation of a power effect. H_{tw} is the effect on voids which results from core subcooling by the feedwater. H_m is the heat consumed in feedwater heating after injection. C represents the feedwater injection which is held constant at the average rate of flow required for the power output level of a particular experiment. H_f is the heat transfer characteristic for heat flow from the generation point inside the fuel element to and including its absorption in the coolant. It includes the factor relating heat generation and flux. H_v is the relation between power and steam or void formation internal to the core. H_s determines steam generation from the net heat. H_{pv} represents the pressure vessel characteristics which determine rate of pressure change when steam generation rate and steam load are not equal. H_{bs} is the dynamic characteristic of the automatic system.¹⁸ Reactor steam load was controlled by manual bypass so H_{bs} had zero input. H_p is a complex function which represents the void effects due to pressure. These include temperature coefficient, flashing, and void boundary change. H_h is the natural recirculation dynamic effect on total void which determines the net core steam void. H_n is the relation between net void reactivity and net core steam void.

Investigation of EBWR pressure effects indicated that at frequencies in excess of 2 or 3 radians per second the pressure feedback loop, H_p , can be neglected, leaving only the power void loop, H_v , since feedwater input, G , is held constant. Preliminary tests without automatic pressure control showed that the frequency of maximum resonant peak response was in excess of 5 radians per second, making the peak response almost independent of pressure variation. Then system stability, which can be expressed in terms of this peak response, can be determined without automatic pressure control, using only a reactivity driving function at A and measuring flux, nv . The latter method, then, was used on a majority of the tests.

Physically, P must become equal to G at high frequencies, since the power feedback, H , becomes very small and GH becomes small with respect to one. Thus:

$$P_1 = \frac{G}{1 + GH_1} = G \quad \text{for } GH_1 \ll 1. \quad (6-9)$$

As P_1 approaches G , phase errors of a few degrees and amplitude errors of several per cent change the calculated value of H_1 by very large



amounts. Therefore, all measurements must be made with the highest possible accuracy. The frequency of the forcing function must be accurately determined, particularly where the transfer function phase shift slope is high.

The reactivity forcing function was generated by a special rod drive which replaced the regular central control rod drive. A variable displacement bell crank was driven by a combination of mechanical and hydraulic transmissions to cover a frequency range of 0.01 to 25 radians/sec. The connecting rod was very long compared to crank displacement; therefore, the control rod stroke was an accurate sinusoid, producing the desired sinusoidal reactivity variation. Static pressure and weight forces in excess of 600 lb and inertial forces of several hundred pounds were neutralized by a spring which biased out the static forces and resonated with the inertia to reduce drive horsepower requirements to a reasonable value.

The amplitude of the reactivity input was limited by both the reactivity drive power requirements and the necessity for maintaining a small input to preserve linearity.¹³ Thus, the output flux signal often had a peak value of less than 5 per cent of the average flux value. The "boiling noise" in the reactor flux output ranged up to a peak value of 5 per cent, or more, of the average flux value, resulting in signal-to-noise ratios of less than one in some cases. In order to measure to ± 5 per cent in amplitude and ± 2 degrees in phase in such unfavorable circumstances, it was necessary to employ a wave analyzer which measured only the fundamental component of the flux wave and compared it with the reactivity forcing function. Operation was based on a correlation tech-

nique in which the reactor flux signal is measured by an ion chamber and then multiplied by a vector which is correlated to the reactivity driving function in both amplitude and phase. It is possible to select a correlation which reduces the fundamental component of the multiplied signal to zero, leaving only a small second harmonic plus any noise present in the original flux signal. Feedback control was used to adjust the amplitude correlation, and phase shift correlation was attained through the use of a mechanical differential. The multiplied signal was passed through a band-pass filter adjusted to pass only the fundamental frequency. This strongly attenuated everything but the fundamental component which then was reduced to zero by proper correlation adjustment.¹⁹

The analyzer measures the ratio N_p/N_0 as determined by the peak value of the reactivity input function. It is necessary to measure the peak reactivity variation to establish absolute gain as indicated earlier in Fig. 6-28. The period method, which was used for the curve of Fig. 6-29, cannot be used in the power tests since the boiling reactor does not generate a period when a rod is pulled out; instead it seeks a power level which generates just enough reactivity in steam voids to balance the excess reactivity inserted by the rod withdrawal.

Calibration was attempted by several means including a rod exchange method which used a hot nonboiling reactivity calibration available for the rod bank from earlier experiments.¹ The data, when normalized by this means, consistently produced power transfer functions which did not return to the zero power transfer function at high frequencies as required by the physical nature of the feedback mechanism. It seemed evident that rod worth was a function of the void condition in the reactor so the data were normalized by arbitrarily requiring that the power transfer function equal the zero power transfer function at a frequency of 25 radians per second. This method was subsequently justified by calculations on the H function.

A possible method of measuring reactivity in voids without resort to rod calibrations employs a simplified version of the above process. A static measurement of P_1 is made, followed by a measurement at a frequency sufficiently high to ensure that P_1 is equal to G . The absolute static value of P_1 is obtained by calculating the value of G for the frequency above and multiplying it by the ratio of the two measured values. It can be demonstrated that the value obtained is $1/H_1$:

$$P_1 = \frac{G}{1 + GH_1} = \left(\frac{GH_1}{1 + GH_1} \right) \left(\frac{1}{H_1} \right) = \frac{1}{H_1}, \quad \text{where } GH_1 \gg 1,$$

or

$$H_1 = P^{-1}. \quad (6-10)$$

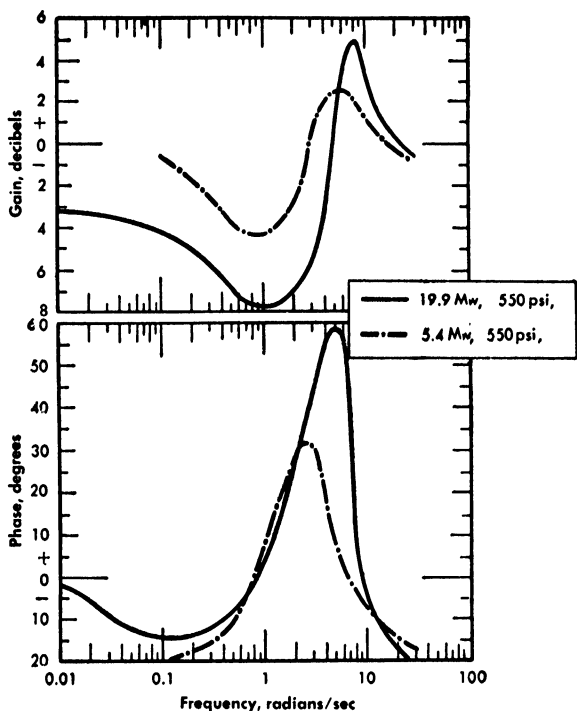


Fig. 6-31. Effects of power on frequency response.

The measurement is incremental by definition, so the slope of the void reactivity vs power curve is obtained by dividing H_1 by the power in megawatts. The reactivity in voids for a specified power may be obtained by plotting and integrating slopes measured for several lower powers, all at constant pressure. The curve obtained by the above means is essential for predicting reactor kinetics at increased powers, since it provides the gain factor for the feedback loop. A demonstration of this will be given later.

Typical power transfer functions for EBWR are shown in Figs. 6-31, 6-32, and 6-33. The curves of Fig. 6-31 demonstrate the effect of a 4-fold increase in power level at constant pressure. Some resonance peaking occurs in both gain curves, the amount increasing with increased power. The gain curve shape above 1 radian per second is largely a result of the power void effect, H_v , acting without H_p (see Fig. 6-30). As the frequency decreases below 1 radian per second, H_p , the pressure void effect becomes large with respect to H_v , causing the net feedback to decrease. H_p continues to increase with decreasing frequency and causes the curves to increase in gain as the low frequency end of the spectrum

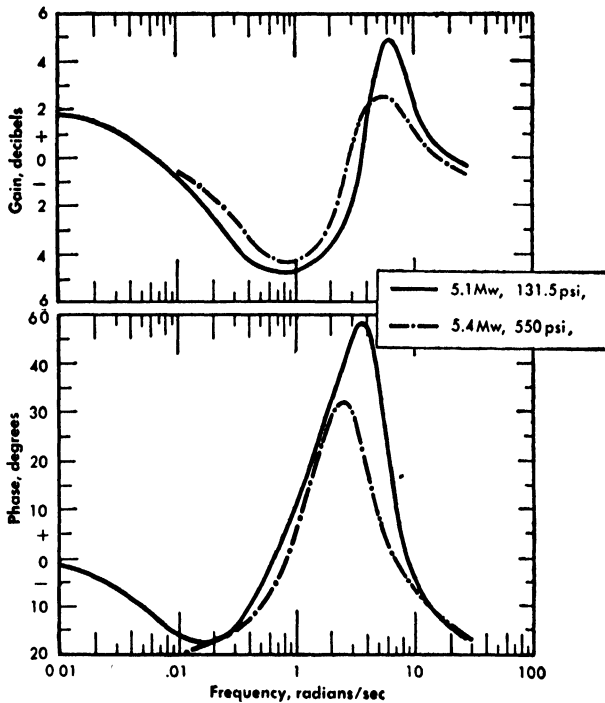


FIG. 6-32. Effects of pressure on frequency response.

is approached. If pressure were held constant so as to reduce H_p to zero, the curves would rise to an asymptotic value only slightly larger than that at 1 radian per second. The value of gain in the dip at 1 radian per second decreases with rising power because of the formation of more core steam voids which increases H_r . The frequency at which peak response occurs rises with increased power largely as the result of a reduction in the time constants associated with H_f and H_v . The time constants are altered by changes in film coefficient as flow and void content change in the core. At the high end of the frequency range both H_v and H_p become ineffective and the curves join as they return to follow the G function.

The curves of Fig. 6-32 demonstrate the effect of a 4-fold decrease in reactor design pressure while holding the power constant. The curve representing operation at 5.4 Mw and 550 psi is the same curve that is shown as the low power example in Fig. 6-31, and requires no comment. The second curve represents operation at one-fourth design pressure. A higher resonance peak is in evidence and the frequency at which peak response occurs is higher just as was the case for a power increase. In fact, this curve is strikingly similar to the 20 Mw curve of Fig. 6-31

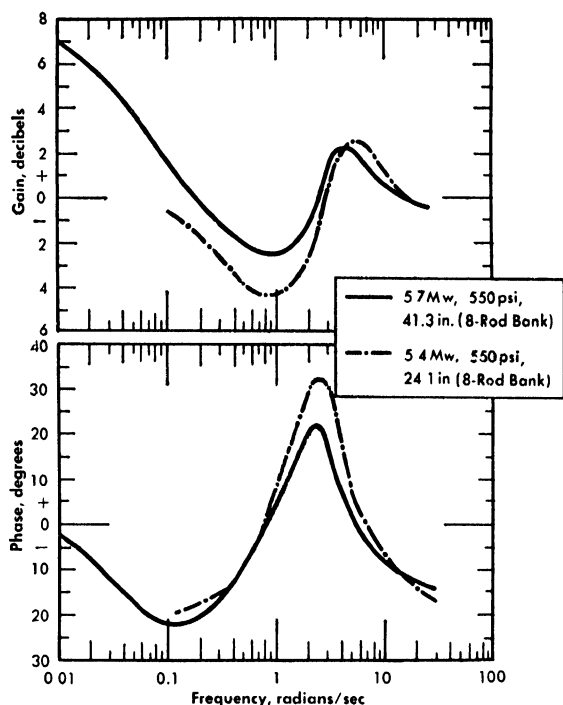


FIG. 6-33. Effects of rod bank position on frequency response.

throughout the resonance region. The explanation lies in the fact that the specific volume of the steam in the core has been increased by the pressure reduction until the void condition is approximately the same as that which exists at 20 Mw and design pressure.

Another way of stating the situation is that the open loop (GH_v) gain for the two cases is nearly equal through the high frequency region, because the reactor forward gain, K_G , is directly proportional to the power change while the feedback gain, K_H , is roughly inversely proportional to the pressure change. The correspondence is limited to the high frequency region where H_v is the controlling factor. The peak responses occur at nearly the same frequency for both cases indicating that the feedback time constants are very nearly equal. These observations led to the development of a method of extrapolating the transfer function which is discussed later.

The effective core length was increased by adding boron to the reactor water so the control rods could be withdrawn. Two power transfer functions which were measured with all reactor conditions the same except for

core length are changed in Fig. 6-33. The height of the peaks was the same while the closed loop gain, P , at low frequencies indicated more reactivity in voids for the shorter core (no boron). These two facts taken together suggest that the heat transfer time constants are longer for the long core, possibly due to decreased flow rate. The resulting increase in open loop phase shift would then offset the decrease in open loop gain that occurred as a result of reduced voids, and the same peaking would appear in both cases.

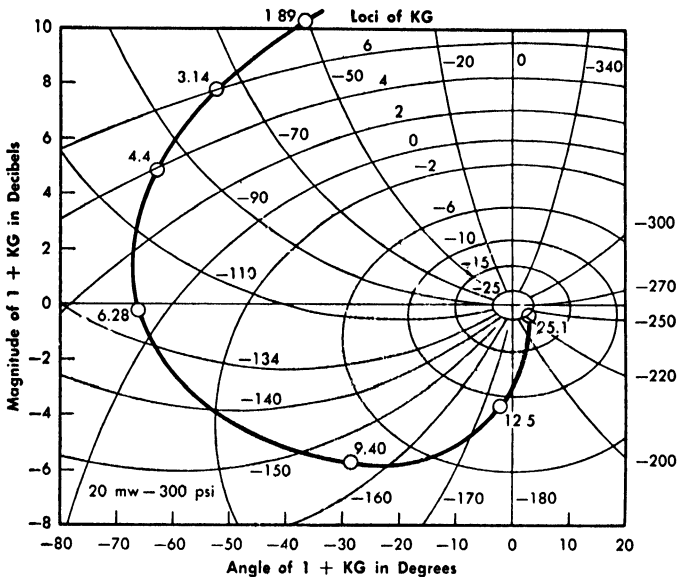
Up to this point nothing has been said about the phase curves that accompany each of the gain curves to give the complete transfer function. The accuracy of the phase measurement permitted its use as a check on the shape of the gain curves. Application of the minimum phase shift theorem requires that the phase shift be equal to the slope of the amplitude curve except as modified by a weighting function.²⁰ The effect of the weighting function is frequently nearly unity so that the zeros in the phase curve correspond to the maxima or minima of the amplitude curve while maxima and minima on the phase curve correspond to maximum slopes of the amplitude curve.

The results obtained in this first phase of the program established that the frequency response of a boiling reactor could be measured with sufficient accuracy to make an analytical investigation worth-while. It also laid the groundwork for a method which permits a safe approach to power by utilizing reduced pressure measurements.

2. Calculation of experimental feedback functions. The second phase of the program consisted of an analysis of the experimental data to evaluate the importance of the various model parameters and to determine how effectively the model represented the reactor. The first problem was to determine the best method for calculating the experimental feedback function from the power transfer function data.

Uncertainty existed as to the absolute gain of the power transfer function, because the peak value of the reactivity input function could not be determined with sufficient accuracy. Also small phase or amplitude errors in the power transfer function appear greatly magnified in the calculated feedback transfer function. This effect is greatest at higher frequencies where the feedback function is the difference between two values both much larger than it is.

Finally, a conformal transformation plane relating GH_1 to $1 + GH_1$ was devised to inspect the curve shape at high frequencies and permit smoothing of data where obvious discrepancies existed. The plane proved a very useful tool in resolving the gain uncertainty mentioned earlier,

FIG. 6-34. Calculation of GH .

because gain change on the plane is accomplished by simply slipping the entire curve up or down along the $1 + GH_1$ magnitude coordinate.

As mentioned earlier the pressure feedback effects are very small at frequencies above 2 radians per second so it is possible to neglect the H_p loop and consider only the power void loop, H_v , above that point. The simplification of the analysis by this means ultimately made it possible to represent the feedback, H_v , by a function with a constant numerator and a cubic denominator. The values of the function were determined by a process involving successive approximations. First a calculation of GH_1 was made, using the conformal plane and a first approximation for the power transfer function gain. A typical case (20 Mw at 300 psi) is shown in Fig. 6-34. Then G was subtracted and the H_1 that resulted was plotted in the form of Bode phase and attenuation diagrams. An asymptotic representation was fitted to the experimental curves with the number and location of the time constants being determined chiefly by the phase diagram. The resulting analytic feedback function was calculated, and then compared point by point with the calculated experimental feedback function. The above step was repeated until no further improvement could be obtained in the phase diagram. At this point the amplitude and phase difference between the experimental function and the analytic function were determined. They indicated in some cases that the experimental function did not obey the minimum phase shift theorem (men-

tioned earlier in connection with the power transfer function). It was found possible to correct the condition by selecting a new value for the power transfer function gain, slipping the curve in the conformal plane to correspond to the gain change, and then repeating the above curve fitting procedure. The H_v function for 20-Mw operation at 300 psi is in the Laplace transform:

$$H_v = \frac{10.1 \text{ db}}{(1 + 0.23 \text{ s})(1 + 0.14 \text{ s})(1 + 0.06 \text{ s})}. \quad (6-11)$$

The above three time constants represent the time required for the heat to pass from the fuel element into the coolant and the time required for the steam generated by that heat to travel to the core exit.

An analog computer study of the model was made at the Princeton Computation Center upon completion of the work on H_v functions. The purpose was to evaluate the effects of the pressure dependent portions of the model and determine how well the model represented the reactor in this area. One finding of the study was that the temperature coefficient was effective only below 0.01 radian per second so this portion of the pressure feedback was dropped out, raising the possibility of an analytic expression for the entire power feedback expression. Another result was verification of the assumptions made earlier to obtain the H_v functions. However, the model still did not represent the reactor very well at frequencies below 1 radian per second.

An analytic expression for the feedback transfer function of the model, using a quadratic numerator and quintic denominator, was developed as a follow-up of the analog study. Constants were evaluated by means of the theoretical model, and the IBM-650 was used to determine the power transfer function. By this means it was found that the discrepancies at low frequencies were related to the part of the feedback transfer function representing the valve orifice. It was clear that this effect was small and limited to frequencies below 0.01 radian per second.

A new expression with one numerator time constant and 4 denominator time constants was set up and a successful representation of the reactor was obtained. The complete feedback transfer function for the 20-Mw 300 psi condition is shown as a typical case:

$$H = \frac{+ 1 \text{ db } (1 + 7.7 \text{ s})}{(1 + 3.2 \text{ s})(1 + 0.23 \text{ s})(1 + 0.14 \text{ s})(1 + 0.06 \text{ s})}. \quad (6-12)$$

The two time constants added to Eq. (6-11) to obtain Eq. (6-12) are determined by complex expressions containing several of the model pa-

rameters. The experimentally derived H functions were used to calculate values for these parameters and to correct the model. The model at this point was an accurate representation of the reactor and could be used to determine the transfer function and stability at increased power.

3. Extrapolation of model by computer to determine stability at 40 Mw.

The discussion in connection with Fig. 6-32 described the effects of pressure variation, and showed that reduction in pressure increased the specific volume of the steam in the core to produce a void condition nearly the same as that for a power equal to the original power multiplied by the reciprocal of the pressure ratio. This effect was made use of to obtain the time constants associated with heat transfer, steam transit, and water recirculation at 40 Mw and 550 psi by making measurements at 20 Mw and 300 psi. Coefficients depending upon steam properties were extrapolated by direct calculation. It was necessary to extrapolate the reactivity in voids versus power curve since the reactivity void coefficient, which set the gain of the feedback transfer function, is the slope of this curve. The extrapolation was performed on the curve shown in Fig. 6-35 which was determined by the hot nonboiling rod calibration. The 40-Mw condition was obtained by using rod position to evaluate the reactivity change between 20 Mw 600 psi and 20 Mw 300 psi and then correcting for the temperature reactivity effect. A

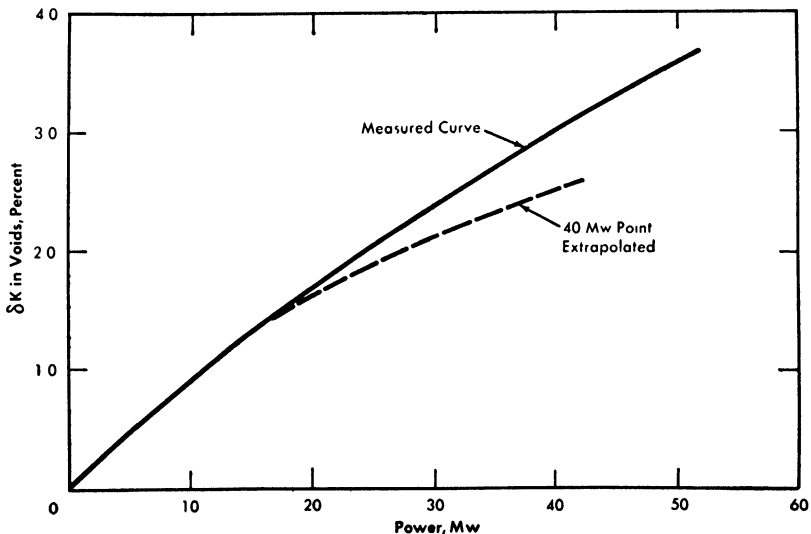


FIG. 6-35. Variation in void radioactivity with power.

smooth curve was drawn through the new point. The void coefficient at 40 Mw was evaluated by taking the ratio of the slope at 40 Mw to that at 20 Mw and then multiplying it by the coefficient which had been measured for 20 Mw 600 psi. Subsequent measurements proved the estimate to be within 16 per cent of the correct value. It is interesting to note that the steady decrease in void coefficient, as power is increased, could eventually lead to increased stability through reduction of the total open loop gain.

All of the above data were combined by means of the model to produce the required 40-Mw feedback transfer function. The IBM-650 was used to combine it with the zero power function to produce the power transfer function prediction for 40 Mw shown in Fig. 6-36. The 40-Mw stability was almost as good as that at 20 Mw with a small decrease in the GH phase margin. The resonance peak magnitude was only about 2 db above that at 20 Mw. At this point a careful survey of the plant was made to determine feasibility of heat removal, fuel element burnout safety factor, and possible hydrodynamic instabilities. The results were combined in an application to the Atomic Energy Commission for permission to operate experimentally at powers up to 50 Mw with the present core.

4. High power operation. The Atomic Energy Commission approved the application after a careful study of the data that were presented to support it. In the meantime preparations to instrument the increased flows had been completed and the plant was ready for the test. On December 2, 1957, the first high power run was made using the reactivity generator to perturb the reactor at 5, 10, 20, 30, and 40 Mw. The maximum power attained on this run was 46 Mw, because the hotwell level was behaving erratically and there was a possibility that the feedwater pumps might be cut off by the low level interlock, which protects the pumps from cavitation. On December 9, after the hotwell and turbine problems were solved, the reactor was brought to xenon equilibrium with rods all out, and transfer functions were run at 35 Mw 550 psi and at 25 Mw 300 psi. Automatic steam bypass and feedwater control was used throughout both these tests to gain operating experience at the increased power. On December 18 the above transfer functions were rerun to secure the low frequency end of the spectrum with the plant on manual control. The experimental data for the 35 Mw 550 psi run are plotted as circles on Fig. 6-36. The 35- and 40-Mw comparison can be made since a 10 per cent difference in power does not change the curve much. Finally on December 23 a short run was made at 50 Mw without xenon in order to test the mechanical capabilities of the plant. In addition the void reactivity curve shown in Fig. 6-35 was obtained. A transfer function run

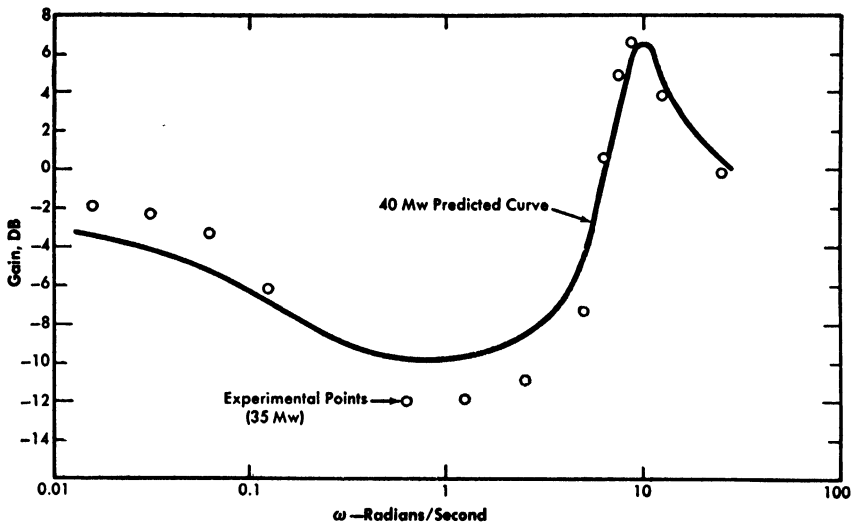


FIG. 6-36. Power transfer function prediction, P for 40 Mw.

was made but lack of plant equilibrium prevented accurate measurements. The open loop GH functions were calculated for the various powers, and the log polar plane (Fig. 6-34) was used to determine gain and phase margins.

The plant was returned to normal 20-Mw operation following the above run. Plans are being made for a shutdown to give the turbogenerator its first yearly inspection and to determine a core rearrangement that will permit running at 50 Mw in xenon equilibrium.

OPERATION AT 61.7 MW OF HEAT

Thorough analysis of the studies described above indicated the possibility of operating EBWR at even higher power than the 50 Mw run made on December 23, 1957. The studies predicted that the system would be stable to a power at least as high as 65 Mw of heat with the existing original 4 by 4-ft core. Accordingly plans were made to again step up the power level, and on March 20, 1958, EBWR was successfully run steadily at 61.7 Mw of heat. Clearly this was above the level which could be held at equilibrium xenon of 40 Mw, but operation at this level served to demonstrate the stability of the core and flow system. The 61.7-Mw level was limited only by the feedwater pump delivery rate.

In this high power run, the power was raised to approximately 50 Mw and the plant held at equilibrium for about 10 min. A steam meter refer-

ence value for 55 Mw was established and power raised to that point and held for approximately 15 min. Plant conditions, including temperature, flow, and pressure have an effect upon the absolute power rating. Upon taking an accurate calibration, it was found that the plant was running at 57 Mw of power.

Flux conditions were normal, no disturbing repetitive oscillation was seen on the high-speed flux instruments and the power was subsequently raised to the referenced 60-Mw value. Power was held constant for about 10 min and a calibration showed that the power was about 60.5 Mw. Again, the flux was observed to be quite normal and the power was raised again. At this point the feedwater valves were run fully open and the pumps were delivering the maximum possible amount of feedwater.

Under these conditions, control was possible only by varying the flux in such a way as to try to prevent a variation in water level, making the reactor power output equal to the delivered water rate. The steam meter was at such a high reading that an accurate power calibration could not be made.

During this run, it was very likely that the power of the reactor was as high as 64 Mw; however, the value officially read for steady calibration was 61.7 Mw. This value was held for about 10 min. Readings and charts were marked and the power was brought back to the normal 20 Mw.

The general appearance of the accurate flux instruments during the highest power runs was satisfactorily steady. However, a later look at the chart showed that the flux oscillation was beginning a cyclical variation.

In requesting permission from the AEC to run at 65 Mw, it had been predicted that at that power the system would reach the beginning of the unstable region. The operators are of the opinion that they were on the very edge of this region and that it is unfortunate that the pump capacity was not greater so that the region of oscillation could have been probed into more thoroughly. Had this region been reached, there would have appeared a very pronounced repetitive variation in flux in a magnitude near 20 per cent of the total readings. The largest variation observed was closer to 10 per cent for these tests and was not truly oscillatory.

As a result of these tests it was concluded that:

1. The reactor operation was satisfactory at 60 Mw.
2. The amount the power level was able to go over 60 Mw and the disturbances inflicted on the plant at this point indicate that satisfactory power operation could have been maintained at 60 Mw had the excess reactivity been sufficient to override the xenon.
3. Having reached the stable 60-Mw performance point, the redesign of the plant with a 5 by 5-ft core to deliver 100 Mw is firmly estab-

lished as correct and the plant output ability to that point can be extrapolated with very good assurance of success.

FORCED RECIRCULATION

One method of increasing reactor power which has frequently been suggested is to recirculate the core water with an external pump. From the foregoing discussion, it is clear that at present, relatively high density power can be achieved without this added plant complication in the case of H_2O moderated and cooled systems.

When this equipment is added, the simplicity of the boiling plant no longer exists. The pump or pumps and associated piping carry radioactive water and cannot be serviced easily. Shielding must be provided under operating conditions.

Forced circulation is, on the other hand, mandatory for the D_2O cooled and moderated core. It is clear that if forced circulation is used, it must pay for itself either by making the more attractive D_2O reactor possible, or by increasing the power output per dollar invested significantly.

The installation of forced circulation and equipment on EBWR will have cost about \$400,000 and the capacity is about 10,000 gpm. The amount of recirculation provided here is about the amount obtained at 30-Mw heat by natural circulation. To reach the equivalent of 100-Mw heat capacity now in prospect would require some \$1,000,000 worth of equipment.

The cost factor associated with higher power through forced recirculation must be carefully considered and not simply accepted as a low cost means of increasing power potential.

It may be very unlikely that an advantage can be gained from the use of forced recirculation if no heat is removed from the water during circulation cycle. When one considers the cost of heat exchangers and boilers which are required to make the removal of useful heat possible in the recirculating system, the plant cost rises rapidly. The tolerable limit is, of course, the point at which the cost of increased heat removal maintains the installed cost per kilowatt at the same value which prevailed before any forced circulation was considered. At this point, one could have accomplished the same objective by simply building a second reactor.

ECONOMIC IMPLICATIONS

While the successful development of a method for predicting reactor performance and providing a safe approach to higher power is important in a technical sense, it is much more important as a step toward the ulti-

mate objective which is to improve the economic picture of nuclear power.

Experience and experimental work have shown that an EBWR size plant, if designed for a single purpose, that is, power production, should have the over-all specifications shown in Table 6-12, where the projected quantities are compared with the present experimental plant.

The construction cost of the present plant is approximately \$4,600,000, including experimental features and facilities. The development cost incurred was about equal to this figure. Increased capacity in the turbo-generator and auxiliaries would cost about \$1,500,000 more if installed when constructed. The total would then be about \$6,000,000 in round numbers, or about \$200/kw installed capacity. The present plant is only about 25 per cent efficient and this would be improved by the single purpose design objective to account for 30 Mw used as an output figure in the projected column in the table. Coal or oil plants in this size would cost \$180/kw or more.

Over-all economics must include amortization, operation and maintenance, and fuel costs. Using the costs listed in an article by L. Link as a basis, Table 6-13 can be made up, keeping the same 80 per cent plant factor used by Link.²²

The basis for computation of the projected figures is explained as follows:

Capital cost. A direct ratio was taken using the expected cost increase and power increase, as explained above.

Operation and maintenance. This will be speculative until a single purpose plant is built. A cost minded operating organization would keep this number low. The projected reduction is to only $\frac{1}{4}$ of the 4.5 Mw

TABLE 6-12
PRESENT AND PROJECTED COSTS

	Present	Projected
Heat output, Mw	20	100
Electrical (net), Mw	4.5	30
Core length, ft	4	5
Core diameter (nominal), ft	4	5
Fuel U ²³⁵ , kg	76.4	250
U ²³⁸ , kg	5594	10,000
Control rods (number)	9	9
Pressure vessel	7 ft dia. 23 ft long	No change
Coolant	H ₂ O	H ₂ O
Coolant circulation	Natural convection	Natural convection
Burnup (reactivity)	~1.5%	~10%

figure. The total cost would be the same regardless of the gross electrical plant output and the resulting per-kwh cost is therefore somewhat arbitrary.

First core fabrication. The present core cost was increased on the basis of the 148 elements for a 5-ft core ratioed to 114 for the 4-ft core and then reduced by the power output ratio. No savings in fabrication cost were postulated, although some reduction in cost could be expected, if the fuel fabrication were done on a commercial basis and especially if the prospect for repeated or volume business was an incentive.

Fuel charges and annual fabrication. Fuel charges include credit for unused uranium and for plutonium. The reduction in unit cost is predicted on longer burnup than that used by L. Link of 2000 Mwd per ton. A value of 4000 to 6000 could be reached in a single purpose nonexperimental plant. It is indeed likely that 10,000 Mwd/ton exposure could be obtained by the use of burnable poison mixed with fuel design or poison inserted in the form of strips which could be removed to replace reactivity. Spiking with enriched fuel assemblies would also increase exposure time and it is possible that spiking with Pu fuel could be considered in a future core.

Economy in the use of boiling reactors will depend upon the development of economical fuel cycles, not upon the cost per kilowatt of installed capacity. This is evident from the fact that at 100 Mw heat or 30 Mw electrical, an EBWR type unit would cost about \$200/kw to construct. The need for enrichment is also established in the use of water-cooled reactors. The future objective is to find a fuel cycle by which the reactor fuel can be operated for 10,000 Mwd/ton or 1 per cent average burnup, with the least amount of enriched feed material.

At least two paths of investigation are available. The first would be a core material probably in oxide form, enriched to about 3 per cent in

TABLE 6-13
PRESENT AND PROJECTED KWH COSTS, MILLS/KWH

	4.5 Mw (elec)	30 Mw (elec)
Capital at 15%	22	4.5
Operation and maintenance	8	2
First core fabrication	4	1
Fuel charges	7	3
Annual fuel fabrication	11	5.5
	52	16

U^{235} with burnable poison reactivity holddown which could be left in place throughout the life of the core. The second is spiking with enriched elements at intervals throughout the life of the core to maintain reactivity.

In the future plans for EBWR, both methods are proposed for investigation. The first project will be to raise the heat output to 100 Mw by adding heat removal capacity. Because of the nature of the Argonne load, the turbogenerator will not be increased in size. Heat removal will probably be by heat dump methods to provide flexibility.

At higher output value, the average burnup rate will be high enough to facilitate fuel cycle studies in a reasonable time. At the original value of 20 Mw heat, the fuel cycle study process would have been too slow to provide useful power plant economy information.

Experience with EBWR so far indicates that the boiling reactor offers a low investment means of producing electrical power from nuclear fission. Progress to date and the assurance of a vigorous continuing program make the natural circulation boiling system attractive. The probable advance in the associated engineering and fuel technology mean that a physical facility, similar to the EBWR plant will not become obsolete for many years.

REFERENCES

1. J. A. THIE, EBWR Physics Experiments, ANL-5711, July, 1957.
2. J. A. DESHONG, JR., Styrofoam Simulation of Boiling and Temperature Effects in the EBWR Cold Critical Experiments, ANL-5697, March, 1957.
3. J. M. HARRER, EBWR Operational Data and Experience. Paper presented at Symposium on Peaceful Uses of Atomic Energy, Sydney, Australia, June 2-6, 1958.
4. E. A. WIMUNC and V. C. HALL, JR., Maintenance Experience of the Experimental Boiling Water Reactor. Presented at the 1958 Nuclear Congress, Chicago, Illinois, March, 1958.
5. J. M. HARRER, A Review of Boiling Water Reactor Performance. Paper submitted for presentation at 5th International Electronic and Nuclear Exposition and Congress, Rome, Italy, June, 1958.
6. R. D. COOK, Fission Product Monitoring and Radioactivity in EBWR. Paper presented by Walter Lipinski at Boiling Water Conference, Oslo, Norway, June, 1957.
7. V. C. HALL, Leakage and Radioactivity in the EBWR, TID-7535, p. 92, May, 1957.

8. R. E. BAILEY, EBWR Water Treatment, Decomposition, and Decontamination Factor, TID-7535, p. 98, May, 1957.
9. W. J. KANN, EBWR Control-rod Drive Mechanism, TID-7535, p. 50, May, 1957.
10. J. A. DESHONG, JR., Upping EBWR's Power, *Nucleonics*, **16**(6), pp. 68-72 (June, 1958).
11. J. M. HARRER, R. E. BOYAR, and DARWIN KRUCOFF, Transfer Function of Argonne CP-2 Reactor, *Nucleonics*, **10**(8), pp. 32-36 (August, 1952).
12. H. CHESTNUT and R. W. MEYER, Servomechanism and Regulating System Design. John Wiley & Sons, Inc., 1951.
13. J. M. HARRER and J. A. DESHONG, JR., Discontinuous Servo for Control Power Reactors, *Nucleonics*, **12**(1), pp. 44-51 (January, 1954).
14. J. M. WEST, J. R. DIETRICH, A. S. JAMESON, G. A. ANDERSON, J. M. HARRER, and H. F. BRUSH, Hazard Summary Report of the Experimental Boiling Water Reactor (EBWR), ANL-5781, November, 1957.
15. H. W. BODE, Network Analysis and Feedback Amplifier Design. D. Van Nostrand Co., Inc.
16. E. S. BECKJORD, Dynamic Analysis of Natural Circulation Boiling Water Power Reactors, ANL-5799, March, 1958.
17. J. A. THIE, Transfer Functions of Boiling Heating Elements. Unpublished work.
18. J. A. DESHONG, JR., and E. S. BECKJORD, Frequency Response Measurements of the EBWR Automatic Steam By-pass Valve Control System, ANL-5726, June, 1957.
19. J. A. DESHONG, JR., Power Transfer Functions of the EBWR Obtained Using a Sinusoidal Reactivity Driving Function, ANL-5798, December, 1957.
20. H. W. BODE, Relations between Attenuation and Phase in Feedback Amplifier Design, *Bell System Technical Journal*, July, 1940, p. 421.
21. B. MAXON and J. A. THIE, Theoretical Basis of Reactor and Control-rod Reactivities, ANL-5711, July, 1957.
22. Experimental Boiling Water Reactor, a special report, Vol. 15, No. 7, pp. 52-64, July, 1957.

CHAPTER 7

VBWR—THE VALLECITOS BOILING WATER REACTOR

The Vallecitos Boiling Water Reactor is one of the major research facilities of the General Electric Company's Vallecitos Atomic Laboratory at Pleasanton, California. It is furnishing steam to a turbine-generator, owned and operated at the site by the Pacific Gas & Electric Company. The 5000-kw of electrical energy produced is being distributed for commercial use by the PG & E system. In this respect it was the first privately-financed nuclear power reactor to go into operation in the United States.

The history of the Vallecitos development is of interest since it has a bearing on the design and construction of the Dresden Nuclear Power Station of the Commonwealth Edison Company described in a subsequent chapter in this book. Early in 1955, the General Electric Company signed a contract with the Commonwealth Edison Company of Chicago and the Nuclear Power Group to build the 180,000-kw Dresden Station near Chicago, Illinois.¹

General Electric officials believed that a small development plant was needed to assure the success of the full-scale Dresden Station, and in the spring of 1955 a proposal was assembled for construction of a 2500-kw reactor to be used to develop operational data for application to the Dresden design.

During the fall of 1955, Samuel Untermyer, who had been closely associated with the early development of boiling water reactors at the Argonne National Laboratory, and who conceived the dual-cycle boiling reactor after joining General Electric in 1954, was given the responsibility for the developmental reactor project. Lester Kornblith and Luther Welsh were placed in charge of reactor and building construction, respectively, under Untermyer.

Shortly thereafter, the size of the proposed reactor was increased to 5000 kw, and the Bechtel Corporation was chosen as engineer-constructor for the project. Negotiations for the purchase of 1600 acres of land near Pleasanton, California, were started, and in December, 1955, General Electric announced plans for construction of its Vallecitos Atomic Laboratory there.² The developmental boiling water reactor was to be a major facility of this laboratory. Shortly thereafter, the Pacific Gas & Electric Company, a member of the Nuclear Power Group, agreed to participate in the developmental reactor project by supplying a turbine-generator

facility which would utilize the steam produced by the reactor. The utility company agreed to contribute \$572,000 toward the turbine-generator installation, and to operate the generating unit as part of its existing electric network.

A construction permit for the reactor was issued by the Atomic Energy Commission in June, 1956, and work on the project was started immediately. Construction was completed by mid-June, 1957, and on August 3, 1957, the reactor first went critical. The 5000-kw generator was first paralleled with the PG & E system on October 24, 1957. As a result of this short, one year building period, the total cost of the reactor plant was substantially less than the amount originally estimated.

Besides its immediate use for testing components of the Dresden reactor, the Vallecitos Boiling Reactor will implement the over-all objective of increasing the steam output obtainable from boiling reactors. This reactor is outstanding in three respects:

Performance. Principally because of its high operating pressure, it has been possible to operate at high power density without excessive steam void content. Average power densities throughout the core exceeding 50 thermal kw per liter have already been obtained with natural circulation; and it is anticipated that the reactor will operate with natural circulation at power densities of 100 kw per liter.

High reactor steam pressure. The reactor system normally produces 1000 psi steam. This increase in steam pressure over the earlier boiling reactors gives a higher thermal efficiency and results in a small steam void content for the same power density as well.

Experimental flexibility. The reactor is completely accessible for replacement of any internal parts. Provisions are included for operation under a wide variety of experimental conditions, with single or dual cycle, natural or forced circulation, and varying degrees of water purity and oxygen content.

After construction of the plant was completed in the summer of 1957, General Electric was issued Power Reactor License No. 1 by the AEC, and after completion of the critical experiments, VBWR was initially operated at power on October 17. On October 19 the reactor developed its maximum licensed output. On October 24 (less than 17 months after ground breaking) the reactor went to full power and over 5000 kw of electricity was being delivered to the PG & E system. Well over one million kilowatt-hours had been produced by January 1958 and no technical difficulties had been experienced.

During the course of power operation, the reactor system was found to be stable against load fluctuations and dynamic effects. The original power limitation of 20,000 kw of heat has been raised 50 per cent to 30,000 kw. The reactor easily generates that output; indeed, fuel temperatures indicate a large heat-transfer margin, and the operators believe that the reactor is capable of producing substantially more than the 30,000 kw presently authorized.

In addition to serving as a pilot plant for the Dresden station, VBWR has the following purposes:

- (a) To serve as a tool for the development, demonstration, and testing of reactor designs, cores, controls, and associated systems and components.
- (b) To provide a training ground in the construction and operation of boiling water reactors and aid in the development of experienced engineers and technicians essential to the construction of nuclear reactors.
- (c) To provide reliable information concerning stability, load response characteristics, safety and fuel economy and to provide an operating prototype which can be used to demonstrate actual performance of boiling reactors.

LOCATION OF VALLECITOS

The Vallecitos Atomic Laboratory site consists of 1600 acres located near Pleasanton, in Alameda County, California. The laboratory derives its name from the land on which it is situated, which was called Vallecitos—Spanish for little valleys—by early settlers. It is east of San Francisco Bay, approximately 33 air miles east-southeast of San Francisco and 19 air miles north of San Jose. The nearest sizable towns are Pleasanton, with a population of 2800, four miles to the north-northwest, and Livermore, with a population of about 10,000, six miles to the northeast. Vallecitos Valley in which the plant is located is a small valley about two miles long and a half-mile wide encircled by barren mountains and rolling hills ranging from 100 to 600 feet above the reactor site. There is very little industry within a 20-mile radius, and very little population. The land immediately adjacent to the site as well as the land to the north and west, is devoted to agriculture.

Water is supplied to the laboratory from the 10½-ft diameter Hetch-Hetchy Aqueduct which provides water to the city of San Francisco. A 14-inch line is installed from this aqueduct to the site, and the pumps provided have a capacity of one million gallons of water per day. The ultimate capacity of the pump house and pipe line is over three million

gallons daily. Studies of the hydrology of the site with respect to paths travelled by materials dispersed near the laboratory indicate that underground water will travel northward toward Livermore Valley three miles away at a speed of 500 to 1000 feet per year. Seismological studies have also been made and though the entire East Bay area is considered as seismologically active, the fault in the vicinity of the laboratory is less dangerous than other faults nearby. The reactor enclosure and the reactor plant have been designed with the possibility of earthquakes in mind and, in addition, a seismograph has been provided to shut down the reactor and close the isolating valve in the main steam line at a point where it passes through the enclosure in the event of an earthquake.

GENERAL FEATURES OF VALLECITOS BOILING WATER REACTOR

The reactor plant includes three structures: the turbine building, the control and the reactor building. The turbine building is a two-story, sheet-metal structure with the turbine-generator located about 13 feet above grade. This building also houses the condensate purification and pumping system, the electrical switchgear, and various turbine auxiliaries. The control building is a two-story steel frame, concrete building containing office space, a small shop, an experimental room, a battery room and a fuel storage vault, as well as the control room. The offices and control room occupy the second floor, which is on the same level as the turbine. An elevated walk connects the turbine area to the control room. Figure 7-1 is a photograph of the laboratory, Fig. 7-2 is a cut-away drawing showing the arrangement of the equipment in the buildings, and Fig. 7-3 is a cut-away view of the reactor.

The reactor is housed in a vapor-tight steel enclosure consisting of a 48-ft diameter cylinder with hemispherical ends, 105 ft in total height. About one-third of this building is below grade. The building interior is divided into an upper and a lower area, separated by a 4-ft thick concrete floor. The reactor is in the lower area, with its top several feet below the upper floor level. It is covered with a removable steel missile and radiation shield 18 inches thick. Also located in the lower area are various reactor auxiliaries.

The reactor is an enriched uranium, boiling light-water reactor. The initial chain reacting core is 3 ft in diameter and 3 ft high, but the core supports can accommodate cores as large as $4\frac{1}{2}$ ft equivalent diameter. A *chimney* section approximately $2\frac{1}{2}$ ft long is incorporated above the active section of each fuel element to promote coolant circulation when operating with natural convection. Fuel elements are 3 in. square and contain either 6 or 9 fully enriched fuel plates. Fuel is fabricated in the

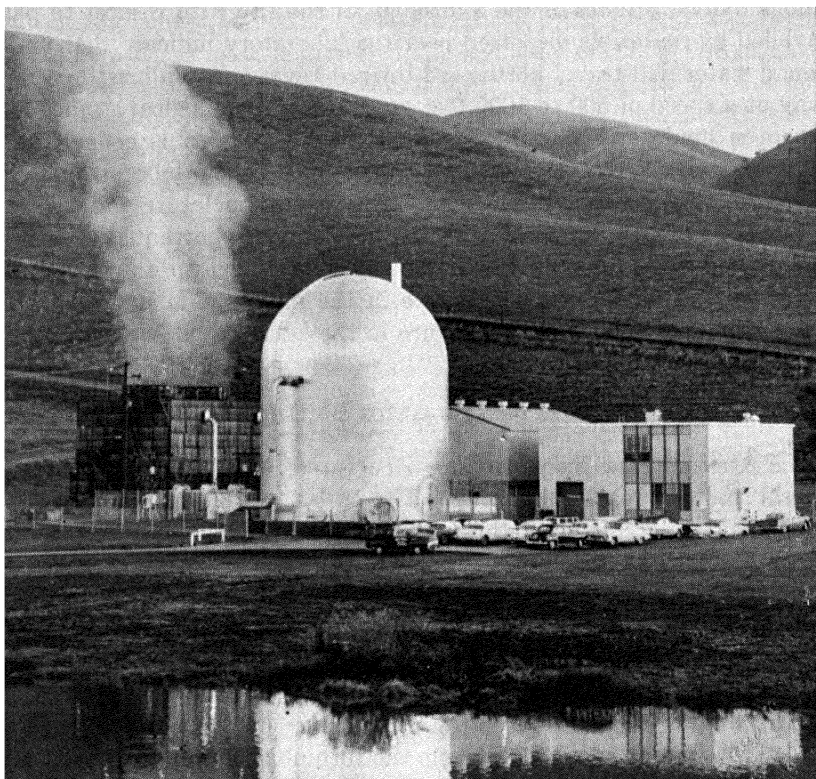


FIG. 7-1. The Vallecitos Atomic Power Laboratory (VBWR) showing, from left to right, the cooling tower, the reactor building, the turbine building, and the control building.

form of plates made from stainless steel, fully enriched UO_2 mixture, encased in stainless steel. These plates are 0.025 in. thick, approximately 3 in. wide, and 3 ft long. Partially enriched uranium oxide, rod-type fuel may be used interchangeably and several of these elements have already been tested in this reactor. The reactor can be operated fully loaded with uranium oxide fuel enriched to 2.3% U^{235} content.

The reactor is contained in a 7-ft diameter stainless steel lined vessel, designed for a working pressure of 1215 psi. This 60-ton vessel, which is 20 ft high, has a gasketed and bolted head, which permits complete access to any portion of the reactor. Four 12-in. ports are also provided to allow refueling without removing the reactor head.

Steam generated in the reactor passes through a moisture baffle into the steam line. Steam generated at 1000 psi in the reactor is reduced to

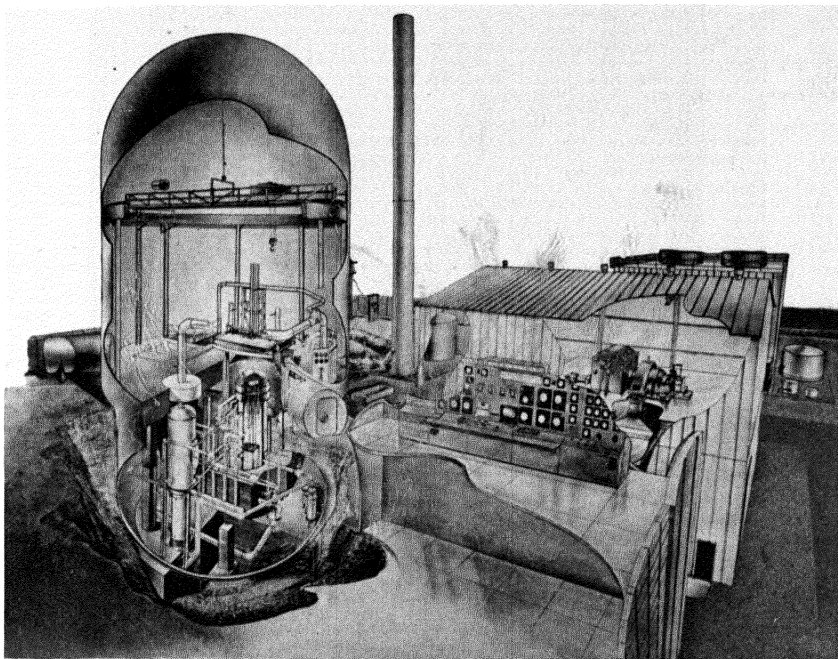


FIG. 7-2. Cutaway drawing of the Vallecitos reactor plant.

450 psi at a reducing valve station, and the reduced pressure steam is used to drive the 5000-kw turbine-generator in the electric generating station.

While the reactor is normally operated on a single cycle with natural circulation, a pump, a heat exchanger, and coolant flow baffles are provided to permit operation on an experimental basis in any of the following cycles:

1. Natural circulation, single cycle.
2. Forced circulation, single cycle.
3. Natural circulation through core, dual cycle.
4. Forced circulation, dual cycle.

A schematic flow diagram (Fig. 7-4) shows how this flexibility of operation is attained.

The reactor is shielded on all sides by thick sections of ordinary concrete. Shielding directly above the reactor is afforded by the water in the vessel, supplemented by an 80-ton movable shield incorporating 18 in. thickness of iron. This upper shield is divided into three sections, and mounted on tracks so that sections may be readily rolled back for refueling. Seven control-rod drives are mounted on the central section of this

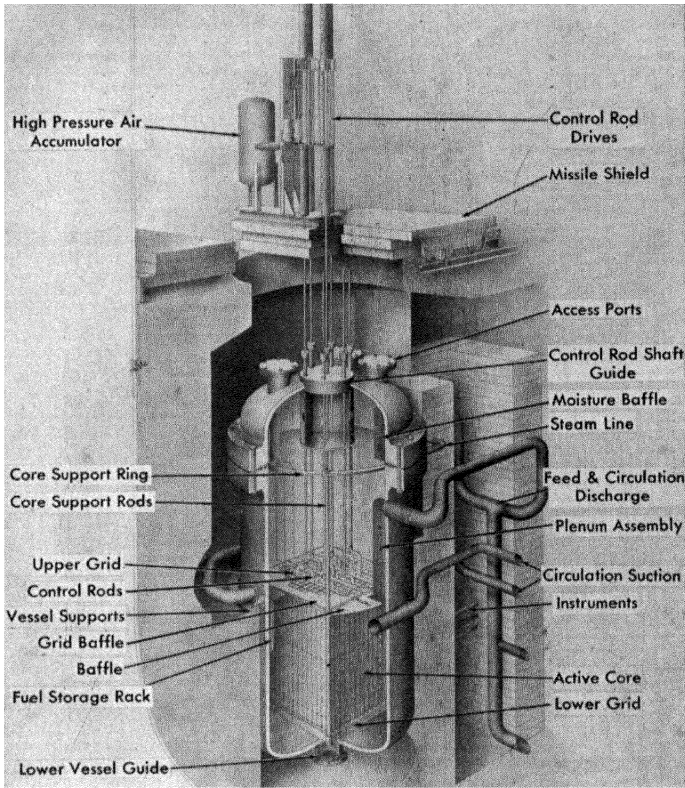


FIG. 7-3. Cutaway view of the reactor.

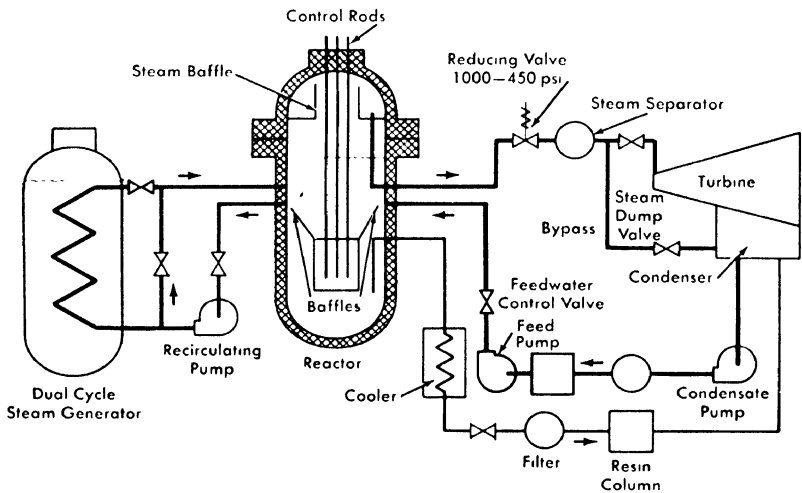


FIG. 7-4. Schematic flow diagram of the VBWR.

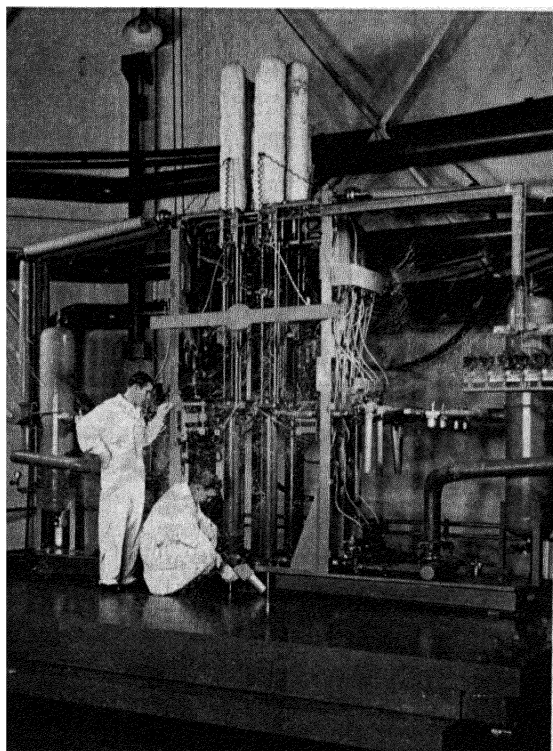


FIG. 7-5. View of upper shield and control-rod drive.

shield, as shown in Fig. 7-5. Control-rod drives enter the reactor vessel through the pressure seals in the top head, and drive blade-type absorbers into the active core. The control-rod drives are normally operated by electricity, provided by storage batteries, but compressed air is used to drive rods into the core in case of emergency. The action of these rods is unusually fast, so that the reactor can be shut down within a fifth of a second after a signal requiring action has been received.

As previously mentioned, the reactor and recirculating loop are contained in a leakproof steel enclosure, 105 ft high, capable of containing the maximum pressure of 45 psi that would be developed if all the coolant in the reactor system should be released. Penetrations through this enclosure are fitted with seals, locks, and valves so that the reactor is automatically isolated under emergency conditions.

The operation of the reactor is controlled from the adjacent building housing the control room. A single control board contains the reactor measuring systems, radiation monitors, steam plant, measuring systems and controls, and generator instruments and controls.

DETAILS OF VBWR COMPONENTS

The reactor. The initial loading of the VBWR consisted of 102 fuel elements, 3.1 in. square containing 22 kg (48.4 lb) of U^{235} . These, as previously indicated, are arranged in a lattice 3 ft in diameter, and the core is also 3 ft high.

Water enters the bottom of the fuel elements; at some point on the way up the element it starts to boil, and the water and steam leave through the chimney which is provided above the element. After leaving the chimney, the water, after having been separated from the steam, flows down around the outside of the core while the steam passes out of the vessel through a pipe at the top. If forced circulation is desired, a baffle can be placed around the top of the core to prevent the downward circulation of the water. Hot water is then removed by pipes above the core, circulated through the pumping loop, and returned to the reactor where it flows through ducts to the region below the baffle. A steam generator may be included in this pumping loop by the manipulation of appropriate valves. In this type of dual-cycle operation, steam is generated by the circulating hot water in the heat exchanger, and this secondary steam enters the turbine at an intermediate stage, at lower pressure than the primary steam, as explained in Chapter 2. In the simulated dual-cycle operation at VBWR the secondary steam will be exhausted to the stack since the present turbine-generator is not equipped with interstage steam supply nozzles.

Reactivity is controlled by 7 control rods made of boron carbide encased in stainless steel cladding. These rods are in 3 rows with one $5\frac{3}{4}$ -in. wide plate and two 12-in. wide plates in the center row, and two 12-in. wide plates in each outside row. The rows are 9.7 in. apart and the plates are $\frac{3}{8}$ in. thick.

The control-rod drives are patterned after those used on the BORAX reactors. The actuating mechanism in this drive is a double-acting air cylinder with the piston attached to the drive shaft. During normal operation, air pressure is supplied to the lower side of the cylinder, driving the shaft up, until a stop on the shaft makes contact with a motor-driven stop. When the motor-driven stop is moved up or down, the stop on the shaft, and the control rod, will follow it. To scram the reactor, air pressure is removed from the lower part of the cylinder and a higher pressure is applied to the upper part. The piston is driven down by the air pressure causing the stop on the shaft to separate from the motor-driven stop. A switch on the stop then opens, causing the motor-driven stop to move down. When it again makes contact with the stop on the shaft, it once more becomes possible to withdraw the rod. A balancing cylinder

at the top of the shaft is provided with reactor pressure so that the cylinder and motor-driven stop do not have to overcome reactor pressure, but only the friction in the mechanism and seals.

The reactor is contained in a 7-ft inside diameter pressure vessel provided with suitable nozzles for the control-rod drive shafts and for the necessary piping and instrumentation. In addition four 12-in. nozzles have been put into the top head to permit refueling without removal of the head. The nominal operating pressure of the vessel is 1000 psia but the design of the safety valve setting is 1300 psia. The normal operating temperature for 1000 psia is 545 F. Normal water level is 12 ft above the bottom of the tank or approximately 7 ft above the active core.

Fuel elements. As already indicated, the fuel is contained in stainless steel clad plates. These plates are 2.950 in. wide and 37 in. long. The active fuel, known as the "meat," has a border, or "picture frame," of stainless steel between $\frac{1}{4}$ and $\frac{3}{8}$ in. wide on the sides and $\frac{1}{4}$ to $\frac{3}{4}$ in. wide on the ends. The meat itself consists of a mixture of 77 per cent Type 304L steel powder and 23 per cent fully enriched uranium dioxide. The picture frame cover plates are also made of Type 304L steel plate.

The fuel elements consist of 6 fuel plates mounted between two side plates, a chimney formed by the side plates and by two stainless-steel plates which are extensions of the outside fuel plates, and suitable top and bottom fittings. Figure 7-6 is a cut-away drawing showing the method of assembly.

The side plates are made of 3 in.-wide strips with cylindrical indentations, are 0.050 in. deep, and have a 5-in.-radius of curvature. Six slots, into which the fuel plates fit, are punched in each indentation. This side plate material is formed and punched and then cut to the proper length. The fuel plates and chimney plates are fitted into the slots and spot-welded in place. The element is held in the lower support grid by a tip formed by narrowing the side plates and putting two cross-pieces between them. The cross-pieces are attached by slipping them into the stamped slots and welding, in the same manner as the fuel plates. The tip is tapered in both directions to simplify insertion into the lower grid. A handle is spot-welded to the top to permit remote handling of the element.

The completed element is 3 in. square and $76\frac{1}{2}$ in. long including the 7-in. long tip. The handle extends another inch above the element. The fuel plates are 37 in. long and the chimney plates $28\frac{1}{4}$ in. The fuel plates start 4 in. above the lower grid.

The dimensions of the tip are held to very close tolerances to insure a good fit into the lower grid. Even more important is the squareness of the shoulder on the side plates. This shoulder rests on the lower grid

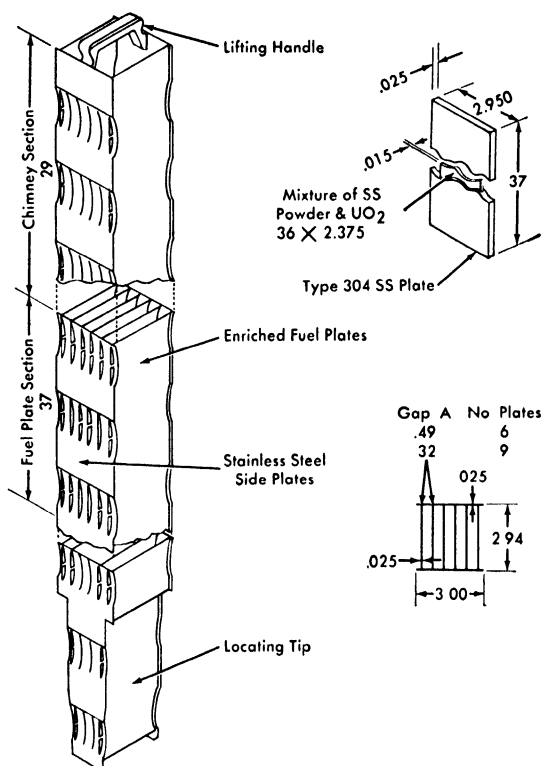


FIG. 7-6. Plate-type fuel element.

and its squareness determines how nearly vertical the element will stand. The tolerance required on this squareness is 0.047 in. in $69\frac{1}{2}$ in.

A sample fuel element was tested in a hydraulic test loop to determine the probable pressure drop under actual operating conditions and the structural stability of the design. A pressure drop of 2 in. of mercury (about 1 psi) was measured at a flow of 185 gpm. It was estimated by the group making the test that this corresponded, so far as pressure drop was concerned, to the rated two-phase flow under actual operating conditions. This value, if in error, is probably too high. The element was tested at flow rates up to 385 gpm, which is at least four times operating pressure drop, and suffered no apparent deformation or damage.

The low-enrichment elements, as mentioned above, are of two types—one containing 9 fuel rods and the other containing 16. The 16-rod elements are intended to simulate the water-to-fuel ratio planned for the Dresden reactor while the nine-rod elements have a more suitable ratio

for this reactor. With this exception, the two types are essentially the same. The fuel rods consist of 0.5 in. diameter by 0.5 in.-long pellets of 2.3 per cent enriched uranium dioxide enclosed in a 0.030-in. thick tube of Zircaloy 2. Zircaloy plugs are heli-arc welded into the ends with space left inside to allow for expansion of the pellets. Each rod is made of two such sections, one about 28 in. long and one about 10 in. long. Long top sections and short top sections alternate in order to stagger the joints. The particular selection of lengths was chosen because the longer segment is the same as that prepared for Dresden.

The rods are enclosed in a $2\frac{15}{16}$ in. square channel made of 0.040 in. Zircaloy. A spacer block is attached to the channel at the top of the fuel rods and the center row of rods (in the nine-rod element) is fastened to this. The outer rows of rods fit into holes in the spacer. A similar spacer at the bottom of the rods is attached to the center row of rods and to the end fitting but not to the channel. The channel thus is free to slide over the end fitting to allow for expansion of the channel. Expansion of the fuel rods is taken up by springs between the rods and the upper spacer. A stainless-steel chimney of the same shape as the channel is fastened to the upper spacer to form an extension of the channel. It is provided with a handle similar to that used on the fully enriched elements. The bottom fitting is a stainless-steel casting machined to fit into the lower grid in the same manner as the plate type elements. Dimensional and squareness tolerance requirements on the two types of elements are the same. The two elements are thus physically interchangeable.

The nine-rod element will contain about 10 kg of UO_2 . An operable loading requires slightly over 100 elements or about 1000 kilograms. An assembly drawing of the nine-rod element appears in Fig. 7-7.

Core support. The principle parts of the stainless-steel core structure are the upper and lower grids. These are suspended on support bolts from a ring which in turn is supported by brackets attached to the pressure vessel just below the flange. The upper grid, in addition to its primary function of holding down and positioning the fuel elements, supports the baffles which control water flow when using forced circulation.

The lower grid consists of an egg-crate type of structure supported in a square frame which is attached to the support bolts. The frame was fabricated by welding together 7-in. by 2-in. by 1-in. angles. Tubing is then welded to the corners and drilled and tapped to receive the support bolts. The egg crate is held in the frame by hold-down bars bolted to two sides of the frame. All welded materials are Type 304 ELC (Extra Low Carbon) steel and all other material is Type 304 steel.

The upper grid assembly consists principally of a frame with six hold-

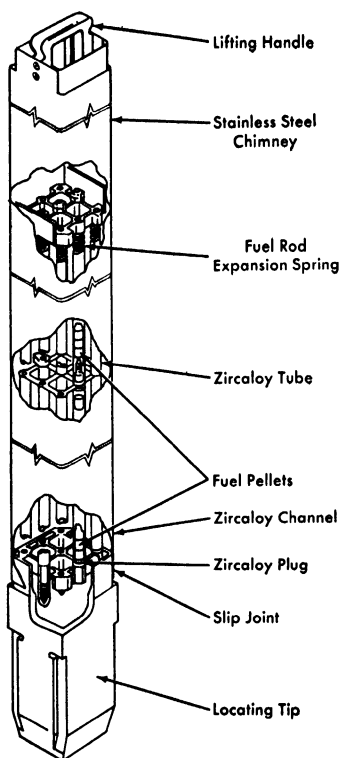


Fig. 7-7. Nine rod low enrichment fuel element.

down grids hinged to it. The frame is a square structure welded from 1-in. by 3-in. Type 304 ELC steel bars placed on edge. Hinge blocks are welded on the top of this and ears are welded on the corners and drilled to accommodate support bolts. Brackets are also welded on the outside of the frame to support various baffle sections.

Proper control of water flow during forced circulation requires that a baffle be placed around the core at some point. Since this baffle must be removed for natural circulation, a location was desired which would make the insertion or removal of the baffle as simple as possible. Hinged baffle sections, which may be swung up to allow circulation, are located outside of the square frame on supports welded to the frame. Inside the frame, the baffles are mounted directly on the hinged grid sections. These baffles reduce the open grid area to 148 elements, which is the equivalent of a $3\frac{1}{2}$ -ft diameter core. The initial operation was with a smaller core of 102 elements. In this case, the remaining space was closed off by filling the core with stainless-steel dummy elements with the water passages stopped.

The support ring, which rests on brackets attached to the vessel wall, is a rolled Type 304 steel ring 2 in. thick and 4 in. wide. It is fastened to the brackets by J-bolts and is fitted with adjusting screws for precision locating. This is necessary in order to align the control rod passages in the core with the control rod nozzles in the upper head of the pressure vessel.

The four support bolts are 1½-in. hardened Armco 17-4PH stainless steel. Monel nuts are used to prevent galling. The bolts are slightly under 14 ft long and are provided with 2¼-in. OD, ¼-in. wall spacers made of Type 304 seamless tubing. These spacers establish the distance between the upper and lower grids and between the support ring and upper grid.

Pressure vessel. The pressure vessel was fabricated in four courses, each 37⅞ in. high, of SA 264 steel plate clad on the inside with stainless steel. The upper head is fastened to the vessel shell with 36 four-inch bolts through flanges that are 17 in. high and more than 9 ft OD. A double gasket (soft iron) arrangement incorporates means of collecting leakage. Figure 7-8 shows details of the pressure vessel.

A 9-in. long piece of 12-in. pipe is welded to the bottom of the vessel and fits into a receptacle in the concrete under the vessel. This positions the vessel but does not support it. The full weight of the vessel is carried by 8 supports welded around the side about 8½ ft from the bottom. These rest on a leveling ring set into the concrete structure surrounding the vessel.

Two 10-in. water nozzles are located on each side of the vessel, 13½ ft above the bottom. Four 10-in. outlet-water nozzles are located 72 deg on either side of the centerline through the inlet nozzles, 9 ft above the bottom. A 6-in. outlet steam nozzle is located 90 deg from the inlet nozzles, 13¾ ft above the bottom.

Seven nozzles for control-rod penetration extend 24-in. above a 7¼-in. thick, 30-in. OD, SA105 (grade II) steel disk in the top of the upper head. Attached to the bottom of the disk is a stainless-steel shell that extends about 30-in. below the head and supports guide bushings for the control-rod shafts.

Four 12-in. nozzles, also located in the head, are access ports for refueling. One of these has a 1½-in. nozzle on the side for water level control during refueling, and two of the others have 1-in. nozzles on the side to supply steam to the balancing cylinders on the control-rod drives. The four access ports are fabricated of carbon steel lined with stainless; all other nozzles are of stainless steel.

Water baffles, extending from each inlet nozzle down to 1 ft below the

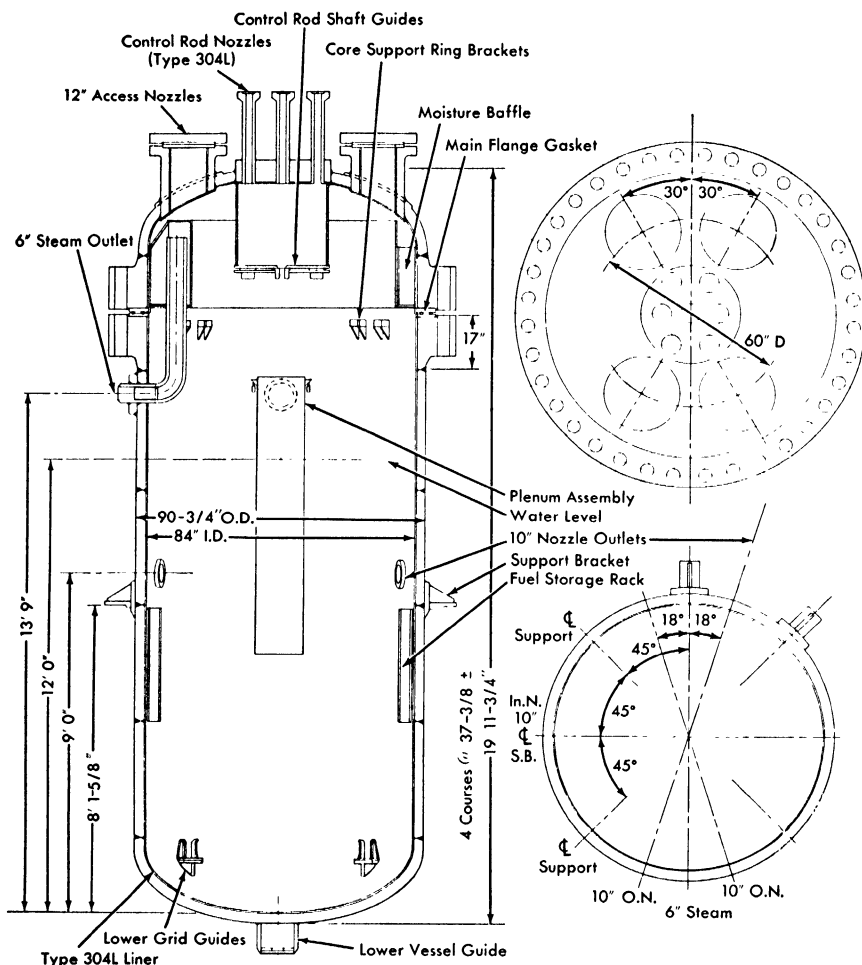


FIG. 7-8. Reactor pressure vessel.

top of the upper grid of the core structure, direct the flow of incoming water to the bottom of the vessel. A steam baffle extends around the inside of the upper head with a pipe connecting it to the outlet-steam nozzle. This baffle serves to remove most of the water entrained in the steam.

A plenum assembly is provided on each side of the vessel to carry water from the inlet nozzles to the area below the baffle. These consist of ducts 10 by 14 in. in cross section. They have openings which fit over the inlet nozzles and vanes for deflecting the water downwards. They extend about 80 in. down from the center of the nozzle to a point approximately one foot below the baffles. These plenums may be easily removed.

Two fuel storage racks are provided, one on the front of the vessel and the other on the back. Each rack will store three irradiated fuel elements for a period of time while the radioactivity is decaying. These racks are attached to brackets welded to the tank near the level of the upper grid. They consist of open top boxes $11\frac{1}{2}$ in. wide, $3\frac{1}{2}$ in. deep, and 36 in. high, made principally of $\frac{1}{8}$ -in. stainless steel. They are located so that the active section of the stored fuel element is below the top grid but above the active section of the core.

Refueling. A 12-in. thick lead coffin is provided for shielding irradiated fuel during removal from the reactor. Fuel may be removed either by disconnecting control rod drives and removing head, or by extracting individual fuel elements through the 12-in. ports in the reactor vessel head. The latter procedure can be accomplished in a relatively short time. For instance, the reactor has been taken off the line, cooled to room temperature, a fuel element removed, and the reactor started up again and placed on the line, all within a 24-hour period.

After removal from the reactor, fuel may be stored in a water-shielded storage pool which is located within the reactor enclosure, next to the reactor shield.

Control rods. As indicated in the general description, the control rods originally used in this reactor consisted of boron carbide plates enclosed in stainless steel. A drawing of a typical control rod is shown in Fig. 7-9. Six 12-in. wide plates and one $5\frac{3}{4}$ -in. wide plate are used, arranged in three rows, with the center row containing two large plates and a small one. The rows on each side contain two large ones. The plates in a row are separated from each other by a space of one-half in.

The absorbing section of the rod is 43 in. long and is filled with boron carbide blocks 2.870 in. square and $\frac{1}{4}$ -in. thick. They are contained between $\frac{1}{16}$ -in. stainless-steel side plates. The side plates are closed on the bottom and sides by stainless steel bars, and stainless steel spacers are provided between each column of boron carbide carbon squares. A piece of $\frac{3}{8}$ -in. thick by 12-in. wide stainless steel forms an extension of the rod through the chimney section of the fuel elements and extends 32 in. above the absorber. It is attached to the side plates and edge bars by welding. Three holes, 8-in. square, are cut in this plate to reduce the weight.

A short piece of stainless steel rod is welded to the top of the control plate to receive the control rod drive shaft extension. This extension is made of stainless steel tubing, $1\frac{1}{2}$ -in. outer diameter and $\frac{1}{8}$ -in. wall thickness, with suitable end fittings welded to it. It extends from the control rod, through the packing gland in the top head, to a coupling

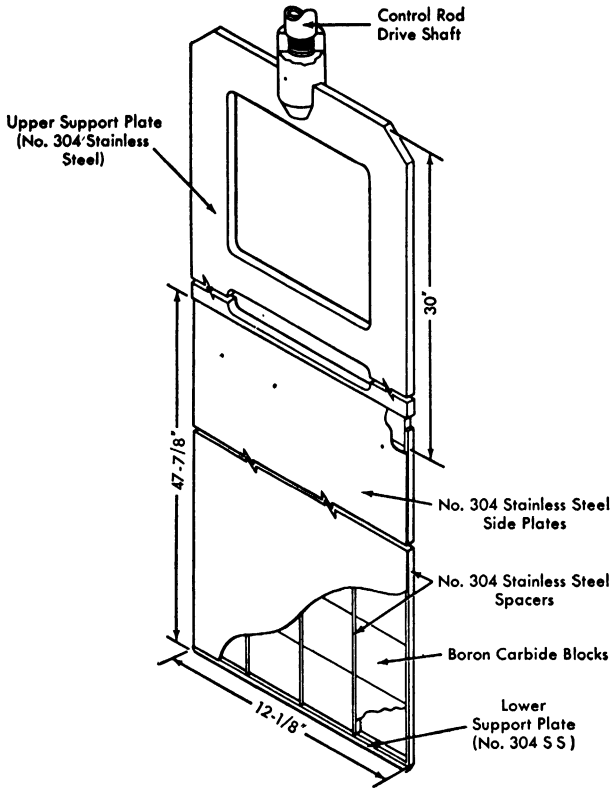


FIG. 7-9. Boron carbide filled control rod.

located above the top head. An 8 micro-inch finish and a maximum diameter variation of 0.0005 in. were specified for these shafts in order that various types of packing gland arrangements may be used without restrictions resulting from roughness or diameter variations.

The boron carbide plates used in the rods are made by hot pressing high purity boron carbide powder in graphite molds. The product has a minimum density of 2.2 grams per cm^3 and contains a minimum of 76.5 per cent boron.

Control-rod drive units. The seven control-rod drive units are mounted on a base plate which is attached to the missile shield (Fig. 7-5). All seven units are identical except the one driving the center rod which, because of space limitations, is arranged slightly differently. The principal difference in this drive is that the motor is inverted and drives the lower end of the lead screw rather than the upper end (Fig. 7-10).

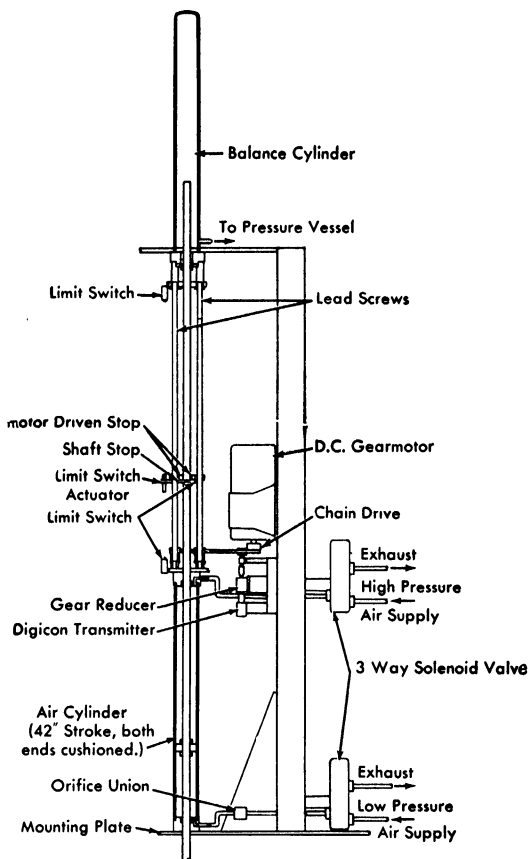


FIG. 7-10. Control-rod drive, schematic.

As mentioned earlier in this chapter, the control-rod drives are patterned after those used in the BORAX reactors, and their operation will not be discussed here. The air cylinder is a double-acting 4-in. bore cylinder with a 42-in. stroke and rated to operate at 200 psi. The heads are modified to have two $\frac{3}{4}$ -in. ports in order to increase the operating speed. The air supply to the lower end of the cylinder is nominally 50 psi. The pressure at scram is around 200 psi. Air cushions are provided during the last $\frac{7}{8}$ -in. length of travel at both ends.

The air cylinders are bolted directly to the base plates of two stands which, in turn, are bolted to the missile shield. One of the stands supports the center drive and three others, and the second stand supports the remaining three units.

REACTOR AUXILIARY SYSTEMS

As considered here, the auxiliary systems of the Vallecitos Boiling Reactor plant are those which are not associated primarily with the main loop to the turbine and back to the reactor. These auxiliary systems include the 188 gpm soft water supply, the emergency heat exchanger system, the reactor water cooler system, the steam generator and recirculating system, and the reactor shutdown pump.

Soft water is supplied by a sodium zeolite water softener designed to provide 188 gpm of water with an average hardness of 2 parts per million or less. From the main supply tank the water passes through the water softener either to a 30,000-gal soft water storage tank or to the emergency heat exchanger. The normal method of operation is to supply water to the steam generator from the storage tank and to supply the emergency heat exchanger directly from the water softener.

The principal purpose of the emergency heat exchanger system is to provide cooling for the reactor core in the event of power failure. It will, however, operate in any case where the reactor system rises above a preset value. A 1200-kw U-tube heat exchanger is used with steam from the reactor passing through the tube side, and cooling water from the soft water supply passing through the shell. The steam is allowed to flow through the exchanger continuously, but cooling water flows only when required by emergency conditions. Flows through both shell and tube sides of the exchanger are by gravity, allowing the system to be independent of electrical power.

The heat exchanger is located outside of the reactor shield, just below the operating floor.

Impurities in the primary water system which leave the reactor vessel with the steam are removed by the condensate demineralizer system, but a large part of the impurities are separated from the steam before it leaves the vessel and remain in the reactor water. In order to continuously remove this contamination, the reactor clean-up system is provided. This consists of two mechanical filters and two mixed resin bed demineralizers arranged so that the water is continuously removed from the lower part of the vessel, passed through the clean-up equipment, and returned to the condenser. Since the resins used in the demineralizers cannot be subjected to a temperature greater than 120 F, a water cooler is provided through which the water passes before entering the filters and demineralizers.

The reactor water cooler is a vertical heat exchanger with the primary water from the reactor passing through the tubes and water from the

cooling tower in the shell. The filters are of the replaceable cartridge type and are designed to remove all particulate matter larger than five microns. The demineralizers are designed to produce water containing not more than 0.004 parts per million of impurities.

Steam generator and recirculating system. The desire to test both single- and dual-cycle operation with either natural or forced circulation was one of the principal reasons for building the Vallecitos Boiling Reactor plant. In order to achieve this aim, a loop consisting of a circulating pump and steam generator is provided. For dual-cycle operation, secondary steam is generated in the steam generator and exhausted to the stack. Operation of this arrangement is not identical with true dual-cycle operation where this steam would be fed to an intermediate stage of the turbine, but is sufficiently similar to provide the necessary experimental data. The simulated dual cycle operation may use either forced circulation by installing the baffle system described earlier in this chapter, or natural circulation without this baffle. To test single-cycle forced circulation operation, the baffle system would be used and the steam generator would be bypassed. For single-cycle natural circulation, neither the baffle system nor the circulating system would be used.

The flow diagram of the loop is shown in Fig. 7-11. Primary water from the reactor enters the loop through four 10-in. nozzles in the pressure vessel. The 10-in. lines from these nozzles feed an 18-in. line which connects to the suction side of the reactor circulating pump. An 18-in. gate valve is included in the suction line. The reactor circulating pump discharge connects to a 14-in. line which leads to a motor operated bypass valve and then back to the reactor through two 10-in. nozzles. A 10-in. line is tapped on to the 14-in. discharge line, upstream of the motorized valve, to connect to the tube side of the steam generator. The outlet from the steam generator passes through another motorized valve and feeds into the line returning to the reactor. By proper adjustment of the two motorized valves, any desired fraction of the circulating water may be made to pass through the tubes of the steam generator. Instrumentation is provided for measuring the flow to the steam generator and through the bypass valve, as well as all important temperatures and pressures. All parts of the circulating system which are in contact with hot water are made of stainless steel. Provisions are made for injecting cool water into the suction nozzles in order to increase the suction head on the pump.

The reactor circulating system suction valve is an 18-in. manual, spur-gear operated split-disk gate valve and is capable of tight shutoff in a 1300 psi circulating system. The reactor circulating pump is a radial-flow, double-suction, diffuser type centrifugal pump, vertically mounted

and driven by a 2-speed induction motor. The pump has a 15-in. diameter, type-347 stainless steel impeller. An oil-lubricated and water-cooled ball bearing is used above the pump, and a carbon throttle bushing and a mechanical seal are provided to minimize leakage along the pump shaft. The driver for the pump is rated at 150 hp at 1200 rpm and 60 hp at 900 rpm.

The 14-in. steam generator bypass line contains a flow nozzle and motor-operated gate valve. The flow nozzle is used to measure the bypass flow which is recorded in the control room and is used to compute power dissipated in the recirculating loop. The valve is a motor-operated, parallel-disk or conduit valve designed for the temperature and pressure conditions which exist in the system. This type of valve combines the low pressure drop characteristics of a gate valve and the throttling characteristics of a plug valve.

The steam generator consists of a vertical U-tube heat exchanger and steam chest contained in a vertical two-diameter cylindrical shell, as indicated in Fig. 7-11. The lower section of the shell is 37-in. in diameter and contains a U-tube heat exchanger. The primary water from the reactor passes through the tubes. The tube bundle is a two-pass type composed of 800 $\frac{1}{2}$ -in. stainless steel U-tubes. The channel is a two-chamber hemispherical shaped 19-in. head which contains the primary water inlet and outlet nozzles.

The lower section of the shell is a 37-in. cylinder 92 $\frac{5}{8}$ in. high, and rolled from forged steel plate. The bottom of this cylinder is fitted with a 300-lb flange which bolts to the tube plate. A 4-in. nozzle is provided in the lower end of the shell and this connects to the suction side of the reactor shutdown pump for blowing down the steam generator. The soft water in the shell side of the steam generator enters the upper section through two 4-in. nozzles and then goes into a circular distributor which is baffled to minimize the possibility of thermal shock. The water flows down the outside of a wrapper tube around the bundle to the bottom of the tank, through openings in the wrapper, and up between the tubes. The water is maintained several inches above the tube bundle by a level control which actuates a diaphragm-operated valve in the soft water inlet line.

The steam-chest section of the generator is a barrel with a 66-in. inner diameter approximately 10 ft high. It contains a 16-in. vapor outlet nozzle which connects through a 16-in. seamless, carbon-steel line to an automatic isolation valve and then continues through the reactor housing and through a 16-in. manually operated gate valve to the stack. Openings are provided in the cylinder for a level control, a level indicator, and an 8-in. line from the main steam relief valve, as well as the 4-in. soft water

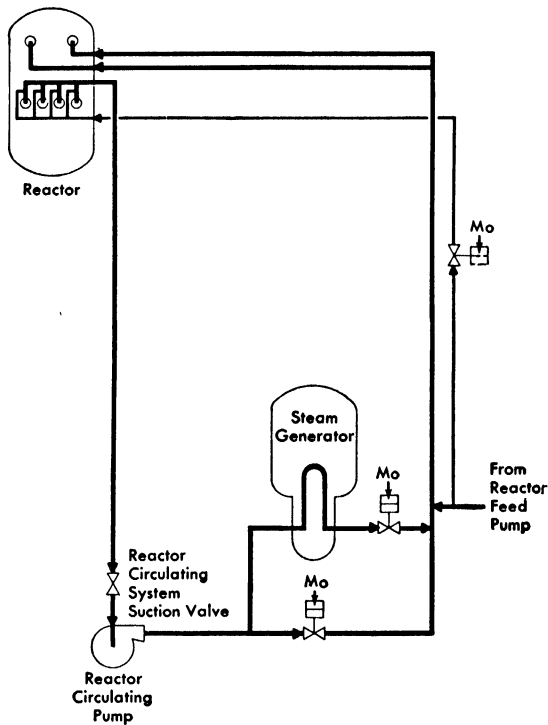


FIG. 7-11. Flow diagram of steam generator and recirculating system.

inlet nozzles mentioned above. A 16-in. manhole is also provided. Design data for the steam generator are listed in Table 7-1.

The 16-in. line to the stack is connected to two 10-in. relief valves relieving into the fuel storage pit. These valves open in the event of high pressure in the steam generator when the isolating valve is closed. The steam generated is then condensed in the fuel storage pit.

The soft water supply to the steam generator is from the soft water storage tank through a 6-in. line. This line is provided with a manually operated gate valve before it enters the enclosure and a check valve on the inside. The water in this line then passes through the diaphragm-operated gate valve which is controlled by the steam generator level controller.

Since the suction head available for the circulating pump without subcooling is barely sufficient, the possibility exists that cavitation can occur if the reactor pressure drops by a small amount. Small pressure variations will normally occur and means must be provided therefore, to prevent cavitation. To accomplish this, a portion of the feedwater

TABLE 7-1
STEAM GENERATOR CHARACTERISTICS

	Shell	Tube
Fluid entering	220,000 lb/hr	3000 gpm
Fluid vaporizing (lb/hr)	220,000	
Temperature in (°F)	60	544
Temperature out (°F)	225	346
Design temperature (°F)	600	600
Operating pressure (psig)	4	985
Design pressure (psig)	300	1300
Test pressure (psig)	450	1950
Pressure drop (psi)		6
Materials	Carbon steel CS-SA-210	Stainless steel Type 304L
Height (ft)	24½	
Steam chest capacity (ft³)	275	
Nozzles	One 16" Two 4"	Two 10"
Heating surface (ft²)	1752	1752

is fed directly to the four suction nozzles in the pressure vessel. The 3-in. line provided to supply this water is equipped with a motor-operated regulating valve which can be adjusted from the control console. This valve will be set to give sufficient sub-cooling to overcome all normal pressure variations.

Reactor shutdown pump. The reactor shutdown pump is provided to serve as a blowdown pump for the shell of the steam generator and also as a method of cooling the reactor core below 212 F. This pump is a horizontal centrifugal pump driven by a 10-hp motor and has a rated capacity of 200 gpm at 225 F with a 100-ft head. The pump casing is designed for a pressure of 300 psi and a temperature of 421 F. The 4-in. inlet to the pump is connected through a gate valve to the shell of the steam generator. The discharge connects through a 3-in. line to an automatic isolation valve and then penetrates the reactor enclosure. The discharge feeds a pair of valves allowing the discharged water to go either to the hold-up sump or to the soft water storage tank. When the pump is used for cooling the reactor water, the latter connection is used. When it is used for blowdown service, the discharge, of course, will go to the sump.

MAIN STEAM AND CONDENSATE SYSTEM

General. The general features of the main steam and condensate are shown in the flow diagram, Fig. 7-4. Steam formed in the reactor core first passes through a baffle which removes most of the entrained moisture. The saturated steam then flows through a 6-in. carbon steel line containing a flow nozzle, an automatic isolation valve, and a manual shut-off valve to the main pressure reducing station. Here the steam is reduced from its original pressure of 1000 psi to a nominal 450 psi for use in the steam turbine. The steam then flows through a centrifugal purifier which is designed to produce steam with an exit quality of 99.5 per cent for delivery to the main turbine throttle valve. The turbine has a nominal rating of 5200 kw.

The main condenser and cooling water system are designed to produce a back pressure of $2\frac{1}{2}$ in. (mercury) at the turbine exhaust. Under certain conditions, however, it may be necessary to also discharge steam into the condenser directly from the reactor. For this reason a bypass line around the turbine is provided as shown in the flow diagram.

Water is removed from the condenser hotwell by two condensate pumps operating in parallel, each capable of pumping 180 gpm at a discharge pressure of 90 psi. This condensate flows through a water purification system prior to entering the reactor feed pump, as already described. The reactor feed pump takes suction at the discharge of the water purification system at a pressure of approximately 10 psi, and develops a discharge pressure of 1060 psi at 280 gpm. This high-pressure feed water flows through a flow nozzle and a control valve, actuated by the reactor level control system, to the reactor. Provision has been made to divert a small portion of the high pressure feed water through two feed water heaters. These are intended as an experiment to determine the erosive effects of extremely wet turbine extraction steam upon feed water heater materials.

Main steam system. Under normal conditions 1000 psia saturated steam will flow through the line between the reactor and the reducing station. The system is designed, however, for a maximum operating pressure of 1300 psi, and this at times will be reached. The remainder of the line carries the saturated steam at a pressure of 475 psi with a moisture content of 1.75 per cent under normal operating conditions. The pressure reducing station can be set for any pressure from zero to 1500 psi.

A steam purifier is installed in the pit in the turbine building to remove excess moisture from the steam. This is done in order to decrease the amount of corrosion and erosion in the turbine and to remove possible

entrained radioactive contamination from the steam. A centrifugal type of purifier is used which is capable of handling 80,000 lb of steam per hour at 475 psi. This purifier is guaranteed to remove 99.5 per cent of the moisture and any entrained particles larger than 10 microns. It is provided with a relief valve set at 485 psig with a capacity of approximately 175,000 lb/hr. The outlet of the relief valve is connected through an 8-in. line to the main condenser.

In order to maintain constant pressure in the reactor, it is necessary to provide means for dumping excess steam. This bypass line has already been referred to and is shown on the flow diagram. Two valves in parallel are provided at the point where the line enters the condenser. One valve is a motor-operated valve controlled manually from the control room; the second is a diaphragm-type regulating valve controlled by a pressure controller on the upstream side of the main reducing station. This valve and control can be used to maintain a constant reactor pressure by bleeding a small amount of steam from the line to the condenser. Since it may be desirable to operate the reactor at steam production rates in excess of the turbine capacity, excess steam may be bypassed through the 4-in. bypass line and the motor-operated valve. The capacity of the system is 65,000 lb/hr of steam. Ten thousand pounds per hour can be dumped through the automatically controlled regulating valve.

Turbine. The turbine-generator consists of a ten-stage turbine directly connected to a 5200-kva, 2200-v, 3-phase, generator operating at 3600 rpm. While the turbine is of standard design, certain changes were made in it to adapt it to the use of saturated steam at the throttle. In normal power plant operation, when superheated steam is used, moisture is largely confined to the final stages of the turbine. In this case, however, with saturated steam at the throttle, moisture forms in all turbine stages and special measures have to be taken to protect the final stages from the unusually high moisture content of the steam in that region. The diaphragms in the second through the sixth stages are protected from water impingement by Monel metal strips. Interstage drain-off orifices through the diaphragms have also been provided to allow water to escape to the four standard extraction points without passing through the turbine buckets. Openings for extraction of steam are provided at four points in the casing, corresponding to pressures of 250, 176, 74, and 17.5 psi. These extractions go directly to the condenser through restriction orifices. Two feed-water heaters are installed as bypass units in the 250 lb and 17.5 lb extraction lines for future experimental purposes.

Main condensate system. The treatment of the condensate is in many respects the same as the treatment in conventional plants, but some special

features have been added because of the necessity of keeping the water of a nuclear system particularly clean. The deaerating hotwell is designed to reduce the oxygen concentration in the condensate to not more than 0.01 cc/liter.

Two condensate pumps are provided, each rated at 180 gpm. Under normal operation one pump is sufficient to handle all of the condensate but during tests, when the reactor is operating at full capacity and steam is being dumped to the condenser through the bypass line, both pumps will be required. The condensate filtering system and the clean-up system have been described and will not be referred to again here.

Oxygen absorbers are provided. These are 2-ft diameter tanks containing six cubic feet of Dualite S-10 resin and are rated at 200 gpm. This resin is an ion exchanger complexed with cupric salt. This cupric salt is reduced to the metallic state during regeneration and absorbs oxygen from the liquid flowing through it during use.

Cooling water for the condenser and other apparatus is supplied from a two-cell cooling tower with two fans driven by 60-hp motors. It is designed for cooling 11,000 gallons of water per minute from 93.8 to 80 F at a wet bulb temperature of 60 F. A vertical circulating pump is used with a design capacity of 12,000 gpm at a 52-ft head.

Waste water system. The waste water system consists of the reactor enclosure and turbine area sumps, the hold-up sump, and the waste water filter and demineralizer. Water from the reactor and turbine sumps flows into the hold-up sump. A sample is taken at this point and analyzed. If treatment is required, it is circulated through the waste demineralizer and filter until the radioactivity is removed, before being discharged, after final analysis, to the pond on the site and then to Vallecitos Creek. If the water in the hold-up sump should become too radioactive for effective decontamination, it can be discharged to a tank truck for disposal.

INSTRUMENTATION AND CONTROL

The instrumentation and control of the Vallecitos plant is separated into two fairly defined areas, one including nuclear and radiation instrumentation and the rod-drive control system, and the other covering the more or less conventional steam and electrical plant instrumentation.

The nuclear instrumentation consists of two startup channels, three linear neutron-level channels, and one period and logarithmic neutron-level channel. Each startup channel uses a boron-lined proportional counter, a preamplifier, an amplifier, and a logarithmic counting-rate meter. One scaler and one recorder, which may be switched to either

startup channel, are provided. Each linear neutron-level channel uses a high temperature, enriched-boron-lined, gamma-compensated ion chamber. One of these chambers feeds a galvanometer, one feeds a safety amplifier, and the third feeds an electronic micro-microammeter; a fourth identical chamber feeds the logarithmic and period amplifiers. The safety amplifier, logarithmic amplifier, and period amplifier drive records, together with the galvanometer circuit, are tied into the scram circuit of the control system.

The principal radiation monitoring instruments are consolidated into a system using a commercial remote-monitoring system. The equipment includes detectors for stack gas activity, reactor building air activity, steam activity, and demineralization activity. Suitable power supplies and measuring equipment are built into the system. All of these circuits, except the demineralizer activity channels, are tied into the scram system.

Also included in the system of nuclear and radiation measuring equipment is a seismographic detector and a high building air pressure detector.

The control-rod drive mechanisms have been described previously. The control system includes the necessary relays, switches and indicators for manipulating the rods, both manually by the operator and automatically in the event of a scram signal. The manual controls include provision for moving any individual rod in or out at the normal speed or at a slower speed, and for simultaneously moving all rods into the reactor at a high speed, as well as for scramming the reactor.

The balance of the instrumentation is referred to as the process instrumentation and includes such items as the reactor level control system, emergency exchanger level control, steam generator control, steam flow measurement system, condensate instrumentation and others of a more or less conventional nature.

Virtually all control is centered on the control console but two auxiliary instrument boards are provided, one in the reactor building and one in the turbine building. These boards contain instruments which the operator does not have to monitor constantly. Industrial television equipment is provided for viewing the board in the reactor building.

Neutron detectors. Two types of neutron detectors are used. At low power levels, enriched boron-lined proportional counters are used, and at high levels, enriched-boron-lined, gamma-compensated ion chambers are used. Two of the former and four of the latter are required. All of these are located in metal sleeves extending through the concrete shielding at the front of the reactor at about the height of the center of the core.

An aluminum cylinder clad in stainless steel is provided inside the

reactor vessel to provide a path for the neutrons from the core to the wall of the vessel. The path is free from variable attenuation which would occur in a water path due to the varying density of the water as a function of temperature. The cylinder is 12 in. in diameter and 18 in. long. It is mounted on the wall of the vessel and extends to a point 6 in. from the outside of the 3-ft core.

For calibration purposes, a neutron flux of approximately 10^{10} neutrons per square centimeter per second at reactor power of 40,000 thermal kilowatts was chosen as the nominal full-power flux for the ion chambers. The chambers may, of course, be located in a lower flux region, if this is necessary because of instrument characteristics.

The proportional counters used are about 15 in. long by $1\frac{1}{2}$ in. in diameter. They are lined with boron-10 and have a sensitivity of 12 counts per second per nv. At the nominal location, a counting rate of one count per second corresponds to about 2×10^{-11} of full power.

The ion chambers are self-compensated enriched-boron coated units, insulated for service at high temperature. The sensitivity to neutrons is 3×10^{-14} amperes per nv and to gamma rays is 10^{-11} amperes per roentgen per hour. The gamma compensation is 98 per cent or better. At the nominal location, full power output is about 300 amperes.

Buildings and shielding. As explained briefly in the general description of VBWR earlier in this chapter, three principal structures are located on the laboratory site. These are the 4000-sq ft control and office building, the 150,000-cu ft reactor containment building, and the 1800-sq ft turbine building. The site may, if necessary, be divided into three zones for contamination control purposes. Zone 1 includes the office space in the control building and all areas outside the control fence. This zone will be contamination-free at all times. Zone 2 covers areas which are normally contamination-free but may become temporarily contaminated by contaminated shoe covers, maintenance, etc. These areas will be cleaned whenever detectable contamination is present. Zone 2 includes the upper room of the reactor enclosure, unshielded portions of the turbine building, non-office areas of the control building and the area enclosed by the control fence. Zone 3 includes the areas from which personnel is normally excluded. Areas in Zone 3 include the lower reactor enclosure space, the area inside the turbine and condenser shielding, and the condensate demineralization area. Admission to Zone 3 with the reactor at full power is limited to that necessary for essential operating and maintenance activities, and restrictions are placed, as required, on the length of time the area may be occupied, the protective equipment necessary, and the type of health physics monitoring required.

The control and office building need little comment since it is a conventional 2-story steel-frame and concrete structure with some special features to protect the occupants in case of an accident. The north end of the building, for example, has one-foot thick concrete walls and an 8-in. thick concrete roof. These features are intended to provide sufficient radiation shielding to protect operating personnel at that end of the building for a period to take whatever alleviating action would be necessary after an accident. This shielding also encourages operators to remain at their position during minor incidents, reducing the possibilities of minor incidents turning into major incidents. It also insures that radiation levels in the control room and offices are maintained at a satisfactorily low level during normal operation.

The turbine building also is a conventional 2-story prefabricated sheet-steel building. The entire building, including the roof, is bolted together to allow easy access for removal of equipment or modification. The only source of radioactivity in the building is from the steam from the reactor. All piping between the two buildings, including the steam line and the radioactive off-gas line to the stack, is buried about six feet underground. In order to isolate all radioactivity within the relatively small condenser-hotwell area, these lines are routed into this area underground. The turbine condenser hotwell area is shielded from the remainder of the building by unmortared solid concrete walls, 2 to 3 ft thick. The area within these walls is accessible during start-up and within minutes after shutdown. The filters, oxygen absorbers, and demineralizers are shielded with $1\frac{1}{2}$ to 2 ft of poured concrete.

All instruments except those required solely for startup are located outside the shielding. These instruments are grouped on panels in three locations: on the turbine platform outside the shielding; above the filter-demineralizer room; and adjacent to the reactor feed pump. Those valves requiring operation during full-power operation are equipped with extension handles through the shielding. These valves include transfer valves for filters, demineralizers, air ejectors, and feed-water heaters as well as steam throttles for air ejectors and gland seal ejectors.

The reactor building, as already indicated, is of special construction, and is designed as a pressure vessel capable of withstanding internal steam pressures of 45 psig at a temperature of 292 F. It is sized to contain all the steam and gas mixture that would be released in the event of a rupture of the pressure vessel. The design also takes into account the earthquake hazard, for which proper allowances have been made.

The hemispherical heads of the enclosure are constructed of $\frac{7}{16}$ -in. plate. The cylindrical section is constructed of $\frac{7}{8}$ -in. plate with the exception of the lower course which is 1-in. plate. The 1-in. course was

installed to provide sufficient strength at the points where temporary supports were attached during construction.

Three tests were conducted to prove the tightness of the enclosure. The first, a soap-bubble test, was made at low pressure. This revealed no leaks. A high-pressure test was then made at 56.25 psig. This is 125 per cent of the rated pressure. No signs of failure were evident and the vessel was formally approved by the State of California inspector.

To ensure safety the design specifications of the shell required the vessel to lose a maximum of 1 per cent of its rated pressure (45 psig) within a 24-hr period through leakage. After the shell had been installed and accepted from the contractor, the reactor operations group was faced with the problem of finding out if the vessel leakage was, in fact, below this limit. The main difficulty was that daily temperature variations cause changes in the absolute pressure large enough to conceal any pressure loss due to leaks. The following account³ describes how this problem was bypassed and how it was proved that the VBWR enclosure leakage was negligible.

Study of temperature records and actual measurements indicated that the ambient temperature at the site normally varies from 15 to 30 F in a period of 24 hours. With the sun striking the steel skin of the vessel, even greater temperature variations could be expected inside the vessel. Clearly such changes in absolute pressure caused by these temperature fluctuations would completely mask a 1 per cent pressure change due to leakage.

The problem was solved by observing the pressure difference between the vessel and a system of airtight drums contained inside the vessel. Since the temperature of the air in the drums is always very nearly the same as that of the air in the container, the pressure difference is not affected by the daily temperature changes and can be used to make an accurate determination of the container leakage rate.

A water manometer was used to monitor the leakage rate. As shown in Fig. 7-12, one leg of the manometer was connected to the pressure vessel and the other was connected to a tank system within the vessel. The manometer provided a very sensitive indication of pressure differential between the two systems. (One pound per square inch corresponded to 27.67 inches differential on the manometer.) At the beginning of the test the two systems were at the same pressure since they were connected by the empty manometer. After water was placed in the manometer, any difference in pressure due to leaks in the reactor containment vessel would be indicated by a differential between the heights of the two water legs of the manometer.

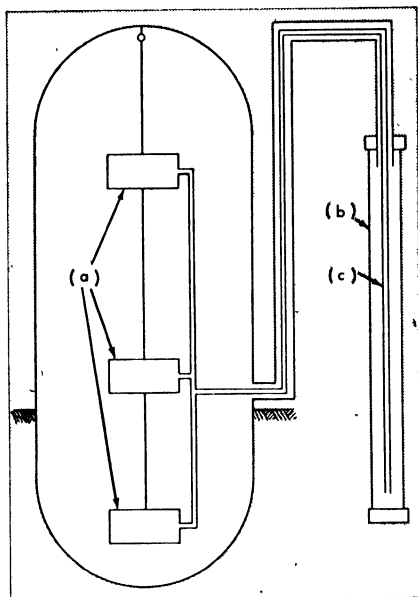


FIG. 7-12. Leak test arrangement for containment shell. Manometer legs (b,c) detect shell leakage by measuring pressure difference between shell and airtight system of barrels (a). Pressure changes due to temperature fluctuations are canceled out since barrels are at temperature of shell.

Test arrangement. Heavy gasoline drums were used for the tank system. Their walls were thick enough not to bulge under the pressure differential yet thin enough to respond rapidly to temperature changes so that the corresponding pressure changes would be quickly equalized on both legs of the manometer. The barrels were suspended by a steel cable in the center of the vessel.

The pressure line from the drums was placed inside the one inch pipe that serves as the pressure line from the reactor container to the manometer location in the VBWR control building. The lines were installed in this manner to minimize the effect of any minute leaks that might be in the tank pressure line.

Results. The weather conditions during the test were quite consistent, and as a result the daily vessel pressure cycles caused by the temperature were repetitive during the 2.3 day period. Pressure differentials caused by the temperature effects on the two legs of the manometer were well compensated for by the three barrel system arrangement within the vessel.

A plot of the manometer readings shows that there was no measurable pressure variations caused by leakage. During the night hours when there was little effect from temperature on the vessel, the readings were about minus $\frac{1}{2}$ inch on each evening for three consecutive evenings. This reading is estimated to be reliable to plus or minus $\frac{1}{8}$ inch. Thus the leakage did not exceed 0.002 per cent per day.

In order to make possible all the qualifying tests for the vessel, it was necessary that construction proceed on temporary supports or legs. After testing, a permanent reinforced-concrete pad was poured under the vessel at an elevation approximately 35 ft below grade level.

OPERATING EXPERIENCE WITH VBWR

During the relatively short time that the Vallecitos plant has been in operation its performance has been very satisfactory. The original operating license limited the output to 20,000 kw of heat. Since its startup, however, there have been a number of amendments with the result that the power limitation has been increased to 30,000 kw of heat, and the reactor performance indicates even larger heat transfer margins. The average output of 50 kw per liter of core, attained when the reactor produced 30,000 thermal kilowatts, is higher than the output per liter required in the Dresden reactor, and it is substantially higher than the output previously obtained from a boiling reactor. Based on these results the designers and operators conclude that still higher steam outputs from boiling reactors are not only feasible but practical.

Nuclear calculations. The two-group diffusion theory was used to calculate the characteristics of VBWR. During the initial loading, these calculations were checked against observed masses, and it was found that the calculations predicted 3 per cent too much reactivity. While this discrepancy is not fully understood, localized flux peaking in excess of that calculated might account for the difference. This explanation is borne out by the fact that the calculated temperature coefficient of reactivity is slightly more negative than the measured value. Calculated void measurements are in qualitative agreement with measured values, and precise agreement with the perturbation theory used in these calculations would not be expected, because of the perturbed flux distribution near the void. Table 7-2 compares measured and calculated values from two of the critical experiments.

Following a number of critical experiments, the 102 element loading shown in Fig. 7-13 was chosen for initial power operation. Some characteristics of this loading are given in Table 7-3.

TABLE 7-2
RESULTS OF TWO CRITICAL EXPERIMENTS

Equivalent diameter of core	2.6 ft	1.6 ft
Number of fuel elements	83	32
Fuel element	6-30	6-40
Critical mass	14.9 kg U ²³⁵	7.4 kg U ²³⁵
Calculated reactivity	1.025	1.028
Measured temperature coefficient	$+6 \times 10^{-5}/^{\circ}\text{F}$	$+3.6 \times 10^{-5}/^{\circ}\text{F}$
Calculated temperature coefficient	$+3.9 \times 10^{-5}/^{\circ}\text{F}$	$+2 \times 10^{-5}/^{\circ}\text{F}$
Calculated central void coefficient	+0.35	-0.12
Measured central void coefficient	+0.46	-0.18

TABLE 7-3
CHARACTERISTICS OF INITIAL LOADING

Maximum authorized reactor output	30,000 thermal kw
Critical mass	22 kg U ²³⁵
Measured void coefficient at 70 F	-0.04
Calculated void coefficient at 545 F	-0.108
Measured void coefficient at 545 F	-0.1
Total available excess reactivity:	
Cold, clean	1.08
Hot, clean	1.07
Hot, clean, boiling	1.06
Hot, poisoned, boiling	1.03

Temperature and void coefficient measurements at low power. Temperature and void coefficients were measured near room temperature, using an external water heater to increase the moderator temperature. Control rods were adjusted as necessary to put the reactor on a slow period, and the temperature coefficient was determined by observing the change in this period as the moderator temperature was varied. In the case of the fully enriched core (Fig. 7-13) which was used initially, the temperature coefficient was slightly positive ($0.6 \times 10^{-5}/^{\circ}\text{F}$) at 70 F, decreasing linearly with temperature being negative at all temperatures over 106 F. The rate at which the temperature coefficient decreased was $0.015 \times 10^{-5}/^{\circ}\text{F}/^{\circ}\text{F}$. Since fission product decay heat normally keeps this reactor at over 100 F, this extremely small positive coefficient at room temperature is not troublesome.

E		b	b	b	b	b	b	b	b		
F		b	b	b	b	b	b	b	b	b	
G	B	B	B	B	B	B	B	B	B	B	
H	B	A	A	A	A	A	A	A	A	B	
J	B	A	A	A	A	A	A	A	A	B	
K	B	B	B	B	B	B	B	B	B	B	
L	B	b	b	b	b	b	b	b	b	B	
M	B	b	b	b	b	b	b	b	b	B	
N		b	b	b	b	b	b	b	b		
4	5	6	7	8	9	10	11	12	13	14	15

FIG. 7-13. Initial power loading of the VBWR.

Fuel element lattice—3.1 in. square.

102 elements—22 kg U^{235} .

Temperature coefficient $+0.6 \times 10^{-5}/^{\circ}\text{F}$ at 70 F.

Coefficient becomes zero at 106 F.

Coefficient becomes $-0.6 \times 10^{-5}/^{\circ}\text{F}$ at 142 F.

Void coefficient at 70 F.

Location	$\Delta k \times \text{water volume}$ void volume
E-8,9 (midplane)	-0.155
Same 9" up	-0.138
F-8,9	-0.085
Same 9" up	-0.157
G-8,9	-0.610
Same 9" up	-0.480
H-8,9	-0.720
Same 9" up	-0.600
J-8,9	-0.450
K-8,9	-0.447

Average integrated void coefficient = -0.04.

Code

First number indicates plates per assembly, second number, grams U^{235} per plate.

6-25 = a

6-35 = c

9-25 = A

9-35 = C

6-30 = b

6-40 = d

9-30 = B

9-40 = D

Heavy rule = rod mostly in; lighter rule = rod mostly out.

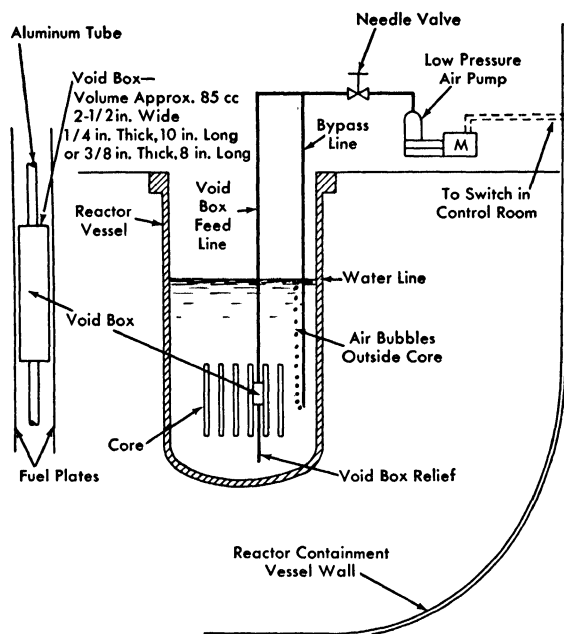


FIG. 7-14. Method of measuring void coefficient.

Void coefficients were also measured by the method diagrammed in Fig. 7-14, using a small air pump to displace water through aluminum tubes connected to aluminum boxes, fitted between the fuel plates in the reactor core. These aluminum boxes were 8 or 10 in. long, $2\frac{1}{2}$ in. wide, and contained about 85 cc. The reactor was placed on a long period, the pump was started, and the water displaced from the void boxes. The new period of the reactor was then observed, and the effect of the void content on reactivity was used to calculate the void coefficient. Void coefficients for the core were found to be slightly negative throughout the entire core.

Startup and operating characteristics. During the startup, the reactor pressure was raised to 1000 psi by heating the water using nuclear power. The turbine throttle was then opened, and the reactor power was raised to maintain steam flow. Control rods were scheduled so that central sheets remained in the core to enhance flux flattening. It was not necessary to bypass steam during load changes, since manual operation of the reactor controls readily maintained stable steam pressure under all transient conditions. Operation of the reactor was extremely stable

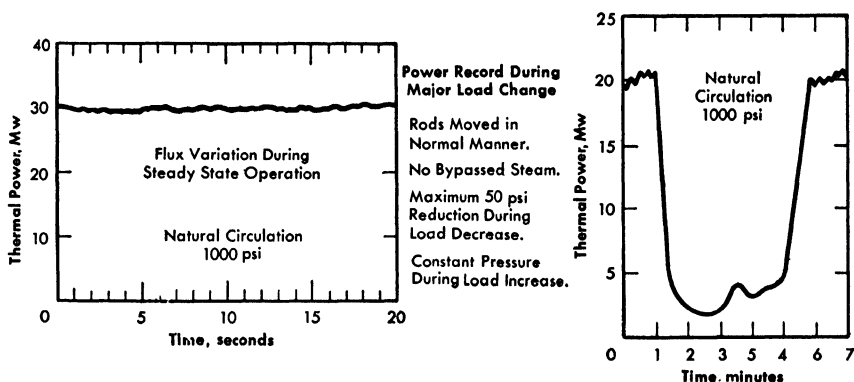


FIG. 7-15. Typical operating records of VBWR.

at full power. The ripple in the neutron flux recorder did not exceed 3 per cent of the stable value when the reactor was operating at 30 thermal kw. Figure 7-15 shows typical operating records.

As expected, it was found that this reactor has a relatively small negative steam void coefficient. The reactivity held in steam voids was measured at various powers, using room temperature control rod calibrations. Results are shown in Fig. 7-16. These data, which are the best presently available, may be subject to uncertainties due to the effects of temperature and steam voids on the control rod strength.

Operation of the reactor following startup was highly satisfactory. By the end of the third month of operation, over 1,600,000 kwh of electricity had been generated by the plant.

Water decomposition. Water decomposition was measured by condensing steam from the main steam line, collecting non-condensable gases over water, and compensating for the amount of dissolved gas in the condensate. Average values for 1000 psi operation are summarized in Table 7-4.

Another feature of interest in the operation of the system is the radioactivity level in various parts of the plant. At 20 Mw, radiation levels in the turbine building are 6 mr (milliroentgen) per hour or less, except in the immediate vicinity of the condenser and the high-pressure end of the turbine. At these points, the levels rise as high as 15 mr per hour. Contact levels as high as 100 mr per hour are measured on the air ejector. Gamma spectrometer analysis indicates that this activity is principally nitrogen-16, which disappears almost immediately after shutdown. Experience indicates turbine personnel receive less than 30 mr per week in

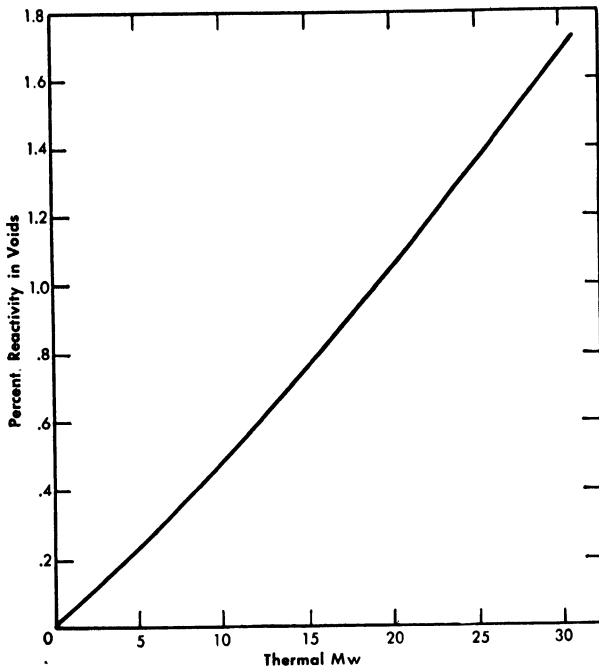


FIG. 7-16. Reactivity held in steam voids.

TABLE 7-4

OXYGEN PRODUCTION AT VARIOUS POWER LEVELS
(1000 psi reactor pressure)

Reactor power thermal megawatts	Liters O ₂ per megawatt hour
4	9
16	11
30	14

the normal performance of their duties. Levels in the upper section of the reactor building, which are normally entered during operation, range up to 6 mr per hour, with levels in the reactor building basement, which is not intended to be accessible during operation, up to a few hundred mr per hour.

Separation factor and residual water activities. After operation, levels outside and inside the turbine casing quickly fall to a fraction of an mr per hour,⁴ so that regular maintenance procedures are entirely practical. Tests were made to determine the separation factor on boiling in the reactor. In these tests, small amounts of NaOH were added to the reactor water and the amount of Na²⁴ activity in the steam and reactor water was measured. In this way a separation factor of 40,000 was found to exist for Na²⁴. Traces of fission products in water, due to surface contamination of fuel, indicated comparable factors for other non-gaseous activities including halogens.

It was found that 1.9 hour fluorine-18 dominated the water activity from a few minutes to a few hours after shutdown. This isotope is produced by the (p,n) reaction on oxygen-18. High energy protons necessary for this reaction are produced by moderation of neutrons in water. This isotope was identified by measurement of the 0.51 positron annihilation radiation. Well over 98 per cent of all reactor water activity after two hours decay is due to F¹⁸.⁷

Since non-gaseous water activities are removed by distillation, noble gases and their daughters predominate in the condenser off-gas system. Noble gas daughters are primarily alkali metals, which are not readily soluble in water, and most of these activities are short-lived, facilitating decontamination and maintenance of equipment after a fuel failure.

High water purity, due to use of deionizers, contributes to the low residual activity levels. Condensate enters the reactor through a large mixed-bed deionizer while the reactor water is purified at a rate sufficient to clean up all the water in the reactor in two hours. The resistivity of the water in the reactor is generally maintained over one megohm, while the water entering the reactor has a resistivity of over five megohms.

Neutron flux distribution. Detailed flux traverses were made throughout the reactor. Besides the conventional uses of flux wires or foils, numerous measurements were made using a miniature gamma ionization chamber ($\frac{1}{4}$ in. in diameter and one inch long). This chamber was inserted in various points throughout the reactor after shutdown. Its reading reflected the fission product activity from adjacent fuel, and hence, power output during high power operation. This method was found to be extremely convenient for determining power distribution during high temperature operation. Comparison with several flux wire measurements showed that the gamma chamber gave a satisfactory indication of local power density.

Flux maps obtained in this way show the pronounced effect of steam bubbles in flattening the radial flux distribution. For example, Fig. 7-17

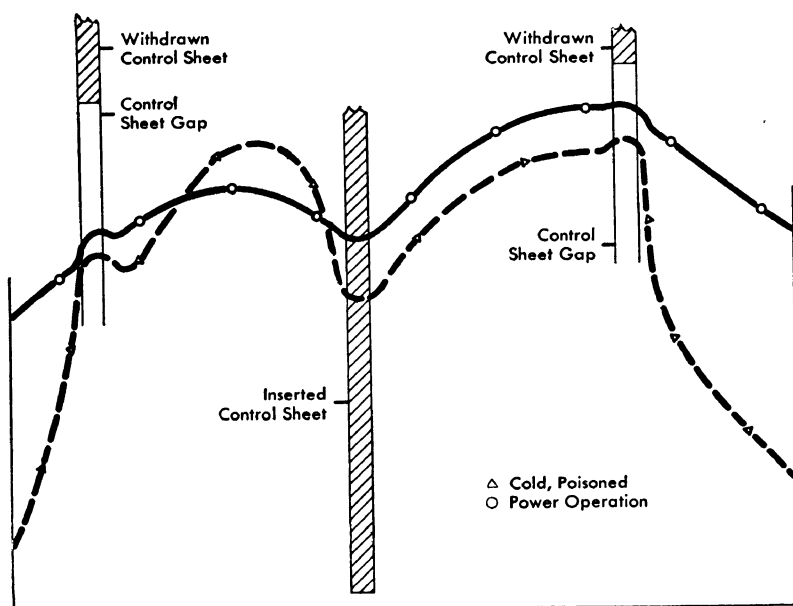


FIG. 7-17. Effect of boiling on radial flux peaking.

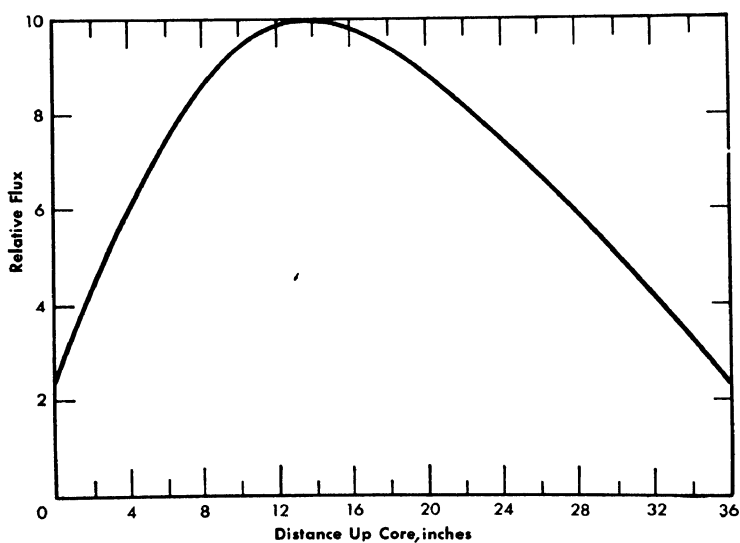


FIG. 7-18. Average axial flux distribution,

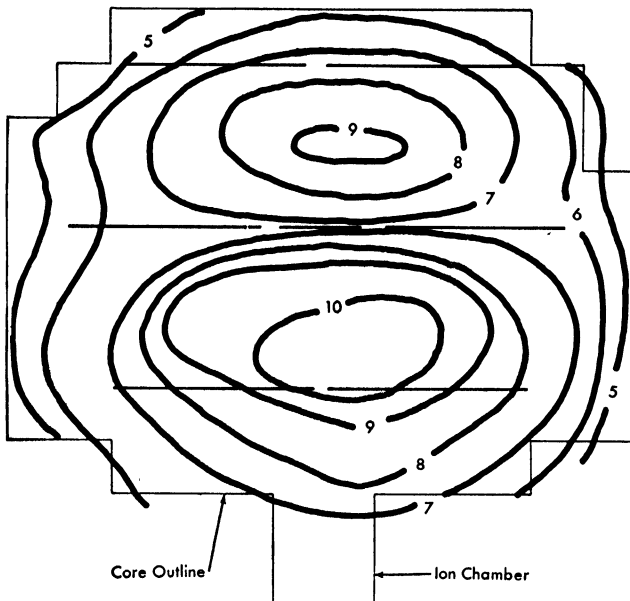


FIG. 7-19. Average radial flux distribution.

shows two sets of flux traverses, made with control rods in the same position. One traverse was made with foils in at low power, using uniform poison to duplicate the operating reactivity. The other traverse was made with a gamma monitor, following full power operation. It is believed that most of the difference in flux peaking between the two traverses can be attributed to "flattening" due to steam voids.

Axial and radial power distributions are shown on Fig. 7-18 and 7-19. Largely owing to the flattened radial power distribution, the over-all ratio of maximum to average power density throughout the reactor is only 2.2. Thus the maximum heat flux at 30 Mw with the present 3-ft diameter enriched core is approximately 200,000 Btu per sq ft per hr. Since burn-out will not occur below 300,000 Btu per sq ft per hr under the conditions in this core, substantially higher power outputs could be obtained from this small reactor core without exceeding the burnout limitations. It is believed that at least 100 kw per liter of core can be obtained without exceeding the heat flux limitations.

Coolant flow and steam voids. Flow limitations through the reactor core must also be considered in establishing maximum reactor output. The average velocity of the water entering the core was measured by observing the time delay before an adjustment in reactor feed affected

reactivity. Feed water enters the top of this reactor, mixing with the water in the downcomer section around the core. The effect of cold feed water is not reflected until the feed addition enters the active section of the reactor core. Experiments were performed in which the rate of feed was suddenly varied and the time delay before reactivity was effected was determined. The average water inlet velocity to the core was calculated using the known volumes for various sections of the flow path within the reactor vessel. A velocity of 2.8 ft per sec was measured with the reactor operating at 20 Mw. Within the precision of this experiment, this value agrees with the calculated velocity of 3.3 ft per sec.

Using measured radial power distribution and coolant velocity, it was calculated that the coolant leaving the hottest element contained 35 per cent by volume of steam at 20 thermal Mw, 46 per cent at 30 Mw, and 72 per cent at 80 Mw. These values are the maximum steam void content leaving any fuel element; the average void content leaving the reactor is much lower. Thus, operation up to 80 Mw is believed feasible with the present core insofar as coolant flow is concerned. A preliminary measurement was performed to verify this estimate of steam void content. Neutron attenuation in room temperature water was compared with the neutron attenuation measured during 20 Mw operation, using a flux wire stretched vertically along the chimney above a central element. This measurement also indicated 35 per cent steam voids above the hottest element at 20 Mw.

Experimental fixtures have also been installed to measure steam voids throughout the core⁵ and the results of this experiment are expected to be available in the near future. A 1/4-in. thimble extends down into the coolant passage within a fuel element, and a flux wire will be irradiated in this tube. Alternate sections of this wire will be sheathed in cadmium, permitting measurement of both thermal and epithermal flux distribution. Since the ratio of fast to thermal neutrons depends upon the amount of moderator present, the ratio:

$$\frac{(\phi_f/\phi_{th}) \text{ non-boiling}}{(\phi_f/\phi_{th}) \text{ boiling}}$$

is related to void content. (In this ratio ϕ_f represents fast neutrons and ϕ_{th} are the thermal neutrons.) This ratio is one in non-boiling regions, falling to zero in places where all the moderator has been expelled by steam. This method correctly indicates conditions in regions large compared to the slowing-down distance of 10 cm in the core.

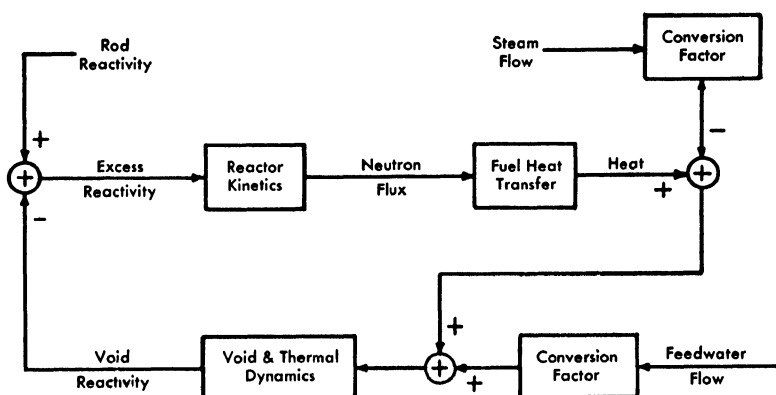


FIG. 7-20. Block diagram for stability analysis.

STABILITY

Like the EBWR and the BORAX reactors, the VBWR is a self-regulating reactor in which the formation of steam reduces the activity. Thus the reactor will *fail-safe* in the event of a rapid reactivity increase.

As already explained in previous chapters, power oscillations may occur in low pressure boiling reactors.⁶ As in the case of EBWR, so in VBWR feed-back theory has been used to gain a better understanding of the intercoupling of nuclear, thermal, and hydraulic effects which may lead to instability. The block diagram shown in Fig. 7-20 illustrates the model used in the VBWR analysis. The detailed analysis will not be given here since it is similar to those made on EBWR but the following parameters were found to have important effects on the stability of the system.

1. *Rate of change in steam voids with power.* This ratio defines the sensitivity of the reactor to disturbances which alter the void content. This ratio depends upon core configuration. A large ratio improves load response, but puts the reactor closer to the threshold of instability. Compared to other operating boiling reactors, VBWR has a small ratio of void change to power.

2. *Hydraulic flow stability.* With natural circulation, steam voids play an important role in hydraulic flow, determining the driving force and also affecting flow restrictions through the core. However, the present VBWR core seems capable of substantially greater output with natural circulation before steam formation blocks the flow path, leading to instability.

3. *Operating pressure.* Pressure affects the transient changes in the void content resulting from disturbances in the system. The higher operating pressure of VBWR contributes to the stable characteristics.

4. *Thermal time response of the fuel.* This factor has a strong bearing on stability. Thermal delay is closely related to fuel dimensions and the thermal diffusivity of the fuel material. Fuels which have a fast time response are potentially less stable than slower responding ceramic fuels used in large power stations.

5. *Coolant storage volume.* The rapidity of pressure transients is affected by the amount of heat stored in the water within the reactor vessel. Inertia of the water may also affect reactor flow characteristics.

APPLICATION OF STABILITY FACTORS TO VBWR

Using appropriate parameters for the initial power loading, it has been calculated that neither hydraulic nor nuclear instabilities will occur below 80 Mw thermal output in the case of VBWR. This is an average of 130 kw per liter of core. However, since this output corresponds to 530,000 Btu per sq ft per hr, maximum heat flux, it would be undesirable to operate this core at so high a power. Thus, the output of VBWR appears to be fixed by heat transfer rather than stability, and power density of at least 100 kw per liter seems feasible. With a larger core, which can be installed interchangeably in the present reactor vessel, much higher reactor output seems attainable.

ROD DROP EXPERIMENT

A series of *rod drop* transients were measured on VBWR and were analyzed by means of an analog computer, to verify analytical procedure. In these experiments the reactor was made critical, hot at zero power, with the rod to be calibrated fully inserted. The rod is then withdrawn and the reactor operated at power. The rod is then rapidly inserted, and a fast response recording of flux is obtained as the reactor returns to its initial condition of lower power criticality. If the time for rod insertion was negligible, the initial drop in neutron flux would measure the amount of reactivity held in steam voids, mathematically:

$$\frac{n}{n_0} = \frac{\beta}{\beta - \rho},$$

where n_0 = initial neutron flux, n = flux after rod is inserted, β = fraction of delayed neutrons, and ρ = reactivity in voids.

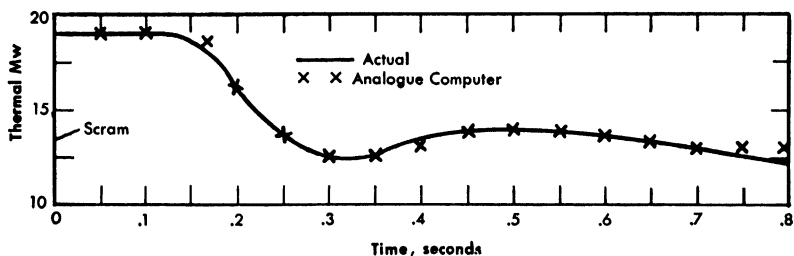


FIG. 7-21. Power transient following reduction in reactivity.

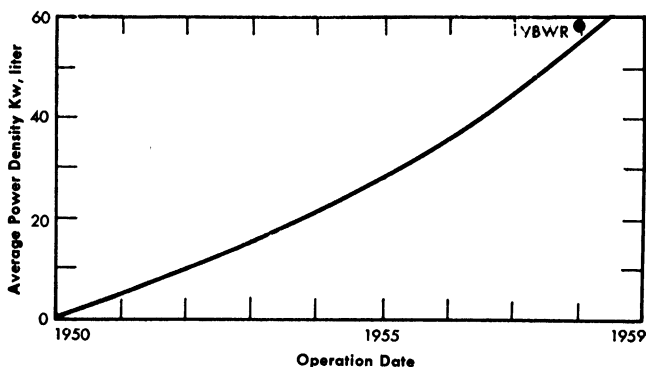


FIG. 7-22. Performance of boiling reactors from 1950 to present.

In the VBWR the rod requires 0.2 seconds for the initial 65 per cent travel, so that the rod insertion time cannot be neglected. The plate-type fuel has a fast time response (about 0.06 seconds) so when the power falls, steam bubbles are swept out of the core, causing a secondary reactivity and power increase. An analog model was applied to this experiment. In this model the unknowns were void reactivity, change in void reactivity with power, and void time response. In Fig. 7-21 an analog representation is compared to a typical rod drop record. A void time constant of 0.1 seconds was found to give a good representation of the experimental rod drop record.

The successful operation of the Vallecitos plant at 20 and 30 Mw with power outputs of 50 kw per liter has been extremely encouraging and the operators of this plant are certain that very much higher outputs can be obtained. The system has shown remarkable stability and excellent response to load changes. Tripping the turbine at full load, bringing the turbine from 10 per cent load to full load in a few seconds, large changes in feed-water rate, opening of bypass valves, and other steam plant maneuvers have produced no reactor transients. Steam generation

can be adequately controlled in a leisurely manner by hand adjustment of the control rod. The pressure of 1000 psi, the highest yet tried in a boiling reactor, has been shown to yield many advantages, such as increased stability, ease of control, reduced rate of water decomposition in the reactor, higher specific power and the possibility of higher temperature and efficiency than heretofore realized. Figure 7-22 shows the gradual increase in the average power density in boiling water reactors since their beginning in 1950. The operators of Vallecitos feel certain that this curve can go considerably higher.

REFERENCES

1. V. D. NIXON, Construction Progress at the Dresden Nuclear Power Station. Presented at Atomic Industrial Forum, New York, October, 1957.
2. L. KORNBLITH, JR., et al., Vallecitos Boiling Water Reactor. Presented at Nuclear Congress of Engineer's Joint Council, Chicago, Illinois, March, 1958.
3. D. L. LAYTON and S. UNTERMYER, Testing Reactor Enclosure Tightness. Paper presented at American Nuclear Society Meeting, Pittsburgh, June, 1957, reported in *Nucleonics*, **15**(8), 66 (August, 1957).
4. SAMUEL UNTERMYER, With Boiling-Water Reactors Turbine Contamination Causes Only Small Problem, *Nucleonics*, **13**(8), 52 (August, 1955).
5. SAMUEL UNTERMYER, Use of Boiling Water as a Reactor Coolant. Presented at Nuclear Science and Engineering Conference (Paper 57-NESC-80), Philadelphia, March, 1957.
6. J. A. THIE, Boiling Water Instability. *Nucleonics*, **16**(3), 102-111 (March, 1958).
7. J. L. RUSSELL, JR., and D. F. RIDER. Fluorine-18 Production in Water Moderated Reactors. Presented at American Nuclear Society Annual Meeting, June, 1958.
8. E. BECKJORD, D. FISCHER, M. A. HEAD, L. K. HOLLAND, L. KORNBLITH, JR., SAMUEL UNTERMYER, and L. WELSH, Operation of a High Performance Light Water Boiling Reactor. Paper OIC-1467 (P/1923), Second International Conference on Peaceful Uses of Atomic Energy, Geneva, 1958.

CHAPTER 8

ALPR—THE ARGONNE LOW POWER REACTOR

The demand for electric power in remotely located, inaccessible areas devoid of natural fuels, offers an opportunity for small nuclear power plant applications. In far northern regions, where heating as well as electric power is in demand most of the year, the fuel supply problem may be critical. The cost of fuel in some isolated areas is inevitably high.

For several years Argonne National Laboratory has made studies leading toward the design of such small nuclear plants, and a boiling water plant of this kind is now under construction at the National Reactor Testing Station in Idaho. This is the ALPR—The Argonne Low Power Reactor—a project which was undertaken at the request of the Army Reactors Branch of the Atomic Energy Commission. This is an enriched uranium-fueled, direct cycle, natural circulation boiling water reactor, cooled and moderated with light water. It is a prototype of *packaged* power plants designed to produce electric power (300 kw) and space heating (400 kw) for remote locations consistent with:

1. Simplicity and reliability of operation and maintenance with minimum supervision.
2. Minimum site construction.
3. Maximum use of standard components.
4. Limited water supply.
5. Utilization of local gravel for biological shielding.
6. Transportability by airlift.
7. Nominal three-year fuel operating lifetime per core loading.

Thus, the ALPR is designed to produce heat as well as electric power and involves an attempt to design a reactor with a core life of three years.

The operating and control experience gained from the BORAX experiments has been incorporated into the development and design of ALPR, together with further control development to provide for automatic adjustment of the reactor power generation rate to the electrical and heating demand.

The principal plant circuits demonstrate unique features in design. The support of the reactor vessel also serves as a substantial part of the thermal shield and as a means of heat removal by a system of built-in water-cooled coils. The cleanup circuit components which become quite radioactive in operation are built into the gravel biological shield. The design requirement of limited availability of water, in turn, limited the available methods

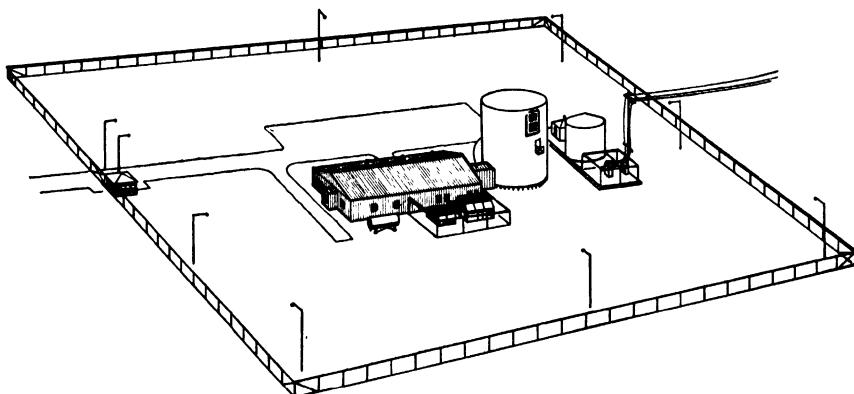


FIG. 8-1. ALPR site.

TABLE 8-1
GENERAL DESIGN DATA

Reactor heat output	3000 kw
Turbogenerator rating	300 kw
Frequency	$60 \pm \frac{1}{2}$ cps
Voltage	$120/208 \pm 5\%$
Phase (4-wire)	3
Power factor	0.8
Design domestic heat	400 kw
Steam production	9020 lb/hr
Operating pressure	300 psig
Operating temperature	420 F
Feedwater temperature (hotwell)	134 F
Standby equipment capacity (diesel electric)	60 kw
Ambient site temperature	-60 to +60 F
Wind velocity (maximum)	125 mph
Building height (maximum above ground)	50 ft
Building site (simulated)	Permafrost
Site materials for construction	Local gravel

of heat removal in the power circuit. Air heat transfer was selected and a system of air-cooled coils was developed to condense the exhaust steam.

The complete facility, shown in Fig. 8-1, occupies an area approximately 350 ft on a side. The permanent design data are given in Table 8-1. The plant is to be housed in a three-story steel tank. The lower level of this tank will contain the reactor and its shielding material, which

consists of gravel of local origin. The second level, the operating floor, will hold the turbine-generator and associated equipment. The third level will house the condenser and special equipment necessary to recirculate and maintain the proper temperature of the cooling air in view of the severe climatic conditions. The building is supported in a manner simulating wood pile construction which may be used in case of future installations in tundra regions.

Referring to Fig. 8-1, the cylindrical steel power plant building adjoins an auxiliary building which provides office space for personnel training, and space for standby equipment. The reactor control and power process instrumentation are located in the end of this building near the power plant. The pump house for the domestic water well located on the site, ground level water storage tank, and sub-station also are shown.

POWER PLANT BUILDING

The power plant building containing the reactor, the turbine-generator and auxiliary equipment is shown in Fig. 8-2. It is a circular, continuous welded steel tank, 38 ft, 7 in. in diameter and 48 ft high. The tank material

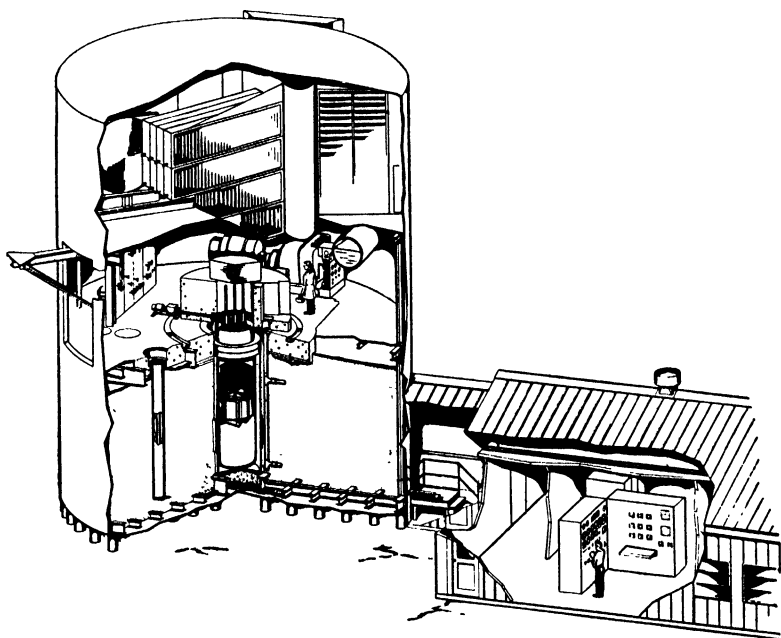


FIG. 8-2. Reactor building, ALPR.

for the prototype is ASTM Spec. A-7 steel, with 30,000 psi yield strength and is suitable for service at temperatures as low as -40°F . It is supported on concrete piers and concrete slab. The structural steel inside the tank is independent of the tank wall and supports two floors, an operating floor at elevation 19 ft 4 in., and a fan floor at elevation 32 ft.

The operating floor contains the top shielding of the reactor, the turbine-generator unit, reactor water storage tank, motor control panel, feedwater pumps, hotwell, boron injection system, bridge crane, purification system, and auxiliary apparatus. The fan floor contains the main steam condensing system consisting of the main condenser, fan, air mixing chamber, building ventilation equipment, and ductwork.

The building is filled with gravel for biological shielding to a point two feet below the operating floor. In addition to the centrally located reactor, the gravel shield contains three fuel storage wells, a reactor water retention tank, purification vault, two storage pumps, reactor instrument wells, ducts, and piping. The exposed gravel area is covered with an

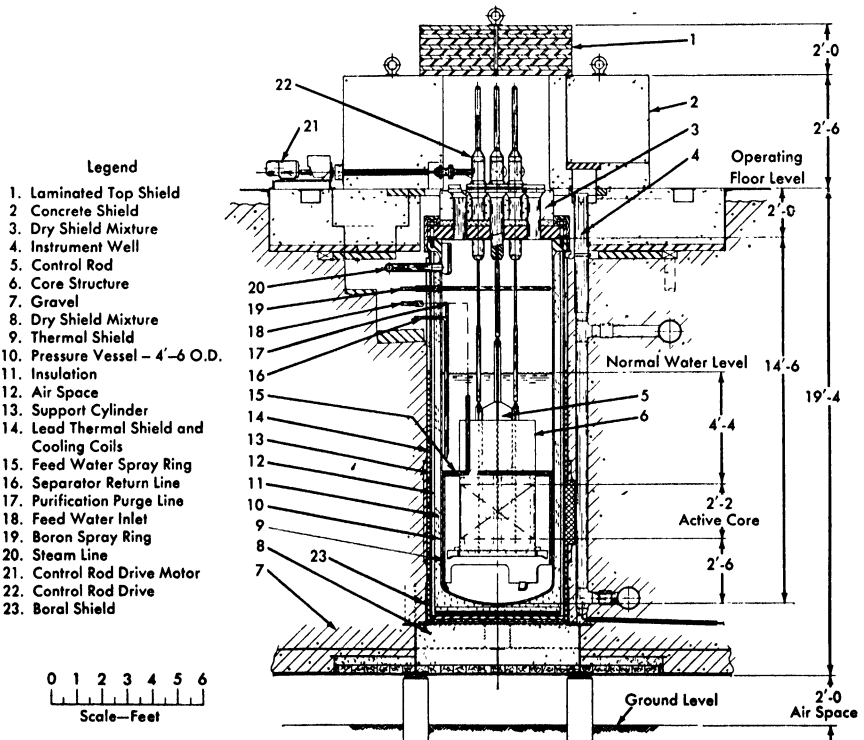


FIG. 8-3. Reactor installation, vertical section.

impervious membrane to contain radioactive gas or dust. The openings in the building are limited to one exterior closed stairway leading to the control room, an emergency exit, and a cargo door.

Construction of the ALPR is well under way; in fact it is entering its final stages. The pressure vessel, steam-turbine generating unit, and other major components have been installed in the reactor building. Critical experiments, startup, and initial operation of the power plant are scheduled in the third quarter of 1958.

THE ALPR REACTOR

Pressure vessel. As shown in Fig. 8-2, the reactor is located on the lower floor in the center of the power plant building. The reactor pressure vessel is fabricated of Type SA-212, Grade B Firebox steel, internally clad with Type 304 steel. A vertical section of the reactor is shown in Fig. 8-3. The vessel is designed for 400 psig at 450 F for water and steam and 500 F metal temperature. The stainless steel clad vessel cover is equipped with nine flanged nozzles to accommodate control rods and drives, two openings for liquid level control, and one opening for source handling. Fuel loading and unloading is accomplished through the control rod nozzle openings. Pressure vessel design data are listed in Table 8-2.

NUCLEAR FUEL AND CORE STRUCTURE

The nuclear and mechanical design of the fuel assemblies (Fig. 8-4) for a three-year operating lifetime resulted in a core structure (Figs. 8-5 and 8-6) consisting of 16 boxes containing a maximum loading of 59 assemblies. The side of the boxes serve as shrouds to define the control rod channels. Each fuel assembly contains 9 flanged fuel plates assembled to side plates by spot welding. Correct spacing of the plates is obtained by the flanged edges. One full length aluminum-nickel strip containing boron-10 is spot welded to a side plate of each fuel assembly. The assemblies are oriented so that these strips face the water channels separating adjacent fuel assemblies. Each fuel plate contains a center portion (meat) of aluminum-nickel-uranium alloy clad with aluminum-nickel.

The aluminum-nickel alloy used in the fuel assemblies and core structure was developed by Argonne National Laboratory. This new material, which evidences excellent corrosion resistant properties, can be readily and inexpensively fabricated and reprocessed and therefore represents an important advancement in the design and operation of this facility.

Table 8-3 lists fuel design data.

TABLE 8-2
PRESSURE VESSEL DESIGN DATA

Diameter, OD	4.5 ft
Wall thickness	
Base material	$\frac{3}{4}$ in.
Stainless steel clad	$\frac{3}{16}$ in.
Height (less head)	14 $\frac{1}{2}$ ft
Design pressure	400 psig
Thermal shield	$\frac{3}{4}$ in. stainless steel
Pressure relief size and settings	
First stage (to condenser)	2 in.—350 psi
Second stage (to atmosphere)	2 in.—385 psi
Level of water above core	4 ft 3 in.
Total weight (empty)	26,000 lb
Average volume of steam dome	80 ft ³
Total weight of contained water	6000 lb
Maximum thermal stresses due to gamma heat in vessel	750 psi
Maximum thermal stresses due to gamma heat in cover	600 psi
Design stress	14,000 psi
Type of closure	Double gasketing with leakoff
Design life	20 yr
Water cleanup circulation rate	3 gpm

CONTROL RODS AND DRIVE MECHANISM

Five cross control rods, Fig. 8-7, will be used in the initial core configuration with additional spaces provided for four T-shaped rods for larger loadings. The active portion of each rod is cadmium sheet confined between aluminum-nickel sheets by edge welding. The cadmium is perforated at intervals, through which the aluminum-nickel sheeting is pressed and spot welded. The length of cadmium provides overlap beyond the active length of the fuel. The center rod has a solid aluminum-nickel plate follower section which extends beyond the bottom of the fuel elements when the rod is fully inserted in the core. The follower section remains in the control rod channel when the cadmium section is withdrawn from the active zone and serves to reduce thermal neutron flux peaking.

TABLE 8-3
FUEL DESIGN DATA

<i>Core</i>	
Horizontal cross section	
Reference loading (40 assemblies)	28 × 28 in.
Maximum loading (59 assemblies)	33 × 33 in.
Fuel loading	
40 Assemblies, U ²³⁵	14.0 kg
59 Assemblies, U ²³⁵	20.65 kg
Metal-to-water ratio	0.5
<i>Fuel Assembly</i>	
Over-all dimensions	3 $\frac{7}{8}$ × 3 $\frac{7}{8}$ × 34 $\frac{1}{2}$ in.
Wt U ²³⁵	350 gm
Number plates	9
Wt B ¹⁰ in boron strip per assembly	0.4 gm
W/o B ¹⁰ in boron strips	0.45
<i>Fuel Plate</i>	
Over-all dimensions	0.120 × 3 $\frac{25}{32}$ × 27 $\frac{13}{16}$ in.
Cladding	0.035 in.
Meat dimensions	0.05 × 3 $\frac{1}{2}$ × 25 $\frac{1}{16}$ in.
Meat volume	74.0 cm
Wt. aluminum in meat	200 gm
Wt U ²³⁵	38.9 gm
Wt U ²³⁸	3.8 gm
W/o uranium in meat	17.6
A/o uranium in meat	2.2

Five cross control rods are worth 14 per cent k in the cold fresh reference core (40 fuel assemblies), and in the fully loaded core (59 fuel assemblies) all nine rods are worth approximately 20 per cent k .

The control drive mechanism is the mechanical rack and pinion type. The pinion drive shaft operates through a seal of the pressure breakdown, labyrinth type. A small amount of water is bled continuously to the seal element to reduce wear and corrosion and to prevent accumulation of decomposition gases produced in the reactor. The external drive is positively engaged with the pinion shaft by a magnetic clutch. Failure of current or a scram signal will release the clutches, and the rods will fall into the core by gravity. Further, the mechanism is designed so that the downward motion will drive the rod down by positive action through an unidirectional clutch. As a backup to power failure, control rod current is supplied by an emergency power system.

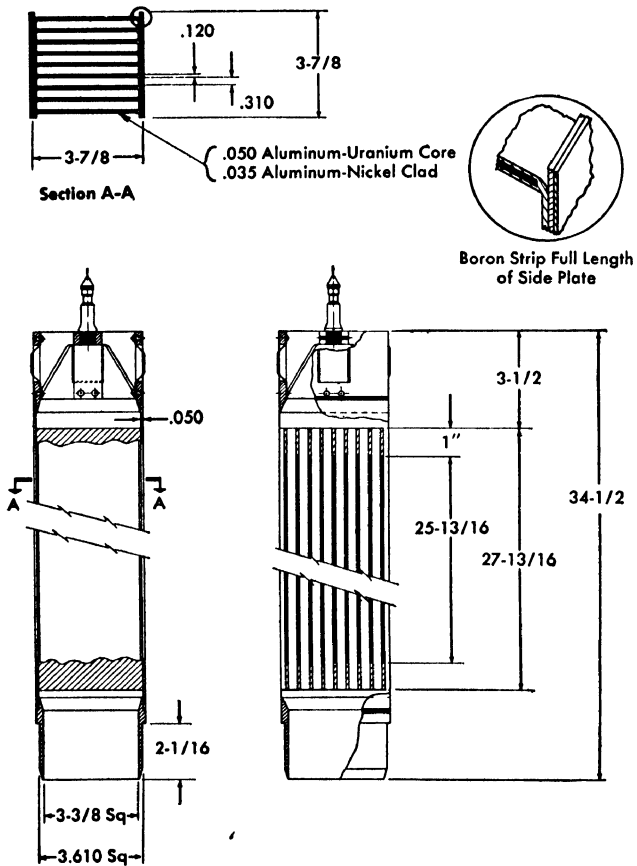


FIG. 8-4. ALPR fuel assembly.

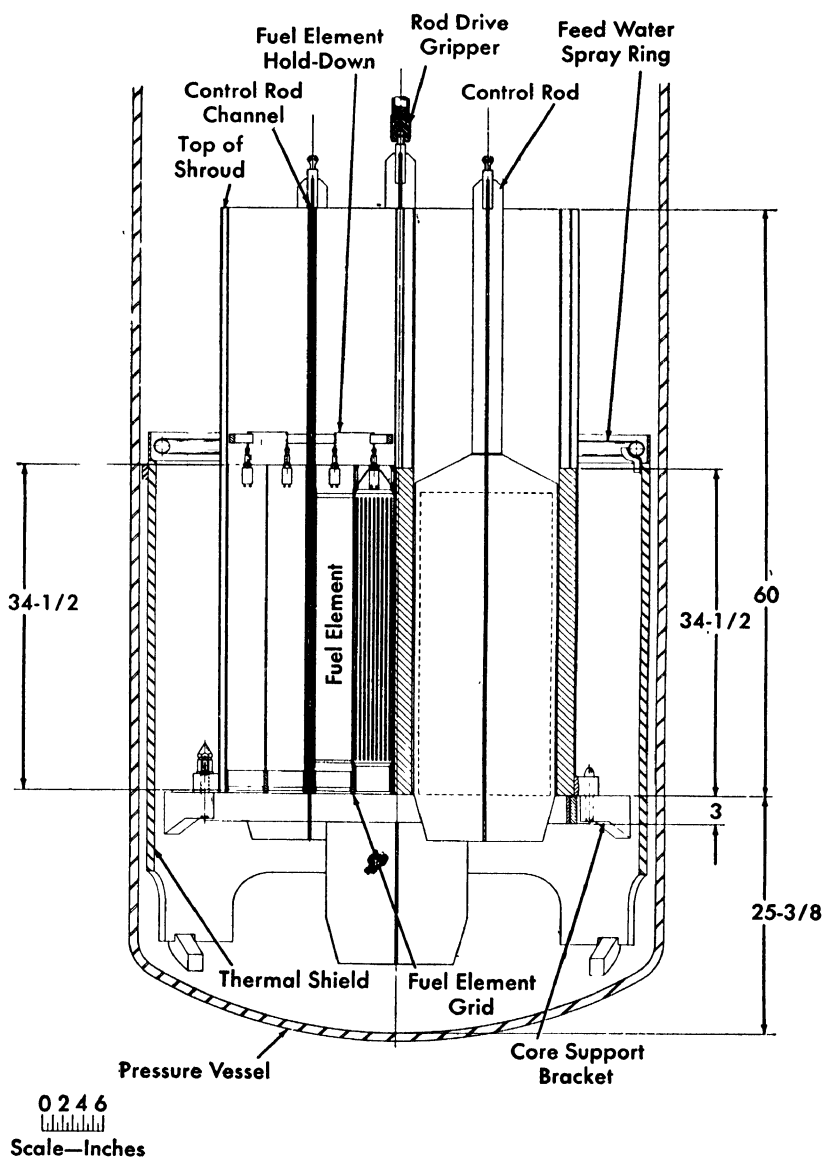


FIG. 8-5. Core vertical section.

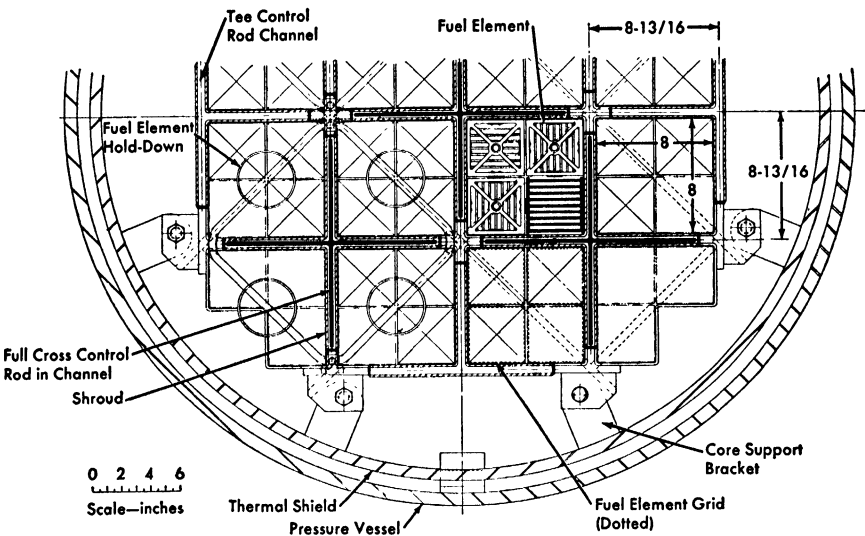


FIG. 8-6. Core plan section.

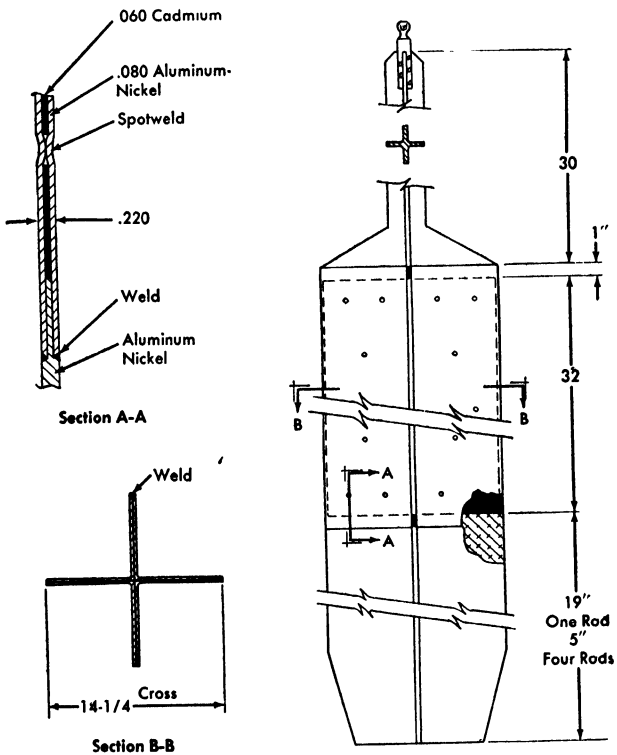


FIG. 8-7. Cross type control rod.

Control rod design data are listed in Table 8-4.

TABLE 8-4
CONTROL ROD DESIGN DATA

Number of cross rods	5
Additional spaces for T-rods	4
Spacing	$8\frac{13}{16}$ in.
Length of cadmium section	32 in.
Thickness	
Cadmium (meat)	0.060 in.
Aluminum-nickel alloy (clad)	0.080 in.
Total travel	31 in.
Scram time	<2 sec
Withdrawal (~ 3 in/min)	0.01% k /sec
Weight of control rod	
Cross rod	49 lb
T-rod	37 lb

TABLE 8-5
NUCLEAR DATA

Average thermal flux in fuel at full power (fresh reactor)	$7.5 \times 10^{12} \text{n}/(\text{cm}^2)(\text{sec})$
Max/avg flux ratio in hot fresh reactor	
Radial	1.6
Control cell	1.3
Axial (control rods out)	1.3
Axial (control rods halfway in)	1.7
Reactor (control rods out)	2.7
Reactor (control rods halfway in)	3.5
Reactivity changes, % k	
Temperature (cold operating)	-1.5 to -2
Xe + Sm	-3
Steam voids	-1.3 to -2
Xenon override	-1 to -1.5
Neutron lifetime (sec)	$4 \text{ to } 8 \times 10^{-5}$ (over range of core conditions)

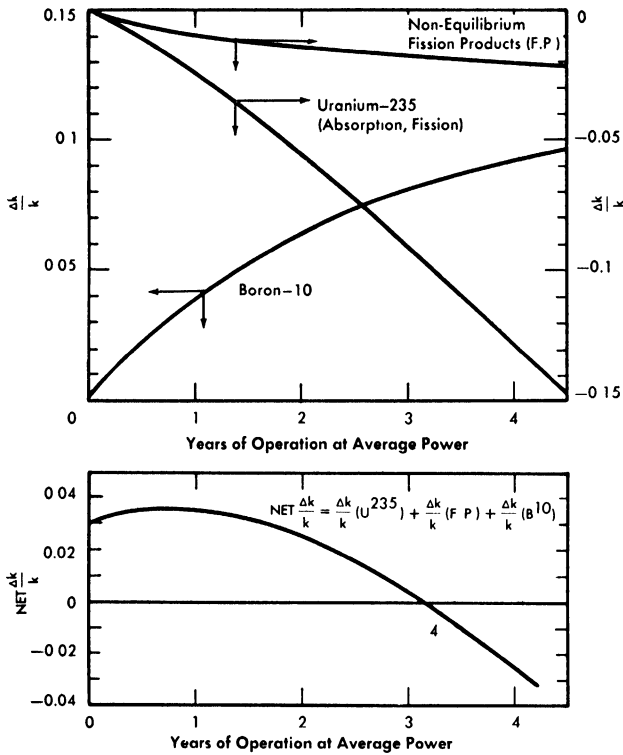


FIG. 8-8. Reactivity variation during core lifetime.

NUCLEAR DESIGN^{1,2,3}

The nuclear design characteristics are listed in Table 8-5.

Diffusion theory was applied to the calculation of over-all reactivity effects in the reactor. The thermal utilization of fuel in the assembly lattice was calculated by a one-group (P_3) method, as was the effectiveness of the control rods in a control cell, a region comprised of four fuel assemblies and half of the surrounding control channels. Two-group diffusion theory was used to calculate the macroscopic or reactor problems after determining the parameters of a homogenized core. The same calculational methods were applied to the BORAX reactors with excellent agreement.

Two-group perturbation theory was used to calculate the reactivity effects of fuel depletion and of boron-10 burnup during the core life. The shape of the real reactor flux was approximated by the product of the radial and axial flux components in the hot fresh reactor. The adjoint

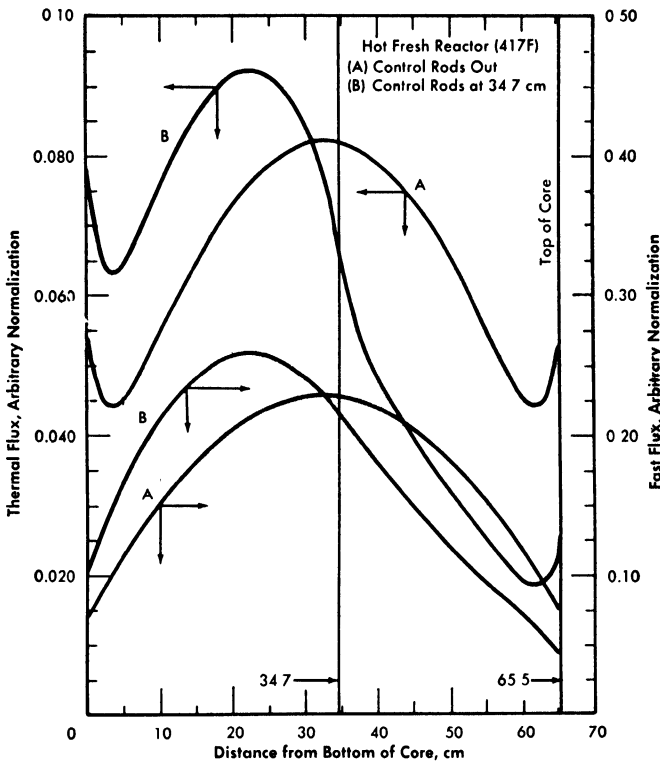


FIG. 8-9. Axial variation of fast and thermal neutron flux.

flux shapes were assumed to be given by the product of the equivalent bare core radial and axial flux components. The perturbation method using the initial flux distribution has been checked against detailed analyses. The lifetime prediction results were in close agreement.

The reactivity components of burnup and the net reactivity are plotted in Fig. 8-8, based on the use of enough boron in the side strips to control 11 per cent in the fresh operating reactor, and on the assumption that the average local thermal neutron flux level is the same in the fuel and in the boron. The reactivity effect of fission product absorption was computed on the basis of published curves.⁴

Multi-region, reflected reactor problems were solved on the UNIVAC to obtain the two-group fast and thermal neutron flux distributions axially (shown in Fig. 8-9) as well as the rod worth (shown in Fig. 8-10). Axially nonuniform steam void distributions were computed and shown to have small effect on the flux distributions. However, the partial insertion of control rods reduces the heat production locally and increases the mean

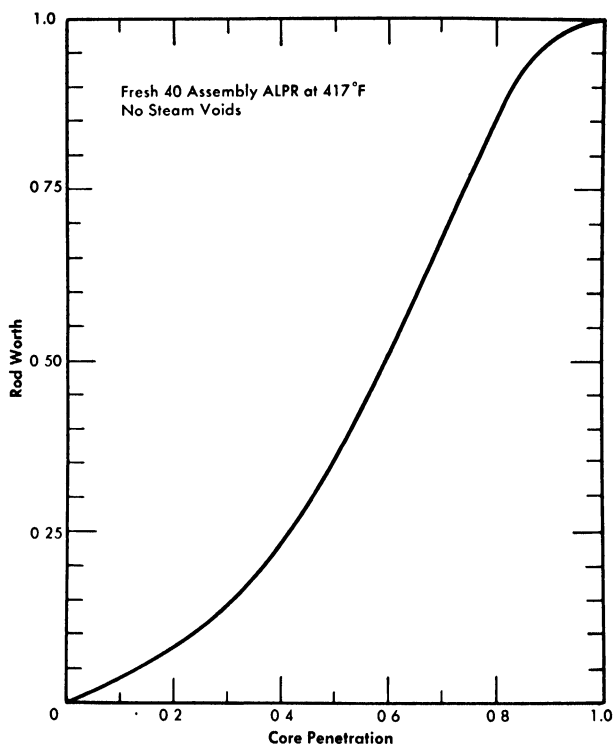


FIG. 8-10. Fractional control rod worth vs. fractional core penetration by rod bank.

steam void in the coolant channels, a negative reactivity effect. This negative reactivity in steam is smaller in absolute value than the steam void reactivities attained in stable operation of the BORAX-II and BORAX-III reactors, operating at 150 psig to 300 psig, respectively.

HEAT TRANSFER CHARACTERISTICS

The empirical factors used in the performance evaluation were obtained from boiling experiments on electrically heated channels, $\frac{1}{4}$ in. and $\frac{1}{2}$ in. wide, and 5 ft long.⁵ In these experiments, with known electrical heat input, measured steam volume fraction, flow rate, and pressure drop, the two-phase frictional pressure drop (liquid and steam) and slip ratios (velocity steam-velocity water) were obtained. Laboratory tests covered a range of pressures from 150–600 psi and up to 80 per cent exit steam voids. The predicted boiling performance is shown in Fig. 8-11.

The heat transfer and fluid flow characteristics are listed in Table 8-6.

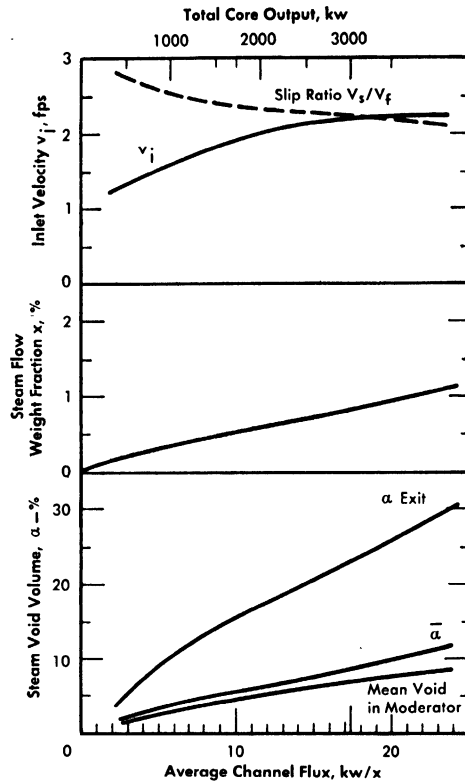


Fig. 8-11. Boiling parameters 300 psi, with 130 to 175 F feedwater (40 elements).

TABLE 8-6
HEAT TRANSFER AND FLUID FLOW DATA

Average power density in core coolant	17.5 kw/liter
Steam flow at 3000 kw	9020 lb/hr
Average steam voids in heated channel	9%
Average steam voids in moderator	7%
Exit steam quality	0.85%
Water recirculation ratio (lb water/lb of steam)	130
Feedwater inlet temperature	175 F
Subcooling at channel inlet	1.8 F
Average boiling length of core	20 in.
Total heat transfer area per 40-element core	475 ft ²
Average heat flux at minimum loading	21,500 Btu/(hr)(ft ²)
Average fuel temperature at centerline	450 F
Average surface temperature plates	440 F

Space heating. The space heating system includes a superheated steam heat exchanger to maintain hot water in a closed heating system. In order to utilize the latent heat of evaporation, the steam side of the heat exchanger is trapped to pass only condensate. The system is controlled by a temperature-sensitive steam throttle valve.

Air-cooled condenser. The exhaust steam is condensed by an air-cooled, finned tube-type aluminum condenser operating at 5 in. Hg absolute pressure. Air cooling was selected as the most practical solution for sites devoid of cooling water.

Condenser cooling is accomplished by controlled mixing of internal circulating and outside air to maintain proper temperature and flow rate. Condenser outlet air is exhausted from the building by a centrifugal blower, inducing an equal flow of colder outside air to enter the mixing chamber. The recirculation rate is automatically adjusted to maintain the condenser inlet air at the lowest permissible temperature above freezing. Since this cooling method is rather unique, its performance will be thoroughly evaluated in the prototype installation. Activity measurements of the condenser gases removed by a conventional air ejector system and discharged to the stack will be made to determine the extent of radiological hazard at this point.

Reactor feedwater system. Condensate from the condenser, air ejector, and space heating system is collected in the hotwell tank and returned to the reactor by one of two feed pumps, provided with automatic changeover. A separate condensate circulating pump provides condensate from the hotwell for the cooling medium in the shield cooling heat exchanger and the air ejector after-condenser. Prolonged stoppage of this pump will result in a gradual temperature rise in the thermal shield, and an automatic shutdown, if remedial action is not taken.

The returning feedwater serves as the coolant for the purification system water cooler, which heats the feedwater from approximately 135 F to 175 F by heat supplied from the purged reactor water. The feedwater is filtered and returned to the reactor above the core.

Primary water purification system. Reactor water is circulated continuously at approximately 3 gpm through a purification system to remove suspended and dissolved impurities and to maintain a pH between 6.5 and 7. The system consists of a holdup tank to reduce N^{16} activity, regenerative heat exchanger for cooling, hermetically sealed pumps, filter, cation exchanger, and a mixed bed demineralizer.

BORAX-III experiments have shown that the resin activity corresponds to the activity of sodium-24. Extrapolation to prototype conditions in-

icates an activity of 5 to 20 r/hr at the surface of the ion column during operation. Gravel shielding reduces this to essentially zero activity above the operating floor.

An improved method has been developed to discharge the resins from the exchange columns into a disposal container by remote control. The purification system is isolated from the high pressure portion of the plant, and water from a large retention tank (also used for reactor water storage) is pumped through the exchangers to force the resins into the disposal tank. This tank is located in a 55-gal storage drum and is shielded by gravel. It has been estimated that the activity at the drum will be approximately 100 mr/hr.

Shield cooling. Two parallel sets of shield cooling copper coils are located in the lead portion of the support cylinder, Fig. 8-3, which also serves to position and support the pressure vessel. Water coolant is circulated through both sets and each set is adequate to remove 65 kw of heat. Since the heat load at full power operation is estimated to be 40 kw (35 kw due to heat generated in the shield by neutron and gamma attenuation and 5 kw of thermal heat from the pressure vessel), failure of one of the parallel circuits would not cause a shutdown.

The shield cooling circuit is a closed forced circulation loop consisting of a condensate cooled heat exchanger, filter, surge tank, and the condensate pump mentioned previously. The circuit also includes a natural circulation radiator and bypass section for emergency shield cooling. Six kw of heat will be removed by natural circulation in the event of a pump failure.

ELECTRICAL SYSTEM

The electrical system components are of commercial manufacture, and with the exception of the emergency power system, require no detailed discussion.

The emergency power system provides uninterrupted power, a necessity for remotely located plants. It consists of a small alternator which normally is driven by a motor connected to the plant electrical system, but upon failure of normal electric power is driven by a motor connected to the station battery.

Equipment connected to the emergency power system will:

1. Close the electrically-operated steam bypass and feedwater valves to prevent pressure buildup in the condenser and consequent loss of reactor water through the condenser safety valve.
2. Drive the control rods inward in the event of rod sticking or failure to scram.
3. Operate essential nuclear and process instrumentation.

INSTRUMENTATION AND CONTROL

Two methods for operating the control rods have been incorporated in the reactor design: (1) manual operation, and (2) automatic operation by steam pressure control. It is anticipated that automatic pressure control will be feasible; however, the transient behavior of the reactor under this type of operation needs to be more fully established. Consequently, plans call for running the reactor initially by manual control.

The control rod drive motors are actuated manually by two control switches, one to control the center rod, and the other to control the peripheral rods. Interlocking circuits prevent the simultaneous withdrawal of two or more rods. Rapid insertion of rods by gravity fall is accomplished by de-energizing magnetic clutches on the control rod drives. A synchro transmitter and receiver effects rod position indication. The instrumentation includes control rod limit switch, safety, alarm, and shutdown circuits for abnormal system conditions.

The nuclear instrumentation provides for measurement of instantaneous reactor power level by four neutron flux detection channels. Two of these channels, one for the detection of very low neutron levels and the other for short-period shutdown, are for prototype operation only and would not be used in future plants. Two channels will be used to monitor air ejector exhaust gamma ray activity and to indicate radiation levels at various locations in the reactor building.

Additional instrumentation incorporates maximum and minimum reactor pressure interlocks and provides for reactor water level measurement and control, as well as various temperature and pressure measurements.

In general, all indicators and recorders requiring continuous surveillance are displayed in the control room, while those requiring less supervision are indicated locally.

REFERENCES

1. M. TRESHOW, A. R. SNIDER, and D. H. SHAFTMAN, Design Study of a Nuclear Power Plant for 100-KW Electric and 400-KW Heat Capacity, ANL-5452, May, 1955.
2. M. TRESHOW et al., ALPR Preliminary Design Study (Argonne Low Power Reactor), Phase I, ANL-5566, April, 1956.
3. A. SMAARDYK, Hazard Summary Report on the Argonne Low Power Reactor (ALPR), ANL-5744, October, 1957.
4. R. W. DEUTSCH, Fission-Product Buildup in Enriched Thermal Reactors, *Nucleonics*, **14**(9), p. 89 (September, 1956).
5. P. A. LOTTES and W. S. FLINN, A Method of Analysis of Natural Circulation Boiling System, *Nuclear Science and Engineering*, **1**, 461-476 (December, 1956).
6. C. R. BRAUN et al., Argonne National Laboratory, Argonne Low Power Reactor, A Prototype Direct Cycle Boiling Water Reactor Package Plant for Electric Power Production and Space Heating, OIC-1222, P/455. Second International Conference on Peaceful Uses of Atomic Energy, Geneva, 1958.

CHAPTER 9

THE DRESDEN NUCLEAR POWER STATION

The Dresden Nuclear Power Station, being built by the Commonwealth Edison Co. of Chicago together with the companies in the Nuclear Power Group, Inc., is of great importance in the field of boiling water reactor technology for several reasons. First, with its net electrical generating capability of 180,000 kw, it is the largest all-nuclear power plant under construction in the United States. Second, it is the first actual dual-cycle boiling water reactor plant undertaken. As explained in Chapter 7, the Vallecitos plant of General Electric, which was built largely as a pilot plant for the Dresden Station, can be operated on a simulated dual cycle, but its purpose is experimental whereas Dresden will be a full-scale dual-cycle commercial electric power plant. Dresden is important for a third reason and that is that its comparatively low construction cost is the lowest of any other commercial nuclear power plant either in operation or under construction at the present time. Finally, this station is important because it is financed entirely by private enterprise.

Associated with the Commonwealth Edison Company in this project is the Nuclear Power Group, Inc., which serves as a means for a number of companies connected with the electrical industry to participate actively in the development of nuclear power.*

Dresden Station is being designed and constructed by the General Electric Co., with the Bechtel Corporation as engineer-contractor. Bechtel is to provide all the construction services for the more conventional portions of the plant. General Electric is doing all the engineering design for systems directly associated with the generation of steam by the nuclear process. Design is being carried out by GE's Atomic Products Equipment Department at San Jose.

*The Nuclear Power Group, Inc., is composed of the following companies: American Electric Power Service Corporation of New York, N. Y.; the Bechtel Corporation of San Francisco, California (the only non-utility company); Central Illinois Light Co.; Commonwealth Edison Company; Illinois Power Company; Kansas City Power & Light Co., Kansas City, Mo.; Pacific Gas & Electric Co. of San Francisco, Calif.; and the Union Electric Company of St. Louis, Mo.

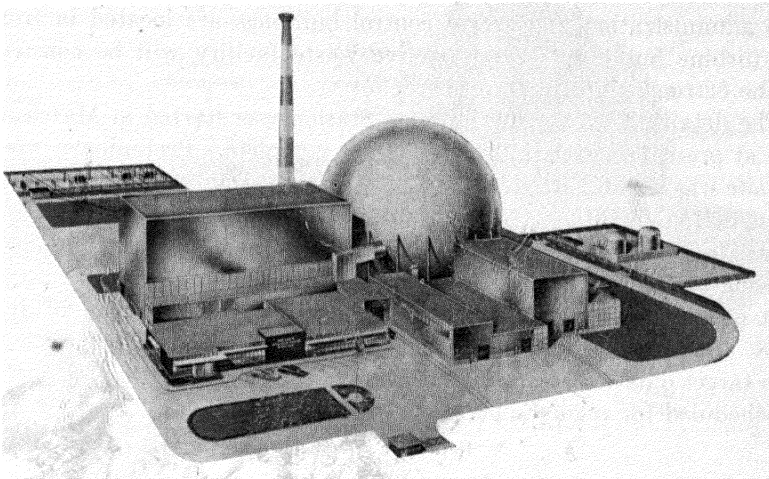


FIG. 9-1. Artist's drawing of Dresden Station.

essential to nuclear power progress. While Dresden as a *going concern* will be a regular producing generating station of the Commonwealth Edison Company system, in a broader sense, the plant will serve as a springboard for future nuclear power development.

SITE AND GENERAL FEATURES

Dresden Station is being built on a 950-acre site in a sparsely populated agricultural area in Grundy County, Illinois, at the point where the DesPlaines and Kankakee Rivers meet to form the Illinois River. This site has many advantages. There is a plentiful supply of water for condensing purposes; the location is isolated; and yet the site is not too difficult for access to rail and highway and transmission line outlets. It is 47 miles southwest of Chicago.

The average population density in the 5-mile radius zone surrounding the site is approximately 15 per sq mi. The most populated areas within 25 miles are the cities of Joliet (14 mi away) and Aurora (25 mi away), each of which has a population of about 50,000.

An artist's conception of the completed plant is shown in Fig. 9-1, and Fig. 9-2 is a ground plan showing the location of the buildings. A steel sphere, 190 ft in diameter encloses the reactor and associated nuclear auxiliaries to contain contamination in the unlikely event of a major nuclear accident. The large rectangular building to the left of the sphere contains the conventional power plant components, the turbine-generator, condenser and auxiliaries, feedwater heaters, and the main control room.

The administration and access control buildings are located in front of the turbine building. A radioactive waste facility will be constructed at the extreme right of the sphere.

The detail design for the Dresden Station was started in March, 1956, and at present is essentially 97 per cent complete. Preliminary work on the site was begun November 28, 1956, with the construction of temporary construction facilities. Actual construction began March 1, 1957, with excavation for the reactor enclosure and turbine building foundations. Testing of the reactor enclosure was completed March 27, 1958, and the first concrete pour for the internal structures was made on March 31, 1958. At present (June 1958) the station is about 30 per cent complete. The target date for completion of construction is late 1959, and the plant is scheduled for regular operation in 1960.

DESIGN OBJECTIVES

The fundamental objective in the design of Dresden is to produce a safe, reliable, and economical central station. The conditions which will satisfy the project objectives are: 180,000 kw, net, electrical output at 85 per cent power factor and with 75 F condensing cooling water; capability for operation at 60 per cent capacity factor and, as already stated, total energy costs approximately comparable to those of the most efficient plants in the Commonwealth Edison system.

Basic criteria were developed after initial studies to guide the design work. Some of the more important of these can be stated as follows:

1. An effective enclosure to prevent the spread of radioactive material over the surrounding area in the event of a nuclear disaster.
2. Personnel may enter the enclosure during normal operation but the requirements will be held to a minimum.
3. Whenever feasible, maintenance will be performed by direct contact rather than through remotely controlled operations.
4. A centralized system of plant control will be provided.
5. The primary nuclear fuel is to be uranium oxide, although modifications of core design and the substitution of fuels of various enrichments after startup will be provided for.
6. Components will be adequate for an anticipated operating life comparable to conventional plant equipment.
7. Maximums for critical reactor parameters are as follows:
 - (a) 346,000 Btu per hr per sq ft maximum fuel element heat flux.
 - (b) 4570 F uranium oxide fuel temperature.
 - (c) 2.5 per cent reactivity compensated by steam voids in the reactor moderator.

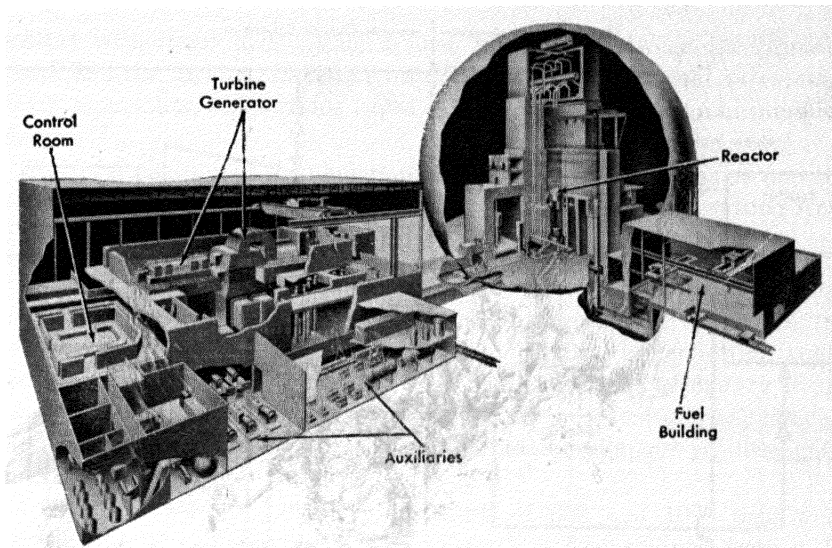


FIG. 9-2. Cutaway drawing of the 180,000-kw boiling reactor plant.

DUAL CYCLE OPERATION

The Dual Cycle Design was selected for Dresden as the best solution to fulfill the design objectives, though other methods were considered. The principles of the dual cycle system are covered in an earlier chapter. It was selected for Dresden, first, to secure a higher power output per unit of core volume, and second, to gain improved reactor response to load demand.

Both of these objectives are related to the variation of nuclear reactivity with density changes caused by boiling. The dual cycle approach is to remove as much energy as possible in the upper part of the reactor without exceeding the limit of reactivity for good stability. A simplified flow diagram of the dual cycle as used in the Dresden design is shown in Fig. 9-3. The boiling portion is controlled by the degree of subcooling at the inlet to the reactor. To accomplish this, the steam-water mixture flows up to a steam separating drum, and the saturated steam, essentially at reactor pressure, then flows directly to the turbine. The water flows down from the drum and is pumped through four heat exchangers which are called secondary steam generators where heat is removed to produce secondary steam at a somewhat lower pressure (500 psig). This secondary steam is admitted to the turbine at the ninth stage.

The dual cycle results in self-stabilized performance under load changes. Most load changes are accomplished by varying the demand from the low

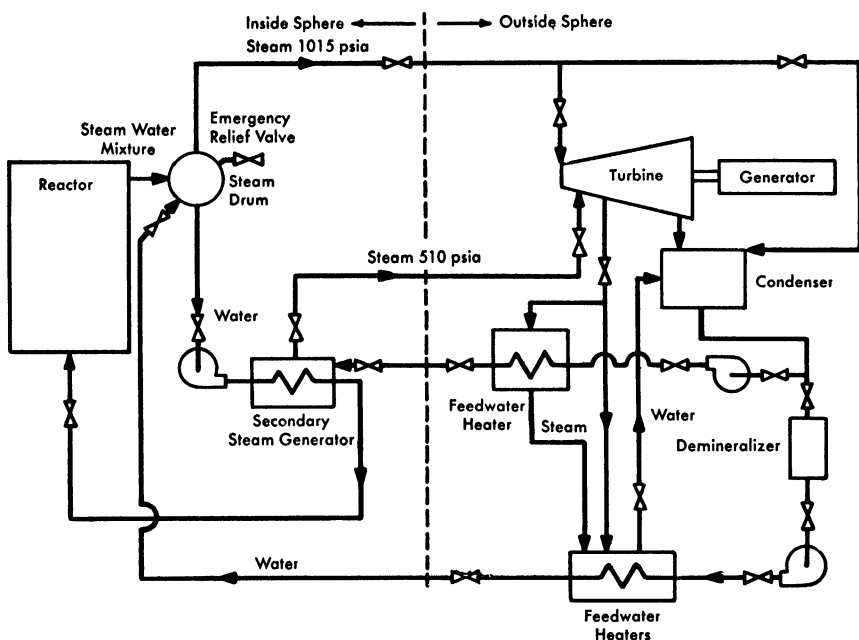


FIG. 9-3. Simplified flow diagram of the Dresden system.

pressure steam supply. When the low pressure steam control valve, which is the governor valve on the turbine, is opened, more heat is extracted from the water, increasing the subcooling and reducing steam-bubble formation within the reactor, thus increasing the power output.

If the reactivity is suddenly increased, steam is formed in the core which tends to shut the reactor off before the fuel overheats; thus the basic safety feature of the simple boiling reactor is retained in the dual cycle system.

SYSTEM PARAMETER CONSIDERATIONS

The design of the Dresden plant is based on a judicious combination of long-established practice and conditions sufficiently advanced to make a substantial contribution to the art. Many of the decisions made in establishing the design are based on a desire to keep the design as simple as possible with a minimum of equipment and to take advantage of the inherent stability, simplicity, and safety of the dual cycle boiling reactor.

Consideration was given to various alternate schemes of the design features comprising the system. Some of the important ones are: a range of system pressures, subcooling by heat exchanger or flash tank, super-

heating with fossil fuel, use of a heat exchanger between the primary steam and the turbine, forced vs natural circulation, internal vs external steam separation, and various types of recirculation pumps and pumping systems.

A brief discussion of the choices made between these alternates and the selection of some of the more important system parameters resulting from technical and economic evaluations follows:

1. *Primary to secondary steam ratio.* To avoid instability, or "chugging" in the reactor, it is necessary to limit the steam formed in the reactor core. Economic studies to determine how this condition could best be satisfied resulted in the selection of a primary to secondary steam ratio of 1.2. These studies required balancing the effect of various coolant flow rates, auxiliary power demands, heat exchanger requirements, and thermal efficiency.

2. *System pressure.* Primary pressures of 600 and 1000 psig were considered. The higher value was selected as fuel savings from a 10 per cent improvement in heat rate outweighed higher capital cost. Surface temperature limitations of the fuel elements made consideration of higher pressures unfavorable. The net station heat rate for the 1000 psig system is calculated to be 11,893 Btu per kwh. For the 600 psig system, the corresponding figure is 13,150.

3. *Core and fuel dimensions.* The principal considerations affecting core and fuel dimensions are as follows:

(a) Existing fabrication and transportation facilities limit the inside diameter of the vessel containing the nuclear core to slightly over 12 ft for a 1000 psig system.

(b) The maximum allowable heat flux is set at approximately 300,000 Btu per hr per sq ft and the calculated maximum center fuel element temperature is 4570 F.

(c) Various mechanical problems made a core length greater than 10 ft undesirable.

Balancing these various factors to obtain optimum conditions results in the selection of a core approximately ten ft in diameter with an active length of nine ft and comprised of 488 fuel assemblies.

4. *Steam separation.* An external drum was selected to insure adequate separation of steam and water. Moisture carryover to the turbine and possible cavitation in the recirculating circuit are considered important.

5. *Forced versus natural circulation.* Forced circulation is used because it provides maximum positive control and distribution of core flow. This is important for a core of this size.

6. *Enclosure type.* A number of enclosure types, including underground vaults, dome-top cylinders and spheres for both complete and partial

plant containment were considered. A sphere containing only the reactor and selected auxiliaries was found to provide the required safety most economically and with better physical arrangement.

7. *Recirculating pumps.* The desire for eliminating shaft sealing problems led to the selection of the canned rotor pump for reactor recirculating duty.

8. *Heat exchanger versus flash tanks.* The heat exchanger rather than the flash tank was selected as the means of subcooling for the following reasons:

(a) Lower auxiliary power required for pumping the reactor recirculating water.

(b) Simplicity of primary flow control.

9. *Heat exchanger for primary steam.* The use of a heat exchanger in the primary steam, as well as the secondary, was considered. This was discarded because it was more expensive, resulted in lower thermal efficiency, and because it was felt that radioactive steam could be handled successfully in the turbine cycle.

GENERAL DESCRIPTION

Heat balance. The heat balance diagram at rated load, shown in Fig. 9-4, indicates the main flow paths as well as cycle performance. The steam flow requirements to meet the 180,000 kw net electrical output at 2½ in. Hg back pressure are 1,407,000 lb per hr at 1000 psig and 1,188,000 lb per hr at 500 psig. The calculated net station heat rate is 11,893 Btu per kwh and the corresponding reactor heat output is 626 megawatts.

The general arrangement of the reactor building, the turbine-generator building and the fuel handling building is shown in Figs. 9-5 and 9-6. The reactor and its auxiliaries are installed in the containment sphere with the turbine-generator and its auxiliaries in the turbine building external to the sphere. The major items of equipment in the containment sphere are the reactor vessel, the primary steam drum, secondary steam generators, and the recirculating pumps. The turbine-generator set, the control room, demineralizer equipment and the usual turbine and condenser auxiliaries are in the turbine building.

Sufficient compartmentalization for major items of equipment is provided so that maintenance can be performed by direct contact during operation of the remainder of the plant. Sufficient biological shielding is provided to allow limited access to the reactor and turbine buildings during operation for observation and inspection purposes.

The fuel handling and storage building is located adjacent to the reac-

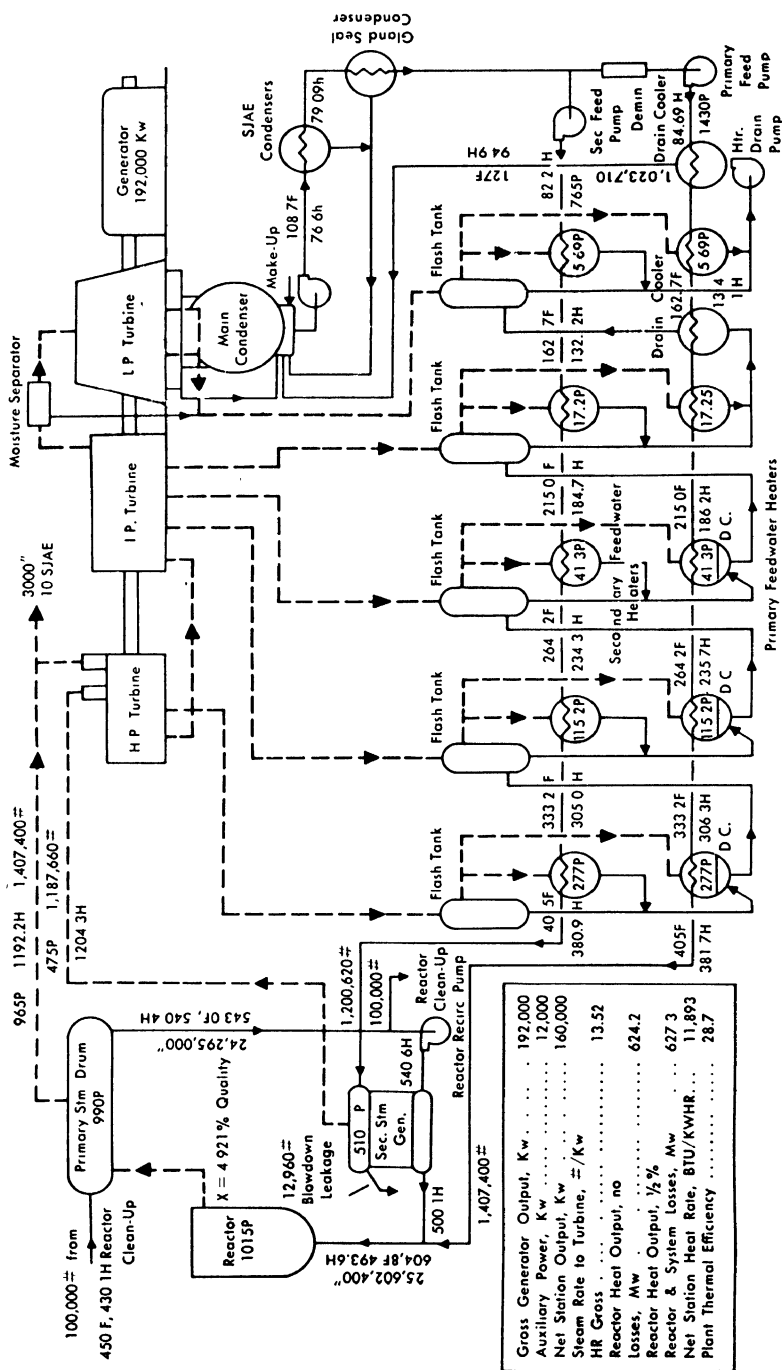


Fig. 9-4. Heat balance diagram of the Dresden dual-cycle system.

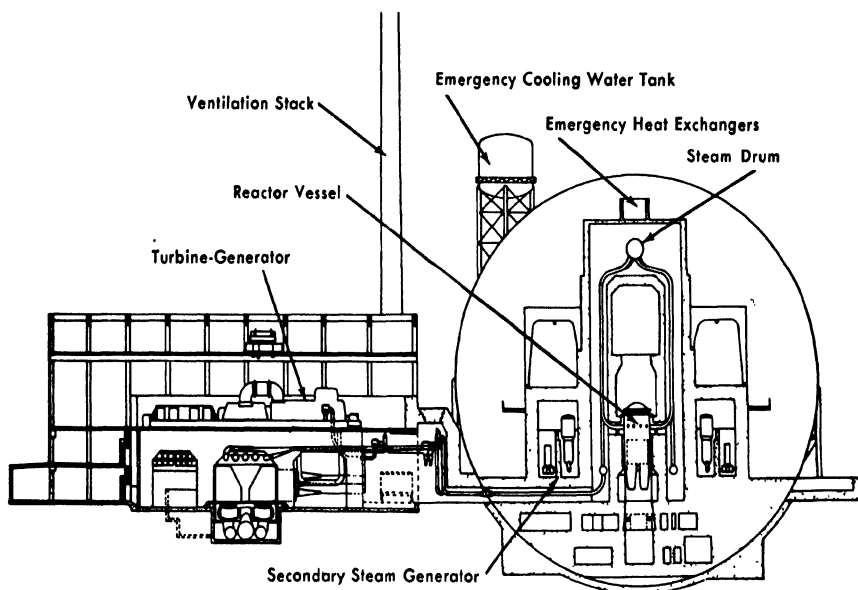


FIG. 9-5. Vertical section of the sphere and turbine building.

tor enclosure. It is connected to the reactor building by an underground canal and a discharge tube which may be isolated from the exterior when not in use. An emergency cooling-water tank provides cooling in the event of loss of the main heat sink, the condenser.

REACTOR ENCLOSURE (CONTAINMENT VESSEL)

The 190 ft spherical steel shell setting 40 ft in the ground housing the reactor but not the turbine was selected as the enclosure type on the basis of extensive studies of the safety and merits of alternative structure types. The studies, conducted by General Electric in collaboration with the Bechtel Corporation embraced a wide variety of potential designs, the more important of which were as follows:

- (a) Above ground and underground structures.
- (b) Steel, concrete, and combinations of the two as structural materials.
- (c) Spherical, cylindrical, and combination (spherical and cylindrical) shapes.
- (d) Enclosures alternatively housing and not housing the turbine along with the reactor.

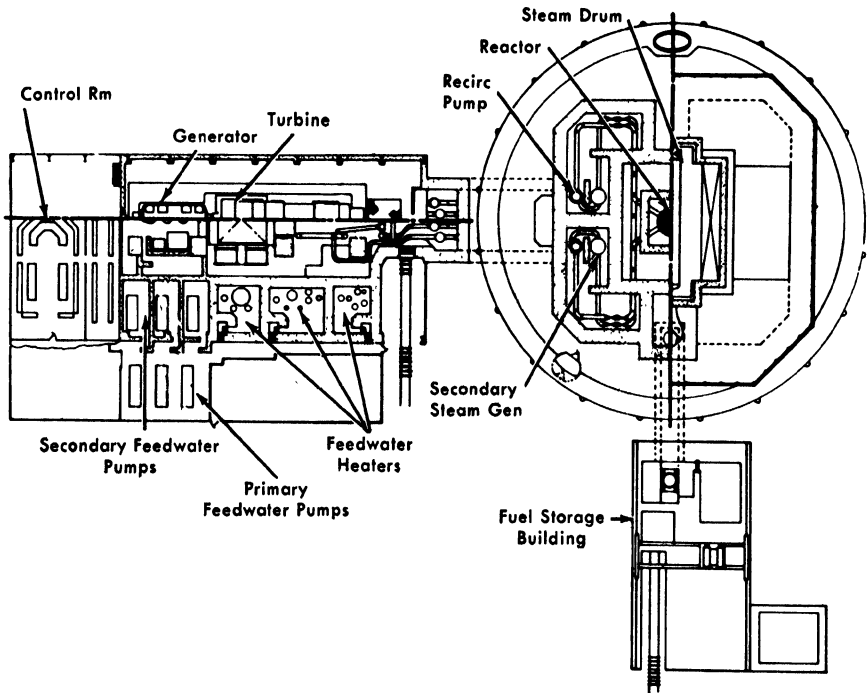


FIG. 9-6. Plan view of the Dresden showing the general arrangement of equipment.

A highest practical degree of structural integrity and reliability was sought as the essential requirement, with safety comparable,—an economical and convenient housing was desired. While it is recognized that experience with large structures of this kind is limited and that, therefore, many important factors had to be weighted by considerations necessarily quite speculative, the enclosure type chosen was judged to be best suited to the requirements. The enclosure size, 190 ft in diameter, was determined by the space requirements for the equipment and accessories enclosed and by considerations of reasonable convenience of layout and equipment.

The choice of a *reactor-only* enclosure (i.e., one not housing the turbine-generator along with the reactor — as in the case of EBWR) was based on the safety comparison summarized below and on the following other considerations:

- (a) It results in a more convenient layout.
- (b) By housing the turbine in a more nearly conventional building, it renders the plant more amenable to future modifications—an important consideration in a developmental project.

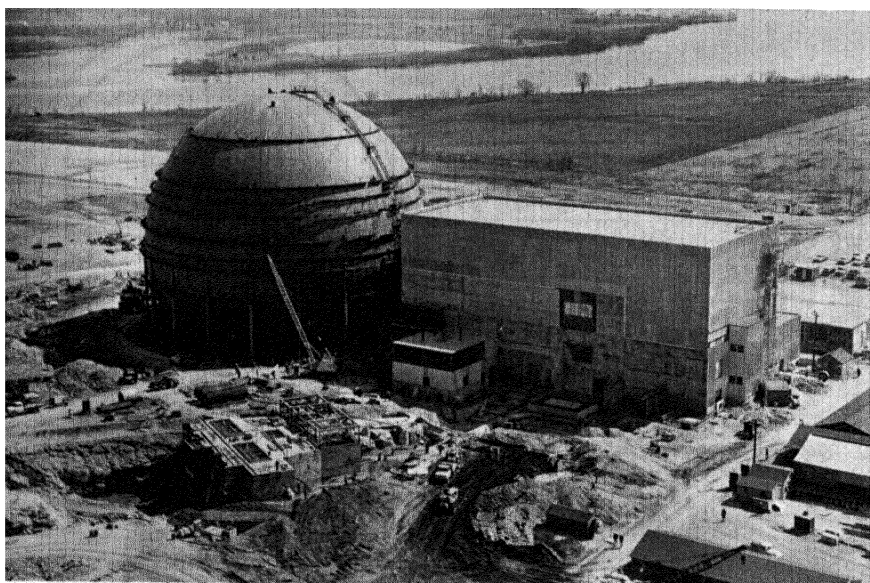


FIG. 9-7. Construction view of Dresden Station, April, 1958, showing the 190-ft sphere complete, and also the turbine building at the right.

- (c) For equivalent safety, the reactor-only enclosure is economically the more advantageous.
- (d) By reducing layout and structural complexities, it represents a trend desirable from the standpoint of the long-range development of the atomic power industry.

From the nuclear safety standpoint too, the reactor-only type of enclosure offers advantages:

- (a) The danger of damage to the enclosure in the event of a turbine explosion is minimized with the turbine outside, rotating in a plant not intersecting the enclosure.
- (b) No penetration through the enclosure wall is required for the generator main leads. This is a particularly vulnerable type of penetration, involving a large-diameter disc of non-magnetic material and temperature differences due to eddy currents.
- (c) The danger of damage to the enclosure by explosion of the hydrogen used to cool the generator is eliminated.
- (d) With less lubricating oil in the enclosure, the fire safety is improved.
- (e) The total area of penetrations is decreased by approximately 30 per cent.

- (f) There are fewer man-hours spent inside the enclosure, unprotected in the event of a nuclear accident.
- (g) With less equipment enclosed, the need to cut temporary openings in the shell for removing or bringing in large pieces is likely to be less frequent.
- (h) The enclosure will have several minor advantages by virtue of its smaller size.

On the other hand, the reactor-only enclosure also has disadvantages from the safety standpoint.

- (a) Valves in the reactor-to-turbine steam lines must be relied on to complete the enclosure in the event of an accident.
- (b) Accidental closure of these valves would separate the reactor from its normal heat sink, the turbine condenser, thus necessitating special provisions to remove the reactor heat by other means.

The design pressure for the sphere is 29.5 psig. Its design conforms to the ASME Code with certain exceptions, these being that no pressure relief valves and no stress relieving of the final welds is included. To conform with the stresses allowed by the Code at the design pressure, the plate thickness is 1¼ in. at the top and 1.4 in. at grade level. The lowest point of the sphere is 39 ft below grade.

Access for personnel and equipment is provided by three pressure locks. Each of these locks has two doors in series, with an interlocking arrangement which assures that one door is closed at all times. A large opening through the sphere is provided by a bolted access door which is 17 ft in diameter. This opening permits the removal of large pieces of equipment during plant shutdown. Seal welding is employed to maintain this opening pressure tight during operation. The sphere is penetrated vertically by a fuel transfer tube 40 in. in diameter. This tube leads into a canal from the fuel building. Additional openings, ranging in size from 2 to 40 in. in diameter, provide for ventilating air ducts, steam, water, instrumentation piping and electrical cables.

Steel columns around the equator of the sphere provided its only support during erection. After the enclosure was completed the space between the rock excavation and the sphere was filled with concrete to within 8 ft of the ground surface to provide a foundation to carry the load of equipment and the concrete inside the sphere. The last 8 ft was filled with sand to distribute the stress when transferring from ground to air exposure. Over 22,000 cu yd of concrete was required for the foundations and shielding for the nuclear reactor.

Basis of internal pressure. The enclosure design is based on adequacy for pressures that could be developed in severe terminal accidents despite the fact that the possibility of occurrence of such accidents is extremely remote.

The probably worst accident that might occur would produce internal pressure from three different sources of energy: (a) the pressurized hot water in the reactor and in those of its auxiliaries which are not separated from it by its solid barrier; (b) a nuclear excursion; and (c) chemical reaction between reactor components, viz., between water and the zirconium cladding in which the uranium-oxide fuel is encased. The hot water contribution (a) is the most important of these; the nuclear and chemical contributions, (b) and (c), would probably be negligible in comparison.

Accordingly, the design pressure of the enclosure, 29.5 psig, has been determined by the pressure that would be created by the release (and thus partial flashing into steam) of the pressurized hot water contained in the reactor and primary recirculation loop under normal operating conditions, with no significant nuclear or chemical energy contributions. Under these conditions, i.e., with the reactor at its rated working pressure (1000 psig) in a boiling condition, operating at its rated power level (626 thermal Mw), a reactor rupture would expose 188 tons of pressurized hot water, and the ensuing partial flashing to steam would create a 23-psig pressure in the enclosure. This 23-psig post accident pressure was calculated on the basis of an instantaneous release. In the event that the release should take a significant time the ultimate pressure reached would be less.

The *worst reasonable* post-accident pressure that might occur would arise from not only the flashing into steam of the hot water in the system but also a nuclear excursion contributing an amount of energy that would melt all the uranium-oxide fuel in the reactor core plus a chemical reaction of 25 per cent of the zirconium in the fuel element cladding with water. Such a worst reasonable accident would give rise to an enclosure pressure of 31 psig. This is still within the 37 psig (125 per cent of design pressure) test pressure.

It is possible, however, to conceive hypothetical combinations of circumstances so extreme that it would not be reasonable to ascribe to them a significant possibility of occurring which could give rise to pressures exceeding those created by the worst reasonable accident just described. However, the structural safety factors used in the design of the enclosure give it a good chance of withstanding pressures significantly in excess of the test pressure.

For example, suppose that the chemical reaction involved in the worst reasonable accident is augmented by burning of the liberated hydrogen (an extremely unlikely possibility) and proceeds with such violence that missiles formed in the explosive reaction rupture the secondary steam generator, so that the pressurized hot water on the secondary side, as well as that on its primary side, becomes exposed to partial flashing to steam.* The enclosure pressure after such an accident would be 34 psig.

These considerations involving the design of the containment have been set forth in some detail because setting the design basis for a reactor enclosure represents a task quite unique in engineering experience. It involves a structure designed for a function that it will almost certainly never be called upon to fulfill. This surrounds the specifications of that function with unique uncertainties. The design of the Dresden Station enclosure will provide highly reliable assurance to the public that it will not be affected with a disaster even in the unfortunate event that the extremely improbable should happen.

A concern of the atomic industry in designing enclosures is that, in view of the uncertainties surrounding the functional specifications, these should not be set with unrealistic pessimism. The direct and indirect costs of an enclosure represent a substantial contribution to the total cost of nuclear reactor power plants, in addition to imposing inconvenient restrictions on plant layout. The designers of Dresden are confident that eventually, through increased engineering wisdom and further substantial reassuring experience with nuclear reactors, the industry will see its way clear to building nuclear plants anywhere without any containment provision. A more immediate objective in this area is the development of more realistic appraisals of the course of any potential reactor accidents, with a view to reducing the amount of *ignorance margin* that must be included in present enclosure design.

THE DRESDEN REACTOR PLANT

In describing the reactor system, reference again is made to the heat balance diagram shown in Fig. 9-4. Energy produced in the reactor core is absorbed by the water flowing through the fuel channels. The resulting steam-water mixture then flows to an external separating drum. Sufficient drum internals are provided to give the desired purity to the steam flowing to the turbine. Good separation is also required to eliminate the possibility of cavitation in the recirculating pumps.

*The possibility of such a rupture is remote because the secondary steam generators are 30 ft away from the reactor and there are 12 ft of concrete wall thickness in between.

The water from the separating drum is pumped through the secondary steam generators which extract heat from the fluid. This produces secondary steam and cools the water to give the desired subcooling before returning it to the reactor vessel.

There are four separate recirculation loops, each consisting of a secondary steam generator, recirculating pump, and secondary valving. Each of these loops can be isolated as a unit.

A physical picture of the reactor plant is shown in the isometric view in Fig. 9-8. This illustration shows the main equipment and piping in skeleton form with the biological shielding. The steam-water mixture produced in the reactor flows through the riser pipes to the primary drum which is centrally located approximately 85 ft above the reactor vessel. The downcomers feed two lower headers which in turn supply the pumps and the secondary steam generators. The primary steam drum is sufficiently high to permit the fuel handling operations to take place in the space between the drum and the top of the reactor vessel.

Each steam generator and pump unit is installed in a separate shielded compartment. After a unit is isolated from the remainder of the system, maintenance in the individual cells can be performed on an 8-hour basis. The unit system was selected for the following advantages:

1. Simplified piping.
2. Reduced water quantity in the primary system.
3. Decreased number of large valves.
4. Lower capital costs.

The plant is capable of stable operation with one or two of the secondary loops out of service.

The primary steam drum is sized to provide approximately 1700 cu ft of water storage volume. It is approximately 8½ ft in diameter and 60 ft long. The drum served the dual role of reservoir and separating device. The drum size is determined by surges between the reactor vessel and the drum during transient conditions rather than on a steam-water separation basis alone. Steam-water separation is accomplished by using conventional stainless steel separator devices. The high drum arrangement produces good natural circulation characteristics and increases the available NPSH on the pumps.

The canned rotor reactor recirculating pumps are completely sealed. The bearings are a Kingsbury type thrust bearing. Water is circulated through the motor by an internal water circulating pump to provide bearing lubrication and cooling. A heat exchanger is provided external to the motor stator to transfer heat through the motor casing to low pressure cooling water. The pump is designed so that the motor and pump inter-

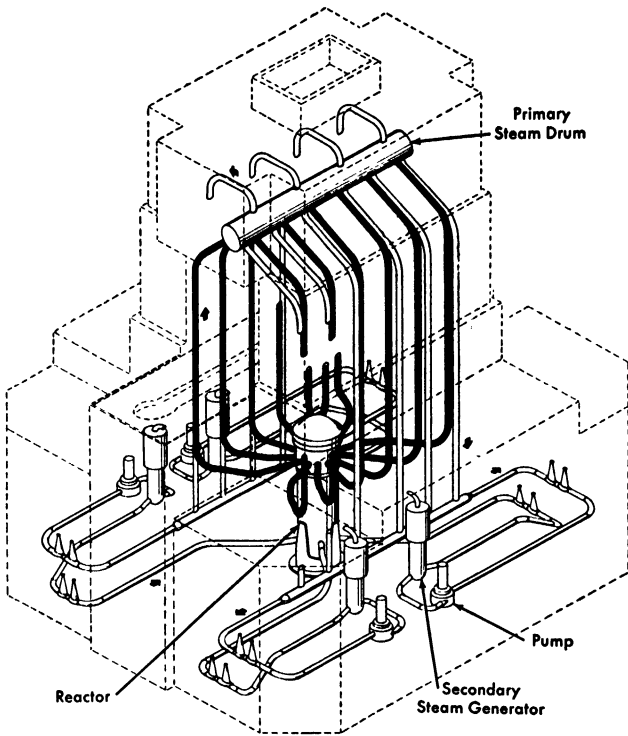


FIG. 9-8. Isometric view of the reactor and steam generators.

nals can be easily and quickly removed from the pump bowl for maintenance, leaving the pump bowl in the piping system. Both the rotor and stator are canned to prevent water from penetrating the windings. A thermal barrier is designed in the pump and motor combination to minimize heat transferred from the pumped fluid to the motor section.

Figure 9-9 shows a section of vertical type secondary steam generators furnished to produce the 500 psig steam. The primary fluid is pumped through the inverted U tubes. The secondary fluid and the steam separation equipment are contained in the vertical cylindrical shell.

An emergency cooling heat exchanger is required to remove the reactor decay heat in the event of loss of the main condenser. Steam from the reactor is condensed by water stored in an emergency storage tank, and this water is evaporated to atmosphere. Sufficient storage is provided in the cooling water tank to take care of the decay heat for several hours after shutdown.

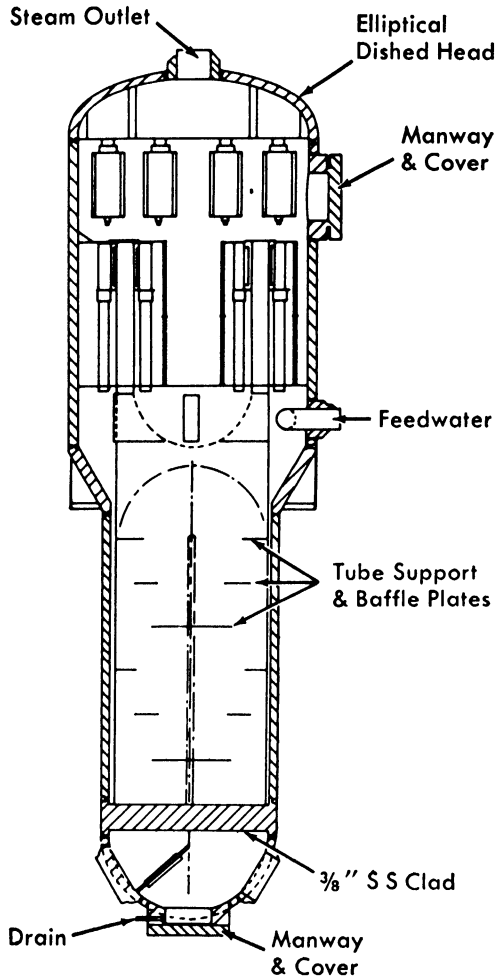


FIG. 9-9. Section of the vertical type secondary steam generator. Four such steam generators are installed.

To remove the reactor decay heat during an unloading operation, a separate circuit is provided. In this system primary water is pumped through a heat exchanger transferring heat to river water.

REACTOR DETAILS

The principal features of the reactor are depicted in Fig. 9-10. Its important design parameters are listed in Table 9-1.

The reactor vessel is a vertical cylindrical pressure vessel, 40 ft, 9 $\frac{5}{8}$ in.

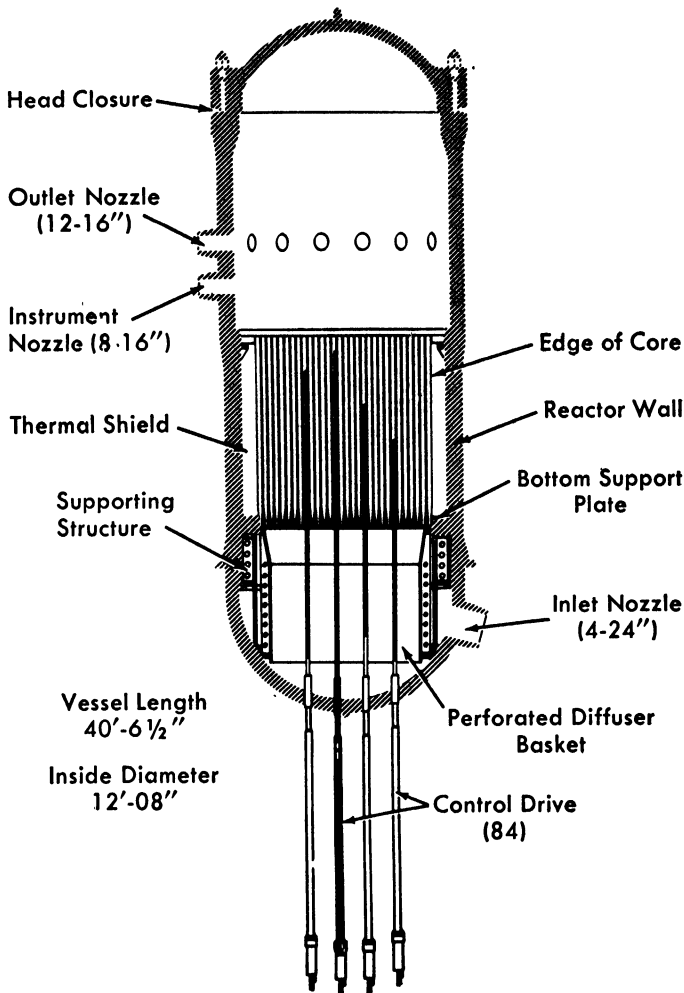


FIG. 9-10. Longitudinal section of the reactor.

over-all inside height (including the dished top and bottom heads) and 12 ft 2 in. inside diameter. It is made of low-alloy steel, clad inside with stainless steel for improved corrosion resistance. The shell thickness opposite the nuclear fuel core is $5\frac{5}{8}$ in. (including the $\frac{3}{8}$ -in. cladding). Table 9-2 gives Reactor Vessel design parameters.

The design pressure is 1250 psig, sufficiently above the 1000-psig normal operating pressure to allow (with margins) for pressure fluctuations arising under various transient operating conditions. In selecting the design temperature of the reactor vessel, account was taken of heating by gamma

TABLE 9-1
SALIENT NUCLEAR CHARACTERISTICS OF REACTOR

Item	Approximate value
Reactor thermal power	626 Mw
Isotopic enrichment	1.5 per cent
Reactivity to control	
Cold to hot	0.1 per cent <i>k</i>
Steam voids	2.3 per cent <i>k</i>
Xenon and samarium	4.0 per cent <i>k</i>
Burnup and maneuvering	5.9 per cent <i>k</i>
Doppler effect	1.0 per cent <i>k</i>
Total Operating Control	13.1 per cent <i>k</i>
Shutdown margin	4.5 per cent <i>k</i>
Total Control-rod Worth	17.6 per cent <i>k</i>
Delayed neutron fraction	
Initial	0.86 per cent
Equilibrium	0.64 per cent
Neutron lifetime	60 microseconds
Temperature coefficients of reactivity	
For slow changes (governed by moderator)	-0.012 per cent <i>k</i> per deg F*
For sudden power increases (governed by Doppler effect)	-0.0008 per cent <i>k</i> per deg F

* With no steam voids. The full-power void coefficient is approximately -0.12 per cent *k* per, per cent increase in voids.

rays and neutrons, in addition to the effect of the coolant temperature. The saturation temperature of the coolant at the 1000 psig operating pressure is 546 F. The maximum temperature difference in the vessel wall between the inside of the stainless-steel cladding and the outside wall is 68 F. This maximum temperature difference occurs opposite the core. Analyses are being made regarding the stress effects of temperature variations that may occur in the course of the vessel's service. The results of these analyses will be used to establish such operating procedures as may be prudent to avoid undesirable high transient stresses.

The 12 ft, 2 in. inside diameter vessel accommodates a core assembly of 10 ft 9 in. outside diameter. A 2-in. width of the 8.5-in. annulus around the core is taken up by a cylindrical core guide support member which also functions as a thermal shield. The remainder of this annulus will be filled with water, which acts as a neutron reflector.

The reactor cooling water enters the mixing plenum below the core

TABLE 9-2
REACTOR VESSEL DESIGN PARAMETERS

Inside length, including heads	40 ft 9 $\frac{5}{8}$ in.
Inside diameter	12 ft 2 in.
Inside throat diameter	11 ft 6 $\frac{1}{4}$ in.
Cladding thickness of stainless steel (Type 304 S.S., 0.20% Co)	$\frac{3}{8}$ in.
Wall thickness of cylindrical portion including $\frac{3}{8}$ inch cladding	5 $\frac{5}{8}$ in.
Maximum rate of change of vessel water temperature during startup and shutdown	100 F/hr
Normal operating pressure	1000 psig
Design pressure	1250 psig
Test pressure, at 120 F minimum metal temperature	1875 psig
Normal operating temperature	546 F
Maximum wall temperature	621 F
Maximum temperature differential across wall	68 F
Design temperature	650 F
Weights	
Head closure	52 tons
Body	201 tons
Internals	30 tons
Core assembly	104 tons
Control assembly	22 tons
Water, normal operation	62 tons
Water, vessel full	129 tons
External water supported by vessel during refueling	149 tons
Weight of vessel	283 tons
Weight of vessel and contents, normal operation	477 tons
Weight of vessel, water and contents, during refueling	693 tons

through four 22-in. inlet nozzles and flows up through the fuel bundles of the core picking up heat to form a steam-water mixture. The resulting steam-water mixture is directed by a turning vane out of the vessel through twelve 16-in. outlet nozzles located near the top of the reactor vessel. These can be seen in Fig. 9-8. Access into the vessel for refueling, changing control rods, and servicing of instrumentation is by removal of the top head assembly. The top head closure is large enough (11 ft, 7 $\frac{1}{2}$ in. inside diameter) to permit replacing all parts of the core assembly if this should become necessary. The head is held in place with 56 five-inch diameter bolts. These are heated during closing and opening to provide a tight closure without the development of undesirable stresses. The head is sealed by means of two O-ring type gaskets. Penetrations through the reactor vessel walls, bottom, and closure head are provided for entry of

control rods, flux monitoring devices, and liquid poison injection. The vent nozzle provides the means for venting non-condensable gases to the steam drum. During the process of shutting down, this vent is used to discharge steam, to assure uniform cooling of the top closure.

Nuclear core. The basic objective of the core design is to achieve low fuel cost with a high degree of reliability, safety, and plant availability. At the same time the core must withstand the mechanical, hydraulic, and thermal conditions resulting from both steady state, transient, and emergency conditions.

The core occupies an approximate cylindrical space 10 ft high by 10 ft, 9 in. diameter. It contains 488 fuel assemblies. Each fuel assembly consists of a 4.29-in. square zirconium channel surrounding 36 fuel rods, 0.563 in. in diameter. These fuel rods, which encase the slightly enriched uranium dioxide, are made of zirconium. The over-all water-to-fuel volume ratio in the core is approximately 2.2. Nuclear core data are shown in Table 9-3.

The core assembly is supported on a bottom support plate which rests on a ring grid girder, which in turn is supported through columns attached to the reactor vessel bottom. A cylindrical core guide support, which also acts as a thermal shield, rests on the ring girder at the bottom and supports an upper core guide assembly. Fuel assemblies rest on the core bottom support plate and are held in transverse alignment at the top by the upper core guide. The core assembly unit, consisting of the cylindrical core guide support, core bottom support plate, core upper guide, and fuel assemblies, is supported through columns to the bottom of the vessel as described above.

Each fuel bundle consists of 36 cylindrical fuel rods in a square array, arranged in 6 rows of 6 fuel rods each as shown in Fig. 9-11. The individual rods are mechanically tied together into an integral unit by means

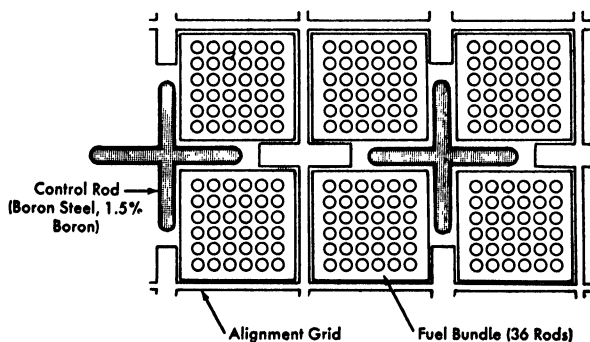


FIG. 9-11. 36-rod fuel assembly.

of various spacer and adapter members. Each fuel bundle is encased in a replaceable square Zircaloy channel which constitutes a fuel assembly. All fuel rods terminate in a common support or transition piece located at the lower end of the assembly. This transition piece forms a holder for replaceable orifices and nose piece units. The resulting bundle fits into and is attached to the channel unit at its upper end by means of cap screws.

Fuel rods are supported at the mid-point and expand in both directions against Inconel springs at the nose piece and top member. The channel holds the fuel assembly in alignment by nodes located at each spacer location. Lifting bails are provided at the upper end of each fuel bundle assembly for handling. The bottom of the fuel bundle terminates in the transition section which fits into a mating hole in the core support plate. A combination orifice plate and nose piece screws into the lower transition section of the channel assembly.

Fuel elements. The basic fuel is 1.5 per cent enriched uranium dioxide fabricated as a sintered, solid, cylindrical pellet, 0.500 in. long and 0.493 in. in diameter. A series of these pellets is encased in a tubular Zircaloy-2 jacket, with 0.030-in. wall thickness, to form a 28-in. long rod. The out-

TABLE 9-3
NUCLEAR CORE DATA

Core diameter (circumscribed circle)	129 in.
Reflector water thickness	4.5 in.
Thermal shield	2 in.
Water space between thermal shield and vessel wall	2 in.
Length of fuel rod assembly	117 in.
Length of fuel zone	112 in.
Active fuel length (cold)	103.6 in.
Active fuel length (at 1000 F average fuel temp.)	104.1 in.
Number of fuel channels	488
Number of fuel rods per channel (6×6 square array)	36
Number of fuel rod positions occupied by instruments	17
Total number of fuel rods	17,551
Channel wall thickness of Zircaloy-2	0.060 in.
Inside dimension of channel (square)	4.29 in.
Centerline distance between channels, inside control rods	4.802 in.
Centerline distance between channels, across control rods	5.160 in.
Center-to-center distance between fuel rods	0.710 in.
Outside diameter of fuel rod	0.563 in.

side diameter of the clad rod is 0.563 in. Four such fuel rod segments are joined by Zircaloy-2 threaded connectors to make up the basic 9 ft, 3¾ in. rod. The adjoining shoulders formed by abutting rod segments provide seating surfaces for spacer plates which hold each rod in proper alignment and maintain the required rod array. Fuel element data are given in Table 9-4.

The fuel cladding can withstand without significant permanent distortion an internal pressure approximately 2300 psi in excess of the external one. It is estimated from calculations together with limited experimental evidence, that the pressure exerted on the cladding by the gaseous fission products at operating or shutdown conditions after 10,000 megawatt-days of irradiation per ton of uranium will be several times below that value. Fuel jacket-to-pellet clearances allow for independence of movement of jacket and pellet under operating conditions to prevent pellet binding.

Control rods and their drives. The control rods, 80 in number, are of stainless steel containing 2.0 per cent boron as the nuclear poison material, and have a 6.5-in. cross-shaped cross-section, as shown in Fig. 9-11. They travel between fuel channels.

These fuel rods are mounted on the bottom of the reactor so that their withdrawal position is below the core. Their upward movement decreases reactivity; their downward movement increases it. Pertinent descriptive data are given in Table 9-5.

The 80 control rods are spaced 9.96 in. apart within an 8-ft, 3-in. diameter cylindrical space in the central region of the core. The rods have an 8-ft, 6-in. active length, an 8-ft, 6-in. total poison length, and an 8-ft, 6-in. travel.

The total nuclear-reactivity worth of the control rods is approximately 18 per cent k . They can be withdrawn only one at a time at a rate of 6-in. per sec, corresponding to a reactivity change rate of not more than 0.013 per cent k per sec. Scram (rapid insertion of all rods) is accomplished in 2.5 sec.

The drive mechanism for both normal operation and scram is an all hydraulic system. Both (a) slow, controlled rod movement and positioning and (b) reliable rapid scram action are provided by hydraulic means. Positioning is assisted by a locking device with 12 discrete positions for each rod. Scram action is accomplished by an independent hydraulic energy source of assured availability.

The positioning device consists of an hydraulically driven piston with a mechanical, ratchet type locking mechanism, which holds the piston rod in any of 12 equal spaced positions in its stroke. A series of magnetically operated switches are located inside an inner cylinder along the drive stroke. These switches are actuated by a magnet built into the piston

TABLE 9-4
FUEL ELEMENT DATA

Compound— UO_2 ; Processing—powdered, compressed and sintered into pellets	
Per cent of theoretical density	94%
U^{235} enrichment	1.5%
Water-to- UO_2 volume ratio	
Over-all in core	2.2
Within channel	1.44
Pellet diameter	0.493 in.
Pellet length	0.500 in.
Number of pellets per fuel segment	52
Number of fuel segments per fuel rod	4
Fuel segment length	28 in.
Fuel rod length, including connectors	117 in.
Zircaloy-2 fuel element connectors—3 intermediate and 2 end connectors	
Fuel cladding thickness of Zircaloy-2	0.030 in.
Volume of UO_2 per fuel rod (cold)	19.75 in ³
Weight of UO_2 per fuel rod	7.45 lbs
Weight of uranium per fuel rod	6.55 lbs
Total weight of uranium in reactor	57.5 tons
Total uranium throughput per year (70% load factor)	31,980 lbs
Weight of zirconium per fuel rod	1.66 lbs
Total weight of zirconium in channels	14,350 lbs
Total weight of zirconium in reactor	43,500 lbs
Zirconium throughput per year (25% channel discard)	9050 lbs

to indicate the position of each control rod. Scram operation is obtained by applying an excess of pressure on the bottom of the piston rod while simultaneously venting the top of the piston rod to the scram dump tank.

The poison section of the control rod is attached to the piston rod by means of a rod coupling device. This coupling engages when the two halves are brought together and the control rod is rotated 90 degrees. Once engaged the coupling is able to transmit any tension or compression load exerted on the rod by the drive mechanism.

Hydraulic scram action and normal controlled movement of the control rod are accomplished by the same piston and cylinder system. Variable positioning of the rod is accomplished by directing water at 200 psi over reactor pressure to either the top or bottom of the drive piston. The opposite side of the drive piston is simultaneously vented to a header held at 30 psi over reactor pressure. Rod movement is therefore ob-

TABLE 9-5
DESCRIPTIVE DATA ON CONTROL RODS

Number of control rods	80
Center-to-center distance between control rods (square array)	9 962 in.
Operating travel	8 ft 6 in.
Shape—cross blade	
Blade width	6 5 in.
Blade thickness	0 375 in.
Active length	8 ft 6 in.
Over-all length	9 ft 7 $\frac{3}{8}$ in.
Poison composition—boron steel	
Chromium	18–20%
Nickel	10–12%
Carbon, maximum	0 02%
Manganese, maximum	2 00%
Silicon, maximum	1 00%
Boron	2 0 – 0 1%
Velocity, normal insertion or withdrawal	6 in./sec
Velocity, scram	8 5 ft/sec
Scram insertion time	1 5 sec
Normal drive—hydraulic	
Scram drive—hydraulic	
Scram energy source—accumulators, backed by vessel pressure	
Accumulator pressure	1400 psig
Control rod weight (in air)	158 lbs/rod
Weight of moving parts during scram	202 lbs/rod
Total weight of all rods and drives	22 tons
Reactivity to control	18%
Cold-to-hot	0 1% <i>k</i>
Steam voids	2 3% <i>k</i>
Xenon & Samarium	4 0% <i>k</i>
Doppler effect	1 0% <i>k</i>
Burnup	5 9% <i>k</i>
Total operating control	13 1% <i>k</i>
Shutdown margin	4 5% <i>k</i>
Total required control	17 6% <i>k</i>
Average strength per rod	0 225% <i>k</i>
Maximum strength per rod	0 259% <i>k</i>
Maximum reactivity addition rate from withdrawal of control rods	0 013% <i>k</i> /sec
Total worth of control rods	18% <i>k</i>

tained by the 170 psi difference in pressure on top and bottom of the drive piston.

Reactor feedwater extracted from the reactor feedwater pump discharge header is the basic hydraulic fluid for the control rod drives. Normal operation of the rod drive uses a water pressure higher than that of the reactor, insuring that all leakage past the piston and piston rod seals is into the reactor vessel. The hydraulic fluid is fed through a pressure regulating valve set for 1400 psi, then through pressure control valves which maintain system pressure at 30 psi or 200 psi above reactor pressure. On a scram signal two solenoid controlled air pilot operated scram valves open to provide accumulator pressure to the bottom of the piston rod and to discharge the water above the piston rod to a scram dump tank.

An accumulator system is provided to supply scram pressure. In each accumulator line supplying the drive mechanisms, a 2-way valve is used to close off the accumulator pressure until a scram signal is produced. One accumulator, with a capacity of about 5.4 gallons, is used for every three control rods. The scram actuating valves are piped to control the scram of three control rods each; thus not more than three rods will fail to scram upon malfunction of any one part of the scram system.

During normal operation some rods will be acting as safety rods in a fully withdrawn position. Other rods will be used as shim rods, in that the rods will be partially inserted and held in a given position for long periods of time. In addition to being available for either of these functions, a rod can also be used as a regulating rod at the discretion of the operator. The electrical control circuit is such that only one rod can be selected and operated at a time.

Back-up safety system. A manually operated system for introducing a nuclear poison bearing solution into the reactor core provides (a) added negative reactivity margin to insure against all longer-term reactivity uncertainties, and (b) an added means of shutting the reactor down—and keeping it shut down, in the extremely unlikely event that a large fraction of the control rods should fail at the same time.

Boron, in the chemical form of sodium pentaborate (having the chemical formula $\text{Na}_2\text{B}_{10}\text{O}_{16}\cdot 10\text{H}_2\text{O}$), will be employed as the nuclear poison. The boron in the liquid poison system (approximately 400 pounds) is enough to give a boron concentration of about 690 parts per million in the cooling water when uniformly dispersed in the 188 tons of water present in the entire system in the cold condition. This boron concentration decreases nuclear reactivity by approximately 20 per cent k . (The poison worth chosen as the design requirement is 18 per cent k

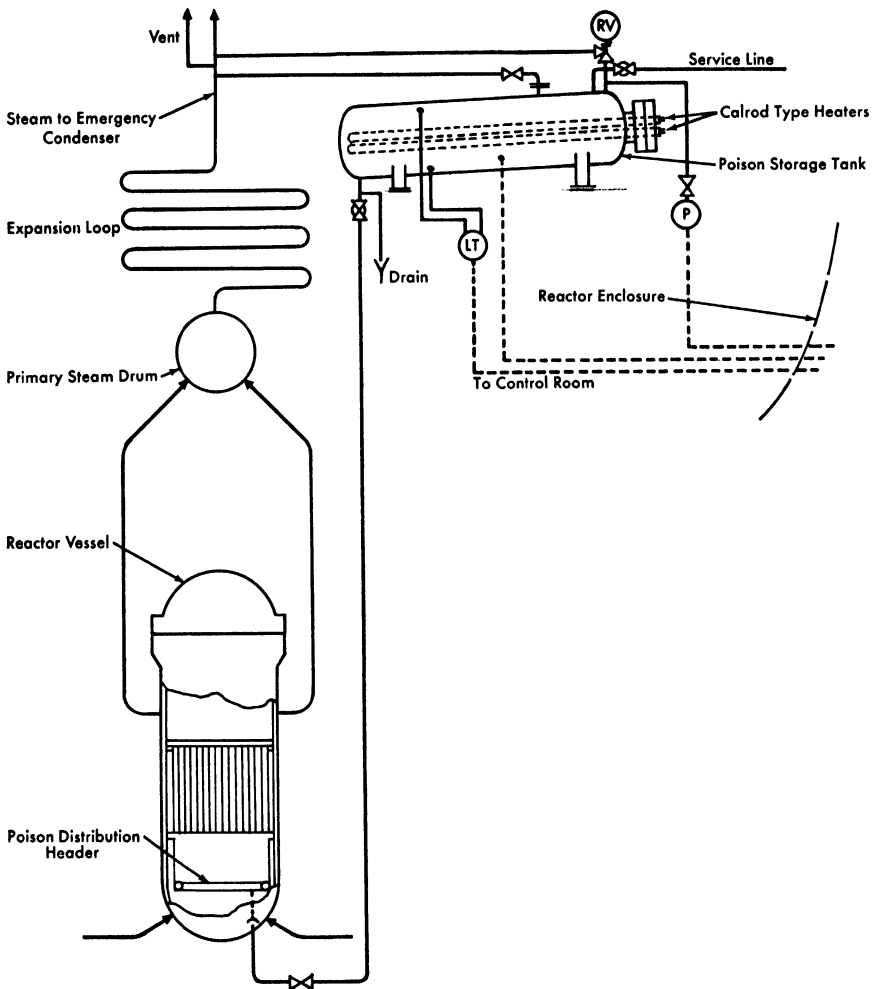


FIG. 9-12. Liquid poison system.

minimum.) The poison injection response time is 15 seconds from the time of actuation.

The liquid poison system consists of a reservoir which contains the poison solution, a pressure-equalizing line which connects the top of the reservoir with the top of the steam drum, and a feed line which runs from the bottom of the reserve to the sparger system located below the core in the reactor vessel. In order to sufficiently poison all the water entering the reactor core, a poison material will be injected at a constant rate compatible with the reactor coolant flow rate. At the maximum reactor coolant flow rate of 8000 lb per sec, the boron flow rate will be

5.3 lb per sec. The poison is injected by density differential-induced gravity feed with reactor pressure being equalized by a line to the top of the steam drum. The system is capable of inserting poison either during operation or during shutdown, including the shutdown condition with the reactor vessel head removed.

Heat transfer and fluid flow. A compilation of significant heat-transfer and fluid flow data on the reactor is included in Table 9-6. The design parameters of the Dresden core can be described through a logical consideration of the heat transfer and fluid flow limitations which were initially established. These limitations concern:

1. The maximum fuel temperature,
2. The void or steam content of the core,
3. The heat transfer conditions resulting in burnout.

To ensure against melting of the uranium dioxide fuel during a power excursion its maximum temperature is limited to less than 5000 F. Within this limitation, the maximum heat flux at the fuel rod surface is 346,000 Btu per sq ft per hr. The corresponding maximum fuel temperature is about 4500 F based on a thermal conductivity for UO_2 of 1.15 Btu/(ft²) (hr).

The over-all design hot spot or flux peaking factor which expresses the effect of power distribution in numerical form is 3.66. This factor allows for a steady-state power excursion of 25 per cent over rated, although the contemplated reactor over-power scram signal setting is 120 per cent of rated power. In addition allowance is made for (1) the axial power distribution, (2) the radial power distribution, (3) fuel burnup, and (4) local flux peaking due to water gaps between channels and between fuel rod segments.

All peaking factors are considered conservatively to occur in coincidence. Because the neutron flux will be subject to controlled variations during the life of the reactor the hot spot factor is history dependent. The magnitude chosen is such that the actual values which will exist during operation will never exceed the design value.

Since the maximum heat flux at 25 per cent overpower is 3.66 times the average heat flux at rated power, the average heat flux is calculated to be approximately 95,000 Btu/(ft²) (hr). The average power density based on the total core volume is 29 kw/liter. This corresponds to 4 kw per foot of fuel rod length.

The void or steam content in the core is established as the result of a compromise between (1) physics limitations, (2) heat-transfer limitations, and (3) economics. If the void content is increased, enrichment, control requirements, and the void coefficient of reactivity increase. On

TABLE 9-6
REACTOR HEAT TRANSFER AND FLUID FLOW

Total rated heat generation	626 Mw
Average heat flux at rated power	107,500 Btu/(hr)(ft ²)
Maximum heat flux at 125% rated power	390,000 Btu/(hr)(ft ²)
Heat transfer area per fuel rod	1 286 ft ²
Total heat transfer area	22,570 ft ²
Maximum fuel center temperature, at 125% rated power	4570 F
Peaking factors	
Axial flux distribution	1.57
Radial flux distribution	1.43
Axial flux skewing	1 15
Local flux peaking	1 15
Fuel burnup	1 11
Overpower trip	1 25
Maximum peaking factor, at 125% rated power	3 625
Maximum specific power, at 125% rated power, and excluding local peaking	15 kw/ft
Power per average fuel rod	35 7 kw
Power per central fuel rod	57 kw
Power density per cc of coolant volume	49 watts/cc
Power density per cc of core volume	27 watts/cc
<i>Fluid Flow</i>	
Total flow rate through core	2 56 × 10 ⁷ lb/hr
Primary steam flow	1 41 × 10 ⁶ lb/hr
Secondary steam flow	1 18 × 10 ⁶ lb/hr
Average steam quality (by weight)	
Core exit	0 0517
Primary steam drum entrance	0 0549
Reactor operating temperature	546 F
Feedwater temperature	405 F
Core inlet temperature	505 F
Core inlet subcooling (based on 1000 psig saturation pressure)	51 4 Btu/lb
Core inlet enthalpy	493.5 Btu/lb
Average channel	
Inlet velocity	4 50 ft/sec
Saturation velocity	4 74 ft/sec
Central channel	
Inlet velocity	6.44 ft/sec
Saturation velocity	6.78 ft/sec
Exit steam volume fraction	0.456
Exit water velocity	12.46 ft/sec
Slip velocity (relative velocity of steam and water)	3 ft/sec
Flow area per channel	0 0662 ft ²
Water area outside channel	0.0373 ft ²
Hydraulic diameter for channel and fuel rods	0 4739 in.
Core pressure drop	7 8 psi

the other hand, flow rate, the ratio of secondary to primary steam flow, heat exchanger requirements, cycle heat rate, and the allowable heat flux decrease. Studies balancing these effects resulted in the selection of a primary to secondary steam ratio of 1.2/1 and a total coolant flow rate through the core of 25.6 million pounds per hour.

Corresponding average coolant conditions in the core are these: The water enters at 505 F, equivalent to a subcooling with respect to saturation at 1000 psia of 51 Btu per lb. The inlet velocity is 4.50 ft per sec.

The velocity increases to 4.75 ft per sec as the water expands to saturation conditions about 5 ft above the core inlet. In the top 4 ft of the core, steam is generated resulting in an exit quality of about 5 per cent by weight and an exit void of about 50 per cent. The average exit water velocity is about 9 ft per second.

Due to its lower density and viscosity, the steam slips upwards relative to the water. The estimated average difference between the steam and water velocities is about 3 ft per sec.

Orifices at the base of the core distribute the coolant flow among the channels approximately in proportion to the heat transfer load. The effect of this is to maintain uniform exit quality throughout the core.

The pressure drop across the core is about 8 psi. During an isostatic power increase to 125 per cent rated, the core flow increases slightly. This unusual condition results from the fact that with increasing quality the additional void in the risers more than compensates for the increase in core pressure drop. Furthermore, since a large fraction of the total system pressure drop is due to single phase flow, the hydraulic flow system is extremely stable.

In every boiling system, there is some heat transfer rate which results in steam blanketing of the heat transfer surface. Because steam is a poor conductor of heat relative to water, the surface temperature increases accordingly. This condition usually results in melting of the heater surface and is called burnout. The heat flux which results in burnout appears to be primarily a function of the local quality, decreasing with increasing quality. At the point in the Dresden core where the worst combination of heat flux and quality occur, the heat flux which would cause burnout is 2.6 times the actual heat flux. This factor represents a large safety factor; hence from the heat transfer point of view, the Dresden core is said to be center temperature limited.

One or two years ago, there was little experimental information upon which to formulate design criteria which meet the center fuel temperature, void and burnout limitations. Today, the position is much improved. Many major areas of concern relative to the adequacy of design with respect to heat transfer and fluid flow have been removed.

TABLE 9-7
SUCCESSFUL IN-PILE FUEL TESTS

Four diametric gap inch	Mwd/t of U	Maximum Btu/(hr)(ft ²)
0 007	187	290,000
0 010	131	340,000
0.003	115	355,000
0 003-0 016	9	370,000
DNPS average		95,000
DNPS max. rated		277,000
NNPS max. over power		346,000

Prototype fuel assemblies containing 9 rods loaded with UO_2 pellets of the same nominal dimensions as will be used in Dresden have been tested in the Vallecitos Boiling Water Reactor.

Table 9-7 summarizes the in-pile fuel tests which have been analyzed to date. The flux conditions listed in this figure correspond to the maximum flux in the fuel assemblies. Because of the difficulty of determining accurately the actual flux conditions in the test fuel, the results must be regarded as preliminary. They are significant, however, since they demonstrate that no damage to the fuel element results from heat fluxes as high as 370,000 Btu per sq ft per hr. Since the maximum heat flux in Dresden is 346,000 Btu per sq ft per hr, even at 25 per cent over-power, these tests confirm the initial design criteria that no melting will result under the worst operating conditions.

The assembly which contained rods with gap clearances between the pellet and the clad varying from 3 to 16 mils showed evidences of melting. But an accurate determination of the heat flux corresponding to incipient melting has not yet been made.

It was mentioned that flow conditions were selected to yield a specific void content and reactivity associated with this void. The void fraction at any point in the core is a function of the local weight fraction of steam as well as the relative velocity or slip between the steam and water. At the University of Minnesota, Argonne National Laboratory, and at the General Electric test facility in San Jose, intensive efforts have been made to understand the two phase flow phenomenon—and in particular, to measure void fraction as a function of weight fraction of steam. The density of the steam water mixture is measured by its ability to attenuate gamma rays. Figure 9-13 shows the design curve that has been used to determine void fraction and on the abscissa, weight fraction or steam

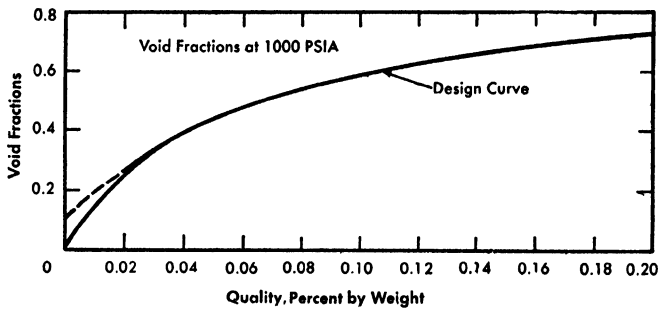


FIG. 9-13. Two-phase flow correlation.

quality. The test was conducted with a horizontal pipe operated at 1000 psia with no heat transfer. The void at any quality would tend to be less in a vertical pipe than in a horizontal pipe, hence these data when applied to Dresden's conditions of vertical flow are conservative. Although this simple correlation tends to oversimplify a very complex phenomenon, the corroboration of the design assumption is encouraging. In order to investigate those conditions which result in vapor binding and burnout, an intensive test program has been conducted with the heat transfer facility at San Jose. The immediate purpose of these tests was to confirm the design curve used for the Dresden Reactor. Figure 9-14 shows the data along with the design curve. On the ordinate is plotted the heat flux at burnout and on the abscissa, the local quality. The heater element used was a vertical thin-walled $\frac{3}{8}$ in. stainless steel tube 9 ft in length. The heater tube was cooled on the outside by forced circulation of water. The flow velocity, boiling length, hydraulic diameter, and pressure were selected to simulate Dresden conditions. The

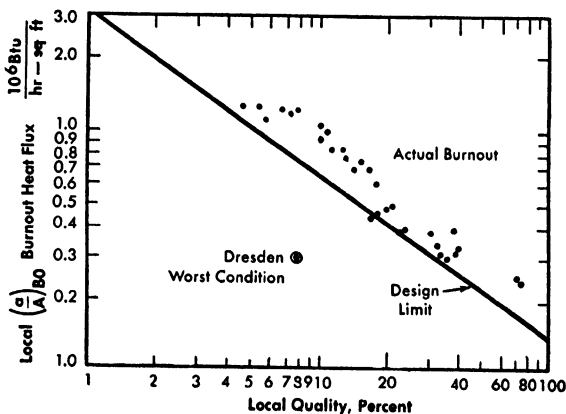


FIG. 9-14. Uniform rod burnout data.

limits on each data point indicate with 95 per cent confidence the possible error in calculated quality. One point of actual burnout resulting in destruction of the heater tube is noted.

The other points represent indicated burnout and hence are slightly conservative. These data clearly confirm the design limit which was devised in the early stages of the Dresden design. For reference, the worst combination of heat flux and quality in the Dresden core is shown. This calculated point results from a detailed consideration of the flux and flow conditions in a hot channel.

Since these data were obtained under stable flow conditions, it follows that the hydraulic loop is highly stable at a power density more than double that of the Dresden Reactor. Because of the hydraulic similarity of the test loop to the Dresden flow loop, this conclusion may well be applied to the Dresden Reactor.

FUEL HANDLING

Operating personnel will load and unload the reactor core using pole-type grapples assisted by crane lifting. Operations are accomplished through approximately 40 ft of water, with visual guidance.

The reactor is prepared for refueling by removing the head of the reactor vessel and other components above the core after filling the fuel-handling canal with water. The fuel is drawn from the reactor with long handle grapples into a water basin above it and is then removed horizontally through a canal to a point near the side of the reactor enclosure, down through a water lock to a level below grade, and horizontally into the storage area outside the enclosure. The fuel is kept under water at all times after it is irradiated and is transported to and from the storage area in groups of fuel assemblies. New fuel is brought from the storage area to the reactor by the reverse route.

Because of the possibility of fuel elements becoming ruptured during reactor operation, provisions are made for removing them from the reactor enclosure in a sealed container if necessary. The container is placed over the suspected element as soon as possible after the opening of the vessel. The reactor hoist carrier and the cover over the fuel handling canal are provided with a ventilation exhaust system to keep fission gas release to a minimum.

Except for the actual refueling operation, all other fuel procedures take place in the fuel storage area outside the reactor enclosure.

To accommodate complete removal of fuel from the reactor, storage capacity for irradiated fuel is provided for 120 per cent of the reactor loading. On-site unirradiated fuel storage capacity for one reactor loading is also provided.

THE POWER GENERATION SYSTEM

The principal features of the steam generating system have already been described and will be covered only briefly in this section. The steam-water mixture leaving the reactor flows to the external separating drum mounted high above the reactor vessel. The water from the separating drum is pumped through the secondary steam generators as already described. There are four separate recirculation loops, each consisting of a secondary steam generator, recirculating pump and secondary valving. As shown in the isometric view, Fig. 9-8, the downcomers from the primary drum feed two lower headers which in turn supply the pumps and the secondary steam generators.

The primary steam drum is approximately 8 ft in diameter and 60 ft long, and holds approximately 1500 cu ft of water storage volume. The high drum arrangement produces good natural circulation characteristics and increases the available net positive head on the pumps.

Figure 9-9 shows a section of the vertical type secondary steam generators which produce the 500-psig steam. The primary fluid is pumped through the inverted U-tubes. The secondary fluid and the steam separation equipment are contained in the vertical cylindrical shell.

An emergency condenser is required to remove the reactor decay heat in the event of loss of the main condenser. Steam from the reactor is condensed by water stored in an emergency storage tank, and this water is evaporated to atmosphere. Sufficient storage volume (30,000 gal) is provided in the cooling water tank to take care of the decay heat for approximately 8 hours after shutdown. To remove the reactor decay heat during fuel change or reactor maintenance (i.e., with the reactor water not boiling) a separate unloading heat exchanger is provided. In this system primary water is pumped through a heat exchanger transferring heat to river water. The heat exchanger, consisting of two independent units in parallel, has a total capacity of 0.8 per cent of rated reactor thermal power, when the reactor is at 125 F at the inlet and 115 F at the outlet.

As indicated in the simplified flow diagram and the heat balance diagram, Figs. 9-3 and 9-4 respectively, the saturated primary and secondary steam enter the dual-admission turbine from the mains which connect the reactor plant to the turbine. The exhaust from the turbine is condensed. The condensate then divides into two flow paths: one through the secondary feed pumps and the secondary-feedwater heaters to the secondary steam generators; and the other through condensate demineralizers, the primary feed pumps, and the primary feedwater heaters to the steam drum.

Five separate feedwater heaters are used on the primary and secondary feeds, giving a total of ten vertical heaters. Integral drain coolers are

used on the three top (highest pressure) heaters of the primary circuit, and external drain coolers are used with the two lower heaters.

The turbogenerator. The basic design objective in designing the steam turbine for Dresden Station was to produce a turbine which, despite the unusual conditions, approaches as nearly as possible the reliability, and freedom from maintenance, expected of turbines built for currently conventional steam conditions.

Since available published information indicated that the radioactivity would not have a significant effect on the turbine materials, erosion and corrosion were considered to be the primary problems with the effect of the presence of an inordinate amount of oxygen being the major unknown.

The tandem-compound turbine consists of high-pressure, intermediate-pressure, and double-flow low-pressure sections exhausting into a common condenser opening. The three sections are mounted on one shaft connected to one 1800 rpm, 245,500 kva 85 per cent p.f. generator. The turbine is designed to produce 192,000 kw at 2.5 in. Hg absolute exhaust pressure while heating the condensate to 405 F in five stages of feed water heating. Under these conditions, the primary admission is at 950 psig, 540 F, and the secondary admission at 460 psig at 462 F. As load and secondary flow are simultaneously reduced, the secondary admission pressure rises to approximately reactor pressure at zero secondary flow.

The 15-stage, opposed flow, high-pressure turbine section exhausts through two crossunder pipes at about 180 psig to the 7-stage intermediate section. The intermediate-pressure section exhausts at about 11.5 psia through a large rectangular crossover to the three-stage double-flow section with 38-in. last stage buckets.

The stage diameters of the 1800 rpm, high pressure section are about equal to those normally used at 3600 rpm, resulting in steam velocities considerably lower than normally encountered at this pressure level.

All end packings of the turbine are steam sealed and are automatically controlled by a steam-seal regulator. These packings are designed to seal the shafts while the unit is on turning gear so that the turbine may be filled, if required, with water and decontaminating fluids prior to maintenance operations. Throughout the turbine, careful attention has been given to the elimination of pockets and crevices in which radioactive material might lodge. Unavoidable pockets have been made self-draining.

The control of moisture in the Dresden turbine was one of the most important problems. The dash state line in Fig. 9-15 shows the equilibrium state of the steam if it were allowed to expand normally through the turbine with no moisture extraction. Under these conditions, the amount of moisture in the low-pressure region of the steam expansion

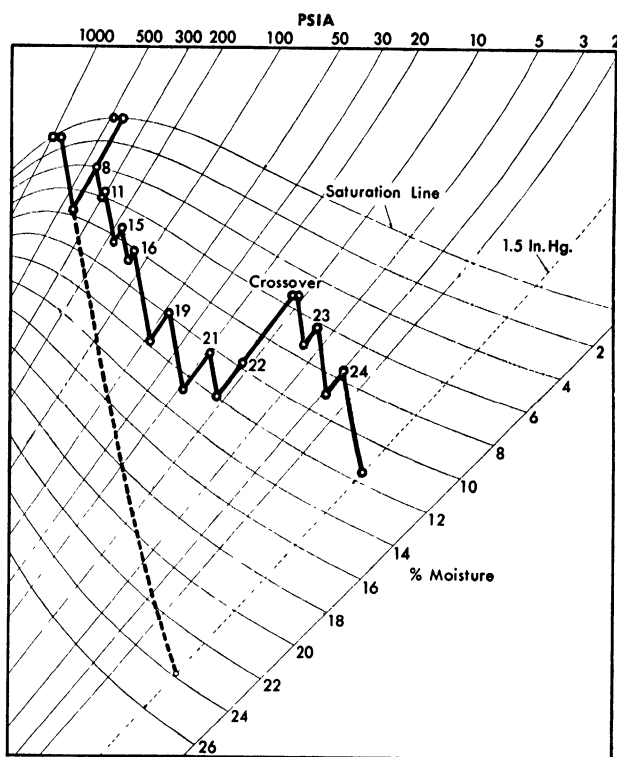


FIG. 9-15. Expansion line indicating steam state in various turbine stages.

would obviously be much greater than that encountered in past experience.

Moisture-extracting buckets and a crossover moisture separator have been employed to control the moisture to reasonable values. The solid expansion line, shown in Fig. 9-15, represents the state of the steam expected as it expands through the turbine, based on a conservative estimate of moisture removal performance. In Fig. 9-15, the numbers adjacent to the various points on the curve identify the stage in the turbine from which moisture extraction takes place. Moisture extracting buckets were first applied to commercial-non-reheat turbines by the General Electric Co. in 1952. Both laboratory and field tests have demonstrated these to be considerably more effective in removing moisture from the steam path than conventional buckets.

The electric generator is a conventional hydrogen-cooled machine operating with a hydrogen pressure of 30 psig. It is a 14,400-v machine.

The condenser is capable of handling the normal steam flow from the turbine plus the heater drains or the bypassed extraction steam. In order to serve as a heat sink for the reactor, it is capable of handling 1,900,000

lb per hr of bypassed primary turbine steam. To prevent leakage of circulating water into the condenser, double tube sheets are provided. The space between these sheets will be filled with demineralized makeup water from which tube leakage can be detected.

It is expected that some oxygen and hydrogen will be produced by radiation-induced dissociation of water in the reactor. Therefore, the condenser and its air-removal apparatus are sized to handle abnormally high quantities of noncondensable gases.

The 40,000 gal condenser hotwell has a compartmented condensate collection arrangement. The compartments are baffled to give a 3 minute holdup time for decay of short-lived radioactivity. Deaeration in the hotwell will limit the condensate oxygen content to 0.001 per cent by volume.

There are 3 condensate pumps, 3 secondary feed pumps, and 3 primary feed or reactor pumps. Two of each set are required for normal full power operation.

AUXILIARY SYSTEMS

Reactor cleanup system. The reactor cleanup system is designed for prevent build-up of undesirable corrosion and erosion products in the reactor primary system and to control water quality in the primary steam drum. The reactor coolant cleanup system consists of two identical loops, each having a capacity of 100,000 lb per hr. Recirculating water is bypassed from the suction header of the recirculating pumps either to one (normal) or both loops, depending upon the requirements, through a booster pump, a non-regenerative heat exchanger, and a demineralizer. Resins used in these demineralizers will exhibit a build-up of radioactive corrosion materials removed from the reactor coolant. The resins will not be regenerated, but rather sluiced from the ion-exchange vessels to the waste system for burial. The regenerative heat exchanger uses the recirculating water from the demineralizer for cooling and the non-regenerative heat exchanger uses water from the cooling water system for removal of heat. The demineralized water is fed back into the primary steam drum through the feedwater lines. It is expected that the reactor recirculating water quality will be maintained at 0.5 ppm maximum total solids and a pH index of 7.2 ± 0.2 .

Condensate demineralizer system. Total solids in the reactor water result from corrosion products in the reactor circulating system, and leakage from the cooling water in the condenser contributes to the impurities present in the feedwater. Demineralizing equipment is provided to remove these trace materials in the reactor feed water.

The full feedwater flow to the reactor is 3100 gpm and this flow passes through 2 of 3 demineralizer beds before entering the reactor feed pump. The condensate entering the demineralizer contains a fraction of a part per million of solids and the demineralizers reduce this concentration to approximately 6 or 8 parts per *billion*, equivalent to a resistivity of 10 megohms per cc. The solids removed from the condensate should not normally contain appreciable quantities of radioactive materials and, therefore, provision is made to regenerate these resins.

Waste disposal system. The radioactive waste disposal system provides a means of removing and disposing (or storing) of all radioactive liquid and solid waste materials from the entire plant.

Wastes will be handled in several ways. The method used in any instance is determined by the type of waste, level of radioactivity and type of contamination. The batch method will be employed. Liquid wastes will be either discharged to the river under controlled conditions, filtered and demineralized and returned to the make-up system, or stored to permit radioactive decay. In the case of acid or detergent wastes, a special tank equipped with an evaporator will be provided to concentrate the liquid for ultimate storage.

Permanent storage is provided for the disposal of solid wastes, including resins from various demineralizer systems, corrosion products and airborne dust trapped by some of the open systems such as the fuel storage basin.

The process area ventilation exhaust will contain trace quantities of dusts, mists, and vapors of process origin. This air as well as the air ejector and gland seal exhausts will be discharged to the atmosphere by means of a 300 ft stack. The air ejector and gland seal exhausts are diluted into the ventilation stream inside the stack. Continuously operated gamma scintillation-type instrumentation will be installed on continuous sample streams of both the total stack flow and the air ejector flow to provide a continuous record and immediate control information on the acceptability of stack gas release.

CONTROL AND OPERATION

Operation and control of the reactor and all other main process equipment is to be centralized in the Control Room, located in the Turbine Building. The Control Room is shielded from all process equipment and from the reactor enclosure. Load regulation as well as plant and power loop instructions originate in this control room.

Control of the reactor power is affected by control rod positioning as well as the amount of subcooling as previously described. The control

rods are positioned from the control room. These rods can be removed or inserted individually. The position of each rod is *continuously* indicated in the control room.

Water levels in the primary and secondary steam drums are controlled by 3-element control systems, comparable to most boiler installations, wherein the feedwater entering the drum is controlled by the drum level and the ratio of steam flow to feedwater flow.

Features of dual-cycle control. The reactor will normally operate under the mechanism of *void* control which depends upon varying the effective moderator density. As the volume of steam voids within the reactor core is varied, the effective density of the moderator is altered. Reactivity decreases as the steam-void volume increases. The positive reactivity determined by the control rod setting is balanced by the negative void reactivity. The core boiling boundary and the core-exit steam quality will vary with subcooling, but the net reactivity of the steam void accumulation, regardless of its distribution, should be unchanged for a given control rod reactivity. The relationship between void reactivity and core-exit steam quality for a given subcooling has been shown in Fig. 9-16.

Control of the plant is accomplished by means of a speed governor and a pressure regulator. These two controls are tied together by a special control linkage system to operate the primary turbine admission valves, the primary steam bypass valves, and the secondary turbine admission valves. Figure 9-17 shows the interconnected mechanical arrangement of these three valves. The pressure regulator holds essentially constant pressure in the reactor vessel to prevent reactivity changes arising from pressure fluctuations. Primary steam is admitted to the turbine in a sufficient quantity to hold the net pressure. If the turbine cannot accom-

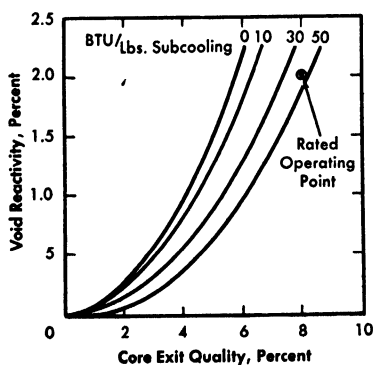


FIG. 9-16. Void reactivity map.

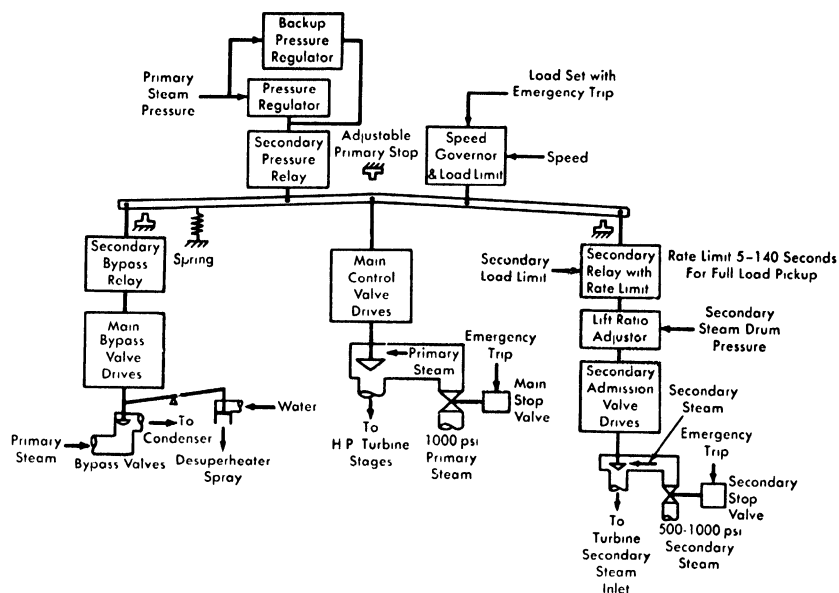


FIG. 9-17. Dual-cycle plant control system.

moderate this required steam flow, the bypass valve is opened and the excess flow of steam is dumped to the condenser. The speed governor admits secondary steam to the turbine to complete the total turbine requirements and control reactor-inlet-water subcooling to maintain reactor power at the proper level. The system is an all-hydraulic control operated from the turbine oil supply.

The control mechanism shown on Fig. 9-17 works as follows. The spring on the pressure regulator link normally holds the bypass valves closed so that the pressure regulator operates the primary admission valves. The speed governor link causes the secondary steam valves to move opposite to the primary valve. The lift ratio adjuster, or pressure compensator, adjusts the linkage gain with secondary drum pressure so that the readjustment of the secondary valve, caused by movement of the primary valve, will not change the load carried. If the secondary valve reaches its closed limit, the bypass valve may be driven open by either the pressure regulator or the speed governor, depending upon load and pressure conditions.

The lift ratio adjuster also serves to limit maximum available secondary steam flow to a fixed amount independent of secondary drum pressure. Secondary steam supplied to the turbine will not significantly exceed rated value regardless of the secondary drum pressure. This prevents excessive secondary steam flow on initial load demand under high rates

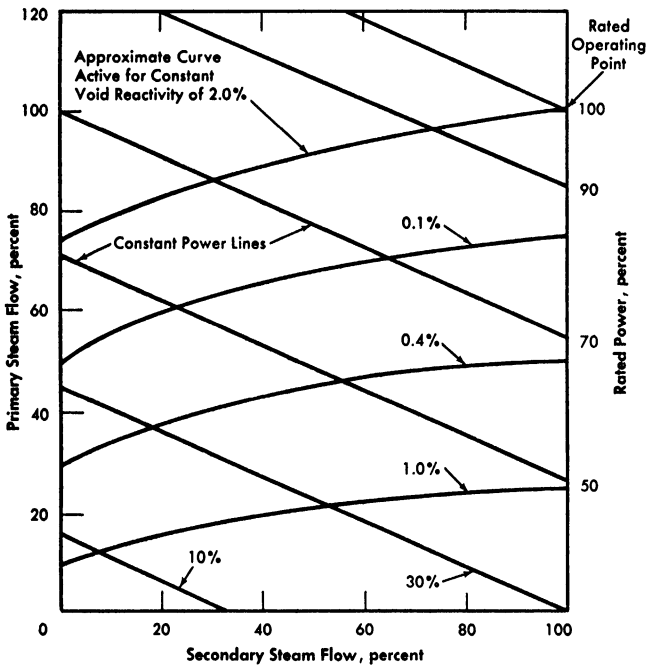


FIG. 9-18. Approximate primary-secondary steam flow relationship for the dual-cycle reactor.

of load pick-up starting from low secondary load (high drum pressure). A back-up pressure regulator is provided to override the main pressure regulator if the latter should malfunction and allow system pressure to rise.

In order to protect the condenser when steam is dumped into the condenser directly from the bypass valves, a desuperheating water spray is initiated by the opening of the bypass valves.

In addition to this linkage tie-in between primary and secondary valve openings another connection between primary and secondary steam flow is a result of the mechanism of void control. As the load on the machine is increased, more secondary steam is admitted to the turbine, subcooling received at the core inlet from the primary side of the secondary heat exchanger is increased, and boiling occurs at a higher level in the core. At first the void reactivity is diminished because the void fraction is decreased; so core power is increased, and then primary steam and the void fraction increase. Void reactivity continues to build up until the same reactivity is tied up in the steam voids as was initially present before the load change (assuming the control rods are not moved). By this process, load control in the range of about 70 per cent to 100

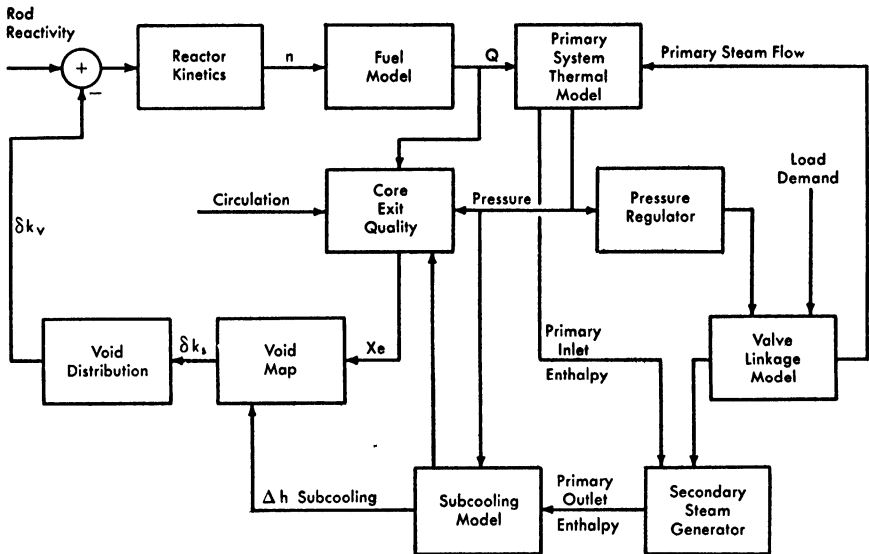


FIG. 9-19. Simplified block diagram of the dual-cycle boiling reactor plant.

per cent of full load is obtained automatically by changes in secondary steam flow and without control rod repositioning.

A control map showing the approximate relationships between primary and secondary steam flow for constant control rod reactivity (constant void reactivity in the steady state) is shown in Fig. 9-18. This control map, derived from the exit quality-void reactivity representation of Fig. 9-16, shows the relationship between primary and secondary steam flow, expressed in per cent of rated values, for constant void reactivity in the steady state. If secondary steam flow is varied by the speed governor setting or load adjuster, the final steady-state effect on primary steam flow may be obtained from the curves of Fig. 9-20. For example, if secondary steam flow is reduced from the 100 per cent load point in a gradual manner (2 per cent per second or so), the primary steam flow will also be reduced in such a fashion as to follow the constant-void-reactivity curve which passes through the 100 per cent load point. This reduction comes about because the decreasing subcooling seen by the reactor as the secondary flow is decreased will initially increase the accumulation of steam voids in the core. This increased void accumulation will initially provide a negative excess reactivity to reduce core power, thus reducing boiling to the point where the void reactivity is the same as before the change in secondary steam flow. In this manner, primary steam flow will continually readjust itself, holding void reactivity

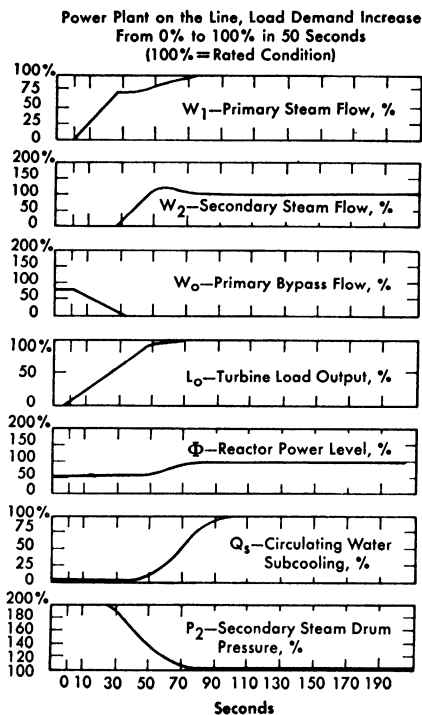


FIG. 9-20. Load pickup transients (approximate).

constant as the secondary steam flow is varied slowly. When zero per cent secondary steam is reached, the primary steam flow will settle out at a lower per cent of primary rated or about 75 per cent of rated primary flow. The resulting interrelation of primary control caused by secondary steam flow adjustments is indicated by the lines of constant void reactivity. If the control rods are repositioned, the net steady-state void reactivity content will change to a new value, and control will take place over another constant void reactivity line. Thus, to operate continuously at low outputs without bypassing large amounts of primary steam, the control rods are readjusted manually to provide control as represented on a lower void-reactivity curve.

If the secondary steam flow is changed very rapidly, the transient interrelation of secondary and primary steam flows will not follow the constant-void-reactivity line initially, but will return to the proper control line in the steady state. For example, if, with the plant operating at rated load, the secondary steam flow were suddenly shut off, the primary steam flow would initially continue at 100 per cent; but, after

a lapse of several seconds, when subcooling changes are felt at the reactor, the primary steam flow would begin to reduce toward the lower final value. It will take about one minute to complete this change. The secondary steam generator itself contributes a time constant to the subcooling change resulting from sudden interruption of the secondary steam flow. The transport time and the reactor-vessel plenum mixing time constant also contribute to the duration of this transient.

Plant performance. The manner in which the primary reactor void-fraction loop is combined with the remainder of the dual-cycle system is shown in the block diagram of Fig. 9-19. The reactor void-fraction loop now appears as only one part of the entire dual-cycle system. Having examined the reactor void-fraction loop to confirm its stability, it remains now to confirm the stability of the entire system. The diagram of Fig. 9-19 indicates how the secondary steam flow drawn by the secondary steam generators determines the enthalpy of the primary coolant leaving the steam generator. The water, at this particular enthalpy, reaches the reactor core after a suitable transport time delay, simulating the circulation from heat exchanger to reactor inlet. The particular subcooling received at the core then determines a new primary power level because of the effects of subcooling on the reactor void-fraction loop. The new reactor power will produce a new steaming rate and, consequently, influence primary system pressure.

If the original secondary steam change is set by the speed governor through a change in demand, then as the primary pressure builds up because of the recent increase demanded of the secondary steam generator, the primary pressure regulator will open the primary admissions valve. Since a linkage exists between primary and secondary admission valves (Fig. 9-17), the secondary valve will close slightly to compensate for the primary steam flow increase. Of course, the slight closure of the secondary steam valve will, in turn, reffect the subcooling reaching the reactor and, consequently, alter somewhat the primary steaming rate.

The entire block diagram of Fig. 9-19 has been rather completely simulated on an analog computer. Results of studies, wherein various system disturbances have been inserted, show that, with the proper selection of pressure regulator parameters, the entire system behavior can be made quite stable.

The control mechanisms and selection of system parameters, as described earlier, allow expected demand changes to be met in a stable fashion without exceeding the design limits of the plant. In general, the demand changes that can be accommodated are as fast as those of a conventional fossil-fueled power plant.

A considerable number of reasonable operating transient conditions can be hypothesized, and these conditions can then be imposed on the plant simulation, with subsequent fluctuations in steam flows, pressures, turbine load, subcooling, and reactor power level being recorded continuously. One severe type of plant transient is that of rapid load pickup. Figure 9-20 represents, approximately, the transient which would arise if full load pickup were demanded over a period of 50 seconds. The machine is assumed synchronized to the system for this transient, and, prior to load pickup, primary steam (about 75 per cent of rated) is bypassed to the condenser. It can be observed from the three recorded steam flows that the secondary steam flow just barely reaches its limit before primary steam flow starts building up rapidly. Nevertheless, the load demand cannot quite be met immediately beyond 95 per cent of rated load until the primary steam has a chance to build up. As a result, the turbine load output finally reaches the 100 per cent demand rather gradually.

If the load demand rose over a longer period of time, the load output curve would rise at a constant rate to the limit, as demanded. On the other hand, if the demands were much more rapid, the load output would begin to lag behind demand at a lower level than occurred under the Fig. 9-20 conditions. During the entire transient the primary pressure regulator will act fast enough to hold the primary pressure constant within ± 10 psi from the set point.

Another operating transient that must be considered is full load trip-out. If the turbine is tripped out while carrying full load, the secondary steam is suddenly shut off, and the primary steam is bypassed to the condenser. As the changed subcooling from reduced secondary steam is felt by the reactor, the primary steam is gradually reduced until a steady value of about 75 per cent of rated primary steam flow is reached. The records of Fig. 9-21 are the system transients obtained from an analog computer study of a dual-cycle system undergoing such a condition. A full load trip-out produces a peak in reactor power because of the short initial peak in pressure occurring before the bypass valve can open and dump steam to the condenser. The pressure transient, as shown, damps out after a few cycles and returns to 1000 psi. Other transients, such as testing the by-pass valves under load conditions, suddenly changing the pressure regulator set point, failure of the main pressure regulator and subsequent operation of the back-up regulator, have also been studied on the analog computer. It has been determined that, in any of the reasonable-to-anticipate operating transients considered, the reactor, the system control mechanisms, and the safety limits perform in such a manner that smooth, stable control is obtained.

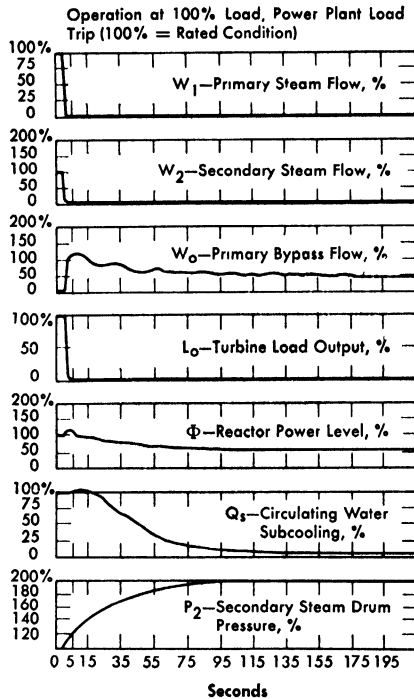


FIG. 9-21. Load loss transients (approximate).

SAFETY SYSTEM

The principal objectives in the design of the Dresden safety system were to achieve: (1) high reliability to produce a system shutdown upon approaching unsafe conditions, and (2) high reliability for continuity of plant operation. False shutdowns due to malfunctions of the safety system should be minimized without sacrificing to any degree the ability of the automatic safety system to protect from unsafe conditions. The concept of a dual safety system was evolved to meet these basic requirements. By this means, two separate power suppliers (motor-generator sets from two separate auxiliary buses) are fed through several relay contacts to pairs of safety system pilot valves. If power is lost to both members of a pair of valves, a group of three control rods will be inserted rapidly into the reactor. In this way, through many pairs of control valves, all 80 control rods would be rapidly inserted to shut down the reactor if power to these scram holding circuits or pilot valves was removed for any one of several scram conditions. In this way, one half of the dual safety system could be de-energized through perhaps equip-

ment malfunction without scramming the reactor. If, in the active half of the safety system, either one of a pair of scram function sensors indicated the approach to an unsafe condition, the other half of the system would be de-energized and a complete scram would be obtained. Thus any one safety system component can fail without scramming the system. Normally, both halves of the dual channel system would be active and the approach to an unsafe condition would de-activate both halves of the system and thus produce a complete scram. Figure 9-22 illustrates the automatic scram functions that the Dresden safety system provides. In addition, it indicates the actuation signals for bringing on the emergency heat exchanger system. Each contact is actually a relay contact where the relay in turn is de-energized in general by the opening of one scram contact. In this way, for the most part, there are then four independent sensors; two sensors for each scram function in each half of the dual channel system. The detail of the high sphere pressure function is typical. The entire safety system is thus fail-safe, with each sensor enunciated. If one out of two of these sensors operates in each half of the system, a full scram results. If only one sensor operates in just one half of the dual channel system, no scram will result.

As explained above, a complete scram will result only if both halves of the dual-channel system are de-energized. When both halves are de-energized, the following scram functions occur. All of the scram solenoid operated air pilot valves and air operated hydraulic scram valves are de-energized to produce hydraulic scram motion of the drive mechanisms. This results in the drive mechanisms rapidly inserting the control rods into the reactor core within a matter of approximately two seconds.

In addition, all sphere ventilation is closed off to eliminate the possibility of sphere leakage should a major accident have occurred since the vent openings are the largest openings in the sphere. In addition, secondary steam flow is closed off to prevent too rapid a drop in primary system pressure and temperature. In addition, after a few seconds time delay following the scram, recirculation pumps are tripped off to eliminate the possibility of pump cavitation due to the pressure decrease in the primary system. (Pressure decreases for the first one or two minutes to approximately 900 psi and then gradually recovers due to decay heat to 1000 psi. The pressure regulator intermittently passes some steam to the condenser to keep the pressure from rising above 1000 psi.)

Automatic scram functions are illustrated in Fig. 9-22. To begin with, the operator has the option of producing a scram himself should he decide plant conditions justify such a move. If high sphere pressure arises, it would occur presumably because of a rupture somewhere in

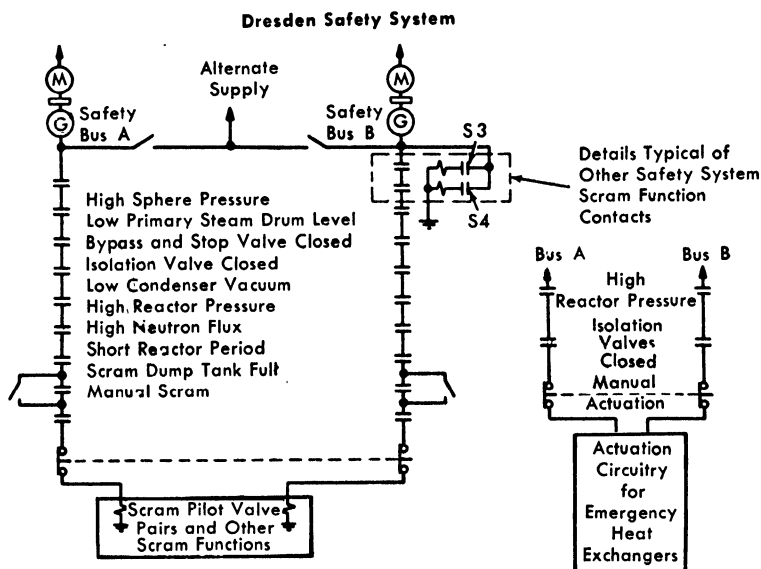


FIG. 9-22. Dresden safety system.

the primary system. Because of this, as sphere pressure exceeds atmospheric pressure by approximately four inches of mercury, a reactor scram would occur. If water level in the primary drum drops too far below normal working level, the system will be scrammed. A scram results from this condition because the water level would fall too low to maintain natural circulation in the system after the total steam voids collapse. For this reason, when a critical level is reached where this situation might possibly occur, the reactor would scram to insure a sufficient level to support this natural circulation. If the reactor steam supply system is completely bottled up, pressure will continue to rise in the reactor vessel, and because of the pressure effect on subcooling and directly on steam volume, reactor power will tend to increase. To guard against the occurrence of such an unsafe condition, if the primary isolation valves are closed a reactor scram will occur (as the operator attempts to close these valves). This normally would never be done in practice. One valve at a time may be closed for test purposes, but if both valves are moved off their full-open position by approximately thirty per cent of travel, a scram will result. Of perhaps even more concern is the possibility of very rapid closure of the bypass and stop valves. These valves can be closed in the matter of only a few seconds compared to the one to two minute closing time of the isolation valves. Such a rapid closure could occur accidentally from failure of the turbine oil supply. For such a

rapid closure, reactor pressure would immediately begin rising at a rate of approximately 18 psi per second. A scram signal should result immediately from the closure of these valves rather than waiting for the high reactor pressure scram signal to come into action at exactly 1050 psi. The high reactor pressure scram function will, however, be available to produce a scram should reactor pressure rise to this operating point of 1050 psi for any reason.

Since the main system heat sink is the condenser, if condenser vacuum is lost it is necessary to shut down the system. A low condenser vacuum scram is provided to scram the system in the event that condenser vacuum falls from the normal working level to approximately 23 inches. At this point a scram occurs. If vacuum continues to drop to 19 inches, turbine primary and secondary admission valves are closed off. If vacuum is dropped further to 7 inches, the system bypass valve is closed up as well; thus, the condenser is completely closed off at this vacuum level to avoid opening the condenser rupture diaphragm.

Of basic concern as a measure of reactor power is the neutron flux signal. If reactor power gets excessive, which could lead to a fuel melting condition or a fuel burnout condition, it would be possible to damage portions of the reactor core. For this reason, if flux exceeds approximately 120 per cent of rated flux, a scram is called for. It is interesting to note that the speed of response obtainable with this high-flux scram is sufficient to obtain protection from a startup accident of the hypothetical type where an operator withdraws control rods one right after the other, with a resulting rapid power rise. This serves as a back-up to the short reactor scram function illustrated which would normally scram the reactor under this startup accident condition well before the power range is reached. Such a classical startup accident is not visualized as an easy achievable condition but serves only as a design limit for safety system response criteria. The short reactor period function is bypassed in the power range to eliminate the possibility of nuisance scrams from boiling fluctuations in the reactor. The final scram function as indicated in Fig. 9-22 is that of a scram resulting from filling the scram dump tank. In order to limit system leakage following a scram and to provide some back-pressure to minimize flashing effects in the water hydraulic system on a scram, a 100 psi scram dump tank is used. This dump tank has an automatic level control that keeps the dump tank almost completely empty normally. However, should a few scram valves fail, the leakage rate would be such that the dump tank would begin filling fairly rapidly. Should the dump tank become completely full, no free volume could exist to accommodate a scram should the necessity of a scram arise. For this reason, several alarms are provided as the level rises and when only

approximately $1\frac{1}{2}$ scram volumes remain, a scram is called for.

Activation of the two emergency heat exchangers is achieved only by high reactor pressure or by closure of the isolation valves. The purpose of the emergency heat exchangers is to reduce reactor power should complete loss of the heat sink (condenser) occur. For this reason, they would normally be brought on only by high pressure in the system. The use of the emergency heat exchangers together with the pilot operated relief valves is sufficient to limit the pressure peak when fast isolation of the system occurs to less than the lowest safety valve setting. However, these pilot operated relief valves are on the downstream side of the isolation valves and upon closure of the isolation valves are no longer available for use. For this reason, a closure of the isolation valves also operates the emergency heat exchangers.

PLANT OPERATION AND STARTUP

Planning for the startup of the Dresden Station has been somewhat more extensive and detailed than is usual for conventional central stations. Two major considerations determined the nature and extent of the preparatory or training program.

1. Commonwealth Edison Co. did not have a pool of experienced reactor operators from which to select the station crew.
2. Dresden will be the first large boiling water reactor plant.

Regular operation. Figure 9-23 shows the organization arrangement that has been selected by the Commonwealth Edison Co. The number of personnel now scheduled for operation is 102. After the station crew gains experience it is believed that the technical and operating personnel can be reduced.

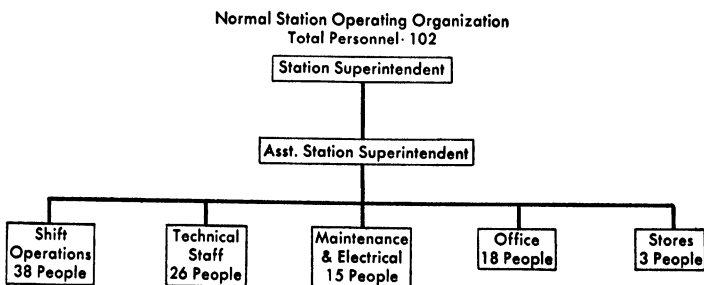


FIG. 9-23. Normal station operation organization.

This organization is for plant operation after Commonwealth Edison acceptance from the General Electric Co. During the time of the first fuel loading and initial operating period, General Electric retains plant responsibility and will technically supervise the startup and operation of the station, using Commonwealth's personnel to perform the actual operating functions. Eight to ten of Commonwealth Edison's supervisory positions will be paralleled by General Electric personnel. During the early phases of startup both companies will have a number of specialists at the plant to collect and evaluate performance data.

Management, supervisory, and technical personnel to staff the organization have been selected and are now engaged in the training program. The remainder of the station staff will be selected in time to participate in field training and the training work at the station.

Training. The training program is divided into two principal categories: (1) Self-study and class room lectures, and (2) on the job training. Each category includes several basic parts, each of which is designed for a specific purpose.

The station startup preparatory program has been designed to assure a thoroughly trained station operating crew and a carefully thought-out and detailed program for the initial loading and early operating periods.

The first part of the training consists of a basic nuclear indoctrination course. This is a simplified course of lectures on nuclear fundamentals which prepares each individual for the remainder of his training. The Dresden Station technology course given to the station supervisory and technical staff was an intensive course of instruction designed to provide a higher degree of knowledge and understanding of the plant characteristics by key station personnel. General Electric engineers and scientists who are working on the station design were used as instructors.

Courses of instruction at the R. A. Taft Sanitary Engineering Center on radiation protection theory and practice will be attended by the radiation protection engineer. The station chemist will be trained in radiochemistry techniques at Oak Ridge National Laboratory. Selected members of the technical staff have been graduated from the Oak Ridge School of Reactor Technology which provided them with valuable background information in the nuclear field.

The preoperational training of the station staff is being supplemented by field training at existing nuclear facilities. The operating crew and technical staff are being assigned to EBWR and VBWR to give them some experience with the day-to-day operating problems of a boiling water reactor. Field training in their respective specialities will be provided members of the technical staff and the maintenance supervisors

at various AEC installations and the Vallecitos Atomic Laboratory. For example, the radiation protection engineer and the maintenance superintendent will be field-trained at Hanford. The nuclear engineer will obtain first-hand knowledge of experimental physics techniques at the Vallecitos critical assembly and VBWR.

The station design course is a condensation of the Dresden technology course and will be given at the Dresden site by the graduates of that course to all personnel directly associated with the operation of the facility. The equipment description course is a detailed discussion of plant equipment by the technical and supervisory staff given to all station operating, maintenance, and technical personnel. A practical course in radiation protection problems and procedures will be given by the radiation protection specialists to all station personnel. The startup and operating instructions will be discussed with operating and technical personnel prior to reactor loading. In addition to the described preoperational training programs the station staff will assist in performing the final testing of the Dresden plant prior to fuel loading as well as the initial loading and test operation of the equipment. During the latter phases of this work operating personnel will be examined for operator's licenses by the Atomic Energy Commission.

Under General Electric's technical direction, the station crew, trained as just described, will perform the work detailed in an initial loading and test operation program prepared in advance.

General Electric is currently engaged in the preparation of this program, which, as now visualized, consists of three major parts: (1) equipment testing and checkout prior to loading; (2) initial loading and low level testing; and (3) power runs and plant performance evaluation tests. Preloading testing will consist of checks of the individual equipment items; for example, strength and tightness tests on pressure vessels. Then as systems are ready, systems tests will be conducted; for example, operation of the reactor safety system by simulated scram signals. Preloading checks will assure the plant equipment is in proper condition for the initial loading. Special attention will be given to the important systems such as the reactor safety system.

As soon as the plant equipment involved has been thoroughly checked and tested, fuel loading will begin. The work will proceed on the basis of previously approved detail plans and procedures. At frequent intervals throughout this entire program, functioning of the critical equipment, e.g., control rods and safety circuits, will be checked. As now visualized, the initial loading and critical testing program will consist of four major steps.

A minimum critical loading will be assembled in a stepwise fashion with source multiplication measurement between steps. The core will

be submerged throughout the loading. When the minimum critical size has been established, rods will be inserted and a small amount of fuel added to allow measurement of the void and temperature coefficients and control rod strength.

Assuming the data collected on the critical loading indicates that the nuclear characteristics are as designed, the assembly will be increased to approximately twice the minimum critical size, again in a stepwise fashion with safety checks between steps. Temperature and void coefficients will be measured in this assembly and data will be collected on rod strength and total excess reactivity.

If the measured quantities agree well enough with those predicted by calculations, loading will then proceed to the limits of the circle circumscribing the control rods. At this point void and temperature coefficients, rod strength, excess reactivity, and flux distributions with various rod configurations will be measured. Again assuming that measured data indicate that loading can proceed safely, the proper size of the fully loaded reactor core will be estimated by measuring the effect on flux distribution of removing peripheral rods.

Loading will then proceed stepwise, with intermediate checks on flux peaking in the peripheral ring, to the fully loaded configuration. Void and temperature coefficients measurements and flux distribution measurements for various rod configurations will be made on the fully loaded assembly, cold.

When the reactor is loaded to its operating size, the reactor vessel closed, and all equipment involved has been carefully checked, the reactor will be brought to criticality and a series of power runs at low, intermediate, and full power will be made. Full power will be approached in a stepwise fashion with a thorough check of plant equipment and radiation levels at steady state as each new power level is achieved. Power generation equipment will be started up and checked out when sufficient reactor steam is available.

The first step will be at about one-tenth power, 300 psig and approximately 25 per cent recirculation flow. Under these conditions the following studies will be conducted:

1. Void coefficients will be measured at several boiling rates.
2. Reactivity effects of variations in reactor pressure, water temperature, and transient poisons will be studied.
3. Transient stability of the reactor will be studied.
4. Power distributions will be measured.

In addition, functioning of plant equipment will be critically examined and plant radiation surveys made. These tests will be repeated at the

same reactor power but at rated temperature and pressure and recirculation flow.

A similar series of tests will be run at each of several intermediate power and rated conditions. In this fashion familiarity with the plant characteristics can be obtained in a safe and orderly manner. The studies at full power will check the actual performance of the plant against designed values and will indicate any corrections or improvements needed. In addition, these runs will be used to make certain that the station staff is thoroughly familiar with its operating duties. AEC operator license examinations for the appropriate station personnel will be conducted during this period.

The program outlined involves well over 100 individuals for varying lengths of time over a four year period. The program is estimated to cost nearly three-quarters of a million dollars including salaries and expenses of individuals involved in training. The thoroughness of this program is essential to the efficient and safe startup and operation of the Dresden Nuclear Power Station.

POWER INCREASE POTENTIAL OF DRESDEN AND DRESDEN TYPE PLANTS

In recent months significant information on UO_2 fuel has been gained from irradiation, critical assembly, heat transfer, and burnout tests conducted at the San Jose and Vallecitos facilities of General Electric Co. In the light of the data obtained in these tests it appears that the Dresden core is a sound, conservative design and that it has considerably higher capabilities. Currently a great deal of effort is being expended to uncover methods that can be reasonably applied for deriving an increase in the electrical generating capacity of Dresden Station or similar type stations. Irradiation tests have demonstrated that increases in thermal output of the core must be limited to those increases associated with improvement of the power distribution within the core, i.e., lowering of the hot spot factor. For purposes of discussion this avenue of approach will be divided into three phases:

1. Increase in capacity with present designed core and additions of turbine-generator unit and auxiliaries.
2. Increase in capacity with redesigned core, new turbogenerator and auxiliaries without making major changes in steam production equipment.
3. Redesign of the Dresden Station for another location.

Phase I. The capability of the plant can be increased to 205,000 kw net with the present core. This represents a gain of 25,000 kw. This is possible by reducing the peak to average power output (based on satisfactory test data) from 3.66 to possibly as low as 2.85. It may be possible to get 10,000 kw more from the present turbine; however, this is contingent upon the turbine passing approximately 5 per cent more than guaranteed maximum flow. Assuming that the turbine will exceed the warranty by 5 per cent, it would only be necessary to add a 15,000 kw turbine-generator.

Phase II. A redesign of the Dresden fuel giving a larger surface to volume ratio (reduce fuel element size approximately 0.1 inch and increase the fuel bundles from 36-rod to 49-rod assemblies with core size remaining the same) would allow the thermal output of 65,000 kw or a Station rating of 245,000 kw net. This would allow the thermal output of the reactor to stay within the central temperature limits of the fuel—which is no melting. This increase in output can be obtained by increasing the primary *steam production*. The reactor recirculating and separating equipment can accommodate a 62 per cent increase in primary steam flow without alteration.

Phase III. For an advanced station design of the Dresden type the following changes would provide the potential of generating a net 300 Mw.

- (a) Redesign of fuel to a 225—0.4 in. OD rod bundle. This would alleviate central temperatures, reduce local hot spots and increase surface area to allow an increase in quality without a decrease in burnout safety factor.
- (b) Increase pumping capacity.
- (c) Increase all steam handling equipment.

Parametric studies have been made of the above phases and many additional methods of increasing generating capacity of the Dresden type reactor.

REFERENCES

1. R. G. LORRAINE, A Review of Boiling Water Reactor Plants for Atomic Power Generation. Paper presented at Virginia Polytechnic Institute, Blacksburg, Va., August 8, 1956.
2. J. R. WOLCOTT, V. A. ELLIOT, E. P. PEABODY, G. M. ROY, J. L. SHANZ, G. SEGE, The Dresden Nuclear Power Station. Presented at ASME Annual Meeting, New York, N. Y., November, 1956.
3. VAUGHN D. NIXON, A Review of Dresden Nuclear Power Station. Presented at the Rocky Mountain Electrical League, Denver, Colo., April 14, 1958.
4. J. W. WEIL et al. Boiling Water Reactor Stability, General Electric Co. Report GER-1468, January, 1958.
5. H. ONGMAN, Water Treatment Process for the Dresden Nuclear Power Station. Presented at the Nuclear Engineering and Science Conference, March, 1958, Chicago, Illinois.
6. S. SEGE, Containment-vessel Design Basis for the Dresden Nuclear Power Station. Presented at the Nuclear Engineering and Science Conference, Chicago, Illinois, March, 1958.
7. C. F. FALK, Radioactive Liquid Waste Disposal from the Dresden Nuclear Power Station. Presented at the Nuclear Engineering and Science Conference, Chicago, Illinois, March, 1958.
8. LESTER KORNBLITH, JR., Instrumentation for a Boiling Water Reactor. Presented at the Nuclear Engineering and Science Conference, Chicago, Illinois, March, 1958.
9. J. LOVE, C. S. DARROW, and B. H. RANDOLPH, Civil Engineering Aspects of the Dresden Nuclear Power Station. Presented at the Nuclear Science and Engineering Conference, Chicago, Illinois, March, 1958.
10. G. SEGE, J. R. WOLCOTT, and V. D. NIXON, Preliminary Hazard Summary Report for the Dresden Nuclear Power Station, May 1, 1957.
11. R. C. THORBURN, C. A. PURSEL, and G. SEGE, Enclosure Section of the Hazard Summary Report for the Dresden Nuclear Power Station, January 24, 1957.
12. K. T. PERKINS and H. K. HOYT, Planning for the Start-up of the Dresden Nuclear Power Station. Presented at the Nuclear Engineering and Science Conference, Chicago, Illinois, March, 1958.

CHAPTER 10

OTHER BOILING WATER REACTORS

The boiling water reactors that are described in the earlier chapters of this book are all either in operation or, as in the case of the Dresden Station, in an advanced stage of construction. Several other boiling reactor power plants are being designed and while technical data on these plants are not complete, enough is known about them to describe their general features. In addition to the plants in the United States, boiling reactor plants are also planned for Latin America, and a boiling reactor plant for producing process steam for heating is being built in Norway.

Three boiling reactor plants are to be built in the United States. These are as follows:

1. The Northern States Power Co. 66,000-kw recirculation boiling reactor to be built at Sioux Falls, North Dakota.
2. The Rural Cooperative Power Association Elk River plant of 22,000 kw (electrical) capacity, at Elk River, Minnesota.
3. Pacific Gas & Electric Co.'s Humboldt Bay Atomic Unit No. 3, with a capacity of 50,000 kw to be built near Eureka, California.¹

The first two of these projects are being constructed under the Atomic Energy Commission's Power Demonstration Reactor program. The Humboldt Bay plant, however, is being financed entirely by the Pacific Gas & Electric Co. The decision of the latter company to build this nuclear unit was due largely to the excellent performance of the Vallecitos boiling reactor plant with which Pacific Gas & Electric Co. is associated, as explained in Chapter 7.

NORTHERN STATES POWER COMPANY'S CONTROLLED RECIRCULATION PLANT

The Northern States Power Company's plant is being designed and constructed by the Allis-Chalmers Manufacturing Co., with Pioneer Service and Engineering Company as Architect-Engineer. Also associated with this project is a group of utility companies contributing financially for research and development. The contractual relationship of the various organizations involved is shown in Fig. 10-1.²

The selection of a boiling water type of reactor for this project was significantly influenced by the wealth of experience the company has had with conventional steam systems and faith in the results obtained with the Experimental Boiling Water Reactor at Argonne. The fact that the

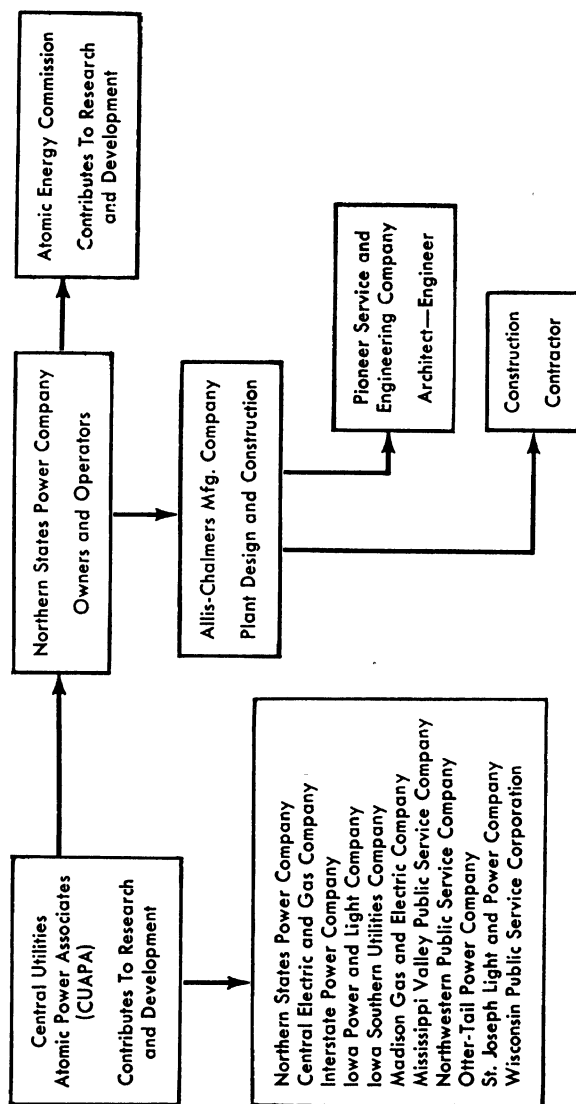


Fig. 10-1. Contractual relationship for Northern States Reactor.

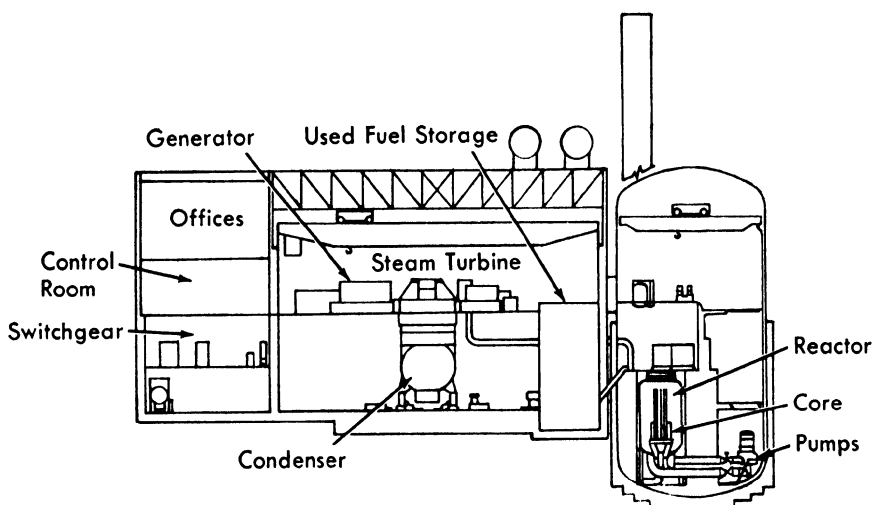


FIG. 10-2. Plant elevation, Northern States.

turbine at EBWR was essentially free from radioactive contamination after a year's operation lends great encouragement to the designers of the Northern States project.

The total cost of the project, excluding operating costs, is estimated at \$28.8 million. The Atomic Energy Commission will participate in research and development up to \$6 million, plus an additional \$1 million waiver of fuel-use charge. The associated utility companies (CUAPA) will contribute \$3,650,000 toward research and development.

The most serious obstacle confronting economic use of nuclear power in the United States is the very high capital cost of the plants. Therefore, it was agreed at the start that effort would be concentrated on reducing capital cost of this plant. The intention is to build a practical nuclear power plant without frills or unnecessary facilities. The use of aluminum fuel elements should materially reduce one factor of the operating cost.

Recirculation. The Northern States plant will be the first large boiling reactor plant designed specifically for recirculation of the water in the reactor. Three large capacity circulating pumps will be installed on the reactor as shown in Figs. 10-2 and 10-3. It is expected that their use will permit a very high heat output from the core while maintaining good flux distribution. Without the pumps the power density in the top of the core would be considerably reduced because of the large void fraction.

A variable-speed control on these pumps provides a means of control in addition to that of the shim rods, and will eliminate any need for continuous bypass of steam to the condenser. Very little power is required

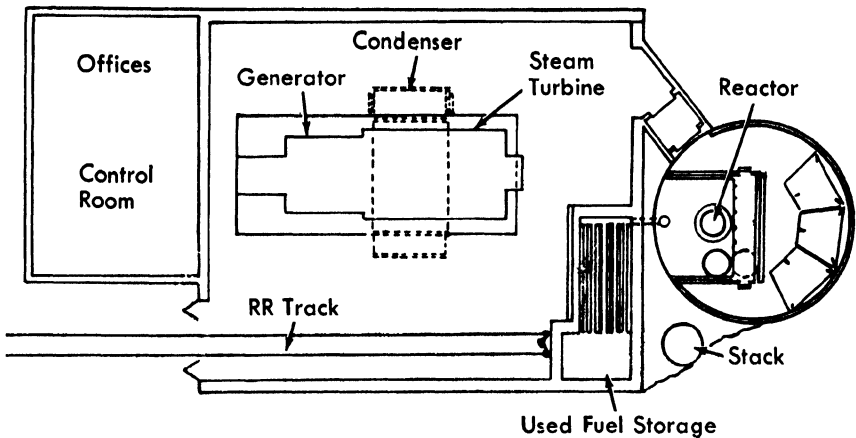


FIG. 10-3. Plan of station, Northern States.

to drive these pumps because of the low head against which they operate. The fact that they will be handling highly radioactive water, however, makes it necessary to install them in shielded enclosures within the reactor containment structure. Valves are provided in the suction and discharge of the pumps to permit servicing one of them while the other two and the reactor are in operation. Shielding is provided between the pumps.

Superheater. An interesting feature of this project is the possible use of a nuclear superheater. The present contract provides for a conventional oil- or gas-fired steam superheater. Such a superheater would provide standard steam conditions, improve plant efficiency and eliminate certain problems associated with the operation of turbines on saturated steam. However, a nuclear superheater has been under study* and this now appears feasible. Its operation can be understood from Figs. 10-4 and 10-5. The reactor core consists of two close-coupled regions. The annular boiling region produces saturated steam which flows to the dome of the reactor vessel. The steam is then conducted down through a multiplicity of tubes containing stainless steel fuel elements which comprise the central core region. Heated to 825 F, the steam then passes through the bottom of the reactor and then directly to the turbine.

Water moderation is provided in the superheating region by the water between fuel-loaded tubes as indicated in Fig. 10-5. Thermal insulation is provided by a double-wall tube with stagnant steam between the walls. Control rods are provided in both regions of the core but, because of the

* See studies on the design of large boiling reactor plants, Chapter 11.

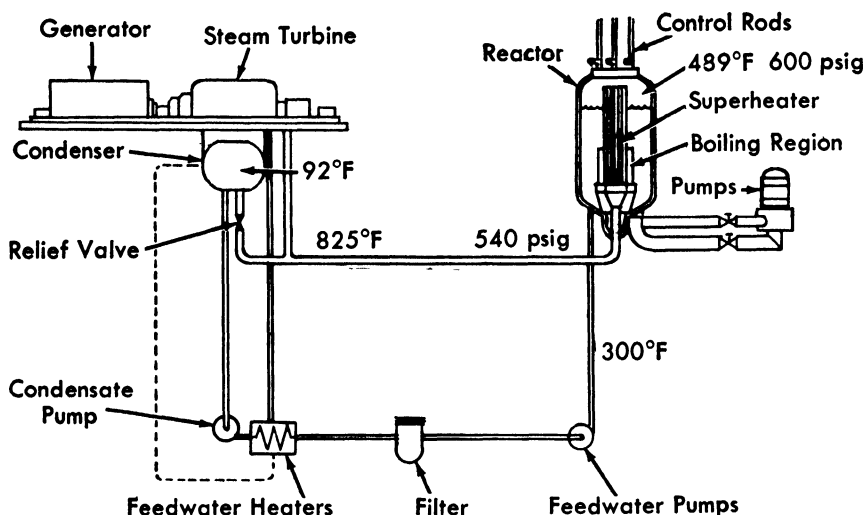


FIG. 10-4. Flow diagram, Northern States.

close coupling between regions, the rods in the superheater will require only occasional adjustment, independent of power demand.

A flooding coefficient, approximately zero, may be obtained by proper design of the superheater, permitting submersion during refueling. This also eliminates the improbable hazard of superheater flooding during operation. The superheater region causes a more uniform power distribution in the boiling region than would be present without the superheater region.

Nuclear fuel costs, using 20 per cent enrichment in the superheater appears to be a little higher than conventional fuels. It is to be noted, however, that *all* the nuclear heat is absorbed by the steam, while a conventionally fired superheater wastes considerable heat. Also, less than 20 per cent of the heat is superheat (see Table 10-1). The future development of low-enrichment fuel elements for superheating should reduce nuclear fuel costs to competitive values.

Steam separator. It is essential that steam bubbles be prevented from being entrained with the recirculating water. This is necessary because bubble entrainment can result in pump cavitation, less efficient reactor operation, and possible reactor instability.

The small surface for steam release and the high water velocity in the downcomer outside the core demand the need for steam-separating devices to remove the steam bubbles from the water.

A principle similar to the cyclone separator is used in effecting this separation of steam from water. Separation is achieved by centrifugal

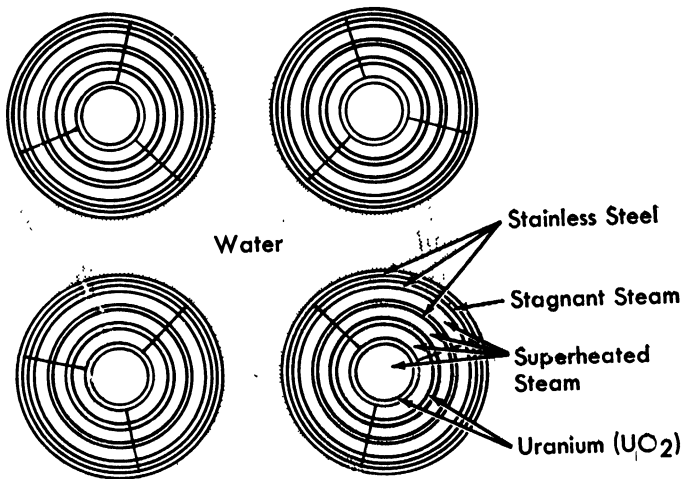


FIG. 10-5. Nuclear superheater diagram.

force in the rotating liquid which floats the bubbles to a void in the center of the separator. Several separators are arranged around the inside of the reactor vessel above and outside the core.

General description of the plant. Figures 10-2 and 10-3 show the general arrangement of equipment. The turbine-generator, which is a 66,000-kw machine, operating at a steam pressure of 600 psi and a steam temperature of 825 F, is located in a conventional building, as is also the associated auxiliary equipment. As in the case of EBWR and VBWR, the steam going through the turbine will be only slightly radioactive and no special shielding will be provided. However, light shielding will be provided over the air ejectors. In addition, a stack will be provided for the discharge of slightly radioactive gases removed from the air ejectors after dilution with air.

The reactor, recirculating pumps, emergency coolers, etc., are located within the containment shell which is a steel cylindrical vessel similar to those at EBWR and VBWR. All this equipment within the shell can be isolated from the equipment in the turbine building by fast-operating valves. A water shield is located above the reactor for deep-water maintenance and refueling operations.

Drives for the cruciform shim rods will be located on the top plug. The plug may be stored in the shield water tank during refueling or lifted out for repairs.

Refueling operation. One-third of the reactor fuel will be replaced three times a year. Used fuel is stored under water in the turbine building, thereby permitting handling of the coffins with the same crane that serves the steam turbine and generator. The sequence of used-fuel disposal is as follows:

The plug is removed from the reactor vessel and a long handled grapping tool is used to lift out the radioactive fuel element. The operator, who can observe the action through the water, drops the fuel element into a chute. It then slides into the fuel storage where it decays for 60 days. The cool fuel is then loaded into a lead coffin, put on a freight car, and shipped to the AEC processing plant.

The cycle is completed by placing a new fuel element into the reactor.

Fuel elements. Aluminum-jacketed UO_2 pellets will be used for the boiling region of the core. Information pertaining to the corrosion resistance of aluminum is promising but not conclusive at this time. Several loops are being installed by Allis-Chalmers to further contribute to the corrosion investigations being done by Argonne National Laboratory and others.

The superheating fuel elements will be made of UO_2 stainless cermet, clad in stainless steel. Three concentric fuel tubes, 6 ft long, will be required in each assembly. Fundamental data are shown in Table 10-1.

Eliminating containment. Included in the Controlled Recirculation Boiling Reactor (CRBR) project is a study of the feasibility of eliminating containment. It is recognized that the containment cannot be eliminated from this "Pathfinder" plant, but it is expected that future plants may be built without containment. Further simplification may also result from the studies.

There is considerable evidence that no reactions between water and metal are credible, and, if they do occur, less than one per cent of the metal is consumed. The CRBR reactor vessel would not rupture if a 1 per cent equivalent amount of slow burning explosive were to be discharged inside it.

Studies of decontamination factors following a major release of radioactivity in the common building which houses both turbine and reactor will be made and an attempt will be made to find the boundaries of *credible accidents*. The designers of this project believe that a very real contribution to the effort of reducing cost will be made when industry is able to eliminate containment.

TABLE 10-1
CRBR WITH NUCLEAR SUPERHEATER PRELIMINARY DESIGN DATA

<i>Plant</i>	
Power, boiling region	164,000 kw
Power, superheating region	39,000 kw
Total core power	203,000 kw
Gross electrical capability	66,000 kw
Net electrical output	62,000 kw
Net efficiency	30.5%
Steam outlet pressure	540 psi
Reactor operating pressure	600 psi
Temperature, boiling region	489 F
Outlet temperature, superheating region	825 F
Gross heat rate	10,370 Btu/kwh
Containment shell size	45 ft × 75 ft
<i>Reactor</i>	
Vessel size	11 ft × 26 ft × 3 in. thick
Total core dimensions	6 ft × 6 ft
Dimensions of superheating region	6 ft × 30 in.
Fuel, boiling region (Al clad)	1.6% enriched UO ₂
Fuel, superheating region (S.S. clad)	20% enriched UO ₂
Fuel loading, boiling region	114 kg (U ²³⁵)
Fuel loading, superheating region	32 kg (U ²³⁵)
Power density, average	41.6 kw/liter
Average heat flux, boiling region	100,000 Btu/(hr)(ft ²)
Average heat flux, superheating region	52,000 Btu/(hr)(ft ²)
Maximum heat flux, boiling region	250,000 Btu/(hr)(ft ²)
Maximum heat flux, superheating region	150,000 Btu/(hr)(ft ²)
Recirculation rate	60,000 gpm
Recirculation pump power	620 kw

THE ELK RIVER PLANT

The Elk River nuclear project of the Rural Cooperative Power Association of Elk River, Minnesota, is another boiling reactor plant to be built under the Atomic Energy Commission Power Demonstration Reactor Program. Contracts for the construction of this plant were signed between Rural Cooperative and AEC and Nuclear Products-Erco, Division of ACF Industries,³ early in 1958, and construction will be started in the near future. The present schedule is to complete the plant by October 1960.

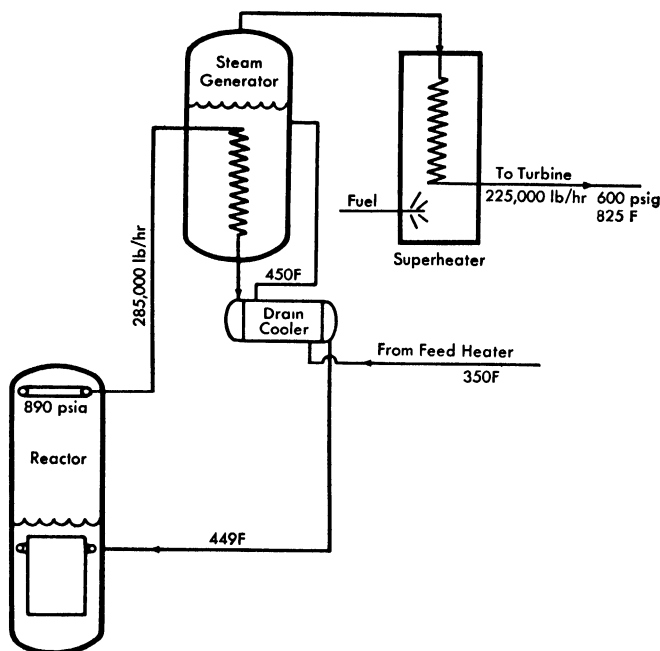


FIG. 10-6. Simplified flow diagram, Elk River.

The Elk River project is unique not only in that it will be a comparatively small commercial nuclear plant—22,000 kw (electrical)—but because it is the first boiling water reactor system to use the indirect cycle. As explained in Chapter 2, the indirect cycle involves two distinct loops linked thermally through a condensing-evaporating heat exchanger. Steam generated in the reactor is condensed on the primary side of the heat exchanger by giving up its heat to generate lower pressure steam on the secondary side.

The system is designed to furnish a steam flow of 225,000 lb of steam per hour to the turbine throttle, at a pressure of 600 psig and at 825 F. The basic elements of the steam generating plant are shown in Fig. 10-6. The primary coolant circulation outside the reactor and recirculation within the reactor are by natural convection. Both circuits are designed for the future application of forced circulation for increased capacity if desired.

In the primary circuit, 258,000 lb/hr saturated steam is generated within the reactor at 875 psig pressure at rated capacity and is passed from the reactor vessel to the primary heat exchanger where the secondary steam is generated. The resulting saturated primary liquid returns through a subcooler to the reactor at 449 F.

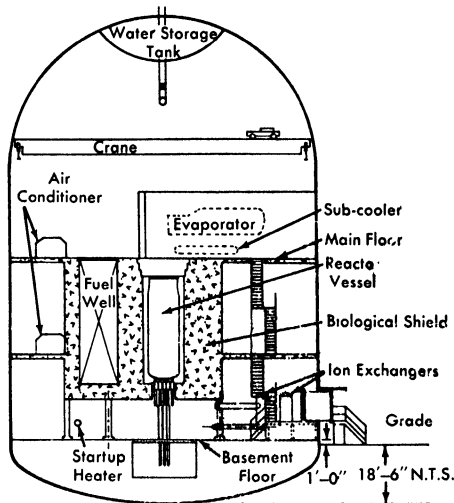


FIG. 10-7. General arrangement, Elk River.

In the secondary circuit, 225,000 lb/hr of feedwater from the fourth stage heater is pumped through the shell side of the sub-cooler for pre-heating, then to the shell side of the primary heat exchanger, converted to steam, superheated in a coal-fired superheater, and delivered to the turbine throttle.

The main advantage of the indirect method for generating steam for electrical power production is the complete absence of radioactivity in the secondary steam, thus permitting continuous access to the electrical plant.

The reactor will produce 58.2 Mw of heat and the separately-fired superheater will produce 14.8 Mw of heat, a total of 73 Mw. The turbo-generator will have an electrical output of 22,000 kw.

The general arrangement of the reactor and superheater installation is shown in Figs. 10-7 and 10-8. The reactor, steam generators, and their auxiliaries will be located in a containment vessel while the superheater and its auxiliaries will be installed in an enclosure adjacent to the existing power plant. It should be noted that an existing, conventionally fired steam-electric plant has been in operation at the site for a number of years. The control room for the new nuclear plant will be located adjacent to the existing plant controls.

Other principal equipment in the reactor building will include the reactor water purification equipment, air conditioning equipment, emergency condenser, fuel storage and waste disposal tanks, and biological shielding. All radioactivity is therefore confined to the reactor building.

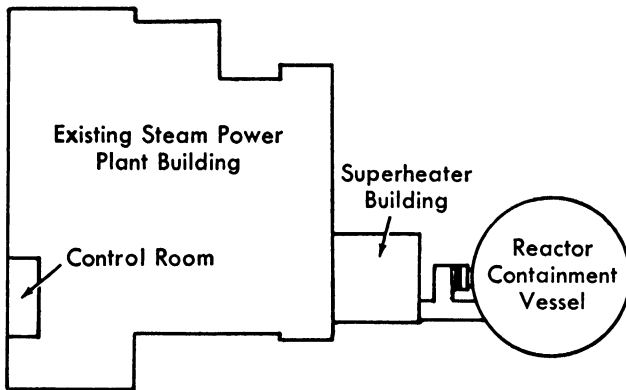


FIG. 10-8. Plan of present building and proposed reactor and superheater.

Ample compartmentalization for major equipment is provided in order that maintenance may be performed while the remainder of the plant is operating at reduced capacity.

The containment vessel, required to prevent the spread of radioactivity in the event of a circuit rupture, may be occupied during normal operation. Normal access for personnel into the containment vessel will be provided by an airlock consisting of two manually operated doors in series with an interlocking arrangement which ensures that one door is closed all the time. A large bolted freight door will be provided to permit removal of large pieces of equipment; this, however, will not be an airlock. Its use will be restricted to periods when the reactor is not operating.

As in the case of Vallecitos, temporary steel columns will be required to support the structure during erection and testing of the vessel. Upon completion of this phase of the project, a concrete foundation will be poured under the vessel for the purpose of carrying the internal equipment and shielding loads.

Radiation levels in the primary system will be sufficiently intense to require biological shielding. The design of the shielding, however, will incorporate compartmentalization to make it possible to service equipment during operation of the remainder of the plant.

Several heating and ventilating units located within the containment vessel will provide normal ventilation. These units will draw outside air into the containment vessel, circulate it, and exhaust it through a vent stack as shown in Fig. 10-7. Both inlet and exhaust ducts will have quick-closing valves actuated by a radiation monitoring system. The exhaust stack is designed to have sufficient volume ahead of the valve to serve as a hold-up tank for a period long enough to allow the valve to close, thus preventing the escape of radioactive gas into the atmosphere.

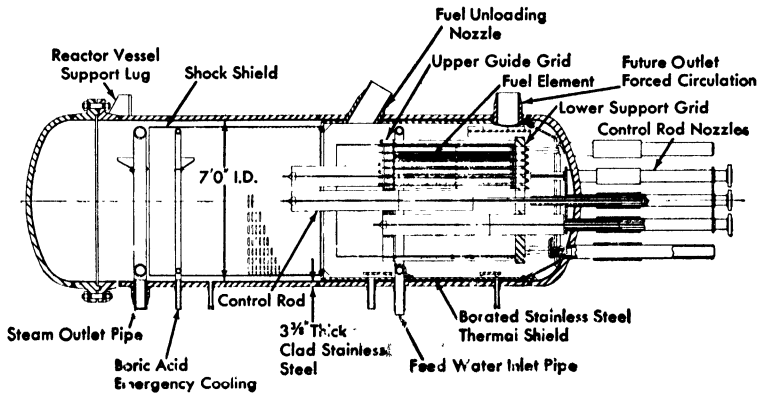


FIG. 10-9. Reactor details, Elk River.

All process and control information will be fed into a centralized control room located within the existing building from which all major operational decisions will be made. An auxiliary instrument area close to the control room will be required to house the equipment associated with indicators or recorders on the control boards.

Where radiation problems exist, shielding and equipment arrangements will be designed so that standard control devices can be used to their fullest extent.

The reactor complex. *Reactor vessel.* The reactor vessel is approximately 84 in. internal diameter, 23 ft high, $3\frac{3}{8}$ in. thick, fabricated of carbon steel and clad internally with a $\frac{3}{32}$ -in. layer of stainless steel. Figure 10-9 shows the reactor vessel arrangement.

A thermal shield of 1-in. thick stainless steel surrounds the core area and is spaced 1 in. from the vessel wall. This design provides an annulus for the flow of water to cool both the shield and the vessel wall.

The core is centered in the vessel. An 11-in. annulus between the core and thermal shield serves as a downcomer for natural water circulation. Feedwater at 449 F enters the reactor through two 6-in. nozzles and a distributing ring above the downcomer. The feedwater cools the water in the reactor as it enters the downcomer with the result that the water entering the fuel elements is subcooled about 3 F. A light louvered shock shield $\frac{1}{8}$ in. thick is installed above the water level line to prevent thermal shock to the pressure vessel as a result of impingement by cold boric acid solution.

The steam dome above the water level is approximately 7 ft, 6 in. high, and designed to provide low steam velocity for good steam-water separa-

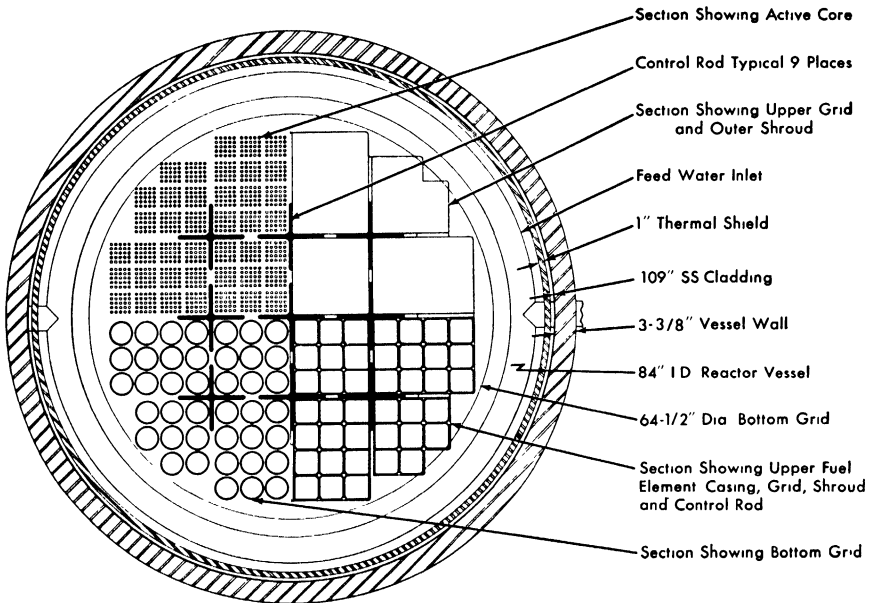


FIG. 10-10. Horizontal section, Elk River.

tion and low steam radioactivity. A dry-pipe collects the steam for discharge from the vessel through two 8-in. steam nozzles at near 100 per cent quality. The design of the reactor vessel also incorporates inlet and exit nozzles for future increase in heat output to be obtained with forced circulation. The water would enter below the core, rise through the core, flow downward peripherally, and leave the reactor by a side outlet at the base. Provisions are made for future installation of a shroud to surround the core and bottom grid for the forced circulation water up through the core elements for increased power at a later date.

Core structure. The reactor core structure consists basically of a lower support grid and upper guide plate fastened together by means of tie rods and encased in a shroud which extends from the lower grid to a point two feet above the upper guide plate. There are 164 fuel element spaces in the lower grid plate, and under normal conditions 148 of these will be occupied by elements. The lower plate will bear the entire weight of the fuel and will transmit this load to the pressure vessel by means of six legs attached to the bottom of the grid and resting on the pressure vessel. The upper guide plate will locate the fuel elements and prevent any local motion. Each fuel element is constructed so that the upper portion forms a spring padded box which can be centered into a guide plate. All structural elements just described are of stainless steel. The proposed core

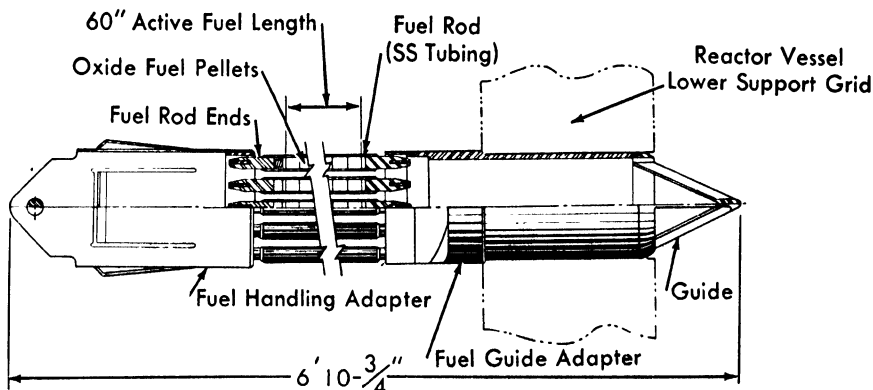


Fig. 10-11. Fuel element assembly, Elk River.

arrangement is shown in Fig. 10-10. The active core height is 60 in. and the active radius is 28.45 in.

The reactor core is reflected on all sides by about 10.5 in. of light water. This is the coolant water surrounding the reactor core and gives good reflector savings to the core in the radial direction. The reflectors on top and below the core are made up of water and steel. This offers additional reflector savings although somewhat less than pure water.

Reactor core performance. The hydraulic characteristics of the Elk River core were examined for the proposed design of 148 elements, each consisting of a 5 by 5 array of 0.450 in. diameter tubes on a 0.75-in. pitch. The heat balance gave a division between heating the water and boiling as shown in Table 10-2. Analysis indicates that boiling will commence on the average approximately 0.9 ft from the bottom of the active element and will extend over a length of about 4 ft. The average indicated water velocity is 4 ft/sec for 58 Mw(t) operation. The recirculation is 28.2 to 1 giving 2.9 F of subcooling at the core inlet and a water recirculation of 18,900 gpm. Performance parameters are shown in Table 10-2.

The calculations used in arriving at these figures were based on the use of a 2-ft shroud above the core; in addition allowance was made for scaling as observed in the Experimental Boiling Water Reactor at Argonne. EBWR operation at an average power density of 62.5 kw/liter, and 40 to 50 per cent exit voids produced no instability or reactor oscillation. These conditions are both more severe than those proposed for Elk River.

A total core life of 27 months may be obtained by inserting the 22 spare elements, each of double uranium loading, at the end of a 15-month core period. In this arrangement, the 22 center elements are withdrawn and are replaced by 22 peripheral elements. The 22 spare elements will be

TABLE 10-2
ELK RIVER REACTOR PERFORMANCE DATA

	Btu/hr
<i>Heat balance</i>	
Heat absorbed in boiling	173,000,000
Heat absorbed in heating the water	25,000,000
Total reactor heat	198,000,000 (58.2 Mw)
<i>Performance parameters</i>	
Average power density in coolant	39.6 kw/liter
Average inlet velocity	4.05 fps
Average Reynolds number	320,000
Average exit void fraction	0.312
Average exit quality	0.036
Average void fraction over boiling length	0.237
Average void core fraction in the core	0.192
Average heat flux	89,800 Btu/(ft ²)(hr)
Total steam rate	258,000 lb/hr
System pressure	875 psig
Boiling temperature	531 F
Inlet water temperature	528 F
Feedwater temperature	450 F

inserted in the initial vacant peripheral positions. The fuel program supposes 15 months operation without disturbing the element arrangement, then the transfer of elements as described for an additional 12-month period.

The proposed fuel elements consisting of thoria-urania pellets clad in stainless steel have been evaluated to determine possible temperatures and stresses. These consist of stainless steel tubes containing stacks of pellets composed of thoria plus enriched urania.

Data of fuel element design, composition, burnup, heat transfer, etc. are given in Tables 10-3 through 10-5. The heat flux, as can be seen from Table 10-4, is well below the burnout point at all locations in the reactor core. Numerous correlations exist for the boiling heat transfer burnout heat flux. The lowest value calculated from these correlations was selected as shown in the table.

The burnup of fuel and accumulation of fission products (Table 10-4) is based on 450 days of operation between refueling and a 100 per cent plant factor.

TABLE 10-3
FUEL-ELEMENT DATA

Tube internal diameter	0.410 in.
Tube wall thickness	0.020 in.
Tube material	AISI type-304 ELC
Thoria-urania pellet diameter	0.407 in. (0.5 in. long)
Total effective pellet length	5 ft
Number of rods per element	25
Total number of fuel elements	148
Total number of rods	3700

TABLE 10-4
HEAT-TRANSFER AND BURNUP DATA

Average heat flux	89,800 Btu/(ft ²)(hr)
Maximum heat flux	310,000 Btu/(ft ²)(hr)
Burnout	900,000 Btu/(ft ²)(hr)
Average metal atom burnup	0.815%
Average total atom burnup	0.271%
Average Mwd per ton (thoria and urania)	5870
Maximum metal atom burnup	2.82%
Maximum total atom burnup	0.938%
Maximum Mwd per ton (thoria and urania)	20,200

TABLE 10-5
FUEL PIN PERFORMANCE

	Heat flux, Btu/(ft ²)(hr)	
	89,800	310,000
Boiling fluid temperature, °F	531	531
Stainless steel outside surface temperature, °F	545	566
Stainless steel inside surface temperature, °F	560	618
Surface temperature of thoria-urania pellet, °F	1396	1421
Center temperature of thoria-urania pellet, °F	1830	3420
Compressive stress in the SS tube, psi	9820	4910
Thermal stress in the SS tube, psi	2520	8700

TABLE 10-6
FUEL REQUIREMENTS (15 MONTHS OPERATION)

Cold clean critical mass	90.0 kg
Temperature and voids	18.0 kg
Fuel burnup	34.2 kg
Burnable poisons	8.6 kg
Equilibrium xenon and samarium	8.2 kg
Fuel needed	159.0 kg
U ²³³ buildup equivalent	11.0 kg (U ²³⁵)
Total U ²³⁵ loading	148.0 kg

Fuel	Mass, kg	Mass per pin, g
U ²³⁵	148	40
UO ₂ (90% or better U ²³⁵)	180	48.6
ThO ₂	4220	1140

The fuel element proposed for the Elk River reactor is shown in Fig. 10-11. It consists of a 5 by 5 square array of fuel tubes complete with a lower fuel guide adapter and an upper fuel handling adapter. The tube is closed on both ends with slotted end pieces which are welded into support grids of the lower and upper adapters. Each fuel tube contains 120 fuel pellets, 2 thermal insulators and a hold-down spring.

The 148 regular fuel elements are loaded with pellets each containing 0.349 g of U²³⁵. Twenty-two spare fuel elements are provided which are loaded with pellets each containing 0.833 g of U²³⁵. Helium gas at 1 atmosphere standard temperature and pressure furnishes the heat transfer bond between the pellet and the tubing wall. A thermal insulating wafer of sintered magnesium oxide is placed at the ends of the fuel pellets to prevent overheating with resulting thermal stresses of the end closures. A hold-down spring is placed in the fission gas space above the pellets. The individual fuel tubes are jointed by AISI type-304 steel tabs at positions one-third and two-thirds of the distance along the length as shown in Fig. 10-11. The separate tabs are brazed to the tubes and spot welded together. The end tab of each row of tubes is welded into a short side plate which thus holds the rows of tubes firmly in position.

The reactor core physics calculations are presented in Tables 10-6 through 10-8. These describe the core condition as a function of reactor operating time. Since the core is similar in size and shape to the EBWR

TABLE 10-7
CONTROL REQUIREMENTS

	Mass kg added	Reactivity $\Delta k/k$ %
Temperature and void for 68 to 531 F, 0 to 18% void	18.0	5.64
Fuel burnout, 1.3 g per Mwd, 26,300 Mwd	34.2	10.71
Fission product poisons	8.6	2.69
Equilibrium xenon and samarium	8.2	2.57
U ²³³ production equivalent	-11.0	-3.45
	58.0	18.16

TABLE 10-8
MISCELLANEOUS DATA, ELK RIVER REACTOR

<i>Poisons, per cent reactivity at end of core life</i>	
Fission product poisons	2.7
Samarium poisons	0.6
Xenon poison (equilibrium)	2.0
Maximum xenon after shutdown	2.9
Total	8.2
<i>Reactivity coefficients</i>	
Mass coefficient, hot critical	0.22 $\Delta k/k$ per kg
Mass coefficient, end of core life	0.34 $\Delta k/k$ per kg
Temperature coefficient, for cold to operating	$-9.6 \times 10^{-3} \Delta k/k$ per °C
Void coefficient, at operating temperature	$-1.444 \times 10^{-2} \Delta k/k$ per % void
<i>Average thermal flux in core</i>	
Clean operating core (58.5 Mw)	1.2×10^{13} n/(cm ²)(sec)
End of core life (58.5 Mw)	1.6×10^{13} n/(cm ²)(sec)
<i>Volume fraction of UO₂ to ThO₂</i>	
At startup	3.79
At end of core life	2.91
Per cent burnup of U ²³⁵ (15 months)	23.1
Conversion ratio, at startup	0.47
Conversion ratio, at end of core life (15 months)	0.55

at Argonne National Laboratory, use has been made of existing information established there.

The worth of the 9 control rods when fully inserted in the core is 15 per cent $\Delta k/k$ at 68 F. So that the reactor will be critical at the end of the 15 month cycle with burnout poisons, xenon and samarium buildup and 18 per cent void in the coolant water, 58 kg of enriched uranium are required. Since the reactivity control of the 9 control rods is 15 per cent, in order to have sufficient safety in the control rods it is necessary to obtain an additional 8.2 per cent excess reactivity by burnable poisons built into the core as shown in Table 10-8. To obtain this amount of control will require 2.7×10^{-4} g of boron-10 per cm^3 of core. The overall control of the rods will be 1.5 times the excess reactivity at clean cold startup.

Thermal shield. The thermal shield will consist of a sheet of 1 per cent boron stainless steel 1 in. thick. The $3\frac{3}{8}$ in. thick reactor vessel is surrounded by 6 in. of cooled steel. This will reduce the temperature of the surrounding concrete below 150 F.

Reactor control rods and drive mechanisms. The upper sections of the 9 cross-shaped control rods will contain strong neutron absorbing material while the lower sections will contain low absorption materials.

The rod drives and seals are attached to thimbles welded to the bottom of the reactor vessel as in the EBWR. See Fig. 10-12. The drive incorporates an attachment to the control rod and a pressure breakdown seal similar to that used on the EBWR. The drive itself consists of an air balancing cylinder and piston, and a lead screw and thrust plate system for controlling the motion of the air piston. Air pressure is supplied to the top of the air cylinder to create a downward force of approximately 100 lb in addition to the reactor pressure. An accumulator will be provided to maintain this pressure in the event of failure of the normal air supply. Should the pressure fall below a safe minimum, the rod will scram. Controlled pressure sufficient to counterbalance the reactor pressure on the drive and the weight of the rod is applied to the bottom of the air cylinder.

In normal operation the bottom of the cylinder is pressurized, thus exerting a positive upward force on the thrust plate. The control rods are raised or lowered by energizing the drive motor which raises or lowers the thrust plate through ball bearing lead screws and chain drive. Scramming is accomplished by exhausting the bottom of the air cylinder to the atmosphere.

Rod position indication is obtained from a synchro-transmitter and receiver system coupled to the rod drive. Scram action of drive will be *fail-safe* since air pressure will always be present on the upper side of the air piston and can thereby force the drive down. The balancing pressure

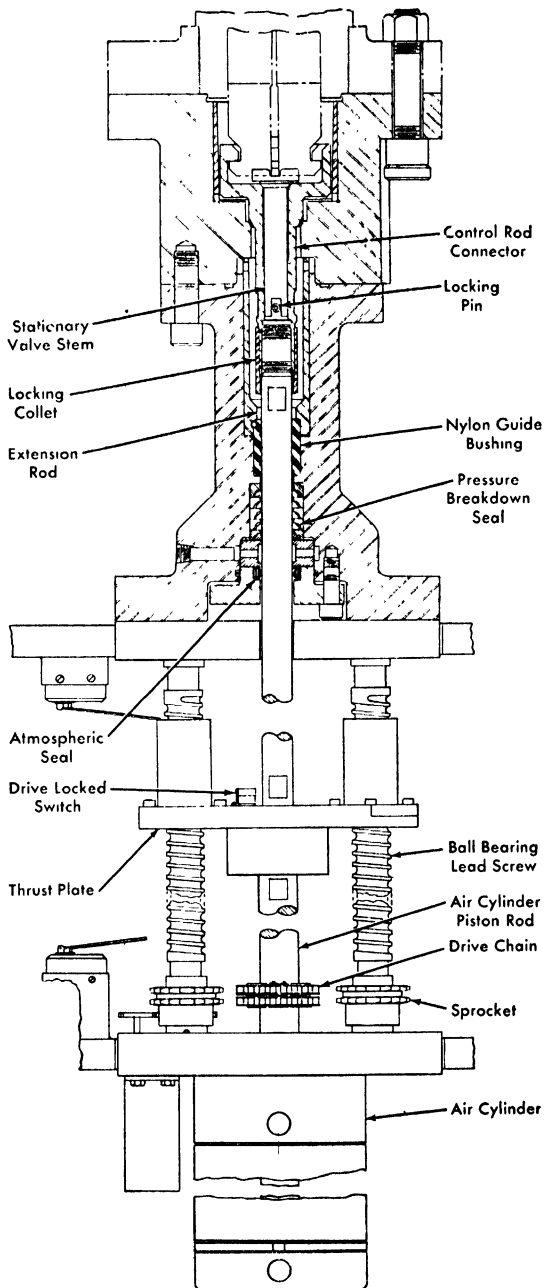


FIG. 10-12. Control rod drive, Elk River.

on the lower side of the air piston will be supplied by a spring and air loaded valve actuated by a solenoid. Thus, an electrical failure will cause this valve to open, exhausting the balancing pressure. This drive mechanism has been tested and proved reliable on the BORAX and SPERT reactors.

Reactor control and instrumentation. The neutron flux will be monitored by 9 independent instrument channels, with 4 different types of channels as follows: Startup channels; Log N and period channels; linear power channels; and level safety channels. At shutdown the neutron density will be between 10^{-8} and 10^{-7} of the full power density. When the neutron density reaches a value of about 10^{-2} of rated power density, the water in the reactor will begin to boil, and at this density the reactor control will be based entirely on the steam power delivered; below this value, control of the reactor power will be by means of the neutron flux sensitive instruments.

A neutron source will be built into the reactor and as a result the startup channels will give a reading of 30 to 60 counts per minute.

Automatic control system. The automatic control system for coordinating reactor output with the turbine-generator demand is outlined in Fig. 10-13. The basic sensing device is a pressure controller located in the secondary steam system at the superheater outlet. Any deviation from normal pressure will result in an immediate change in the fine control rod position, thus effecting a change in the reactor power level. Therefore, changes in load or other incidents resulting in secondary pressure changes will automatically be compensated for by the resultant correction in reactor power. This mechanism permits the reactor to follow the turbine-generator load. Minor fluctuations in pressure will be corrected by the negative temperature coefficient inherent in the reactor design.

Each primary steam loop will contain a pressure limit switch to limit the pressure rise within those loops. This provision is considered desirable in order to back up the basic pressure sensing device in the secondary system in case it failed to operate or in a situation where the primary system pressure rises without an accompanying secondary pressure rise. These limit switches, of course, will be set so they will not interfere with normal operating control.

A three-element feedwater control system will maintain the water level in each steam generator within proper limits. Superheater outlet steam temperature will be maintained constant by means of a temperature controller operating in connection with the butterfly control valves in the pulverized coal feed lines to the separately-fired superheater and also

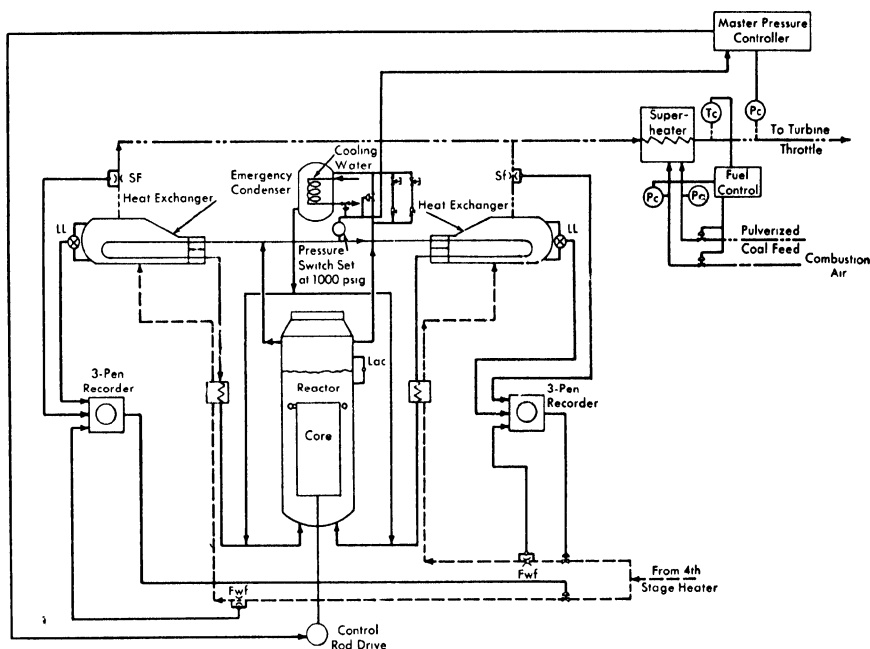


FIG. 10-13. Plant control system, Elk River.

the dampers in the combustion air lines. Loss of fuel will be announced in the control room but will not require automatic shutdown of the reactor. Loss of combustion air on the other hand would result in a dangerous condition and to avoid this the combustion air supply will be interlocked with the fuel control valves so that loss of combustion air will shut off the fuel supply to the superheater. Interlocks will be provided to purge the superheater furnace before coal is re-introduced into the furnace after a failure of the air supply.

Steam generating plant. The steam generator is designed to produce 225,000 lb of dry saturated steam per hr at a throttle pressure of 600 psig. The reactor will produce 285,000 lb/hr at about 875 psig. The steam generator is a two-shell, horizontal, closed condensing-boiler exchanger with condensing in the tubes. These are stainless steel, welded into stainless steel-clad tube sheets. A dry pipe is installed in the shell. A two-shell design provides extra plant reliability since the plant can operate at reduced capacity in the event of a tube failure in either shell. Access openings in the tube bundle headers, isolating valves, and biological shielding around each shell permit plugging a failed tube without shutting down the entire plant.

An alternate steam generator design is being considered and may be substituted in the final plant design. This consists of a single shell vertical heat exchanger with primary steam condensing in the shell and secondary steam generated in the tubes. The vapor-water mixture from this generator will be discharged into a separate horizontal steam drum containing a battery of cyclone separators. This vertical design has the advantage of more positive drainage of the primary condensate and a lower pressure drop, thereby facilitating natural circulation of the primary fluid. It also permits the inclusion of means for the separation and removal of dissociated gases carried over from the reactor with the steam. The single shell design, however, does not provide the flexibility of the two-shell unit but special provision for extra reliability is probably not warranted due to the relatively low temperature of the heating fluid.

A drain cooler is provided to subcool the condensate leaving the steam generator to about 450 F before re-entering the reactor. Cooling is provided by the evaporator feedwater of the secondary system, the feed being heated to about 50 F below its saturation temperature.

Another element of the steam system is a recombiner to recombine the hydrogen and oxygen resulting from dissociation of the primary water in the reactor. The steam evolved by the recombination of these gases and the steam carried over with the gases from the steam generator will be condensed in a small condenser and returned to the primary system.

The superheater, designed initially for use of pulverized coal, will receive essentially dry and saturated steam from the steam generator and superheat it for delivery to the turbine throttle at 825 F.

An emergency condenser is to be installed to receive steam from the reactor through a suitable relief valve system in case the reactor pressure reaches a predetermined value due to a sudden dropping of load or failure of the controls. This condenser will be large enough to condense the entire steam output of the reactor. This unit may also be used as a heat sink for testing the reactor at higher than the design output. Based on EBWR performance it is believed that such tests will prove the reactor to be capable of operating continuously at 80 to 100 Mw electrical.

HUMBOLDT BAY ATOMIC UNIT No. 3

The latest boiling water reactor to be proposed is for the Humboldt Bay plant of the Pacific Gas and Electric Co. It is of interest to view the construction of this plant against the background of the Pacific Gas and Electric system because this nuclear unit will occupy a place in the system that was originally intended as a conventional fuel-fired plant. Also, this is the first United States nuclear power plant chosen for economic

reasons. The Humboldt Bay plant is located near Eureka in the extreme northwestern corner of the Pacific Gas and Electric Co. system, isolated by mountains from hydro plants to the east, and the principal north and south transmission lines. Additional electrical generating capacity will be needed in this area by 1962. This could be provided by (1) constructing a 115-mile transmission line at a cost of about \$8 million for the line plus another \$8 million for steam generating capacity in large system units located in the San Francisco Bay region, or (2) a conventional 60 Mw steam unit at the existing Humboldt Bay Station at a cost of about \$11 million, or (3) a 50 to 60 Mw nuclear unit at Humboldt Bay at a capital cost of about \$20 million. The choice, economically, lies between the conventional 60 Mw steam and the nuclear unit at Humboldt; but to determine which is most economical requires consideration of the comparative fuel, operating, and maintenance costs.

The cost of the first nuclear core, including all materials and fabricating costs except the fuel use charge, will not exceed \$3 million. The fuel elements will be good for a life of 10,000 Mwd per ton of uranium. Any element failing to reach this exposure level will be replaced at a cost prorated on the basis of its actual useful life. The fuel will be enriched to permit achieving an exposure of 15,000 Mwd per ton, and it is expected that such an average exposure will be realized.

The second core is estimated at this time to cost under \$2 million and is expected to achieve an exposure of 15,000 Mwd per ton.

Experience in the operation and maintenance of the Vallecitos plant has led the company engineers to believe that the nuclear unit at Humboldt Bay Station will have substantially the same operating and maintenance expense, exclusive of fuel, as a conventional unit. This represents a substantial reduction in estimated cost compared to earlier studies.

Based on the capital cost previously mentioned, and expected operating and maintenance expense, it is estimated that with the second nuclear core and the current credit for plutonium produced in the reactor, power can be produced at a cost of about 8 mil/kwh with the unit operating at 90 per cent capacity factor, and having a capability of 60,000 kw.

The 90 per cent capability factor is believed proper because the nuclear unit should produce for the Eureka area and for the system all the kwh of which it is capable, since the nuclear fuel is less than half that required to generate the same number of kilowatt hours from conventional plants.

System load studies indicate that the nuclear unit operating in the manner described would reduce conventional annual steam generation in the Humboldt Bay and system steam plants located in the San Francisco area by approximately 470 million kwh annually. The expected saving in system fuel cost by generating these kwh with the nuclear unit will

offset the annual charges on the additional capital cost of that unit, plus nuclear insurance, thereby producing economically competitive power for this location.

Integration of nuclear unit with existing plant. The Humboldt Bay plant is located on the mainland shore of Humboldt Bay, about 4 miles south of Eureka, which has a population of about 30,000. Arcata, to the north, with a population of about 4000, is the only other center with more than 2500 people within 50 miles of the site.

An important reduction in capital cost is obtained by installing the nuclear unit in the existing Humboldt Bay plant where a 50,000-kw conventional outdoor unit is now in service and a second similar unit is under construction. This second unit is scheduled for completion late in 1958. The installation of a third conventional unit would result in a greater saving because in the case of the nuclear unit, additional property insurance is required for units 1 and 2. The advantage, however, is still in favor of using this site for the nuclear unit.

General features of the nuclear unit. The relative locations of the generating units in the Humboldt Bay plant are shown in Fig. 10-14 with the proposed nuclear unit at the right. Perhaps of much greater significance than the initial saving in cost, so far as progress in the application of nuclear power is concerned, is the installation of a nuclear power unit side by side with conventional units. This should be of significant value in comparing the operation of these two types of power plants.

Humboldt Atomic Unit No. 3, according to the designers, represents a continuation of the determined attack, begun at Vallecitos, on unit power costs, through further simplification of equipment, plant design, and operating requirements.

Simplification is achieved without compromising safety or reliability. It is made possible by reducing presently high margins in design and operation, as a result of the favorable experience at Argonne and Vallecitos, along with incorporating, wherever practicable, advances that have been made in conventional power plant design.

Humboldt Atomic Unit No. 3 represents an enlarged, advanced Vallecitos rather than a scaled-down Dresden. It is a design blending of the expected low fuel cost of Dresden with the low capital cost of Vallecitos.

The reactor will use natural circulation with complete internal steam separation and drying, and will have the largest steam flow using the single cycle principle of any reactor design for generating power. The single cycle reduces the complexity of the plant in comparison to the dual cycle by eliminating the secondary heat exchangers with their required

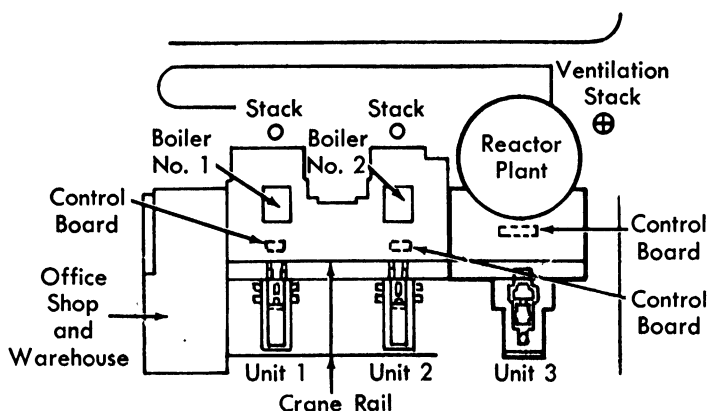


FIG. 10-14. Humboldt Bay power plant plot plan.

circulating pumps. Rod type, Zircaloy clad, oxide fuel elements will be used. The 3600-rpm outdoor turbine will have a throttle pressure of 1000 psi, saturated, and will be served by the existing gantry crane. Two stages of feedwater heating will be used. These features permit simplification, not only in the reactor design, but in the plant layout as well.

The designers of this nuclear unit emphasize that the Humboldt conception came into being as a result of a reappraisal of the costs of generating power by nuclear energy in the light of present knowledge and as applied to a medium size generating unit required for commercial power in 1962. Scientific advancement in this case was not a prime requisite any more than it would be with a conventional unit. The basic requirement was that it should produce economical power for the load growth of the Eureka area as would have been the case had a conventional steam unit been selected. The aim of Humboldt Bay project is to produce competitive power. This demonstrates that economical generation of electricity by nuclear energy is further along than many had realized.

NORWAY'S BOILING REACTOR AT HALDEN

Nearly all the work done in the development of the boiling water reactor has been done in the United States—much of it by Argonne National Laboratory. For some time, however, Norway has been working on the design of a boiling reactor at Halden, Norway, and this plant is now under construction.⁴ This is an experimental unit with a capacity of 20 Mw (thermal), but when it goes critical later in 1958 it will be the world's first reactor to use its heat output to supply an industrial plant. The boiling reactor will deliver 482 F, 588-psi steam to a paper and pulp mill.

The primary function of the reactor, however, will be to serve as an experimental facility, largely in the development of a reactor type suitable for ship propulsion. Thus nuclear steam will be supplied on an intermittent basis depending upon the experimental program. Operating under these conditions, the Halden plant will not be able to pay its way as a steam reactor; however, there will be an opportunity to collect cost data and operating experience.

The core of the Halden Reactor will be contained in a stainless-steel-lined mild steel pressure vessel of 8 ft, 10 in. internal diameter and 13 ft, 2 in. in cylindrical internal height. The tank itself is of conventional shape with a curved bottom and a flange ring at the top. The lid, however, is highly unconventional for the pressure and diameter in question. Owing to the necessity of having all fuel elements similar, conventional design has been replaced by a lid consisting of a 2 ft, 4 in. flat steel slab, bolted directly to the flange ring. Some 26 holes for steam outlet and condensate return pipes and 335 holes for fuel elements and control rods have been bored in the top lid. The tank and lid will probably be nickel-plated prior to installation to prevent corrosion.

The reactor will use cadmium control rods. All control rods are exactly alike in design and can be used either as seram, shim, or fine control rods depending upon the motor drives and electrical system to which the individual control rod is connected.

The control of this reactor is built around a graphic panel which is an almost complete flow diagram with valve and pump annunciators and indicators for the most important process variables.

The reactor is expected to go critical early in the summer of 1958.

REFERENCES

1. C. C. WHELCHER, Vallecitos and Humboldt Bay Power Plants. Paper presented at the American Power Conference, Chicago, Illinois, March 26, 1958.
2. C. B. GRAHAM, Controlled Recirculation Boiling Reactor. Paper presented at the 1958 Nuclear Congress, Chicago, Illinois, March 17-21, 1958.
3. Nuclear Products—Erco, Division of ACF Industries, proposal to the Atomic Energy Commission.
4. Norway Installing Halden Components, *Nucleonics*, 16(1), p. 21 (January 1958).

CHAPTER 11

CURRENT AND FUTURE RESEARCH AND DEVELOPMENT PROGRAMS

The successful operation of the BORAX reactors, the Experimental Boiling Water Reactor, and the G. E. Vallecitos plant has demonstrated the feasibility of the boiling process for power production. These reactors, though designed and operated with a high degree of conservatism, have indicated an approach which seems to yield greater economy than any of the other present reactor power plants. Admittedly, problems still exist which must be solved before boiling reactors can become economically competitive central station power plants, but the potential of natural circulation reactors appears very good and less limited than has generally been believed. The *art* of boiling reactor design is considerably more complex and less established than that of pressurized water reactors. Further experimental work is necessary to determine the true capabilities of the boiling water reactor concept. Some of the problems that remain and the experimental work in progress will be considered in this concluding chapter.

SAFETY TEST PROGRAMS

One of the more important problems that remain to be investigated is that relating to reactor safety. The large number of proposed reactors (boiling water and other types) increases the opportunity for an accident and the move from rural to urban areas increases the penalty of such an accident. Since it is likely that a serious accident might set back the nuclear-power industry for many years, it must be avoided by maintaining the admirable record of safety that has characterized the preceding years.

The BORAX experiments focused attention on the underlying problems of reactor transients and served as a starting point for the continuation and extension of experimental studies of this kind. The SPERT (Special Power Excursion Reactor Tests) program, operated by Phillips Petroleum Company at the National Reactor Testing Station in Idaho for the Atomic Energy Commission, was initiated partly to answer new questions raised by the BORAX work.^{1,2,3}

A runaway accident can be initiated in numerous ways, but each of these can be characterized as a reactivity disturbance under a particular set of initial conditions of the reactor system. The disturbance must be further specified by its magnitude and speed of occurrence. This brings into consideration the various means of reactivity introduction: malfunctioning of the reactor, accident damage, sabotage, operator error, and even inherent instability. The means of introduction of the disturbance and its time-scale, magnitude, and probability of occurrence are intimately related to the design details of the individual reactor plant.

Following the reactivity disturbance, events are determined by the response characteristics of the reactor system which are a combination of the largely non-nuclear characteristics of appendages like the control system and inherent characteristics which are strongly nuclear. In the most extreme accidents the inherent characteristics will fully determine the results.

Finally, additional problems arise such as the containment of explosions and factors affecting the dispersal of fission products.

Reactor response is of major importance in reactor safety, and the reactor kinetics program is concerned with the study of factors influencing reactor response.

Four types of experiments have been made in the kinetics program:

1. Step transients
2. Ramp transients
3. Stability tests
4. Static measurements

Ramp and step studies are concerned mainly with accident behavior with the ramp transient more representative of conditions surrounding an actual accident. The step transient is more simple to interpret and more representative of an extreme accident. Stability is an ever present problem rather specially related to the behavior about an equilibrium power under normal operating conditions; however, its role as a source of major reactivity disturbances needs investigation. Static measurements are very important as part of the calibrations for the experiments and because these basic qualities are required in the direct interpretation of the experiments.

The SPERT unpressurized work has included experimentations in step and ramp transients. Stability studies were made of the conditions for onset of divergent power oscillations as functions of reactivity in the system, temperature and water-head. As a part of the static measurements series, a detailed investigation of void coefficients was carried out.

The work to date has been done with the SPERT-I Reactor Facility but other SPERT reactors and other cores are planned for the continuation of these important experimentations.^{2,3}

SPERT-I consists of an open vessel into which has been placed initially a plate-type, enriched uranium-aluminum core. The reactor is water-moderated and -reflected, with external heating supplied for operation up to boiling. No provisions for heat removal or coolant circulation through the core are included. It is a minimum facility providing for a limited adjustment of water temperature and hydrostatic head. The reactor construction is simple and readily modified. The core is easily accessible for static measurements, core modifications and instrument installation, within the limits of individual core interior dimensions. SPERT-I and the experiments performed upon it will be described in greater detail later in this chapter.

SPERT-II is an intermediate pressure facility providing considerable flexibility in pressure, temperature, and flow. The system is designed primarily to permit selection of various moderator and reflector materials, principally light and heavy water. Both up and down controlled flow is provided for. The facility is relatively complex and much less flexible than SPERT-I in regard to design modification, static measurements, and instrument installation.

SPERT-III provides for a wide range of temperature, pressure and flow, plus heat removal capacity for power operation. Because of the complexities introduced by high pressure and high temperature, it is by far the least flexible with regard to either plant modification or experimental work.

SPERT-IV is in the proposal stage and would provide shelter and facilities for construction and testing of small reactor prototypes.

SPERT-II will be designed to operate at pressures up to 300 psi and temperatures up to 400 F, whereas SPERT-III will operate at pressures up to 2500 psi and temperatures up to 670 F. In addition SPERT-III will have coolant flows up to 20,000 gpm.

There are at this time six reactor core types planned:

1. Al plate
2. Al plate, adjustable spacing
3. Stainless steel
4. Zirconium
5. Al, low concentration
6. Mixed combination of 1 and 4

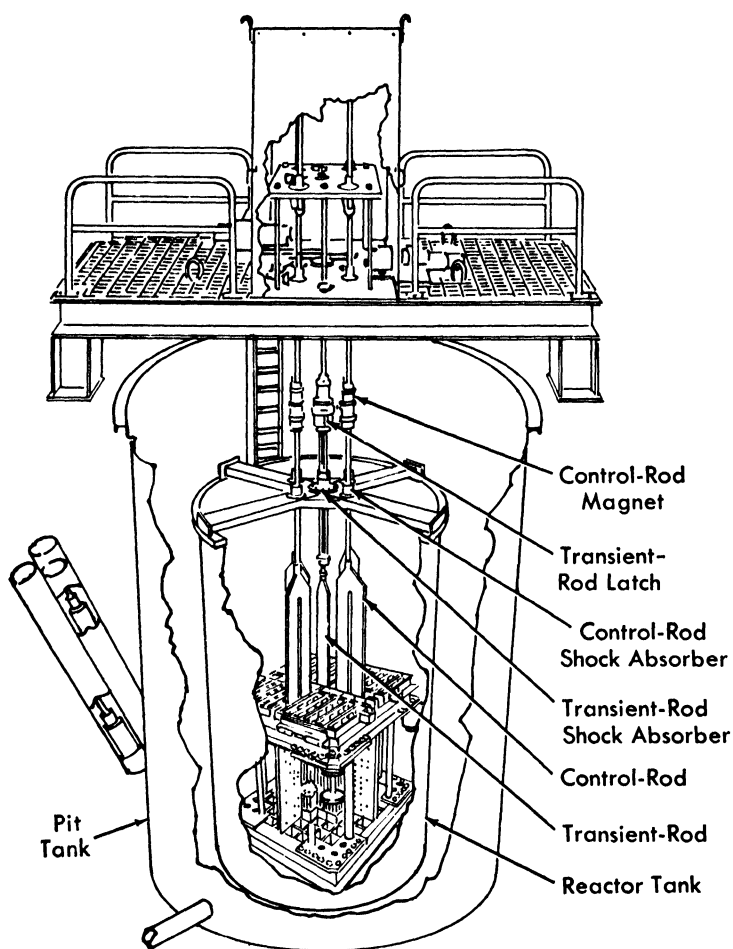


FIG. 11-1. Pictorial cutaway of SPERT-I reactor.

SPERT-I

As the starting point for the reactor transient behavior and safety studies, every effort was made to provide a remotely-operated, economical reactor facility of the very simplest design. The SPERT-I reactor is a non-pressurized, light-water-moderated and -reflected reactor using highly enriched fuel in modified MTR-type elements. Fig. 11-1 is a cutaway drawing of the reactor.

The reactor building is an uninsulated, 24 ft by 18 ft galvanized iron

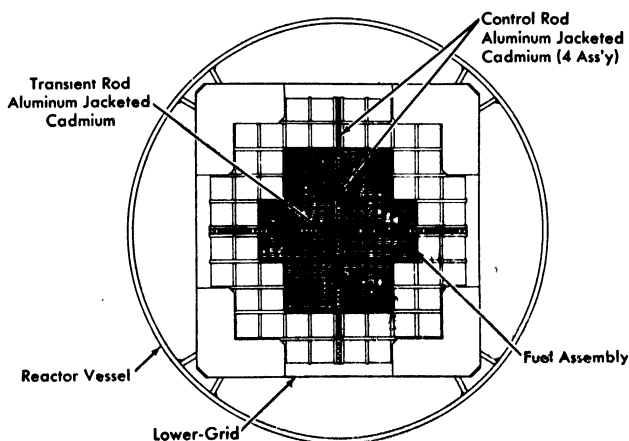


FIG. 11-2. SPERT-I lattice, 28 assemblies.

structure which houses the reactor and associated equipment, electrical switchgear and other auxiliary facilities.

The reactor vessel is situated in a 10 ft diameter by 16 ft deep pit below grade with the mechanisms for driving the control rods supported on a bridge structure spanning the pit, as shown in Fig. 11-1. The pit is equipped with a drain and sump pump for automatic removal of waste water to a leaching pond outside the building. Adjacent to the reactor pit, and also embedded in the building floor, are 18 6-in. diameter by 14 ft long tubes for the temporary storage of fuel or other radioactive materials. The initial core is composed of plate type fuel assemblies whose plates are fabricated of 93.5 per cent enriched uranium-aluminum alloy clad with aluminum (Fig. 11-2).

Four cadmium blade-type control rods function as a group for reactor operation; a similar, central control rod is used to initiate excursions. The control-rod group has a continuously variable speed adjustment on withdrawals up to 10 in. per min; this is about 0.1 per cent (or 13¢ sec) in reactivity. The poison section of the shim rods is a 4.75 in. wide by 0.020 in. thick cadmium sheet clad with aluminum. The poison section is located in the upper section of the blades; therefore, in the normal rest position of the rods (down), the cadmium is fully inserted in the active lattice. The fifth rod, referred to as the transient rod, is located in the geometrical center of the reactor. This rod, which in its normal position of rest has the cadmium portion outside and below the active section, is lifted upward to bring poison into the reactor. Transients are initiated by releasing the suspended rod and driving it downward out of the reactor to remove the poison. The coupling between this rod

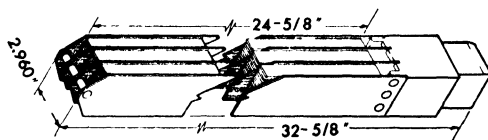


FIG. 11-3. SPERT-I fuel assembly.

and its drive is mechanical. The release is effected by action of small air pistons, and an acceleration of about 10 g is supplied by another air cylinder. The travel time varies between 80 and 120 milliseconds depending on the release position, which is determined by the amount of reactivity to be injected.

The four shim, or control, rods are driven by a single motor. A second motor is used to drive the transient rod. A total of about 25 in. is permissible. Both drive systems consist of a $\frac{1}{2}$ -hp, 480-v induction motor driving a variable-speed transmission. The output speed of the transmission may be remotely controlled from the control building. Chain and sprocket drives acting on a ball nut and screw permit control rod withdrawal rates up to 10 in. per min but rates to 35 in. per min have been achieved by changing the size of the drive sprocket.

The shim rods are coupled to the drive rods by means of magnetic clutches. Interruption of the magnet current permits the shim rods to be released and poison inserted in the core. For the first two inches of their travel downward the shim rods are accelerated by means of small air pistons and plungers. With 65 psi air pressure acting on the piston, a force of 260 lb is applied to the shim rods. The measured average rod drop time in water from interruption of the magnet current to the rod seat (24 in.) is about 150 milliseconds.

The fuel assemblies (Fig. 11-3) consist of an end box, two aluminum side plates, two stiffener plates, a lighting pin, and 51 fuel plates. Although it was desired to keep the reactor system similar to BORAX for ease of data comparison, the fuel assemblies for SPERT-I had to be strengthened because part of the testing program involved tests in the period region between the fastest nondestructive test and the destructive test performed in the BORAX experiments. The stiffening members, which effectively separate the assembly into three sub-assemblies, decreased water-channel width by a factor of three.

Relief holes are provided in the square end boxes on the bottom end of the assemblies. These and the bottom-grid design provide for some coupling of the water channels between assemblies with other channels. The corresponding channel was blocked in the BORAX experiments. These changes should be most apparent in those factors that are greatly

TABLE 11-1
CHARACTERISTICS OF SPERT-I CORE

Critical mass (23 fuel assemblies, 18 C)	3.86 kg U ²³⁵
Excess reactivity with 28 fuel assemblies	4% k_{eff}
Prompt neutron lifetime	$50 \pm 5 \mu\text{sec}$
Reactivity loss from 20 to 95 C (boiling)	1.05% k_{eff}
Temperature coefficient at 20 C	$-0.9 \times 10^{-2}\% \Delta k_{\text{eff}}/^{\circ}\text{C}$
Temperature coefficient at 95 C (boiling)	$-2.0 \times 10^{-2}\% \Delta k_{\text{eff}}/^{\circ}\text{C}$
Average void coefficient for core	$-3.5 \times 10^{-4}\% \Delta k_{\text{eff}}/\text{cm}^3 \text{ void}$
Maximum void coefficient (center of core)	$-7.2 \times 10^{-4}\% \Delta k_{\text{eff}}/\text{cm}^3 \text{ void}$
Reflector void coefficient (adjacent to core)	$-0.69 \times 10^{-4}\% \Delta k_{\text{eff}}/\text{cm}^3 \text{ void}$
Volume of water in the core	$54 \times 10^3 \text{ cm}^3$

influenced by the hydrodynamical system of the nuclear-thermal-hydrodynamical complex. Presumably, the thermal characteristics of the plates were left unchanged. The lattice used for most of the transient tests consisted of 28 assemblies.

Each assembly of 51 fuel plates contains 168 grams of U²³⁵. The central portion of each plate consists of uranium-aluminum-alloy 0.020 in. thick and is clad on each side by 0.020 in. of aluminum making a total plate thickness of 0.060 in. These plates are separated by 0.117-in. water channels. The reactor was just critical at ambient temperature with 23 fuel assemblies loaded into the core (3.86 kg of U²³⁵) and with all rods completely out of the core. The loading of 28 assemblies resulted in a total excess reactivity of about 4 per cent Δk at 18 C, as determined by a series of shim rod worth measurements which were made by uniform poisoning methods. Characteristics of the SPERT-I reactor core are shown in Table 11-1.

The initial experimental step is to determine the control bank's critical position, which varies with changes in such things as water purity or temperature. The critical position is measured at a power level of about 200 watts with the poison section of the transient rod entirely out of the reactor. The system is made subcritical and its power reduced by pulling the transient rod upward into the core in preparation for starting a transient. During the decay of the power to a level of several watts, the control bank is withdrawn to its position for the transient. Ejection of the transient rod will then make the reactor supercritical by the desired amount. When the operator decides to proceed with the transient tests, he arms the center-rod ejection system. At any time thereafter the transient sequence can be started by activating the transient sequence timer. According to a preset schedule, the timer starts

cameras, recording equipment, etc., initiates the ejection of the center rod, and at the desired time scrams the control rods and shuts off equipment. Tests can be run as frequently as once every ten minutes, but the average time is about an hour per test. Fig. 11-4 shows diagrammatically the sequence of events in a typical transient test.

Meters in the control room indicate the gamma-ray level near the reactor. Thus, it is not necessary to approach the area to determine when safe entry can be made. Personnel working in the reactor area are always equipped with portable health physics instruments. This is especially important immediately after reactor operation because the radiation levels vary enormously within the area. For example, permissible working times may be as short as minutes at the edge of the pit. Ordinarily, the pit building can be re-entered 10 to 20 minutes after the test.

Usually the oscillograph charts are processed immediately after a test and are available for an initial measurement of the periods, peak power, and peak temperature within 10 minutes. At a later time a Benson-Lehner data handling system is used to read and replot in real units traces such as appear in Figs. 11-5 and 11-6 which show some of the characteristics of typical reactor transients. The first part of the tests, which covers the power region from 2 watts to about 100 kw, is not shown. The reactor temperature initially was about 18 C. In both cases the power rise was essentially exponential for the last several decades. Most of the shutdown effects appear in the neighborhood of the power peak. In the 100-msec case, the damping of the system returns the power gradually to a stable value without any undershoots in the period of observation. The 13-msec case exhibits a large undershoot, but the second rise in power also appears strongly damped and does not show additional oscillation.

Comparison of the two transients also shows a definite change in the rate at which the shutdown takes place relative to the initial period. The long period run has a rising slope steeper than the shutdown slope. In contrast, the slope on the shutdown side of the short-period transient is greater than on the rising side. Stated another way, plots of power versus reactor period as the unit of time show shape changes after the power peak for differential initial reactor periods.

SPERT-I EXPERIMENTS

Scope of experiments. The nondestructive BORAX-I and II transient work covered the period region between 100 msec and 5 msec at boiling temperatures, and between 100 msec and 13 msec subcooled to

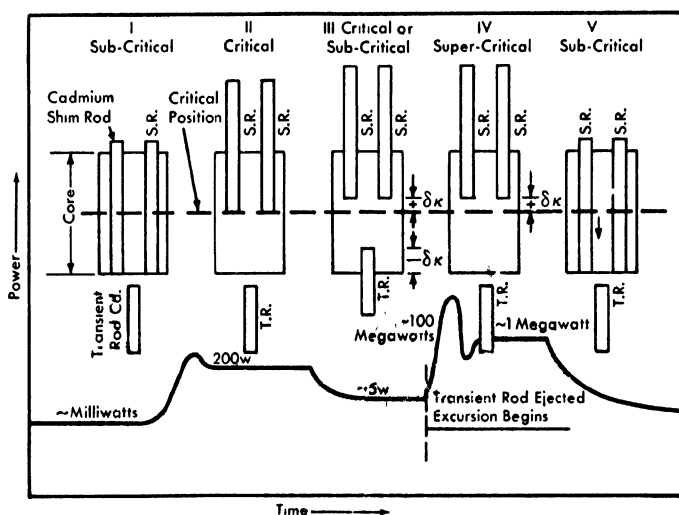


FIG. 11-4. Transient test sequence, SPERT-I reactor.

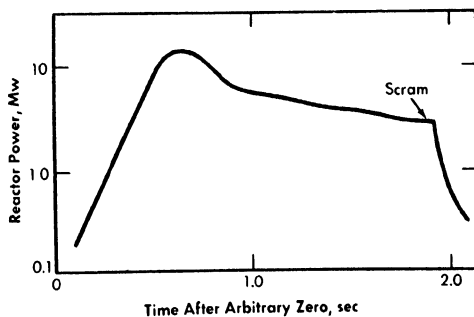


FIG. 11-5. SPERT-I transient test, 100-msec period core water initially 18 C.

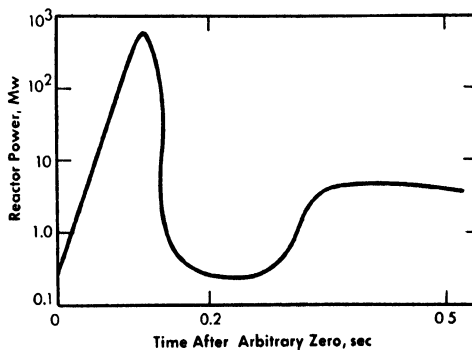


FIG. 11-6. SPERT-I transient test, 13-msec period core water initially at 18 C.

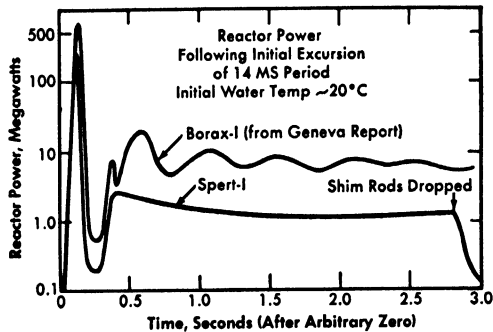


FIG. 11-7. Reactor power for BORAX-I and SPERT-I after 14-msec-period excursion. Initial water temperature about 20°C.

ambient temperatures. Burst behavior was explored as a function of temperature for a selected number of periods. In addition, one subcooled destructive test was run at a period of 2.6 msec.

A part of the SPERT effort was to repeat BORAX work (Fig. 11-7) and at the same time extend work into the region between destructive and nondestructive tests. The SPERT step transient tests have covered the regions from 10 sec to 5.5 msec subcooled to ambient, from 10 sec to 5 msec boiling, and from 10 sec to 10 msec subcooled to the intermediate temperature at 85°C. Ramp transient explorations have been conducted for a range of starting powers of a factor of 10^9 (i.e. from 10^{-4} watts to 10^{+5} watts), and for a range of reactivity injection rates from 0.01% $\Delta k/\text{sec}$ to 0.36% $\Delta k/\text{sec}$. Stability and static measurements were made as was stated earlier.^{4,5,6,7}

Boiling ramp tests. Severe instabilities were found in SPERT-I in the form of a series of diverging power oscillations — pulses whose amplitude continued to increase throughout the period of observations with no apparent tendency to approach a limiting value. The instability was observed in an experiment where an attempt was made to insert 2% excess reactivity at a linear rate of approximately 0.08%/sec with the water at boiling temperature. The reactor was initially critical at a power level of 100 kw at which point approximately 0.15 per cent excess k was compensated by steam. As the rods were withdrawn the power rose more or less uniformly until about 1.6 per cent additional reactivity had been added. At this point divergent oscillations appeared and the reactor was scrammed manually when the added reactivity reached 1.95 per cent, giving a total of 2.1 per cent reactivity added to the system in excess of the low power critical condition.

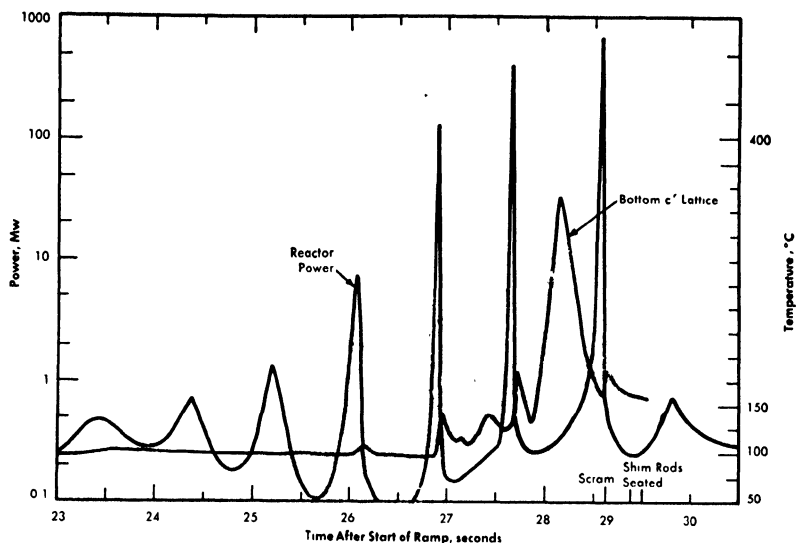


Fig. 11-8. Boiling ramp test, $1.9 \Delta k$. Reactivity was added at approx. 0.07% per sec from critical at 100 kw and terminated by shim rod scram at 28.58 sec for a total insertion of approx. 1.9%.

Examination of the oscillograph record (Fig. 11-8) revealed that the last power surge reached a peak of about 700 Mw and exhibited a minimum period of less than 10 msec. The power in this burst was rising more rapidly than an exponential and a rough analysis indicated that reactivity was increasing at a rate of the order of 10 per cent per sec during the burst. The rate of growth of the successive power surges indicated that one or two more pulses would probably have exceeded the safe limits for the reactor.

These results focused attention on the necessity for examining long term stability in the reactor in addition to the initial transient behavior. Such investigations had been postponed because of the radiation problems associated with long-run-out time required. In view of the violent instability observed, it was decided to alter the program and experimental procedure to investigate the problem more fully at this time.

The procedure was therefore changed so that the ramp addition of reactivity was terminated at a preselected reactivity addition, but the reactor was to be allowed to run for an additional few minutes unless scrammed manually in the interest of safety. All tests were started from boiling at the 5 w level and the ramp rate was approximately 0.08 per cent per sec. The total reactivity addition was 1 per cent in the

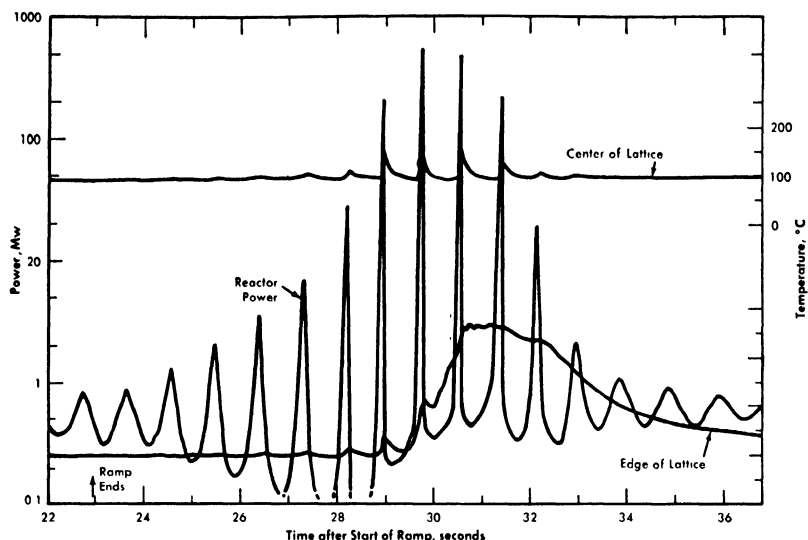


FIG. 11-9. Boiling ramp test, 1.8% Δk . Reactivity was added at approx. 0.08% per sec from critical at approx. 5 w and terminated after 22.94 sec for a total insertion of approx 1.8%.

first test and was increased in 0.25 per cent increments in the following tests until instability was observed at 1.75 per cent total addition. At this point the reactor oscillations increased in amplitude until a considerable quantity of water was ejected from the tank. Thereafter, the oscillations converged and the power approached a steady value with only minor fluctuations (Fig. 11-9). It was found that the water head above the core had been reduced to one foot from the initial value of two feet. This suggested that the stability was strongly dependent on the hydrodynamics. To check this point the water head was increased to four feet above the core and the rod withdrawal adjusted for a total reactivity addition of 1.5 per cent since a similar test at two feet head had exhibited stable behavior (Fig. 11-10). The increased head resulted in divergent oscillations (Fig. 11-11), and the reactor was scrammed to avoid damage.

Following the boiling tests the long term stability of the reactor at room temperature was investigated. The procedure was essentially the same as that for the boiling ramps with the exception that the effectiveness of the rods was somewhat greater due to the lower critical position. The resulting ramp rate was approximately 0.09 per cent per sec. Tests were made in which the total added reactivity was 1%, 1.5%, 2%, 2.25% and finally 2.5%. All but the last run exhibited stable behavior with

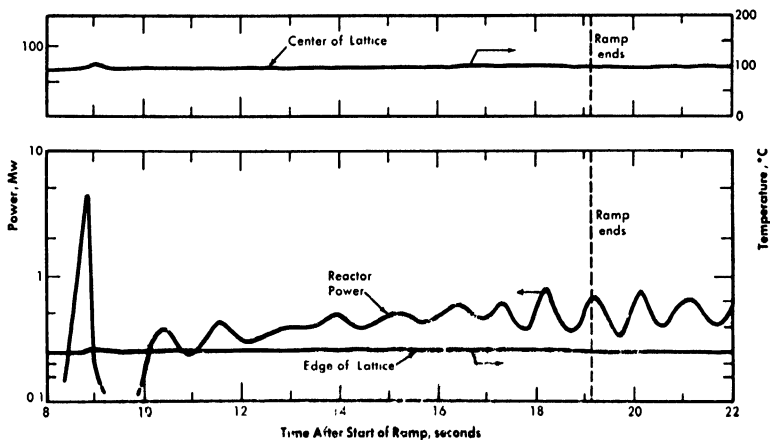


FIG. 11-10. Boiling ramp test, $1.5\% \Delta k$. Reactivity was added at approx. 0.08% per sec from critical at approx. 5 w and terminated after 19.15 sec for a total insertion of approx. 1.5% .

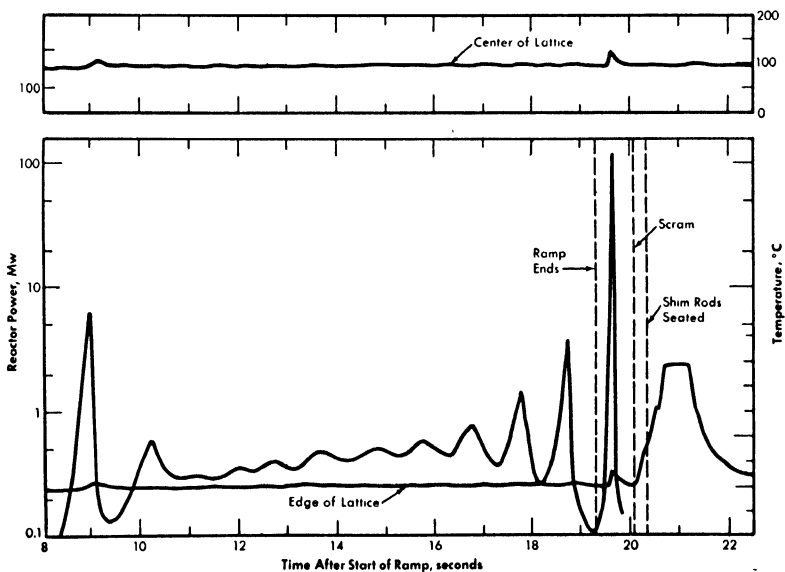


FIG. 11-11. Boiling ramp test, $1.5\% \Delta k$ (high head). Reactivity was added at approx. 0.08% per sec from critical at approx. 5 w and terminated after 19.25 sec for a total insertion of approx. 1.5% .

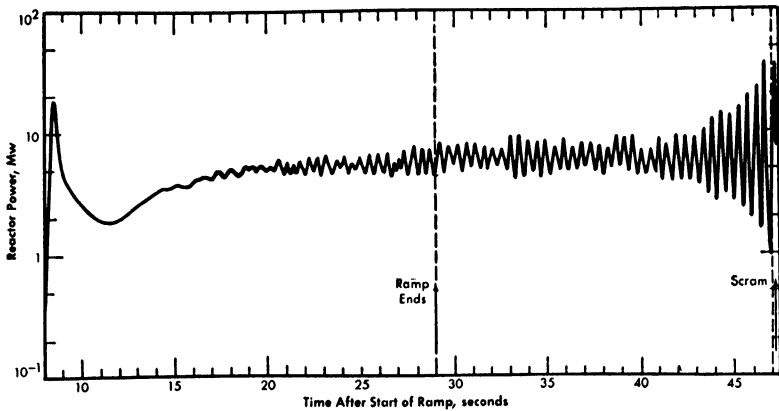


FIG. 11-12. Subcooled ramp test, 2.5% Δk . Reactivity was added at approx. 0.09% per sec from critical at approx. 5 w and terminated after 29.2 sec for a total insertion of approx. 2.5.

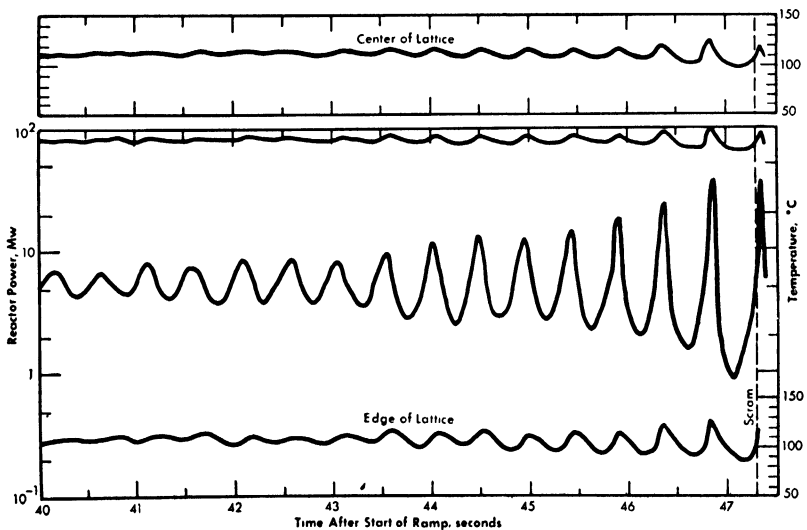


FIG. 11-13. Divergent portion of preceding test (Fig. 11-12).

only mild oscillations of a non-divergent nature. The head was two feet in all tests except one of 2.25% reactivity addition at four feet head. The 2.5% ran for approximately 40 sec in a stable manner and then the oscillations began to diverge (Fig. 11-12). The rate of divergence was moderate (Fig. 11-13) giving ample time for observation before a scram became necessary.

The following observations can be made:

1. Under certain conditions the reactor becomes extremely unstable resulting in large power peaks having very short effective periods. The rate at which reactivity increases in the system under these conditions is large compared to the rate at which it was originally added to the system and is actually comparable to the injection rates attainable with the transient rod used in the stop tests. In other words, the reactor has an intrinsic mechanism for rapid injection of large amounts of reactivity.

2. The maximum reactivity which the SPERT-I reactor can compensate stably in the boiling condition with a two feet water head is somewhat less than 1.75%. The instability is quite sensitive to the hydrodynamic situation. The safe limit is less than 1.5% for a four foot water head.

3. The ability of the core to survive these tests stems in part from the fact that the more violent surges expel water permanently from the core, a condition which is not necessarily typical of many research reactors.

4. A reactor in which a large amount of reactivity is held in voids may be hazardous if there is any mechanism by means of which void disappearance may occur suddenly. (An example of this mechanism is the splashing of water in a boiling reactor as described for SPERT-I.)

For the subcooled condition the maximum reactivity of stability in SPERT-I at room temperature (approximately 2.5%) is greater than for boiling. Also when oscillations occur they diverge less rapidly than those at boiling. Stability appears to be less sensitive to the hydrodynamic situation.

Thus, while the existence of severe instability in the SPERT-I reactor is certainly established, these preliminary tests give rise to many questions regarding the causes that remain to be answered. While the oscillations occur, the conditions under which they can be excited have not been fully determined.

Conclusions:

(1) The extension of the subcooled work from the 13 msec region covered by BORAX to the 5.5 msec region has now covered the major part of the unexplored zone between 13 and >2.6 msec. Now the division between the unsafe and disastrous zones has been located with considerably greater precision. It has been narrowed down to the period-range 5.5 to >2.6 msec. It has been predicted that it begins about 4 msec.

(2) Extrapolation of the curves of the measured quantities to the 2.6 msec region becomes more accurate. On this basis it seems unlikely that there were any discontinuities in the BORAX curves of the energy released, the temperatures developed, or the peak powers reached. The discontinuity in the pressure that was developed in the BORAX destructive test, which would not be predicted from the SPERT data, is probably due to the simple steam explosion brought about by melting of the fuel plates.

(3) An increase in water head or water temperature lowered the reactivity-compensation level at which oscillations diverged; that is, it decreased stability. Or in other words, instability appears to be aggravated by operation near boiling and with a large head.

The SPERT experiments have also resulted in certain other consequences of general application to all reactor systems:

(1) The reactivity-compensation necessary to turn a burst is sensibly less than the injected reactivity for an important class of tests. This rate is influenced very strongly by the shape of the burst. The practical importance of this conclusion is that, for incidents involving reactivity injections up to the region of control by prompt neutrons alone, the emergency control of a reactor does not require a fast acting, strong control. A fast acting, small control will produce the needed delay in the burst to permit the slower insertion of the strong control. This advantage is limited to reactivity additions not greatly in excess of prompt critical. This simplified view does not adequately describe reactors which can be autocatalytic for part of the burst.

(2) Severity of the ramp-produced burst is strongly dependent on ramp rate and weakly dependent on initial power over a wide range. The results imply first that in many cases, the source strength is unimportant and that only a very weak source is necessary, and second that the reactivity injection rate must be limited.

(3) Stability can become an important factor in a runaway initiation as it is possible that small initial reactivity disturbances characteristic of stability problems can lead to large reactivity changes involved in a runaway incident. The SPERT instabilities have shown that it is not adequate, for reactor safety, to show that reactivity cannot be introduced at the rate to produce an accident, but it must be shown that the reactor itself cannot do this.

(4) Static measurements of reactor properties are necessary, in detail, to interpret theories on transient behavior and safety evaluation, particularly such fundamental constants as effective delayed neutron fraction and prompt lifetime.

These extensive experiments performed upon SPERT-I with the first core have led to a detailed set of reference measurements for the particular combination of characteristics of this system. From this point, the effect of varying kinetic behavior, produced by varying individual parameters, can be studied in detail. The program upon the various SPERT reactors and cores includes investigations of variations of void and temperature coefficients, prompt neutron lifetime, thermal characteristics of fuel plates, fuel plate geometry, moderator hydrodynamics, pressure, temperature, and coolant flow. The SPERT program has already made a substantial contribution to reactor theory and practice, and there is every reason to believe that, with its continuing program, more valuable data upon reactor kinetics will be forthcoming.

FURTHER BOILING REACTOR INVESTIGATIONS

As part of an Advanced Water Reactor Development Program, Argonne National Laboratory has in progress a series of studies having as their objective the development of very large boiling reactors for central station use. Conceptual designs of heavy-water boiling reactors are being developed.

These investigations are still in their initial stages but the fact that nuclear superheating is involved in one of them is of great importance because if steam temperature of 850 F can be attained, such systems will more fully meet the steam requirements of modern central stations than the low temperature systems so far developed.

ADVANTAGES OF D_2O IN BOILING REACTORS

The use of heavy water also is important. Iskenderian, Link, Treshow and West have considered the use of it in boiling reactors.¹¹ Heavy water comes very close to meeting the technical specifications for an ideal moderator material. Of the materials commonly considered as moderators for nuclear reactors, the slowing down power of heavy water is second only to light water. The combination of good moderating power and low absorption cross section for thermal neutrons gives heavy water a moderating ratio (slowing-down power divided by absorption cross section) far greater than for any other material.

Heavy water also is the only moderator which can be seriously considered in a high performance power reactor using natural uranium. With reasonable care in the design to prevent excessive amounts of neutron-absorbing structural materials in the core, a large, natural uranium power reactor should be able to operate with a conversion ratio

near unity. This high conversion ratio assures maintenance of reactivity for the high fuel burn-up which is necessary for economic reasons.

The intermediate slowing-down power and low absorption cross section of D_2O encourage lattice designs in which the volume ratio of D_2O to uranium is very high. Common values of this ratio for D_2O and H_2O reactor cores are in the neighborhood of 20 and 2 respectively. The small volume of uranium makes the core design, fuel handling, and control of the D_2O reactor much simpler than for its H_2O counterpart. The small quantity of uranium and low absorption cross section of the moderator account for a prompt neutron lifetime more than 10 times as long in a D_2O moderated reactor as in the H_2O type. This is a very important factor from the standpoint of safety, inasmuch as short-period excursion need not be considered in D_2O reactors. Since a significant part of the capital expenditure for a nuclear power plant is usually associated with safety provisions, the relative safety of a D_2O moderated reactor is expected to have an important bearing on the cost of electricity.

Until recently the chief disadvantage of D_2O as a moderator and coolant has been its high cost. In recent years it has become evident that the cost of producing D_2O can be decreased markedly by mass-production methods. The present quotation for D_2O from large production plants in the United States is \$28 per lb. At this price and with reactor designs which minimize the D_2O investment per unit of heat output, the direct charges against the D_2O can be reduced to less than 1 mill per kwh of electricity produced. The cost of D_2O becomes significantly less than the cost of fuel in the D_2O reactor and very much less than the cost of fuel reactors using other moderators in conjunction with enriched uranium.

Direct-cycle boiling water reactor power plants are particularly well suited to use D_2O as a moderator and coolant. The only D_2O required outside of the reactor is a very small amount in the form of steam and condensate. In a pressurized water system the amount of D_2O required in the external pipes, steam generators, and pumps approximates that of the reactor itself.

The advantages of D_2O moderator are best realized in reactors of large physical size. To achieve criticality and operate at high temperature with natural uranium fuel, D_2O -moderated reactor cores must be 10 ft or more in diameter. When the reflector is taken into account, a pressure vessel having a diameter of at least 13 ft is required. For a boiling reactor generating 600 psi steam, the thickness of the pressure vessel wall would be about 4 in. By comparison, a pressurized water reactor operation at 2000 psi in order to generate 600 psi steam in exter-

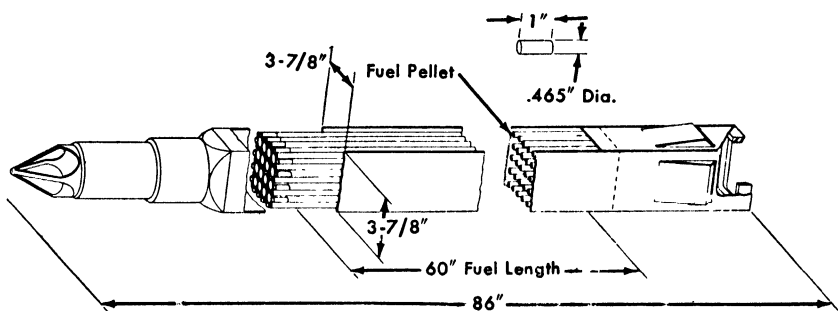


FIG. 11-14. Fuel assembly of EBWR-III core

nal steam generators, would have to have wall thicknesses of the pressure vessel so great that fabrication would be extremely difficult, if not impossible.

The choice of an optimum reactor design for power production depends upon the assumptions which must be made regarding the availability of slightly enriched uranium and on the amount of power desired per reactor. If only natural uranium is available, D_2O reactors are necessarily large, and H_2O reactors are excluded entirely. A D_2O natural-uranium reactor, which is large in physical size and capital investment must produce large quantities of heat if the unit cost of this heat is to be reasonable.

The detailed design of a large boiling D_2O reactor using natural-uranium fuel, capable of producing 1000 Mw of heat was described in a paper¹¹ for the 1955 Geneva Conference, but it is more likely that if a D_2O moderated boiling reactor of this size is built now, it will use the uranium oxide type of fuel. BORAX-IV was the first UO_2 -fueled boiling reactor, and by virtue of its high temperature characteristics and corrosion resistance, the use of this fuel for large reactors is certain to increase in the course of reactor development.

PLANS FOR 100 MW THERMAL EBWR OPERATION

The successful stable high power operation in EBWR with only 114 assemblies having a 4 ft active length has resulted in plans for filling the core with 148 elements having a 5 ft active length. The proposed 148-element core is expected to produce over 100 Mw of heat with no increase over the power density now reached at 60 Mw thermal. This exceeds the capacity of the present steam system and heat removal equipment will have to be added to make possible a full-scale demonstration.¹²

Fig. 11-14 is a view of the proposed design of a new fuel assembly.

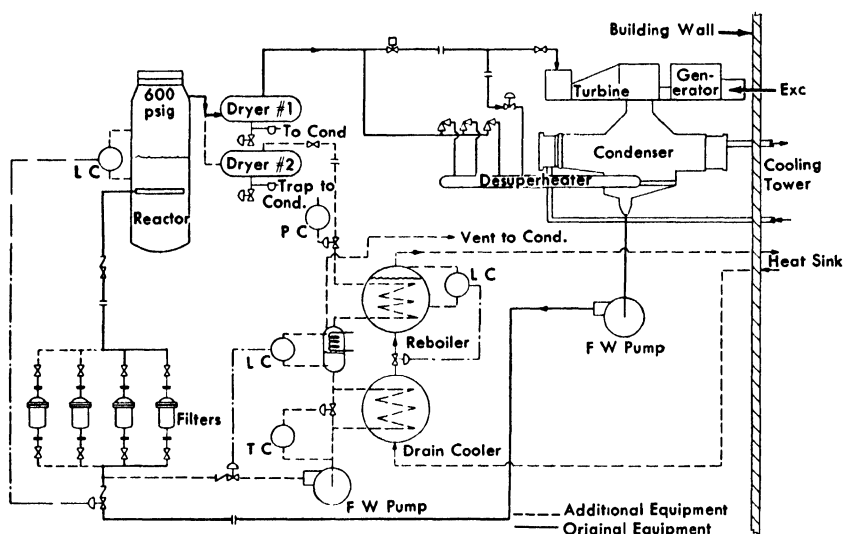


FIG. 11-15. System for removing 100 Mw, thermal, from the EBWR.

It is 86 in. long and $3\frac{7}{8}$ in. square. The components of each assembly are the tip, the bottom fuel end fitting, the side plates, 25 fueled tubes, spacer tubes and clips, and the top lift ring. The bottom end fitting, the tip, the side plates, and fuel tubes are all of stainless steel. The fuel tubes are 0.500 in OD with a 0.015 in. stainless steel wall. The fuel pellets are 0.465 in OD by 1 in. long, cold pressed and sintered UO_2 of low enrichment. The exact percentage of enrichment has not been established.

The present plan for the type and arrangement of equipment to be used for 100 Mw of heat output is shown in Figure 11-15. This is a schematic diagram and does not indicate the required number of pumps or heat exchangers, or piping changes. Space in the containment building is sufficient to allow the installation of heat exchangers, drain coolers, pumps, filters, etc. In no instance, will radioactive fluid escape the vapor container since only the tertiary fluid reached the heat sink. The only equipment outside the vapor container will be the heat sink, which may be used as a heating or process steam supply to the Laboratory. A cooling tower will be used in warm weather to provide loading flexibility for experimental work. Operation is planned for 1959.

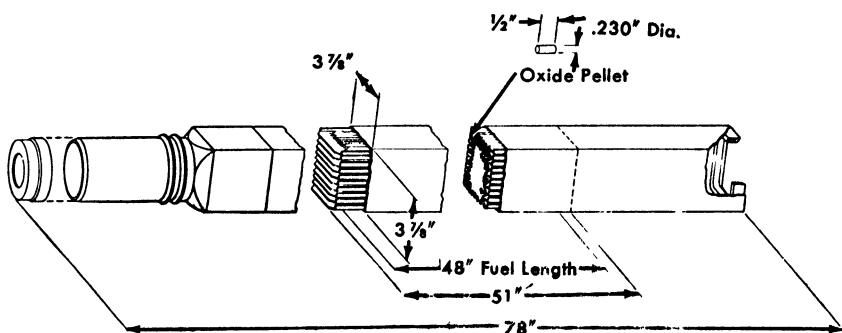


FIG. 11-16. Fuel assembly of EBWR-1F core using D_2O .

HEAVY WATER AND FORCED CIRCULATION CORE FOR EBWR

EBWR was designed for the eventual use of D_2O as moderator and coolant as explained in Chapter 5. The design for the new H_2O core for 100 Mw(t) operation referred to above has not yet been completed. In the interim, a test may be conducted with forced circulation using a urania-thoria core.¹² (Fig. 11-16)

In this design the fuel assemblies are 78 in. long and $3\frac{7}{8}$ in. square. The components of each assembly are the orifice tip, the bottom fuel end fitting, the side plates, 10 fueled plates, and the top lift ring. The orifice is machined from stainless steel. A variety of openings are supplied to permit variable orificing of the core. The bottom fuel end fitting is machined from a stainless steel casting.

The side plates are $\frac{1}{16}$ in. aluminum-nickel alloy X-8001. The fuel plates are extrusions of this same alloy and are similar to the BORAX-IV plates. The fuel pellets are 0.230 in. OD by $\frac{1}{2}$ in. long cold pressed and sintered ThO_2-VO_2 mixture. In fabricating these elements, the fuel plate is first sealed on one end. The fuel pellets are then placed into the holes provided in the extrusion. When filled with the proper amount of fuel the tubes are evacuated and the end sealed. The plate is hydraulically pressed to form a good mechanical contact between the aluminum clad and the fuel pellet.

The 10 fuel plates are welded to the subassembly just described, forming the completed box and the bottom fuel end fitting is riveted to the bottom of the box. Finally, the top lift ring is welded into the top of the box to complete the assembly.

TABLE 11-2
CAPITAL COST 30 Mw(E) NET ELECTRICAL PLANT

Buildings	\$ 300,000
Containment	1,000,000
Site development	400,000
Reactor system	2,000,000
Electrical	500,000
Turbogenerator and equipment	1,800,000
	\$6,000,000

30 Mw (ELECTRICAL) PACKAGE DESIGN

Recently drawings were made showing how a 33 Mw(e) turbine-generator could replace the 5 Mw(e) unit now located in the EBWR shell. Had the data on performance now at hand been available at the time EBWR was built, the plant could have accommodated this higher output unit. The containment shell and safety equipment, reactor vessel, storage well, and water treatment equipment are the same. For the larger plant output the turbine generator, steam and feedwater lines, steam dryer-emergency cooler, air ejectors, and condensers and pumps are changed. All this equipment would fit into the present containment shell.¹²

The cost of the new plant is estimated to be about \$6,000,000 based on single purpose, natural circulation H₂O design.

Allowing 3000 kw for the house auxiliary load, this plant can produce a new output of 30 Mw, electrical, at an installed cost of \$200 per kw. Capital cost breakdown in dollars is shown in Table 11-2.

This cost of \$200 per kw is comparable to the cost of coal-fired plants in the 30,000 kw output size. The operator of a power system can now build a nuclear power unit of this type, including all containment and safety provisions needed in the populated area, without the usual large investment of capital funds called for in more complicated nuclear plants.

Attractive as this figure may be, it does not mean that the cost of power produced will be economical or competitive with fossil fuels in any particular location or under any particular set of conditions. A supply of uranium enriched in U²³⁵ is needed. The original enrichment of the fuel in the EBWR reactor of 1.44 per cent U²³⁵ will not suffice for this plant. The cores will require an average of 2.5 to 3.5 per cent. The final operating cost will depend upon an economical fuel cycle.

TABLE 11-3
ESTIMATED COST FOR 33 Mw(e) EBWR TYPE PLANT

Cost basis	Yearly	Mills/kwh	Mills/kwh
21 × 10 ⁷ kwh/yr			
Enrichment, 3%			
Conversion ratio, 0.5			
Capital, \$6,000,000 at 15%	\$900,000	4.3	
Operation and maintenance, 20 persons at \$12,000/yr	240,000	1.1	
Subtotal capital cost			5.4
Fuel fabrication, \$640,000 at 15%	\$ 96,000	.45	
Uranium rental, \$3,500,000 at 4%	140,000	.67	
Annual burnup (10,000 Mwt day/ton)			
Pu credit at \$30/g	170,000	81 to 1.5	
or			
Pu credit at \$12/g	320,000		
Annual fabrication	225,000	1.1	
Subtotal fuel cost			3.03 to 3.72
Total cost			8.43 to 9.12

If conversion ratio is of no great importance the core may be loaded with burnable poison which would allow keeping one core in place for several years. An alternate scheme is the use of enriched fuel spikes, originally planned in EBWR but abandoned for the first loading in the interest of getting the plant in operation. Recycling of plutonium as an enrichment spike is definitely a possibility.

Fuel cost for 30 Mw(e) plant. It is estimated that the construction cost for the fuel assembly shown in Fig. 11-14 would be \$970 per assembly exclusive of fuel or \$140,000 per core. The fuel cost for a core is about \$3,400,000 including all inventory. If spiking is used to maintain reactivity the cost of enriched fuel for an assembly would be about \$15,000.

The number of spikes or slightly enriched fuel elements needed per year of operation is in doubt until a fuel cycle has been demonstrated. A complete fuel cycle study is not available but it is possible to propose the estimated cost breakdown shown in Table 11-3 for reference purposes, using the cost information available at this time.

The most reliable cost figures presented are those covering plant and

fuel fabrication. It is clear that unless a fuel cost of near 3 mills per kw hr is obtained, low cost power is not possible. A demonstration plant must be run by a cost conscious utility organization. The actual operating cost cannot be determined accurately if the plant is used extensively for experimental work, as is the case with the EBWR. The cost breakdown presented here is more of an objective rather than an actual proposed cost. Nevertheless it can be achieved if proper emphasis is placed on economical construction and fuel cycle.

BOILING REACTORS AND FUTURE NUCLEAR POWER DEVELOPMENT

While it is too early to speculate on the probable place the boiling water reactor will occupy in future nuclear power development, there are good reasons for believing that this type of reactor will be regarded with increasing favor by utility company engineers. The quantity and quality of boiling water technology is steadily growing and the doubt that boiling reactors could achieve stability of operation has been replaced by a state of confidence. There is every assurance that the remaining uncertainties will be resolved favorably. Compared to the other reactor concepts that are being tried out in the United States at the present time, the boiling reactor plant seems to offer both the lowest construction cost and the lowest net power cost. The estimated power cost range of the 33 Mw(e) EBWR type plant outlined above (8.43 to 9.12 mills per kwh) comes close enough to the cost of power from conventional power plants to make this size and type of plant worthy of further study and investigation. The estimated power costs of the Dresden (see Chapter 9) and Humboldt Bay (see Chapter 10) boiling reactor plants lend further support to the attractive possibilities of the boiling reactor in future nuclear power development, and the operation of these plants will be regarded with great interest to see whether the favorable estimated costs can be realized.

There is little question that the capital costs of all types of nuclear power plants will be reduced as experience is gained, but here again the boiling reactor appears to have advantages over other more complex types of nuclear plants because the boiling reactor plant more nearly resembles the conventional steam plant than any other nuclear plant thus far proposed. The operation of Vallecitos and the proposed operation of the Dresden Station at approximately the same pressure is within the range of pressures used by many modern steam stations and this requires no special types of pumps or piping or other auxiliary equipment. The use of oxide fuels will further advance the art of boiling water reactor design and may make possible the development of nuclear superheating. Nuclear superheating is uniquely suited to

boiling water reactor systems and has considerable potential for reducing the capital costs of boiling reactors.

While it is true that power densities, at least for the present natural circulation designs of boiling reactors, appear to be definitely limited, the use of forced circulation will no doubt reduce this limitation. The dual cycle attempts this in a somewhat less obvious way, by reducing the percentage of reactor power utilized in steam void production. Another less publicized design, since it is of recent vintage, is a method of increasing the coolant flow by means of high chimneys or risers above the core. It appears that high pressure will certainly increase the stable limits of operation. There are, however, definite economic disadvantages to extremely high pressure systems. Forced circulation operation can certainly increase the limiting power density either by tending to stabilize the hydrodynamic system or by merely forcing more coolant through the core and hence reducing steam void fractions for a given power density. The application of high chimneys is somewhat similar in this respect. However, if additional driving force is provided it becomes necessary to contend with a problem of instability of another type. This problem may be stated simply that at certain power densities, inlet temperature, and driving pressures, the channel can operate at more than one flow rate. Properly designed inlet orifices can eliminate this behavior but introduce additional pressure drop requiring extra pumping power or still higher chimneys. This further increases complexity and plant cost.

It should not be assumed that the problems which still exist will impose limitations to the ultimate utility of the boiling reactor power plant. They merely reflect the present stage of development; one which permits a better understanding of the more specific problems through experimental investigations but which has not provided sufficient experience to predict the ultimate, most effective solutions. It would be rather naive to expect that single, direct solutions to these problems will ever be accepted, but there certainly exists a large area for improvement of present designs.

The ultimate development of nuclear power plants is, of course, dependent upon economics. At the moment, in the United States, nuclear power is not necessary because of the still abundant fuel reserves. At the rate that the use of electric power is increasing, however, the time is not far distant when the production of fossil fuels will no longer, comfortably, be able to meet the demands and then it will become expedient to turn towards nuclear power. Since the early part of this century, the generation of electricity in the United States has roughly

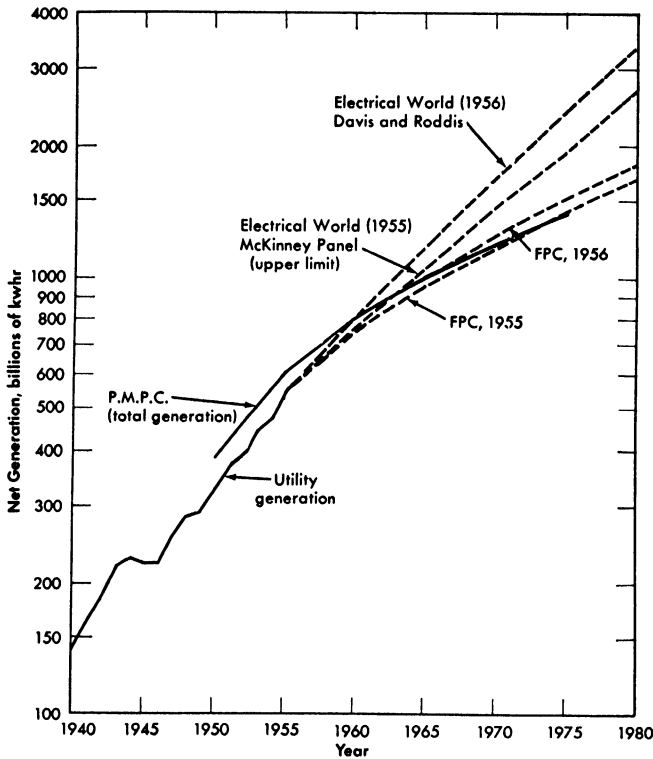


FIG. 11-17. Net generation of electrical energy, 1940 to 1956, and estimates of future requirements.

doubled every ten years. This rate of growth had held until the present time but obviously the doubling of output cannot be expected to continue indefinitely. Figure 11-17 is a plot on semi-logarithmic coordinates of the future demand for electricity. The semi-logarithmic plot is used to show the rate of growth visualized by the several estimates. The actual growth curve of net utility generation is shown by the heavy line to 1956; future estimates are indicated by dotted lines. There are several schools of thought regarding the future rate of growth of electric power requirements. The conservative side is represented by the estimates of the Federal Power Commission. Even after the rapid developments of the five year period following the PMPC (The President's Materials Policy Commission)¹³ report, their estimates are only slightly higher. The estimates of the *Electrical World* and the McKinney Panel¹⁴ are much more optimistic, and approach closely the historic 10-yr doubling of output for the period 1955 to 1975.

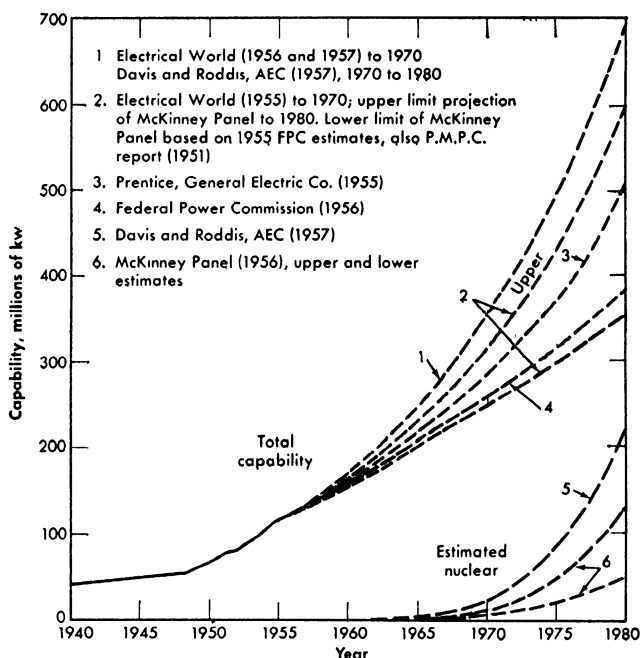


FIG. 11-18. Generating capacity, 1940 to 1956, with estimates of future requirements.

Figure 11-18 shows graphically the various estimates in terms of installed capability. This plot is made using rectilinear coordinates, which tends to give visual emphasis to the optimistic aspects of the various predictions in relation to past performance. Actually, the growth depicted by the most optimistic forecasts shows a continued doubling of capacity every ten years beginning in 1940. Despite the wide variation in estimates there is general agreement that substantial increases in capability will be required by 1970 and 1980, perhaps in an amount intermediate between the upper and lower projections of the McKinney Panel.

At this stage of development of nuclear power reactors, it is most difficult to forecast the share of future capability to be realized by nuclear installations. Predictions of the McKinney Panel and Davis and Roddis for the future development of nuclear power are shown in Fig. 11-18. It seems reasonable to accept the premise that the growth of nuclear power will approach the lower limit curve shown; indeed it can be argued that it is necessary from the long-term view of our energy needs that we maintain a development program of sufficient

magnitude to ensure some such rate of development. However, in view of our supplies of fossil fuels, the chances are much less than anything like the upper limit capabilities will be attained as early as the time schedule indicates.

Aside from improvements in the design of plants the essential factor which could accelerate the development of nuclear power in the United States in the further reduction in nuclear fuel costs and this means largely a higher rate of burn-up. By increasing the life of fuel elements fewer replacements are necessary and this results in less frequent incurrence of fabrication and processing costs, which are at present major factors in total fuel costs. It is difficult to estimate in advance the life of any given fuel element since this is dependent upon various factors such as radiation damage, corrosion, decreases in reactivity, and by accumulation of poison products. Reliable operating data on fuel life will be available only after several power reactors have been in operation for several years.

The use of oxide fuel, such as is used in BORAX-IV, shows considerable promise in effecting lower fuel costs. Uranium oxide or uranium-thorium oxides seem able to give very good fuel element life, perhaps 25,000 megawatt-days per ton or more at moderately high temperatures. Their use, however, tends to reduce reactor specific power by virtue of the low density and low thermal conductivity of the material.

An essential point in evaluating whether a new plant should be nuclear fueled or fossil fueled would be the cost performance of the plant over its entire useful life. Thus a decision might be made to build a nuclear plant even though it would not be competitive at first, because nuclear fuel costs would be expected to decrease over the next 30 years while conventional fuel costs increased.

Another factor in reducing the cost of nuclear power will be the increased standardization in nuclear plants of pumps, heat exchangers, valves, and other components. This has been attempted at Vallecitos. Experience also indicates that in a number of areas standard equipment can replace the specially designed components used initially. This again is a reflection of increased knowledge and confidence which can result only from actual operating experience.

At this early stage in the development of power reactors, it is necessary to impose very stringent specifications on the reactor and much of the associated equipment in order to reduce to an absolute minimum the possibilities of breakdown or failure. To the extent that these conditions bear on public safety, there can be no relaxation of specifications. However, in many cases the stringent requirements are merely an attempt to avoid shutdown for replacement or routine main-

tenance, or are a hedge against anticipated problems which actual experience may indicate do not exist.

As experience is gained and more becomes known about the reliability of reactors and their components, it will be possible to have less stringent specifications, leading to substantial reductions in the cost of fabricating these parts. Use of standard melts of steel, gasket closures, and reduced fabrication tolerances on pressure vessels are examples. In the case of the boiling water reactor, the moderate pressures, of course, considerably reduce the costs of the pressure vessels which need to be only strong enough to withstand the operating steam pressure. Also, the large containment structures now being provided in virtually all plants and which now contribute so importantly to costs may prove unnecessary.

The very low capital costs for conventional power stations are the result of the experience of building hundreds of power stations. It should not be surprising, therefore, that the first nuclear power stations have capital costs considerably higher — \$300 to \$400 per kw of capability compared to \$100 to \$200 per kw for conventional plants. With experience this first cost can be lowered, but it is a moot question whether it can be lowered to the very low values which coal-burning plants now enjoy. Fortunately, the lower costs expected for the nuclear fuel can offset the higher capital cost.

REFERENCES

1. W. E. NYER, S. G. FORBES, F. L. BENTZEN, G. O. BRIGHT, F. SCHROEDER, and T. R. WILSON. Experimental Investigations of Reactor Transients, IDO-16285, April, 1956.
2. T. R. WILSON. An Engineering Description of the SPERT-I Research Facility. IDO-16318, June, 1957.
3. W. E. NYER and S. G. FORBES. Spert Program Review. IDO-16415. September, 1957.
4. W. E. NYER, S. G. FORBES et al. Transient Experiments with the SPERT-I Reactor. *Nucleonics*, Vol. 14, pp. 44-49, June, 1956.
5. F. SCHROEDER et al. Experimental Study of Transient Behavior in a Subcooled, Water-Moderated Reactor. *Nuclear Science and Engineering*, Vol. 2, No. 1, p. 96. January, 1957.
6. S. G. FORBES, F. SCHROEDER, and W. E. NYER. Instability in the SPERT-I Reactor Preliminary Report. IDO-16309, October, 1956.
7. S. G. FORBES, F. SCHROEDER, and W. E. NYER. Instability in SPERT-I. *Nucleonics*, Vol. 15, pp. 41-43, January, 1957.
8. T. R. WILSON. The SPERT-III Reactor Facility. IDO-16314, May, 1957.
9. R. W. MILLER. Calculations of Reactivity Behavior During SPERT-I Transients. IDO-16317, June, 1957.
10. F. L. BENTZEN. SPERT-I Instrumentation, IDO-16316 (1957).
11. H. P. ISKENDERIAN, L. E. LINK, M. TRESHOW, and J. M. WEST. Heavy Water Reactors for Industrial Power, Including Boiling Reactors. Paper No. 495 Geneva Conference, 1955.
12. J. M. HARRER, J. A. DESHONG, JR., V. C. HALL, J. A. THIE, W. R. WALLIN, E. A. WIMUNC, R. E. BAILEY, et al. Performance Evaluation of Direct Cycle Boiling Water Nuclear Power Plants Based on Recent EBWR and BORAX Data. A paper to be presented at the 1958 Geneva conference.
13. Resources for Freedom, a report of the President's Materials Policy Commission, U. S. Government Printing Office, Washington, D. C., June, 1952.
14. Report of the Panel on the Impact of the Peaceful Uses of Atomic Energy Volume II, Background Material, U. S. Government Printing Office, Washington, D. C., 1956.
15. HARLAN W. NELSON. Fuel Availability and Future Power Resources. Paper presented at the American Power Conference, Chicago, Ill., March 26-28, 1958.
16. WALTER H. ZINN. An Appraisal of the United States Nuclear Power Program. Presented at the American Assembly, Columbia University, December, 1957.

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