Argonne National Laboratory

**HAZARDS EVALUATION REPORT
ASSOCIATED WITH THE OPERATION
OF EBWR AT 100 MW**

by

E. A. Wimunc and J. M. Harrer

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ARGONNE NATIONAL LABORATORY
P. O. Box 299
Lemont, Illinois

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Contributors

N. Balai	W. C. Lipinski
L. W. Fromm	P. A. Lottes
L. Genens	J. F. Matousek
V. C. Hall	M. Petrick
D. R. Jacobson	D. J. Roy
V. M. Kolba	J. A. Thie
C. V. Pearson	

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PREFACE

A "Hazard Summary Report on the Experimental Boiling Water Reactor (EBWR)" was issued as ANL-5781.

Since that time numerous tests have been made on the reactor and plant up to a power level of 62 Mw(t). From the results of these tests it appeared that, with some modifications, EBWR could feasibly operate at 100 Mw(t). This Addendum to ANL-5781 is submitted with a discussion of changes in the hazards evaluation that may be encountered in operating EBWR at 100 Mw(t). Additional equipment is installed to handle the increased power output.

This report is not intended to repeat information already presented in ANL-5781; rather, it provides to supplement the information. One exception is Section VI, entitled "Enumeration and Evaluation of Possible Hazards," which has been completely revised and is presented in full in this report.

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HAZARDS EVALUATION REPORT ASSOCIATED WITH THE OPERATION OF EBWR AT 100 MW

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I. INTRODUCTION

This hazards report covers the basic modifications to the EBWR plant as designed to increase the present operating design rating of 20 Mw(t) to a new rating of 100 Mw(t). It is written as a supplement to ANL-5781 (the original hazards report) and follows the outline of that report for purposes of cross reference.

Successful stable operating experience up to 61.7 Mw(t) power has been achieved with a 4-ft diameter core. Predictions have been made for the performance of the 5-ft maximum diameter core, which can be installed in the present reactor, and stable operation at 100 Mw(t) is indicated.

Changes and additions to the plant, core, reactor vessel and other major components are shown. An analysis of the effect of these changes on the performance and safety of the reactor is presented.

A re-evaluation of the previous hazards (reported in ANL-5781) was made and some additional hazards brought about by the increased power level modification are covered.

II. SUMMARY

The Experimental Boiling Water Reactor (EBWR) was originally designed as a test reactor to prove the feasibility of a direct-cycle boiling water reactor integrated power plant. The reactor was designed to produce 20,000 kw of heat in the form of 600-psig steam, which was fed to a turbo-generator which then generated 5,000 kw of electricity. This power level was considered as a minimum design in order to extrapolate this information for future use in designing large central station power plants.

Experiments at this and higher powers resulted in a prediction of stable operation as high as 66 Mw(t) with the initial 4-ft diameter core, and short-term operation of the plant at 61.7 Mw(t) has confirmed the prediction. A detailed study of EBWR stability was made by taking a series of transfer function measurements, relating flux or power level to a reactivity input. Data taken at various power levels, steam pressures and control rod position were analyzed and extrapolated to predict stability at higher operating powers.⁽¹⁾

A simplified model of the EBWR kinetics was developed.⁽²⁾ With each increased power level operation sufficient data were obtained whereby stability was predicted to at least 66 Mw(t). The AEC granted permission for a test run to 65 Mw(t). On March 20, 1958, the reactor was successfully operated at 61.7 Mw(t). The power was limited at this point by the fact that the feedwater pumps were operating at maximum capacity.

Although frequent shutdowns were required to expedite the experimental program, the reliability of the plant was demonstrated by several runs, one of these extending for over three months.

One important outcome of these experiments has been to demonstrate a method whereby boiling reactor stability can be evaluated safely in a quantitative manner. By converting the above results to a 5-ft diameter core, which can be installed in the present reactor vessel, operation at 100 Mw(t) appears to be quite feasible. The core will be cooled by natural convection of H₂O and moderated by the H₂O coolant. Modifications and additions will be made to absorb or dissipate the 100-Mw(t) power.

Experimental work in the enlarged facility will be concerned initially with gathering operating data and evaluating the characteristics of the 5-ft diameter core design in order to permit comparison with similar information obtained from the original 4-ft diameter core. This phase will be followed by prolonged maximum power testing of the core to demonstrate the lower fuel costs attainable through extended fuel burnup.

Eventually plutonium-spiking demonstrations and testing are anticipated. In regard to the plutonium and fission product content of the core during 100-Mw(t) operation, it should be pointed out that hazards analysis of very high burnups (1% of total uranium has fissioned) has been made (ANL-5781, page 156). At the start of the 100-Mw(t) operation, the plutonium and fission product content (even should one of the spikes be of plutonium) is well below that already analyzed. The use of plutonium spikes will be prefaced by a separate hazards analysis report.

A. Modifications

1. Initial Preparations

Before any modifications are made in the reactor vessel, the present core loading will be completely removed and transferred to the fuel storage pit. The fuel racks in the storage pit are fabricated of boron-stainless steel and are calculated to prevent approach to criticality under any conceivable arrangement of stored fuel. However, the pit will be monitored for criticality during the storage operation.

The existing control rods will be detached from the drive mechanisms and removed from the vessel by a remote operation. The rods will be stored temporarily in the fuel storage pit and eventually removed from the building. The existing 4-ft stroke drive mechanisms will be removed from the vessel nozzles in the sub-reactor room.

Operations in the sub-reactor room may be complicated by the presence of a high radiation background in the area. In order to reduce hazard and inconvenience to a minimum, the vessel and system will be cleaned as thoroughly as possible prior to the installation of the new vessel components described below, and prior to recharging the core.

2. Core

The 5-ft diameter core will comprise the following sub-assemblies:

- 112 partially enriched (1.44% U^{235}) uranium fuel (existing)
- 7 natural uranium fuel (existing)
- 1 source dummy (existing)
- 28 highly enriched uranium fuel (new)
- 148 total core locations

The new fuel subassemblies will have 5-ft active lengths, as compared with the 4-ft active lengths in the present fuel, in order to permit their continued use later when the present fuel is replaced with 5-ft long assemblies.

3. Control Rods

In order to provide adequate control for the new 5-ft long fuel subassemblies, control rods having 5-ft absorber lengths are necessary. Nine cruciform rods having 2% boron-stainless steel absorber sections similar in dimensions (except for the increased length) to the boron-stainless steel rods presently in use will be fabricated. The new rods will be provided with followers, each containing 200 gm of U^{235} fuel, equivalent to about one-third the present partially enriched thin fuel subassembly, in order to enhance the control capabilities of the rods. The voids in the control rod channel will be about 63% at the worst condition [max flux = 145,000 Btu/(hr)(ft²)].

A tenth control rod, smaller and lighter than the others, will be fabricated for use in the center. The control rod oscillator mechanism will be attached to this rod during transfer function and stability tests. This rod will feature a 2% boron-stainless steel absorber section and a Zircaloy-2 follower.

4. Control Rod Drive Mechanisms

Again in order to accommodate the new 5-ft active length fuel subassemblies, control rod drive mechanisms capable of lifting the rods through a full 5-ft stroke are required. The present mechanisms provide a 4-ft stroke and are not adaptable to longer strokes because of column limitations and seal problems. Rack-and-pinion-type mechanisms which have been tested at the Laboratory will be installed.

The 5-ft stroke drive mechanisms will be locked to the new control rod extension fittings and bolted to the vessel nozzles in the sub-reactor room in the same manner as the existing drives. Some slight electrical changes will be made to accommodate the rack-and-pinion mechanisms, and an additional platform will be constructed to support the new drive motors, which must be located at a higher elevation for this type of drive.

5. Core Riser

Experimental hydrodynamic investigations at the Laboratory have shown that the addition of a riser or chimney to the top of the reactor core will materially increase the rate of flow of coolant by natural recirculation and enhance the likelihood of achieving high power densities. The riser produces a greater head differential between the low-density steam-water mixture leaving the core and the water in the downcomer space. The resulting increased recirculation permits a greater steam production rate for the same steam void fraction in the core. The latter is an important factor in reactor stability.

The riser will be fabricated from Type 304 stainless steel ($\frac{1}{8}$ in. thick). The lower cylindrical portion will be bolted to the core support plate using the studs originally provided for the forced-circulation shroud. The upper portion of the riser is conical to an elevation where the cross-sectional area is equal to the flow area of the core, at which point a transition to a cylinder is made. The conical reduction in core effluent area effects a corresponding increase in downcomer area. This, in turn, reduces the downcomer flow velocity and enhances the disengagement of steam from the recirculating water.

The upper conical portion of the riser is removable in order to permit access to the outer rows of fuel subassemblies for loading operations. Bayonet-type locking bolts will hold the upper shroud in place.

The lower cylindrical portion of the riser will be mounted on the core support plate after completion of the modifications to the vessel proper. The upper conical portion will be placed in position after the core has been loaded.

6. Pressure Vessel

In order to permit removal of the quantity of steam corresponding to 100-Mw(t) operation, removal of the present steam collection ring and replacement of the present 6-in. steam outlet nozzle with one of a larger size is necessary. If the present arrangement were allowed to remain, excessive steam velocity, pressure drop, and surface erosion might result.

In order to effect this replacement, it will be necessary to remove the shielding concrete blocks, lead bricks, and cooling coils from part, or all, of the upper pipe tunnel in the East face of the reactor shield (between the condenser floor and main floor). The present steam piping on the outside, and the steam collection ring on the inside, of the vessel will be cut away, using a maximum of precaution to prevent the dropping of chips into the vessel. A hole of proper size to accommodate a 10-in. nozzle will be cut (with a special boring machine) at the same location as the present nozzle, and the new nozzle welded in place. New 10-in. steam piping will be welded in place. All pipe welds within the biological shield will be completely radiographed.

A system of baffles fabricated from $\frac{1}{8}$ and $\frac{1}{16}$ -in. thick Type 304 stainless steel will be installed in the top of the reactor vessel to assure collection of steam from the highest feasible point in the vessel. The system is made up of a box section welded to the steam outlet nozzle.

The baffle system is designed to permit operation with the higher water level in the reactor and still provide a means of steam separation.

A new 6-in. feedwater inlet nozzle and distribution ring will be installed above the top of the conical section of the new core riser. The new arrangement will permit the larger feedwater flows required for 100-Mw(t) operation without excessive pressure drop, and will provide for injection of the colder feedwater at the top of the downcomer where it will be most effective in collapsing entrained steam bubbles in the recirculating water, and in increasing the driving force for natural recirculation.

The radiation levels that will be encountered in performing work on the reactor vessel will not be known until the core and the shielding have been removed. The portions of the vessel which are affected by alterations are near the top. The removable stainless steel shield plug, which has been in place in the top of the vessel during operation, has shown negligible induced activity.

7. Power Plant

The present turbogenerator will be used to absorb 20 Mw of thermal energy to generate 5 Mw of electrical energy. Additional heat-removal equipment will be installed to transfer 80 Mw of thermal energy into the steam distribution system at the Laboratory site. An intermediate steam loop has been incorporated to prevent virtually any possibility of contaminated steam from escaping into the heating system. Air-cooled steam-condensing units and the necessary heat exchangers are also added to ensure (1) continuous operation of the plant at maximum capability during the summer months when the Laboratory steam demands are sharply reduced, and (2) availability of low-temperature reactor feedwater for greater flexibility in experimental operation. The selection of air-cooled steam condensers to dissipate the heat was dictated by the increasing site-wide shortage of available water.

8. Containment Vessel Penetrations

A number of new penetrations will be required through the gas-tight steel containment shell which houses the power plant building. The additional openings will accommodate power, control, and instrumentation cables to service the new equipment within the building, and several steam effluent and condensate return pipe lines interconnecting the power plant building and the adjacent reboiler building.

Prior to startup, all new and old penetrations and closure gaskets will be leak tested to ensure original airtightness of the containment vessel. The maximum acceptable leakage is 1000 ft³/day.

B. Plant Parameters

The major design and operating features of the EBWR Facility, as modified, are summarized in Table I.

TABLE I

Plant Parameters

<u>General</u>	
Type	Direct-cycle boiling reactor
Heat output	20,000 kw for turbine; 80,000 kw for heating
Gross electrical output	5,000 kw
Operating pressure	600 psig
Operating temperature	489°F
Coolant	H ₂ O

TABLE I (Cont'd.)General (cont'd.)

Moderator	H ₂ O
Fuel	U ²³⁵ , U ²³⁸

Core

Active diameter	5 ft
Active height	5 ft
Total uranium content	6.4 tons
Total U ²³⁵ content	65-90 kg
Average water/uranium volume ratio	3.64
Structural material	Zircaloy-2 and Al

Fuel Assemblies

Total number in core	147 (~20% of assemblies are spiked)
Cross-sectional dimensions	3.75 x 3.75 in.
Number of plates per assembly	6
Number of rods per spiked assembly	49
Side plates	0.060-in. thick Zircaloy-2
Cladding	0.020-in. thick Zircaloy-2
End fittings	Stainless steel
Composition of fuel "meat"	93.5% U, 5% Zr, 1.5% Nb (by weight)
Composition of spikes	Zr, Al structure with U ₃ O ₈ (>90% enriched)
Thickness of meat in thin plates	0.174 in.
Thickness of meat in thick plates	0.239 in.
Thickness of meat in rods	0.325 in. diameter
Width of water channels between thin plates	0.428 in.
Width of water channels be- tween thick plates	0.360 in.
Power generation in aver- age assembly	≤ 0.68 Mw

TABLE I (Cont'd.)Nuclear Data

Average thermal flux	$3 \times 10^{13} \text{ n}/(\text{cm}^2)(\text{sec})$
Neutron lifetime	$6 \times 10^{-5} \text{ sec}$
Reactivity to control	$\sim \leq 13\% \text{ k}$
9 control rods with 8 fuel followers	$\sim 15\% \text{ k}$

Heat Transfer and Fluid Flow

Average power density in core coolant	
Spiked zone	135 kw/liter
Unspiked zone	115 kw/liter
Maximum power density in core coolant	
Spiked zone	418 kw/liter
Unspiked zone	356 kw/liter
Steam flow (at 100 Mw)	300,000 lb/hr
Average steam voids in heated channels	18.9%
Average steam voids in total moderator	16.5%
Average exit steam quality	3.2%
Lb of water circulated per lb of steam	32
Subcooling at core inlet	12°F
Nonboiling height of core	40% (avg.)
Average coolant inlet velocity	6.3 ft/sec
Average heat flux	
Spiked zone	136,000 Btu/(hr)(ft ²)
Unspiked zone	157,000 Btu/(hr)(ft ²)
Maximum heat flux (Based on max/avg heat flux ratio of 3.1)	485,000 Btu/(hr)(ft ²)
Estimated burnout heat flux	>1,000,000 Btu/(hr)(ft ²)
Maximum surface temperature	514°F

TABLE I (Cont'd.)Heat Transfer and Fluid Flow (cont'd.)

Average fuel centerline temperature	
Thick plate (enriched)	579°F
Thin plate (enriched)	567°F
Rod (Spike)	558°F
Maximum fuel centerline temperature	
Thick plate (enriched)	767°F
Thin plate (enriched)	724°F
Rod (Spike)	694°F

Control Rods

Total number	9
Spacing	12.75 x 12.75 in.
Shape	cruciform: 10 x 10 in.
Material	Boron-stainless steel
Thickness	0.250 in.
Boron content	2 wt %
Penetration of absorber into core	60 in.
56-in. travel time	1.35 sec.
Maximum withdrawal rate	0.4 in./sec. ~0.01% k/sec
Strength of 9 rods	~15% k

Pressure Vessel

Diameter, inside	7 ft
Height	23 ft
Working pressure	600 psig
Design pressure	800 psig
Thickness of cylindrical portion	2 $\frac{3}{8}$ in.
Material	SA 212 Grade B boiler plate
Cladding	0.1-in. Type 304 stainless steel
Thermal shield	1-in. 18-8 stainless steel containing 1% boron

TABLE I (Cont'd.)Pressure Vessel (cont'd.)

Relief valve settings	650, 700, 725, 750 and 775 psig
Total weight	60 tons
Weight of contained water	14 tons

Power Plant

Condenser pressure	$2\frac{1}{2}$ in. Hg abs.
Flow rate to cooling tower	13,650 gpm (max)
Feedwater flow rate	600 gpm
Feedwater temperature	120°F
Generator output voltage	4,160 volt

Steel Building

Diameter	80 ft
Height	119 ft
Net volume	~400,000 ft ³
Design pressure (internal)	15 psig
Design pressure (external)	0.5 psig
Maximum leakage rate at 15 psig	1,000 ft ³ /day
Type steel	ASTM A-201 Grade B Firebox Quality
Sides of tank	$\frac{5}{8}$ in. thick (cylindrical)
Bottom of tank	$\frac{5}{8}$ in. thick (elliptical)
Top of tank	$\frac{3}{8}$ in. thick (hemispherical)
Lining of tank below grade	2 ft (min) concrete
Lining of tank above grade	1-ft thick concrete
Capacity of water reservoir	15,000 gal
Flow rate of sprinkler system	1,000 gpm
Ventilation air flow rate	3,000 cfm
Personnel access	Air-lock doors
Freight access	Bolted and gasketed door

III. DESCRIPTION OF THE EBWR FACILITY

A. Site

A new population census has not been made since the previous hazard summary report was submitted. However, no unusual changes in population have occurred to alter materially the original analysis given in ANL-5781.

B. Reactor

1. Mechanical Design

The present reactor plant will remain essentially intact but will have additional heat-removal equipment to handle the increased power output.

Modifications to the reactor vessel include additional fuel, longer control rods with rack and pinion drives, a new shroud, and larger steam and feedwater nozzles to handle the five-fold increase in flow.

(a) Core

The core, shown in Fig. 1, consists of vertical, closely packed fuel assemblies, supported at the bottom by a lower grid plate and at the top by a metal guide structure. A fuel assembly can be placed in each of the 4-in. square locations shown. The fueled follower control rods, guided in metal channels, are driven up out of the core by control rod drive mechanisms located in the sub-reactor room. As the poison sections of the rods are removed from the core, the fueled portion of the follower rod is inserted into the active lattice.

The bottom grid is $5\frac{3}{4}$ -in. thick, Type 304 stainless steel plate installed at the bottom of the pressure vessel to support the weight of the core assembly and to fix the position of the fuel assemblies and control rods. The grid support members are designed to provide the necessary flexibility for thermal and elastic variations. Cooling water holes are provided to reduce temperature rise due to gamma and neutron heating. This grid and the guide structure are removable from the pressure vessel in the event that a revised core arrangement is desired.

Each of the present 4-ft active length (Core I) fuel assemblies contains six fuel plates (see Fig. 2). Zirconium side plate extensions and stainless steel end fittings make the overall length of the fuel assembly $79\frac{5}{8}$ in. The fuel assemblies are of two types: natural, using a natural uranium alloy, and slightly enriched, using a low-enrichment uranium alloy. The plates for the natural assemblies contain a heat-treated

FIG. 1

MODIFICATIONS IN EBWR FOR OPERATION AT 100 Mw

ternary alloy of 93.5 wt % natural uranium (0.72% U^{235}), $1\frac{1}{2}\%$ niobium, and 5% zirconium, clad with 0.020 in. of Zircaloy-2. The plates for the slightly enriched elements contain a "meat" of 1.44% U^{235} enriched uranium, clad with 0.020-in. Zircaloy-2. The Zircaloy-2 side plates are thick enough to permit welding to the fuel plates and are perforated to stretch (creep) if the fuel plates elongate under irradiation.

The "spike" concept for attaining 100-Mw operation is predicated on (1) a minimum amount of new fuel element construction; (2) the maximum utilization of the existing Core I fuel elements; (3) the first step is the enlargement of the core to a full 5 ft x 5 ft core. The

spike fuel assembly contains forty-nine rods (see Fig. 3). The fuel rods are a dispersion type, having U_3O_8 of greater than 90% enrichment dispersed in an X-8001 aluminum alloy matrix and clad with Zircaloy-2. The bond between the "meat" and clad is by means of mechanical contact. The fuel rods are positioned in the assembly by slotted end fittings. Differences in thermal expansion of the rods are allowed for by fastening one end of the rod to the grid support structure and allowing the opposite end to move axially, guided by the slotted end fittings.

FIG. 3
SPIKE FUEL ROD ASSEMBLY AND FRAME

The fuel assembly is of the reversible type, being symmetrical end for end, and slides into a fuel-element frame. Since the spikes have a 5-ft effective fuel length and the balance of the core has 4-ft effective fuel length, the reversible feature allows the ultimate full use of the fuel.

The fuel frame (see Fig. 3) consists of Zircaloy-2 in and above the core region with a stainless steel locating tip end fitting. Two sets of Inconel springs are located at the top of the frame for positioning with regard to the present 4-ft fuel elements and the future 5-ft fuel elements. Nine fuel assemblies occupy the 12-in. square grid cells between control rod guide channels.

Two different thicknesses of fuel plates will be used (as in the present loading) until the 4-ft elements are removed. The thin plate (0.174 in. U + 0.040-in. Zircaloy-2 clad) assemblies give a water-to-uranium volume ratio of 4.416. The thick plate assemblies (0.239 in. U + 0.040-in. Zircaloy-2 clad) give a water-to-uranium ratio of 2.759.

The design of the core permits some flexibility of loading. There are a total of 148 fuel positions which, when filled, produce a core 5 ft in diameter. For 100-Mw(t) operation, all 148 positions are filled, 147 with fuel and 1 as a source. Of the 147 fuel assemblies,

56 are Enriched Thick Elements	(4 ft long)
56 are Enriched Thin Elements	(4 ft long)
7 are Natural Thick and Thin Elements	(4 ft long)
28 are Enriched Spike Elements	(5 ft long) .

The 28 spikes will be located in a square surrounding the central 36 four-foot long fuel elements. The amount of reactivity produced by the U^{235} in the spiked region will be determined experimentally during pre-startup operations.

Uniform lateral support for the top ends of the fuel assemblies is provided by the stainless steel and Zircaloy-2 end fittings with spring leaves on all sides, which prevent vibration of the fuel assemblies. These spring leaves, in conjunction with the tapered seat at the lower ends, provide a degree of freedom of movement in case of fuel assembly distortion. A total of $\frac{1}{4}$ -in. distortion in two directions can be accommodated. No provisions are made for holding the fuel assemblies down (they are free to expand and contract lengthwise) as no forces which could dislodge them are envisioned. Should less than a full loading be used, stainless steel dummy fuel assemblies are provided to fill the vacant holes.

The positioning of the spikes in the core will cause the flux at the outer periphery to be raised and a degree of power flattening achieved.

(b) Control Rods

Control of the reactor is effected by nine cross-type control rods which can be withdrawn individually and inserted individually or simultaneously. All nine control crosses (see Fig. 4) are $\frac{1}{4}$ in. thick x 10 in. wide and are made of 304 stainless steel containing 2% boron by weight. In addition, the 10-in. wide portion of the control rod follower will contain fuel. The fueled crosses will be $\frac{1}{4}$ in. thick x 10 in. wide and 36.5 in. long. The fuel will be UO_2 of greater than 90% enrichment dispersed in a 304 stainless steel matrix and clad with 304 stainless steel. The poison portion of the rod will be welded to the fueled follower section. This subassembly, approximately 8 ft long, will be detached from the 3-in. follower section for ease of handling in a coffin when removed from the core.

FIG. 4
5-FOOT CONTROL ROD WITH FUELED FOLLOWER

The nine rods are estimated to have a combined strength of 15% k. The total amount of k to be controlled is expected to be about 13%, distributed approximately as follows:

Operating voids (18.9% of coolant channels)	~ 4%
Temperature density effects from room temperature to operating temperature (489°F)	~ 1%
Xenon (at room temperature)	~ 2%
Burnup	~ 6%
	~ 13%

Additional control can be obtained by adding 1% boron-stainless steel strips ($\frac{1}{16}$ in. thick) to three sides of the spike fuel sub-assemblies. Assuming (1) no self-shielding of the boron and (2) all spikes contain a maximum of 3 strips, calculations (a simple combination of diffusion and transport theory) indicate the reactivity worth of the boron averaged over all the spike elements would be about 15% k. The number of strips inserted and their relative positions with respect to each other, the control rods, etc., will determine the amount of self shielding for a given arrangement. The self-shielding factors will be evaluated experimentally.

When in operation, the control rods are held in the shroud structure above the core (see Fig. 1). The upper portion of this shroud structure serves to position the upper ends of the fuel assemblies.

The control rods and fuel follower are guided in a fuel and control rod guide structure consisting of a sheet metal assembly, 94 in. high, mounted on the bottom grid. It is made of $\frac{1}{8}$ -in. zirconium and stainless steel plates spaced with zirconium and stainless steel bars. This assembly provides nine crossed-shaped channels for the fueled follower control rods and a number of cells for the guidance of the upper ends of fuel assemblies. The lower portion in the region of the active core is zirconium; the upper part is stainless steel. Available data on radiation damage indicate no distortion of this structure need be expected for at least 10 years.

(c) Control Rod Drive Mechanisms

The design of the control rod drive mechanisms, located outside the reactor shield and below the reactor, is shown in Fig. 5. The mechanism incorporates a rack and pinion for translation of the rotating drive motion into linear motion of the control rod. A magnetic clutch is used as the connection between the pinion shaft extension and the drive motor. Upon interruption of its electrical circuit, the magnetic clutch is released, allowing rapid rod insertion.

The weight of the rod is sufficient for the rod to drop the 56-in. free fall into the core in 1.35 sec. A restricted section in the bottom of the vessel makes it impossible for the rods to drop beyond the core, and a mechanical stop at the end of the rack limits upward travel. All rods can be released for rapid insertion simultaneously.

A one-way clutch is provided to assist rod insertion. A torque-limiting device is used to prevent damage to the control rod and its components that may be mechanically restrained from dropping into the core.

Only one rod at a time can be withdrawn from the core. The drive for the mechanism will be a standard gear motor with a magnetic brake. Change gears are used to vary the speed as required for different control rods and in order to limit the maximum rate of addition of Δk to $\sim 0.01\%/sec$.

Motion	28 in./min
Position Indication	within 0.020 in.
Travel	60 in.
Quick return travel of 56 in. in 1.35 sec. The last 4-in. stroke is slower due to water dashpot action.	

The seal design used in the mechanism is of a breakdown labyrinth type. This type of seal breaks down the pressure to atmospheric conditions. The labyrinth breakdown rings fit on the shaft with a definite clearance. Extensive tests have been conducted previously on this type of seal and are reported in ANL-5220.⁽⁸⁾

FIG. 5
CONTROL ROD DRIVE MECHANISM

The same facility as was previously used on the original 4-ft rods was used to test the mechanism. Some 10,000 cycles have been completed (including 1,000 full-stroke rapid insertions) with negligible wear on the rack, pinion, bearings, seal shaft and seal rings. The test conditions were room temperature at 600 psig. Since the temperature of the reactor thimbles is held below 200°F, the application of heat was not essential to the test. In principle, this test was run to check out the design and construction. It has been found that reactor conditions cannot be simulated more realistically. Operation with simulated dirt and deposits does not provide meaningful data.

(d) Pressure Vessel

In order to remove the quantity of steam corresponding to 100-Mw(t) operation, the present steam collection ring will be removed and the present 6-in. steam outlet nozzle will be replaced with a nozzle of larger size. If the present arrangement were allowed to remain, excessive steam velocity, pressure drop, and surface erosion might result.

A system of baffles will be installed in the top of the reactor vessel to assure collection of steam from the highest point in the vessel. The baffle system is designed to (1) permit operation with a high water level necessitated by the new riser in the reactor; and (2) prevent a large slug of water from entering the system.

A larger feedwater inlet nozzle and distribution ring will be installed just below the top of the conical section of the new core riser. This arrangement will permit increased feedwater flows without excessive pressure drop and will provide for injection of the colder feedwater at the top of the downcomer, where it will be most effective in collapsing entrained steam bubbles in the recirculating water and in increasing the driving force for natural recirculation.

(1) Applicable Design Code

The design and installation of these nozzles meet all the requirements of Section I "Power Boilers" of the ASME Boiler and Pressure Vessel Code which was the original basis of design for the EBWR pressure vessel. Moreover, the design meets the supplementary mandatory requirements of the applicable ASME Code Case rulings for nuclear pressure vessels.

(2) Thermal Heating and Stresses

The EBWR pressure vessel design was based on operation with heavy water at a power level of 40 Mw(t). Operation with light water at a power level of 100 Mw(t) is much less severe and therefore does not present any additional problems of thermal stresses due to neutron and gamma heating.

The 1-in. thick thermal shield originally installed for D₂O operation will not be required for 100-Mw(t) light water operation. However, it will not be removed from the vessel.

(3) Irradiation Effects

As part of the irradiation surveillance program for the EBWR pressure vessel, tensile and impact specimens of the parent SA 212-B plate stock were irradiated in the MTR to an integrated fast (>1 Mev) neutron flux of 0.9×10^{20} nvt (or higher). All samples showed an appreciable increase in the ductile-brittle transition temperature. Hardness and brittleness values increased at room temperature.

Similar samples exposed to fluxes of 2×10^{18} nvt under comparable conditions by other workers did not show any appreciable change.

Calculations indicate that the EBWR pressure vessel has accumulated a total integrated fast (>1 Mev) neutron flux of 8×10^{17} nvt. The fast flux calculated for the worst point at the vessel wall is 1.8×10^{10} nvt for the 20-Mw reactor, and 1.7×10^{11} nvt for the 100-Mw reactor. Numerous foils and wires were inserted (June, 1959) into the annulus between the vessel wall and biological shield to verify the calculated fast neutron flux values, and to obtain information about the fast neutron spectrum. The data are being analyzed. Future supporting studies will include irradiation of SA 212 B samples in various test locations in the EBWR.

Recent data of Trudeau* indicate that a fast neutron exposure of steel to 1×10^{19} nvt (>1 Mev) may be tolerated. Using the calculated fast fluxes as a measure of damage, the EBWR can be operated for about 60,000 Mwd before the value 1×10^{19} nvt is attained.

(4) Insulation

The vessel is provided with thermal insulation consisting of 3 in. of stainless steel wool on all surfaces. Thus, the thermal heat loss from the vessel is about 12.5 kw, whether reactor is operating at 20 or 100 Mw.

(e) Primary Shield

The water-cooled lead shield surrounding the vessel was designed for 40-Mw operation with D₂O. Consequently, the system is

*L. P. Trudeau, INCO (Canada), personal communication.

capable of removing the heat transferred from the vessel to the lead, and the nuclear heat generated in the shield, during 100-Mw operation with H_2O .

Shield effectiveness is greater than necessary; measurements indicate that at 100 Mw(t) the background radiation should be well below allowable levels.

(f) Fuel Handling and Storage

New fuel elements are not radioactive and can be lowered into the reactor vessel directly by means of an unloading rod. When all loading operations are completed, the reactor vessel cover is installed, thermal insulation and top shielding plugs set into place, and finally the holddown beams are locked in position.

A positive locking device is incorporated into the unloading rod gripper mechanism to prevent the fuel from suddenly dropping into the core during loading operations. The fuel element cannot be released from the loading rod until all weight is released from the loading rod gripper. A positive unlocking action must be performed by the operator before the fuel element can be released.

Irradiated fuel is removed from the reactor by means of a lead shield transfer coffin on a carriage. The loaded coffin is moved to a position over the water-filled fuel storage pit and the fuel assembly is lowered through a shielded chute into the pit, where it is placed in a boron-stainless steel storage rack.

The fuel storage area is a shielded concrete pit (24 ft deep) integral with the reactor shield. The water in the fuel storage well provides adequate shielding above the fuel elements. The storage rack is mounted in one side of the pit, approximately 15 ft below the water surface and adjacent to the shielded chute. The storage rack is a rectangular grouping of 4-in. square boxes formed by interlocking $\frac{1}{8}$ -in. thick sheets of stainless steel, and 1 and 2 wt % boron-stainless steel. The stainless sheets are oriented to isolate rows of seven elements. Calculations indicate 29.5% reactivity may be controlled by this arrangement when the storage rack is loaded with 148 - 1.4% enriched elements. Additional boron-stainless steel may be placed in the storage rack when storing the 28 - 5-ft core fuel elements (fully enriched spikes).

The well has water inlet and drain connections which may be used to circulate the water through filters and coolers or ion exchangers, if necessary.

Spent control rods and other irradiated core components are stored at the bottom of the fuel storage pit in such a manner as to permit free pit floor space for a shipping coffin. Spent fuel is loaded

into the shipping coffin by lowering the coffin to the bottom of flooded pit, transferring the elements to the coffin, closing the coffin loading door and then removing the loaded coffin from the storage pit.

(g) Startup Heating

The reactor vessel and its contents will be heated to the startup temperature (325°F) before the control rods can be withdrawn. This has been the previous practice and will not be changed.

(h) Shutdown Cooling

The decay heat which must be removed from the reactor drops from a maximum of 6% of operating power immediately after shutdown to about 1% after 7 hr. During a normal shutdown the reactor is cooled by a secondary cooler in the reactor purification system. From previous operating experience, this cooler should be adequate to handle the cooling requirements when the reactor is shut down from 100 Mw. In the event that one cooler is insufficient, the standby cooler can be utilized.

The circulating water pumps will be in operation to supply cooling water for the main condenser to condense any steam that enters the condenser from the steam traps in the steam lines.

In event of a shutdown caused by a loss of electrical power, the Emergency Shutdown Cooler, which is located within the original steam dryer, is used to remove the decay heat. Water flowing by gravity from the 15,000-gal water storage tank is boiled inside the tubes, condensing reactor steam on the shell side. The condensate returns to the reactor by gravity flow. The steam-water mixture is returned to the overhead tank to heat up the 15,000 gal of water; circulation is established by the difference in density in the water supply and steam-water discharge lines. The main steam valves are closed when this cooler is in use, so that the reactor water level will not fall except for what little is lost through the steamtraps. The cooler is designed to remove 1000 kw when the steam pressure is at 600 psig.

An evaluation to determine whether the emergency cooler initially installed in EBWR would be sufficient to dissipate the decay heat in the event emergency cooling was required after operating at 100 Mw was made. Figure 6 shows the decay heat expected from the 100-Mw core as a function of time after shutdown, as based on work performed at the Laboratory.⁽⁹⁾ The curve depicts decay power rates to be expected after an infinite time of reactor operation.

Results from a test indicated that, after 20-Mw operation, the total heat removed by the emergency cooler plus the system heat losses during the first hour after shutdown was about 2 Mw-hr;

FIG. 6
100-Mw(t) ENERGY GENERATION vs. TIME AFTER SHUTDOWN

consequently, it is reasonable to predict that the cooler is capable of handling this heat input over an extended period, since there is ample heat-removal capacity in the 15,000-gal low overhead storage tank (Fig. 7).

FIG. 7
STORAGE TANK WATER TEMPERATURE VS. TIME AFTER EMERGENCY SHUTDOWN

A laboratory water line capable of supplying 80-100 gpm of makeup to the storage tank is controlled by a float valve; therefore the availability of the storage tank as a decay heat sink is dependent only upon

the ability to supply makeup within 19.7 hr after the failure of the water supply. The maximum makeup rate required would be 7 gpm/Mw of decay heat, provided there was no recovery due to the steam condensing on the underside of the dome and draining into the tank.

Figure 8 depicts the pressure rise that would occur in the reactor as a function of time after emergency shutdown if there were no means of relieving the pressure. It can be seen from the curve that, even in the event of a complete failure of all the safety and relief valves, the design pressure of the reactor vessel, 800 psig, would not be exceeded. The reactor is equipped with an 8 in. by 4 in. steam-powered regulating valve (Foster Relief Valve) designed to limit the reactor to a preset pressure of 650 psig by allowing steam to flow to the main condenser via the desuperheater section. Integrating the area under the curve in Fig. 8 between the highest pressure value to be expected and the 650 psig set point and dividing by the latent heat content of the steam passed through the regulating valve yields an estimate of the amount of water that will be lost from the reactor during the period that the regulating valve is open. Calculations indicate that approximately 1200 lb of reactor water will be flashed and the water level in the reactor will drop about 7.5 in. It is reasonable to assume that the large, cold, condenser surface will be sufficient to cool this amount of steam, even should there be no flow of coolant, without exceeding the 20-psig limit set on the condenser by the relief diaphragm.

FIG. 8
100-Mw(t) REACTOR PRESSURE vs. TIME AFTER EMERGENCY SHUTDOWN

As the pressure in the reactor rises, the temperature will also rise and the system heat losses will increase, since the total temperature differential between the system and the heat sink will become

greater; thus, the 2-Mw cooling rate is conservative. The integrated decay heat was based upon energy generation to be expected after an infinite reactor operating time; therefore, this figure is also on the conservative side. Since the calculations were based on conditions worse than those of actual operation and still yielded results that indicate satisfactory performance, the present EBWR steam dryer-emergency cooler and the Foster Relief Valve-condenser complex should be adequate for handling 100-Mw(t) emergency cooling requirements.

(i) Boron Injection

The high-pressure boric acid injection system is a standby system designed to reduce the reactor core reactivity below criticality if the control rods fail to operate properly under shutdown conditions.

The automatic injection feature will be eliminated since: (1) rapid boric acid injection is not necessary (it is a backup safety measure to be employed only after due consideration of conditions in the reactor); (2) at equilibrium-controlled operation, insertion of one rod is sufficient to override the void-controlled reactivity and reduce power.

The volume of concentrated boric acid solution will be increased to handle more than 13% reactivity. Manual injection of the solution will be accomplished by means of a pushbutton on the reactor console, by a handwheel on the wall of the control room, or by a lever in the power plant building.

(j) Reactor Water Purification System

The existing Reactor Water Purification System will be adequate for the proposed 100-Mw operation.

2. Instrumentation and Control

(a) Control Rod Drive Mechanisms

Each of the nine rod drives is operated by a reversible electric motor to give a rod speed, in terms of reactivity, of $\sim 0.01\%$ k/sec. One of the nine rods, probably the center one, will be selected as a regulating rod. A separate set of switches controls this rod. An additional set of switches controls the remaining eight rods, in conjunction with a rod selector switch which prevents simultaneous withdrawal of any two or more rods. It is possible to insert eight of the nine rods simultaneously.

Each drive is provided with an electromagnetic clutch which engages the pinion extension shaft to the drive linkage. Upon release of the clutch, the rod falls, due to gravity, to the "full in" or position of minimum k in the reactor. A travel of 56 in. (full out to dash pot) occurs in not more than 1.35 sec. Prior to startup, the drop time is measured to ensure that this value is not exceeded.

The rod clutch release used for this test releases only the rod selected, to facilitate individual rod-drop reactivity calibrations. A shutdown push button releases all the rods for dropping into the reactor. Interlocks operated by each rod at its "full-in" position are arranged to prevent the regaining of control power after a shutdown unless all rods are at the "full-in" position.

(b) Rod Position Measurement, Indication and Limit Switching

Since rod position measurement is by Synchro, with the transmitter geared to the pinion side of the magnetic clutch, the position indicated is that of the control rod. A current relay used with the magnetic clutch operates an indicating light to show that the magnetic clutch is energized and connected to the drive. A limit switch operated by the rod as the piston enters the dashpot activates an indicating light. The limit switch is also used to determine free-fall insertion times. The "rod out" limit also operates a light indicating this condition. An indicating light is operated by the rod selector switch to identify which rod drive can be operated. A number of warning lights throughout the plant area indicate that the rod drive control power is on, and a one-minute alarm bell sounds each time the control power bus is energized, announcing plant-wide warning that the reactor is operating or can be operated.

(c) Safety and Alarm Circuitry

Table II is a compilation of the signals that cause loss of control power with resultant insertion of all rods. Signals which cause alarm only will be added as required.

Interlocks require that before the reactor may be started up the reactor safety circuits listed in Table II must be cleared. In addition, the linear flux instrument shunt must be set to its most sensitive scale, all rods must be in their maximum shutdown positions and the period trip circuit must be operative.

TABLE II

Rod Insertion Signals

Plant area radioactivity high
 Reactor flux high - No. 1* electronic relay
 No. 1 magnetic-sensing instrument
 Reactor flux high - No. 2* electronic relay
 No. 2 magnetic-sensing instrument
 Reactor flux high - No. 3* electronic relay
 No. 3 magnetic-sensing instrument
 Reactor pressure high
 Reactor water temperature below 325°F
 Condenser pressure high***
 Reactor water level high
 Reactor water level low
 650-psig relief valve open
 Reactor safety valves open (700, 725, 750, 775 psig)
 Intermediate system safety valves open (375, 400 psig)
 Primary steam valve trip**
 Reactor period under 5 sec.**
 Turbine throttle trip
 Feed water pump power failure (ACB 3 and 4)
 Reboiler house radioactivity high
 Deaerator safety valve open

*Two of these electronic trips must operate in coincidence to effect a shutdown.

**Used only for startup with a key switch provided to lock it out of operation.

***Circuits have bypass switches to permit plant startup.

(d) Power and Period Measurements

There will be no change.

Figure 9 presents the present operating overall picture of the coverage and overlapping of the instrument channels. The scale is given in terms of reactor heat generation power expressed in watts.

(e) Temperature Interlocks

The present temperature interlock system is designed to prevent withdrawal of the control rods until the reactor water is heated to 325°F by an external heat exchanger using site steam. No change will be made in this system.

(f) Pressure and Rate of Change of Pressure Interlocks

The interlock for rate of change of steam pressure will be disconnected. This was used originally as an extra means of protection against sudden steam-valve closure. When the steam valve was closed at 20-Mw operation, a 3 psi/sec pressure rise occurred. The two-out-of-three electronic high-flux, one-out-of-three magnetic high-flux and any one of the high-pressure shutdown circuits provide adequate protection for the reactor without the added complication of the rate-of-pressure interlock.

(g) Reactor Water Level Measurement and Control

The feedwater can be controlled manually by motor-operated throttling valves in the reactor feedwater line. Bypass valves around both automatic feedwater valves can be operated from the control room. A second control system incorporating the components necessary

for a three-element feedwater controller is added to the new cycle. Steam flow, feedwater flow, and reactor level can be used for automatic control of feedwater flow from the reboiler cycle, as well as from the present turbine cycle. The turbine cycle feedwater control can also be arranged to control solely on the condenser hotwell level.

(h) Temperatures

All temperatures of auxiliaries closely allied to reactor operation, such as the reactor water ion-exchange system, shield cooling system, etc., are measured by thermocouples and registered on corresponding indicators or recorders.

3. Physics

(a) Introduction

The physics data reported in ANL-5781 were based on a core comprising 71 natural uranium and 42 fully enriched "spike" elements. However, the reactor has been operating with 114 elements, over a hundred of which have always been slightly enriched (1.44% U^{235}). It was not believed at that time that this discrepancy between the anticipated and the actual fuel elements would affect significantly any conclusions regarding reactor safety.

Similarly, it must be reported at this date that a discrepancy must necessarily exist between the anticipated cores and those actually to be used. The use of spikes with slightly enriched elements is, in a sense, a core concept, more closely resembling the spiked core of ANL-5781 than the core initially run in the reactor. However, the percentage of spiked elements used with the 1.44% enriched elements probably will be smaller (i.e., about 20% of all the elements are spikes) than that already reported for use with natural elements (37%).

Because of this similarity of core types, there is little significant change in the physics except for the plutonium buildup. It is estimated that the existing core, loading No. 51 (see Fig. 10) after intermittent use during 1959, will have a total irradiation of about 10,000 Mwd. With 6-7 kg of Pu^{239} present at that time, the spiking and subsequent additional irradiation will lead to somewhat larger Pu contents during 1960. It is hoped to accomplish by this the demonstration of boiling reactor behavior with substantial burnups present.

The reactor physics of $Pu^{239}O_2$ dispersed in X-8001 Al is being studied at present. Although seed-blanket-type cores are being investigated, the physics is somewhat similar to cores having a more uniform distribution of plutonium, such as occurs in the course of burnup.

FIG. 10
CORE LOADING NO. 51

The significant hazard aspect of plutonium-bearing cores (whether seeded or not) is the effect of the total Pu^{239} content on the delayed-neutron fraction. The total Pu inventory is also of interest. Relevant information will be included in the hazards analysis report preceding plutonium loadings.

(b) Reactivity Loss

Table III shows the results of room-temperature criticals during the reactor life. The reactivity due to burnup was obtained by correcting the overall reactivity loss for the individual fluctuations due to minor loading changes.

TABLE III
Reactivity Loss at Room Temperature Due to
Samarium, Fission Products, U^{235} Loss, and
Pu Buildup

<u>Loading No.</u>	<u>Mwd at End of Loading's Operation</u>	<u>% Reactivity above Loading 46 Due to Loading Change</u>	<u>% Burnup Reactivity Loss</u>
46	0	0	0
46	410	0	0.62
46	2120	0	1.4
46	3060	0	1.82
47	3856	0.76	2.18
48	4629	0.609	2.55
49	6667	0.629	3.47
50*	6987	0.76	3.63
51		0.48	

*Same as Loading No. 47

The rate of reactivity loss is seen to be 0.46% per 1000 Mwd at room temperature. It is of more practical interest to know the rate of loss at 489°F. A single experiment at 6987 Mwd shows 1.93% is lost at this temperature, or $\sim 0.7\%$ less than at room temperature. The rate of loss at 489°F is believed to be 0.36% per 1000 Mwd. Presumably this shows the positive contribution of plutonium to temperature coefficient.

Reactivity in Voids

According to theory, this quantity is more sensitive to control rod position than to burnup per se, or to minor loading changes. Table IV summarizes results obtained for this quantity, defined as the

reactivity present in a period if all the voids would collapse. The control rod position effect was explored by water boration during early experiments.⁽³⁾ This effect was confirmed approximately by equilibrium xenon rod positions (at 20 Mw), corrected for burnup during reactor life.

TABLE IV

Reactivity in Voids

Conditions	Results
19.6 Mw, all rods at 29.8 in. 0 Mwd, equilibrium xenon	2.2 %
13.5 Mw, all rods at 26 in. 6987 Mwd, no xenon	1.65 %
20.4 Mw, all rods at 28.5 in. 6987 Mwd, no xenon	2.2 %
Reduction in reactivity in voids by all rods moving out one inch at constant power:	
By water boration experiments	0.0278%
By burnup experiments	(0.036 ± 0.036)%

Temperature Coefficient

Four definitions of the reactivity loss between room temperature, 68°F, and 489°F have been found useful:

- a) Difference in all rods out, reactivities at 68°F and 489°F.
- b) The amount of subcriticality at 489°F with all rods at their 68°F critical position.
- c) The amount of supercriticality at 68°F with all rods at their 489°F critical position.
- d) An integral of small reactivity differences measured by maintaining criticality with the rods at all points between 68°F and 489°F.

The contribution of plutonium toward reducing the magnitude of the temperature coefficient is seen in Table V. According to definition d) the coefficient will be near zero in another 7000 Mwd. It should be pointed out, however, that it has been necessary to base these results on only one measurement at 6987 Mwd.

If the temperature coefficient continues to change linearly with Mwd exposure, the present core will probably remain negative during 1959. It is not anticipated that a ring of spikes added for the 100-Mw core should contribute any significant positive component to this coefficient. Hence, it is reasonable to expect the temperature coefficient, as defined by d), to be negative at the start of the 100-Mw core operation.

TABLE V

Per Cent Reactivity Loss from 68 to 489°F

	<u>a</u>	<u>b</u>	<u>c</u>	<u>d</u>
At 0 Mwd	0.91	3.13	2.41	2.94
At 6987 Mwd	0.26	1.55	1.19	1.45

The prompt or metal temperature coefficient used in safety analyses remains essentially unchanged from its initial negative value at zero Mwd. Therefore, when water-density and neutron-temperature contributions to the overall temperature coefficient are zero, or even slightly positive, this does not, ipso facto, represent a serious hazard. If a slight positive water-temperature coefficient develops, its presence is more readily apparent in mild excursions (i.e., periods longer than 0.1 sec) than in serious transients (i.e., periods much shorter than 0.1 sec). The magnitude of the excursions in the former case increases because ample time exists for the water to heat up. In the short-period excursions the time delay for heat to enter the water is long enough so that any positive reactivity addition would still be almost negligible and previous analysis is valid. However, the loss of inherent safety might be significant should a strong positive water-temperature coefficient develop in the course of time.

Value of the Dollar

For initial experiments⁽³⁾ the dollar was taken to be 0.825% reactivity (D. J. Hughes data). Calculations⁽⁴⁾ show that the effect of uniformly distributed plutonium is to change this to 0.825 (1 - Mwd x 10⁻⁵)%. Although experiments⁽⁴⁾ at 4000 Mwd agreed with this calculation, results at 9687 Mwd indicate the dollar to be 0.825 (1 - 0.13)% . The non-uniform distribution of plutonium may account for the discrepancy. As with the falling temperature coefficient, the dollar deflation does not appear to pose any serious hazard in the near future.

Core Life at 20 Mw Prior to a 100-Mw Change Over

In its design phase no particular lifetime was assigned to the initial core loading. To predict core life from the above experiments it will be defined as the number of Mwd at which 20-Mw operation with equilibrium xenon occurs with all control rods out. The result is a life of

$$(\% \text{ reactivity added to Loading 51}) \times 2780 + 9200 \text{ Mwd}$$

(c) Reactor Kinetics

The increasing macroscopic cross section for absorption, Σ_a , during the first 10,000 Mwd of reactor operation may lower the initial calculated neutron lifetime of 6×10^{-5} sec by 5%. Although subsequent irradiation further reduces the lifetime, the smaller Σ_a of the spiked zone will nullify this effect.

During the first 5000 Mwd of operation several room-temperature control rod calibrations were made. By comparison with the calibration obtained in December, 1956 (ANL-5711)⁽³⁾ at the same rod settings, it was possible to detect the changing value of the dollar. Figure 11 shows that the somewhat inaccurate experimental results confirm the theoretical curve. The ordinate is the deflationary factor on the dollar due to the lower delayed-neutron fraction for plutonium. An approximate theoretical description of this factor is

$$1 - \frac{\text{kg of Pu}^{239}}{65}$$

It is not anticipated that the combination of spiking and burnup will lead to deflationary factors less than about 0.6.

FIG. 11
EFFECT OF IRRADIATION ON KINETIC EQUATION FOR EBWR

(d) Core Loading and Control

The previously reported (ANL-5711)⁽³⁾ control rod calibration was obtained with 48 in. of poison (Hf and B) with Zr followers in a nonspiked core. With fuel followers having a total U^{235} content of

1.8 kg, the total rod worth may increase about 1% reactivity. Changing from Hf to B in 5 of the 9 rods should not reduce the total strength more than about 1% reactivity. Location of the rods in a spiked zone will cause their worth to vary slightly during core life due to differential burnout effects between the spiked and nonspiked zones.

The shape of the differential control rod worth curve previously reported (ANL-5711³) may change slightly due to the 33.5-in. fuel followers. This can be seen from the fact that during the first 7.25 in. of insertion there is a net motion of fuel toward the center of the reactor. This may nullify the negative 0.2% reactivity due to poison moving down 7.25 in.

This change in rod characteristic per se does not represent a serious hazard. However, in combination with a series of three low-probability events, this change, along with reduced shutdown time, does represent a loss of safety: 1) all rods are fully out of the reactor; 2) an amount of excess reactivity somewhat in excess of prompt critical is added, 3) this reactivity is somehow present before the fuel is significantly heated and before the rods are fully inserted. It can be seen that failure to lose 0.2% reactivity (or perhaps even for some reason a gain of 0.1%) during the first 7.25 in. of an insertion will enhance the gravity of the event.

Notwithstanding these theoretical estimates, when the new control rods are initially calibrated in the reactor,

- (a) the reactivity per inch downward insertion is always a loss;
- (b) the reactivity per inch downward insertion can be a gain for the first few inches, and the integrated gain is $<0.1\%$; or
- (c) the reactivity per inch downward insertion can be a gain for the first few inches, and the integrated gain is $>0.1\%$.

The course of action to be followed is, respectively:

- (a) operate without restriction on rod travel;
- (b) limit the upward travel of the rods by an operating rule to the point where $dk/dx \cong 0$; and
- (c) limit the upward travel of the rods by insertion of a mechanical stop at the point where $dk/kx = 0$.

Design calculations are not sufficiently accurate to predict the proper location of a mechanical stop if case (c) should exist; therefore experiment must be relied on. During burnout, the flux shifts upward and some change in the

point at which $dk/dx = 0$ can be expected. Whether or not any significant change has occurred in the previous calibration will become apparent when the rods are moved in that region.

The insertion time for the new control rods will be almost three times as long as previously experienced. This represents some loss of safety. However, the loss is unimportant, since, even with the shorter insertion time, the rods offered no real shutdown assistance during accidents giving rise to periods in the region of milliseconds. For this reason the hazard analysis of EBWR has always ignored negative reactivities from control rod insertions during an accident giving rise to a short period.

Nine control crosses are adequate for control of the reactor when loaded with sufficient reactivity to accommodate the burn-ups anticipated. Two new features of the reactivity control system that make this possible are:

- (a) the use of fuel in the control rod followers: this increases the worth of the rods by a factor of ~ 1.1 ;
- (b) the use of burnable poison strips: this reduces the net amount of reactivity necessary to control with the rods.

In all probability, the reactivity worth of the control rods will be determined during initial loading experiments. The rods will be run as a bank, all at nearly the same level in the 100-Mw core. This simplifies both calibration and analysis and is, in other respects, essentially as advantageous as any other scheme. Determination of the rod worth as a bank will be based on:

- (a) experimental measurements of the rod worth in the previous core loading;⁽³⁾
- (b) calculations supporting the values obtained by experiments (a); and
- (c) calculations of the rod worth in the proposed core, with appropriate correction factors evolved from (a) and (b).

The reactivity worth of the individual rod is not considered essential. For all practical purposes, the average worth is $\frac{1}{9}$ of the total, and the maximum differential worth has been determined during oscillator experiments.⁽¹⁾

The initial loading experiments will determine the exact number and location of the spikes and the burnable poison strips. Figure 12 shows the estimated spiked core loading arrangement; it is essentially loading No. 51 (Fig. 10) with a ring of spikes sandwiched in.

The upper two feet of the spikes do not undergo any significant burnup during the first half of the spike lifetime, since (a) the surrounding fuel is only 4 ft high; (b) the control crosses reduce the flux; (c) the steam voids reduce the flux. Therefore, about midway in the life of the spike, it will be turned upside down.

4. Heat Transfer and Fluid Flow

(a) Power Distribution

Figure 13 shows the calculated power distribution for 100-Mw operation based on the core loading shown in Fig. 12. The radial maximum to average of 1.50 is considerably less than that reported for the present EBWR core. The flattening is caused by the introduction of the 5-ft spike elements.

(b) Recirculation Flow Rates

The proposed riser and shrouding shown in Fig. 1 should more than double the core recirculation flow rates obtained to date provided the steam carryunder problem can be resolved. The calculated recirculation flow rate is 9.5×10^6 lb/hr as compared to the measured value of 4×10^6 lb/hr with the present core operating at 60 Mw. The radial inlet core velocity profile is shown in Fig. 14. The average inlet velocity to the core is approximately 6.3 ft/sec. The high recirculation flow rates are due primarily to the action of the upper portion of the riser on the outer peripheral fuel assemblies. The velocity, instead of dropping off at the core periphery, remains fairly constant even though the power density decreases. This stems from the fact that the outer fuel assemblies have lower steam volume fractions, and hence less frictional resistance, but still share the high driving head of the common riser. The riser will have a higher steam volume fraction than many of the outer fuel assemblies.

FIG. 14
INLET CORE VELOCITY VS. CORE RADIUS

The velocity dip in the spike zone (see Fig. 14) is due to several effects: (1) The equivalent diameter of the spike fuel assembly is 0.5 in. as compared to 0.680 in., the mean value of the thin and thick fuel assemblies; and (2) the extra foot of length of the spike fuel assembly adds additional frictional flow resistance and also decreases the effective riser height above it.

(c) Stability Considerations

Laboratory tests on a natural circulation system at 600 psig with a geometry similar to the flow path in EBWR showed that an instability threshold existed at $\alpha_e \cong 0.75$. The exit steam volume fraction in the $Q_{\max}$ channel of the present core when operating at 61 Mw was estimated to be 0.70. It has been predicted that the present (114-assembly) reactor core would be unstable at 65 Mw. The exit steam volume fraction in the spike zone fuel assemblies at 100-Mw operation will be $\cong 0.60$ (see Fig. 16). Thus, although it appears that the spike region fuel assemblies are well below the probable threshold for instability, an important factor that must be considered is the additional resistance of the extra foot of length of the spike fuel assembly.

The distribution of resistance in the flow circuit is known to affect the system stability. In general, it is believed that restricting the outlet of the boiling section in a natural circulation system tends to make the system more unstable. The effect of the tall riser on the stability problem is also unknown, but it may be quite beneficial. Therefore, in view of these uncertainties it appears that the spike-region fuel assemblies may reach the threshold of hydrodynamic instability.

Another factor that is being considered is the problem of parallel channel operation. This problem is normally associated with a forced circulation system since, in a sense, the necked-down portion of the riser actually behaves as a pump. The reason for this is that the net driving head in the individual fuel assemblies is a substantial percentage of the total head available. As a result, the net driving head for each fuel assembly is automatically varied by any occurrence which may tend to increase the steam volume fraction within the assembly, such as a local power surge, flow restriction, etc., that is, each fuel assembly still possesses some degree of self-regulation. The percentage of total net driving head that exists within the fuel assemblies as a function of radius is shown in Fig. 15 for 100-Mw operation. Another beneficial aid is the division of the lower portion of the riser into 16 sectors by the control rod drive superstructure. This also tends to provide some local self-regulation.

FIG. 15
NET DRIVING HEAD VS. CORE RADIUS

The possibility that the water level may be below the top of a riser or shroud has existed in the past and can also exist after shutdown from operation at 100 Mw. Experiment⁽³⁾ has shown that it is possible to operate at a few megawatts (initial startup) with conditions

that might be referred to as hydraulic instability. However, there is no conclusive evidence that low water level in the riser is a source of instability.

During shutdown cooling, the nuclear component of the feedback required to promote instability is not present. Therefore, variations in the core steam void fraction (as a consequence of shutdown decay heat generation) may change the reactivity of the system, but the reactor cannot attain criticality because the control rods are inserted.

There is no danger of core meltdown due to decay heat, provided the core is submerged. The riser section need not be filled with water. The feedwater flow rate can be regulated from the control room.

(d) Core Steam Volume Fraction

The variation of the radial and axial steam volume fraction is shown in Figs. 16 and 17.

FIG. 16
RADIAL VOID PROFILE

The peculiar radial void distribution (Fig. 16) results from the rather unique velocity profile obtained with this type riser and from the normal radial power distribution. As expected, the peak voids occur in the spike zone. Integrating and weighing the radial and axial void profiles yields a value of 0.189 for the mean core steam volume fraction at 100-Mw operation.

(e) Vapor Separation - Steam Carryunder and Liquid Carryover

The probability of achieving 100-Mw operation will depend,

FIG. 17
AXIAL STEAM VOID PROFILES AT VARIOUS CORE RADII

(e) Vapor Separation - Steam Carryunder and Liquid Carryover

The probability of achieving 100-Mw operation will depend, to the largest degree, on solving or circumventing the problems of steam carryunder and liquid carryover. The magnitude of the steam carryunder problem will be governed by the amount of steam entrained and the rapidity with which the entrained steam bubbles are quenched by the feedwater. It is conceivable that the entrained steam bubbles are not collapsed immediately because of insufficient mixing in the downcomer; consequently, the steam would be carried down a considerable distance before condensing. Should this occur, the performance would be limited since the net driving head would be reduced. A reduction in net driving head would, in turn, reduce the recirculation flow rate, increase the core steam volume fraction and, in general, reduce the probability of achieving 100 Mw.

The problem of water carryover by the steam is acute because of the reduction in the volume of the steam dome with the use of the taller riser and the sharp increase of the superficial steam velocity (based on reactor vessel diameter) at 100 Mw, which is 1.89 ft/sec. The bubble layer above the riser must be kept at a minimum to ensure the maximum possible steam dome. This is accomplished by starting reactor operation with the hot saturated water level just at the top of the riser. Under such conditions the estimated height of the bubble layer at 100-Mw operation would be 2.5 ft above the top of the riser. This would leave another 2.5 ft of steam dome for primary separation.

The degree of primary steam separation that will take place within the reactor vessel is unknown and can only be estimated. Utilizing the information on natural vapor separation from the Russian literature,⁽¹⁰⁾ the liquid carryover could range from 0.10 to 3% by weight, depending on the location of the interface. This correlation is based on the cu ft of steam discharged per hr per cu ft of steam dome. Although it is doubtful that the correlation would hold over extreme geometry ranges, it may be applicable to the present EBWR geometry. The superficial steam velocity is known to affect primary vapor separation through its effect on the steam disengaging height (length) at the interface. The steam disengaging length is defined as the length of the travel upward required for transition from the steam-water mixture to essentially pure steam. The disengaging height increases with increased superficial steam velocity and, in turn, with increased reactor power. This has been shown by measurements in EBWR up to 30 Mw. Extrapolation of these results to 100 Mw shows the steam disengaging height to be approximately 22 in., which is approaching the height of the steam dome.

(f) Operating Characteristics

Table VI summarizes the data relative to the operation of EBWR at 100 Mw. All void and flow data are based on calculation methods developed from laboratory experiments.

(g) Burnout Considerations

In general, burnout in a system such as the EBWR can occur as a result of two distinct effects. Burnout can occur in heated sections with or without two-phase flow if heat fluxes are high enough to produce vapor layers over a sufficiently large area of the heated surface. Recent experiments indicate that this type of burnout would require heat fluxes in excess of 1,000,000 Btu/(hr)(ft²) at velocities and qualities in the range of interest to the EBWR. Burnout can also occur in heated sections at lower heat fluxes if there is insufficient liquid to maintain reasonable heat transfer coefficients from surface to bulk liquid. Other previous work has usually produced the result that the "dry pipe" type of burnout occurs at steam qualities near 80%.

C. Power Plant

1. Steam System

(a) General

The major components necessary to handle the excess steam developed under increased power output of the reactor are shown in the schematic flow diagram, Fig. 18.

TABLE VI

Calculated Heat Transfer Data (at 100 Mw)

Design Power of Reactor	100 Mw
Operating pressure	600 psig
Operating temperature	489°F
Steam flow	300,000 lb/hr
Average steam voids in reactor core	18.9%
Average steam voids in total moderator	16.5%
Maximum to average radial flux	1.5
Maximum to average axial flux	1.5
Equivalent diameter of coolant channel in spiked element	0.5 in.
Equivalent diameter of coolant channel in unspiked element	0.68 in.
Fuel heat transfer area in spiked zone	540 ft ²
Fuel heat transfer area in unspiked zone	1940 ft ²
Total fuel heat transfer area	2480 ft ²
Average power density in spiked zone	135 kw/liter
Average heat flux in spiked zone	136,000 Btu/(hr)(ft ²)
Average power density in unspiked zone	115 kw/liter
Average heat flux in unspiked zone	157,000 Btu/(hr)(ft ²)
Maximum heat flux in reactor (based on max/avg heat flux ratio of 3.1)	485,000 Btu/(hr)(ft ²)
Average fuel center line temperature	
Thick plate (enriched)	579°F
Thin plate (enriched)	567°F
Rod (spike)	558°F
Maximum fuel center line temperature	
Thick plate (enriched)	767°F
Thin plate (enriched)	724°F
Rod (spike)	694°F
Maximum surface temperature	514°F
Average surface temperature	507°F
Average recirculation rate	32 lb H ₂ O/lb steam
Average exit steam quality	3.2%
Inlet subcooling	12°F
Average core inlet velocity	6.3 ft/sec

EBWR-100MWT FLOW DIAGRAM

FIG. 18
100-Mw FLOW DIAGRAM

(b) Turbine-Generator

The turbogenerator unit will remain intact (with minor modifications such as piping, controls, and relief valves) and will be used to generate up to 5,000 kw of electricity. There is no change in this section and the information given in the previous hazard summary (ANL-5781) is not affected.

(c) Main Steam System

The steam leaving the reactor will go directly to the turbine and reboilers. Steam will also pass through the steam dryer-emergency cooler tank (where any excess moisture is stripped out) to the turbine and pressure-regulating valve. Mounted to a parallel line are a back-pressure-regulating valve and two safety valves which discharge through a desuperheater to the condenser.

The steam dryer contains cooling coils which can dissipate core shutdown heat even when the main steam stop valves are closed and power is lost to the condenser circulating water pumps. Cooling is accomplished by water flowing down from an overhead water tank mounted on the underside of the dome of the steel building. Reactor steam is condensed on the outside of the tubes and the condensate flows by gravity back to the reactor. Steam is generated within the tubes, and the lower density of the steam-water mixture sets up a natural circulation flow from the overhead water tank. Operation of the emergency cooler is initiated manually from the control room. This cooling source is turned on automatically by loss of instrument air or loss of 125-volt, dc power.

Although the main condenser is designed to handle 20 Mw of heat output from the reactor at a $2\frac{1}{2}$ in. Hg abs, a much higher heat load can be accommodated by allowing condenser pressure to rise. A rupture disk in the condenser wall will open at 15-20 psig.

The 6-in. steam collecting ring will be removed from the vessel. A steam duct will be installed in the uppermost section of the vessel to eliminate "slug" carryover. A larger steam line will be installed leading to the steam dryer.

The steam system will operate in essentially the same manner as originally designed, except that additional equipment will be operating in parallel with the original system to accept the increased power.

Additional relief valves to handle 100-Mw operation will discharge into the condenser.

(d) Steam Bypass System

The steam bypass system originally installed and operated in EBWR will be used primarily to bypass all the steam the turbine was utilizing in the event the generator is accidentally tripped off the line. No steam will be bypassed through this valve during normal operation of the turbine-generator.

Reactor pressure will be maintained by the reboiler pressure-regulating valve. This valve will also handle any load variation demanded by the turbogenerator up to its full rating. When the electric load decreases and the turbine inlet valves close proportionately, the reactor pressure will tend to increase and the reboiler bypass pressure-regulating valve will open to limit the pressure rise in the reactor. The reverse sequence will occur when the electrical load increases. The EBWR electrical output is such a small portion of the capacity of the utility system to which it is connected that changes in load demand should be minor except in the very rare case of a power line failure.

(e) Overpressure Relief System

To supplement primary steam pressure system safety, one back-pressure relief valve and conventional pop safety valves are installed to accommodate the capacity of at least 100 Mw of steam at 650 psig. These valves are mounted adjacent to and discharge to the main condenser; their inlet nozzles are connected to the reactor piping by an unobstructed header. The back-pressure-relief valve is of the steam-powered type with an adjustable pressure range. It is set to open at 650 psig; at 675 psig, the valve would be fully open and could pass approximately 200,000 lb of steam per hour. The safety valves are standard ASME code approved, set to open at 700, 725, 750, and 775 psig, and having blowdowns of about 30 psig.

The intermediate system will also have safety valves set at 375 psig and 400 psig, which will dump into the condenser. These valves will have a capacity of about 80 Mw of steam at these pressures.

Reactor safety rods are inserted automatically when any one of the aforementioned valves open. The cause of the excess pressure would be determined and corrected before operation was resumed.

Since the relief and safety valves discharge to the condenser, the condenser is designed to accommodate this influx of relatively high-pressure and temperature steam. In addition, the condenser is equipped with a 15-20 psig rupture disk to protect the equipment in the event circulating water is stopped or if the reactor is not shut down by one or more of the shutdown signals mentioned above.

Rupture of the disk in the condenser will discharge the steam directly to the building. The radioactive steam would be detected by a monitor, which, in turn, would trip close the inlet and outlet ventilating valves of the plant. This trip also shuts down the reactor if a previous signal has not done so.

(f) Condenser

The condenser system will remain essentially the same. A slight additional load of about 10,000 lb of steam per hour will be discharged from the primary deaerator. Additional capacity relief valves will be installed to discharge into the condenser.

The air ejectors will handle the increased load of non-condensable gases.

(g) Feedwater System

In order to handle the increased reactor power output, two additional feedwater pumps will be installed, one on a standby basis. Each of these pumps will have a capacity of 600 gpm. These pumps will be in parallel with the original pumps.

The feedwater regulating valve on the original pump will be controlled by the condenser hotwell level and turbine steam flow, and will have an output of about 120 gpm at 120°F. The new pump feedwater regulating valve will utilize conventional 3-element flow control with signals from the reactor level, reboiler steam flow, and its feedwater flow. Each pump will normally pump 480 gpm at 120°F.

The feedwater will pass through disposable cotton-fiber filter cartridges housed in 4 parallel filter vessels. When the pressure drop through one of these units becomes excessive, the unit will be valved off and the cartridges replaced.

(h) Air-drying and Fluid-recovery System

No change is anticipated for this system.

(i) Circulating Water System

The only change in this system involves arrangements for diverting about 1000 gpm of circulating water (out of a 13,000-gpm pumping capacity) through the primary subcooler. This flow will be used to subcool the feedwater to about 120°F for experimental operation. It will raise the circulating water temperature 3-5°F before the water returns to the cooling tower.

Since the primary fluid will be on one side of the cooler, a leak could transfer some radioactive fluid to the circulating water. A radiation monitor will be used to detect a leak.

(j) Feedwater Makeup System

This system will remain unchanged. The capacity is sufficient to handle the added load.

(k) Retention Tanks

The two 3,000-gal storage tanks located in the basement of the power plant area have sufficient capacity to handle the increased load and will not be altered.

2. Electrical System

The only change in this section concerns paragraph (a). This change involves the addition of transformers, switchgear and breakers to handle the additional equipment necessary to operate the plant at a power level of at least 100 Mw.

D. Building

1. General

In addition to the 20-Mw design facilities, the vapor containment plant will house the following major equipment:

- 1 Flash Deaerator
- 1 Subcooler
- 2 Feedwater Pumps
- 2 Full-flow Feedwater Filters.

The related controls and instrumentation will be mounted on an additional panel installed in the control room.

A new reboiler building will be built near the vapor-containment building. This building will house (1) the primary reboilers, drain tanks, drain coolers; (2) the intermediate steam system; and (3) the major electrical units. The intermediate system will consist of the following major equipment:

- Secondary reboilers
- Secondary drain coolers
- Intermediate drain tank
- Intermediate flash tank

Intermediate feedwater pumps
Drain tank
Blow-off tank
Flash condenser condensate return unit.

The electrical units will include:

4160-volt switchgear
440-volt unit substation
440-volt motor control center
120/208-volt distribution transformer.

Air-cooled condensers will be installed adjacent to the re-boiler building to dissipate the excess heat when the plant steam requirements are low (during summer months). This arrangement will enable continued full-power operation independent of steam load demands.

2. Missile Protection

In ANL-5781 the design of the upper section of the reactor vessel was incorrectly described as being 50% stronger than the rest of the vessel. The design pressure of the vessel is 800 psig. A 3-in. armor plate ring, rather than a 2-in. plate described, is used for a shield and to retain removable shielding.

3. Spray System

A 15,000-gal water-storage tank is suspended from the dome of the steel shell. A building spray system, consisting of spray rings located above the main floor and in the basement, is supplied from this tank at the initial rate of 1,000 gpm until the reserve has been used up. Subsequent supply is from the Laboratory water system at the rate of 100 gpm. In the event of a rupture of the pressure system, releasing steam into the building, the spray system is turned on manually, either from the control room or from the Power Plant area.

The water in the reactor system (primary) will amount to about 41,000 lb. If the water in the reactor and additional primary system equipment were flashed to steam from 600 psig into the building at an ambient temperature of 90°F, the resultant temperature and pressure would be approximately 205°F and 32 psia.⁽⁵⁾ In calculating this temperature and pressure, no consideration was given to condensation, which can be expected to take place on the cold building and equipment surface. Thus, a pressure and temperature somewhat less than the calculated figures can be expected.

In the event of a vessel rupture accompanied by a failure of one or more fuel plates, the washing effect of the sprays would be helpful in reducing the radioactive contamination of the atmosphere in the building. Thus, radioactivity in any ensuing leakage from the gastight building would consist mainly of gaseous fission products.

4. Air Lock

This section remains as reported in ANL-5781.

5. Connections through Gastight Enclosure

All electrical cables and pipes leading from the Power Plant Building are sealed to maintain the internal building pressure of 15 psig. There are five penetrations through the shell to accommodate the following primary system piping to and from the adjacent Reboiler Building:

- (1) One 8-in. main steam line to the reboilers.
- (2) One 1-in. vent line from the drain tanks and reboilers to the existing condenser.
- (3) Two 4-in. condensate lines from the drain coolers to the deaerator tank.
- (4) One $\frac{3}{4}$ -in. control air line.

Two air-operated isolation valves are installed in each of the pipe lines in Items (1) and (3). One back-pressure-regulating valve is installed in the single vent line [Item (2)]. The control air line features a solenoid valve inside the containment shell, which will close on plant shutdown.

Building ventilation is limited to a 3,000-cfm fresh air and exhaust system in order to use small ducts and dampers. Exhaust air leaves the plant near the top of the steel shell through a duct designed for a 2-sec holdup time. Weight-operated, quick-closing solenoid-controlled butterfly valves are located in the inlet and outlet ducts. These valves close upon detection of activity in the plant or upon power failure. Closing time is less than one second.

Heat loss of the power equipment into the air in the building is removed by a recirculating building air system consisting of four unit air conditioners, two on the ground floor and two in the basement. The plant circulating water system provides coolant for the air-conditioning condensers. Building heating, when required, is accomplished with steam coils in these same units. Steam will be supplied from the Laboratory steam plant.

6. Reboiler Building

The entire primary system outside the containment shell is housed within the relatively gastight Reboiler Building. The building atmosphere is monitored and vented through a stack in the same manner as is the existing EBWR stack. Should the activity in the Reboiler Building stack rise above normal, the monitor trip circuit will:

- (1) shut down the reactor, and simultaneously close one of the air-operated valves in the main steam line from the reactor;
- (2) close the air-operated valves (close upon removal of air pressure) in the primary condensate lines;
- (3) the back-pressure-regulating valve in the vent line from the drain tanks and reboilers normally will be closed; a pressure greater than 15 psi will open the valve to allow vent gases to flow to the main condenser in the containment shell.

At present, a trip of the activity monitor in the existing EBWR vent stack closes the inlet and exhaust air ducts in the containment shell, and shuts the reactor down. The system has been modified to effect closure of the air-operated valve in the main steam line to the Reboiler Building.

All of the air-operated valves in the main steam line and the condensate lines interconnecting the two buildings are backed up by motorized valves which, in the event of normal power failure, can be powered by the emergency power supply. In summation,

- (1) If activity is detected in the Reboiler Building, one air-operated valve in all lines, except the vent line, is closed and the reactor is shut down. The valve in the vent line remains closed whenever the pressure in the Reboiler Building equipment is less than 15 psig.
- (2) If the activity in the containment shell is above normal, and the activity in the Reboiler Building is normal, reactor is shut down and the steam line to the Reboiler Building is closed.
- (3) The operator has control of all valves at all times and can either close or open them as necessary. The automatically set conditions simply provide a starting point.

7. Facilities for Viewing Power Plant Area

The use of a thick concrete liner for the gastight shell precludes the use of viewing windows in the wall of the building. To provide for viewing the water column sight gauge from the Control Room, a television monitor located in the Control Room will be used in conjunction with a camera located in the Power Plant area. This has been the previous practice and has proved satisfactory.

IV. OPERATING PROCEDURE

A. Preliminary

After the 5-ft control rods and drives are installed, the units will be tested for smoothness of operation and to ensure a rapid insertion time of about 1.5 sec. All the newly installed instruments, interlocks and alarms will be given a pre-operational test. The high- and low-pressure boron injection systems will be on standby for use during loading, if the need arises.

- (1) The existing strong antimony-beryllium source ($\sim 10^{10}$ neutrons/sec) will be used for loading and startup testing. The source will be placed in a stainless steel dummy element and will be positioned adjacent to the fuel during loading. Source strength will be maintained by subsequent irradiation in EBWR.
- (2) BF_3 counters will be used to check the counting rates during the loading. These counters will register neutron counts in the control room.
- (3) An ion chamber will be installed in the pressure vessel and will feed a signal to a sensitive electrometer or micro-microammeter to indicate multiplication or approach to critical.

B. Initial Loading of Fuel

Since new control rods will be installed for 100-Mw operation, the fuel loading will conform with the procedure outlined in the previous Hazard Summary Report (ANL-5781).

C. Preparing for Operation

Effectively, the same procedure originally used to calibrate rods and to determine the loading for power operation will be followed as was described in ANL-5781. The inner-shield plug is not required with H_2O operation; therefore, the plug will not be placed in the reactor vessel.

D. Normal Startup

Several modes of normal startup operation may be employed, each differing primarily in the sequence of warming up the plant components. However, irrespective of the sequence employed, the first step will be to test and to clear the Shutdown Annunciator Board on the operating console in the control room. One mode of operation is as follows:

- (1) Set the pressure-regulating valve in the primary steam line to the reboiler at a pressure greater than 600 psig, say 610 psig.
- (2) Close intermediate steam line valves downstream of primary reboilers.
- (3) Start present turbogenerator portion of plant, per past standard operating procedure, to attain a thermal power level of between 20-40 Mw. The turbogenerator will absorb steam to produce 5000 kw of electricity; any excess steam will be bypassed to the desuperheater of the main condenser. This steam can later be diverted to the reboiler cycle.
- (4) Start intermediate feedwater pumps; all tubes in primary reboilers are to be completely submerged (normal operating condition).
- (5) Open primary system to reboilers for initial warmup by gradually opening bypass valve.
- (6) Start large reactor feedwater pump. A minimum flow through the bypass across the pump will be used for dissipating heat generated in pumping water until sufficient feedwater begins to flow to the reactor.
- (7) Manually switch turbine-condenser feedwater control from reactor level control to hotwell level control. Experience has shown that this indirect method of control is satisfactory.
- (8) Manually switch the new feedwater control from manual to automatic reactor level control.
- (9) Gradually load the reboilers with primary steam by reducing the setting of the primary system pressure-regulating valve to about 600 psig. [The bypass valve will now be closed manually if it was opened for warming up the reboilers as outlined in step (5)]
- (10) As intermediate steam pressure builds up to 350 psig, the manual valve downstream of the reboilers will be opened. The pressure controller just upstream of the air-cooled condenser will be set to maintain intermediate steam pressure at 350 psig.
- (11) After venting the intermediate flash tank, the air-cooled flash condenser fan and pump will be started.
- (12) The pressure-regulating valve in the intermediate steam line upstream of the secondary reboiler will be used to control the amount of site steam supplied by the reactor.

- (13) Reactor power will be increased gradually until the desired operating level is reached. At the same time, the pressure-regulating valves to the turbogenerator system and the reboiler system will be adjusted so that no steam bypasses the turbine, and 600 psig is maintained in the reactor by the reboiler cycle, respectively.
- (14) The air-cooled condenser, drain cooler and subcooler will be adjusted as required by the load demands.

E. Normal Operation

Normal operation of EBWR should be quite simple. The electrical or turbogenerator portion will operate as previously and can be an automatic or manual operation.

Since steam will not be dumped to the condenser by way of the bypass valve, load variations will be taken up by rejecting steam to the reboiler circuit for a decreasing electrical demand, or by diverting steam from the reboiler circuit for an increasing electrical demand.

Similarly, an automatic or manual adjustment of the pressure-regulating valve upstream of the secondary reboiler will maintain the required steam flow to the ANL system. The function of the pressure control valve upstream of the air-cooled condenser is to maintain 350 psig pressure in the intermediate steam system. If the pressure of the ANL steam increases (above 200 psig) the regulating valve will close proportionately and transfer less heat to the ANL system. Since the intermediate steam pressure will tend to increase, the pressure-control valve will open to maintain 350 psig pressure in the intermediate steam line.

The reactor pressure will be maintained at 600 psig by the pressure-regulating control system.

Emergency operating conditions are discussed in Section VI.

F. Normal Shutdown

- (1) The power house will be notified of a forthcoming shutdown so that their boilers can take over the production of steam required by the Laboratory.
- (2) The secondary reboiler will be unloaded to the air-cooled condensers at the same rate as steam is generated at the power house.

- (3) The reactor power will be reduced by gradually inserting the control rods into the core. Since the turbine bypass-regulating valve is set at a pressure just above 600 psig, it will not be affected and the pressure-regulating valve in the primary steam line upstream of the reboilers will gradually close to maintain reactor pressure at 600 psig.
- (4) Insertion of the rods will continue until the reboiler pressure-regulating valve carries about 10% of rated flow to the reboilers.
- (5) The setting on the turbine bypass pressure-regulating valve will be lowered to 600 psig, or slightly below, to assure that this valve takes over control of pressure from the reboiler regulating valve. When this occurs, the setting on the reboiler pressure-regulating valve will be lowered until this valve closes and transfers the excess steam flow through the turbine bypass valve.
- (6) The turbogenerator portion of the plant will be shut down as per existing procedures.
- (7) All air-cooled condensers and coolers will be drained whenever freezing conditions exist or are imminent.
- (8) Feedwater pumps will be shut down.

V. INHERENT SAFETY OF EBWR AGAINST REACTIVITY ADDITIONS

The nuclear characteristics of the revised core are substantially the same as reported in ANL-5781. Consequently, the inherent safety analysis (including the experiments of Appendix E) described in ANL-5781 essentially remains valid. It is necessary, however, to "renormalize" the safety features to prompt excess reactivities at irradiation times greater than zero Mwd.

Accordingly, the inherent self-protection of the reactor against rapid room-temperature additions of excess reactivities of up to 1.1% initially reported (ANL-5781), actually corresponds to 0.34% (1.1% minus the initial value of the dollar) prompt excess reactivity. This number remains valid even though the dollar is changing. Therefore, during reactor life the total excess reactivity that can be added suddenly and safely is 0.34% plus the deflated dollar value at that time (which is given in Section III-B-3). For example, at 7000 Mwd where the dollar was measured at 0.87 times its original value, the total reactivity readily compensated safely is $0.87 \times 0.755 + 0.34 = 1.0\%$ (based on Keepin data).

Since the publication of ANL-5781, theoretical and experimental studies have contributed significantly to the understanding of the instability phenomena. Transfer function measurements at 20 Mw^(1,7) were used to predict stable operation at 40 Mw. Also, an a priori theory was found to agree within 1 decibel with transfer function experiments at 20 Mw, and within 6 decibels with similar experiments at 50 Mw.^(1,2) Calculations of the latter type for the 100-Mw core are described in Appendix H.

The foregoing studies have also revealed hitherto unknown complexities of the problem of stability. A boiling reactor could have an unstable point defined as that power at which linear noise-free analysis of transfer functions would lead to an infinite transfer function. Yet there is evidence⁽⁷⁾ which indicates that in the presence of nonlinearities and boiling noise, safe operation at powers beyond this point has been possible.

In discussing possible spontaneous oscillations, it is convenient to deal with a damping factor magnitude, ξ . As the reactor approaches the oscillating or unstable state, the flux trace shows small spontaneous oscillations which disappear in a matter of one or two seconds. The flux trace under these conditions can be fitted to the formula

$$\text{Power} = \exp(-\xi \omega t) \sin \omega t.$$

In BORAX-II at 300 psig and 4.2% void reactivity, ξ was evaluated as -0.068. This was the most negative value ever encountered.⁽¹⁾ In EBWR at 62 Mw,⁽⁶⁾ 600 psig, and 4% void reactivity, ξ was evaluated as -0.04. A quantitative understanding of how to predict ξ is not available. It is also

not possible to say how far negative ζ may go in the practical case before the divergent oscillations do not disappear. Therefore, in the case of EBWR at 100 Mw it is not possible to predict what value of ζ will be encountered or what value will render operation impossible.

This approach supplements the transfer function analysis given in Appendix H. In this linear analysis, stability is shown up to 100 Mw of thermal power.

Because of the sensitivity of the transfer function resonance to reactor parameters, it is not possible to make accurate stability calculations until the detailed core design has been set and its characteristics have been calculated. Nevertheless, it is possible to indicate at this time the bearing of certain factors on stability.

- (a) The EBWR transfer function at 50 Mw early in core life had a resonance of $G = \frac{\Delta N}{N} / k_{in} = 55$ decibels at $\omega = 10.1$ radians per second. Its width ~ 3 decibels down was $\Delta\omega = 0.2$ radian per second. At 100 Mw, the power density would be $2 \times 114 / 147 = 1.55$ times larger, and the value of the dollar will be somewhat smaller. These effects should raise and sharpen the resonance. BORAX-IV has operated with a 60-db resonance.⁽⁷⁾
- (b) The reactivity in steam voids should be comparable to that observed at 60-Mw operation of EBWR (about 4%), although this quantity is quite sensitive to core life and control rods. BORAX IV has operated satisfactorily with 6.9% reactivity in steam voids at a lower pressure, 322 psig.⁽⁷⁾

As in raising the power to 60 Mw, further power increases will be prefaced by transfer function determinations in order to assure a safe procedure of entering into a potentially unstable region.

VI. ENUMERATION AND EVALUATION OF POSSIBLE HAZARDS

A "Hazard Summary Report on the Experimental Boiling Water Reactor (EBWR)" was presented in ANL-5781 and was based on operating EBWR at 20 Mw(t). The same items have been re-evaluated, based on operating EBWR at 100 Mw.

A. Shutdown Cooling - Loss of Water

The removal of the decay heat from the fuel elements of the EBWR following a regular shutdown is by boiling of the coolant. The heat contained in the steam generated is, in turn, rejected to the condenser.

On shutdown of EBWR from the 20-Mw operating power level, the heat removed (excluding heat losses) during the first 10-min period does not exceed the heat-removal capacity (100 kw) of the Reactor Water Purification System. Only the secondary cooler of the Reactor Water Purification System is used. Tests run on this cooler indicate that it is capable of removing approximately 500 kw. There is one standby purification system whereby its secondary cooler can also be utilized. If both coolers do not provide sufficient shutdown cooling, the pressure-relief valve will open at 650 psig and dump steam into the main condenser.

Under abnormal conditions which result in loss of water from the reactor vessel, some cooling of the fuel is provided, depending upon the conditions which exist.

- (1) A major rupture of the reactor vessel that results essentially in the confinement of air within a waterless reactor vessel will cause fuel elements to melt (Appendix A). Some auxiliary cooling is provided in the event of such an accident: water from the 15,000-gal overhead storage tank can be admitted to the vessel through the Boric Acid Spray Ring, or Laboratory Water can be admitted through this ring via the Boric Acid Crystal Tank. This is done automatically when both the water level and pressure in the reactor drop below the set limits. It can also be done manually from the control room.

However, with approximately five times as much decay heat encountered under the new operating conditions, it is doubtful that this spray of about 10 gpm would be of much use if such a major rupture occurred. Furthermore, the new riser will prevent efficient spraying over the core to prevent melting as claimed originally. If the spray ring were not damaged, water would be admitted to the vessel, flash and provide some cooling by steam. Consequently, the spray ring will not be modified or removed and will be used to add water to the downcomer area.

- (2) A major rupture of the vessel that results in the loss of water from the vessel and is of such nature that building air can flow through the core will probably result in the fuel plates reaching a temperature at which the fuel elements would melt. The zirconium will oxidize, exposing the hot uranium, which will probably burn. Fission products will be released to the atmosphere. Some cooling may be afforded by the spray system described in the preceding paragraph. Nevertheless, there is no assurance that this high temperature will not be reached. The building must contain the fission products which are released when fuel melts.
- (3) A minor rupture of the vessel could result in the eventual loss of all water from the vessel. If the rupture were above the top of the core. The flashing operation would prevent overheating of the fuel elements until the pressure had dropped to a point where water would be injected automatically. The core could be kept flooded by water from the Laboratory main. If the rupture occurred below the core, the operator would reduce pressure in the reactor as rapidly as possible by opening either or both bypass valves and dumping the steam to the condenser or reboilers. This would automatically permit low-pressure injection of water from the Laboratory water main or from the overhead tank through the Boric Acid Spray Ring.

B. Power Failure

Following loss of power to the magnetic clutch, the control rods would drop to essentially their maximum shutdown position in less than 1.5 sec (fail-safe operation). The longer insertion time is partially due to the 25% increase in length. The use of a new enclosed rack-and-pinion drive has removed the high initial (5 g) acceleration feature of the original drives.

If the control rods do not drop to a point where the reactor will become subcritical, the operator may initiate the High-Pressure Boric Acid Solution injection from the control board or do so manually with a release lever in the shell wall as a backup shutdown method. Water from the overhead storage tank can be admitted to the Emergency Cooler upon signal from the control room. Flow of water to the Emergency Cooler is automatic on loss of dc power or instrument air.

In case of loss of power from both incoming utility lines, the reactor is shut down and the turbine-generator is tripped off. The pressure-regulating valve in the turbine steam line bypasses steam to the main condenser, preventing reactor pressure from rising above 600 psig. Continued operation of the pressure-regulating system is assured by an oil accumulator

tank. The pressure-regulating valve for the primary reboilers will close automatically upon a power failure. The circulating water pumps are without power, so the flow of cooling water to the condenser is by natural convection and is at a low rate. The natural circulation is effected by the automatic opening of a valve which bypasses water from the condenser outlet directly to the storage basin of the cooling tower. The height of the cooling tower risers would prevent circulation.

The ventilation openings in the Power Plant Area building shell are closed automatically on power failure. Automatic valves are instrumented to fail safe on loss of power.

C. Pump Failure

The most probable reason for failure of pumps is loss of power.

1. Circulating Water Pumps

Failure of one of these pumps would lead to a slight increase in back pressure in the condenser with a correspondingly higher temperature of water in the hotwell. Introduction of hotter water into the reactor by the turbogenerator feedwater pump would result in more steam per unit of power and cause some drop in power.

2. Feedwater Pumps in Turbine-Generator System

If the feedwater pump should fail without loss of power, the standby pump would be started. In the event both pumps fail to operate, the feedwater pump in the primary reboiler system would tend to increase its output until cut off by a low-water level interlock in the primary flash de-aerator tank. This event would shut down the reactor. An alarm would also indicate a high water level in the hotwell.

A power failure to the feedwater pumps would alert the operator. He would shut down the reactor.

3. Feedwater Pumps in the Primary Reboiler System

If the feedwater flow from the primary reboiler portion of the cycle should fail because of a power interruption, the reactor would be shut down automatically. If the pump failed for other reasons, the water level in the reactor would drop and the low-water level interlock would shut the reactor down.

4. Intermediate Feedwater Pumps

A failure of an intermediate pump will cause the pressure in the reactor to rise, since water will not be fed to the reboiler to remove reactor heat. The turbine bypass valve will begin to open at about 605 psig to maintain this pressure in the reactor. The operator will have approximately one minute to start the standby pump in order to resume normal operation. If the pump fails to start, or if the sequence of events is ignored, one of the following will occur, depending on conditions:

- a) Reactor overpressure interlock will initiate reactor shutdown.
- b) Hotwell level will rise and initiate alarm in control room.
- c) Low-level interlock in primary flash deaerator will cut off feedwater pump; power interruption on pump interlock will initiate reactor shutdown.
- d) Primary reboiler shell low-level alarm will warn operator that water is being boiled away with little or no makeup.

5. Cooling and Purification Pumps

In case of failure of the auxiliary cooling water pumps, cooling water is lost to the building air-conditioning system, vapor recovery, reactor purification system secondary coolers, and other coolers. The loss of cooling for the purification system (ion-exchanger circuit) would automatically shut off the pump in this circuit. The operators would shut down the reactor if cooling water could not be restored from the Laboratory main.

6. Shield Cooling Pumps

In case of failure of a shield cooling pump, an alarm in the control room would notify the operator of low flow in this circuit, whereupon he would turn on the spare pump. If the operator did not turn on the spare pump, the primary shield would be heated to approximately 450°F. It is possible, although not likely, that some crumbling of the concrete in the primary shield would occur if the concrete were maintained at 450°F for a long period. However, the results would not be hazardous.

Approximately 12 kw/hr will be removed by the cooling water during 100-Mw operation. This was the amount of heat removed during 20-Mw operation and is not expected to vary.

D. Instrument Failure

Instruments which are capable of inserting the rods upon receipt of the proper signal will also insert the rods when the instruments themselves are not functioning properly. Similarly, alarm circuits are designed to give alarms when malfunctioning of the circuits develops.

The pressure-control system in the turbine-generator system fails in the closed position and will increase reactor pressure. This will automatically increase power level in the reactor. The pressure-control valve in the primary reboiler system will open to allow the reboilers to accept the added load and maintain reactor pressure at 600 psig.

The motorized feature of the turbine bypass valve provides a means of setting the minimum opening to which the valve can be driven by springs. This can be used to limit the rate of pressure rise in case of failure of the control system. The pressure-control valve in the primary reboiler system will tend to close on a drop in reactor pressure or fail to a closed position. The reactor will be shut down when this valve fails to a closed position since an overpressure or high-flux interlock will drop the control rods.

E. Safety Rod Failure

Since there will be nine safety rods whose combined strength at operating conditions is greater than the maximum amount of k they will ever be required to control, failure of one, or even two, safety rods does not create a dangerous situation. Any safety rod which does not function properly will be repaired or replaced. The reactor never will be operated with less than the number of safety rods required to control all available excess k . Failure of rods to reduce reactivity is backed up by the borated water injection.

To accomplish shutdown from normal operation, only enough rods need to be inserted to compensate for the amount of reactivity ($\sim 4\%$) held in steam voids. If, because of open valves, the reactor temperature decreases, the additional reactivity compensation is easily achieved by driving rods in through over-riding clutches or by injecting boron.

F. Change in Power Demand

Previous operating experience has shown that the changes in electrical demand on the generator have been negligible. Generator output will be varied by adjusting the position of the turbine admission valves manually.

A small increase (say 5%) in electrical load will result in diverting steam from the reboiler load, since the present bypass valve will be closed during operation. The reboiler bypass valve will close slightly due to a drop in pressure in the reactor until the pressure holds constant at the set point near 600 psig. The valve will retain an equilibrium condition near 600 psig in the reactor when conditions are satisfied. The intermediate system will gradually adjust itself automatically by diminishing its output because less steam is entering the reboiler.

A decrease in load causes the pressure to rise in the reactor and will automatically adjust itself in reverse order of the above until equilibrium is re-established.

If the pressure exceeds 625 psig, the reactor is shut down automatically by an overpressure shutdown interlock. If the pressure increases to 650 psig, the relief valve is opened, dumping steam to the condenser. Further increases in pressure cause opening of safety valves which are set to open at 700, 725, 750 and 775 psig, which will also dump steam into the condenser. If the condenser pressure exceeds 20 psig, which may result if all steam valves opened, a rupture disk in the condenser would open and discharge steam into the vapor-containment building. Radioactivity will close the ventilating valves on the building. Opening of any of the relief valves will shut down the reactor.

If the turbine-generator is suddenly tripped off the line, its bypass valve would dump all its steam to the condenser. The pressure in the reactor will remain essentially constant during this transition. The operator can adjust both bypass valve pressure reference settings if a small deviation from 600 psig is encountered in the reactor.

G. Too Much or Too Little Water in Reactor

Safety rods cannot be withdrawn if the water level in the reactor is too low or too high. The rods will be inserted automatically if the water level exceeds preset limits during operation. If the high-level safety trip should not work or be bypassed by a wiring change, and if the tank were filled entirely with water, the reactor would be less safe under fast excursion conditions because expulsion of water from the core would be difficult. It is not sufficient to just fill the reactor with water completely. Some way must also be found to add a large amount of k to the reactor suddenly. This would require rapid insertion of fissionable material or rapid removal of poison. The maximum Δk likely to be encountered by inserting a single element is about 2%. The mechanical design is such that a fuel element cannot be added suddenly. This amount of reactivity could be controlled by the metal temperature coefficient.

A large amount of k could also be added by withdrawing the safety rod suddenly. Before this can be done the cam clutch must be disconnected from its drive mechanism. Some means must be provided for driving the rod out of the core at a rapid rate. The new rod weighs about 200 lb.

If the safety rods could be withdrawn from the reactor while it is cold and clean and with the water level well below the top of the core, and if water were added to the reactor with 4 pumps at the feedwater pumping rate of 1600 gpm, the total available k of 9.6% could be added over a period of about 19 sec. The maximum rate of addition of k might be as much as 1.0% per second. Safety features are incorporated in the design to prevent the occurrence of this situation; these include the releasing of rods for low water level and a water temperature below 325°F.

H. Water Leaks from the External System

The dominant radioactivity in the steam circuit will be due to 100-Mw fast neutron irradiation of O^{16} to form N^{16} . Since the molecular density of steam is very low compared with pressurized water, the amount of N^{16} will also be correspondingly low. The short half-life (7.4-sec) of the N^{16} prevents accumulation of this activity in the vicinity of a leak. Even if all of the water in the system should flash down to atmospheric pressure, the integrated exposure from the N^{16} will not pose a serious hazard to operating personnel in the building.

Development of faults in condenser tubes would permit circulating water to leak into the condenser. An increase in electrical conductivity of condensate or of reactor water will be the first indication of major leakage. Water-level indicators would eventually show an increase in water inventory. Upon noting evidence of a leak the operator would isolate that half of the condenser (as indicated by increased conductivity) by closing butterfly valves in the inlet and outlet circulating water lines. The water boxes and tubes on the leaking side would then be drained to a tank in the basement of the Power Plant Area. The condenser would still be able to condense the turbine exhaust and/or bypassed steam, although at a slightly higher back pressure. The operator would open the valve admitting spray water into the bypass desuperheater to prevent damage to the condenser due to high thermal gradients.

Condenser tubes have failed in the present condenser by erosion due to steam deflecting off condenser bracing. The above sequence was followed. The faulty tubes were located and plugged. Circulating water was restored to both halves of the condenser. The entire operation took place without reducing reactor power from its 20-Mw(t) level.

If a break occurred in the primary reboiler tubes, radioactive fluid would enter the 350-psig intermediate system. This system will be monitored constantly for radiation, and if radioactivity is detected it would indicate a tube failure. This leak would also show up as a loss in inventory of the 600-psig primary system. Operation could be continued in one reboiler under reduced power while the tubes were being repaired or plugged in the isolated reboiler.

The same analysis applies to the primary drain cooler and the primary subcooler, except that in the latter the cooling tower water leaving the unit will be monitored. Either vessel can be bypassed for continued operation while corrective measures are taken to repair these units.

I. Cladding Failure

Zircaloy-2 is quite resistant to corrosion at a maximum fuel surface temperature of 514°F. However, in the fabrication of a very large number of plates and rods, it cannot be guaranteed that all welds are perfect or that the cladding is free of defects.

Assuming no corrosion inhibitor is contained in the fuel, the consequences of a clad rupture would be a reaction of the uranium with water, with subsequent swelling and discharge of oxide particles to the coolant stream. Prior to irradiation, the current EBWR fuel alloy (5% Zr - 1.5% Nb) corrodes at a very slow rate in water at 488°F. However, a limited number of tests indicates that the corrosion resistance is markedly reduced by irradiation in a reactor. To date, the EBWR has not experienced a clad failure.

Since the possibility exists that a cladding failure may occur after a long irradiation period, conservatism demands an assumption that the corrosion rate of the fuel will be high. The bulk of the radioactivity will remain in the reactor or be removed by the reactor water purification system. The decontamination factor for nongaseous radioactivity getting into the external system was found to be at least 10^4 by sodium tracer technique measurements performed in EBWR.

Calculations indicate the maximum centerline temperature of a thick enriched fuel plate will be about 767°F (408°C) at 100 Mw. As a consequence, the fuel alloy in the subassemblies to be reloaded (estimated maximum burnup = 0.3 - 0.4 at %) may exceed the critical swelling temperature. The swelling may block the coolant channel and bring about a condition of overheating, but may not necessarily lead to melting.

Metallurgical studies are under way on an EBWR fuel subassembly (having one of the highest burnups) to determine the maximum temperature that can be attained safely with this fuel. Preliminary tests

indicate: (1) swelling occurs at about 550°C (1022°F); (2) a relatively long fuel life can be expected at 500°C ; and (3) a relatively short life at 600°C .

Concurrent studies are in progress to determine the effect of scale deposits on the heat transfer surface of fuel elements. These studies, in combination with previous reactor flux-distribution measurements, are designed to ascertain the maximum fuel surface temperature that may be promoted by the scale deposit. Calculations to date indicate the temperature will not exceed 500°F .

J. Startup Accident

The existing antimony-beryllium neutron source, previously irradiated to saturation and maintained at full strength by operation of the EBWR, will be used in the new core.

A period meter, set to release safety rods at periods less than 5 sec, will be used in all reactor startups. It is not desirable to use a period trip meter at full power because of the short period variations of the neutron flux.

Safety rods cannot be withdrawn unless the flux-indicating galvanometer is on its most sensitive scale and the water temperature is 325°F or above. The rate of addition of k by rod withdrawal is limited to $\sim 0.01\%$ per sec. With this rate of addition of reactivity, the water will begin to boil as soon as the fuel plates reach a temperature just slightly higher than the boiling point of the water. The negative steam and temperature coefficient will automatically compensate the reactivity added by rod withdrawal. Since the burnout heat flux is at least twice the maximum operating heat flux, this does not appear to be a dangerous situation. Burnout occurs at a steam volume fraction greater than 80%.

Experience with EBWR has shown that the safety procedures employed during startup are adequate.

K. Melting of Fuel Plates

Any metal-clad fuel element will melt if an arbitrarily large amount of k is added suddenly to the reactor. The BORAX-I reactor was able to shut itself down with periods as short as 5 msec. The shutdown mechanism was almost entirely one of steam formation, with a minor contribution from expansion of the fuel elements. An excursion with a period of 2.6 msec destroyed the core. EBWR is at a disadvantage relative to BORAX-I in that it must form much more steam per unit area of heat transfer surface for control of a given amount of excess k . This steam must be formed by transferring heat through zirconium cladding, which is a very poor thermal conductor relative to the aluminum cladding on the BORAX

fuel elements. The EBWR reactor has other important shutdown features, however. The metal temperature coefficient (Doppler effect and metal expansion) together with delayed neutrons is estimated to be able to compensate for the sudden addition of 0.34% plus the deflated dollar value (see Section V).

If the pressure vessel should rupture above the water line as a result of a short-period excursion, at least 30% of the water could be lost from the reactor as the stored energy flashed to steam. Some quantity of bulk water also would be expelled. The resulting pressure increase in the plant would be considerably less than the 15 psig which the building is designed to withstand. Any fuel blown out would be expected to cool sufficiently by radiation, conduction to objects with which the element is in contact, and air convection to keep it below the melting point. The building spray system would aid in condensing the steam in the building.

Possible radiation dosages resulting from leakage of fission products and plutonium from the gastight building were given in Appendix D of ANL-5781 as based on 20-Mw operation. Although the 100-Mw core will contain approximately 60% more fuel, Appendix D (entitled "Evaluation of Radioactive Hazards in the Surrounding Area") should remain applicable and conservative.

L. Excessive Pressure

The safety rods are inserted when the pressure exceeds 625 psig. A pressure-relief valve opens at 650 psig. Safety valves set at 700, 725, 750 and 775 psig further protect the reactor by releasing steam to the condenser. A rupture disk in the condenser will open if the condenser pressure exceeds 20 psig and permit steam to be discharged into the vapor-containment building. Opening of these valves will reduce the power level of the reactor while the flashing is occurring. The safety rods are inserted by the opening of any one of these valves and the reactor is shut down. The safety rods are backed up by the boric acid system. Injection of the high-pressure boric acid is completed in about 20 sec after initiation.

M. Sudden Closure of Steam Valve

The steam-system design is such that there are no valves other than the present turbine trip-throttle valve and the primary trip valve which can close suddenly in one of two main steam lines.

When the reboilers are not in use, experience indicates that one of the following three events will occur:

- (1) The reactor will be shut down by the turbine trip interlock and the bypass valve will remain essentially unchanged until shutdown cooling is initiated.

- (2) If the turbine trip interlock is bypassed with a key switch, the bypass valve will open and maintain reactor pressure at the set point. The reactor will continue operating without sensing the interchange.
- (3) Should the bypass valve fail to make the interchange in Case (2), control rods would be inserted by one or two shutdown interlocks: high-flux and/or overpressure. EBWR has been shut down by the high-flux interlock when the power rate of increase was approximately 1 Mw sec. This corresponded to a rate of increase of pressure of about 3 psi/sec.

A sudden closure of the turbine trip valve will have little effect on the reactor in the event the reboilers are in use. Assuming the turbine trip valve interlock was bypassed with the key switch, a minor adjustment of the regulating valves may be required following the interchange. If the interlock was not bypassed the reactor would be shut down.

If the interchange fails to respond, the reboiler pressure-regulating valve will open further to maintain 600 psig in the reactor. Either the reactor power will be decreased, or the secondary system will be adjusted to accept the 20% increase in steam flow.

If the reboilers are in use and the primary trip valve closes suddenly, the reactor will be shut down by the valve closure interlock. If the valve closure interlock fails, the reactor will be shut down by the high-flux interlock. If this trip fails, the reactor will be shut down by the overpressure interlock.

N. Failure of Steam System

A failure of the main steam system may be promoted by:

(1) rupture of the steam piping, steam dryer, or a valve upstream of any control mechanisms; (2) rupture of the steam piping or valves downstream of the control mechanisms, the turbine or condenser casing, a reboiler, or tubes in the reboiler; or (3) internal mechanical failure of turbine valves or control mechanisms.

Rupture of the main steam piping or steam dryer upstream of the shutoff valves on the primary reboiler will permit the steam to escape into the gastight building. In case of a wholesale failure, the rate of steam flow will be sufficient to reduce the reactor power instantaneously, due to flashing. The safety rods will be inserted by one of the following shutdown interlocks: (1) when the general level of radioactivity in the building rises above a predetermined level; (2) when the water level in the reactor drops; or (3) when the water temperature drops to 325°F, if the operators had not already inserted the rods.

The effect on the reactor of a failure of the condenser or turbine casing or other elements downstream of the control mechanisms would be similar to that discussed in the preceding paragraph.

In the event the EBWR turbine shaft or casing should fail, the reinforced concrete lining of the steel shell will prevent puncture of the shell by flying fragments. The failure of the turbine will either trip the throttle valve or completely open the governor valves, depending on the sequence and nature of the failure. Tripping the throttle valve will shut off steam to the turbine and open the pressure-regulating bypass valves. Control rods will be inserted into the reactor by any one of the following shutdown interlocks: (1) by high radioactivity in the building; (2) low water level in the reactor; (3) loss of vacuum in the condenser; or (4) by the operators. Opening of the governor valves will increase the flow of steam from the reactor and, in addition to the shutdown signals just mentioned, will reduce power in the reactor temporarily, due to flashing in the core. The reactor after being shut down will be cooled by one of the cooling systems previously described, since the condenser will be inoperative.

In the event the primary side of a reboiler vessel should rupture, control rod insertion will be initiated by one of the following shutdown interlocks: (1) high radioactivity in the reboiler building air; (2) low water level in the reactor; (3) low water level in the primary flash deaerator cutting off the feedwater pump; or (4) by the operator. Shutdown cooling will be accomplished by one of the cooling systems described in Section VI-A.

In order to formulate a safe system for handling leakage from the primary steam system to the intermediate steam system in the event of a tube failure in one of the primary reboilers, the following parameters were considered:

1. quantity of fluid being transferred;
2. a fluid "sump" to handle leakage; and
3. effect of leakage on other system components.

Based on failure of a $\frac{3}{4}$ -in. diameter tube in the steam phase of one reboiler handling a 40-Mw load (560 psig steam on one side and 350 psig pressure on the other), the flow through the open tube will be 12,000 lb/hr. This amounts to approximately 10% of the flow or about 4 Mw.

The existing condenser at EBWR can safely handle an added load of 20 Mw without an appreciable rise in pressure. Since one tube failure represents a 4-Mw additional load for the condenser, five simultaneous primary reboiler tube ruptures can be sustained by the existing condenser.

O. Sudden Introduction of Cold Water

Since the reactor has a negative temperature coefficient, introduction of cold water increases the reactivity. This could happen, for example, if the feedwater pump was not turned on until the reactor had been raised to its operating temperature. However, the rate of change of reactivity is quite slow, since only about 15% of the total water in the reactor is added per minute by the feedwater pumps. Perhaps the most dangerous situation of this type could come about if the main steam line is closed and the reactor brought up to temperature before the feedwater pump is turned on. The reactor power level would merely rise to a point where the increase in temperature and steam formation nullified the increase in k due to addition of cold water. The temperature at which this occurred would be somewhat less than the temperature existing when the pump was turned on. That the excursion is slow can be deduced from the fact that cooling all of the water in the reactor from 488°F to 70°F increases the reactivity about 1%.

P. Improper Charging of Fuel

The maximum Δk likely to be encountered by inserting a single element is about 2%. If the reactor is just barely subcritical when the fuel is inserted, an excursion of about 5-msec period would be initiated (with the initial value of the dollar, but correspondingly less as P_u builds up).

The low-reactor water temperature interlock would have tripped all rods before the vessel cover was removed, making the reactor very subcritical. An extra precaution is taken to reduce the chances of dropping a fuel assembly during loading by attaching a positive locking device to the handling rod, as described in Section III-B-1f. This stop will hold the fuel just above the control rod guides. The fuel element is lowered into the core the remaining distance by a hand-operated winch.

Careful monitoring and accounting procedures will ensure that the U^{235} content of any enriched fuel assembly is not materially greater than the design value.

Q. Earthquake

Earthquakes in the Chicago area have been infrequent and mild. The most serious **consequences** of a severe earthquake, if it should occur, would be to rupture the vessel or piping and jam the safety rods. Concentrated boric acid can be injected into the reactor by the operator if control rods have not reached their minimum positions of reactivity after a shutdown.

In case of rupture of the reactor vessel, enough water would be lost from the reactor, suddenly or gradually depending on circumstances, to shut the reactor down. Shutdown cooling would be accomplished through

the boric acid spray ring, as previously described. The piping for this cooling system is designed to prevent damage from tremors.

The concentration of fuel in the core of the reactor could not be changed by earthquake, since the fuel elements are bound closely together. Thus, there should be no danger of an increase in reactivity.

R. Fire

The reactor and its building are essentially fireproof. Hazardous amounts of combustible materials are not allowed in the building.

S. Sabotage with Fissionable Materials or Explosives

In any heterogeneous reactor the fuel can be melted by the sudden addition of an arbitrarily great amount of excess reactivity. This can be done by injecting fissionable material or by withdrawing poison. The proposed reactor should sustain 0.34% prompt excess reactivity without mishap.

Proper placement of explosives could: (1) rupture the vessel to drain the coolant; (2) rupture the emergency shutdown cooling system; and (3) destroy the gastight integrity of the containment shell. Although this series of events is technically feasible, it is not possible to incorporate safety features consistent with the aims of a dedicated saboteur.

T. Sabotage by Pressurizing Reactor Tank with Gas

By pressurizing the reactor with air or other gases prior to startup, the saboteur could raise the boiling point of the water and inhibit the inherent safety mechanism of steam formation. A large number of gas cylinders would be required to raise the pressure to a point where any hazard is involved. To do any damage, the saboteur must also introduce a large amount of k suddenly. The introduction of gas would merely be a way to make the excursion resulting from sudden introduction of k more serious, but would not be a hazard in itself.

U. Explosive Chemical Reaction between Fuel Elements and Water

High-order explosion criterion of vessel failure was presented and described in ANL-5781. Corrected information is presented in Section III-D-2 of this report.

APPENDIX A

Shutdown Cooling

In the event of a sudden shutdown from 20-Mw operation, accompanied by loss of water in the reactor core, it was indicated that some of the core would probably melt. Should a similar event occur after 100-Mw operation, the fission gases and products that result from melting would be confined to the vapor containment vessel if the products escaped from the reactor vessel.

APPENDIX B

Hydrology and Seismology

This section is not affected by increasing the power output of EBWR.

APPENDIX C

Metal-Water Reactions

Additional information is included in Section III-D-2, entitled "Missile Protection," and also in Appendix G of this report.

APPENDIX D

Evaluation of Radioactive Hazards in the Surrounding Area

The previous evaluation of this section (reported in ANL-5781) was based on a 4-ft diameter core, four feet high. The 100-Mw core will not exceed the above-mentioned core by 60%.

The following discussion is based on Part II of Appendix D, entitled "Dosages from Spherical Radioactive Cloud," as was reported in ANL-5781.

In analyzing doses from radioactive clouds, the operating power level of the hypothetical reactor affects in direct proportion the concentration of radiation in the cloud. Ground doses show a slight inverse dependency on the power level because the height of rise of the cloud is proportional to the power level raised to the 0.3 power. This means that, using the same assumptions as in ANL-5781, the calculated gamma and beta doses due to a maximum accident situation will be greater by less than a factor of five. It is evident that if the cloud rises, the doses are

still rather nominal, and if the cloud does not rise, they are already hazardous. Doses due to rainout are large for the 20-Mw analysis, while those calculated for a slow leakage situation do not become prohibitive when increased by a factor of five.

The inhalation doses are based on 1% burnup of all uranium in the core. This corresponds to 56,000 Mwd. EBWR has now logged about 7000 Mwd and the original fuel will never achieve burnups of this magnitude. The current plutonium inventory is estimated to be about 4.5 kg, compared to the 22 kg used for the hazard analysis. This indicates that the use of some elements spiked with plutonium will increase the plutonium inventory somewhat, but will not approach the figure used in previous calculations. Thus the inhalation considerations set forth in ANL-5781 are still adequate.

APPENDIX E

Inherent Safety of the Reactor

The function of this section was to support Section V in the previous hazard report ANL-5781.

Section V has been revised and has references cited for its support.

APPENDIX F

Climatology (Meteorology Group)

This section is not affected by increasing the power level of EBWR.

APPENDIX G

Reactor Support and Containment Structure

This section is not affected by increasing the power level of EBWR.

APPENDIX H

Reactor Stability

The stability of EBWR was investigated and reported.⁽²⁾ An analytic expression for the feedback transfer function was obtained and

fitted to experimental data. Power coefficients and time constants for various modes of operation were determined. The power coefficients and time constants obtained at the present power levels can be used to extrapolate the data and thereby predict reactor performance at higher power levels with remarkable certainty.

Figure 19 shows the open loop gain for the product of the reactor function (G) and the feedback function (H) at 180° of phase shift. The phase margin remaining when the gain of the product (GH) is unity is also shown. Linear theory predicts that the reactor will not oscillate spontaneously until unity gain and zero phase margin (180° phase shift between reactor neutron level and reactivity input) occur simultaneously. Extrapolation of the data of Fig. 19 shows that oscillations will occur at 68 Mw with a 114-element, 4-ft core.

The dynamic behavior of the large reactor will be determined by power density. It is anticipated that the reactor will produce 80 Mw ($147 \times 62/114$) with 147 fuel elements (5-ft diameter core, 4 ft high), and will have approximately the same gain and phase margin as 114 fuel elements (present core) operating at 62 Mw. If the element length is increased to 5 ft, the enlarged core will produce 100 Mw ($80 \times 5/4$) and will have approximately the same stability characteristics as the present core at 62 Mw.

Transfer function data taken at low power levels in the larger cores will permit more accurate extrapolation to reactor performance at high power levels. The present core was successfully operated at 62 Mw following an experimental program of transfer function measurements. A similar procedure will be used prior to operation at 100 Mw.

The extrapolation of linear data to show that spontaneous oscillations will occur at a specified power level does not mean the reactor cannot be operated satisfactorily at this or at higher power levels. The inherent non-linear features of the reactor limit the magnitude of the oscillations. The reactor may be operated satisfactorily with a large amount of noise in the neutron flux signal, but the reactor cannot be operated when it has reached a condition of diverging oscillations. The personnel involved at ANL are familiar with the unstable condition and will recognize it quickly if it occurs during an experiment.

The margin of practical stability between the linear oscillatory limit obtained through the transfer function tests plus computations and the power at which divergent oscillations will occur is currently being investigated. The methods applied here do not provide any absolute assurance of stability in the power range indicated. However, the data show that there is reasonable assurance of stability for the reactor at powers to over 100 Mw thermal. Direct extrapolation could place the expected lowest unstable power at 110 Mw ($147/114 \times 5/4 \times 68$). Tests in this range will provide the information essential to further analysis.

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