CIVILIAN POWER REACTOR PROGRAM

PART III

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Book 5

Status Report on Boiling Water Reactor Technology as of 1959

UNITED STATES ATOMIC ENERGY COMMISSION

MASTER

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Technology as of 1959*

Published : 1960

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INTRODUCTION

An intensive but brief evaluation of the technological and economic status of boiling water reactors has been performed by personnel of the General Nuclear Engineering Corporation under contract to the U.S. Atomic Energy Commission. Particular thanks are due Argonne National Laboratory for supplying reports and memoranda dealing with the work which has been carried out at ANL. Certain data were also supplied by the Allis-Chalmers Manufacturing Co. on those boiling reactors for which it has primary responsibility.

It was specified by the AEC that considera-

tion should be given only to boiling water reactor power plants in the 25 to 325 eMw range. Ground rules were also furnished by the AEC for computing the cost of electricity produced by these plants.

This report describes work already accomplished dealing with the current status of boiling water reactors from a technological standpoint. This will be followed by an appraisal of outstanding problems, how they might be resolved, and the benefits which can logically be expected to accrue from efforts to resolve these problems.

SUMMARY

The boiling water reactor concept has evolved from nonboiling reactor designs because of advantages apparent in the simplicity of the direct steam cycle, the lower operating pressure for a given steam temperature, the higher thermal efficiency obtainable, and the decrease in capital costs attributable to the above advantages. There was some concern originally with such problems as the effect of steam voids on stability, reactivity, and safety, and the effect of radioactivity associated with using steam directly from the reactor. However, research, development and operation of boiling water reactors have led to a better understanding of the concept and now there is no question of the feasibility of satisfactorily operating these reactors. Boiling water reactors in operation have already demonstrated safety of operation, stability at relatively high power densities with large percentages of reactivity in steam voids, and lowered capital costs as a result of simplicity of plant design.

Further improvements in the designs are already being studied with the ultimate objective being the generation of nuclear superheated steam at pressures and temperatures comparable to modern fossil fuel steam plants.

Summary statements regarding some of the more important general development aspects of work completed or underway follow. More detailed data on these aspects, as well as information on boiling water reactors which have contributed to this development or are under construction, are included in the main part of the report.

1. The Boiling Water Reactor Concept

Boiling H₂O reactors fall into the category of thermal converters and cannot be made to

operate on natural uranium. Typical reactors use approximately 1 to 3 percent enrichment depending on the fuel, clad, and burnup requirements. Light water has been used as both moderator and coolant in the pressure vessel type reactors designed in the United States. Graphite or D₂O can also be used as moderator in pressure tube type boiling water reactors. By proper design, cooling of the boiling water reactor can be accomplished with: natural circulation within the vessel; natural circulation through an external loop; forced circulation; or a combination of forced and natural circulation.

2. Technological Status of Pressurized Boiling Water Reactors

Although the first boiling water reactor experiment was made only 6 years ago, an adequate technology exists today to proceed with boiling water reactor designs of all sizes. Much of the boiling water technology available from non-nuclear plants and the nuclear technological background obtained from pressurized non-boiling and research type water reactors were applied to the boiling water reactor concept. This allowed development of this type to proceed rapidly in all phases of design. A brief summary of the status of these developments follows.

a. Physics

Although the physics of boiling water reactors is complicated by the presence of steam voids, sufficient data are available to allow for conservative extrapolation to large power reactor designs.

b. Heat Transfer and Fluid Flow

Most of the research and development to date on heat transfer and fluid flow has been centered on natural circulation boiling systems. Data are adequate to establish conservative reactor design criteria but are not sufficient for obtaining the optimum performance necessary to yield the most economic design.

Neglecting stability requirements, heat transfer performance is limited by the center melting temperature of the fuel ($5,000^{\circ}\text{F}$. for UO_2) and by the maximum surface heat flux allowable at a particular flow velocity and sub-cooling, before burnout occurs. Except for the economics of fuel fabrication, the central temperature of UO_2 would not severely limit the maximum heat flux since subdividing of the fuel can be used. However, present economic studies have indicated that a UO_2 fuel pellet size of approximately 0.4 to 0.5-inch O.D. should be used. A typical Zircaloy-2 clad, 0.45-inch O.D. pellet could operate at a maximum heat flux of approximately 400,000 Btu/hr-ft² with a centerline temperature of $4,900^{\circ}\text{F}$.; this is based on an average UO_2 thermal conductivity of 1.0 Btu/hr-ft- $^{\circ}\text{F}$. This value of heat flux is well below established burnout heat fluxes for typical boiling reactor operating conditions. Assuming a typical maximum-to-average heat flux of 3.5 results in an average heat flux of approximately 115,000 Btu/hr-ft², which, for a reactor design using a representative water-to-fuel ratio of around 2.5, results in an average power density of 39 tKw/liter of core and an average specific power of approximately 14 tMw/ton of UO_2 for a core containing no control rod channels. When typical space requirements for control rods are taken into account, the average power density drops to about 34 tKw/liter which is representative of the values to which typical boiling water reactors could be designed today. Most boiling reactors at present are being designed to slightly more conservative values because of the lack of detailed operating data.

Natural circulation reactors in small sizes

have been shown to be hydrodynamically stable over a wide range of conditions; and, with proper design, it appears possible that hydraulic stability can be considerably improved. Little work has been done on forced circulation boiling systems, but it is expected that hydraulic stability would be readily achievable even in large reactors, since the orifices needed to insure stable flow represent no problem where a pump is available to overcome the resultant pressure drop.

c. Fuels and Materials

The most widely accepted fuel material for water reactors is uranium dioxide (UO_2). This is due to its excellent corrosion resistance in water and its radiation damage resistance. In-pile tests indicate that low enrichment uranium dioxide fuel is radiation damage resistant to burnups greater than 10,000 megawatt-days per tonne, is relatively inert in hot water, retains fission products to a large degree, and is compatible with zirconium, stainless steel, and aluminum cladding. Uranium dioxide pellets can be fabricated in various forms and with densities of approximately 95 percent of the theoretical maximum. The thermal conductivity of UO_2 is low; typical data have indicated an average value of approximately 1.0 Btu/hr-ft- $^{\circ}\text{F}$. at operating temperatures.

Three cladding materials are being used in boiling water reactors: aluminum, zirconium, and stainless steel. Aluminum-nickel alloys have shown increased water corrosion resistance, but at present their long term integrity is questionable at temperatures exceeding 400°F . Zirconium and zirconium alloys have been developed to the point where these materials are favored as fuel clad and core structural material in water reactors. Good strength and water corrosion resistance properties have been reported for temperatures as high as 660°F . The low neutron absorption cross section of zirconium offsets its high cost relative to stainless steel. Stainless steel has been used extensively

in reactors and has excellent corrosion and strength properties. In direct cycle boiling reactors, there has been little evidence of radioactive cobalt buildup in the steam loop because of corrosion of stainless steel structural components. In forced circulation loops and where an indirect cycle is used, reactors using stainless steel for fuel cladding may show significant accumulation of radioactivity.

Two types of control rod materials have been developed and are being used extensively in water reactors—boron-stainless steel and hafnium. Both of these materials show excellent resistance to hot water corrosion. Boron-stainless steel is being used in most reactors because of the high cost and limited supply of hafnium; otherwise, hafnium would be preferable inasmuch as it does not suffer isotopic depletion or radiation damage. Other materials such as the rare earths have not been developed to a point where they are acceptable.

d. Reactor Safety

Boiling water reactors which have large steam expansion volumes above the core have been shown to be inherently safe against plausible reactor excursions. Possibilities of metal-water reactions have been virtually eliminated by the use of oxide fuel, which does not undergo an exothermic reaction. The fuel clad, if zirconium, aluminum, or stainless steel, can enter into a metal-water reaction. However, the possibility of such a reaction occurring is remote since conditions normally required for such a reaction (fine dispersal of the molten material into the water) are not readily attainable.

e. Reactor Stability

Although there has been much discussion about the instability of boiling water reactors, these reactors have shown excellent stability over a wide range of power with relatively large amounts of reactivity compensated by steam voids. Boiling Reactor Experiment-IV (BORAX-IV) held approximately 6.9 percent reactivity in steam voids at 322 psig

and achieved a power density of approximately 45 tKw/liter of core. Progressive understanding of the reactivity feedback effects due to voids, temperature, flow, and pressure has substantially raised the core power density for stable operation in boiling reactors. Increases in operating pressure to 300, 600, and 1,000 psi have shown progressively greater stability at higher power levels than previous reactors. Present boiling water reactor designs expect to use operating pressures of 1,000 psi or higher. The use of uranium dioxide fuel has led to a much longer thermal time constant in the fuel. This, in turn, has led to greater stability of the reactor with respect to transient reactivity changes. Hydraulic stability can be improved in boiling water reactors by introducing a restriction in the flow path outside the core zone. The extra pumping head requirements will adversely affect the natural circulation type reactor but is no problem in reactors using forced circulation.

f. Coolant Chemistry

The major operational difference between the coolant systems used in fossil fuel boilers and boiling water reactors is the decomposition of water in the reactor and the deposition of radioactive corrosion products, "crud", in the heat transfer loop. No deleterious effect on steam piping, turbine, or condenser has been noted in Experimental Boiling Water Reactor (EBWR) as a result of the higher oxygen content which results from the decomposition of the water. Most of the crud deposition occurring on the fuel plates in EBWR is apparently due to corrosion of dummy aluminum-nickel fuel elements not normally used in power reactors. Very little crud deposition has been noted in the primary steam loop, and radioactivity of the primary loop has remained low even when operating with defected UO_2 fuel elements.

g. Components and Auxiliary Systems

In general, boiling water reactors have been able to use components and auxiliaries already

developed for conventional steam plants or for nonboiling water reactors. Specific developments have been required because of the water activity, water decomposition, and corrosion of materials in saturated steam.

Materials and technology are available for the design and construction of pressure vessels for 700° F. and 1,500 psi operation in diameters up to 15 feet. Corrosion of stainless clad vessels has not presented a problem as yet.

Corrosive and erosive effects of wet, oxygenated steam have been slight in the turbine of EBWR, which has led to increased confidence in the use of direct steam from the reactor.

In the field of control rod drives, many types of drives have been developed—screw jack, magnetic jack, rack and pinion, hydraulic—all of which can be made to operate satisfactorily. Some bottom-mounted drives appear to suffer from buildup of radioactive crud which settles to the bottom of the thimbles, making maintenance more difficult. Methods of minimizing this problem in future reactor designs are being evolved.

Instrumentation for reactors has been developed rapidly by the electronics industry, and commercial units are readily available. The greatest need for instrument development is in the in-pile measurement field where the reactor environmental conditions have seriously limited the type of materials which can be used.

h. Containment

Almost all reactors are now being surrounded by gas-tight steel shells, which can contain the products of a maximum credible accident. Although the feasibility of constructing these shells is unquestioned, code specifications are such that costs are very high.

i. Fuel Failure Detection

Detection of even a minor fuel failure is easily accomplished in a boiling reactor by monitoring the exhaust gases leaving the condenser for the fission product, xenon-135. Location of the particular fuel element which has failed is more difficult. More development effort is required toward this end.

3. Reactor Designs

Data for boiling water reactors which have been operated or are under construction are given in the main part of the report. The largest of these now under construction in the United States is the Dresden Nuclear Power Plant. This plant is rated at 626 Mw thermal and 180 Mw net electrical. The reactor will use forced circulation and will operate at 1,000 psig with a core inlet coolant temperature of 505° F. and a core outlet coolant temperature of 546° F. Zircaloy-2 clad UO₂ fuel is used in a segmented type fuel element. An operating power density of approximately 30 tKw/liter of core and a specific power of approximately 11 tMw/tonne of UO₂ are expected. The turbine for this power plant is a saturated steam, dual admission unit operating at 965 psig (540° F.) and 475 psig (461° F.). Modern steam plants operate at 1,450 psig or higher, with superheated steam at 1,000° F.

To approach the modern steam plant operating conditions would require superheating of steam in a separately fired superheater or within a nuclear reactor. The AEC is presently sponsoring studies of reactor designs which will incorporate nuclear superheat and yield data sufficient to design for conditions similar to modern steam plants.

A. THE PRESSURIZED BOILING WATER REACTOR CONCEPT

The pressurized *boiling* water reactor concept differs from the pressurized *nonboiling* water concept in that bulk boiling of the water is allowed in the core, and the reactor pressure is obtained by the steam so generated. Because of this, for a given operating temperature and steam cycle efficiency, the operating pressure is much lower in a boiling reactor than in a non-boiling reactor.

Boiling H_2O reactors fall into the category of thermal converters and cannot be made to operate on natural uranium. However, rather low enrichment fuel is required for boiling reactors primarily designed for economic power production—of the order of 1 to 3 percent depending on the fuel and clad used.

Various fuel types can be used in these reactors if canned with a suitable clad material. The most common fuels used to date are metallic uranium alloys and UO_2 alone or in combination with ThO_2 . Metallic uranium alloys used to date are not very corrosion resistant to hot water and have been displaced by the fissile oxides, the most common being UO_2 , although PuO_2 may be usable and ThO_2 can be incorporated with either. Zircaloy-2, aluminum, and stainless steel are the most common clad

materials used or contemplated for water reactors.

All boiling water reactors in the USA have been designed with light water (H_2O) as both the moderator and coolant. However, in Russia a boiling water plus direct superheated steam reactor is being constructed with graphite as the moderator and light water and steam coolant flowing in pressure tubes of the boiling and superheating regions, respectively.

Although H_2O -moderated and cooled boiling water reactors employ lattices of relatively low moderator-to-fuel ratio, the moderating ability of H_2O is such that the neutron energy spectrum producing fissions is in the thermal range.

By proper design, cooling of the boiling reactor core can be accomplished with: natural circulation within the vessel; natural circulation through an external loop; forced circulation; or a combination of forced and natural circulation.

The greatest flexibility of this type reactor lies in its use in almost any conceivable steam cycle—direct cycle, indirect cycle, direct plus indirect cycle. Since H_2O is the reactor coolant, it is possible to reheat or superheat directly in the reactor instead of in an intermediate heat exchanger.

B. OBJECTIVES

The boiling water reactor concept has as its prime objective the generation of economic power. At present, such an objective can be realized if capital costs can be decreased and, above all, if fuel costs can be lowered. Both of these objectives can be achieved in this type reactor by:

1. Increasing the thermal efficiency of present boiling reactors by operation at higher pressures and temperatures; using some form of superheat, either fossil fuel or nuclear; or using various cycles for decreasing coolant pumping power, such as reheating the turbine steam.

2. Increasing the allowable fuel exposure by the development of fuels and cladding materials compatible with H_2O at high temperature and for long periods of time. Further improvement of UO_2 fuels and zirconium alloy clads is possible.

3. Decreasing the physical size of the reactor and thus the capital cost of the plant by the use of smaller cores of higher power density. Here, the use of forced circulation and the optimization of exit steam

quality, within safety and stability limits, are anticipated.

4. Increasing the specific power of the fuel to lower the fuel inventory and thus the fuel costs. This would benefit all types of reactors. In this type of reactor, the greatest increase in specific power would come by power flattening and by subdivision of the fuel.

5. Optimizing the conversion ratio in this type of reactor by the proper moderator-to-fuel ratio and use of low parasitic absorber clad.

6. Simplification and/or elimination of components of the plant which are not really necessary but which are presently used because of lack of sufficient experience or because of extreme safety precautions resulting from limited understanding of possible or impossible nuclear accidents.

7. Independent of the reactor concept, obtaining economical nuclear power can best be aided by decreasing fuel fabrication and reprocessing costs, since the best one can do in decreasing capital costs is to approach the cost of fossil fuel boilers.

C. WORK COMPLETED

Although the first boiling water reactor experiment was made in 1953, this reactor concept has become the most widely adopted reactor for electric power generation of all types. One large factor in its acceptance in so short a time lies in its relative simplicity of steam generation and inherent safety. With much of the water technology already available from non-nuclear plants and the existing nuclear background of pressurized and unpressurized water reactors, development of this type of reactor has proceeded rapidly.

1. General Research and Development

The design and construction of boiling water reactors have been preceded by general research and development within the areas of fuel and material properties, heat transfer and fluid flow, coolant chemistry, reactor safety, physics experiments, reactor components, and auxiliary systems. Summaries of important information gained in these areas are given in the following section.

a. Physics of Boiling Water Reactors

The designs of the boiling water reactors that have been operated to date have not involved a large amount of physics analysis. These designs, although quite complex from the physics viewpoint, were able to use the available pressurized water physics data. Also, these reactors are very conservative in the physics design and do not reflect the most efficient or economical design as applied to commercial plants.

To date, most of the reported physics data specifically applicable to boiling water reactors have been obtained from BORAX (3-12)

Special Power Excursion Tests (SPERT) (13-19) and EBWR (1, 2, 20, 21, 22) experiments. These data have been largely concerned with the over-all behavior of boiling water reactors, and the physics data obtained are not of the basic type but are more in the nature of prototype experiments. Thus, while these experiments have led to increasing confidence in the predicted behavior of a boiling water reactor, much of the detailed information which is necessary for optimizing a nuclear reactor is still lacking. For example, most boiling water reactors being constructed or proposed for civilian power production are fueled with low enrichment UO_2 fuel rods. However, at present, no reactor completely loaded with this type of fuel is in operation. Therefore, uncertainties in the static and dynamic behavior of each core make it necessary to provide safeguards to insure meeting the operational specifications. These safeguards add to the cost of the reactor.

The presence of steam in a boiling reactor introduces a variable in the analysis which is not found in other reactor types. The steam coefficient of reactivity, usually identified as the void coefficient, is composed of a sum of negative and positive components. The void coefficient may vary as a function of position within the reactor. For example, in a large fully enriched reactor, the reactivity associated with steam is positive for steam formation at the center of the reactor and negative for steam formation at the outside of the reactor. Since the variation of void coefficient with temperature is quite large (variations by factors of 5 to 10 could occur between room temperature and operating conditions), it may be extremely difficult to have a negative void coefficient for

the cold-clean condition of the reactor without having too much reactivity in steam voids at full power.

The specification of the void coefficient implicitly determines the water-to-fuel ratio in the reactor. For a reactor of specific size, a change in the water-to-fuel ratio primarily changes the neutron resonance escape probability and thermal utilization. The rate of change of resonance escape decreases with increasing void while the opposite is true for the thermal utilization. By using the proper water-to-fuel ratio, the desired void coefficient at a particular operating condition can be specified. For large reactors, using oxide fuel, a water-to-fuel ratio of about two appears to be optimum. For smaller reactors, a larger water-to-fuel ratio is required because increased neutron leakage increases the magnitude of the void coefficient.

The presence of voids also affects the power distribution. It will tend to peak the reactor power toward the bottom of the reactor and raise the axial factor compared to the usual cosine distribution. However, it also tends to flatten the power in the radial direction. Control rods inserted from the bottom of the reactor can be used to compensate for the higher axial peaking caused by the void. Strategic radial positioning of the control rods will reduce the overall peaking factor, and the effect of the control rods in flattening the power is a very important parameter in steady state operation.

Control requirements of a light water boiling reactor are somewhat different from a non-boiling light water reactor of the same size. The nonboiling reactor will have a lower water-to-fuel ratio; and, as such, will have a larger reactivity change with temperature. The lower reactivity change in the boiling reactor is offset by the additional reactivity requirements due to the steam coefficient of reactivity. However, the conversion ratio in the nonboiling reactor will tend to be higher, and the reactivity loss with burnout will be less. The net result is that boiling reactors usually require slightly

more total control than nonboiling reactors.

The plutonium produced as fuel burnup progresses will contribute a positive component to the temperature and void coefficients. This is not expected to cause any trouble except in reactors which have positive, or only slightly negative, coefficients with fresh fuel in the core. It is believed that plutonium will not affect the sign of the operating void coefficient for 10,000 Mwd/T operation, but it could cause the void coefficient to become positive at room temperature in those reactors which have a high water-to-fuel ratio. This may operate to the disadvantage of natural circulation reactors by dictating a smaller water-to-fuel ratio in the core, thereby increasing the pressure drop.

In most other considerations, the physics design of a boiling reactor is similar to other reactors. The same questions arise as to the accuracy of analytical methods such as the solution of multigroup diffusion equations, which include the accuracy of the nuclear constants, multigroup coefficients, neutron spectra, etc. Increasing the accuracy of theoretical predictions not only requires the development of analytical methods, but constant revision of these methods as more information becomes available.

(1) PRESENT STATUS OF REACTOR STATICS

(a) *Lattice Parameters*

During the past several years, an extensive program has been undertaken at the Bettis plant and Brookhaven National Laboratory to measure the properties of slightly enriched uranium metal and UO_2 light-water-moderated lattices (23). These data, which are room temperature measurement, permit evaluation of theoretical methods used to determine lattice parameters.

Typical data of the variation of lattice parameters and infinite medium multiplication factor k_∞ , as a function of water-to-fuel volume ratio are shown in Figure 1 and Figure 2. These curves represent a compilation of data from experiments of 0.600-inch uranium metal

rods in light water (24). The data for oxide lattices are shown in Figure 3 and Figure 4. These curves represent a compilation of data on 0.387-inch UO_2 rods of 10.53 g/cc density in light water (23).

(b) *Reactivity and Control.*

Experimental measurement of reactivity effects and reactor control of an operating power reactor has been reported only on the EBWR (25) and BORAX-IV (2) reactors. Since EBWR uses plate-type, metallic uranium alloy fuel elements and BORAX-IV uses a $\text{ThO}_2\text{-UO}_2$ mixture with a very high water-to-

FIGURE 1—Thermal utilization (f) and resonance escape probability (p) for uranium-water lattices. [Refs. 23, 24.]

FIGURE 2—Fast effect (ϵ) and infinite multiplication factor (k_∞) for uranium-water lattices. [Refs. 23, 24.]

fuel ratio, it is not possible to make direct use of the experimental information in a rod-type UO_2 core.

Despite the apparent lack of information, there is a high degree of confidence that initial criticality can be predicted to within 2 to 3 percent Δk for simple fuel elements, such as rod clusters with water-to-fuel ratio of about 2 or greater. The reported error in EBWR criticality at startup, based on calculations (no critical experiments), was 3.39 percent Δk (25).

FIGURE 3—Thermal utilization (f) and resonance escape probability (p) for UO₂-water lattices.

Similar accuracy was attained for the total worth of control rods.

The advent of two-dimensional, multigroup diffusion theory has aided in removing a good measure of the geometric uncertainties. Therefore, for rod-type fuel arranged in a similar fashion to the cold critical experiments used in determining lattice parameters, it is reasonable not to expect a large deviation for the initial criticality. However, further verification of a more detailed nature is required to check the calculational accuracy.

The calculation of void and temperature coefficients suffers from lack of direct experimental measurement. However, cold critical

mockups of void coefficient are in good agreement with theory ($\sim 10\%$) for EBWR experiments. The temperature coefficient calculations for BORAX-IV are a factor of 2 lower than the measured values in the temperature range of 65° F.-207° F. and at 420° F.

(c) Power and Void Distributions

The experimental measurements of flux and power distribution in EBWR and BORAX-IV show close correlation with calculations. On the other hand, one has no knowledge of the operating void distributions except from theory.

The EBWR and BORAX-IV data indicate

FIGURE 4—Fast effect (ϵ) and infinite multiplication factor (k_{∞}) for UO₂-water lattices.

the type of power distribution expected in a boiling reactor. However, the EBWR is relatively small (4-foot diameter and height) and the BORAX-IV was even smaller so that a direct extrapolation of the power distribution to a large core is not possible.

(d) Fuel Burnup

No boiling reactor has operated to a long burnup, although some of the fuel in EBWR has been irradiated to a maximum of 3,000 Mwd/T. No theoretical calculations of reactivity changes occurring in an operating reactor have been compared with the measured changes.

(2) PRESENT STATUS OF REACTOR DYNAMICS

The experimental boiling water reactor program which encompasses the BORAX reactors, SPERT Program, and EBWR has been primarily concerned with studying the dynamic characteristics of the reactor. Since the early theoretical studies indicated that reactivity in steam voids exceeding the delayed neutron fraction ($\sim 0.72\%$) could give rise to power oscillations, it was necessary to make a detailed study of the effect of steam formation on reactor stability. The main incentive for performing these experiments is to develop theoretical understanding to allow an increase in design power densities.

These experiments have furnished valuable information of the effect of increasing operating pressure and decreasing the thermal conductivity of the fuel elements. For example until very recently, based on the early metallic fuel experiments, a maximum of 2 percent Δk in void has been used as a design criterion to avoid instabilities. However, in the BORAX-IV experiment, using oxide fuel, 6.9 percent Δk in void was achieved with no instability occurring.

b. Heat Transfer and Fluid Flow

Early heat transfer and fluid flow experiments were centered about determining the

characteristics of natural circulation boiling systems, since natural circulation flow is less understood than forced circulation. Important problem areas included flow rate, steam-water mixture density, hydraulic stability and two-phase pressure drop.

A series of experiments performed at Argonne National Laboratory (ANL) (2) provided enough information to develop confidence in the feasibility of the boiling systems for use in nuclear reactors and demonstrated that reactors can be designed to have high circulation rates with natural convection.

These experiments produced data at three pressure levels—atmospheric, 260 psia, and 600 psia—in both open and closed loops and for single and parallel channels.

The systems and apparatus used are listed in Table 1.

The data emanating from these experiments have provided considerable qualitative understanding of the mechanics of boiling systems but are still short of providing the precise quantitative information required for optimizing the design of boiling reactors.

The original data on two-phase friction pressure drop from the work of Martinelli and Nelson (35) on flow at low pressure in horizontal pipes has been largely confirmed in the boiling tests at ANL. Additional work by Petrick (36) and Sher (37) have added to these data. No information on entrance and exit losses is available. Enough information exists for reasonable evaluations of reactor performance, but detailed design data are lacking.

(1) STEAM VOLUME FRACTION AND SLIP RATIO

One of the greatest unknowns in determining steam volume fraction is the velocity of steam relative to the velocity of water. A recent paper from ANL (73) summarizes all the data obtained to date at Argonne National Laboratory. Parameter ranges studied are: pressure, 150–2,000 psi; mixture quality, 0–25 percent, superficial liquid velocity, 0.5–8 ft/sec; and flow channel equivalent diameter, 0.4–2.0 inches.

TABLE 1

SYSTEMS AND APPARATUS USED IN HEAT TRANSFER AND FLUID FLOW EXPERIMENTS AT ARGONNE NATIONAL LABORATORY²

Apparatus	System type	Number of channels and geometry	Maximum power density, Kw/liter	Maximum inlet velocity, ft/sec
Table top visual density study apparatus (atmospheric). ¹	Open - - -	3 annuli, $\frac{1}{16}$ x $\frac{1}{2}$ I D. x 36 in. $\frac{1}{8}$ x $\frac{1}{2}$ I D. x 36 in. $\frac{1}{4}$ x $\frac{1}{2}$ I D. x 36 in.	60 (AC) - - -	2.8
Atmospheric boiler ² - - - - -	Open - - -	4 to 20 rectangular, $\frac{1}{8}$ to $\frac{1}{2}$ in. x 5 x 9 to 24 in	110 (DC) - - -	- - -
260 psia multichannel boiler ³ - - -	Closed - - -	6 rectangular, $3\frac{3}{16}$ x $\frac{1}{2}$ x 60 in. - - -	50 (DC) - - -	3.0
260 psia singlechannel boiler ⁴ - - -	Closed - - -	Single rectangular $3\frac{3}{16}$ x $\frac{1}{2}$ x 60 in. -	60 (DC) - - -	3.0
600 psia multichannel boiler ⁵ - - -	Closed - - -	3 rectangular, $\frac{1}{16}$ x $4\frac{1}{4}$ x 48 in.; 3 rectangular, $\frac{1}{16}$ x $4\frac{1}{4}$ x 24 in.	50 (AC) - - -	1.8
600 psia singlechannel boiler ⁶ - - -	Open - - -	Single rectangular, $\frac{1}{2}$ x 2 x 60 in. - - -	50 (AC) - - -	3.1

¹ The table top apparatus was used to determine the effect of various additives on natural circulation rates and showed very little effect

² The atmospheric boiler was used for preliminary circulation tests

³ The 260 psia multichannel boiler was used to determine flow rates and steam voids by use of a gamma ray source. Considerable scatter of data was caused by inaccurate flow-measuring equipment and incompletely understood effects of flow rate on slip ratio

⁴ The 260 psia singlechannel system was a refinement of the previous setup.

⁵ The 600 psia multichannel boiler was constructed to obtain information for EBWR conditions. In this rig, steam void measurements could be made at any channel location

⁶ A 600 psia singlechannel assembly was substituted for the multichannel rig, and the external piping modified to provide greater flexibility in order to investigate more completely the various phases of boiling. Known as the Armadillo Test Loop, it provides for accurate flow measurement, ease in changing test sections, ability to accommodate test sections up to 5 ft in length, provisions for varying downcomer resistance, pressure taps for two-phase pressure drop measurements, provisions for forced circulation cooling, and provisions for determining chimney effects

Both natural and forced circulation flow was tested using both air-water and steam-water tests.

The data show that the velocity ratio is affected primarily by pressure, mixture quality, superficial velocity, and to a lesser degree by the flow channel geometry. (Superficial velocity is defined as mass velocity divided by liquid density and would correspond in the steam-water tests to the inlet water velocity.)

The effect of quality on the slip ratio is shown in Figures 5 through 12. These curves cover a pressure range from 150 to 1,500 psi and a flow channel equivalent diameter range of 0.44 inch to 2.0 inches. The slopes of these curves are plotted versus pressure on Figure 13 showing the decreasing effect of quality on slip ratio as the pressure increases. A plot of slip ratio versus quality with pressure as a parameter is shown in Figure 14. The 1,500 psi curves were taken from two tests. Data from

the test resulting in the dotted line is questionable due to probable errors in flow measurements. Data from several investigators were used for these curves. The quality effect on slip ratio was also observed to diminish with increasing mass velocity. This was noted at higher mass velocities (magnitude unspecified) but sufficient data have not been accumulated to quantitatively establish the magnitude of the effect.

The effect of mass velocity on slip ratio is shown on Figures 7, 12, and 15 for the steam-water system and Figure 5 for an air-water system and shows a decrease in slip ratio with increasing inlet mass velocity.

Steam volume fractions were also measured by ANL for subcooled boiling and were found to be a function of the bulk water temperature, velocity, heat flux, operating pressure, and channel dimensions. An adequate correlation was not found.

A recent letter from ANL (41) claims that steam void fractions can be predicted to within ± 15 percent over the range covered by the above reported paper. No verification has yet been made within a reactor system.

Analyses of the ANL heat transfer data have been made in GNEC-39 (77) with the arithmetic difference between the velocity of the steam and the velocity of the water as a parameter of correlation. These analyses led to the following conclusions (among others) with regard to the relative velocity (velocity of steam minus the velocity of water).

(a) The velocity of steam itself and the velocity of steam relative to water both markedly increase with distance up the boiling channel.

FIGURE 5—Representative results from air-water studies. [Ref. 73.]

FIGURE 6—Effect of geometry on velocity ratio at 600 psig. [Ref. 73.]

FIGURE 7—The effect of quality on velocity ratio at 150 psig. [Ref. 73.]

FIGURE 8—The effect of quality on velocity ratio at 250 psi— $V_o \approx 1.5 \text{ ft/sec}$. [Ref. 73.]

FIGURE 9—Effect of quality on velocity ratio at 600 psig. [Ref. 73.]

FIGURE 10—The effect of quality on velocity ratio at 1000 psi $V_0 \approx 2.5 \text{ ft/sec}$. [Ref. 73.]

FIGURE 11—Velocity ratio as a function of quality at 1500 psig. [Ref. 73.]

(b) The local velocity of steam relative to water is primarily a function of the local steam void fraction, the relative velocity increasing with α .

(c) At a given pressure, the velocity of water has little influence on the relative velocity of steam. The decrease in slip ratio (*ratio* of the velocity of steam to the velocity of water) with increasing inlet water velocity is largely due to the fact that as the water velocity is increased, the steam velocity increases by the same *amount*, not by the same *ratio*.

(d) Relative steam velocity decreases significantly as the pressure is increased.

(e) The “apparent” velocity of steam with respect to water throughout the cross section of the channel sometimes reaches values of 10–15 ft/sec at the exit of boiling channels. However, the velocity profile is believed to be such that water adjacent to boiling heat transfer surfaces is relatively stagnant while that at the center of the channel moves at high velocity. If so, the rapidly-moving steam need not be moving at high velocity *with respect to the water*. The modest relative velocity required to agree with experimental results is consistent with the buoyant and drag forces involved.

FIGURE 12—The effect of superficial velocity on velocity ratio at 1200 psia—Data of Hughes. [Ref. 73.]

(2) STEAM SEPARATION

The knowledge of the mechanism and various factors influencing the separation of steam from steam-water mixtures in reactor vessels or in separate steam drums is quite incomplete. The present trend of increasing the power and operating pressure of boiling water reactors has caused steam separation and carryover to become one of the problems in boiling water reactor plant engineering.

Little work has been done that has appeared in the English literature; most of the work has been done by Russian engineers (45-52). Their experimental data obtained by different techniques is, in some cases, inconsistent, and the desired equations do not predict correctly the fraction of moisture carried by the steam for certain ranges of steam loads and pressures. Better understanding of the moisture carry-over problem would make possible progress in the direction of decreased size of reactor vessels, steam drums, and steam dryers.

(3) TWO-PHASE WATER LEVELS

Another aspect of the two-phase boiling density problem is determining the density of the two-phase mixture above the reactor core so that its height may be estimated from an external water leg reading.

The height of the two-phase mixture cannot be obtained by purely analytical means. Recourse must be made to experimental data on vapor holdup through stagnant beds of water. Unfortunately, no data exist for flow of steam through stagnant water at 600 psi in large vessels. However, such data are available on air-water systems at atmospheric pressure and in vessels up to 36 inches in diameter (69). Additional data have been reported by Behringer (70) for steam flowing through water in a 2-inch pipe at various pressures up to 600 psi. These data tended to check the work of Zinola (69) and served as a basis of extrapolation of the air-water data to higher pressures for the large diameter vessels.

FIGURE 13—Slope of $\frac{V_g}{V_t}$ quality lines as a function of pressure. [Ref. 73.]

FIGURE 14—Velocity ratio as a function of quality at various pressures. [Ref. 73.]

FIGURE 15—Velocity ratio as a function of superficial velocity at 150 & 600 psig $X \approx 0.03$. [Ref. 73.]

Calculations of the height of the vapor-mixture interface have been made for EBWR (69) at 40 Mw by ANL and tend to explain the increase in carryover actually measured at that power level, (71) since they show that the interface would have been very close to the steam ring at these powers.

A recent ANL report, Test Report 62A, (76) shows the results of water level measurements in the EBWR vessel. Tubes were placed within the vessel and samples of steam and water extracted with the reactor water level being varied. Sample properties were determined by N-16 activity, enthalpy, flow rate through sampling wire, and noncondensable gas content. Figure 16 shows the results of enthalpy measurements at one location. This curve indicates that the thickness of the vapor-mixture interface increases with power. Measurements of the other properties gave similar results. A curve is also given in this report

showing the discrepancy between sight-glass readings and true water levels.

More work must be done in the field of the transport of vapors through low-velocity water at the higher pressures before optimum vessel sizes can be chosen.

(4) COOLANT CIRCULATION

Under certain idealized laboratory conditions, natural circulation flow rates can be quite high. Laboratory tests on a natural circulation loop showed that recirculation velocities up to 10 ft/sec could be obtained (53). Basically, the system consisted of a heated section of 4-foot long pipe, 1 inch in diameter, with a 2-inch diameter, 8-foot long riser above it. It should be noted that these geometric conditions are more favorable than would be encountered normally in an economic reactor design. The large hydraulic diameter of the

FIGURE 16—Enthalpy of sample water in line A₃.

heated flow path implies a low power density core and hence a large diameter pressure vessel. Conversion ratio would also be adversely affected. The long riser section would require excessive vessel and building height for large reactors, but would be satisfactory for small reactors. A natural circulation system without any riser yields recirculation velocities which are generally low, about 2-4 ft/sec. Since some riser height is normally available without undue penalty, velocities somewhere between these limits can be expected for natural circulation systems.

(5) HYDRAULIC STABILITY

Laboratory tests have shown that natural circulation stability is a function of exit steam volume fraction, pressure, and system geometry. These tests have shown that there is a threshold value of the exit steam volume fraction beyond which hydrodynamic instability occurs. An out-of-pile loop experiment made by ANL indicated flow instability begins above a value of 0.75 exit steam volume fraction (74). This relationship is also a function of pressure; as the system pressure increases, the system will become unstable at a lower value of the exit steam volume fraction, but at a higher power density.

Hydrodynamic stability is strongly affected by the system geometry. Placement of restrictions in the downcomer tends to increase the stability of the system; that is, the threshold value of the exit steam volume fraction at the inception of instability increases. This can only be done, however, by penalizing the flow capabilities of the hydraulic loop and thereby downgrading the power-generating capabilities of the reactor system. A series of tests which were run to study the effect of riser geometry showed that for a system where the riser flow area was one-half that of the test section, hydrodynamic instability was observed at a power density of 120 kw/liter at 600 psi; the inlet section velocity was approximately 4 ft/sec. By comparison, stable operation was

obtained at a power level of 350 Kw/liter for the riser geometry of twice the flow area of the test section. The recirculation velocity in this instance was 8 ft/sec. The effect of riser height on reactor stability is unknown.

It should be noted that, in general, natural circulation systems are hydrodynamically stable over wide ranges of conditions and from information gained recently it appears that the stability of such systems can be improved considerably by proper design. More information is needed, however, especially on high riser systems of large hydraulic diameter and of varying relative flow areas to the core.

Forced convection systems with multiple flow channels are also subject to hydrodynamic instabilities which result from parallel channel effects. This problem is discussed in detail in Bonilla's book, *Nuclear Engineering* (72). These effects are due to possible variations in the resistance of each channel while they are being maintained at a fixed pressure differential common to all. Orificing of the individual fuel assemblies of a forced circulation system can be used to eliminate the possibility of flow fluctuations occurring.

The Russians (59) in startup experiments with their superheater reactor encountered a similar problem in switching assemblies from a boiling condition to superheating. They found that when the water boils and the steam content reaches 15-20 percent by weight, a pulsation of the steam-water mixture flow occurs. However, as soon as the mixture starts coming out of the separator, pulsations cease and do not occur during the further increase of the steam phase. Pulsations at the moment the water boils were eliminated by installing low pressure drop orifices which had a pressure drop of 15 psi for a coolant flow of 2,200 lb/hr. Changeover was effected more than 50 times with no disruptions of the stability of the flow observed.

Little additional laboratory information is available. A theoretical criterion for predicting forced circulation hydraulic instability for

relatively high pressure drop systems is given by H. Chilton in a 1957 paper (64).

(6) CRITICAL HEAT FLUX AND BURNOUT

A great deal of experimental data on burnout in water-cooled reactors has been accumulated. Most of this information has come from the Navy program and is applicable to non-boiling systems at 2,000 psia. Some data have been compiled in these programs for boiling systems, but, again, most of this is limited to 2,000 psia. The most complete compilation of these data is contained in WAPD-188, (57) which includes correlations predicting burnout in boiling systems with estimates of pressure effects. A critique of this report is given in KAPL-M-D1G-TD-1(61) and KAPL-M-D1G-TD-2 (62) which claim refined correlations of the same data and elimination of a point of discontinuity. Some Russian data became available at Geneva in Geneva Paper P/2319 (59).

Some general conclusions deduced from available references are listed below:

(a) The critical heat flux is primarily dependent upon the local enthalpy of the coolant—the critical flux decreasing with increasing enthalpy. The manner in which enthalpy is introduced into the correlations differs with each investigation, but the agreement is unanimous as to the qualitative dependency. There is no doubt that the burnout heat flux is less in a boiling system than in a nonboiling one.

(b) The critical heat flux is also dependent upon pressure. For subcooled water, the critical heat flux increases with increasing pressure. In the quality region, ANL reports the critical heat flux decreases with increasing pressure over the range 1,000–3,000 psi. WAPD-188 (57) reports critical heat flux essentially the same at the 600–800 psia and 1,850–2,000 psia levels with a peak of magnitude, 1.20, occurring at 1200 psia and a linear variation in between.

(c) The critical heat flux is dependent upon flow rate. WAPD-188 (57) reports that burn-

out flux is a continuously increasing function of mass flow rate above 1.6×10^6 lb/hr-ft² and has no observable dependency at lower flow rates. Other reports (58, 59, 63) indicate that in the quality region a maximum occurs in this function. An analysis of Bettis data (58) reports that the mass velocity at which this maximum occurs decreases with increasing fluid enthalpy.

(d) The critical heat flux is indeterminately dependent upon channel size and/or geometry. There is disagreement as to whether this dependency is upon the channel diameter or L/D ratio. ANL (54) reports no indication that channel length or L/D ratio affects critical heat flux but shows an effect of decreasing burnout flux with *increasing* flow diameter. Bettis (57) shows an effect of decreasing burnout flux with increasing L/D, from which it can be inferred that burnout flux decreases with decreasing flow diameter, in contradiction to the ANL findings. However, Bettis made no attempt to correlate with the diameter itself. The Bettis tests were based upon varied combinations of channel diameter and lengths.

(e) Disagreement exists on the effect of surface condition on burnout. Bettis (57) reports no noticeable effect whereas ANL (54) reports a significant increase in critical heat flux with time attributed possibly to the effect of surface irregularities (caused by corrosion and corrosion products) which tend to promote turbulence and also to offer large numbers of points at which nucleation can take place.

Table 2 demonstrates the variation in calculated burnout flux using different correlations. Conditions are: pressure, 2,000 psia; mass flow, 1.6×10^6 lb/hr-ft² (~ 9 ft/sec); quality, 5 percent.

It should be noted that because of the large scatter in data, it is not possible to select an accurate value for burnout heat flux. However, even the lowest values are substantially greater than the heat fluxes at which boiling reactors actually operate at present.

Some work has been done (55, 56) toward increasing burnout heat flux by use of the cen-

trifugal force of the liquid. Preliminary results indicate:

(a) Centrifugal forces are effective in breaking up vapor films on the heated surface and increasing the burnout heat flux in swirling flow.

(b) At equal pumping power, swirling flow gives higher burnout heat fluxes than straight flow.

TABLE 2
COMPILATION OF BURNOUT HEAT FLUX MEASUREMENTS*

Source	Burnout heat flux, Btu/hr-ft ²	Channel configuration
USSR (59).....	807,000	(0.355 in. I.D. tube).
ANL (54).....	730,000	(0.304 in. I.D. tube, L/D=60).
	630,000	(0.436 in. I.D. tube, L/D=41).
Bettis (57).....	875,000	(L/D=60) Round tubes.
	896,000	(L/D=41) Round tubes.
	**875,000	($G \leq 1.6 \times 10^6$) Rectangular
	**1,130,000	($G \geq 1.6 \times 10^6$) channels.
KAPL (62).....	1,110,000	Rectangular channels.
Combustion Engineering (58).	1,460,000	Rectangular channels.
Dresden (60).....	***915,000	Solid rods.

*P=2,000 psia, $G=1.6 \times 10^6$ lb/hr-ft² (~ 9 ft/sec), Quality=5%.

**Point of discontinuity in correlation.

***Data corrected to 2,000 psia using method recommended by Bettis.

(7) FUEL TEMPERATURE

The problem of determining maximum fuel temperature in UO₂ rod elements is complicated by the fact that the thermal conductivity of UO₂ is not precisely known and that the effect of thermal stress cracking of the pellets and the dilution of the helium bond by low thermal conductivity fission gases is unpredictable. UO₂ thermal conductivity at temperatures of interest in reactor design has been measured as 1.0 Btu/hr-ft-° F. with an estimated accuracy of ± 20 percent. The reactor designer in calculating fuel temperatures uses an idealized model which consists of an unfractured pellet of thermal conductivity equal to 1.0 and an undiluted gas gap retaining predictable dimensions and properties. Limited tests (75) using PWR-Shippingport UO₂ fuel

rods at high surface heat flux (420,000 Btu/hr-ft²) and prolonged irradiation 13,500 (Mwd/T) have indicated the possible limits on power generation capabilities of this type fuel rod and serve as an experimental basis for UO₂ fuel designs.

c. Fuel and Core Materials

(1) METALLIC FUELS

(a) Low Uranium Content Alloys and Dispersions

The highly enriched U-Al alloys used in BORAX and SPERT and U-dispersions (Al-UO₂ & S.S.-UO₂) used in the SPERT Reactors and the Army Package Power Reactor (APPR) all have the advantages of moderate fabrication cost, excellent resistance to uranium corrosion, high resistance to irradiation damage and good thermal conductivity. However, generally speaking, the aluminum alloys are suitable only for low temperature ($\sim 400^\circ$ F.) applications. Slightly higher temperatures may be tolerable if the aluminum is alloyed with ~ 1 wt. percent nickel and if phosphate ion is added to the reactor coolant. Both dispersions have the disadvantage of requiring highly enriched uranium as fuel, which means high fuel burnup costs. The combination of high burnup costs and low temperature application effectively eliminates the aluminum-uranium alloys from consideration for all but military or other special applications. The UO₂-S.S. dispersion elements use highly enriched U but have excellent corrosion resistance in high temperature water and steam, and may have some application in superheat reactors if high burnup at elevated temperature can be demonstrated.

(b) High Content Uranium Alloys

Only one high uranium content alloy fuel has been operated in boiling water reactors to date. This is the Zircaloy-2 clad, U-5 wt. percent, Zr-1.5 wt. percent, Nb alloy plates, used

in the EBWR. This fuel type was developed by ANL as part of the AEC efforts to produce a corrosion resistant, radiation stable alloy of uranium. The plates are manufactured by a rolling technique in which a corrosion resistant layer of Zircaloy-2 is metallurgically bonded to the uranium alloy core. This bonding process is every expensive and extreme care must be exercised during all stages of manufacture to prevent any contamination of the Zircaloy-2 clad which will impair either its corrosion resistance or bond integrity.

The EBWR type fuel has the advantages of low parasitic neutron absorption, relatively good thermal conductivity, and high fuel density. A disadvantage of this material is its low corrosion resistance in high temperature water which results from the heat treatment required to produce a radiation damage resistant alloy. The integrity of the element is therefore dependent on the Zr-2 clad. A specimen EBWR type element has been irradiated to a burnup of about 8,000 Mwd/T(U) and shows excellent resistance to radiation damage, relative to other metallic uranium fuels. High fabrication cost and poor corrosion resistance in hot water may bar this type from any wide spread power reactor application.

(2) CERAMIC FUELS

Two types of ceramic fuels have been, or are proposed for use, in boiling water applications. The BORAX-IV reactor was fueled with a mixture of $\text{ThO}_2\text{-UO}_2$; the Dresden Station will be fueled with UO_2 . Both ceramics are used in the form of short cylinders which are stacked within tubes of sufficient length to produce the required fuel element length. The jacket tubes used in BORAX-IV are an alloy of Al-1.0 wt. percent Ni and the Dresden Station tubes are Zircaloy-2.

(a) *Thoria-Urania ($\text{ThO}_2\text{-UO}_2$)*

The aluminum-nickel alloy jacketed $\text{ThO}_2\text{-UO}_2$ fuel was developed expressly for BORAX-IV. The fuel is made by pressing a

mixture of the $\text{ThO}_2\text{-UO}_2$ powder to the desired size compact and sintering at high temperature to produce maximum density. This fuel has the advantages of low fabrication cost, excellent resistance to both corrosion and radiation damage, a high melting point and contains fertile material which can be converted to a fissionable isotope of uranium. However, highly enriched uranium must be added to ThO_2 as the fissionable material in the mixture, and difficulties (chiefly gamma radioactivity) are encountered in separating and recovering the U-233 produced by the Th. Furthermore, the low thermal conductivity exhibited by this material largely cancels the benefits of the high melting point. Additional development of suitable reactor cycles and reprocessing methods will be needed if this type is to be useful for economic power.

(b) *Uranium Dioxide (UO_2)*

Zircaloy-2 clad UO_2 fuel was developed for the pressurized water reactors and, as a result, has been investigated more extensively than any other power reactor fuel in use. UO_2 has been produced by swaging and extrusion as well as by the more expensive powder compacting. However, the latter method is preferred since the pellet density achieved by the first two methods is lower than that produced by powder compacting. This fuel has the same advantages as the $\text{ThO}_2\text{-UO}_2$ system without as many of the disadvantages. Although UO_2 is only ~ 50 percent as dense as metallic uranium fuel, low enrichment can be used and the recovery of produced plutonium is not much more difficult than is the case with metallic fuels. Average exposures of the order of 10,000 Mwd/T(U) are predicted for UO_2 and have been demonstrated by limited in-pile tests. A more statistical evaluation of Zr-2 clad, UO_2 fuel elements will be obtained as a result of the Shippingport PWR experience.

A brief resume of the general characteristics of UO_2 type elements for water cooled reactors is given in Table 3.

TABLE 3

GENERAL CHARACTERISTICS OF UO_2 TYPE ELEMENTS IN WATER COOLED REACTORS*A. UO_2 —pressed and sintered to 92–95 percent of theoretical density*

1. General properties (78–87):
 - Mechanical..... Extremely brittle; fractures under thermal stress.
 - Melting point..... $\sim 5,000^\circ \text{F.}$, however, poor thermal conductivity brings the center temperature of fuel pellets under typical operating conditions close to the melting point.
 - Thermal conductivity.. Low; drops to approximately 1 Btu/hr-ft- $^\circ \text{F.}$ for reactor operating conditions.
 - Density..... Low—10.97 g/cm 3 theoretical; the ratio of $\frac{\text{metal atoms/cm}^3 \text{ in } \text{UO}_2}{\text{metal atoms/cm}^3 \text{ in metal}} = 0.515$
 - Thermal stability..... Poor; gross radial cracking occurs with some circumferential cracking. No further cracking is noted after the initial.
2. Radiation stability (78, 79, 80, 86, 87). Excellent; no indication of any changes other than those due to thermal effects.
3. Fission product release (78, 79, 80, 88, 89, 90):
 - Fission gases..... 40 to 50% of the gases released from 65% dense packed UO_2 ; approximately 1% released from 95% of theoretical density pressed and sintered pellets.
 - Other fission products.. Small particles of UO_2 , which form because of the severe thermal fracturing, can be released to the coolant.
4. Corrosion (78, 80, 91).... Excellent resistance in 650°F. water and 650 to 750°F. steam. Inert in water containing any measurable excess of hydrogen.
5. Chemical reaction..... No exothermic reaction.
6. Bonding:
 - Helium..... Elements require a gap between the fuel pellet and the can which changes when fuel pellets fracture. Under irradiation the fission gases mix with the helium, lowering its conductivity and increasing the pressure in the fuel can; low thermal conductivity of gas requires close tolerance of gap to maintain low ΔT . Clad failure allows water to replace helium, leading to further failure (87).
 - Lead..... Better thermal conductivity than helium and allows the use of larger gaps between the fuel pellets and fuel can. Little data exists on compatibility of lead with canning material at high temperature.
7. Temperature limitations (80, 88). Maximum operating temperature of the UO_2 is probably limited to the melting point of $5,000^\circ \text{F.}$ until more data are available about the failure rate with center melting. Other limitations are associated with the compatibility with the clad materials.
8. Compatibility with clad materials:
 - UO_2 –aluminum (92).... No reaction below 750°F.
 - UO_2 –Zircaloy-2 (81, 93, 94)..... No reaction below $1,100^\circ \text{F.}$ Slow reaction at $1,300^\circ \text{F.}$ Rapid reaction at $1,600^\circ \text{F.}$
 - UO_2 –Stainless Steel.... No reaction below $2,200^\circ \text{F.}$
9. Fuel lifetime..... 10,000 Mwd-ton or more expected.

B. Canning Material

1. Aluminum.....
 - (Poor mechanical properties at elevated temperatures.
 - Low melting temperature.
 - Poor corrosion resistance in water (78, 80, 95, 96).
 - Metal-water reaction possible (78, 97 101).
 - Good thermal conductivity.
 - No reaction with UO_2 below 750°F. (92).
 - Low neutron absorption cross section.
 - Low cost.

TABLE 3—Continued

2. Aluminum-nickel alloy M-388-----	Poor mechanical properties at elevated temperatures. Low melting point. Improved corrosion resistance in 500° F. distilled water (102-108). Complete oxidation in superheated 840° F. (300 psi) steam.
	Good thermal conductivity. Metal-water reaction possible (1, 97-101). Medium cost.
3. Zircaloy-2-----	Higher absorption cross section (0.0151 cm^{-1}) than Al but still quite low. Good mechanical properties at 650° F. (79, 106-109). High melting point. Good corrosion resistance in water at 650° F. (79). Relatively low thermal conductivity.
	Metal-water reaction possible (1, 97-101). UO ₂ -zirconium couples show a diffusion reaction above 1,200° F. (81). Fretting corrosion a problem where vibration exists (110). Hydriding of the clad can occur (110). Low neutron absorption cross section (0.0076 cm^{-1}). High cost of metal and of close tolerance tubes.
4. Stainless steel-----	Excellent mechanical properties at elevated temperatures. High melting point. Excellent corrosion resistance in 650° F. water.
	Relatively low thermal conductivity. No reaction with UO ₂ below 2,200° F. Metal-water reaction may be possible because of high chromium content (79, 98). High cross section (0.267 cm^{-1} -S.S. 304).

(3) CLADDING AND STRUCTURAL MATERIALS

(a) *Aluminum*

Aluminum has been used for both fuel cladding and core structural materials in BORAX-I, II and III, SPERT and the Argonne Low Power Reactor (ALPR). Its major advantages are the ease with which it can be fabricated into the required shapes, and its relatively low parasitic absorption of neutrons. However, the lack of strength and corrosion resistance at temperatures above 420° F. limits its usefulness for commercial power reactors of interest. Various alloys have been developed in an effort to overcome these shortcomings without notable success to date. Principal among these is the ANL X8001, aluminum-1 wt. percent, nickel alloy. Improved corrosion resistance, relative to commercial alloys, has been reported, particularly if the coolant contains sufficient phosphate ion to yield a pH of 3.5. However, an offsetting tendency towards pitting and increased gal-

vanic corrosion in the presence of stainless steel has been observed when using coolant having this treatment, along with some intergranular attack of sensitized 304 stainless steel. Based upon the results of work accomplished, there appears to be little future for aluminum in large size economic power reactors.

(b) *Zirconium*

Zircaloy-2, an alloy of zirconium, has been used extensively as both a cladding material and a core structural material. Its outstanding advantages are excellent corrosion resistance in high temperature water and low parasitic neutron absorption. The major disadvantages are its high cost, the extreme care which must be used in its fabrication and the lack of cold formability inherent in the material. Heating or welding must be done in an inert atmosphere to prevent contamination by nitrogen and oxygen, which are highly detrimental to both corrosion resistance and formability. Welding or brazing of Zircaloy-2 to stainless steel has been

under study but as yet very little success has been achieved.

However, procedures have been developed for the fabrication of Zircaloy-2 within its limiting characteristics at reasonable cost and future cost reductions are indicated. This material appears to be the most promising of those proposed for use in the commercial power reactor program. Table 3 contains a brief review of the general characteristics of UO₂ canning materials.

(c) Stainless Steel

The excellent corrosion resistance, ease of fabrication and extensive technological background of stainless steel has resulted in its use as both cladding and structural components in a number of mobile and central station power reactor designs. However, the high parasitic neutron absorption of this material requires more enrichment than would be needed if aluminum or Zircaloy-2 were used. As a result, it appears that appreciably higher fuel fabrication costs for less absorptive materials can be tolerated without increasing total power costs. Reductions in Zircaloy-2 fabrication costs make it more preferable than stainless steel as a fuel jacket or structural member in the core proper excepting where high temperature corrosion resistance is impaired as in the case of integral superheat designs.

(4) CONTROL ELEMENT MATERIALS

(a) Hafnium

A primary control element for use in high temperature power reactors is hafnium. This material is closely related to zirconium and has essentially the same properties excepting that its corrosion resistance is not quite as sensitive to nitrogen and oxygen absorption. The high price of this material, and the short supply due to defense commitments limits its use in civilian power reactors. Although processes under development at present are expected to alleviate

the price situation, there appears to be little hope for any great increase in supply

(b) Boron Stainless Steel

Type 304 stainless steel containing up to 2.0 wt. percent boron was successfully produced and fabricated into control blades for EBWR. Ingots of the material are produced by vacuum casting. Strip and plate are produced by hot rolling and some slight cold finishing. Alloys containing 2.0 wt. percent B are quite brittle and sensitive to edge cracking. The 1 percent alloys are markedly less so. Both alloys possess essentially the same properties as 304 stainless steel excepting the previously mentioned reduced ductility and reduced weldability. All the boron alloys exhibit greater radiation induced embrittlement than normal stainless steels. However, this material seems to be an adequate replacement for hafnium if the handicap of periodic removal and replacement of control elements can be accommodated. Iron alloys having greater than 3 wt. percent B have been produced in England but these alloys must be canned in stainless steel to prevent corrosion.

(c) Cadmium

Although quite useful for low temperature applications, cadmium suffers from poor corrosion resistance to high temperature water. To be useful for commercial power reactors of interest, a suitable corrosion resistance cladding is required. There appears to be little economic incentive for this approach in light of the boron stainless steel development.

(d) Rare Earths

The use of oxides of gadolinium, samarium, europium and other rare earths has been under study for some time. However, relatively little data are available which indicate that these materials will be economically suitable for civilian power reactors.

d. Coolant Chemistry (1)

The areas of principal concern to date have been moderator decomposition and effects of "crud" produced by corrosion. Most of the experimental data available come from the operators of the EBWR.

(1) RADIATION DECOMPOSITION

Three factors are important:

(a) *Effect of O_2 in Steam*

Virtually no experimental work has been done on this in EBWR, but 2 years of operation indicate that the presence of 20 cc (STP) of O_2 per Kg of steam has no gross deleterious effect on construction materials in steam piping, turbine, or condenser. The most notable effect is a heavy coating of red oxide inside the condenser.

More work should be done on this question, since if O_2 has no deleterious effect, there is no need to worry about radiolysis in H_2O moderated boiling reactors. There is, of course, an economic problem if D_2O is used.

(b) *Mechanism of Radiolysis*

The net decomposition observed is due to liquid phase recombination of H_2 and O_2 being hindered by the stripping of these gases from the liquid to the vapor phase by the steam bubbles. ANL has reported that by bubbling CO_2 through water under gamma-irradiation, the recombination is completely inhibited and the water decomposes at the maximum theoretical rate.

In EBWR at 20 Mw (thermal) and 600 psi, O_2 is produced at a rate of about 9 liters/min. The calculated rate is 2 or 3 times this figure. Thus recombination is not completely inhibited. At least two mechanisms can be proposed:

(1) There is significant vapor phase recombination in the steam bubbles as they rise through the coolant channel.

(2) Complete equilibrium does not exist

between the steam of the bubbles and the water. Hence H_2 and O_2 are not completely stripped, and some stay in solution in the reactor water. The H_2 and O_2 in solution are carried back down the outside of the core, due to natural circulation, and are here recombined mostly by the gamma flux.

In one experiment in EBWR, steam from the steam dome was passed through the reactor core before it was sampled. The experiment is not fully complete but results indicate that vapor phase recombination of H_2 and O_2 in the steam at 20 Mw and 600 psi is not significant.

Samples of the downcomer water were taken from positions level with the top and bottom of the core. Results showed that O_2 concentrations at the top of the core were 0.2 to 0.4 ppm which had decreased to less than 0.1 ppm at the bottom.

Thus, (2) is the most probable mechanism. The quantities involved are quite sufficient to account for the deviation of the measured decomposition rate from the theoretical maximum.

The above experiments are to be repeated in more detail and results obtained as a function of reactor power and pressure.

If mechanism (2) is true, then decomposition would be increased if recirculation ratio decreased and would also be increased if better stripping took place, i.e., more voids or increased contact time between bubbles and water.

(c) *Prevention of Radiolysis*

It has been found that if H_2 is added to the reactor feedwater at a rate such that the H_2 concentration in the core was 45 cc/Kg, the radiolysis is inhibited to such an extent that O_2 concentration in the stream falls to less than 0.4 cc/Kg of steam at 20 Mw and 600 psi.

If the presence of O_2 does prove harmful, this effect may have promise as a solution to the radiolytic decomposition problem.

The effects of other additives are:

NH₃ and N₂H₄—increase decomposition rate.

N₂—no effect.

O₂—no effect in BORAX;
not yet tried in EBWR.

Decomposition rates for various reactors are shown in Figure 17.

(2) THE "CRUD" PROBLEM

A red-brown deposit of corrosion products has been noticed on the surface of EBWR fuel plates ever since the reactor started steady operation. This deposit has been increasing steadily and now averages 0.004 to 0.005 inch in thickness.

The deposit is a mixture of oxides of Al, Ni, and Fe and is roughly 65 percent aluminum oxide, 25 percent nickel oxide, and the rest iron oxide. The Al probably comes mainly as a result of corrosion of the Al-Ni dummy fuel elements, although some may come from corrosion of the condenser tubes. The main source of the Ni is probably the Ni plate of the turbine, which was showing distinct signs of erosion and flaking off during the last inspection. Some also comes from the Al-Ni dummies.

Some of this deposit is flaking off and falling to the bottom of the reactor vessel, causing high radiation fields in the thimbles below the reactor and the control rod drives. Enough deposit is accumulating around control rod mechanisms to prevent proper seating when the rods are lowered into the reactor.

The rate at which crud is being deposited is higher than the rate it is flaking off since the thickness of the deposit has increased steadily over the past 18 months.

The main activity in this deposit is Co-58 from the fast neutron reaction Ni-58 (np), Co-58. Deposits from the fuel elements show a Co-58 specific activity of about 2.54 d/(min) (mg crud), and a Co-58/Co-60 ratio of four. It is interesting to note that crud from the Al-Ni dummy element shows a Co-58 specific activity of 9×10^5 d/(min) (mg crud) and a

Co-58/Co-60 ratio of about one. This lower ratio is probably due to the greatly decreased fast neutron flux in the dummy fuel elements and indicates that the crud in the reactor is *not* mobile, but remains in the place where it is deposited.

Because of the high radiation fields involved and because of the poor heat transfer of this deposit it is important to find some method of preventing deposition, and of removing the deposit already present.

In systems containing only steels and Zircaloy, it is known that operation at pH 10-11 is effective in preventing crud deposition. In systems containing Al, this high pH could not be tolerated and either Al must be removed or some other suitable inhibitor be found.

As yet very little, if any, "crud" has been carried over into the primary steam loop.

e. Reactor Safety (1-11)

(1) EXCURSION ACCIDENTS

The boiling water reactor has a built-in safety characteristic because of the formation of steam voids which reduce reactivity. This unique feature of boiling water reactors has been well demonstrated by the BORAX experiments, in which reactor transients with periods as short as 0.006 second were safely limited by the mechanism of void formation.

Since power reactors using oxide-type fuel will have longer thermal time constants as compared with the BORAX reactors (1-5 second vs. 0.006 second), it is not expected that these reactors will be able to accommodate as large a *step* function change in reactivity as the BORAX reactor. However, these large step changes do not occur in power reactors, and excursions due to ramp changes in reactivity and step function changes slightly above prompt critical can easily be handled by the combined effects of oxide temperature rise (Doppler broadening), bulk moderator temperature rise, and the steam void formation. Because of the longer time constant of this fuel, it is also expected that relatively fast

FIGURE 17—Water decomposition rates in Experimental Boiling Water Reactors.

transient reactivity changes will be considerably damped, thus tending to decrease any regenerative effects.

To indicate the safety of boiling reactors, it is postulated that under operating conditions, the reactor operator or a malfunction causes an absorber rod to move out of the core. In most reactors, power will continuously increase on

a period determined by the amount of reactivity added even after the rod reaches its limiting position. In the case of a boiling reactor, steam voids will form sufficient to remove the reactivity added by the rod and the power rise will be limited. Although the formation of steam voids normally results in a negative reactivity effect, it is possible to obtain a positive

void coefficient under certain fuel-to-water ratio conditions. Most boiling water reactors are designed to have a negative void coefficient from approximately room temperature to operating temperature.

(2) METAL-WATER REACTIONS (97-101)

All water reactors are subject to the problem of a metal-water reaction. This type of exothermic reaction can occur between water and zirconium, stainless steel (chromium only), uranium metal and aluminum. Experimental data have indicated that, when small droplets of the above metals are dispersed in water, such reactions can occur. In the design of the containment building and holddown features for boiling reactors, consideration is given to this remote possibility.

When metallic fuels are used in water reactors, it is reasonable to assume that this fuel can reach the melting point during an excursion and possibly be dispersed so as to cause a metal-water reaction; this has been one of several possible explanations of the BORAX-I destruction experiment. Since UO_2 does not undergo an exothermic reaction in water, it represents a much safer fuel material, leaving only the possibility of the clad entering such a reaction. The possibility of the Zr clad, which is in contact with water, reaching its melting point and becoming dispersed in the water is extremely remote.

(3) STABILITY

In any reactor system, if a change in operating power results in a change in reactivity, the system is said to have a power coefficient of reactivity. Most reactor systems will have several important power coefficients, and it is the magnitude and response time of these coefficients which determine the stability of a system. As in the case of any servo system, the characteristics of these coefficients can be experimentally determined by sinusoidally oscillating the reactor power as a function of the

parameter under study, and over a given frequency range. The resultant amplitude and phase of the reactivity change as a function of frequency determine the transfer function of the particular power coefficient. Experimental data have been obtained of the transfer function of EBWR (111, 112, 113) and these have been successfully applied to predict a further safe operating power. However, there was some discrepancy between the experimentally determined data and the calculated feedback parameters of EBWR. Thus, direct extrapolation of EBWR data is valid for EBWR but is not applicable for predicting the calculated stability of other boiling water concepts.

The chugging phenomena of the BORAX reactors was the first indication of stability limitation in boiling reactors. BORAX-I showed a definite oscillatory behavior when ~ 1.6 percent k_{eff} was held in voids at atmospheric pressure. At 135 psig, BORAX-I showed an oscillating behavior with ~ 2 percent k_{eff} held in voids, which indicated more stable operation at higher pressure. BORAX-IV also showed oscillatory tendencies at atmospheric pressure with ~ 1 percent reactivity in voids while at 322 psig, it operated stably with 6.9 percent reactivity. EBWR has demonstrated better stability at pressures between 150 psig and 600 psig than below 150 psig. VBWR has also reported excellent stability at the 1,000 psig operating pressure.

The effect of the steam void coefficient of reactivity on stability of reactors having fuel of relatively short time constants (0.006 second) was demonstrated by operation of BORAX-II and BORAX-III (2-11) at 300 psig and a power density of 22 Kw/liter. At this power density BORAX-II had 2.6 percent k_{eff} in steam voids and the power fluctuated ~ 15 percent. BORAX-III had only 1.5 percent k_{eff} held in steam for the same power density and the power fluctuation did not exceed 4 percent.

More recent data obtained on BORAX-IV (1, 10) may indicate that one of the parameters which affects the maximum power operation of a boiling reactor, with respect to stability, is

the rate at which voids increase as a function of power.

A plot of power density for BORAX-IV versus reactivity in voids showed that at 300 psig, the rate of change of reactivity in voids with respect to power was decreasing and at 95 psig the rate of change of reactivity with respect to power was increasing. The reactor is more stable at the higher pressure, and it appears that a reactor may be more stable if the second derivative of the power versus reactivity in voids is positive. More data on different reactors are required to substantiate this conclusion.

A significant parameter which has a large effect on the stability is the thermal time constant of the fuel. Thermal time constants of various reactors are: 0.006 second for BORAX-III, 0.9 second for BORAX-IV, and 0.8 second for EBWR. Stable operation was obtained in BORAX-III at 300 psig with 2.0 percent k_{eff} maximum held in voids, in BORAX-IV at 300 psig with up to 6.9 percent k_{eff} maximum held in voids, and in EBWR at 600 psig with 4.1 percent k_{eff} maximum held in voids.

In the case of BORAX-IV and EBWR, it would appear that the fairly fast boiling noise fluctuations of reactivity are filtered out by the longer time constant of the fuel, which allows a greater amount of reactivity to be held in voids. In the case of the short time constant of BORAX-III, the power change in the reactor due to a boiling noise fluctuation in reactivity is fed back to the water to form more or less voids. When the negative feedback is 180° out of phase with the initiating signal, unstable operation can result. In essence, the long time constant acts as an inertia to keep the boiling system from changing rapidly with rapid changes in reactivity.

The stability of a boiling water reactor is also determined by hydraulic considerations. In boiling reactors with long time constant elements, it is probable that the hydraulic stability will determine the maximum power. Out-of-pile tests by ANL (1) have indicated

that hydraulic instability occurs at ~70 percent exit voids from a channel. Data from EBWR show that the predicted instability from transfer function test analysis occurs at nearly the same power as the equivalent out-of-pile (70 percent exit voids in several channels) flow instabilities. One method of improving the dynamic response of the hydraulic system is to introduce an orifice outside the core zone. This has been done by Russians in the superheater reactor. Of course, the restriction requires a larger pressure head to be generated for the same flow through the core. In the case of forced circulation reactors, this extra head is no problem.

f. Components and Auxiliary Systems

In general, boiling reactors have been able to utilize components and auxiliaries already developed for pressurized water reactors and conventional steam plants. Much of the specific development work required to date can be attributed to questions raised by the presence of radiolytic oxygen in the system and the use of saturated steam in the direct cycle.

PRESSURE VESSELS

The largest pressure vessel built for a boiling reactor to date is the 12 foot-2 inch I.D., 283-ton vessel for the Dresden Nuclear Power Station, which is designed for 1,250 psig (2). It has walls 5¼ inches thick, made of low alloy steel (ASTM A302B) and is clad with a ⅜ inch layer of type 304 stainless steel. The vessel head is held in place by 56 five-inch bolts and is sealed with two O-ring type gaskets. The top head is removed for servicing the reactor, providing an 11 foot-7½ inch opening through which any part of the core assembly can be removed, if necessary. Fabrication and preliminary pressure testing of this vessel have been successfully completed.

Smaller vessels have been completed for the VBWR and EBWR, weighing approximately 60 tons each. The 84-inch I.D. VBWR vessel

was made of SA 264 steel plate, lined with 304 L stainless steel to give a total wall thickness of $3\frac{3}{8}$ inches (2). This vessel was designed for a working pressure of 1,215 psia and has a bolted and gasketed head sealed with double soft iron gaskets.

The 84-inch I.D. EBWR vessel is made of SA-212 Grade B carbon steel with a Type 304 stainless lining, and is designed for a maximum pressure of 800 psig (22). The vessel head consists of a flat cover plate $8\frac{1}{16}$ inches thick, held in place by 44 bolts, $2\frac{1}{2}$ inches in diameter. Two concentric flat gaskets maintain the seal between the vessel and the cover plate. After testing a variety of gaskets, the EBWR staff developed a spiral wound metallic gasket filled with teflon and asbestos, which performed satisfactorily. Many other designs tested did not perform well, and obtaining a satisfactory closure using flat gaskets was apparently quite difficult.

Field tests on the EBWR vessel also demonstrated that galling of closure bolts and washers could be avoided by the use of a graphite-mineral oil suspension.

Some rusting of the EBWR vessel cladding has been observed, with pits 21 to 26 mils deep occurring in places (1). Also, instrumentation tubing (30 mil wall) passing through one of the nozzles in the vessel head was found to be perforated and extremely brittle at the end of a 90-day experimental run. General inspection of the vessel cladding is not possible because of thermal shields, etc., so the extent of the deterioration of the vessel lining is not fully known.

Development work on high-strength alloys for pressure vessels was undertaken by ANL (1). The objectives of this program were to reduce vessel weight, reduce fabrication costs, insure uniformly desirable properties, and reduce heat generation in vessel walls. As a result of cooperative work between ANL, U.S. Steel, and Babcock & Wilcox, two promising steels are now being investigated. One is the well known $2\frac{1}{4}$ Cr-1 Mo alloy; the other is known as HY-80. The $2\frac{1}{4}$ Cr alloy has been

produced in 6-inch gage plate with 105,000 psi tensile strength and with an impact strength of 15 ft-lbs at -50° F. at the $\frac{1}{2}$ gage point. Welding procedures have been devised by Babcock & Wilcox which permit the welding of 6-inch plates without any sacrifice in strength. Test plates are now being readied for irradiation tests in the Materials Testing Reactor (MTR). Properties of HY-80 are also quite promising. Impact strength of specimens taken from $\frac{1}{4}$ and $\frac{1}{2}$ gage points from a large slab exceeded 15 ft-lbs at -100° F. Welding tests are currently being made on this material by Babcock & Wilcox and irradiation tests will follow. Work is also being considered to determine high-temperature creep strengths for the Hy-80 material.

(2) PIPING, PUMPS, AND VALVES

To date, reactor piping has been conventional Type 304 stainless steel designed to Section VIII of the ASME Code. The chief problem has been to obtain enough flexibility in these heavy lines to prevent excessive stresses from being developed in going from the cold to the hot conditions.

Recently, centrifugally cast stainless steel pipe has been produced with strengths superior to that of wrought seamless pipe (1). Improved strength and weldability are obtained by intentionally maintaining slightly higher chromium and slightly lower nickel than the normal 18-8 stainless. Metallurgically, this produces an alloy which is about 90 percent austenite plus 10 percent delta ferrite. Long-term testing will be required to establish high-temperature creep properties, etc., but the results to date are promising.

Consideration has also been given to the use of lower steel alloys in place of austenitic stainless steel for piping, pumps, etc. Lack of control of pH and oxygen content may make this substitution less attractive for boiling reactors than it is for pressurized water types. However, if crud problems can be avoided, the lower alloys should have promise, especially

for the more expensive components such as pumps and valves.

Little development work has been done on pumps and valves specifically for boiling reactors. Both canned pumps and pumps with mechanical seals are available from other development programs, however. Pumps with mechanical seals are less expensive, but as might be suspected, the chief uncertainty is obtaining satisfactory performance of mechanical seals at high temperatures and pressures, and more work on seals would be desirable.

Valves have received even less attention than pumps because there are off-the-shelf designs available to meet most of the requirements.

(3) TURBINE AND CONDENSER

Corrosive and erosive effects of wet oxygenated steam have been the principal concern in the designs of turbines for service with boiling reactors. The work of Cataldi, et al, (114) and experience with the EBWR turbine have done much to allay early concern over these effects.

Originally, the EBWR turbine and steam piping were plated by the Kanigen process in the belief that the nickel plate would provide corrosion protection. Later examination showed that the nickel has been completely removed in many places, but otherwise turbine wear has not been excessive.

In the design of the Dresden turbine, several ground rules were adopted to avoid problems (115):

(a) Moisture content of steam was held within acceptable limits by use of throw-out stages and moisture separators.

(b) Stage energy and pressure drops were held to about one-half the normal values for the high pressure section.

(c) Leakage paths subject to wiredrawing were eliminated wherever possible, and where they could not be eliminated, they were faced with erosion resistant materials.

(d) Connections were provided for introducing decontamination solutions for turbine cleanup.

As a result of the development work and EBWR experience, manufacturers are now considerably more confident of the performance of turbines using steam directly from a boiling reactor.

Condenser designers have been able to utilize recent developments in the condenser field to good advantage in nuclear plants. Welded tube construction which is desirable for other types of plants is now available as a proven technique with exactly the characteristics needed for nuclear applications. De-superheaters to permit dumping steam directly to the condenser are also desirable for once-through boilers; so, again, the nuclear plants can take advantage of developments for conventional plants. The presence of high oxygen in the steam has apparently not produced serious corrosion in the EBWR condenser, indicating that conventional materials are satisfactory for nuclear service.

(4) ROD DRIVES

A variety of rod drives are available for boiling reactors, including screw types, rack and pinion, magnetic jacks, and hydraulic units. Each of the above types has been subjected to extensive tests and, in some cases, actual reactor operation.

In the EBWR, bottom-mounted drives were used, employing reciprocating shafts working through linear seals. The reciprocating shafts were driven by external screw jacks. In the course of EBWR operation, problems were encountered with crud collecting in the nylon guide bushings and restricting the movement of the rods. Redesign of these bushings, using teflon instead of nylon, eliminated movement difficulties, but crud settling into the seal housings has caused very high radiation levels (as high as 40 r at 2 inches). The existing rod drives are now scheduled to be replaced with rack and pinion drives with rotary seals similar to the drives used on APPR. The rack and pinion drives are designed so that they be

flushed out, and it is hoped this will prove to be an effective means of eliminating crud (1).

The magnetic jack-type drives will be tested under actual operating conditions, both on the EBWR and the ALPR. To date, their most serious limitation has been lack of a satisfactory means of indicating rod position.

Hydraulic drives are planned for the Dresden reactor. These drives use a piston arrangement with full system pressure on both sides of the piston. Movement is achieved by creating slightly unbalanced pressure across the

piston. This type of mechanism is capable of a very fast scram action. In the case of Dresden, the scram insertion time is only 1.5 seconds.

(5) CONTAINMENT

Except for test reactors operated at remote locations, all boiling reactors constructed to date have been provided with steel containment shells which were designed on the basis of extremely pessimistic assumption. Design data for containment buildings for boiling reactors as summarized below (116):

Reactor	Shape	Volume, ft ³	Design Pressure, psig	Material and thickness	Leakage*
EBWR.....	Cylinder.....	400, 000	15	A201 (A300) 5/8 in.	0. 25
VBWR.....	Cylinder.....	125, 000	45	SA-212 3/8 in.	1. 0
Dresden.....	Sphere.....	3, 600, 000	29. 5	A201 (A300) 1.4 in.	0. 5

*In units of "percent of shell volume (S.T.P.) per 24 hours."

All of the above designs are based on calculated maximum pressures resulting from instantaneous release of primary fluids taking no credit for pressure reductions due to heat lost to the building.

(6) FUEL FAILURE DETECTION

Although most of the reactor auxiliaries have required very little development, fuel failure detection and location have presented some unique problems. Detection of fuel failures has been accomplished quite effectively by monitoring the gases leaving the condenser air ejectors for Xe-135. Location of fuel failures has been more difficult due principally to the fact that the ceramic fuels do not react appreciably with water when the cladding breaks, and once the initial puff of active gas has been released, the faulty element bleeds at a rather low rate. ANL experience has shown that by raising and lowering the power in a faulty element, it can be made to release bursts of activity, presumably due to expansion effects connected with the temperature variations (1). Thus, it may be possible to determine approximate location of faulty elements by varying control rod positions and observing changes in air exhaust monitor readings. Attempts to lo-

cate samples by inserting probes to sample water above faulty fuel during operating met with no success in actual plant tests in EBWR. Obviously, more work will be required on this important problem.

2. Experimental Reactors (2-19)

Complementing the general research and development on boiling water reactors was the successful experimental demonstration of the feasibility of the boiling concepts. The early concepts were known as the BORAX reactors and are discussed briefly below.

a. BORAX-I (Boiling Reactor Experiment I)

(1) DESCRIPTION

This reactor, constructed at National Reactor Testing Station, Idaho in 1952, consisted essentially of a core of Al clad, highly enriched U-Al fuel plate assemblies contained in a steel pressure vessel. Light water was used for the coolant, moderator, and reflector, the water height being ~4 feet above the core. Four control rods were designed to hold the reactor subcritical and the fifth (center) rod, was designed to be ejected rapidly from the core under experimental conditions. Cooling was by natural

circulation within the vessel, with the steam released to the atmosphere.

(2) PURPOSE

This reactor was basically an experimental tool to test the feasibility of the boiling water concept and check the inherent shutdown mechanism due to void formation.

(3) CHRONOLOGY

See Figure 18.

(4) SIGNIFICANT DESIGN DATA

See Table 4.

(5) SIMPLIFIED FLOW DIAGRAM

Since there is no electrical plant attached to this reactor, no flow diagram is included.

(6) OPERATING EXPERIENCE

In the short life span of BORAX-I, a large number of excursion type tests and a small number of steady-state boiling experiments were run. These tests demonstrated that boiling water reactors could operate stably and had a high degree of inherent safety due to shutdown by void formation. Power transients with progressively shorter periods (as low as 0.005 second) were obtained in these tests by the rapid addition of reactivity. In these cases, the reactor shut down by steam void formation; further excursions during the test were prevented by insertion of safety rods. In the short period excursions, some deformation of the fuel plates occurred due to pressure transients in the core.

BORAX-I was also operated as a steady-state boiler with ~ 1.5 percent k_{eff} held in steam voids. Although fluctuations in power

FIGURE 18—Schedule for Boiling Water Reactors.

TABLE 4

SUMMARY OF SIGNIFICANT DESIGN PARAMETERS FOR THE BORAX REACTORS

	BORAX-I	BORAX-II	BORAX-III	BORAX-IV	
Contractor.....	ANL	ANL	ANL	ANL	
Thermal power, max., tMw.....	1. 2	6. 0	16. 0	5	20. 5
Electric power, max., eMw.....	-----	-----	3. 5	-----	3. 5
Pressure, max., psig.....	135	300	300	Atm	322
Core volume, liters.....	120	222	592	255	455
Core volume, ft ³	4. 24	7. 88	21. 0	9. 0	16. 1
Fuel material.....	U-A1	U-A1	U-A1	UO ₂ -ThO ₂	
				UO ₂ -90%	
				Fuel-7%	
Enrichment.....	90%	90%	90%		
Fuel element:					
Clad.....	Al	Al	Al	Al+Pb bond	
Plate spacing, in.....	0. 177	0. 324	0. 324	0. 604	
Plates per subassembly.....	18	10	24	6	
Size of subassembly, in.....	3 x 3	3 x 3	3 x 3	3 x 3	
U-235 per subassembly, gms.....	138. 6	93. 4	137	283	
Heat transfer area per unit core volume, ft ² /ft ³	120	66. 7	63. 9	34	
H ₂ O to metal ratio in core.....	1. 59	2. 04	2. 78	1.92 (incl. UO ₂)	
H ₂ O fraction of core.....	0. 614	0. 672	0. 735	0. 656	
Temperature coefficient of reactivity (80- 360° F.), %k _{eff}	3. 5	1. 16	0. 8	2.5 (measured)	
Void coefficient (calc.), %k _{eff} /° steam.....	-0.24 (207° F.)	-0. 10	-0. 10	-0.32 (207° F.)	
				-0.28 (421° F.)	
Effective neutron lifetime, 10 ⁻⁵ sec.....	6. 5	7. 5	7		
Reactivity in steam voids, max., %k _{eff}	2. 0	3. 1	2. 0	(Atm) 1. 25	(322 psig) 6. 9
Power density, max.:	(Atm)	(135 psig)			
Kw/liter of core.....	8. 4	10. 0	27	27	19. 6 45
Kw/ft ³ of core.....	238	283	765	765	555 1270
Kw/liter of coolant.....	14. 7	17. 5	38. 5	38	30. 5 67. 5

of the order of 15 percent were noted, no instability tendency was detected. Beyond ~1.5 percent k_{eff} in voids, a pattern of oscillations was established which led to unstable operation or "chugging" when ~2.5 percent k_{eff} was held in voids at atmospheric pressure. Operation at higher pressure increased stability and the reactor operated stably at ~1,200 kw, 135 psig. This represented a power density of ~10 kw/liter or ~283 kw/ft.³

A final experiment, which resulted in the self-destruction of BORAX-I, was made by adding ~4.0 percent k_{eff} in 0.2 second with the reactor water initially at room temperature. The power during this test increased on a period of 0.0026 second and reached a maximum of 18,000 megawatts. A total energy release

of 135 Mw-sec was obtained. No reliable records of maximum fuel plate temperature or reactor pressure are available for this experiment. Examination of the damaged pressure transducer indicated a peak pressure of >6,000 psig. Rupture of the vessel and other component damage also indicated high pressure. Photographic records showed that a light flash associated with the nuclear excursion lasted about 0.003 second and was extinguished before any ejected material appeared above the top of the shield tank. This indicated the nuclear shutdown was accomplished rapidly. It is significant to note that relatively little damage to the equipment outside the pressure vessel or to the control rod drive mechanisms above the vessel occurred.

Examination of the debris around the reactor showed that a large fraction of the fuel plate material had been melted. However, it is not known whether an explosive chemical reaction occurred between the molten metal (Al and U) and the water. If such a reaction did occur, it was far from complete, as evidenced by the large quantity of unoxidized aluminum fuel fragments.

b. BORAX-II (Boiling Reactor Experiment II)

(1) DESCRIPTION

This reactor was similar to BORAX-I but had a core approximately twice the size of BORAX-I and a pressure vessel for operation of the reactor at 300 psig. Fuel elements were again U-Al alloy clad with aluminum, but the plates were spaced on 0.324-inch centers as compared to the 0.177-inch plate spacing of BORAX-I.

(2) PURPOSE

The main objective of the BORAX-II was to continue the study of the natural circulation boiling reactors at higher pressure.

(3) CHRONOLOGY

See Figure 18.

(4) SIGNIFICANT DESIGN DATA

See Table 4.

(5) SIMPLIFIED FLOW DIAGRAM

Since there is no electrical plant attached to this reactor, no flow diagram is included.

(6) OPERATING EXPERIENCE

This reactor was only operated for a short time in 1954 and 1955 before it was decided to install a turbine generator. During the short operation, tests similar to those conducted on

BORAX-I were made at atmospheric and at 300 psig pressures. The tests served to further indicate that stable steady-state boiling operation at power densities suitable for power reactors could be obtained with pressurized, boiling water reactors. Stable operation under steady-state boiling was obtained, in BORAX-II, pressurized to 300 psig, with a power density of ~ 27 kw/liter of core (765 kw/ft³) with up to 3.2 percent k_{eff} in voids. Tests were also conducted with this reactor to determine the response of the reactor when the main steam valve was closed suddenly and when the safety valves opened suddenly. When the main steam valve was closed the operating pressure and reactor power increased to a new level in a stable manner. Sudden opening of the safety valve caused pressure and power to fall. When the valve reset itself the power would readjust to approximately its initial value. Because of the stable reactor behavior during the tests conducted, it was decided to install a turbine-generator plant with this reactor.

c. BORAX-III (Boiling Reactor Experiment III)

(1) DESCRIPTION

This reactor was essentially the same as BORAX-II with new fuel elements and a turbine-generator added. The new fuel elements were structurally stronger and had a larger water-to-metal ratio (2.78) than the fuel in BORAX-II (2.04) or BORAX-I (1.59).

(2) PURPOSE

The main purpose of BORAX-III was to study the operation of a boiling water reactor as a component in a power generation system.

(3) CHRONOLOGY

See Figure 18 and Figure 19.

(4) SIGNIFICANT DESIGN DATA

See Table 4.

FIGURE 19—Operating history of BORAX III and BORAX IV.

(5) SIMPLIFIED FLOW DIAGRAM

Figure 20 is representative of the single, direct cycle within this plant.

(6) OPERATING EXPERIENCE

The BORAX-III power plant was successfully operated at maximum power levels of 16 tMw and 3.5 eMw and represented the first boiling water reactor to generate electric power. Over a period of 1½ years of life it accumulated a total of ~13,000 tMwh and ~1,600 eMwh. (See Fig. 18). It operated at a power density the same as BORAX-II, 27 Kw/liter of core (765 Kw/ft³) but at a higher power density per percent reactivity in voids.

The reactor plant operated stably at all pres-

ures from atmospheric to 300 psig and achieved a maximum power generation of 16 tMw. Other tests performed on this reactor were concerned with:

Water Quality Control—This was not a major problem, although better quality control is required in a reactor system than is usually required in a fossil fuel boiler.

Radioactivity—Radiation levels during operation at 8 tMw as shown in Table 5 are relatively low. After shutdown the radioactivity level dropped rapidly to a negligible amount.

Water Decomposition—This problem was studied because of the possible corrosion effects of the oxygen on the reactor and turbine components. Some data were obtained

which showed that the addition of hydrogen decreased the oxygen evolution by a factor of ~ 20 . This method, however, requires a large amount of gas to be handled and would probably be uneconomical.

TABLE 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 resistiv-			
ity, ohms/cm ³	270,000	490,000	150,000

Survey Location

Steam line-reactor build-			
ing, 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

BORAX-III accumulated a total operational lifetime of 13,000 tMwh and 1,600 eMwh over a period of 1½ years as shown in Figure 19. This was insufficient operational time to obtain any data as to long-term maintenance problems to be expected with these reactors.

d. BORAX-IV (Boiling Reactor Experiment IV)

(1) DESCRIPTION (1, 2, 10, 11)

BORAX-IV was the BORAX-III reactor plant with a new core substituted for the BORAX-III core. The new core consisted of fuel elements using ThO₂-UO₂ fuel canned in aluminum and bonded with lead.

(2) PURPOSE

This reactor had as its initial purpose the testing of a new type of fuel element developed

FIGURE 20—Simplified flow diagram direct cycle.

by ANL ($\text{ThO}_2\text{-UO}_2$ fuel fired in air). A second purpose was to study the behavior of this reactor for possible application of the results to the EBWR. Although the EBWR uses metallic uranium alloy clad with Zircaloy-2 as its fuel element, the thermal time constant of the EBWR element (~ 0.8 sec) was comparable to the BORAX-IV element (~ 0.9 sec).

(3) CHRONOLOGY

See Figure 18 and 19.

(4) SIGNIFICANT DESIGN DATA

See Table 4.

(5) SIMPLIFIED FLOW DIAGRAM

Figure 20 is representative of the single, direct cycle used in this plant.

(6) OPERATING EXPERIENCE

The BORAX-IV reactor went critical in December 1956 and has operated for a total of $\sim 7,000$ tMwh and ~ 800 eMwh as indicated on Figure 19. Primary emphasis in the experimental program of this reactor was given to obtaining more detailed data on the stability and safety of boiling water reactor plants. Transient and steady-state power runs similar to the previous BORAX experiments were made and reactor transfer function measurements were made using oscillator techniques. At atmospheric pressure, this reactor was found to be unstable at a power level of 4.57 tMw with 1.5 percent k_{eff} held in voids. As the pressure was increased to 322 psig, stable operation was achieved up to 20.5 tMw with 6.9 percent k_{eff} held in steam voids.

Maximum power densities achieved were: 19.6 Kw/liter of core (555 Kw/ft³) at atmospheric pressure with 1.5 percent k_{eff} in voids and 45 Kw/liter (1275 Kw/ft³) at 300 psig with 6.9 percent k_{eff} held in voids.

Besides the steady-state and transient operating data, BORAX-IV has provided infor-

mation on radiolytic decomposition (Figure 17) and experimental data on the contamination of the reactor coolant loop due to fuel clad failures. These fuel clad failures were detected in February 1958 by the sudden increase in activity in the turbine room to 70 mr/hr. Activity levels of 400 mr/hr were noted in the turbine room and 8,000 mr/hr in the hot well following further reactor operation with the failed elements. Examination of the BORAX-IV fuel elements later indicated that at least 17 elements had failed. These failures were attributed to leaks caused by the collapse of the aluminum into the void space above the oxide pellets after repeated thermal cycling in the reactor. It is significant that operation with the failed fuel elements, although causing a relatively high activity level in the plant during operation, caused no significant contamination of the turbine and little increase in the gross activity of the water. The radioactivity in the turbine was very low when the reactor was shut down. This observation would indicate that mostly gaseous fission products were released by the oxide elements.

Surface activities at various points in the BORAX-IV plant were measured while the reactor operated with the failed fuel elements. The results of these measurements are shown in Table 6 for various operating powers and for 17 hours after shutdown.

Other difficulties encountered with this plant were due to control rod drives, instrument cables, and other components. These difficulties, however, do not represent operational problems for boiling reactors since these components were part of previous BORAX reactors and had been designed for short-term use.

e. SPERT (*Special Power Excursion Reactor Tests*) (13-19)

To provide information on water reactor hazards, a special program, SPERT, was initiated by the AEC for reactor safety investigations. This program is a continuation of the BORAX experiments and is designed to pro-

TABLE 6
BORAX-IV SURFACE ACTIVITIES IN mr/hr DURING OPERATION WITH FUEL DEFECT

	<i>Before operation</i>	<i>2.4 Mw (thermal)</i>	<i>6.0 Mw (thermal)</i>	<i>17 hours after shutdown</i>	<i>Normal activities at 12 Mw (thermal)</i>
Steam pipe turbine building	0	250	400	0	200
Turbine exhaust casing (external)	0	150	190	0	50
Turbine exhaust casing (internal)	0.7 γ 15 $\beta + \gamma$			0.7 γ 15 $\beta + \gamma$	
Condenser hotwell	0	2,200	5,000	1.5	500
Air ejector after-cooler	0	15,000		0	420
Air ejector discharge (2- by 2- by 1-ft, AEC type filter)	0	30,000	50,000	1	

vide comprehensive information on the kinetic behavior of light water moderated reactors using highly enriched fuel. The reactor operated at atmospheric pressure with natural circulation cooling. Five blade type rods are used as in the BORAX experiments. In general this facility provides data similar to, but more refined than, that described under the BORAX experiments. Since testing is at atmospheric pressure the data is not expected to add significantly to the design of low enrichment oxide fueled high pressure boiling water reactors. Details of this reactor are therefore not included here although a bibliography of reference reports is supplied.

3. Power Demonstration Reactors

a. EBWR (*Experimental Boiling Water Reactor*) (1, 2, 20, 21, 22)

(1) DESCRIPTION

The Experimental Boiling Water Reactor at Argonne National Laboratory, Lemont, Ill., is a complete experimental nuclear power generating station designed to provide operating and design data on boiling water reactors suitable for power generation.

The EBWR reactor is a light water moderated and cooled, natural circulation boiling water reactor which normally produces 20 Mw of heat and 60,000 lb/hr of 600 psig steam for the generation, in a direct cycle, of 5 Mw electrical power. The heat output capability of the EBWR has been demonstrated to be greatly in excess of the design rating; the EBWR has been

operated stably at thermal power levels up to 61.7 Mw for a short period.

The EBWR core is essentially a right circular cylinder, 4-foot diameter, 4 feet high. The core contains 113 fuel assemblies of the flat-plate type. Since no critical experiments preceded the loading of fuel into the EBWR, two plate thicknesses and two fuel enrichments (natural and 1.4 percent) were used to cover uncertainties in optimum moderator to fuel ratio and critical enrichment. The fuel within the plates is a metallic alloy of uranium with 5 percent by weight of zirconium and 1.5 percent of niobium. The fuel plates are clad with 0.020-inch thick Zircaloy-2.

Nine cruciform control rods, with 10-inch span are provided for power regulation and shutdown control. The central five control rods are made of 1/8-inch thick hafnium and the outer four of 1/4-inch thick, 2 percent boron stainless steel; Zircaloy-2 extensions are attached below the poison absorber section. The control rod drives, mounted below the reactor vessel are the roller-latch, lead-screw type.

The stainless clad SA 212 Grade B pressure vessel is a 7-foot I.D. cylinder with a semi-ellipsoidal bottom and a flat plate gasketed closure head. A 1-inch thick 1 percent boron stainless steel thermal shield is provided to minimize the thermal stresses in the 2 1/4-inch thick vessel wall during future operation with heavy water. The biological shield surrounding the pressure vessel consists of 3 inches of lead, 5 feet of magnetite concrete and 3 feet of light concrete.

The reactor and turbine-generator equipment are housed in a steel cylindrical vessel 80 feet in diameter and 119 feet high with a semi-ellipsoidal bottom and hemispherical top head. Approximately half the vessel height is located below ground level. It is designed to withstand an internal pressure of 15 psig. Below the ground level main floor the vessel is lined with 2 feet of concrete to resist penetration by accidental missile fragments. Above the main floor the lining is 1-foot thick up to the level of the concrete ceiling slot.

The turbine, generator, and main pumps are located within the reactor containment building. Waste heat is dissipated in a cooling tower located outside.

Control of the reactor and power generating equipment is from a control room in an adjacent service building. This building also contains the electrical switchgear, a laboratory, offices, and sanitary facilities.

(2) PURPOSE

The EBWR is primarily an experimental plant intended to provide as much information as possible for use in designing large central station units. An unusual amount of flexibility has been incorporated in the design of the EBWR. The unit is designed to permit testing of reactor cores up to 5 feet in diameter. Provision for operation with forced circulation cooling and with heavy water have also been included in the design. Valuable information intended to be gained by the operation of the EBWR includes system stability, radioactivity

of components, corrosion and erosion, leakage, water quality control, component maintenance, water decomposition, etc.

(3) CHRONOLOGY

See Figure 18 and Figure 21.

(4) SIGNIFICANT DESIGN DATA

See Table 7.

(5) SIMPLIFIED FLOW DIAGRAM

Figure 20 is a simplified schematic flow diagram illustrating the direct cycle type of operation of EBWR.

(6) OPERATING EXPERIENCE

(a) Power Generation

As of January 1, 1959, the cumulative thermal and gross electrical energy generated by EBWR were the following:

Thermal...	180,000 Mwh
Electrical...	28,000 Mwh

The experimental program for the EBWR facility requires frequent shutdown periods for preparation and carrying out of planned experiments, which makes it different from a similar size powerplant specifically devoted to steady state electric power generation. This type of operation is reflected in the relatively low thermal and electrical energy accumulated since the start of power generation in January 1957.

FIGURE 21—Operating history of Experimental Boiling Water Reactor.

TABLE 7
SIGNIFICANT DESIGN DATA OF BOILING WATER REACTORS

1 REACTOR PLANT		EBWR	VBWR	ALPR	RCPA	NSP		Dresden	SENN	Russian Superheater	
						Boiling	Superheater			Boiling	Superheater
2 POWER, Thermal	Mw	20	30	3	73	164	39	626	507.8	285	
Electrical, gross	Mw	4.6	5.2	0.3	23	66		192	160	100	
Electrical, net	Mw	4.1			22	62		180	150		
3 EFFICIENCY, Thermal											
Gross	%	23.0			31.6	32.5		30.6	31.5	35.1	
Net	%	20.5			30.2	30.5		28.7	29.5		
4 MODERATOR		H ₂ O	H ₂ O	H ₂ O	H ₂ O	H ₂ O		H ₂ O	H ₂ O	Graphite	
5 COOLANT		H ₂ O	H ₂ O	H ₂ O	H ₂ O	H ₂ O	Steam	H ₂ O	H ₂ O	H ₂ O	Steam
Pressure	psig	600	1000	300	922	600		1000	1000	2265	1605
Temperature, avg. core inlet	° F	484		418.2	534.1	486	489	545.3	512	572	607
avg. core outlet	° F	489	546	420	536.5	489	825	505	546	655	950
6 CORE SIZE, Length	ft	4	3.08	2.15	5.0	6	6	9.75	9.3	19.7	
Diameter, equivalent	ft	4	2.5-3.5	2.62	4.52	6 O D - 2.5 I D	2.5	10.37	9.16	23.6	
Volume, core	ft ³	50	15.1-29.6	11.55	80.2	140	29.5	820	614	8600	
7 FUEL											
Weight, fuel total (UO ₂ or UO ₂ +ThO ₂)	10 ⁶ grams	Metallic	0.022	0.00171	4.54	6.66	0.0303	59.3	47.0	Metallic	
Weight uranium (U or U+Th)	10 ⁶ grams	5.5	0.019	0.00151	0.18+3.82	5.87	0.0267	52.3	41.4	90.0	
Enrichment	wt %	1.2	92	91	4.5	1.7	92	1.5	1.84	1.34	
Temperature, maximum, centerline	° F	660	600		3420	4200	1095	3840	3866	752	1022
Clad, material		Zr-2	SS-304	X8001 A1	SS-304	Al-Ni alloy	SS	Zr-2	Zr-2	SS	SS
Clad, temperature maximum	° F	515	575	440	551	550	1080	585	574	670	987
8 HEAT FLUX											
Maximum	Btu/hr-ft ²	155,000	500,000	75,250	314,400	432,000	170,000	277,000	285,800	194,000	177,000
Average	Btu/hr-ft ²	45,500	111,100	21,500	91,000	131,000	85,000	94,700	97,600		
Ratio, max/avg		3.41	4.5	3.5	3.45	3.3	2	2.93			
9 POWER DENSITY, Total core											
	Kw/ft ³	398	1415	260	724	1170	1320	765	827	33.1	
	Kw/liter	14.1	50	9.1	25.6	41.4	46.7	27	29.2	1.17	
10 SPECIFIC POWER, Thermal											
Fuel, avg	Mw/10 ⁶ gm	3.38		1.755	12.8	24.6	1.28	10.6	10.8		
U or (U+Th)	Mw/10 ⁶ gm	3.62			14.5	27.9	1.46	12.0	12.26		
11 CONVERSION RATIO, Initial		0.74	Zero					0.7	0.721	0.65	
12 CONVERSION FACTOR											
Equilibrium core at rated burnup	gms Pu/Kg U				7.34 (U+Th)			5.0 (10,000 MwD/T)			

(b) System Radioactivity

The radiation levels in the EBWR plant as of January 9, 1958 were as follows: (Plant operating at 20 tMw).

Location	Activity in mr/hr
Top of reactor -----	1
Steam dryer -----	150
Turbine Inlet -----	50
Turbine casing -----	4
Air ejector after cooler ----	400
Condenser hot well -----	40
Stack exhaust duct -----	3
Sub-reactor room -----	10
Feedwater filters -----	10
Ion exchanger pre-filter	
(inside shielding) -----	500/1000
Ion exchanger	
(inside shielding) -----	500/1000
Ion exchanger after filter	
(not shielded) -----	500/1000
Startup heater -----	400

The above readings showed no measurable change from similar readings over the previous 6 months.

One day (24 hours) after shutdown the following typical activity levels are noted.

Location	Activity in mr/hr
Steam dryer -----	0.5
Main steam piping -----	0.2
Turbine trip valve -----	0.2
Condenser:	
Steam dome -----	0.2
Hot well -----	2.0

The activity levels encountered while servicing equipment after a reactor shutdown are the following:

Location	Activity in mr/hr
Feedwater filters -----	5
Control rod bushings -----	2
Ion exchanger:	
Top of resin bed -----	1,000
Pre-filter -----	500
After-filter -----	500
Inside turbine (downstream of nozzle block) -----	0.1 to 0.2

(c) Equipment Problems

On three occasions failure of the bearings in the feed-water pump occurred. This is strictly

a problem of pump design rather than a difficulty peculiar to a nuclear plant.

Frequent difficulty with "sticking" of the control rods was experienced during the early operation of the EBWR. This problem is associated with "crud" accumulation in the rod seal housing at the bottom of the reactor vessel. It has been largely corrected by substituting bushings made of softer material.

This crud accumulation in the control rod thimbles results also in a high activity level in the immediate vicinity of the control rod thimbles. Activity levels of the order of 65-70 r/hr have been reported. This would probably not be a major problem if provisions had been made for flushing the control rod thimbles.

Difficulties were experienced with the H_2-O_2 catalytic recombiner during the early operation of the EBWR. Frequent explosions of the hydrogen-oxygen mixture have been experienced. This system is now operating satisfactorily.

Failure of the stem of the turbine trip throttle valve occurred. A combination of improper operating procedures and incorrect design of the stem were given as reasons for the failure.

A turbine blade failure, not associated in any way with nuclear phenomena was experienced. Servicing operations were carried out on the slightly radioactive (~ 2 mr/hr) machine in a conventional manner.

A leak was detected in a tube of the main condenser. At reduced power level no difficulty with activity was involved in plugging the defective tube.

The majority of operating problems that have been experienced, except for the control rods, were on the so-called conventional equipment in the plant. There have been no significant problems that were nuclear in origin.

(d) System Problems

Load Following.—With the pressure control by-pass valve system there has been no difficulty in synchronizing the power outputs of the reactor and the load of the turbine gener-

ator. Operation has also indicated that manual operation of the reactor without the bypass valve in service can be accomplished to handle slow load changes on the turbine. The operating staff is confident that an automatic control rod system could be installed successfully at any time.

Activity due to Fuel Element Defects and Fuel Failure.—Three sets of experiments have been run in EBWR on ThO_2 ceramic fuel enriched with 10 percent fully enriched UO_2 .

Test #1—A specimen, fuel diameter 0.300 inch and fuel length $2\frac{1}{2}$ inches, was sheathed in stainless steel through which a 0.010-inch hole was drilled at the center line of the fuel. This specimen was exposed in EBWR at a maximum thermal neutron flux in the fuel of $8 \times 10^{12} \text{ n cm}^{-2} \text{ sec}^{-1}$ at 20 Mw.

Test #2—A larger specimen, fuel diameter 0.312 inch and fuel length 48 inches, was also sheathed in stainless steel and a 0.020-inch defect drilled level with the top of the fuel. This specimen was exposed at a maximum thermal neutron flux of $8 \times 10^{12} \text{ n cm}^{-2} \text{ sec}^{-1}$ at 20 Mw.

Core temperatures in Test #1 did not exceed $1,650^\circ \text{ F.}$ and in Test #2 did not exceed $2,350^\circ \text{ F.}$ at 20 Mw power.

Test #3—Four specimens, each containing fuel of a different density were sheathed in separate stainless steel tubes. Provisions were made so that gas could be swept past the fuel during irradiation and this sweep gas could be analyzed for its pickup of fission products. The four specimens were irradiated in a maximum thermal neutron flux in the fuel of 5 to $6 \times 10^{12} \text{ n cm}^{-2} \text{ sec}^{-1}$.

The results from the first two tests were:

(i) Only fission product gases were evolved to any significant extent, and the rate of release was much less than 1 percent of the production rate of fission gas. Thus, the only activities seen were due either to fission gases or their decay products. Fission products, such as Mo-99, with no gaseous precursors were not observed. Release of radio-iodines was significantly low.

(ii) When the quotient R/Y , where R is the release rate of a nuclide from the defect in atoms per second and Y is its fission yield, was plotted against $1/\sqrt{\lambda}$, where λ is the decay constant, a straight line resulted passing through the origin. Radio-xenons fell on one line and radio-kryptons fell on a separate line of higher slope. These results indicate that some diffusion controlled mechanism is operative within the fuel element.

(iii) No increase in activity was registered on any radiation monitor in the reactor system higher than that normally registered at a given power, with one exception. During reactor start-up, or during power increase, a transient burst of activity was registered on the gamma monitors at the air ejector. Activity levels fell to normal within a few minutes. This burst of activity during increase of reactor power is a familiar phenomenon with unbonded ceramic fuels and is attributed to the "water-logging" phenomenon.

(iv) Samples of reactor water taken from above the element containing the defected specimen showed no increased counting rate either due to radio-iodines or to delayed neutrons, when compared to similar samples taken from above fuel elements containing no defected specimen.

One defected sample was located close to No. 3 control rod. When this rod was moved to increase the flux in the specimen a burst of activity was recorded on the gamma monitor at the air ejectors. Movement of any of the other eight rods did not affect monitor readings. This indicates that defects in the close vicinity of a control rod could be detected by proper manipulation of that rod.

During almost 4 months' steady operation during Test #2, no accumulation of any fission product could be detected in the turbine or steam pipe.

The results of Test #3 were:

(i) Again, the only significant radioactivities released were due to fission gases. Ra-

dio-iodines could be detected this time, however.

(ii) Release of fission gas increased as oxide density decreased.

(iii) A plot of R/Y vs $1/\sqrt{\lambda}$ again showed a straight line passing through the origin. The slope of this line increased with increasing power. The "specific slope", i.e., slope per Mw, also increased with increasing power, indicating a temperature effect.

g. VBWR (Vallecitos Boiling Water Reactor) (26)

(1) DESCRIPTION

The VBWR is a boiling water reactor generating steam which is utilized directly in a turbine generator. The reactor operates at a nominal pressure of 1,000 psig. The steam pressure is reduced to 450 psig to accommodate the second-hand turbine. The plant is not designed for high efficiency but rather as a test plant.

The reactor is licensed to operate at 30 Mw thermal and the turbine generator is rated at 5,200 Kw. Excess steam is bypassed to the condenser. Cooling is by natural circulation.

Operation as a dual cycle reactor with forced circulation is to be tested later. The dual cycle equipment consists of a secondary steam generator and recirculating pump.

The reactor is housed in a vapor tight steel enclosure with the turbine generator in a separate building.

The VBWR uses an enriched uranium, light-water-moderated core having an active length of 37 inches and an equivalent diameter variable between 30 inches and 42 inches. It is contained in a 7-foot diameter pressure vessel. The fuel elements are made up of very thin (0.025 inch) plates of fully enriched UO_2 in a matrix of stainless steel and clad in stainless steel. Either 6 or 9 plates are held by side plates to make up an assembly which is approximately 3 inches x 3 inches square and 37 inches long.

(2) PURPOSE

The VBWR was built as an experimental facility to test the design features of Dresden and to serve as fuel testing facility for Dresden fuel.

(3) CHRONOLOGY

See Figure 18 for the tentative schedule.

(4) SIGNIFICANT DESIGN DATA

See Table 7.

(5) SIMPLIFIED FLOW DIAGRAM

A simplified schematic flow diagram indicating the typical flow diagram of a direct cycle reactor is shown in Figure 20.

(6) OPERATING EXPERIENCE

Because of the high operating pressure (1,000 psig), a very high power density was achieved in this reactor without excessive steam voids. An average power density of ~ 50 Kw/liter of core was obtained at 30 Mw thermal operation with ~ 1.75 percent k_{eff} held in steam voids. Because of the thin fuel elements used in this reactor (0.025 inch) and the large separation of fuel plates (0.49 inch), this corresponds to a power density of only ~ 58 Kw/liter of coolant. The possibility of obtaining stable operation at power densities of greater than 100 Kw/liter of core with natural circulation is anticipated with this thin, fully enriched fuel element.

Radiolytic decomposition of the water decreased with the higher operating pressure, as was found in comparing the decomposition rates of BORAX-IV at 300 psig and EBWR at 600 psig. (see Fig. 17)

At the highest power operation, radiation levels in the turbine building were 6 mr/hr or less except in the vicinity of the condenser, and ~ 15 mr/hr at the high pressure end of the turbine. This again substantiates the previous data obtained from EBWR and BORAX-IV

which indicated that although some activity is present in the direct steam system, little if any shielding is required.

Measurements of the steam void content from the hottest element indicated that at 20 Mw thermal approximately 35 percent by volume of steam was contained in the coolant. Assuming stable operation at 80 Mw thermal, it was calculated that ~72 percent by volume of steam would be in the coolant from the hottest channel.

The fuel element in this reactor has a thermal time constant of ~0.06 second, which is much faster than that obtained with oxide fuel. It is expected that more stable steady state operation will result when an oxide fuel is used in combination with the higher operating pressure.

A pump has been successfully decontaminated and repaired.

h. ALPR (Argonne Low Power Reactor) (27)

(1) DESCRIPTION

The Argonne Low Power Reactor (ALPR) is a natural circulation, light-water moderated and cooled boiling reactor which operates at 300 psig, 420° F. and produces 3,000 thermal Kw of heat. The plant generates a gross electrical output of 300 Kw and provides 400 Kw of space heat.

The reactor is fuel with highly-enriched uranium in the form of aluminum-uranium alloy plates, 0.050 inch thick by 3.50 inches wide by 28.5 inches long, clad with 0.035 inch of aluminum-1 wt percent nickel (X8001 alloy). The reference core has an equivalent diameter of 2.61 feet, is 2.15 feet long, and contains 40 fuel assemblies of 9 plates each.

The reactor is controlled by five cruciform-shaped cadmium control rods clad with X8001 alloy. The control blades are 14.25 inches across tips and have a 32-inch long poison section. Additional spaces are provided within the core for four T-shaped control elements. Supplementary burnable poison may be included as removable strips of boron attached

to the side plates of the fuel assemblies. A boric acid, gravity feed, injection system is provided for emergency shutdown.

Since the reactor produces both electricity and space heating steam, a heat exchanger is mounted in a sealed loft in the containment shell in which 40 psig superheated steam is used to transfer 400 Kw of heat to the water of the space heating system. The balance of the reactor output is sent through a saturated steam turbine located on the reactor floor. The turbine exhausts to an air-cooled condenser also mounted in the sealed loft in the top of the gastight steel containment building.

The reactor control room is located in a conventional building adjacent to the containment shell. Since the reactor is intended for use in remote areas, the reactor shield consists of local gravel which fills the lower one-third of the shell. The reactor vessel is mounted in a steel well located on the vertical centerline of the building within the gravel such that the top of the vessel is level with the top of the gravel. A concrete ring and two blocks, one of concrete and one of laminated steel and masonite, form the top shield. A layer of lead containing imbedded cooling coils surrounds the steel well and serves as thermal shielding.

(2) PURPOSE

The ALPR is the prototype of a reactor intended for military applications in arctic areas. The reactor was designed to be transportable by air and to require a minimum of bulky materials of construction which would also need to be brought into the site by air. Hence, the use of local gravel for shielding and thermal insulation to prevent thawing of the tundra on which the plant is to be erected. A secondary purpose is to train military personnel in the operation and maintenance of a power plant of this type.

(3) CHRONOLOGY

See Figure 18.

(4) SIGNIFICANT DESIGN DATA

See Table 7.

(5) OPERATING EXPERIENCE

The reactor has been in operation for approximately 6 months and has operated satisfactorily with one notable exception. The failure of an oil seal in the deep well pump supplying the reactor make-up water resulted in the introduction of a significant quantity of oil into the reactor primary system. As a consequence, the demineralizers were rendered use-

less and extensive contamination of the fuel plates, core structure, reactor vessel and piping occurred. The oil-contaminated water has been removed and the system cleaned to the greatest extent possible. The detrimental effects of this incident upon over-all plant performance, if any, will not be known for some time.

In general, because of the special design aspects of this reactor and its use of highly-enriched fuel, very little data will be forthcoming of a nature suitable for extrapolation to more economic power reactors. However, operating history of the components will be of value.

D. WORK UNDERWAY

1. General Research and Development

The general fields in which development of boiling water reactors is being advanced are: design of large boiling water reactors of high power densities; design and development of boiling water reactors with integral nuclear superheat; development of fuels and fuel elements.

Further experiments and development work are being pursued at ANL on basic problems such as: control of water decomposition rates; "crud" control; physics and engineering analyses of heat transfer studies of advanced boiling water concepts; engineering analysis of two-phase flow; fuel element failure effects and detection systems; steam separation; fuel element development; and activity buildup. A summary of the data to date has been already discussed. Further data will become available through future progress reports from ANL. ANL is presently enlarging the EBWR core using 5-foot long U-Zr alloy fuel elements interspersed with the original EBWR-I fuel elements. This reactor core is expected to reach a thermal power of 100 Mw and a power density of ~ 58 Kw/liter of coolant. Stability limits of this new core will be investigated. Present plans call for irradiation behavior data to be obtained on the U-Zr fuel elements to burnups of 10,000 Mwd/T. Flow and water chemistry data are also to be obtained.

Other AEC development programs associated with boiling water reactors are reviewed below.

a. Boiling Water Reactor Design Study (TID-8500) (28)

Ebasco Services and the General Electric Company have made a design study to deter-

mine the optimum concept and plant size capable of producing the lowest power generation cost based on technology consistent with the start of plant construction in July 1960.

Except for a higher power density (36 Kw/liter) in the core and a larger assumed allowable burnup of UO_2 fuel, no major extrapolations of the privately financed General Electric Company's Dresden and Italian SENN reactor designs were attempted in the concept. Power costs are estimated on the basis of equilibrium fuel and an average burnup of 15,000 Mwd/T. Most of the development required for this reactor has to do with particular mechanical component design problems. Some effort would also be required on fuel element fabrication techniques.

Some of the general characteristics of this reactor design are:

Power, thermal -----	980 Mw.
Power, electrical	
gross -----	320 Mw.
Power, electrical net..	306 Mw.
Primary (direct)	
steam flow, pres-	
sure and tempera-	
ture -----	2,830,000 lb/hr at 950 psig; 540° F.
Secondary (indirect)	
steam flow, pres-	
sure and tempera-	
ture -----	870,000 lb/hr at 450 psig; 460° F.
Fuel element -----	Rod type, UO_2 enriched to 1.91%.
Fuel assembly -----	120 assemblies, such contain- ing 146 fuel rods.
Core, length x height..	130 inches x 133 inches.
Control rods -----	97 boron-steel.
Average core power	
density -----	36 Kw/liter of core.
Average heat flux ---	108,400 Btu/hr-ft. ²
Pressure vessel -----	12.3 feet I.D.; 46 feet long; 5.4 inches thick.

b. Puerto Rico Water Resources Authority (PRWRA) (29)

The AEC contracted with the PRWRA and General Nuclear Engineering Corporation for the first phase of a study program to evaluate the application of nuclear superheat in a plant having a gross output of about 15 eMw. The purpose of this study was to determine the pertinent problems involved in utilizing nuclear superheat and to generate a preliminary design for a small prototype plant. The concept studies was a UO_2 fueled core where boiling is done in the center of the core and the saturated steam is superheated in specially designed elements around the periphery of the boiling section.

The gross electric output and throttle steam conditions have been set at 17,000 Kw, 850 psig, and 900° F. to conform to the requirements of an existing standard turbine. Saturated steam will normally be produced in the boiling zone of the reactor at 900 psig and 534° F. The temperature of the steam will be raised to 900–950° F., varying with operating conditions, as it passes through the superheater zone. A controlled attemperator in the external steam line will reduce the temperature to 900° F. before the steam reaches the turbine. About 17,000 Kw gross electricity can be generated from steam delivered to the turbine at 850 psig and 900° F., when the reactor is producing 50 Mw of heat. Of this total, 12.5 Mw are generated in the peripheral superheater zone and 37.5 Mw in the central boiling zone.

With regard to overall feasibility and economy, the net result of the work which has been accomplished on the Boiling Nuclear Superheat (BONUS) project thus far has been an increase in ground for optimism. Existing uncertainties are mostly in the nature of questions regarding which, of several available alternates, will prove most desirable rather than major doubts about finding any suitable solution. Other questions pertain to long-term effects such as fuel integrity, corrosion, radioactive contamination, etc. Questions of the

former type will be answered as the BONUS study and experimental development work progress.

c. Superheater Research and Development Program

(1) PHYSICS

The program is concerned with the development of nuclear design techniques of precision sufficient to provide a firm basis for specification of a reactor possessing integral nuclear superheat. This work includes not only the development of techniques but also the evaluation of their validity by a systematic comparison with results of critical experiments designed to simulate as much as possible the actual reactors being considered.

The unique characteristics of a reactor involving nuclear superheat are such that critical experiments already carried out do not provide adequate information for reactor design. The problems to be investigated include:

1. The safety of draining or filling the superheat fuel elements with water.
2. The division of power between boiling and superheater fuel regions.
3. The effect of voids in boiling and superheater
4. Determination of control complement required for operation and shutdown of the reactor.
5. Effect of fuel burnout on the reactivity and power distribution.
6. Determination of lattice parameters for promising superheater designs.
7. Correlation of experimental results with theoretical analysis.

(2) FUEL AND MATERIALS

The advent of integral nuclear superheat for boiling reactors has produced a new set of fuel and materials requirements which are currently being investigated under AEC contracts. The areas of chief concern are: (1) corrosion resistance of thin cladding materials in high temperature steam; (2) carryover of radioactive corrosion products to the turbine and other parts of the plant circuit as a result of the high steam velocities and temperatures required in

superheater elements; (3) effects of strain cycling on clad integrity; (4) thermal insulation of the superheater elements from the surrounding water; (5) methods and requirements for supporting fuel for high temperature operation; (6) methods of accommodating differential thermal expansion in fuel assemblies; (7) effects of moisture carryover on clad integrity; (8) methods of assuring adequate removal of heat from superheater fuel in case of failure of internal joints or mechanical couplings.

Many of these problems are closely related to the problems associated with a typical gas cooled reactor and much of the research currently being undertaken for gas cooled types will be directly applicable to superheater design. However, the gas coolant is considerably less corrosive than the steam, oxygen, and hydrogen mixture found in the integral superheater. Also, since in most gas cooled designs the gas is not used directly in the turbine, the integral superheater must be more carefully designed to avoid problems due to carryover in the steam.

In the designs employing UO_2 pellets for fuel, the temperatures encountered in superheater applications impose unusual demands principally on the cladding materials rather than on the pellets themselves since such pellets customarily operate at equally high or higher temperatures in other applications. For designs which use dispersion type elements made of enriched UO_2 dispersed in stainless steel the conditions encountered in superheater operation are unusual for both the meat material and the cladding, and further work will be needed to establish burnup and exposure limits for this type of fuel. Unless rather high burnups (50-90% of the U-235 atoms) are achieved, this type fuel may not be economically competitive.

(3) HEAT TRANSFER

With regard to heat transfer problems integral superheaters have practically identical

problems to those being investigated for gas cooled reactors. Reasonably accurate correlations are available for predicting film coefficients for simple fuel geometries. The principal areas being investigated are local variations in heat transfer coefficients due to effects of spacers and local variations in velocity profile as a result of complex channel geometry. These effects tend to cause hot spots which can result in serious variations in clad temperature and warping of fuel elements due to the large temperature differential controlled by the gas film coefficient. Unfortunately, problems of this type so far have required empirical tests of actual prototype elements and the development work done is generally applicable to only one specific design.

(4) FLUID FLOW

Increasing pressures, higher power densities, and incorporation of nuclear superheat make water separation problems more difficult. Although the carry-over and carry-under problems currently being studied are not new, it is essential that requirements for steam-water separation be clearly established, particularly in the case of integral superheat designs where separation is done completely inside the pressure vessel.

(5) COOLANT CHEMISTRY

Corrosion problems peculiar to the use of high temperature oxygenated steam are just beginning to be investigated. The specific areas of interest are indicated in Section D-1c(2).

(6) REACTOR SAFETY

Most current designs using nuclear superheat require flooding of the superheater with water as part of the normal shutdown procedure. Designs must be arranged so that minimum changes in reactivity occur with flooding. Also, superheater fuel must be cooled prior to flooding to avoid fuel damage due to thermal shock.

Emergency shutdown cooling of the superheater requires special attention, since cooling must be accomplished either by maintaining a continuous flow of steam through the superheater at all times or by controlled leakage of enough heat from the superheater to prevent overheating the fuel following a scram.

Safety problems associated with the use of moving fuel and controlled circulation are of particular concern in superheat designs since both of these control methods appear to have characteristics which make them attractive for integral superheat reactors.

(7) COMPONENTS AND AUXILIARY SYSTEMS

In integral superheaters, requiring more than one pass of steam through the core, seals are generally required between the superheater elements and the permanent core structure or between elements making up the various passes. These seals must be essentially water-tight, resistant to corrosion by both high temperature water and steam, and must be adaptable to manipulation under water with remote handling tools. The pressure differential across these seals will generally be less than 100 psi. The sealing requirements are unique for each reactor. Development work is being undertaken in an effort to devise a satisfactory seal for service in the BONUS reactor.

In view of the crud problems encountered with bottom-mounted drives, flexible tape drives which can be mounted radially alongside the pressure vessel are being investigated as a possible substitute in the BONUS reactor. The tape drives can be left connected to the control rods during fuel handling and should not tend to trap corrosion products.

Attempts are still being made to devise pressure vessel seals which will permit rapid opening or closing of pressure vessels. For relatively low pressures self-sealing rings of temperature-resistant organic materials have been tried, but without success. The best techniques available to date are those using spiral-wound

metallic gaskets in conjunction with hydraulic bolt tensioning devices.

d. Fuel Element Research and Development

The Fuel Cycle Development Program of the AEC has been initiated to furnish increased general knowledge, improved fabrication methods, and reduced fuel costs on all phases of fuel cycles. The work will include effort in the general areas of fuel element technology listed below. Studies of greatest immediate interest to boiling reactors are indicated with asterisks.

- (1) Extrusion of uranium oxide fuel elements*.
- (2) Development of metallic uranium fuel element of improved irradiation stability*.
- (3) Development of uranium carbide and uranium carbide-type fuel elements.
- (4) Utilization of gas-pressure bonding to fabricate ceramic, cermet and dispersion fuel elements*.
- (5) Synthesis and fabrication of refractory uranium compounds*.
- (6) Development and testing of uranium oxide fuel elements*.
- (7) Development and testing of homogeneous ceramic fuels*.
- (8) Development of an apparatus for ultrasonic roll bonding of fuel elements.
- (9) Development of a uranium carbide fuel element clad with niobium alloy.
- (10) Program to increase specific power of oxide fuel for boiling water reactors*.
- (11) Development of graphite matrix nuclear fuel elements.
- (12) Development of uranium-plutonium oxide fuel cycle*.
- (13) Fabrication of clad massive uranium oxide fuel elements by hot extrusion*.

The following are some of the proposed specific tasks to be undertaken in the general areas of UO_2 development.

- (1) Determination of the effects of swaging temperature and UO_2 properties on the density of swaged UO_2 .
- (2) Determination of the effects of temperature, UO_2 properties, die design, and extrusion methods on extruded UO_2 density.
- (3) Improvements in the methods used for fabricating UO_2 compacts.
- (4) Investigation of methods for obtaining zonal enrichment of UO_2 .

- (5) Improvement of Zircaloy-2 and stainless steel jacketing techniques for UO_2 fuel.
- (6) Determine effects of cold work on stainless steel fuel jacket properties.
- (7) Investigation of welding and brazing methods for joining either stainless steel or Zircaloy-2 components.
- (8) Investigations of the effects of metal bonding agents on heat transfer and fission gas retention of UO_2 fuels.
- (9) Improvement in fuel element end closure and fuel assembly fabrication techniques.

2. BORAX-V Experimental Facility (1)

(1) DESCRIPTION

The BORAX-V reactor to be built at National Reactor Testing Station, Idaho is being designed as an advanced boiling plus superheating reactor experimental facility with great flexibility. Although no selection has been made as to the type of fuel or fuel element, it is planned to make the fuel element spacing flexible so that moderator-to-fuel spacing can be varied somewhat more rapidly than if a new set of fuel element assemblies were required for each test.

It is also planned to begin the tests with a section of the core operating as a superheater. Design is such that the superheat zone will first be located at the perimeter and later at the center of the reactor core. Distributed elements are also possible but each setup requires an entirely new core. Therefore, the number of changes may be held to the two zones indicated.

The superheat elements will be placed in individual pipes or shrouds which allow conducting saturated steam from the vessel head downward over the elements for superheating. The superheat elements will probably be water-moderated. The superheater fuel zone will probably consist of a 1- to 1½-inch square section containing several fuel plates of UO_2 -S.S. matrix fuel clad in stainless.

The steam conditions will probably be 600 psi at 850° F. and the peak power will be between 20 and 40 tMw; probably 20 tMw with superheat.

(2) PURPOSE

1. The determining of operational feasibility will probably be the primary purpose of this experimental reactor. Studies will be made of safe operational procedures of an integral nuclear superheater. One of the basic problems to be studied is operational procedures for the safe voiding and flooding of superheater elements.

2. Some fuel element tests and performance data will be obtained. This reactor probably will not operate long enough to produce utility-type data.

3. Performance data is expected to be obtained that will establish the possible efficiency, economics, and behavior of more suitable proportioned cores.

(3) CHRONOLOGY

The estimated construction and operation dates are shown on Figure 18.

(4) SIMPLIFIED FLOW DIAGRAM

Not available.

3. Power Demonstration Reactors, AEC Projects (2)

a. Rural Cooperative Power Association (RCPA-Elk River)

(1) DESCRIPTION

The Elk River plant is a natural circulation, light water, indirect cycle boiling water reactor with a separate coal-fired superheater. The reactor fuel is 93.5 percent enriched uranium (as UO_2) mixed with ThO_2 , resulting in an initial enrichment of 4.5 percent. Control is by bottom-mounted, boron-stainless steel rods. Net electric output is 22 Mw; reactor thermal power is 58.2 Mw, and superheater thermal power is 14.8 Mw.

(2) PURPOSE

The project is part of the AEC's second-round power demonstration program and will

be used for power production in the Elk River, RCPA system. This will be the first boiling water reactor operating on a wholly indirect cycle and the first to add a fossil fuel superheater to the nuclear plant cycle.

(3) CHRONOLOGY

Design started June 1958 and is expected to be completed by September 1959. Construction started August 1958 and is expected to be completed by August 1960. Scheduled start of operation is February 1961 (Figure 18).

(4) SIGNIFICANT DESIGN DATA

Significant design data are listed on Table 7.

(5) SIMPLIFIED FLOW DIAGRAM

A simplified flow diagram representative of the indirect, natural circulation cycle is shown on Figure 22.

(6) OPERATING EXPERIENCE

There is no operating experience to date.

b. Northern States Power Company (NSP) (30-34)

(1) DESCRIPTION

The Pathfinder plant features a direct cycle, controlled circulation, boiling reactor with a core having an integral nuclear superheater

FIGURE 22—Simplified flow diagram of indirect cycle with fossil fuel superheating.

and an aluminum alloy clad boiler section. Fuel for the boiler will consist of conventional assemblies of rod type elements containing UO_2 pellets. However, to help flatten the axial power distribution, the rod diameter will be varied along the length of these elements. For the superheater section concentric tubular elements, consisting of highly enriched UO_2 particles dispersed in a stainless steel matrix and clad with stainless steel, will be used. Steam will be produced initially in an annular boiling region surrounding the central superheater. After flowing up through moisture separators located above the boiler, it will make a single pass downward through the superheater, emerging at 825°F . into a collection plenum beneath the core which is in turn connected directly to the main steam line.

Thermal isolation of the high temperature superheater from the rest of the core is accomplished by mounting each superheater element in a double-walled, stainless steel tube. The 53-mil annular gap between the double tube walls is filled with stagnant steam which acts as a thermal barrier to prevent excessive heat loss to the water surrounding the superheater tubes. The moderator-to-fuel ratio for the superheater is chosen to minimize reactivity changes associated with flooding the superheater. The use of aluminum cladding for the boiling elements has limited the operating pressure of this reactor to 600 psig and the saturation temperature in the boiling zone to 489°F . Zirconium cladding is being considered as an alternate to aluminum.

(2) PURPOSE

This plant was proposed under the Third-Round Power Demonstration Program. Its main purpose was to demonstrate economic power production using aluminum clad fuel elements, control of power by varying recirculation ratio and use of an external fossil-fueled superheater. Recently, an integral nuclear superheat section has been substituted for the conventional superheater.

(3) CHRONOLOGY

Figure 18 shows the tentative schedule for this reactor.

(4) SIGNIFICANT DESIGN DATA

See Table 7.

(5) SIMPLIFIED FLOW DIAGRAM

A simplified flow diagram of the direct cycle with integral superheat is shown in Figure 23.

4. Power Demonstration Reactors, Private Projects

a. Dresden Nuclear Power Station (2, 117, 118)

(1) DESCRIPTION

The Dresden Nuclear Power Station is an all nuclear boiling water reactor plant of 626 tMw output which will produce 180 Mw net electrical power. It is financed entirely by private enterprise and is being built by Commonwealth Edison Company of Chicago together with the companies in the Nuclear Power Group Inc.

The reactor core is light water cooled and moderated and is made up of 488 fuel assemblies slightly over 9 feet long. Each assembly consists of 36 half-inch diameter slightly enriched UO_2 , Zircaloy-2 clad rods enclosed by a Zircaloy-2 channel. There are 80 control rods which enter the bottom of the vessel and are driven up into the core.

The reactor plant is enclosed in a 190-foot steel spherical structure with the turbine generator housed in a separate building. The reactor plant equipment consists of the reactor, reactor recirculating system, piping, associated auxiliaries, and necessary shielding. The recirculating system consists of the reactor, steam separating drum, four recirculating pumps, four secondary steam generators, and the inter-connected piping.

Direct and indirect steam are generated by the reactor. The direct steam from the reactor enters the turbine at ~ 980 psig and 546°F . and the indirect steam is generated in a separate heat exchanger and enters the turbine at

FIGURE 23—Simplified flow diagram of direct cycle with integral nuclear superheating.

~500 psig and 469° F. Normal load control is by the governor controlled secondary steam admission valves.

(2) PURPOSE

This plant will produce power for the Commonwealth Edison System.

(3) CHRONOLOGY

The estimated time schedule for the reactor is shown in Figure 18.

(4) SIGNIFICANT DESIGN DATA

See Table 7.

(5) SIMPLIFIED FLOW DIAGRAM

Figure 24 is representative of the direct plus indirect cycle used in this plant.

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b. Pacific Gas and Electric (PG&E)

(1) DESCRIPTION

The PG&E is a privately-financed plant for the Pacific Gas and Electric Company. The design and construction will be made by Bechtel Corporation and General Electric as the architect-engineer and nuclear manufacturer, respectively. The plant will consist of a light water moderated, natural circulation, UO_2 fueled reactor and associated equipment. The plant is rated at 50 eMw and the reactor operates at 1,000 psia pressure. The reactor is currently scheduled for operation in mid-1962.

(2) PURPOSE

This reactor is primarily designed to generate power using a natural circulation type reactor.

FIGURE 24—Simplified flow diagram direct+indirect cycle.

(3) CHRONOLOGY

The estimated construction and operation dates are shown on Figure 18.

(4) SIGNIFICANT DESIGN DATA

No data are available to date besides the power.

(5) SIMPLIFIED FLOW DIAGRAM

Figure 20 is representative of the typical direct cycle plant.

5. Power Demonstration Reactors, Foreign Projects

a. *Societa' Elettronucleare Nazionale (SENN)* (119)

(1) DESCRIPTION

The SENN plant is very similar to the Dresden plant except that the flow of secondary

steam is much smaller relative to the flow of primary steam. In SENN, the ratio of primary steam flow to secondary steam flow is 3.22 : 1; while in Dresden, it is 1.11 : 1. This, coupled with a better condenser vacuum, makes the SENN plant gross efficiency (31.5%) slightly better than Dresden (30.6%). As far as design and operating characteristics are concerned, the SENN plant is a typical direct plus indirect cycle unit, producing saturated steam at 950 psig and 450 psig, which is then sent to a dual admission turbine. The secondary steam admission is governor-controlled, while the primary steam admission valves are pressure-controlled. Reactor power can be automatically matched to the load for changes up to 35 percent of rated power by the governor-controlled secondary steam system which carries the sub-cooling to the reactor.

(2) PURPOSE

This reactor is being privately financed by an Italian power syndicate (Societa' Elettro-nucleare Nazionale) for construction in Italy to produce power on a commercial basis.

(3) CHRONOLOGY

See Figure 18.

(4) SIGNIFICANT DESIGN DATA

See Table 7.

(5) SIMPLIFIED FLOW DIAGRAM

Figure 24 is representative of the direct plus indirect boiling water reactor cycle.

b. Rheinisch-Westfalisches Elektrizitätswerk (RWE)

(1) DESCRIPTION

The RWE is an indirect or closed cycle boiling water power plant being designed by the General Electric Company for construction in Germany. This plant is basically similar to the Elk River plant but will not use a superheater. The reactor will operate at a pressure of 1,000 psia, a thermal power of 58 Mw and an electrical capacity of 15 Mw. Dry saturated steam from the secondary side of a heat exchanger will be delivered to the turbine.

It is not expected that any significant information will be obtained from this reactor which will not be obtained from the Elk River reactor.

(2) PURPOSE

This plant will be used for the generation of power.

(3) CHRONOLOGY

The estimated schedule for this reactor is shown on Figure 18.

(4) SIGNIFICANT DESIGN DATA

No data, other than material above, is available.

(5) SIMPLIFIED FLOW DIAGRAM

Figure 22 is a simplified flow diagram of the RCPA reactor, which is basically similar to RWE except for the inclusion of a fossil-fired superheater.

c. Russian Superheater (120)

(1) DESCRIPTION

This reactor, to be constructed in the Ural Mountains of Russia, is a graphite-moderated, pressure-tube-type, boiling water plus integral superheat reactor. The reactor plant, having a 100 Mw electrical output, is to be the first unit of a 400 eMw power station. Fuel assemblies are installed vertically and each fuel assembly consists essentially of a 20-foot long cluster of seven tubes in a cylinder of graphite. Six annular U-Mg (1.3% enriched) fuel elements canned inside and outside with stainless steel surround a seventh centrally located stainless steel coolant tube. Water is fed down the central tube and distributed to the internal hole of the six fuel elements. The water is allowed to boil as it flows through these elements and the exit mixture is taken to a steam separator where the steam and water are used to generate saturated secondary steam in a heat exchanger. This secondary steam is returned to the superheat fuel elements, which are similar to the boiling elements, where the steam is superheated and sent directly to the turbine. The pressure tubes in which the superheating is done are located at an intermediate radius such that reactivity is affected very little by interchanging water with steam inside the tubes.

At startup, the superheater tubes are filled with water. With the reactor operating at 30 percent of full power, secondary steam from the heat exchanger is used to force the water from the superheater tubes.

The turbine inlet conditions are 1,320 psig and 930° F. These conditions are comparable to those used in modern steam plants, and a good deal better than conditions achieved in U.S. reactor plants.

Many of the features of this reactor concept, including prototype fuel elements under boiling and superheating conditions, and the transition from boiling to superheating conditions, have been checked out in the Russian APS-1 reactor.

(2) PURPOSE

This reactor is part of the Soviet power plant expansion program, and represents only one of the various types of large power plants being constructed.

(3) CHRONOLOGY

No dates of construction or startup were indicated by the Russians although it was stated that this plant is in construction. An estimated chronology for this plant is shown on Figure 18.

(4) SIGNIFICANT DESIGN DATA

See Table 7.

(5) SIMPLIFIED FLOW DIAGRAM

A simplified flow diagram for the boiling-superheating system is shown on Figure 25.

d. Russian BWR (121)

(1) DESCRIPTION

A pressurized boiling water reactor with a 50 Mw electrical capacity is under construction at the Volga Test Station in Russia. The reactor pressure vessel is designed to operate up to 2,000 psia. Fuel elements are to be similar to their nonboiling pressurized water reactor; fuel assemblies are full length, not segmented, zirconium alloy (Zr-2.5 wt% Nb) clad

FIGURE 25—Simplified flow diagram Russian boiling+superheating system.

UO₂. Control rods come through the top of the reactor and steam is fed directly to the turbine. A forced circulation coolant system is being provided. Very little other data were available on this reactor.

(2) PURPOSE

This plant is to serve as an experimental unit for checking the operation of a boiling reactor under a variety of conditions. Operating pressures up to 2,000 psi will be investigated as well as various cycles.

(3) CHRONOLOGY

Although no dates for operation of this reactor were given at the 1958 Geneva Conference, it was said to be under construction. An estimated chronology for this plant is indicated in Figure 18.

(4) SIGNIFICANT DESIGN DATA

No design parameters except the power rating, pressure capability, fuel element type, and coolant system discussed above, were available. A model of the plant was exhibited.

E. CONSTRUCTION AND OPERATION SCHEDULE

The construction and operation schedule of various boiling water reactors which have been or are being constructed is shown on Figure 18. In general, reactors below approximately 50 eMw of the natural circulation type will average approximately 3 years for construction

through criticality and full power operation. A progressively larger period is required as the plant size is increased. At a generating capacity of 320 eMw, approximately 4.5 years would be needed.

F. INHERENT PROBLEMS AND LIMITATIONS

1. Physics

At the present time, uncertainties in physics information lead to over conservatism in both the nuclear and mechanical design of boiling reactors. Furthermore, these uncertainties are not totally removed with the successful operation of the reactor plant because detailed physics measurements are usually not made. Since the uncertainties are interrelated, it is difficult to resolve the individual effects based on total reactor performance.

There is an urgent need for experimental information involving practical engineering designs. While it is impossible to have a general set of experimental information that would provide a direct experimental check on all conceivable boiling reactor designs, there are certain areas where experimentation can be used as a basis for the analytical techniques, which in turn are to be directed at specific designs.

Specifically the following problem areas need to be investigated:

a. Reactor Stability

Stability considerations dictate the value of void coefficient of reactivity which in turn sets the fuel/moderator ratio, or its equivalent the average density of fuel in the core. Therefore, the average power density of the core is not only determined by thermal and heat transfer characteristics but by dynamic considerations as well.

The main incentive for developing a theoretical understanding of the factors affecting the stability of boiling reactors is to allow the increase of design power densities to the point where they are limited by considerations other than reactor stability.

All the experimental information has been gathered from small natural circulation reactors having fuel elements and water-to-fuel ratios considerably different from those which appear to be desirable for large commercial plants. Extensive experimental information of much more detailed type is required on high power density forced circulation systems before boiling water reactor cores can be designed to their highest capability with complete confidence.

The experimental information desired pertains to the stability characteristics of large reactors as a function of pressure, water-to-fuel ratio, geometry, and coolant flow velocity. The pressures considered should be between 1,000 psi and 2,000 psi. Water-to-fuel ratios should be in the range of 1 to 3. The coolant flow velocity should be varied from 6 to 30 feet per second.

b. Power Distributions

In order to increase reactor performance, it is necessary to minimize peak-to-average power. While the calculations of gross power distributions are reasonably good, the local effects introduced by water channels, fuel end caps, control rods, non-uniform fuel, etc., are not well known. Since these effects can add as much as 50 percent to the peak-to-average power, a much more comprehensive understanding of these effects is needed to determine optimal designs.

c. Lattice and Neutron Spectrum Measurements

Present lattice measurements are for uniform fuel rod distributions and initial isotopic composition. It is desired to have lattice measure-

ments which include the effect of fuel burnup, plutonium formation, changes in void distribution, and the effect of control rods.

Similarly, it is desirable to have direct neutron spectrum measurements of close packed lattice with the purpose of determining the magnitude of the positive contribution of the temperature coefficient when Pu-239 is present in large quantities. Further, the effective cross section of Pu-240 as a function of lattice spacing should be measured, since its presence represents a possible limit to improving neutron economy by tightening lattices.

Measurement should be made of the resonance escape probability of highly subdivided fuel elements, such as elements consisting of several annular regions, with and without void, to permit more reliable calculational formulas for the resonance integral and the void coefficient.

d. Xenon Instabilities

Large reactors operating with high thermal fluxes are subject to xenon oscillations. These oscillations produce a flux tilt in that the reactor power will shift from one section of the reactor to another. This type of instability may become an important consideration for light water reactors when the linear dimensions of the reactor exceed about 7 feet. The production of voids will tend to damp this effect, so it may not be as serious in boiling reactors as in other reactor types. Only the operating data from large power reactors can furnish the experimental information to compare with analysis.

e. Fuel Cycles

A fuel cycle that produces uniform burnup and power flattening with a minimum of shutdown time is of vital concern to achieve low power costs. An experiment in fuel management to determine the optimum "shuffling" procedure as fuel irradiation progresses would be extremely valuable. This should be done in an actual operating reactor.

f. Design of Boiling Water Reactor Utilizing Digital Computer Codes

The core design of boiling water reactors considering only reactor statics is rather complex since physics, heat transfer, and hydraulic information are all factored into the calculations. Digital computer codes which assemble all this information into one package can increase calculational accuracy and can also greatly assist in reducing the development costs associated with any new design. Although one such code now exists (BORE), it is limited in its scope and the hourly charge just for using the code is quite large. A more generalized code, if available at a nominal charge, in which control and power flattening methods can be introduced as variables would contribute to the advancement of core design.

Similar codes for reactor dynamics would tend to eliminate a major source of the uncertainties in the transient calculations.

2. Fuel and Materials

a. Metallic Fuels

(1) LOW URANIUM CONTROL ALLOYS AND DISPERSIONS

The highly enriched aluminum-uranium alloys are limited to low temperature applications by their inherent lack of strength and corrosion resistance in high temperature water. This fact, coupled with the high fuel costs attendant to fully enriched uranium systems, effectively eliminates them from any major role in economic commercial boiling reactor power plants. Similarly, the use of highly enriched uranium makes stainless steel-uranium alloys and dispersions uneconomical for conventional boiling reactor programs. However, because of the excellent resistance to high temperature steam corrosion and the strength exhibited at elevated temperature, stainless steel is admirably suited to the superheating zones of integral superheat reactors. This factor may outweigh the disadvantages of higher fuel costs.

(2) HIGH CONTENT URANIUM ALLOYS

The major problems of high content uranium alloys are metallurgical instability under irradiation and the basic lack of corrosion resistance. Gratifying results have been obtained in the areas of radiation stability through the use of heat treatment and alloying. However, exposure levels comparable to those demonstrated as acceptable for UO_2 have not been obtained. Furthermore, there is little expectation that they will be achieved in the near future. In the areas of corrosion resistance, there has been practically no success at all. The only improvements have been changes in the rate at which complete oxidation occurs from a matter of hours to a matter of days. The outlook for a reasonably quick solution to the corrosion problem is quite bleak.

b. Ceramics

High cost rules out thoria-urania systems until major advances are made in thermal breeding reactor concepts and thoria processing methods. Uranium dioxide, however, appears to have a bright future. Three problems, basic to this fuel, are reduced uranium density, poor heat transfer properties, and high fabrication cost. There is little which can be done about the first problem other than to design more fuel into a given core volume. The second is covered in Section F-3,

Heat Transfer.—An extensive research and development program, both private and AEC sponsored, has been undertaken to solve the third. The highlights of the AEC sponsored program to improve UO_2 fuel fabrication techniques and reduce the cost of fabrication is given in Section D-1-d,

Fuel Element Research and Development.—There is every reason to believe that lowered fuel cycle costs will result from this effort.

c. Cladding and Structural Materials

(1) ALUMINUM

The inherent characteristics of low strength at elevated temperatures and poor corrosion

resistance in high temperature water, which eliminated aluminum fuel types from consideration for economic power reactors, also applies to their use as a structural material. Although attempts have been made to improve these properties, only limited success has been achieved. The future outlook promises no better results.

(2) ZIRCONIUM

Zirconium alloys are the most promising metals for use in the active zone of a boiling reactor core. These materials meet all of the necessary requirements with one exception—both the base cost of the metal and the cost of fabrication are high. However, the savings effected in neutron economy are sufficient to make the use of Zircaloy competitive with stainless steel, price handicap notwithstanding. The AEC fuel development program, mentioned previously, has as one of its goals improvements in the techniques and methods of zirconium fabrication. These efforts should affect significant cost reductions.

(3) STAINLESS STEEL

The role of stainless steel in structures external to the active portion of the core is assured. Moreover, the excellent corrosion resistance of stainless to high temperature steam, a property lacking in Zircaloy, makes it the only material which can safely be used, at present, in the superheating portion of integral superheat cores. Further, if improvements in strength can be achieved permitting reductions in required thicknesses, other core applications may once again be possible.

d. Control Element Materials

(1) HAFNIUM

The basic problems associated with hafnium are its extremely short supply and high cost. New techniques for the production of ductile hafnium from sponge are under development and are expected to reduce the cost by at least 50 percent over the present crystal bar process.

However, there seems little which can be done to improve the supply situation. The long range outlook is for continued shortage.

(2) BORON STAINLESS STEEL

The lack of knowledge regarding the effects of radiation on the properties and structure of boron stainless steel is the major problem area. Recent data (124) indicate that 1.0 wt percent boron stainless steel specimens start to lose strength after 10 to 20 percent burnup of B-10 atoms and have lost a good deal of their strength after a burnup of 90 percent of the B-10 atoms. The operation of the EBWR, which contains four 2 wt percent boron stainless control rods, and long exposure in pile tests now in progress, should do much to answer these questions. A secondary problem area is cost. At the present time, 1 wt percent material in plate form costs approximately \$3.50 per pound and 2 wt percent material about \$6 per pound. However, increased production should reduce these prices to a level more nearly comparable to the columbium stabilized stainless alloys.

(3) CADMIUM AND RARE EARTHS

The problems associated with the jacketing of cadmium and the rare earths, which is necessary if these materials are to be used in high temperature water or steam, are the major deterrents to their use in boiling power reactors of interest. However, if the results of the boron stainless steel programs, now underway are not successful, the use of cadmium and the rare earths will bear further investigation.

3. Heat Transfer

Aside from stability considerations, the limitations on power density, which apply to both boiling and nonboiling reactors, are the limitations on heat transfer to water (which if exceeded cause burnout of fuel elements) and the limitations on heat conduction in the fuel element (which if exceeded cause excessive tem-

peratures at the center of the element). Although these limitations are hard to discuss in terms of an arbitrary fuel element, a consideration of their relations to the fuel elements currently in use can clarify the basic relations.

At given coolant conditions (flow velocity, subcooling, etc.), the occurrence of fuel element burnout is determined by the heat flux at the fuel jacket-water interface. At a given surface heat flux, the central temperature of a fuel element of given shape and composition depends on the size of the element; and for a given maximum permissible centerline temperature, the limiting surface heat flux can always be increased by decreasing the element size. Curve A of Figure 26 shows, for a typical metal-jacketed UO_2 fuel element, the calculated surface heat fluxes which will result in a central temperature of $4,800^\circ\text{F}$., as a function of fuel element diameter. It is generally accepted that heat fluxes as high as $400,000\text{ Btu/hr-ft}^2$ can be used in boiling reactors without danger of burnout (the calculated maximum heat flux in EBWR when operating at 60 tMw with metallic plate-type fuel elements was $464,000\text{ Btu/hr-ft}^2$). From Figure 26, it is evident that such a heat flux will require a fuel element diameter for UO_2 elements at least as small as 0.44 inch. If a higher heat flux, such as $600,000\text{ Btu/hr-ft}^2$, were regarded as permissible, the fuel temperature limit would require that the fuel element diameter be reduced to about 0.28 inch. It should be noted that the reduction of fuel element size increases the power density and specific power, not only because higher heat fluxes may be used but also because more heat transfer surface can be packed into a given core volume. Curves B and C of Figure 26 show the attainable maximum power densities and specific power as functions of fuel element diameter, on the assumption that a constant ratio of H_2O to fuel (UO_2) can be maintained as the fuel element size is decreased, and tolerance specifications must become more rigorous.

The differences in thermal performance between boiling and nonboiling reactors, as they

FIGURE 26—Surface heat flux, specific power, and power density as limited by UO_2 centerlines temperature of $4,800^\circ\text{F}$. All values are maximum (not average) values; T (ambient) $= 550^\circ\text{F}$; h (water film) $= 5,000$ Btu/hr ft 2°F ; k (UO_2) $= 1.0$ Btu/ft hr $^\circ\text{F}$; k (Zr) $= 8.2$ Btu/ft hr $^\circ\text{F}$; and k (HE) $= 0.14$ Btu/ft hr $^\circ\text{F}$.

evolve in practice, can be examined by means of Table 8. In this table, the Shippingport reactor and EBWR are included simply for reference. The comparison which may be considered as typical of the present state of the art is between the Yankee, Consolidated Edison, and Russian reactors as representatives of non-boiling designs, and the Dresden and the Belgonucleaire reactors as representatives of boiling designs. The maximum heat fluxes in the

nonboiling reactors are higher by from 30 to 60 percent than those used in the boiling designs. The average heat fluxes, however, in the two cases are not greatly different. It is difficult to say whether there is an inherent difference in the degree of power flattening attainable in the two reactor types; certainly it cannot be concluded that the differences in maximum-to-average heat flux ratios shown in the table have general meaning, because the re-

TABLE 8

THERMAL OUTPUT CHARACTERISTICS OF H₂O REACTORS

	Shipping- port*	Yankee	Consoli- dated Edison	USSR	EBWR	Dresden	Belgonu- cleaire
Thermal output, Mw-----	**120	392	500	760	***60	626	242
Fuel load, tons of oxide-----	16	25. 1	22. 2	44	-----	65. 8	23. 53
Oxide diameter, in. -----	0. 360	0. 290	0. 312	0. 346	Metal fuel	0. 494	0. 384
Jacket thickness-----	0. 026	0. 021	0. 020	0. 028	-----	0. 030	0. 015
Maximum heat flux, Btu/(ft ² - hr).	~354, 000	450, 000	450, 000	443, 000	464, 000	277, 000	365, 000
Average heat flux, Btu/(ft ² -hr)	~48, 300	87, 000	112, 500	83, 000	136, 500	94, 700	79, 300
Maximum heat flux-----	7. 33	4. 18	4. 00	5. 34	3. 40	2. 90	4. 60
Average heat flux-----							
Surface/volume ratio of fuel...	11. 1	13. 8	12. 8	11. 6	~8. 5	8. 1	10. 4
Specific power, Mw/ton of oxide.	7. 5	15. 62	22. 5	17. 3	-----	9. 5	10. 3
Volume H ₂ O-----	1. 45	1. 44	~1. 54	1. 90	Metal fuel	2. 17	3. 08
Average power density, Kw/ liter of core.	23. 3	60	66	43	~44	28. 9	25. 5

*Quantities apply to blanket only.

**225 Mw total, 105 in seed, 120 in blanket.

***Experimental operation.

actor designs have been made at different times and by different groups and have been carried to different degrees of refinement. Whatever the general significance of the maximum-to-average ratio, it is clear that the higher specific powers exhibited by the nonboiling reactors in the table are due largely to the greater subdivision of the fuel (higher surface-to-volume ratio). The higher specific power, when combined with the higher ratio of fuel volume to water volume in the nonboiling reactors, leads to power densities which are roughly twice as high as those exhibited by the boiling reactors. A progression toward higher power densities is evident in boiling reactors, however, and this difference between boiling and nonboiling types is expected to narrow markedly as more operating experience accumulates, particularly on forced circulation boiling reactors. It must, of course, be recognized that the power density which characterizes a given design results from an economic optimization process

and that the economic incentives toward high power density are not as great in boiling as in nonboiling reactors.

It can be seen from the data reported in C-1 that the comparatively high enthalpies at which boiling reactors operate at any given pressure result in lowered burnout heat fluxes. This may be partially compensated by an inherent flux flattening advantage of boiling water cores. The data also indicates that considerable margin exists between the operating heat fluxes of present designs and even the lowest values of burnout flux from the correlations. A further margin should be recognized in that the minimum burnout flux in a boiling water core will always exist at a channel exit which is at a comparatively low operating heat flux. The maximum operating heat flux in a boiling water reactor normally occurs in the lower portion of the core in the subcooled region. The burnout heat flux at this point can be quite high. It is not correct to compare

the maximum operating heat flux of a boiling core with the minimum burnout flux since these values will always be calculated at different axial points.

As fuel element technology improves there will be a greater incentive to increase operating heat fluxes, since the fuel temperature limit will, in all probability, be relaxed. Burnout heat fluxes will then be approached more closely than in present designs. Argonne National Laboratory reports (74) the possibility that at high mass flow rates the problem of burnout may be eliminated entirely since the surface temperature excursion may be small and not result in destruction of the heat transfer surface.

As heat fluxes are pushed much higher in the future the boiling reactor designer will need to know the factors which affect burnout more accurately. In addition to a better understanding of the available data, more information is required on burnout under transient conditions, on pressure dependency, on geometric dependency, on effects of parallel multi-channel systems, on effects of non-uniform power generation, and on flow direction and orientation.

Bettis points out that (57) "while the present empirical burnout heat flux correlations account for the effects of the principal variables, the scatter of data ($\pm 35\%$ on flux) remains approximately three times larger than can be accounted for by experimental error. Consequently, additional theoretical work is needed, with the aims of understanding the fundamental burnout mechanism and determining the correct parameters for correlation."

4. Coolant Stability and Temperature

a. Steam Pressure and Temperature

The rather large differences between the steam pressures attainable at the turbine with boiling and nonboiling reactors is due to the characteristically steep rise of water vapor pressure with temperature in the range of interest. If the reactor is to be so designed that

no boiling occurs at the surface of the hottest fuel element, then the reactor must be pressurized to a level equal to or greater than the vapor pressure of water at the surface temperature attained by the hottest fuel element under the most adverse operating conditions. This temperature will be higher than the secondary steam temperature by some amount, ΔT , which will depend on the film temperature drop at the fuel element and boiler tube surfaces, on whatever hot channel factors are appropriate in the reactor, and on any additional allowances which may be made for overshoots of reactor power. If a value of 150°F. is taken for ΔT , a relation between secondary steam pressure and reactor pressure results which comes close to fitting current nonboiling designs. This relation is plotted in Curve A of Figure 27.

In the indirect-cycle boiling reactor, the only contribution to ΔT arises from the heat-transfer temperature drop in the secondary boiler. One example of such a design is Elk River (RCPA). A temperature drop of approximately 48°F. is taken in the heat exchanger and a constant value of 48°F. has been drawn as Curve B of Figure 27. In the direct-cycle case the steam pressure is very nearly equal to the reactor pressure, being lower only by the pressure drop in the moisture separator and the piping. A constant value of 25 psi has been assumed in Curve C of Figure 27.

The question of how much the higher steam pressure is worth, or alternatively how much advantage results from the lower reactor pressure of boiling reactors is difficult to answer. For large central station applications it may be predicted that boiling reactors will operate in the relative high pressure range, 1,000 psi and above, because of the heat removal advantages which result from high pressure. Thus a typical comparison might be between a boiling reactor operating at 1,000 psi and nonboiling reactor operating at about 1,500 psi and producing steam at a pressure of about 500 psi. A curve of efficiency as a function of pressure for saturated steam is given in Figure 28. As

CURVE A - NON-BOILING REACTORS ($\Delta T = 150^{\circ} F$)
 CURVE B - INDIRECT CYCLE BOILING REACTORS ($\Delta T = 48^{\circ} F$)
 CURVE C - DIRECT CYCLE BOILING REACTORS ($\Delta p = 25 \text{ PSI}$)

FIGURE 27—Possible relationships between turbine inlet pressure and reactor pressure for boiling and non-boiling reactors.

FIGURE 28—Calculated turbine cycle efficiency as a function of pressure for saturated steam. Five stage regenerative cycle: Optimum feedwater rise; 1.5 in. Hg. back pressure; and 100 to 200 Mw capability.

steam pressure is raised the moisture problem becomes more important, and it may be expected that superheating for boiling reactors will be even more attractive in order to realize the fuel potentialities of the reactor type. At present, modern fossil fuel plants use superheated steam at throttle conditions of 1,000° F. and 1,450 psi or higher.

A further advantage of high steam pressure is the gain possible in power density. It is now well established that an effective method of increasing the stability-limited power density of a natural circulation boiling reactor is by increasing the operating pressure. As pressure is increased the volumetric heat content of a steam bubble increases in almost direct proportion, and consequently, a greater quantity of heat can be removed from the reactor for a given water flow velocity in the coolant channels and a given steam volume fraction. This effect is greatly obscured at very low steam pressures and low coolant velocities by the very strong effect of the variation in velocity of slip of the steam through the water. However, at pressures which are of practical interest for electric generating plants, the variation in slip velocity with pressure is much reduced, and the effect of pressure on power output is quite important. Furthermore, experience with operating reactors has demonstrated that operating pressure has other inherent effects on stability which make possible stable operation with larger amounts of reactivity compensated by steam. In the EBWR it was found that when the operating pressure was reduced to one-quarter the normal value, a reactor transfer function very similar to the normal transfer function was obtained if the reactor power also was reduced to one-quarter of its normal value. These effects have been attributed at least in part to the lower pressure coefficient of reactivity which characterizes high pressure operation. In effect the lower pressure coefficient of reactivity is simply a manifestation of the fact that steam bubbles in equilibrium with saturated liquid are less compressible when the operating pressure is high.

b. Coolant Stability

Recirculation of the coolant through the reactor can be achieved by either natural or forced circulation. Laboratory tests have shown that natural circulation stability is a function of system geometry, pressure and exit steam volume fraction. Changing system geometry by placing restrictions in the downcomer increases stability but imposes a power capability penalty on the reactor system due to decreased flow. Increasing system pressure results in instability occurring at a lower value of exit steam volume fraction, but at a higher power density. A threshold value of exit steam volume fraction (estimated to be 75%) exists beyond which hydrodynamic instability has occurred in several experiments.

Forced circulation systems, because of the non-oscillatory characteristics of the pumping head, are expected to be hydrodynamically stable. In multichannel systems, at the high pressure drops associated with some forced circulation systems (where density head differentials are small compared with the friction drop), an effect of flow choking or stoppage can be postulated. The effect can be controlled by the introduction of suitable flow restrictions at the flow channel entrances. The additional pumping power necessitated by the orifices represents an almost trivial economic penalty.

5. Components

Since inherent limitations in plant components have generally resulted in economic penalties rather than design barriers for boiling reactors, they have attracted less attention than some of the more pronounced effects, such as fuel temperature limits, etc.

a. Pumps and Valves

High pressure mechanical shaft seals are presently available for service up to 1,250–1,500 psi. For higher pressures and temperatures, canned motor pumps are available which can withstand suction pressures as high as 3,000 psi or more (122). Although pump seal prob-

lems have not been eliminated and the costs are high, rapid advances have been made in this field in recent years, and pump development has kept pace with the demands of boiling reactor plants. Imposition of zero leakage requirements in early plant designs resulted in severe cost penalties. With increased operating experience, it should be possible to relax these requirements somewhat.

Providing the necessary submergence, or net positive suction head, required to prevent cavitation in circulating pumps handling saturated water, makes it necessary to locate these pumps 20 feet or more below the reactor water level in most cases. This increases the cost of loop piping and limits equipment location within the containment shell.

Although entrainment of steam in downcomers has not been a serious problem at the pressures and power densities encountered so far, as pressures increase it will become more of a problem. If appreciable quantities of steam are carried into the pump intake lines, pump cavitation will occur and mechanical steam separators will have to be included either inside the reactor vessel or as part of the external loop.

b. Pressure Vessels

Both fabrication techniques and manufacturing facilities impose limits on size and pressure ratings of reactor vessels. With present techniques, vessel manufacturers estimate that vessels for 1,500 psi service can be made with inside diameters as large as 15 feet; with slightly smaller diameter, pressures up to 2,500 psi are feasible. There are two basic reasons for these limits. First, as plate thickness increases, it becomes more difficult to assure metallurgical uniformity and freedom from internal faults. Second, forming mills and annealing ovens are limited in the size of the pieces they can handle. Transportation fa-

cilities are also a major factor in limiting vessel size if the plant site is not near a navigable stream.

c. Turbines

Saturated steam turbines are subject to excessive wear and to loss of efficiency if excessive moisture is allowed to accumulate. Present practice is to limit stage moisture to about 15 percent. In order to meet these requirements at pressures of 1,000 to 1,400 psi, it is necessary to use special moisture removal stages. If still higher pressures are used, the moisture problem increases very rapidly and the incentives to use superheated steam increase accordingly. If superheated steam is provided, turbines are now available which can handle steam conditions up to 5,000 psi and 1,200° F (123); well beyond any nuclear requirements for the near future.

d. Control Rod Drives

A large amount of development work is being done on various control rod drives, and much of it is aimed at reducing cost of drives and obtaining a more nearly ideal set of characteristics, such as reduced crud contamination, less interference with fuel handling, fast scram, etc. The rod drives which are available today are reasonably sound mechanically and are subject to essentially the same pressure limitations as the pumps discussed in Section 5a.

e. Instrumentation

Most out-of-pile instrumentation has developed rapidly with no indication of severe problems. However, in-pile instrumentation is severely limited by radiation tolerances, temperatures, and pressures. Practically all organic materials undergo some radiation damage at a total accumulated dosage of $\sim 10^7$ rads. Development is needed on transducer elements using non-organic insulation and on all metallic component elements.

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