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ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

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SAFETY ANALYSIS ASSOCIATED WITH THE
PLUTONIUM RECYCLE EXPERIMENT IN EBWR

by

B. J. Toppel, P. J. Vogelberger, Jr.,
and E. A. Wimunc

Contributors

R. Avery

C. H. Bean

J. B. Heineman

W. C. Lipinski

W. Reardon

(General Electric, Hanford)

A. D. Rossin

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FOREWORD

A Safety Analysis Report was originally submitted to the AEC-COO in March of 1964. After some suggested changes were agreed to, the original report was revised. It was conditionally approved in the fall of 1964 after limited copies were printed and submitted to the AEC in a version dated November 6, 1964. General distribution of this report was not made until the fall of 1966 owing to delays in power operations approval resulting from preparation and review of the reactor vessel survey which was initiated following observation of cracks in the cladding.

Results of the inspected reactor vessel and the surveillance plans are detailed in an Argonne National Laboratory report, ANL-7117, dated November 1965.

Zero power critical experiments performed on the system and improved physics calculations have revealed some differences with the predicted values in this report. These newer values will form the subject of subsequent topical reports and papers. They do not change any of the basic conclusions in regard to the safety of the system.

The commission has been notified through formal channels of all major changes and findings made since the original version of this report (November 6, 1964) was submitted.

This report is identical to the November 6, 1964, report and does not include results of the vessel inspection, piping repairs, and recent physics results.

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I. INTRODUCTION

The Experimental Boiling Water Reactor (EBWR) was originally constructed to demonstrate the feasibility of a direct-cycle, natural-circulation, boiling-water reactor integrated power plant. The reactor was designed to produce 20,000 kW of heat in the form of 600-psig saturated steam which was fed directly to a turbogenerator, generating about 5,000 kW of electricity.¹ This power level was considered as a minimum value for producing useful information which could be extrapolated and used in the design of large, central-station power plants. The reactor achieved criticality during December 1956, and began full-power operation at 20 MWt on December 29, 1956.

Subsequent experiments at power levels ranging from 20 to 40 MWt indicated that stable operation at powers as high as 66 MWt was possible with the initial 4-ft-dia core. This was affirmed in 1958 by the short-term operation of the plant at 61.7 MWt. Further power increase at this time was precluded by the feedwater pumps which were operating at maximum capacity.

A detailed study of EBWR stability was made by taking a series of transfer function measurements relating power level to reactivity input. Data at various power levels, steam pressures, and control rod positions were accumulated, analyzed, and extrapolated to predict reactor performance at higher power levels. The results of this study were the subject of a paper presented at the 1958 Geneva Conference.²

Subsequent analyses and projection of these data to a 5-ft-dia core indicated that with some modification of the core structure and pressure vessel internals, and with additional heat-removal equipment, EBWR could operate at or near 100 MWt. In July 1959, the reactor plant was shut down preparatory to the onset of proposed modifications.³ Physics analysis for the 100 MW operation is discussed in ANL-6305.⁴

On May 25, 1962, permission was granted by the U. S. Atomic Energy Commission to operate EBWR at power levels up to 100 MWt. Administrative approval by the International Atomic Energy Agency was granted on July 11, 1962.

Upon completion of the modifications, the reactor was subjected to a series of stability tests at successive power level increases up to 70 MWt. After careful analysis and scrutiny of the data, it was decided that operation at 100 MWt could be achieved safely. This power was attained during November of 1962.

The significant observations and reactor performance data obtained during the period from July 1959 to December 6, 1962 are summarized in Paper No. CN-15/3 presented at the IAEA Conference on Operating Experience with Power Reactors in Vienna, Austria in June of 1963.⁵ Association with the IAEA was terminated shortly after completion of the 100 MW operation.

In anticipation of release of the EBWR from the Water Cooled Reactors Program, the Division of Reactor Development early in 1962 requested Argonne to participate in the Plutonium Recycle Program with particular emphasis on that part of the work which would be appropriate using the EBWR facility.

The joint Argonne-Hanford program, which has evolved as a result of the above request, is now underway for the purpose of obtaining information useful for the utilization of plutonium as a fuel in light water, thermal systems. Such utilization could be made in either an all thermal plutonium recycle system, or as the thermal part of a mixed fast-thermal reactor complex.

An encouraging facet to this cycle is that studies carried out on fast systems have indicated that the blanket may not be able to pay for itself in such systems. The above system would alleviate the charges against the blanket since these costs would be the same as those required for fueling the thermal core. Also, it appears that plutonium containing much of the higher isotopes is advantageous, in terms of neutron economy, as fast reactor feed material.

The physics information required for the development of either the all thermal or the mixed fast-thermal concept is essentially the same. Since the attainment of certain rather basic numbers, in particular obtaining data on the isotopic behaviour of plutonium as it is irradiated in a thermal reactor, is the main objective, it appears desirable that the experiment be made as simple as possible while still yielding this information so that it would be most amenable to analysis.

In specifying an EBWR experiment which would be useful in the Plutonium Recycle Program the following considerations were kept in mind. It is advisable that modifications to EBWR should be kept to a minimum. It should remain a boiling H₂O system. EBWR should be used primarily as an irradiation facility rather than in support of an active reactor experimental program.

This report describes general improvements to the EBWR plant, as well as facility modifications and physics results specific to the planned plutonium recycle experiment. Details of the facility which have not changed from previous reports¹⁻⁹ are not repeated here.

II. SUMMARY

A. Description of the Experiment

In the planned experiment on EBWR, the 148 element core (Fig. 1) is divided into four zones; the central 6 x 6 square, the immediately adjacent 28 element row, the next 32 element row (excluding the fuel elements on the 45° symmetry lines), and the remaining 52 outside elements. One of the outside elements will be replaced with an Sb-Be neutron source.

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Fig. 1. Schematic Layout of Plutonium Recycle Experiment Core Loading

will contain europium and samarium oxide burnable poisons to minimize the reactivity variation with burnup.

The outer 52 elements contain natural uranium and will not be replaced during the experiment.

Each fuel element contains a 6 x 6 array of Zircaloy tubes about 0.372 in. ID with 0.025 in. walls (0.945 cm ID with 0.064 cm walls). These elements are similar to those of the EBWR Core 2 reference design.¹⁰

The central zone represents the plutonium test region. It contains mixed plutonium-uranium oxides.

The two rows surrounding the Pu region may be referred to as the enriched uranium shim zones, and it is these rows which may have their reactivity periodically increased in order to restore the reactivity of the over-all system. They will initially be identical and

One of the control rods will be similar to that used previously for the 100 MW operation, being fabricated from hafnium and will again be used for oscillator measurements. The remaining eight control rods will be boron-stainless steel rods as before but the boron will be enriched to approximately 90% in B^{10} rather than containing natural boron. A new design is also being incorporated to eliminate the previous problems encountered due to differential steel-zirconium expansion. (See Section III.C.)

The plutonium fuel elements are designed for a burnup of 0.025 (~25,000 MWd/tonne), although no definite duration has been set for the experiment. It should be continued as long as feasible and as long as useful information on isotopic change behavior can be obtained. It is anticipated that the interesting range may only be up to about 9,000 MWd/tonne. The higher design burnup allows for continuation if deemed worthwhile.

The fuel element boxes in the central region are somewhat different than in the remainder of the core. They are of a "take apart" design and allow the withdrawal and replacement of individual rods within the element. This feature is necessary to permit isotopic analysis studies of the central region as the burnup proceeds. These studies are the main objective of the experiment.

It will be interesting to note throughout this report similarities between the proposed operating conditions and previous EBWR experience. For example, the above initial loading corresponds to 13.3 kg of Pu^{239} whereas EBWR at the time of shutdown contained 9 kg of Pu (Ref. 11).

Prior to the start of irradiation, critical experiments will be performed. These experiments will be performed partly at Hanford, using rods similar to those to be shipped to Argonne for assembly into the fuel elements. At Argonne, critical experiments will be performed within the EBWR vessel. In particular, it will be highly desirable to obtain a clean, i.e., no control rods inserted, critical configuration of only the plutonium fuel elements where the system is controlled with boric acid. The usual reactivity coefficient and other standard measurements will also be made. In particular transfer function measurements to test the system stability will be made after loading of the driver zones and during the approach to power. The fuel will then be irradiated a predetermined amount, of the order of 2,000 MWd/tonne. This would require an elapsed time of about 4 months, assuming sustained operation at an initial power of about 40 MW although correspondingly shorter times would be involved if higher powers are attained. At this point a major investigation of the irradiated fuel would be undertaken. This would consist of removal of some plutonium rods from the system. A selection of the order of 5-10 rods might be sufficient to give a thorough mapping of the plutonium region. The rods would then be examined for isotopic analysis. Further measurements would at

this time be made on the reactivity of the system. It would again be highly desirable to come to a clean condition, that is, where the assembly only contains the plutonium elements and the system made critical by the addition of the proper amount of boric acid. Further measurements of items related to various reactivity coefficients would also be made at this time. After these measurements are made the irradiation would proceed for the next major time step, perhaps again of the order of 2,000 MWd/tonne. Owing to burnup of the plutonium zone and redistribution of the power, this second step would require an elapsed time of about 5 months, again assuming sustained operation at a power of about 40 MW.

The observed reactivity and control rod worth variations during the course of the irradiation will determine the desirability of alteration of the overall reactivity of the system. It is expected that if such variation is desirable, it would be accomplished by altering the composition of the shim zones.

The program outlined above of irradiation and periodic sampling and analysis of the plutonium fuel will be continued as long as the useful information derived thereby is consistent with the interest in the experiment and the effort required to achieve the required irradiation.

Comparison of the isotopic analyses obtained for the plutonium fuel with calculational predictions during the course of the irradiation will facilitate the setting up of a mathematical model which could serve as the basis for future calculations for prototype systems of either the all thermal recycle type, or of the fast-thermal complex.

B. Safety Considerations

In considering the safety of a plutonium loaded reactor, the two distinguishing areas of concern are related to the potential hazard associated with dispersal of the plutonium and effect on the kinetic behaviour due to the reduced delayed neutron fraction. However, it is shown in this safety analysis that the proposed experiment and the previous EBWR core are closely related in regard to plutonium content and effective delayed neutron fraction, as well as plant configuration.

The initial core contains only about 50% more plutonium than did EBWR at the time of shutdown after the 100 MW operation. In addition, since oxide fuel is to be used exclusively, the proposed experiment has a significant safety advantage over the previous EBWR core with regard to the degree to which plutonium could be dispersed if a fuel rod failed.

The β_{eff} of the full system varies from about 0.005 initially to over 0.006 at a plutonium zone burnup of 0.006. By comparison, the β_{eff} for

EBWR was about 0.0054 (Ref. 7) at the end of the 100 MW operation. For the plutonium critical configuration, since Pu^{239} is the only fissile isotope, β_{eff} is lower and has a value of about 0.003. The control rod drive speed will be reduced from 28 in./min used previously to about 14.5 in./min so that the maximum reactivity insertion rate for the full system will be about $2/3$ of that encountered during the 100 MW operation. For the plutonium critical configuration, the central rod speed will be reduced to about 4.3 in./min. This speed results in a reactivity insertion rate of about 80% of that encountered in the full core loading, and about half of that experienced during the 100 MW operation. These changes represent a considerable conservatism compared with previous operation.

Even though the expected operating power is less than that previously attained safely in EBWR, the stability of the proposed system will not be taken for granted. The stability of the system will be investigated during the approach to power using essentially the same techniques and equipment which were successfully used during the 100 MW program.

Data and calculations pertinent to the brittle-ductile transition temperature for the EBWR pressure vessel indicate that no danger exists of brittle failure at operating temperatures. However, higher stresses are involved during hydrostatic tests. At some point in the program, it may be desirable to perform such tests above room temperature. This is easily done with the existing facilities. The need for increasing the temperature during hydrostatic testing as irradiation proceeds will be considered in the light of results obtained from surveillance samples to be included in the core loading.

The proposed system exhibits the same inherent safety characteristics as did the previous EBWR cores. In addition, the present oxide fuel loading has significant advantages, from a safety point of view, over the previous metal fuel core loadings. These advantages result from the higher melting point, lower thermal conductivity, and improved corrosion resistance of oxide as compared to metal fuel. Events which previously did not lead to serious core damage will again not be damaging for the present core. Events which previously would have caused core damage may now not be damaging. For example, whereas the previous core could sustain a step reactivity insertion of about 0.3% above prompt critical, the corresponding conservative value for the present core is about 1%.

The same potential "maximum credible accident" is postulated for the present experiment as was postulated for the previous core. This accident which involves no nuclear violation corresponds to complete loss of water during full power operation and results in melting of the fuel. Owing to the use of oxide fuel, the consequences of this accident would be less severe for the proposed experiment owing to the lower likelihood of chemical reaction with the water and subsequent plutonium dispersal as

compared to the previous predominantly metal fueled cores. In particular, rupture of the containment shell is not believed to be credible for the proposed experiment. In the case of the previous metal fueled core, such rupture was considered to be credible in the event of an extensive metal water reaction. Since the integrity of the containment shell is maintained, only slow leakage from the building would contribute to the resulting small radiological hazard in the surrounding area following the maximum credible accident.

It can be concluded that owing to improvements in the plant and to the fact that an oxide core is involved, together with the benefit of experience and confidence gained during previous EBWR operation, and to the similarity of the proposed and previous system that the plutonium recycle experiment is as safe as the system previously encountered in EBWR and in fact represents a significantly more reliable reactor-plant configuration.

III. DETAILED DESCRIPTION OF Pu RECYCLE EXPERIMENT

In this section we describe the details of the system which are specific to the plutonium recycle experiment. In Section IV, details of the general plant improvements are described. These general improvements are the result of previous experience with EBWR and would have been made independent of the nature of the use to which the EBWR had been put.

A. Core Description

The core shown in Fig. 2, 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. The fuel is located in a 4-in. square pitch lattice in the grid plate. The Zircaloy-2 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 Zircaloy-2 portion is inserted into the active lattice. This procedure has been followed in all past operations and does not constitute a change.

Each of the 4-ft active length fuel elements contains 36 fueled rods. These elements are similar to those of the EBWR Core 2 design.¹⁰ The fuel rods containing natural and slightly enriched UO_2 are positioned in the assembly by slotted end fittings. Differences in thermal expansion of the rods are allowed by locating one end of the rod on the lower grid support structure and allowing the opposite end to move axially, guided by slotted end fittings. The PuO_2 fuel rods are made so that the lower end is snapped into a holder and the upper end is held in a guide. These rods can be individually removed from the assembly without dismantling the assembly. A special tool will be used to grasp the upper end of the rod and a firm pull will release the lower end of the rod from its socket. The ability to remove individual Pu fuel pins is essential to the success of the experiment since the periodic investigation of the isotopic composition of the Pu zone is the main objective of the program.

Details of the fuel rods and elements are shown and described in Section III.B.

Table 1 summarizes the plant parameters. Some of the values are based on the calculated initial power of 42 MW although the system will be operated at as high a power as is consistent with safe operation, probably not more than about 70 MW. The maximum operating power will obviously be limited by the available reactivity, and the void coefficient of reactivity. Previous experience with EBWR has shown that no significant liquid carry-over occurs below about 70 MW, so this power is taken as an upper limit from mechanical considerations, assuming the calculated maximum operating power has been underestimated. In any case, however, stability requirements will always be controlling in regard to the maximum power level.

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Fig. 2. Cross Section of EBWR Reactor Vessel

TABLE 1. Plant Parameters

<u>General</u>		
Type	Direct cycle boiling reactor	
Heat output	20,000 kWt for turbine; ~22,000 kWt to reboiler	
Gross electrical output	5,000 kW	
Operating pressure	600 psig	
Operating temperature	489°F	
Coolant	H ₂ O	
Moderator	H ₂ O	
Fuel	PuO ₂ and UO ₂	
<u>Core</u>		
Active diameter	5 ft (152 cm)	
Active height	4 ft (122 cm)	
Total uranium content	~3930 kg (or 4470 kg UO ₂)	
Total U ²³⁵ content	~108 kg	
Total Pu content (8% Pu ²⁴⁰)	~14.5 kg	
Average water/fuel volume ratio	~2.7	
Structural material	Zircaloy-2	
<u>Fuel Elements</u>		
<u>Type</u>	<u>Materials</u>	<u>No. in Core</u>
Pu	1.5 w/o PuO ₂ in UO ₂	36 elements
	Zircaloy-2 tubes	
	Stainless steel end fittings	
Shim	6.0% U ²³⁵ content of U in UO ₂	60 elements
	Burnable poison	
	Eu ₂ O ₃ = 0.158 w/o (av)	
	Sm ₂ O ₃ = 0.0288 w/o (av)	
	Zircaloy-2 tubes	
	Stainless steel end fittings	
Natural Uranium	Natural (0.7% U ²³⁵ content of U in UO ₂)	51 elements
	Zircaloy-2 tubes	
	Stainless steel end fittings	
<u>General</u>		
Number of rods per element	36	
Cladding	~0.064 cm wall Zircaloy tubing x 0.945 cm ID (~0.025 in. wall x 0.372 in. ID)	
Fuel element end fittings	Stainless steel	
UO ₂ pellet dimensions	0.927 cm dia x 1.27 cm long (0.365 in. dia x 1/2 in. long)	
PuO ₂ dimension	0.945 cm dia vibratory compacted (0.372 in. dia)	

TABLE 1. (Contd.)

General (Contd.)

Power generation in average element (based on 42 MW operation)	286 kW
Heat transfer surface area	16 ft ² /element
Over-all average heat flux (at 42 MWt)	61,600 Btu/hr/ft ²

Nuclear Data

Average thermal flux	$\sim 1.6 \times 10^{13}$ neutrons/cm ² /sec
Neutron lifetime	$\sim 30 \times 10^{-6}$ sec
Reactivity to control	$\sim 15\%$

Control Rods

Total number	9
Strength of 9 rods	$\sim 17\%$
Shape	10 x 10 in. cruciform
Spacing	12.75 x 12.75 in.
Maximum withdrawal rate	~ 14.5 in./min ~ 4.3 in./min for center rod during Pu zone criticals

<u>Material</u>	<u>Boron Stainless Steel</u>	<u>Hafnium</u>
No. of rods	8	1
Thickness (nominal)	0.25 in.	0.125 in.
Boron content	2 wt% ($\sim 90\%$ enriched)	-
Absorber length	60 in. (48 in. required)	46 in.
Drop time	~ 1.35 sec/56 in.	~ 1.2 sec/ 48 in.

Pressure Vessel

Weight of contained water	~ 13.8 tons (hot)
Volume of contained water	4160 gal

Heat flux values and other quantities related to operating power are given in Table 1 for the predicted 42 MW power. Values for other powers may be obtained by simple proportionality. At 70 MW, the core average and core maximum fuel centerline temperatures are 1135 and 2460°F respectively. These values are close to the results which can be obtained by scaling the 42 MW values for the fuel temperature rise from the constant 489°F water temperature (see Table 18) by the factor 70/42. Even at the maximum power of 70 MW, the heat flux is still a factor of two below the predicted burnout flux (see Section III.H).

B. Fuel Elements

The reactor core 152 cm (5 ft) dia loading is initially a three zone loading with a total of 148 fuel elements having a 122 cm (4 ft) effective fuel length. The plutonium test fuel is located in a central 6 x 6 square zone of 36 elements. An intermediate zone, two rows adjacent to the central zone, contains 60 elements fueled with enriched uranium. The outer zone contains 52 natural uranium elements.

All fuel elements contain 36 rods loaded with oxide fuel in a square lattice. The fuel is contained in 0.945 cm (0.372 in.) ID x 0.064 cm (0.025 in.) wall Zircaloy-2 tubing. All loaded rods are evacuated, back-filled with dry helium or mixed argon-helium gas at atmospheric pressure, and sealed with Zircaloy-2 plugs welded to the jacket tube at each end.

A gap is maintained between the top level of the fuel and the bottom of the top end plug of each loaded fuel rod. This gap is intended for the collection of fission gas released during irradiation of the fuel to a designed burnup of ~25,000 MWd/tonne although no definite duration has been set for the experiment. It will be continued as long as feasible and as long as useful information on isotopic change behaviour can be obtained. It is anticipated that the interesting range may only be up to about 9000 MWd/tonne. The higher design burnup allows for continuation if deemed worthwhile.

Successful irradiations of UO_2 to greater than 100,000 MWd/tonne have been achieved.¹² Studies on mixed PuO_2 - UO_2 fuel¹³ have indicated that the mixed oxide should exhibit no unexpected irradiation effects as compared to the UO_2 experience. The fuel for the proposed experiment should therefore exhibit no radiation damage problems even if the higher design burnup should be approached.

The central plutonium bearing test fuel rods have a gap of 13.97 cm ($5\frac{1}{2}$ in.) above the fuel and all other enriched and natural uranium fuel rods have a gap of 8.89 cm ($3\frac{1}{2}$ in.) above the fuel. A stainless steel spring loaded to 2.268 kg (5 lb) is positioned between the fuel column and the top end plug to maintain the fission gas space and prevent shifting or dislocation of the fuel column during shipping, handling, or fuel loading at the reactor.

The 36 fuel rods in each fuel element are supported and positioned at each end of stainless steel support grids and at the center by means of a Zircaloy-2 center spacer grid. All support and spacer grids are contained within and fastened to a 9.52 x 9.52 cm ($3\frac{3}{4}$ x $3\frac{3}{4}$ in.) square Zircaloy-2 element can. This can has a 0.157 cm (0.062 in.) thick nominal wall fabricated from rolled, formed, and welded strip stock. A stainless steel locating tip end fitting attached to the bottom of the element can completes the fuel element.

The central plutonium test fuel is 1.5 w/o PuO_2 -98.5 w/o depleted UO_2 mixed oxide loaded into the Zircaloy-2 tube by vibratory compaction. This mixed oxide was prepared by blending 6 kg lots of ceramic grade plutonium and uranium oxide which was then jacketed in a double wall can, heated to 1200°C , evacuated, sealed, and compacted by a high energy rate process. This is a high temperature, high pressure impacting process which fuses the two oxides and keeps them from separating, thereby maintain a uniform dispersion of the PuO_2 in the UO_2 . The mixed oxides as compacted have a minimum density of 98% of theoretical and do not classify or segregate during loading, compaction, or during irradiation. A completely homogeneous dispersion of PuO_2 in UO_2 is maintained.

The high energy rate compacted material was removed from the can, crushed, screened, and loaded into the fuel tubes by vibratory compaction. The mixed PuO_2 - UO_2 was compacted to produce a 122 cm (48 in.) long column of fuel in each tube with a minimum density of 90% of theoretical. An additional 1.27 cm (1/2 in.) of depleted UO_2 was loaded on top of the fuel to prevent contamination of the tube end from plutonium fuel during subsequent handling and end closure operations. A 0.635 cm (1/4 in.) thick zirconia disc was inserted on top of the compacted oxide to act as a bearing surface for the stainless steel spring in the fission gas collection space and to retain the loaded oxide fuel.

About 2% of the fuel rods in the Pu zone will be special experimental rods. The properties of these special rods are discussed in Appendix C.

Figures 3 and 4 show details of the plutonium fuel elements and fuel pins.

Fuel for the intermediate or shim zone is in the form of UO_2 pellets enriched with 6 w/o U^{235} which are mechanically loaded into the Zircaloy-2 tubes. This shim fuel also contains a burnable poison, 0.158 w/o Eu_2O_3 and 0.0288 w/o Sm_2O_3 . A master blend of high fired rare earth oxide and UO_2 was crushed, screened, and blended with ceramic grade UO_2 to maintain a distribution of the rare earth oxide with the desired homogeneity of $\pm 3\%$ of the amount specified. The oxide was compacted into pellets without additives or sintering aids, sintered in hydrogen at 1650°C , and centerless ground to size. Finished pellets were 0.927 cm (0.365 in.) OD x 1.27 cm (0.500 in.) long with a density of 10.2 to 10.6 gm/cc. Pellets were stacked in 121.920 ± 0.025 cm (48.000 ± 0.010 in.) long columns, weighed, and loaded into the jacket tubes.

A number of considerations were involved in the selection of the burnable poisons and fabrication of the shim fuel.

It was desired that the poison be distributed homogeneously throughout the fuel so that there would be no self-shielding as in the case of discrete

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Fig. 3. $\text{PuO}_2\text{-UO}_2$ Fuel Element

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Fig. 4. PuO_2 Fuel Pin

poisons. Because of the different densities of the rare earth oxides and UO_2 there is a tendency for them to segregate during mixing. This was overcome by preparing the pre-fired master composition of rare earth oxide in UO_2 which was then blended uniformly with the rest of the UO_2 . Both chemical analyses and neutron scanning in the Juggernaut reactor have shown no detectable variation in rare earth additions between individual fuel pellets nor within end-to-end measurements for any one pellet.

A special X-ray fluorescence spectrometry procedure was developed by the Mallinckrodt Laboratory for detection of the low levels of europium and samarium specified. Results were reported with an analytical error of 5 ppm for europium and 3 ppm for samarium.

It is expected that the poison material will have physical and mechanical properties comparable to those of the fuel. Both Eu_2O_3 and Sm_2O_3 are ceramics with properties comparable to UO_2 .

These materials have high thermal stability, high compressive strength, low thermal expansion, high modulus of rupture, high hardness values, and low thermal shock resistance. The total rare earth addition has no measurable effect on the final density of fuel pellets. Density values for 100 pellets ranged from 10.23 to 10.31 gm/cm^3 with an average of 10.27 gm/cm^3 . The minimum specified density was 10.20 gm/cm^3 . With these properties the rare earth addition will have no adverse mechanical or physical stress effects on the performance of the fuel.

Weight loss measurements were obtained at ANL for mixed rare earth-urania oxides in a vacuum of 1×10^{-6} Torr and temperatures up to 2200°C. The rate of volatilization of europia, samaria, and urania were sufficiently similar to discount any probability of oxide separations due to thermal gradients.

It is desirable that burnable poison should not form any reaction product with the fuel material, fuel jacket material, or reactor coolant in the event of a rupture in the cladding material. Such reactions could cause swelling and subsequent damage to the reactor core.

The rare earth oxides are stable compounds with melting points in excess of 2000°C. At the sintering temperature of the fuel pellets, 1650°C,

the rare earth additions form a solid solution with the UO_2 , an optimum condition for promoting and maintaining homogeneity. There are no known reactions between Eu_2O_3 , Sm_2O_3 , and the Zircaloy-2 cladding material.

Water corrosion studies were conducted at ANL on compacts of 97 w/o UO_2 -3 w/o Eu_2O_3 , and 97 w/o UO_2 -3 w/o Sm_2O_3 . Specimens were tested in both oxygenated and deoxidized 360°C water at a pressure of 2750 psi for test times up to 738 hr. All specimens with rare earth additions showed a greater resistance to corrosion than did control specimens of UO_2 without the rare earth additions. It was concluded that lesser additions of rare earth oxides to UO_2 would certainly not adversely affect the corrosion resistance of the fuel in the event of accidental exposure to coolant water.

It is important that the burnable poison materials should impart a minimum of radiation damage to the fuel in the form of swelling or redistribution of poison due to thermal gradients. Unlike boron, the rare earth additions do not aggravate volume swelling tendencies of the fuel as they do not produce helium atoms.

Because the compositions of materials which form solid solutions are uniform and stable over a broad temperature range, it is not possible for thermal gradients which can occur in the fuel to cause migration of the burnable poison.

Although boron compounds have been successfully used as burnable poisons in the Shippingport and Army Package Power Reactor, there is no record of mixed Eu_2O_3 - Sm_2O_3 having been used as a burnable poison in UO_2 fuel. The experience gained during the course of the present experiment will supply valuable information concerning the use of these rare earths in a power reactor. This experience constitutes an important part of the experiment.

The outer elements were loaded with natural UO_2 pellets. These pellets, without additives, were compacted, fired, and finished by the same procedure as was used for the enriched pellets. The finished pellet dimensions and loading were also the same as for the enriched pellets.

Figure 5 shows details of the uranium fuel elements.

In addition to the fuel described above, it is possible that some of the spike elements from the previous 100 MW operation will be used in the shim zones. This allows considerable operational flexibility in adjusting system reactivity both initially and after irradiation.

Detailed records were maintained for all materials used for structural and fuel components, inspection of all structural components, and

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Fig. 5. Natural and Enriched UO_2 Fuel Element

for all fuel fabrication, testing, inspection, and loading. Each fuel rod and each fuel element have individual identification markings cross referenced with all fabrication and inspection records.

A record will be kept of the orientation of all fuel elements in the core. Tabs or notches on the fuel elements will be used for identifying this orientation. Tabs are used on the shim elements, and notches are used for the plutonium and natural uranium elements. The latter two are distinguished both by their different lengths and by the removable nature of the plutonium fuel pins. Any fuel element which is removed from the core can therefore be replaced in its original orientation.

C. Control Rods

After completing the 100 MWt experimental program but prior to unloading the fuel, all nine control rods were examined. One control rod at a time was disconnected from its drive mechanism and was lifted out of its guide shroud so that the poison section and its Zircaloy follower could be examined under water in the flooded reactor vessel. Eight rods were made of 2% boron stainless steel whereas the ninth or center rod was made of hafnium. Although the hafnium rod was used in conjunction with an oscillator for obtaining transfer function data and underwent about 100,000 cycles, it appeared to be in good condition. It was later transferred to the fuel storage pit and will be used in future oscillator tests.

Surfaces of all rods and followers were examined with a strong light and a mirror. A boroscope was used on questionable areas. All surfaces were covered with the usual reddish-brown iron oxide. The only defect noted during this examination was a missing rivet head. The rod was, therefore, subjected to a "rap & wedge" test and two more rivet heads broke loose.

Examination of the areas of failure in the rivets indicate that the areas had been subjected to intergranular cracking and appeared to have undergone severe mechanical distortion as evidenced by the removal of individual grains from their surfaces.

Because of the questionable condition of the 432 rivets used in nine rods, and that rods of greater worth were desirable for the new core, eight new enriched boron stainless steel rods and one hafnium rod were ordered. The hafnium rod will be a spare. The new design described below eliminates the use of rivets in the boron stainless steel and Zircaloy junction.

Control of the reactor is effected as in previous EBWR operation by nine cross-type control rods which can be withdrawn individually and inserted individually or simultaneously.

One control cross with hafnium for the absorber material is 0.317 cm (0.125 in.) thick x 25.4 cm (10 in.) wide with a 117 cm (46 in.) long control section. The hafnium is welded to a 98 cm ($38\frac{1}{2}$ in.) long Zircaloy-2 follower section. The remaining eight control crosses have boron stainless steel absorber sections which are 0.635 cm ($1/4$ in.) thick x 25.4 cm (10 in.) wide. The absorber sections are 152 cm (60 in.) long and are made of Type 304 stainless steel containing 2 w/o boron which is highly enriched, approximately 90 w/o, in the isotope B^{10} . The 87.6 cm ($34\frac{1}{2}$ in.) long follower sections are made of Zircaloy-2.

The absorber portion of the boron stainless steel rod is joined to the follower section by means of an assembly rod. A tongue-and-groove joint replaces the original riveted transition sections. Rivets in these transition sections have failed in service. Due to the differential thermal expansion between Zircaloy-2 ($7.63 \times 10^{-6}/^{\circ}\text{C}$), and stainless steel ($17.3 \times 10^{-6}/^{\circ}\text{C}$), sufficient strain can occur in the outermost rivets at the design operating temperature (0.010 cm at 250°C), to cause a shear type failure of the rivet shank. The tongue-and-groove joint eliminates the stress due to differential thermal expansion of the absorber and follower materials.

The control rod components are held in alignment when assembled by means of a central 1.27 cm ($1/2$ in.) square Type 304 stainless steel bar which is integral with the cross guide follower and extends throughout the length of the control rod. By removing the handling tip from the upper end of the control rod, the rod may be disassembled for ease of handling or replacement of either the absorber or follower section independently.

When in operation, the control rods are held in the shroud structure above the core (see Fig. 2). The upper portion of this shroud structure serves to position the upper ends of the fuel elements.

The control rods and followers are guided in a fuel and control rod guide structure consisting of a sheet metal assembly, 238.7 cm (94. in.) high, mounted on the bottom grid. It is made of 0.318 cm ($1/8$ in.) zirconium and stainless steel plates spaced with zirconium and stainless steel bars. This assembly provides nine cross-shaped channels for the Zircaloy-2 control rod followers and a number of cells for the guidance of the upper ends of fuel elements. 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 ten years.

Figure 6 shows details of the B-SS control rods and Fig. 7 gives details for the original and spare Hf control rods.

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Fig. 6. Five Foot Control Rod with Zircaloy-2 Follower

D. Control Rod Drive Mechanisms

1. Drive Speed

For the 100 MW operation of EBWR, the insertion and withdrawal drive speed for each of the nine control rods was fixed at 28 in./min. This rate of motion when expressed in units of dollars per second of reactivity change will have values that vary according to which control rod is being considered, the particular loading configuration in which the rod is involved, and finally the current value of β_{eff} , the effective delayed neutron fraction.

When comparing the Pu recycle experiment with the previous EBWR operation with regard to determining an acceptable control rod drive speed, the most meaningful quantities for comparison are the values of β_{eff} in the two cases. On the other hand, if one attempts to make comparisons based on relative control rod worths, the problem becomes ill defined owing to the wide variations encountered during the previous operation and expected for the proposed experiment.

As indicated in Sections III.G.2e, 3e, and 4c, for the fully loaded system, β_{eff} varies from an initial value of about 0.005 to over 0.006 as the irradiation proceeds. On the other hand, for the Pu critical configuration, β_{eff} has a calculated value of about 0.003. By comparison, at the end of the 100 MW operation, β_{eff} had the value 0.0054.⁷

In spite of the similarity of β_{eff} for the present experiment and previous operation, it has been decided to reduce the drive speed for each rod in the case of the full system to about 14.5 in./min. Maximum rates of reactivity variation due to control rod motions will, therefore, be only about 2/3 of those encountered previously during the 100 MW operation and, as such, represent conservatism for this experiment.

In the case of the plutonium critical configuration, the small value of β_{eff} and large central rod worth (see Section III.G.4) indicate the desirability of further reducing the central rod drive speed.

A drive speed of about 4.3 in./min will be used for the central rod during the plutonium zone critical experiments. This speed results in a reactivity insertion rate of about 80% of that encountered in the full core loading, and about half of that experienced during the 100 MW operation.

2. Materials

Prior to the 100 MW operations of the EBWR reactor, the control rod drive mechanism racks, pinions, and seal shafts were changed from 17-4 PH H-900° stainless steel construction to 17-4 PH H-1100° stainless steel. This change was accomplished to eliminate susceptibility to

transverse cracking which was reported in the nuclear industry in other cases where 17-4 PH H-900° stainless steel was being used. Upon disassembly of the drive mechanism subsequent to completion of the 100 MW program, these 17-4 PH H-1100° racks, pinions, and seal shafts were non-destructively examined for integrity by the magnaglow process and by ultrasonic methods. These examinations revealed no indications of any kind of impending failure after 74,620 MWh of operation. It is concluded, therefore, that the 1100° heat-treated components are adequate for the control rod drive application and will be re-used for the plutonium recycle core.

3. Oscillator Drive Unit

In order to conduct certain physics and stability experiments, an oscillator drive unit will be connected to the hafnium control rod to provide a periodically varying reactivity input. The oscillating equipment and connections will be those used in previous EBWR transfer function experiments. As in the past, the control rod connected to the oscillator drive unit will not be available for rapid insertion nor for simultaneous insertion with the remaining eight rods at the normal drive unit speed. When the transfer function experiments are completed, the hafnium rod will remain in the reactor as a normal control rod, and will be operated by a drive mechanism identical to each of the other control rods.

E. Fuel Storage

The entire core and source were removed from the reactor vessel following the 100 MW operation and were placed in the West and South storage racks located in the fuel storage pit. There were no fuel failures during the operating history of EBWR and, therefore, the handling and storage of a failed element has never been encountered.

At the present time, the West rack contains the 32 fully enriched spike fuel elements used to spike the original core and the South rack contains the plate type elements from the 100 MW operation. The plate type elements will be removed from the present South rack before startup of the experiment and this rack will be replaced with a new rack. The new rack will consist of a rectangular grouping of 143, $4\frac{1}{16}$ in. square boxes formed by interlocking $1\frac{1}{4}$ in. thick sheets of nominal 1.2 w/o boron stainless steel. Conservative calculations indicate that the new South rack will provide more than a 15% shutdown margin for the most reactive configuration during the course of the experiment.

F. High Pressure Boric Acid Injection System

The high pressure boric acid injection system will be retained and at the normal reactor operating pressure of 600 psi is capable of increasing

the reactor boric acid concentration by about 9.35 gm of H_3BO_3 per gallon of water at the hot operating no void density ($\rho = 0.794$). This results from mixing the 41.4 kgm of H_3BO_3 in 210 gal of room temperature water, with 4160 gal of hot water in the reactor.⁸ The operation of this system produces a δk of about -7.4% for the initial system and about -7.2% for the system after a burnup of 0.006. (See Sections III.G.2d and 3d.) Assuming no control rods are inserted and the system is at operating temperature, the following conditions would exist. With a 3% reactivity contribution due to equilibrium Xe, the system at 0.006 burnup will be about 1% subcritical. The initial system will have a k_{ex} of about 0.3% and hence will continue to operate at a low power even after injection of the boric acid. If no Xe is present even the system at 0.006 burnup will require additional boric acid to completely stop power generation. Assuming a boric acid addition rate of 520 gm/min from the control boric acid system (see Section IV.C), the order of 40 min would be required to completely stop power generation for the reactor without Xe. With Xe, the initial system could be completely shutdown in about 4 min.

The inability to stop power generation completely using only the high pressure boric acid injection system was not considered to be a serious drawback during previous operation of EBWR,³ nor is it so considered for the proposed experiment.

G. Physics

1. Calculational Model

Multigroup cross sections have been generated using the GAM-1¹⁴ and SOFOCATE¹⁵ codes. Resonance parameters for Pu^{240} and U^{235} have been inserted into the GAM-1 library so that resonance self-shielding is calculated directly for Pu^{240} , U^{238} , and U^{235} . The GAM-1 results for Pu^{239} have been modified using self-shielding factors derived using the RIC code.¹⁶ An atom density of 0.022×10^{24} has been used for all of the Pu and U oxides.

Results obtained from SOFOCATE have been corrected using thermal flux depression factors which were derived from one-group DSN¹⁷ results. For the DSN calculations involving cylindrical symmetry, a thick pure scattering region was added as the outermost zone as suggested by Honeck.¹⁸

Four, seven, and eight group cross section sets have been used for the various calculations. The energy limits for the cross section sets are given in Tables 2-4.

TABLE 2 Group Structure of 4 Group Cross Section Set

Group	Lower Energy, eV
1	5.53×10^3
2	1.44
3	0.532
4	Thermal

TABLE 3 Group Structure of 7 Group Cross Section Set

Group	Lower Energy, eV
1	1.83×10^5
2	5.53×10^3
3	275
4	6.48
5	1.44
6	0.532
7	Thermal

TABLE 4 Group Structure of 8 Group Cross Section Set

Group	Lower Energy, eV
1	1.35×10^6
2	498×10^3
3	302×10^3
4	183×10^3
5	5.53×10^3
6	1.44
7	0.532
8	Thermal

All materials other than the europium and samarium isotopes to be used as burnable poisons in the shim zones, were derived in a spectrum appropriate to the plutonium region, which for the hot cross sections was assumed to have an average void fraction of 0.3. The burnable poisons were derived in a spectrum appropriate to the shim region and again for the hot cross sections, for a void fraction of 0.3. Cross sections were derived for both 0 burnup and 0.006 burnup in the plutonium zone. Intermediate burnup values and values for burnups greater than 0.006 were obtained by linear interpolation or extrapolation.

One-dimensional seven, eight, and four group diffusion theory calculations were made using the REX code which is an IBM-704 version of the code written for the UNIVAC.¹⁹ Seven group burnup calculations were performed using the CYCLE code²⁰ which relies on REX for multigroup flux information and for reactivity evaluation.

The PDQ codes^{21,22} and the 20 GRAND code²³ were used for the two-dimensional diffusion theory calculations using the 4 group cross sections. PDQ-2 was used in RZ geometry in conjunction with the hydrodynamics code CHOPPED²⁴ to obtain regional void distributions as a function of power and burnup. The predicted operating power is based on these results. (See Section III.H.)

Control rod worths were obtained using PDQ-3²² and a logarithmic boundary condition for the thermal group such that for the thermal group, $(D d\phi/dx)/\phi = -C$. Here D is the diffusion coefficient of the medium bordering the rod. As in the physics calculations for the EBWR core 1A,⁴ C was set equal to 0.46948. The control rod energy and spatial self-shielding for the epithermal groups was evaluated using the DTK code²⁵ in slab geometry. The worth of the central Hf rod has been calculated assuming it to be 2% natural boron in stainless steel.²⁶

In order to fit the elaborate isotopic chains²⁷ of europium and samarium into the CYCLE code, an abbreviation of these chains as well as the U-Pu chain was necessary. The simplified chains considered in the burnup calculations are listed in Fig. 8. The microscopic and macroscopic

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Fig. 8. Simplified Neutron Burnup Chains Used for CYCLE Calculations

cross sections have been modified for the burnable poison so as to attempt to maintain approximately the same macroscopic absorption cross section as for the detailed chains.

The B188 RP code²⁸ was used with eight group real and adjoint fluxes determined by the REX code¹⁹ to calculate effective delayed neutron fractions and prompt neutron life-times. These parameters were used in the space independent, one energy kinetics codes A130 RP²⁹ and A129 RP³⁰ to determine the system asymptotic period vs. reactivity, and to examine the influence of the Doppler reactivity coefficient on total energy release for various super-prompt critical reactivity insertions.

The transient temperature distribution

in a typical fuel pin and Zr clad during such prompt critical events was calculated using the ARGUS code.³¹

For the calculation of β_{eff} , thermal fission delayed neutron decay constants and group yields were used for Pu^{239} and U^{235} and fast fission values were used for U^{238} and Pu^{240} (Ref. 32). The detailed spectra for the delayed neutron groups was taken from the work of Batchelor and McK. Hyder.³³

The group reciprocal velocity values used in the calculation of the prompt neutron lifetime were derived, except for the highest energy group and thermal group, assuming the flux to vary as $1/E$. The thermal fission spectrum of Pu^{239} (Ref. 34) was assumed for the highest energy group. For the thermal group, using the SOFOCATE derived value for "UNIT A" (the cross section of a hypothetical material having a pure $1/v$ behaviour and a 1 barn cross section at an energy of 0.0253 eV) we have $(1/v)_{thermal} = 100 \times \text{"UNIT A"}/2208.8$.

2. Initial Configuration

a. Loading

The initial configuration tabulated in Table 1 (Section II) is discussed here for convenience.

The average fuel, zirconium, and water volume fractions for the unrodded core are respectively 0.21663, 0.18854, and 0.59483. For

control rod calculations, since the rod regions are treated separately, the corresponding volume fractions are 0.23468, 0.16301, and 0.60231 for fuel, Zr, and H₂O. All oxides are assumed to have an atomic density n (in units of 10^{24} atoms/cc) of 0.022 and zirconium is taken to have an atomic density of 0.0425. At room temperature, the n for water is used as 0.0334 and the operating no void density is assumed to be 0.788. For the moderator, room temperature is assumed to be 293°K and operating temperature is taken to be 527°K. Operating fuel temperatures are taken to be 1473°K.

Table 5 lists the regional volumes and equivalent cylindrical radii used for the one-dimensional calculations.

TABLE 5. Regional Dimensions

Region	No. Elements	Regional Volume, liters	Equivalent Cylindrical Radius, cm
Plutonium	36	511.68	36.55
1st Shim	28	397.85	48.73
2nd Shim	32	455.14	59.69
Natural U	52	738.44	74.1
Outer Reflector	-	2047.6	104.1

Table 6 lists the regional enrichments and shim zone burnable poison contents.

TABLE 6. Regional Enrichments

Region	Enrichment	Weight Percent	
		Eu ₂ O ₃	Sm ₂ O ₃
Plutonium	0.015 ^a	-	-
1st Shim	0.06 ^b	0.158	0.0288
2nd Shim	0.06 ^b	0.158	0.0288
Natural U	0.0072 ^b	-	-

^aPlutonium in depleted uranium; of this, 8% is Pu²⁴⁰.

^bU²³⁵ in U²³⁸.

b. Void Distribution

The initial configuration was used in a series of PDQ-2--CHOPPED iterations to obtain regional voids vs. total system power (see Section III.H). For these calculations, the 4 ft high core was axially divided into three zones of equal height. The calculated voids vs. power in the various radial zones and for the three axial parts of each zone together with the top reflector are shown in Fig. 9. Figure 9 also shows the regional

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Fig. 9. Regional and Average Voids at Various Reactor Powers for the Initial System

average voids for use in one-dimensional calculations. The relative regional average voids are not very sensitive to power over the range shown. The predicted initial hot operating power is about 40 MW. The average voids in the Pu, 1st Shim, 2nd Shim, and Natural U zones are respectively in the ratios 1.0/1.0/0.9/0.3 corresponding to an average core void of about 0.21. The regional power fractions under these condition are respectively 0.36, 0.29, 0.26, and 0.09.

c. Rod Worth and Shutdown Margin

The rod worths for various configurations are listed in Table 7 for the initial reactor at room temperature.

TABLE 7. Initial Cold Rod Worths

Configuration	Worth, %
All rods	17
Central rod, remaining rods in	4.8
Central rod, remaining rods out	2.2
Off center, corner rod, remaining rods in	6.0
Off center, non-corner rod, remaining rods in	6.4

The cold, rodded k_{eff} is calculated to be 0.98. For eight rod shutdown one therefore requires a boric acid content of about 4.0 gm/gal corresponding to a δk of $\sim -1.1\%$ per gm/gal (see Section III.G.2d).

d. Reactivity Effects

Table 8 summarizes the initial reactivity behaviour as a function of water density and temperature.

TABLE 8. Zero Void Reactivity Summary

Moderator Temperature, °K	Water Density	k_{eff}
293	1	1.15
293	0.788	1.12
527	0.788	1.11
527	0.788	1.08 ^a

^aAllowing 3% for Xe.

At the hot operating condition, the void coefficient of reactivity $\delta k/\delta V_{Pu} \approx -0.33$ where δV_{Pu} refers to changes in the Pu zone void with relative changes in the other zones so as to maintain the initial void distribution. In terms of average core void, the coefficient is $\delta k/\delta V_{avg} \approx -0.45$ which corresponds to a power coefficient of reactivity $\delta k/\delta P \approx -0.001 \text{ MW}^{-1}$.

At room temperature, the void coefficient of reactivity $\delta k/\delta V \approx -0.124$. This value decreases as boric acid is added to the water but is still negative for additions of over 10 gm of boric acid per gallon of room temperature core water. Table 9 gives $\delta k/\delta V$ for various boric acid contents.

TABLE 9. Cold Void Coefficients of Reactivity for Various Boric Acid Contents

Boric Acid Density, gm/gal (cold)	$\delta k/\delta V$
0	-0.10
1.0	-0.094
5.0	-0.059
10.0	-0.024

The boric acid worth for the cold, initial system is about -1.1% in δk for an addition of 1 gm/gal of core water to the unpoisoned system and about -0.9% for an addition of 1 gm/gal for the system which contains 9 gm/gal. The boric acid worth decreases as the water density decreases, owing to hardening of the spectrum. At the hot operating temperature, but with no voids (water density $\rho = 0.788$), the boric acid worth is about -0.68% for an addition of 1 gm/gal of equivalent room temperature density water ($\rho = 1$). Owing to non-linearity of boric acid worth with concentration, an addition of 12 gm/gal of unit density water produces a δk of about -7.6% corresponding to an effective worth per gm/gal over this range of about -0.63%.

The initial reactivity behaviour and cold rod worths are summarized in Fig. 10. In Fig. 10 the Pu, 1st Shim, 2nd Shim, and U zone void fractions were in the ratios 1.0/1.0/0.9/0.3.

e. Prompt Neutron Lifetime and Effective Delay Neutron Fraction

Table 10 lists β_{eff} and prompt neutron lifetime ℓ for the initial cold unrodded system made critical using boric acid, and for the hot operating system.

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Fig. 10. Initial Reactivity Behaviour and Cold Rod Worths for Unpoisoned System

TABLE 10. β_{eff} and l for the Initial System

Condition	β_{eff}	$l, 10^{-6} \text{ sec}$
Cold, poisoned, unrodded	0.0053	25.5
Hot operating	0.0049	30.9

These values of β_{eff} can be compared with the value of about 0.0054 at the time of shutdown of EBWR.⁷

f. Fuel Element Worth

The worths of a central Pu fuel element and of a fuel element at the center of one side of the first shim row have been determined for the cold, initial configuration. These elements are worth respectively about 0.6 and 0.3% relative to core water. For the 100 MW operation, by comparison, the maximum δk due to inserting a single element was calculated to be about 2%.³ The low worth of the present fuel elements is consistent with the large void coefficient of reactivity exhibited by the present undermoderated loading.

3. Irradiation Behaviour

a. Reactivity vs. Burnup

The CYCLE code was used to calculate regional compositions and system reactivity as a function of burnup in the plutonium zone. The regional voids for the hot operating case were taken to be 0.3, 0.3, 0.3, and 0.15 respectively for the Pu, 1st Shim, 2nd Shim, and natural uranium zones. Variation of cross sections with burnup were taken into account by assuming a linear behaviour with Pu zone burnup.

Figure 11 shows the system k_{eff} vs. Pu zone burnup for the hot operating system and the cold unrodded system. The contribution

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Fig. 11. Reactivity Variation with Burnup

of fission product Sm buildup has been included. The buildup of Xe has not been included but rather a constant 3% reactivity loss has been assigned to Xe. The isotopic compositions derived from the CYCLE calculations have been used to study reactivity effects and rod worths as a function of burnup.

The influence of the assumed voids on the predicted plutonium zone isotopic variation was observed by rerunning the hot operating CYCLE calculation assuming the same average voids as above but with relative voids corresponding to the distribution expected at a Pu zone

burnup of 0.006. Although the reactivity variation with burnup was somewhat greater than that shown in Fig. 11 owing to a larger rate of burnout of the shim burnable poisons, the Pu zone isotopic composition changed by less than 2% at 0.006 burnup compared with the original calculation.

b. Void Distribution at 0.006 Burnup

The compositions at 0.006 Pu zone burnup were used in a series of PDQ-2--CHOPPED iterations to obtain regional voids vs. total system power as for the initial system. Figure 12 shows voids vs. power corresponding to the initial Fig. 9. Figure 12 also gives the regional average voids. As for the initial configuration, the relative regional average voids are not very sensitive to power over the range shown. The predicted hot operating power for 0.006 Pu zone burnup is about 30 MW. The average voids in the Pu, 1st Shim, 2nd Shim, and natural U zones are respectively in the ratios 1.0/1.4/1.3/0.55 corresponding to an average core void of about 0.19. The regional power fractions under these conditions are respectively 0.22, 0.32, 0.33, and 0.13.

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Fig. 12. Regional and Average Voids at Various Reactor Powers for the System at 0.006 Pu Zone Burnup

c. Rod Worth and Shutdown Margin at Various Burnups

Table 11 lists the rod worths for various configurations and burnups at room temperature. The 0 burnup values from Table 7 are repeated here for convenience.

TABLE 11. Cold Rod Worths at Various Burnups

Configuration	Pu Zone Burnup	Worth, %
All rods	0	17
	0.003	15
	0.006	14
Central rod, remaining rods in	0	4.8
	0.003	0.4
	0.006	0.2
Central rod, remaining rods out	0	2.2
	0.003	0.6
	0.006	0.3
Off center corner rod, remaining rods in	0	6.0
	0.006	7.7
Off center, non- corner rod, remaining rods in	0	6.4
	0.006	6.0

The cold rodded k_{eff} at Pu zone burnups of 0.003 and 0.006 are respectively 0.996 and 0.999. This decreased shutdown margin with burnup is due to the loss of central rod worth which is partially compensated for by the increasing worth of the corner rods. Since the rodded k_{eff} is essentially unity at 0.006 burnup, one requires a boric acid content of about 9 gm/gal for eight rod shutdown corresponding to a $\delta k \sim -0.8\%$ per gm/gal (see Section III.G.3d).

d. Reactivity Effects at 0.006 Burnup

Table 12 lists the reactivity behaviour as a function of water density and temperature for a Pu zone burnup of 0.006.

At the hot operating condition, the void coefficient of reactivity $\delta k / \delta V_{\text{Pu}} \approx -0.38$ where δV_{Pu} refers to changes in the Pu zone void with relative changes in the other zones so as to maintain the relative void distributions. In terms of the average core void, this corresponds to $\delta k / \delta V_{\text{avg}} \approx -0.39$ and in terms of reactor power, to $\delta k / \delta P \approx -0.0013 \text{ MW}^{-1}$.

TABLE 12. Zero Void Reactivity Summary at 0.006 Pu Zone Burnup

Moderator Temperature, °K	Water Density	k_{eff}
293	1.0	1.14
293	0.788	1.10
527	0.788	1.09
527	0.788	1.06 ^a

^aAllowing 3% for Xe.

At room temperature, the void coefficient of reactivity $\delta k/\delta V \approx -0.16$. Table 13 shows the influence of boric acid on the void coefficient of reactivity for the system at a Pu zone burnup of 0.006.

TABLE 13. Cold Void Coefficients of Reactivity for Various Boric Acid Concentrations and a Pu Zone Burnup of 0.006

Boric Acid Density, gm/gal (cold)	$\delta k/\delta V$
0	-0.16
1	-0.16
5	-0.13
10	-0.026

As in the case of the initial system, the reactivity coefficient though appreciably smaller than for the unpoisoned case is still negative for boric acid contents over 10 gm/gal of room temperature core water.

The boric acid worth is about -1% in δk for an addition of 1 gm/gal to the cold unpoisoned system for a Pu zone burnup of 0.006. An addition of 1 gm/gal to the system which contains 9 gm/gal results in a δk of about -0.7%. This addition would for the hot system, correspond to a worth of about -0.6% per gm/gal of unit density water.

The reactivity behaviour and rod worths at 0.006 Pu zone burnup are summarized in Fig. 13. In Fig. 13 the Pu, 1st Shim, 2nd Shim, and U zone void fractions were in the ratios 1.0/1.4/1.3/0.55.

e. Prompt Neutron Lifetime and Effective Delayed Neutron Fraction

Table 14 gives β_{eff} and ℓ for the system at room temperature for a Pu zone burnup of 0.006. The system is unrodded and contains sufficient boric acid to be critical.

Fig. 13

Reactivity Behaviour and Cold Rod Worths at 0.006 Pu Zone Burnup for Unpoisoned System

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TABLE 14. β_{eff} and ℓ for the Cold System at 0.006 Pu Zone Burnup

Condition	β_{eff}	$\ell, 10^{-6} \text{ sec}$
Cold, Poisoned, Unrodded	0.0066	26.1

The increase of β_{eff} with burnup (see Table 10) is due to the shift in power from the Pu zone out into the enriched U shim zones. This shift is indicated quantitatively by the relative power fractions and void distributions given above in sections 2b and 3b. This behaviour is of course opposite from that of EBWR where β_{eff} varied from an initial value of about 0.82 to about 0.54% (Ref. 3) at the end of the 100 MW operation.⁷

4. Clean Cold Plutonium Critical Characteristics

a. Reactivity vs. Burnup

Reactivity as a function of the composition determined by CYCLE, has been evaluated for the cold water reflected plutonium zone using the REX code. Figure 14 shows the results. Also shown on Fig. 14 are the effect of density and temperature changes corresponding to 325°F for the unpoisoned case. Figure 14 indicates that for burnups greater than about 0.003, some reactivity will be needed above that supplied by the Pu zone itself. This additional driver if required would consist of some part of the shim zone. It will be desirable of course to maintain as simple a configuration as possible to facilitate comparison of calculation with experimental results.

b. Rod Worth and Shutdown Margin

Table 15 gives the rod worths for the initial, cold, unpoisoned Pu critical configuration.

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Fig. 14. Reactivity Variation with Burnup for the Pu Critical Configuration

TABLE 15. Rod Worths for Cold Initial Unpoisoned Pu Critical Configuration

Condition	Worth, %
All rods	16
Central Rod, remaining rods in	9.7

The cold, rodded k_{eff} for the initial Pu critical configuration is 0.93. Eight rod shutdown would therefore require about 2.0 gm/gal of boric acid corresponding to a $\delta k \sim -1.3\%$ per gm/gal. (See Section III.G.4d.)

c. Prompt Neutron Lifetime and Effective Delayed Neutron Fraction

Table 16 lists β_{eff} and ℓ for the initial Pu critical configuration at room temperature. As before, the system is unrodded and contains sufficient boric acid to be critical.

TABLE 16. β_{eff} and ℓ for the Initial Cold Pu Critical Configuration

Condition	β_{eff}	$\ell, \times 10^{-6} \text{ sec}$
Cold, poisoned	0.00307	41.7

The small value of β_{eff} is expected due to the system being fueled only with Pu^{239} as the fissile species. The increased prompt neutron lifetime as compared with the fully loaded system results from replacing the heavily absorbing shim and natural uranium zones with the water reflector.

d. Reactivity Effects

The cold, void coefficient of reactivity for the initial Pu critical configuration is $\delta k / \delta V \approx -0.18$ for the unpoisoned system, and

$\delta k/\delta V \approx -0.11$ for the system containing about 6 gm of boric acid per gallon of room temperature core water, which corresponds to the initial clean critical condition. The boric acid worth is about -1.3% in δk for a 1 gm/gal addition to the unpoisoned system and about -1% for a 1 gm/gal addition to the system which contains 9 gm/gal.

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Fig. 15. Critical Number of Pu Fuel Elements as a Function of Boric Acid Content for the Initial Pu Configuration

e. Fuel Element Worth

A central plutonium fuel element in the initial, cold, Pu critical configuration has a calculated worth of about 2% relative to the core water. The other elements are worth less. For example, an element in the center of a quadrant (see Fig. 1) is worth about 0.8% relative to core water, while a corner element is worth only about 0.2%. The average worth of an element in the Pu critical configuration is less than 1% relative to core water.

Figure 15 shows the calculated critical number of plutonium fuel elements as a function of boric acid content for the initial plutonium critical configuration.

5. Compensation of Super-prompt Critical Reactivity Insertions by the Doppler Effect

a. Calculational Procedures

The Doppler broadening of resonances in U^{238} , Pu^{240} , and Pu^{239} is the only shutdown mechanism required to predict the prompt excursion behaviour of the present core. This was found also for recent SPERT-1 experiments on oxide rod cores.³⁵ No credit has been taken for shutdown due to expansion effects owing to uncertainties of the behaviour of oxide fuels, particularly after irradiation and operation at elevated temperatures. Since there are no pressure effects from rapid boiling^{35,36} which might damage the core, the determining factor with regard to the core damage was assumed to be melting of the oxide fuel (2750°C) at some point in the core. This criterion for core damage is actually conservative in that the temperature distribution in the individual fuel pin is not being considered. Thus, even though melting does occur over some part of the oxide fuel pin volume, as long as the outer volume of the oxide is solid, the event is presumably not damaging.

The first step in the calculation was to obtain the cross section changes due to elevation of the fuel temperature. A series of GAM-I (Ref. 14) and RIC (Ref. 16) calculations were run for fuel temperatures ranging from 293 to 2293°K with the moderator held at room temperature. The cross section set group structure was that given in Table 2, and cross section changes at the various fuel temperatures were obtained for group 2 for U^{238} and Pu^{239} , and for group 3 for Pu^{240} .

The next step in the calculation was to determine the spatial temperature distributions in the core for a variety of conditions and energy inputs to the system. For an assumed spatial power distribution and under the assumption that no heat is lost from the fuel, the spatial temperatures are simply related to energy inputs using tabulated values for the specific heat of UO_2 as a function of temperature.^{37,38}

Negative reactivity effects due to various energy inputs were then obtained by running diffusion theory calculations having spatially dependent cross sections appropriate to the temperature distribution corresponding to the particular energy input and assumed power distribution. The reactivity coefficients so obtained give the reactivity feedback due to the Doppler effect as a function of energy input and may be used directly in the A129 RP code³⁰ which in turn calculates integrated energies as a function of various reactivity insertions. Finally then, again for the assumed power distribution and using the specific heat for UO_2 , the spatial temperatures resulting from various reactivity insertions are obtained.

It should be noted that the above procedure represents a significant refinement over earlier calculations^{1,3} which were based upon space independent calculations of the temperature coefficient due to the Doppler effect assuming a uniform temperature for all of the fuel. In particular, the variation of the Doppler coefficient with temperature was approximated by a single, temperature independent value. The only attempt to include spatial effects for this original work was through the device of including an assumed maximum to average factor of 1.6 for the power peaking.

In the present calculations, since the reactivity effects were evaluated using either the one-dimensional REX code¹⁹ or the XY-geometry PDQ code,²² the effects of axial power shapes are taken into account in a conservative manner. That is, the power distributions are based on an axial average power value, whereas in fact the axial power and temperature distributions will have maxima of perhaps 1.4 times as high as the average value. Since these maxima occur in regions of high importance, the negative reactivity feedback being calculated for a given energy input is conservative and therefore the energy release calculated for a given reactivity insertion is overestimated.

Two control rod configurations have been considered with regard to the assumed power distributions. The first is for the case of the cold, unrodded core. This configuration was chosen mainly for orientation purposes since it corresponds most closely to the original simple approach.³ It is, however, a very unrealistic configuration with regard to serious reactivity additions. That is, for this case, one is calculating the consequences of large, rapid, reactivity insertions for a configuration in which a mechanism for such insertions cannot be readily invented.

These unrodded cases were studied using the REX code.¹⁹ The plutonium zone and first shim zone were each split into two regions while the second shim and natural uranium zones were treated as one region each. Each of these six regions contained cross sections modified according to the temperature expected in the region concerned. Power distributions were used as appropriate to the initial system, and to the system after a plutonium zone burnup of 0.006. The initial unrodded plutonium critical configuration was also considered. In this case, the plutonium zone was divided into four regions, each having a temperature and thus cross sections appropriate to the energy input being considered.

A realistic control rod configuration with regard to a serious reactivity insertion is that for which one rod is out and the remaining eight rods are in. Such a situation was in fact previously considered with regard to a maximum credible accident in EBWR.³⁹ In this configuration, there is a very marked change in the power distribution so that Doppler results obtained for the unrodded configuration are not at all applicable. In particular, about 70% of the power is contained in the quadrant with the corner rod out.

This configuration was studied using the PDQ-3 code²² and taking a corner rod to be the only rod not in the core. For these calculations, the azimuthal temperature variation was approximated by dividing the core into the quadrant without the corner rod, and the rest of the reactor. In addition, as before, the plutonium and first shim zones were divided into two regions while the second shim and natural uranium zones were not divided. In all, eleven different regions were considered with regard to particular cross section and temperature assignments.

A second criterion in regard to core damage concerns the integrity of the zirconium cladding both with regard to melting and with regard to rupture due to excessive thermal stress during a transient event. The transient heat transfer characteristics of the oxide fuel for the small periods being considered are such^{37,40} that the question of clad integrity is probably not a major consideration with regard to core damage for events which are limited to those which just lead to melting of the oxide fuel.

In order to examine this point quantitatively, the ARGUS code³¹ was used to calculate the transient temperature distributions in the fuel, clad, and water for some of the power excursions described above to determine the thermal stresses and temperatures in the zirconium clad. The DTK code²⁵ was used to determine the radial power distribution in the cylindrical fuel pin which is needed as input to ARGUS.

Some of the results obtained, which are in qualitative agreement with those reported for SPERT-I³⁷ are given in Section III.G.5.b.

b. Results

The methods outlined in Section III.G.5.a were used to determine the step reactivity insertions which would just lead to melting of the oxide fuel for the cold initial system and the cold system at a 0.006 plutonium zone burnup, both for the unrodded system, and for the case of all rods except one corner rod fully inserted. The unrodded, cold initial plutonium critical was also studied. The results are summarized in Table 17 which lists the calculated values for k_{ex} and the corresponding integrated energy releases and reactor periods for the various configurations. The specific heat of UO_2 is such^{37,38} that the melting point will be reached for an energy density of 8.22 kW-sec per cc. The values given in Table 17 for total energy release all correspond to this energy density at the hottest point of the particular system being considered. As discussed in Section III.G.5.a, these calculated values are conservative.

TABLE 17. Step Reactivity Insertions Leading to Fuel Melting

System	Step k_{ex} , %	Integrated Energy Release, MW-sec	Reactor Period, sec
Cold, 0 Burnup, Unrodded	1.6	1000	2.4×10^{-3}
Cold, 0.006 Burnup, Unrodded	1.5	1100	3.0×10^{-3}
Cold, 0 Burnup, Only corner rod not inserted	0.95	190	6.2×10^{-3}
Cold, 0.006 Burnup, Only corner rod not inserted	1.1	170	5.8×10^{-3}
Cold, 0 Burnup, Unrodded Pu Critical Configuration	1.5	390	3.3×10^{-3}

The k_{ex} of 1.6% for the initial, unrodded system corresponds to about 1.07% prompt excess reactivity. This value may be compared with the previously quoted limit for EBWR³ of 0.275% prompt excess reactivity for a step insertion event which would just melt the fuel. The rodded results, which are the values based on a realistic configuration with regard to a serious accident, allow a prompt k_{ex} of over 0.4% before fuel melting occurs.

The reactor periods given in Table 17 were obtained using the A130 RP code²⁹ with delayed neutron parameters and prompt neutron lifetimes determined from the B188 RP code²⁸.

Some typical results from the ARGUS studies of the transient temperature distributions in the fuel pins during an excursion are shown in Fig. 16.

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Fig. 16 Transient Temperature Distributions for a Step Insertion of 0.94% k_{ex} and for the Cold Initial System with Only One Corner Rod Withdrawn

The event shown there is for the cold initial system with all rods except one corner rod inserted, and for a step insertion of 0.94% k_{ex} . This event carries the center of the fuel pin to just below the oxide melting point. The temperature distributions at three moments in time are displayed: 0.145 sec corresponding to the peak of the burst, 0.180 sec corresponding to the time of maximum temperature difference across the clad, and 0.295 sec at which time the integrated energy release is essentially constant in time. The typically flat radial temperature distribution in the fuel^{37,40} is evident, as is the fact that the outer clad temperature at the time of the peak of the burst has increased by only a few degrees above the initial 20°C.

The maximum difference in temperature across the clad is about 210°C for this event. If one assumes a Young's modulus of 11.5×10^6 psi, a linear coefficient of expansion of $7.6 \times 10^{-6}/^\circ\text{C}$ and a Poisson's

ratio of 0.34 for Zircaloy-2, the above temperature difference would result in a calculated stress of about 14,500 psi.⁴¹ This may be compared with the short term tensile strength of about 35,000 psi and with the elastic limit of about 18,000 psi for Zircaloy-2.

A more severe event, corresponding to a step insertion of 1.0% k_{ex} carried the center of the oxide fuel to several hundred degrees above the melting point, although the outer surface of the oxide did not melt. For this more serious event, the maximum temperature difference across the clad was about 270°C corresponding to a maximum thermal stress of about 18,700 psi.

These examples lend support to the statement that the reactivity insertions given in Table 17 are actually quite conservative as far as damage to the core is concerned and indicate that the present oxide core greatly exceeds the previous EBWR metal fueled system with regard to its ability to sustain, without permanent damage, severe reactivity insertions. This result was of course expected from the SPERT-1 experience.³⁵⁻³⁷

H. Heat Transfer and Hydrodynamics

Past experience with EBWR has shown that the hydrodynamic characteristics are adequate to permit sustained operation at powers well above 40 MW. The plutonium recycle core is composed entirely of rod-type fuel pins, which are geometrically very similar to those used in the "spike" region of the Core IA loading during operation to 100 MW. Since the average power density in the new core will be lower than that in Core IA, it is to be expected that these elements will function without hydrodynamic difficulty.

Still unresolved in the Core IA experimental results is the disparity between the predicted recirculation flow rate at 40 MW ($\sim 10^7$ lb/hr) and the measured rate ($\sim 7 \times 10^6$ lb/hr). It was conjectured that perhaps the scale which accumulated on the plate-type elements during early Core IA operation had roughened the surface so that the core friction loss was substantially increased at power levels greater than 20 MW; however, recent pressure loss experiments using these irradiated elements have shown no change in frictional characteristics when compared to the original, clean condition. A likely explanation is the reduction in the natural circulation driving head due to a high void fraction in the downcomer resulting from carryunder of steam bubbles from the riser, although measurements of void fraction in the lower downcomer did not reveal this condition.

Although the power distribution and the average hydraulic equivalent diameter in the new core are somewhat different from the values found in Core IA, it is expected that the recirculation flow rate will be quite similar

to that measured in Core IA at 42 MW. Because of this and the aforementioned observations, some experimental values from Core IA operation⁴² have been incorporated in the following calculations, in order to predict more accurately the actual operation of the plutonium recycle core.

A number of power-void iterations were carried out using two codes programmed for the IBM-704 computer--the PDQ neutron diffusion theory code and the CHOPPED hydrodynamic program. The CHOPPED code is a revision of RECHOP, previously used extensively for EBWR calculations. The new code provides a faster iteration procedure and it was possible to produce relationships between local average void fraction and power level at a given burnup, as shown in Sections III.G.2.b and III.G.3.b, Figs. 9 and 12.

The curves shown in Figs. 17 through 21 indicate the distributions for an initial power level of 42 MW. Figures 17 and 18 show the radial and axial power density distributions, respectively. Note that the highest average power density and the peak power density occur in the first shim zone. Figure 19 indicates a fairly uniform inlet velocity distribution, even though the natural uranium zone has a much lower power density than the other core zones. This occurs because the outer zone receives a large share of the driving head in the upper riser, while the friction loss is low because of the low void fraction in the zone. Figures 20 and 21 present, respectively, the radial exit void fraction variation and the axial behaviour of the void fraction at the average power density shell in each zone.

Table 18 presents the most pertinent heat transfer characteristics of the core. The maximum heat flux, 191,000 Btu/hr-ft², which occurs at the innermost edge of the first shim zone at a height of about 1.1 ft ($L^+ = 0.285$), is well below the predicted burnout flux of approximately 800,000 Btu/hr-ft². All power densities are expressed per liter of coolant contained within the fuel element boxes.

Fig. 17

Radial Power Density Distribution, 42 MW
Initial Power

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Fig. 18. Axial Power Density Distribution at Various Radii, 42 MW Initial Power

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Fig. 19. Inlet Velocity Distribution, 42 MW Initial Power

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Fig. 20. Radial Exit Void Fraction Distribution, 42 MW Initial Power

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Fig. 21. Axial Void Fraction Distribution at Various Radii, 42 MW Initial Power

TABLE 18. Predicted Heat Transfer Data at 42 MW and 0 Burnup

Reactor Power	42 MW	Plutonium Zone	
Operating Pressure	600 psig	Av Power Density	61.75 kW/ℓ ^a
Operating Temperature	489°F	Total Power	15.06 MW
Average Steam Void Fraction (Based upon fuel box coolant vol)	0.262	First Shim Zone	
Average Steam Void Fraction (Based upon coolant in typical core lattice)	0.210	Av Power Density	64.98 kW/ℓ ^a
Equivalent Diameter of Coolant Channel	0.0461 ft	Total Power	12.32 MW
Total Core Heat Transfer Area	2327 ft ²	Second Shim Zone	
Core Power Density--Av	42.18 kW/ℓ ^a	Av Power Density	50.58 kW/ℓ ^a
--Max	130.5 kW/ℓ ^a	Total Power	10.96 MW
Core Heat Flux--Av	61,600 Btu/hr-ft ²	Natural Uranium Zone	
--Max	191,000 Btu/hr-ft ²	Av Power Density	10.59 kW/ℓ ^a
Total Recirculation Flowrate	7.51 x 10 ⁶ lb/hr	Total Power	3.66 MW
Average Exit Steam Quality (Core exit)	0.0235	Fuel Centerline Temperatures	
Estimated Carryunder	0.264	Core Av	881°F
Subcooling at Core Inlet	2.0 Btu/lb	Core Max	1672°F
		Integrated Fuel Conductivity, $\int k(\theta)d\theta$	
		Core Av	5.2 W/cm
		Core Max	16.1 W/cm

^aLiter of coolant within the fuel element boxes.

IV. DESCRIPTION OF GENERAL PLANT IMPROVEMENTS

In addition to the aforementioned changes required for the performance of the plutonium recycle experiment, a number of other plant changes and modifications have been accomplished since the writing of the previous 100 MW operation hazards analysis,³ based on previous operating experience and in the interest of increased reliability and safety. A brief description of these modifications is given in the paragraphs which follow.

A. Instrumentation and Control

1. Startup Counters

In order to provide visual source range neutron level information to the reactor operator during startup and while shutdown, the two startup scaler channels⁸ have been replaced by two source range count-rate meter channels. These new instruments feature boron-coated, gas-filled, proportional counter detectors located in the radial instrument holes, pre-amplifiers located adjacent to the instrument holes, discriminator amplifier integrating units located on the instrument gallery adjacent to the reactor shield, detector high voltage power supplies also in the instrument gallery cabinets, and logarithmic count-rate meters, period meters, and decimal scalars located at the reactor control console in the control room. The new count-rate channels have a sensitivity of 10 cpm/nV and a meter range of 10 to 10^7 cpm with count rates below 10 cpm being covered by the control room decimal scalars. The high voltage to the detectors may be turned off from the reactor console when neutron level exceeds the source range, so as to extend the life of the counter chambers. One of these new startup channels is shown in block diagram form in Fig. 22.

2. Period Instrument

In an effort to eliminate frequent spurious period trips during startup, a new low-microphonic compensated ion chamber will be installed in the log-period instrument. This new chamber will have a sensitivity of 4×10^{-14} Amps/nV and be supplied by highly regulated positive and negative power supplies. An isolation transformer in the a.c. power supply to the log-period amplifier will serve to eliminate the effect of noise on the period instrument.

3. Source

The reactor startup background Sb-Be source will be re-irradiated to saturation and installed in the core opposite the startup counter instruments to provide approximately 10^{10} neutrons per second. The resulting source flux, assuming a partial fuel loading having a multiplication of 10, will be approximately 5000-50,000 cpm as read on the

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Fig. 22. Source Range Count-rate Instrument

permanently installed source range instruments. As in the past, source strength shall be maintained by subsequent irradiation in the EBWR core during power operation.

4. Temperature Interlock

During the experiments preliminary to 100 MW operation, a startup temperature interlock by-pass key switch was installed to permit cold critical experiments to be performed without the use of a temporary jumper in the scram relay cabinet. The use of the temperature interlock by-pass is prohibited during normal startup operations.

5. High Water Level By-pass

A new by-pass key switch will be installed on the reactor control console to permit by-pass of the reactor high water level scram during shutdown periods when the reactor vessel head is removed and the water level is raised above the scram point level for shielding purposes. The use of a key switch for this frequently used by-pass function eliminates the use of a temporary jumper in the scram annunciator relay cabinet. This high water level scram by-pass key switch operates in the same manner as other console keys and will be included under the rules for use of console key switches.

6. Level Scram Selectors

Two new key-type selector switches will be installed in the reactor high and low water level scram switch circuits. The purpose of these switches is to permit reliable, positive selection of the desired high and low reactor water level scram settings. Formerly, the selection of proper scram points for operation with or without the core riser or its one foot extension was accomplished by manipulation of electrical leads in a terminal box, a procedure which is inherently unreliable and not readily double-checked. These key switches will be installed on the local water column terminal box and will have suitable nameplates indicating the instrument number and trip level of the level switch selected. Figure 23 is a schematic diagram of the high and low level scram circuits with the new selector switches included. The keys for these switches are removable in any position and will be included under the rules for use of console key switches. It is anticipated that for normal operation during the present experiment, high level switch H-7-A and low level switch H-8-B (see Fig. 23) will be used unless operation without the four-foot core riser or with the one-foot riser extension become desirable.

7. Process Instruments

As a result of operational experience during the 100 MW experiment, several additional process instruments have been installed

Fig. 23

Reactor Water Level Scram Point Selector
Switches and Associated Circuitry

in the Intermediate Steam System. Differential-pressure cell level instruments have been added to the shell side of the primary reboilers and to the flash tank. These instruments measure the differential pressure on a 0-50 in. water column and transmit a level signal to indicator-recorders mounted on control room panel #12.

A 0-500 psig pressure transducer has been installed on the air-cooled condenser header to transmit a pressure signal to an indicator located on control room panel #12

B. Purification and Chemical Control Systems

1. Primary Purification System

The reactor primary water purification system has been extensively analyzed and re-designed to perform its major functions reliably and efficiently. These include:

- a. Control of primary water quality at a specific resistance of about one (1) megohm to minimize deposition of suspended and dissolved impurities on core heat transfer surfaces and buildup of radioactivity in the external steam and water circuits.
- b. Dissipation of reactor decay heat for normal shutdown cooling.
- c. Removal of boric acid from the primary water.

Operating experience and special tests⁴³ have consistently indicated failure of the purification system canned-rotor pumps and subcooler heat exchangers to perform according to original design specifications. The system will therefore be modified in accordance with the improved design as shown in Fig. 24. The principal features of the re-designed system are:

- a. Installation of two 350-ft discharge head, 50 gpm, horizontal canned-rotor pumps.
- b. Relocation of the new pumps immediately downstream of the regenerative heat exchangers to provide the required NPSH.
- c. Installation of two heliflow-type non-regenerative subcoolers having heat removal capacities of 612,000 Btu/hr each at 12 gpm flow and 350,000 Btu/hr each at 50 gpm flow. These units will normally be operated in series for shutdown cooling.

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Fig. 24. Reactor Primary Water Purification System Piping Diagram

d. Valving arrangements to permit shifting from normal operation to shutdown cooling from the control room panel and to permit continuous reactor water purification and boric acid monitoring while shutdown cooling is in progress.

e. Adequate flow and temperature instrumentation to permit performance of heat balances to evaluate system and measure reactor decay heat load.

During the course of the 100 MW operations, numerous changes were made to the purification system. The most significant of these changes was elimination of two unshielded mixed-bed polishing resin units and conversion of the shielded prefilter units to mixed-bed purification demineralizers with 0.6 cu ft each of 1:1 mixed cation and anion resins. This change has been incorporated into the re-designed system.

2. Control Boric Acid System

The control boric acid addition system is the system normally used to increase reactor water boric acid concentration whether the reactor is at the hot, operating temperature and pressure or at ambient temperature and pressure conditions.

This system has been modified by the removal of the variable-speed, positive displacement boric acid metering pump⁹ which proved to be of no significant value in boric acid management and which was of very limited capacity. The two fixed-rate, positive displacement boric acid addition pumps have been retained. Each of these pumps is capable of delivering 1.4 gal/min of stock boric acid solution (182 gm/gal) to the reactor at the 600 psig operating pressure. These pumps are operated by remote switches on the control room panel number seven and are considered to provide adequate boric acid management for all plutonium recycle experiment operations.

The previous continuous monitoring system will be retained. Bi-hourly determinations of the boric acid content will also be made using an indicator-titrimetric method. The previous bi-hourly determination based on neutron absorption by the B^{10} in the boric acid will now be used as a periodic check on the indicator-titrimetric results.

C. Power Plant

1. Steam Systems

a. The reactor steam flow and by-pass steam flow instrument differential-pressure transducers will be replaced with 0-230 in. units. This change was necessitated by the fact that the originally installed flow

nozzles were fabricated to measure steam flow at 800 psig rather than the 600 psig at which the system operates, thereby limiting the 0-150 in. installed transducers to 85,000 lb/hr instead of the 100,000 lb/hr intended. Replacement of the 0-150 in. transducers with new 0-230 in. units will permit measuring of 100,000 lb/hr steam flow at 600 psig without changing the flow nozzles themselves.

b. The intermediate steam system safety relief valve settings were changed to 400 and 410 psig prior to the 100 MW program. These settings will be retained for future operations.

c. In order to provide remote addition of make-up water to the intermediate steam system, a solenoid-operated valve has been installed in the demineralized water make-up line to the flash tank. This make-up valve will be operated from a switch on control room panel #12 where the new flash tank level indicator is located.

2. Main Condenser

a. The main condenser will be completely retubed with 18 B.W.G. phosphorical Admiralty Brass tubes replacing the original 6063-T831 aluminum alloy alclad tubes. After completion of the 100 MW program, inspection of the circulating water side of the aluminum main condenser tubes revealed very severe general pitting; and eddy-current device wall thickness measurements indicated that, of the 26% of the 1158 tubes measured, 78% had less than 75% of the original 0.058 in. wall remaining. In view of these conditions, it appeared unlikely that the condenser tubes would give reliable service during the long continuous runs required for the plutonium recycle experimental program, particularly since internal corrosion was likely to be accelerated due to the virtual elimination of the pure aluminum protective cladding by the severe pitting. Inhibited Admiralty Brass which was the original alternate material specified for the main condenser tubes, was selected for the replacement tubes since the aluminum alloy did not meet the expected performance and because operational water chemistry and corrosion product distribution experience indicate that the brass constituents will not introduce any significantly different phenomena.

The cooling tower water treatment methods will be adjusted to accommodate the change of material in the main condenser tubes.

b. The main condenser desuperheater pipe will be modified to eliminate back-pressure buildup in the deaerator vent line which has resulted in the leakage of radioactive gases at the packing of the deaerator safety relief valve. The modification consists simply of eliminating free communication between the desuperheater pipe and the deaerator vent line by cutting the common pipe, welding a plate over the open end of the desuperheater pipe, and adding a diffuser plate to prevent impingement of deaerator steam on the condenser tube bundles. Figure 25 illustrates the details of this change.

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Fig. 25. Main Condenser Desuperheater Modification

3. Air Drying and Fluid Recovery System

The power plant air drying and fluid recovery system was originally designed and installed to minimize outleakage of working fluid from turbine shaft seals and certain valve and pump glands and to prevent dilution of heavy water, if used, by moisture contained in air inleakage to the system.⁸ Since heavy water will not be used as a working fluid in the present experiment, the air drying feature of this system is not required for operation of the steam plant. In order to evaluate operation of the system without the air drying equipment, an isolation valve and by-pass valve and piping will be installed to permit by-passing of the air drying system and discharge of the gland exhaust non-condensables directly to the off-gas system.

If this mode of operation proves entirely satisfactory, the modification could be made permanent by elimination of the air drying equipment.

4. Air-lock Doors

The containment shell personnel air-lock doors may be modified by replacing the existing manually operated latch mechanism with an electro-hydraulic operated latch mechanism. This change would not alter the present sealing arrangement of the doors but merely alter the manner in which the doors are brought to bear on the gasketed closure. The existing system of interlocks, which normally prevents opening of both air-lock doors or equalizing valves, will be maintained by means of electrical limit switches and solenoid operated equalizing valves. Manual operation of the latching mechanism will be possible in the event of failure of electrical or hydraulic components. A key switch will permit by-pass of the electrical interlocks during periods when simultaneous opening of both doors is allowed by the EBWR operating rules.

5. Containment Shell Test

Upon completion of air-lock door modifications and prior to loading fuel into the reactor vessel, a containment shell 15 psig internal pressure test will be performed to check building integrity and insure the leakage rate is within permissible limits.

D. Modification of the 6 in. Forced Circulation Connections

In the original EBWR design, four 6 in. Tees were installed in the bottom of the reactor vessel for possible conversion from a natural convection to a forced circulation system. Although one end of three Tees was connected to a blowdown system, radioactive corrosion products collected in the side branches and could not be cleaned. It is believed that bubbling air through these Tees during some experiments caused the corrosion products to settle in the branch of the Tee. The fourth Tee is connected to the water purification system and has never accumulated such products.

There are four feet of shielding between the reactor vessel and the Tees. The Tees are located in the subreactor room alongside the control rod drives. At times the radiation levels emanating from the Tees have approached 100 R/hr at 2 in.

Because of this intolerable condition the three Tees will be removed from the reactor nozzles that extend through the shielding. Pipe reducers will be welded to the nozzles and drain lines provided for blowdown as was done previously. Welding and testing will conform to previous practices.

V. OPERATIONAL PROCEDURES

A. Rules for Core Loading

The core loading criteria which were specified for the 100 MW operations and which will continue in effect for the plutonium recycle experiment are stated below.

1. The system, which may contain boric acid, must always have the capacity to be made subcritical at any time by the use of only eight (8) of the nine (9) control rods.

2. Any loading shall be subcritical with all nine (9) control rods fully inserted and no boric acid in the reactor core.

B. Initial Approach to Plutonium Zone Critical

1. Prior to commencement of loading the plutonium fuel assemblies into the reactor, the following preliminary preparations and checks shall have been satisfactorily accomplished:

- a. The new control rods will be installed and tested for smoothness of operation and to insure a rapid insertion time of about 1.5 sec. The 4.3 in./min control rod drive motor will be installed in the center control rod drive mechanism.

- b. All newly installed and overhauled nuclear instruments, interlocks and components shall be given a pre-operational test.

- c. A freshly irradiated Sb-Be neutron source shall be installed in a suitable location near the plutonium zone.

- d. Two fission chamber counting channels and one ionization chamber micromicroammeter channel shall be temporarily installed in suitable reactor core locations to monitor reactor multiplication and approach to critical.

- e. When the permanently installed nuclear instruments provide adequate indication of reactor behavior, the temporary in-core counters and ion chamber may be removed.

2. Since no critical experiment will have been run prior to the plutonium fuel loading, a careful, deliberate approach to a critical loading will be conducted.

a. The expected minimum critical mass in terms of the number of plutonium fuel elements has been pre-determined and will be used as a guide in loading. (See Fig. 15.)

b. Plutonium fuel assemblies will then be loaded between the source and the counters to approach the minimum predicted critical mass.

c. Following the loading of the selected number of fuel assemblies, control rods will be withdrawn one at a time to permit measurement of clean core multiplication. Inverse multiplication as indicated by the two in-core counters will be plotted vs. the number of fuel assemblies loaded and extrapolations made to predict the approximate critical loading prior to loading the next selected number of fuel assemblies into the reactor.

d. When a cold critical mass has been loaded, a reference run to critical shall be performed and the response of the permanently installed nuclear instruments checked against the temporary in-core instruments for proper performance.

e. The full 36-assembly plutonium loading will then be approached in steps by loadings of fuel assemblies at several boric acid concentrations. Multiplication will be determined by rod withdrawal after each loading increment in order to predict the critical loading for the new boric acid concentration.

C. Plutonium Critical Experiments

With the plutonium zone fully loaded, the usual reactivity coefficient measurements such as rod worth and boric acid worth and other standard experiments will be conducted to check predicted characteristics and to establish reference data upon which to base future evaluations of system behavior at the successive burnup steps. Control rod oscillator experiments to measure prompt neutron lifetime and effective delayed neutron fraction may also be included. Various critical experiments will also be run at elevated temperatures using the startup heater to increase the water temperature. These hot critical experiments will be run with the reactor vessel head installed.

D. Determination of Final Loading

The final design power loading of uranium shim zones and driver zone fuel assemblies will be performed in a series of steps with boric acid concentrations maintained to insure eight-rod shutdown at all times and criticals performed at each step to determine critical rod positions. Adjustments to the system reactivity may be made at this time so as to guarantee the 9-rod shutdown condition. These adjustments will be primarily made by altering the composition of the shim zones. When the full

core has been loaded, it is intended that the 4.3 in./min drive motor in the center control rod drive mechanism be replaced with the 14.5 in./min motor. Reactivity coefficients, rod worth, and boric acid worth experiments will be repeated for the final system loading.

E. Initial Approach to Power

Immediately prior to power operation, the nuclear instruments will be calibrated to read approximate power level by operating the reactor at a low power of about one megawatt and comparing the instrument readings with the thermal power level as calculated from the measured temperature rise of the reactor plus the various heat losses.

With high flux trip instruments set to shutdown the reactor at about ten percent of expected operating power level, the reactor will be operated at a low power level until sufficient data is obtained, including control rod oscillator stability measurements to permit an increase to design maximum operating power. (See Appendix A.)

F. Normal Operation

After initial physics experiments have been completed and reactor characteristics determined, reactor and power plant operations will be conducted in accordance with the EBWR Operating Manual. The major steps in the normal operating procedures are outlined below:

1. Normal Startup

- a. Conduct pre-startup plant check of all systems and instruments.
- b. Pre-heat reactor to clear Reactor Water Temperature Low startup interlock at 325°F, using the startup heater.
- c. Clear the shutdown annunciator panel to obtain control power.
- d. Conduct control rod drop tests.
- e. Startup reactor and raise power level to about 1 MW to complete heatup.
- f. Energize required steam plant auxiliaries and draw a vacuum on the main condenser.
- g. Set steam by-pass valve system to maintain 600 psig reactor pressure and shift to automatic control.

- h. Add feedwater to maintain reactor water level as required using manual control.
- i. Warm up turbogenerator unit and place on governor speed control. Parallel generator on laboratory bus and pick up load as permitted by reactor power level and steam by-pass valve flow.
- j. Commence boric acid removal from reactor water.
- k. Increase reactor power level to by-pass a minimum of 15,000 lb/hr steam to the main condenser.
- l. Adjust reboiler plant feedwater system and deaerator and place in manual control operation.
- m. Line up primary and secondary reboilers for operation and commence warmup of the reboiler steam system using the air-operated by-pass valve.
- n. When reboiler steam system is warmed up, transfer reactor pressure control from the turbine steam by-pass valve to the reboiler reactor pressure regulating valve.
- o. Transfer the reboiler plant feedwater regulating valve to automatic control of reactor water level and turbine plant feedwater regulating valve to automatic hotwell level control.
- p. If required to maintain intermediate steam pressure below 350 psig, warm up and place the air-cooled heat exchanger and intermediate pressure control valve in operation.

2. Routine Operation

When startup procedures have been completed, the turbogenerator portion of the plant will operate as in the past. Variations in generator electrical load will be absorbed by the reboiler plant reactor pressure control valve which will reject excess reactor steam to the primary reboilers or divert steam from those reboilers to compensate an increase in generator load.

The secondary reboiler will be base loaded by automatic control of the intermediate steam condensing pressure. The air-cooled heat exchangers will function to automatically control intermediate system pressure as the output of the primary reboilers varies with reactor power level or turbogenerator load.

The reactor pressure will be maintained at 600 psig by the reboiler reactor pressure control valve or by the turbine steam by-pass valve in the event of reboiler plant shutdown.

Periodic shutdowns will be scheduled to permit verification of the nine rod shutdown requirement as fuel burnup proceeds.

3. Normal Shutdown

a. The laboratory boiler house will be notified of the planned shutdown and the secondary reboiler will be unloaded to the air-cooled condenser or by reducing reactor power as the laboratory boilers pick up the additional steam load.

b. When reactor power has been reduced below 35 MW and steam flow to the primary reboilers is less than 45,000 lb/hr, transfer reactor pressure control to the turbine steam by-pass valve, shut the reboiler steam stop valve and transfer reactor water level control to the turbine plant feedwater regulating valve.

c. Unload and trip the turbogenerator unit off the bus and shut down the steam turbine.

d. Reduce reactor power level and maintain 600 psig reactor pressure by adjustment of the turbine steam by-pass valve.

e. When steam by-pass valve is closed, shut down reactor.

f. Dissipate decay heat to maintain reactor pressure at or below 600 psig by means of the turbine steam by-pass valve.

g. If plant cooldown is not anticipated, shut down air ejectors and break vacuum.

h. Add boric acid required for cold, eight-rod shutdown margin.

G. Periodic Investigation of the Pu Zone

At fuel burnup steps of about 2000 MWd/tonne in the plutonium zone, major investigations of the irradiated system will be performed. Assuming sustained operation at a power of about 40 MW, the first such step would require an elapsed time of about 4 months. Subsequent steps would require somewhat longer elapsed times owing to gradual decrease in reactor power, burnup of the plutonium zone, and redistribution of the power in the reactor. If one assumes no change in power level, about 14 months of sustained operation would be required to reach a plutonium zone burnup of 0.006 (~6000 MWd/tonne).

The periodic major investigations of the system will include:

1. Determination of cold, critical rod bank position at a given boric acid concentration for the full core system. Substitution of the 4.3 in./min control rod drive motor for the 14.5 in./min motor in the center control rod drive mechanism.
2. Complete unloading of all enriched and natural uranium fuel assemblies to permit measurements of the plutonium zone parameters.
3. Performance of reactivity coefficient measurements and control rod worths in the critical plutonium zone. For burnups greater than about 3000 MWd/tonne it may be necessary to include some shim assemblies in order to obtain a critical mass.
4. Removal of selected plutonium fuel rods for isotopic analysis.
5. Replacement of plutonium fuel rods with fresh fuel and determination of reactivity changes.
6. Reloading of all the uranium fuel into the reactor core. At this time, adjustments to the system reactivity may be made by altering the composition of the shim zones. The 14.5 in./min control rod drive motor will again be reinstalled in the center control rod drive mechanism.
7. Re-determination of cold, critical rod bank position for the full core for comparison with that determined at the beginning of the investigation.
8. Performance of oscillator transfer function measurements at low power to verify stability after burnup has produced changes in power distribution.
9. After several of these burnup steps, probably not more frequently than every 6000 MWd/tonne, a major reloading of the shim zones may be required.

H. Hydrostatic Testing

Appendix B presents data and calculations pertinent to the brittle-ductile transition temperature for the EBWR pressure vessel. The results given there indicate that it may be desirable after some point in the program to perform hydrostatic testing of the system at an elevated water temperature. The need for increasing this temperature during hydrostatic testing as irradiation proceeds will be considered in the light of results obtained from surveillance samples to be included in the core loading. The expected required maximum temperature of 180°F at the end of 80 MW-yr (Appendix B) is well below the capability of the preheat system and presents no technological problem.

VI. SAFETY CONSIDERATIONS

A. Introduction

Experience gained during previous operation of EBWR (Refs. 2,5) at power levels up to about twice the power level anticipated for the proposed experiment, together with the basic similarity of the proposed and previous reactor and plant configurations serve as a basis for confidence concerning the safety of the plutonium recycle experiment. Since the main objective of the experiment is the attainment of data concerning the long term isotopic behaviour of plutonium in a thermal light water moderated reactor, the experiment has been kept as simple as possible while still yielding this information so that it will be most amenable to analysis. Modifications to the EBWR have been kept to a minimum. Previous sections of this report have described general improvements to the EBWR plant as well as facility modifications and results of physics calculations specific to the planned plutonium recycle experiment.

In this section, the influence which the above changes may have on the safety of the system will be considered. Owing to the confidence and extensive experience gained during the operation of the EBWR, the previous EBWR safety considerations will be used as a standard by which the safety of the proposed system can be measured. Various examples of abnormal operation which are unchanged from the previous EBWR loading will also be discussed here for convenience.

The safety of the system will be demonstrated by showing that the proposed plutonium recycle experiment is as safe as the system involved in previous EBWR operation, and in fact represents a significantly more reliable reactor-plant configuration than has previously been encountered in EBWR.

B. Comparison of Proposed Experiment with Previous EBWR Experience

1. Core Loading

a. Plutonium Content

The initial core loading (Section III.A) will contain about 13.3 kg of Pu^{239} and 1.2 kg of Pu^{240} . This represents only about a 50% increase in plutonium content as compared with the 9 kg of plutonium present in EBWR at the time of shutdown.¹¹ In addition, the proposed experiment involves only oxide fuel whereas the previous core contained mainly metal plate-type fuel elements. The fact that the plutonium is chemically combined in the proposed core represents a significant safety advantage over previous EBWR operation with regard to the degree to which plutonium could be dispersed following a fuel rod failure.

Other advantages of the oxide vs. metal fuel are discussed below in Section VI.B.3.

The similarity of plutonium content is reflected in the similarity of the effective delayed neutron fractions, β_{eff} , in the proposed and previous cores as indicated below.

b. Delayed Neutron Fraction

In the proposed experiment, the fully loaded initial system has a calculated β_{eff} of about 0.005 and at a plutonium zone burnup of 0.006 (~6000 MWd/tonne), β_{eff} is over 0.006. This increase in β_{eff} with irradiation is due to the shift of power distribution from the plutonium zone out into the enriched uranium shim zones. Previously, in EBWR, the trend in β_{eff} was of course decreasing rather than increasing, with β_{eff} varying from an initial value of about 0.0082 (Ref. 3) to about 0.0054 at the end of the 100 MW operation.⁷ Thus the proposed experiment exhibits a β_{eff} initially comparable to that of the previous core at the time of shutdown, and β_{eff} increases during the irradiation.

Although the largest part of the proposed experiment will be concerned with the operation of the fully loaded system, the periodic investigations of the plutonium critical configuration form an important part of the overall experiment. In this case, since Pu^{239} is the only fissile species, β_{eff} is less than for the fully loaded system and has a value of about 0.003.

The comparison of the β_{eff} for the proposed experiment and previous operation is involved most directly in considerations of control rod drive speeds and stability. These are considered below.

c. Control Rod Drive Speed

When comparing the plutonium recycle experiment with the previous EBWR operation with regard to determining an acceptable control rod drive speed, the most meaningful quantities for comparison are the values of β_{eff} in the two cases. On the other hand, if one attempts to make comparisons based on relative control rod worths, the problem becomes ill defined owing to the wide variations encountered during the previous operation and expected for the proposed experiment.

For the 100 MW EBWR operation, the control rod drive speed for each of the nine control rods was fixed at 28 in./min.³

For the proposed experiment, the drive speed is being reduced to about 14.5 in./min so that the maximum reactivity insertion rate for the full system will be about 2/3 of that encountered during the 100 MW operation. For the plutonium critical configuration, the central rod speed

will be reduced to about 4.3 in./min. This speed results in a reactivity insertion rate of about 80% of that encountered in the full core loading, and about half of that experienced during the 100 MW operation. These changes represent a considerable conservatism compared with previous operation.

d. Stability

Although the plutonium content and the values of β_{eff} are similar for the proposed experiment and the previous operation, the magnitude of the negative void coefficient of reactivity is larger in the proposed experiment. This is due to the central plutonium loading which increases the neutron importance in that zone and makes the core effectively smaller than in the previous EBWR core loading. Thus even though the expected operating power is less than that previously attained safely in EBWR, the stability of the proposed system will not be taken for granted since a large negative void coefficient of reactivity could adversely influence stability.

The stability of the fully loaded system will be investigated using essentially the same techniques and equipment which were successfully used during the 100 MW program.⁵ Transfer function measurements will be made at several power levels to obtain sufficient data which can be extrapolated to determine an upper power limit for stable operation. These measurements will be made at power at selected intervals during the experiment to detect any decrease in stability due to changes in power distribution.

The stability of the plutonium critical configuration need not be considered since power operation is not involved for that case.

2. Inherent Safety

By the inherent safety of the reactor is meant its ability to withstand the effects of accidental reactivity additions without responding in such a way as to endanger its surroundings, and to withstand these effects either indefinitely or until the probability that shutdown will be accomplished by external means (safety circuits, manual scram, etc.) is for all practical purposes unity. The EBWR has exhibited a high degree of inherent safety principally because of the large and relatively quick-acting negative power coefficient of reactivity characteristic of a normal boiling water reactor. In addition, the prompt fuel temperature coefficient of reactivity has provided protection against very rapid reactivity insertions.

The proposed experiment in EBWR qualitatively exhibits the same inherent safety characteristics as did the previous cores and benefits from the properties of the oxide fuel which are discussed in the next section.

3. Oxide Fuel Considerations

The present oxide fuel loading has significant advantages with respect to safety over the previous predominantly metal fuel core loadings. As was pointed out in Section III.G.5, the present core can sustain much larger rapid reactivity insertions without core damage than could the previous metal fueled system. This is due to the higher melting point and lower thermal conductivity of the oxide fuel. For example, whereas the previous core could, by virtue of the Doppler effect combined with metal expansion, sustain a step reactivity insertion of about 0.3% above prompt critical,³ the corresponding conservative value for the present core based solely on the Doppler effect is about 1%.

The advantage of an oxide vs. a metal fueled core has been well documented by SPERT-1 data.³⁵⁻³⁷ Recent tests of an H₂O moderated SPERT-1 core consisting of 0.5 in. dia rods of 4% enriched UO₂, have indicated that for short periods, the only important shutdown mechanism is due to Doppler broadening of the U²³⁸ capture resonances. In particular, in none of the cases examined did the cladding surface temperature at peak power exceed the saturation temperature of the water, so that void formation did not contribute to the shutdown mechanism. Calculations discussed in Section III.G.5 agree qualitatively with these results. In the most recent test⁴⁴ the core which was initially at a power level of 0.2 watt and at room temperature sustained a \$2.70 reactivity insertion corresponding to a 2.2 msec period. The peak power was 19,000 MW thermal and the total energy release was 160 MW-sec. In this test, which had been intended to destroy the core, only two of the 590 fuel pins were ruptured. It had been originally feared³⁵ that sudden rupture of an oxide fuel rod during a short period transient might produce progressive rupturing of neighboring pins due to local steam formation. This anticipated "chain reaction" for rupturing fuel pins did not occur and the core sustained no structural damage.

This behaviour is in marked contrast with earlier experiments on metal plate-type fuel which exhibited extensive damage after a total energy release of one fifth the above amount.⁴⁴ The significant difference between the oxide rod fuel and metal plate fuel is due to the greatly decreased thermal conductivity of the oxide compared with the metal. In the latter case, the rapid transfer of heat from the meat to the cladding can cause sudden rapid boiling with subsequent development of destructive transient pressures, and surface melting due to the decreased surface heat transfer coefficient. Because of the ability of an oxide core to withstand large excursions without damage, it has been suggested that a rod type oxide core may provide the basis for an efficient burst facility.³⁵

Measurements and calculations have been performed at the Hanford laboratory⁴⁵ which were concerned with the possibility of a time delay in the Doppler effect owing to the heterogeneous nature of the mixed

PuO₂-UO₂ fuel. This possibility exists since the heat generation occurs mainly in the Pu²³⁹ in the PuO₂ while the Doppler reactivity effect is due mainly to raising the temperature of the U²³⁸ in the UO₂. The Hanford studies were based on mixed oxides similar to those which are to be used in EBWR. Transfer of heat from the Pu to the U was assumed to take place by conduction and direct deposition by fission fragments. For the 1.5 w/o PuO₂ in UO₂, the time required for the UO₂ to reach 63% of the PuO₂ temperature was about 1 msec. This value is conservative since radiative heat transfer was neglected, although calculations indicate that the radiative mechanism would contribute significantly to the total UO₂ heating. The periods given in Table 17 (Section III.G.5) are all significantly longer than the expected time delay in the heating of the U²³⁸ and therefore this phenomenon will not alter the previous results based on the assumption of a prompt acting Doppler effect.

The improved corrosion resistance of oxide as compared to metal fuel represents another safety advantage of the proposed experiment in the event of a cladding failure exposing fuel to water since the likelihood of plutonium dispersal is reduced. As indicated in Section III.G.5, failure of the cladding would not be expected except for very large excursions accompanied by wholesale melting of the oxide fuel. For example, a calculated step insertion of 1.0% k_{ex} for the worst control rod configuration carried the center of the oxide fuel at the hottest point in the core to several hundred degrees above the melting point. However, the outer surface of the oxide did not melt, and the maximum temperature difference across the Zircaloy-2 clad was about 270°C corresponding to a maximum thermal stress of about 18,700 psi. This may be compared with the short term tensile strength of about 35,000 psi and the elastic limit of about 18,000 psi for Zircaloy-2. Even for this severe excursion the clad would therefore not be expected to fail.

4. Soluble Poison Systems

a. High Pressure Boric Acid System

The high pressure boric acid system is being retained. As before, at the normal reactor operating pressure of 600 psi it is capable of increasing the reactor boric acid concentration by about 9.35 gms of H₃BO₃ per gal of water at the hot operating no void density ($\rho = 0.793$). Assuming no control rods are inserted and that the system is at operating temperature, the following conditions would exist. With a 3% reactivity contribution due to equilibrium Xe, high pressure injection of boric acid would result in the system being about 1% subcritical at a plutonium zone burnup of 0.006. With equilibrium Xe the initial system would have a k_{ex} of about 0.3% and hence would continue to operate at a low power. If no Xe is present, even the system at 0.006 burnup would require additional boric acid from the control boric acid system to completely stop power generation.

The inability to stop power generation completely using only the high pressure boric acid system was not considered to be a serious drawback during previous operation of EBWR,³ nor is it so considered for the proposed experiment.

b. Control Boric Acid System

Other than the modifications discussed in Section IV.B, the control boric acid system is the same as that used successfully for the 100 MW operation. As before, the boric acid control system is not an emergency system but rather a holddown device to be used in a carefully planned manner. In general, boric acid will be present whenever the reactor is cold and a core suitable for high power operation is installed. The boric acid will for example permit satisfying the eight-rod shutdown requirement. As before boric acid will gradually be removed as required either when the reactor is being raised to operating temperature on nuclear heat or while the reactor is at power. Since power operation with boric acid in the system has been demonstrated to be feasible and safe, there is no need to continue removing boric acid past the point required for reactor operations.

As in previous operation of EBWR, the presence of boric acid reduces the magnitude of the negative void coefficient of reactivity and if present in sufficient concentration can reverse the sign and make the void coefficient positive. For example, for the initial full system, the void coefficient of reactivity is calculated to be zero for a boric acid concentration of about 13 gm/gal. As shown in Section III.G, the initial system requires a boric acid concentration of about 4.0 gm/gal of cold water to satisfy the eight-rod shutdown requirement. At a Pu zone burnup of 0.006, about 9 gm/gal are required. In each case, the void coefficients of reactivity though considerably reduced from the initial unpoisoned values, are still negative.

The nine- and eight-rod shutdown criteria will be checked during the course of the experiment and if necessary, adjustments will be made to the system reactivity by altering the composition of the shim zones.

The presence of the boric acid should have a negligible effect on the stability of the system. In general at operating condition very little, if any, of the boric acid will be present. The effect of the boric acid on the void coefficient becomes progressively less important as the system goes up in temperature and void content. What small boric acid remains at operating condition should enhance stability by reducing the magnitude of the negative void coefficient.

5. Reactor Vessel Irradiation Effect

Appendix B presents data and calculations pertinent to the brittle-ductile transition temperature for the EBWR pressure vessel. It is

pointed out there that based on the past irradiation history of EBWR and correlation with other data, no danger exists of brittle failure at operating temperatures. On the other hand, during hydrostatic tests, higher stresses are involved. At some point in the program it may become desirable to perform hydrostatic tests above room temperature. This is easily done with existing facilities. The need for increasing the temperature during hydrostatic testing as irradiation proceeds will be considered in the light of results obtained from surveillance samples to be included in the core loading.

6. Plant Improvements

As indicated in Sections III and IV, the same basic plant is involved as before. The modifications, which were kept to a minimum, have all been in the sense of improving reliability or increasing the safety of the system. Several examples of such improved reliability are cited below.

The improvement in the water purification system lessens the likelihood of fouling the heat transfer surfaces on the fuel pins.

The nuclear instrumentation system has been generally improved by rerouting signal cables and supplying a separate ground return. These modifications reduce the chance for spurious scram signals and hence lead to more reliable operation of the system.

New instruments have been installed in the reboiler plant providing greater ease of operation.

The expected power of about 40 MW is well below the 70 MW range where liquid carryover commences.⁵ Previous operation with liquid carryover greatly taxed the various components of the plant involved. This lower power also allows a greater margin with regard to the capability of the reboiler plant. That is, since the total capacity of both reboilers is about 65 MW whereas the present experiment will be utilizing only about 20 MW of this capacity, a wide margin exists for handling abnormal steam loads before depending upon the protective systems.

The absorber portion of the boron stainless steel control rods is joined to the Zircaloy-2 follower sections by means of an assembly rod. A tongue-and-groove joint replaces the original riveted transition sections. The present design eliminates the stress due to differential thermal expansion of the absorber and follower materials which produced shear type failure of the original rivet shanks.

C. Discussion of Possible Hazards

1. Listing of Hazards Enumerated in Previous EBWR 100 MW Hazards Report and Relation to Proposed Loading

Owing to the basic similarity of the proposed system and the previous system, most of the abnormal conditions considered before are equally applicable in the proposed experiment.

In this section are listed the items enumerated for the 100 MW operation³ as possible hazards, and their relation to the proposed loading is briefly indicated.

Shutdown Cooling--Loss of Water

This is discussed below in Section VI.C.3 as a potential "maximum credible accident." It is pointed out that owing to the use of oxide fuel, this incident has less serious consequences as compared with the previous cores since the likelihood of metal water reactions and subsequent plutonium dispersal are reduced. Otherwise, there is no significant change.

Power Failure

No significant change.

Pump Failure

No significant change.

Instrument Failure

No significant change.

Control Rod Failure

The modification of the boron-stainless steel control rods (Section III.C) lessens the likelihood of control rod structural failure. Otherwise, although there is some change in the reactivity in voids, the previous comments are basically unchanged.

Change in Power Demand

The proposed system has a somewhat different power coefficient of reactivity but the conclusions presented are basically unchanged.

Too Much or Too Little Water in Reactor

Although the reactivity coefficient related to changes in water height of the proposed system may differ somewhat from the previous core, there is no significant change in the conclusions. The consequences of dropping a fuel element, presented in this section before, are discussed below in Section VI.C.2b. It is shown there that the lower fuel element worth and greater ability of the proposed oxide core to sustain a large reactivity insertion without core damage represent a significant advantage with regard to this type of accident.

Water Leaks from the External System

No significant change.

Cladding Failure

The proposed system is much less susceptible to cladding failure, as discussed in Section VI.B.3.

Startup Accident

This is discussed in Section VI.C.2a. There is no significant change.

Melting of Fuel

The proposed fuel consists of oxide rods rather than metal plates as before. Section VI.B.3 discusses the much lower likelihood of melting of the proposed fuel compared to the previous core loadings. The present system can tolerate a much greater excursion without core damage, and owing to the chemically bound nature of the oxide fuel, has a much lower likelihood of dispersing plutonium in the event of a fuel rod failure.

Excessive Pressure

The two safety valves which were set at 750 and 775 psig have been reset to 650 and 665 psig. Otherwise, no significant change.

Sudden Closure of Steam Valve

No significant change except that the lower power for the proposed experiment allows a wide margin for handling abnormal steam loads before depending upon the protective systems.

Failure of Steam System

No significant change.

Sudden Introduction of Cold Water

The proposed system has a larger temperature coefficient of reactivity than did the previous core. This would lead to a larger reactivity insertion rate in the case of a sudden introduction of cold water. However, similar arguments to those presented previously minimizing the severity of this abnormal situation can be given here. We consider as before the situation for which the main steam line is closed and feedwater is not added until the reactor has reached its operating temperature. The feedwater pumps can add water at a temperature of about 130°F and at a maximum rate of about 1,600 gal/min. This rate is reached only after complete opening of the slow acting feedwater valves. The feedwater would mix with the naturally circulating 4000 gal of water in the reactor at a temperature of 489°F. Starting with a normal water height in the system, the high water level scram point would be reached after about 750 gal had been added to the system. Thus, the reactor would be shut down in about 30 sec due to scrambling of all control rods. If one assumes instantaneous mixing during this process, the system temperature at this time would have dropped about 60°F and the water density would have increased by about 6%. From the reactivity coefficients given in Section III.G, the resulting reactivity increase for the initial system would be about 0.8%. This is of course an overestimate owing to the time delay involved in reaching temperature equilibrium in so large a system.

After the reactor had scrammed, the feedwater pumps would continue to add water to the reactor until the source of feedwater was exhausted. This would occur in less than one minute, at which time the pumps would automatically be shut down.

Improper Charging of Fuel

This case is discussed in Section VI.C.2 and 2b where it is shown that the proposed core is significantly safer than the previous core. For the proposed core, inserting a fuel element into the reactor when it is just barely subcritical, would lead to a period much longer than the period calculated for the initial 100 MW core. In addition, as the irradiation proceeds, the resulting period from such an event would increase for the proposed experiment rather than shorten as in the previous core.

Sabotage with Fissionable Materials or Explosives

No significant change other than the greater inherent ability of the proposed core to withstand large excursions without damage.

Sabotage by Pressurizing Reactor Tank with Gas

No significant change.

Explosive Chemical Reaction between Fuel Elements and Water

The improved corrosion resistance of the proposed oxide fuel as compared with the previous metal fueled core will reduce the magnitude of any metal water reaction in the event of dispersal of the fuel into the water following a cladding failure. Chemical reaction of the Zircaloy-2 cladding with the water would still be possible.

Boric Acid System Mismanagement

Although the reactivity coefficients in the proposed core are different from those of the previous system, the consequences of mismanagement of the boric acid system are essentially unchanged from those discussed for the 100 MW operation. The boric acid system is basically the same as that used previously. Modifications are discussed in Section IV.B.2.

It can be concluded that the proposed system exhibits no hazards greater than those previously anticipated for EBWR. For many of the previously enumerated hazards, the proposed system is seen to be significantly improved from a safety point of view.

2. Accidents Involving Reactivity Additions

Many of the hypothetical accidents involving reactivity additions are preceded by the system lacking adequate shutdown margin.

Inadequate shutdown margin could result from:

- (1) Failure of operator to add boric acid when required, inadvertent loss of boric acid, or nonuniform dispersal of boric acid.
- (2) An erroneous loading, due to error on the part of loaders and lack of awareness on the part of operators and supervisors, or a blunder in the estimation of reactivity associated with a loading change.

The consequences of boric acid mismanagement have been thoroughly discussed elsewhere³ and are essentially unchanged for the proposed core.

Improper charging of fuel could result from interchange of the plutonium and shim zone fuel. However, the uranium and plutonium fuel assemblies can be distinguished from each other. In particular, the Pu fuel element cans are about $3\frac{1}{2}$ in. longer than are the U elements although the active lengths are the same. (See Section III.B.) Therefore, although it is physically possible to insert any type fuel assembly into any position of the core, the chance of a misloading with regard to the Pu vice U fuel

is highly unlikely. In addition, the reactivity change associated with such an interchange is very small. For example interchanging a corner assembly from the first shim row with a central Pu assembly in the initial system produces a δk of less than 0.2%.

Accident situations involving reactivity additions can be classed into two general types according to the rate of addition of reactivity to the system. In the first case, the reactivity is introduced at a low enough rate that the temperature distributions in the reactor approximate the equilibrium values during the event. The more serious accidents concern rapid reactivity insertions for which the transient temperature distributions differ greatly from the equilibrium distributions.

An example of the former case would be that of a startup accident involving violation of the shutdown criterion and withdrawal of a control rod beyond criticality using the normal drive unit. A second example would be the case of slow removal or dilution of boric acid in the system. This case is essentially unchanged from previous discussion in the 100 MW Hazards Report.³

Examples of the second more serious type of accident are more difficult to invent. One such case, which involves a gross error in judgement on the part of the operating crew as well as a clear exception to the rules, would postulate the use of the building crane to either smoothly withdraw a control rod, at about 13 times the normal 14 in./min, or to abruptly free a control rod which has become jammed. These situations were previously considered for the case of the 100 MW analysis.³⁹

In the following sections, further consideration will be given to these two classes of accident situations.

a. Startup Accident due to Withdrawal of Control Rod at Normal Rate

The situation in regards to a startup accident is similar to that for the case of the 100 MW operation. This accident results from the continued withdrawal of a control rod, at the normal rate, beyond criticality. A number of points can be made which are pertinent.

An antimony-beryllium neutron source irradiated to saturation will be maintained at strength by operation of the EBWR.

A period meter, set to release control 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.

Normally, control rods cannot be withdrawn unless the flux-indicating galvanometer is on its most sensitive scale and the water temperature is 325°F or above. As indicated earlier, the control rod drive speed is being reduced from the previous 28 to about 14 in./min. In the case of the plutonium critical configuration, the central rod speed is being reduced to about 4.3 in./min. The rate of addition of k by rod withdrawal for the case of the Pu critical configuration is approximately half that for the 100 MW operation. For the full system, the rate of reactivity addition is about $2/3$ that previously encountered for the 100 MW operation. As before, with these rates of addition of reactivity, the water will begin to boil as soon as the fuel rods 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 about four times the maximum operating heat flux, this does not appear to be a dangerous situation.

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

b. Rapid Reactivity Additions

One may conceive of two mechanisms by which reactivity might be added to the system sufficiently rapidly that the normal self-regulating nature due to void formation would not be of assistance as a shutdown mechanism. These involve either rapid insertion of fuel, or rapid removal of a control rod.

Rapid Insertion of Fuel

In the previous core, the maximum δk due to inserting a fuel element was calculated to be about 2%. If this were accomplished by dropping the fuel into the system when it was just slightly subcritical, this amount of reactivity would have led to fuel melting and core damage. In the proposed experiment owing both to the improved inherent safety afforded by the oxide fuel and decreased fuel element worth, dropping a fuel element in the fully loaded system would not result in core damage. For example, a central plutonium fuel element is worth about 0.6% relative to core water, and an element in the center of the first shim zone is worth about 0.3% relative to core water. These values are much less than the conservative values given in Table 17 for step reactivity insertions just leading to fuel melting at the hottest point in the core. For the plutonium critical configuration, the average fuel element worth is less than 1% relative to core water and this is again conservative as far as fuel melting is concerned. The central elements in the Pu configuration are worth somewhat more than the amount indicated in Table 17 needed for fuel melting. However, in any case, it should be remembered that in order for this postulated accident to occur, the system would have to be just

subcritical before the fuel element is inserted, and there would have to be a gross failure of either the fuel element structure itself or of the fuel element handling mechanism. As in previous operation, a positive locking device is incorporated into the loading tool gripper mechanism to prevent fuel from suddenly dropping into the core during loading operations. The fuel element cannot be released from the loading tool until all weight is released from the loading tool gripper. A positive unlocking action must be performed by the operator before the fuel element can be released. The fuel elements are lowered into the core by a hand-operated winch.

High Rate of Withdrawal of a Control Rod

The building crane could be attached to a control rod, and add reactivity by withdrawing the rod. The maximum crane speed is 180 in./min (minimum speed is 9 in./min) compared with the 14 in./min speed of the control rod drive mechanisms. Even a more violent event could be postulated if it is assumed that the building crane is being used to remove a control rod which has become jammed. If it is assumed that the control rod weighing about 180 lb is jerked out by the energy stored in the steel cable which is assumed to be under 3600 lb tension and has been stretched, the rod would accelerate to 25 ft/sec in 0.079 sec and would require a total of 0.2 sec to clear the core.

Neither of the above conditions would result in a nuclear incident unless the eight-rod shutdown requirement is first violated. However, with the shutdown margin lacking and with the vessel cover removed, both the smooth continuous withdrawal, or the rapid ejection of a control rod could lead to fuel melting and serious core damage.

The exact details of the incident would depend upon the configuration involved, which control rod is being removed, and the irradiation history of the system. For example, the full system initially has rod worths varying from about 4.8% for the center rod to about 6.4% for a side rod. After a plutonium zone burnup of 0.006, the center rod worth has dropped to only about 0.2% and a corner rod worth is about 7.7%. At the same time, β_{eff} has increased from an initial value of about 0.0053 to about 0.0066. Thus, maximum rates of reactivity addition for a rod being withdrawn at the building crane rate of 180 in./min could have values ranging from ~\$0.3/sec to over \$1/sec. For the higher rates of addition, although the Doppler reactivity feedback reduces the magnitude of the event, an energy release sufficient to produce fuel melting would be reached in two to three seconds. This time scale is so short that melting would probably occur before void formation could appreciably influence the incident.

In the case of jerking out a jammed rod with the building crane, if one assumes a linear worth for the rod and 0.2 sec for ejection,

reactivity insertions ranging from about \$1.5/sec to ~\$50/sec would be possible, again depending upon the configuration and rod involved. The total energy release for the higher rates would lead to extensive core melting, although complete melting of all 147 elements would probably not occur since this would require ~3700 MW-sec.

The excursion could be terminated by expulsion of steam, resulting in a mechanical destruction of the core. The possibility of a zirconium-water chemical reaction also exists. The extent of such a reaction would depend upon the mechanism by which the molten zirconium was dispersed in the water remaining after the incident occurred. On the other hand, the improved corrosion resistance of the oxide fuel compared with metal fuel is a distinct advantage for the proposed experiment in reducing the extent of possible metal water reactions.

It can be concluded that the reactivity accidents considered here involving rapid control rod withdrawals present no hazards greater than those previously recognized in EBWR.

3. Maximum Credible Accident

A very serious accident could occur without any nuclear violation. This accident would involve cracking of the reactor vessel on a vertical "seam" while the reactor is operating at full power.

The rupture of the pressure vessel would result in flashing of the high temperature water to steam. This would, however, not result in failure of the containment shell since, as has been pointed out in previous analyses,¹ the 400,000 cubic foot containment volume is sufficient to accommodate the internal pressure resulting from flashing all of the high-pressure water to steam. Assuming the worst case in which all of the water in the reactor and additional primary system equipment (41,000 lb) was 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 with about 30% of the water in the vapor phase.³ A decrease in pressure following the initial pressure buildup would result because of heat transfer to the cold building and equipment surfaces. Actuation of the building spray system would increase the rate of pressure decrease.

After loss of the coolant, the bare core would eventually melt due to the decay heat following power operation. A considerable interval of time would elapse, however, before the melting point of the fuel is reached. For example, if heat losses are ignored, it is estimated that more than two hours would be required to reach the oxide fuel melting point following extended 40 MW operation. Heat transfer involving conduction or

radiation would, of course, lengthen the time interval between loss of coolant and core melting. Until core melting occurs, the decay heat does not greatly contribute to supplying energy to the containment building except for heat losses from the vessel.

The above sequence of events is judged to be the "maximum credible accident" for the proposed experiment.

The same sequence of events was considered to lead to a potential "maximum credible" accident for the 100 MW operation.³⁹ However, owing to the use of oxide fuel, the consequences of this accident would be much less severe for the proposed experiment as discussed below.

As pointed out in previous analyses,⁸ the EBWR facility was designed to withstand the effects of chemical reaction of about 25% of the core metal, corresponding to an energy release of 8×10^8 cal (~ 3300 MW-sec). This assumes that the released hydrogen does not burn. For the proposed experiment, however, only the Zircaloy-2 cladding would be likely to be involved in such a reaction owing to the chemically bound form of the fuel.

The rapid reaction of as much as 50% of the Zircaloy-2 cladding is not believed to be credible. Many studies have been made on the subject of metal-water reactions⁴⁶⁻⁴⁹ substantiating this conclusion. Also, it is interesting to note that of the order of half of the metal-water reaction occurs before the cladding melts.^{46,47,49} This slow oxidation of the solid cladding significantly reduces the amount of material available for more rapid metal-water reactions after the cladding becomes molten.

If one makes the extremely pessimistic assumption that 50% of the Zircaloy cladding reacts with the water and that all of the hydrogen produced (both during the initial slow oxidation and the later more rapid reaction) burns, a total energy of about 8×10^8 cal would be released of which a significant fraction occurs slowly and therefore does not contribute to the likelihood of rupturing the containment. In any event, this amount of energy would result in a very small increase in pressure in the building since it will be essentially absorbed in the latent heat of vaporization of the water. For example, ignoring the addition of the spray system water, this energy release would only add about 2 psig to the building pressure. The pressure increase would be even less at later times owing to the addition of the cold spray water. As was pointed out before, the building spray system is available to reduce the building pressure differential. This 15,000 gallon spray system can supply room temperature water at a rate of 1000 gpm. After the initial 15,000 gallon supply is exhausted, addition of water continues at a rate of 100 gpm as provided by the laboratory water supply.

Analysis of the pressure and temperature history following the maximum credible accident assuming flashing of all primary system water,

release of 8×10^8 cal at time zero, and inclusion of decay heat, indicates that with the action of the spray system, the building pressure would fall below 15 psig in less than 2 min and below about 8 psig in about an hour. In the absence of the spray system, the building pressure would be about 15 psig or less and would decrease at a slow rate due to heat losses to the surroundings. In any event, rupture of the containment is not of concern for the present experiment.

a. Radiological Hazards in the Surrounding Area

Previous analyses in Appendix D of ANL-5781¹ concerning radioactive release were performed both for the case of rupturing the containment shell, and for the much less serious situation, corresponding to the present case, of slow discharge from the building. The unique aspect of the present experiment is that rupture of the containment shell is not believed to be credible. In the case of the previous metal fueled core, such rupture was considered to be credible in the event of an extensive metal water reaction.

In regard to the calculation of slow radioactive release from the containment shell, the previous analysis¹ assumed a leak rate based only on pressure differences due to barometric fluctuations of the order of 1% of ordinary atmospheric pressure. This low internal pressure was the result of the action of the building spray system following the postulated incident. The resultant leak rate of 100 cubic ft/day¹ is based on the pessimistic assumption of turbulent flow in which case leakage is taken to be proportional to the square root of the pressure differential. If the leakage points produced streamline flow, the leakage rate would be smaller. The 100 cubic ft/day value is also based on the maximum allowable leak rate of 1000 cubic ft/day at 15 psig. The building leak rate is periodically experimentally measured by pressurizing the building to 15 psig.

Although the spray system would act to reduce the internal pressure following the postulated maximum credible accident, a conservative approach to the radioactive release problem will be adopted here. It will be assumed that for the present experiment a leak rate of 1000 cubic ft/day (0.25% per day) is appropriate. This would correspond to taking no credit for a pressure decrease following the incident, and is a pessimistic assumption. It will be assumed therefore that the results presented in Appendix D of ANL-5781¹ for the case of slow leakage should be multiplied by 10 for the present experiment. These earlier calculations will be referred to now in regard to their relevance to the present experiment.

The results in Tables XVII-XX of Appendix D of ANL-5781¹ concerned with slow release were based on a 100 cubic ft/day leak rate,

operation at 20 MW, and mixing of the entire core fission products in the building air. No credit was taken for settling of particles owing to confinement of the air in the building or for the fact that the spray system would wash out some of the isotopes.

Recent studies⁵⁰⁻⁵³ have shown that the previous assumption of 100% release of fission products following fuel melting is very unrealistic. Thus, for example, in a report written by a group from the Division of Licensing and Regulation of the AEC,⁵⁴ it was assumed that fuel melting would release 100% of the noble gases, 50% of the halogens, and 1% of the solids in the fission product inventory to the building. This represents approximately 15% of the gross fission product inventory.⁵⁴ Therefore the results given in the above mentioned tables after multiplying by 10 to account for the higher leak rate, are conservative estimates for the present experiment even assuming operation at a power as high as 70 MW. However, the data pertinent to "immediate" release are not applicable for the postulated incident since a considerable time interval will elapse between rupture of the pressure vessel and release of radioactive material. The latter will occur, of course, only following a core meltdown.

It should be noted that in regard to plutonium release considerations, the present core plutonium inventory will be less than assumed earlier.¹ For example, at the beginning of life, the core will contain 13.3 kg of Pu²³⁹ and 1.2 kg of Pu²⁴⁰, and after a 9000 MWd/tonne burnup, requiring about 3 years of operation, will contain about 15.5 kg of Pu²³⁹, 4 kg of Pu²⁴⁰, and 1.3 kg of Pu²⁴¹. These values can be compared with the assumed inventory (Table XXI of Ref. 1) of 22 kg of Pu²³⁹, 6.4 kg of Pu²⁴⁰, and 0.94 kg of Pu²⁴¹ for the previous analysis. Also the previous total uranium inventory was taken as 5500 kg, whereas the present core contains about 3900 kg of uranium.

As pointed out before, interest in the present experiment is not expected to extend to burnups much beyond about 9000 MWd/tonne. This is due both to the long times involved and to the fact that the isotopic variation with burnup will tend to reach an asymptotic behaviour and, hence, result in less information of additional interest as burnup proceeds. Thus, even though from gas space considerations the fuel has been designed for a maximum exposure of 25,000 MWd/tonne, the plutonium inventory expected during the course of the present experiment would be considerably less than the inventory assumed for the earlier calculations.¹

The transport of the PuO₂ in the containment shell following the core meltdown is not expected to be different from that of the UO₂. This is due to the method of fabrication of the fuel which involves the high temperature, high pressure impacting process. As discussed above, the process fuses the two oxides and produces a completely uniform and homogeneous dispersion of the PuO₂ in the UO₂. This fabrication technique should be

clearly distinguished from a simple mechanical blending process in which the PuO_2 and UO_2 could classify or segregate. After fuel melting an even more uniform solution of the PuO_2 in the UO_2 could be expected.

Knowledge of the exact details of the dispersal of the core material are not required since results obtained assuming total release of the core material still lead to very small calculated inhalation dosages as indicated below.

The inhalation dosage hazard for the present experiment is very conservatively based on total release of the core material as in the previous hazards analysis.¹ The inhalation dosages presented there (Table XXIV of Ref. 1) concerned with slow release are however considerable overestimates with regard to the present core. This is due to the decrease in plutonium and uranium inventories, and to the very pessimistic previous assumption¹ of total release of the core material following core meltdown. Although information regarding release of Pu and U is not available, it is reasonable to assume that only 1% of the solids in the fission product inventory will be released.⁵⁴ Also, as before, no credit is taken for settling or washing out due to the spray system.

For convenience, the previous results given in Table XXIV of Ref. 1 are presented here in Table 19. These previous results, based on

TABLE 19. Inhalation Dosages (Rems) Calculated for Steady-state Leakage Case (1000 cubic ft/day)^a

Isotope	Distance, meters			
	25	200	1600	16,000
<u>Bone Dosages</u>				
Sr^{89}	429	19.5	0.82	0.03
Sr^{90} - Y^{90}	4290	195.	8.2	0.27
Pu^{239}	8.6	0.37		
Pu^{240}	8.6	0.37		
Pu^{241} - Am^{241}	2.6	0.12		
<u>Lung Dosages</u>				
Pu^{239} and Pu^{240}	2420	105	4.7	0.15
Am^{241}	975	42.9	1.9	0.06
<u>Thyroid Dosages</u>				
I^{131}	3630	160	7.0	0.22

^aObserver resides at the center of the streamer for 1 hr.

the originally assumed inventory (Table XXI of Ref. 1), have been multiplied by 3900 to account for the number of kg of uranium in the proposed core and by a factor of 10 to account for the assumed higher building leakage rate. All numbers have been converted to Rems assuming a RBE of 10 for the alpha emitters and 1 otherwise. The previous bone dosages are assumed to have been given in rads.

Owing to the very pessimistic assumption of complete release of the core material, these results for inhalation dosages must be considered to be severe overestimates of the expected situation for the present experiment.

b. Direct Radiation Following a Maximum Credible Accident

The direct radiation dosages and dose rates which might be encountered external to the containment building following occurrence of the maximum credible accident have been estimated assuming a 50% fission product release and uniform distribution of the released fission products within the shell. Any settling or washing out due to the spray system would result in significant dose rate reductions since half of the containment shell is below grade level. Table 20 presents some typical results assuming no such settling and for an assumed 42 MW operation.

TABLE 20. Some External Dosages (R) and Dose Rates (R/hr)
Following the Maximum Credible Accident

Time after Incident	Juggernaut		Bldg. 310		CP-5	
	Dose Rate	Dose	Dose Rate	Dose	Dose Rate	Dose
1 min	252	-	25	-	1	-
10 min	154	32	15	3.2	0.6	0.1
1 hr	91	126	9	13	0.4	0.5
10 hr	46	-	5	-	0.2	-
1 d	35	1260	3	126	0.1	5
10 d	17	-	2	-	0.07	-
100 d	5	-	0.5	-	0.02	-

It should be noted that Table 20 assumes release of fission products to the building at time zero. Actually such radioactive release will occur only after core meltdown occurs.

Although the results presented in Table 20 overestimate the severity of the expected exposures, it would no doubt be necessary to initially evacuate laboratory personnel from the 300 area until a comprehensive survey of dose rates in the various buildings could be completed and then to establish controlled access to the area.

D. Conclusions

In considering the safety of a plutonium loaded reactor, the two distinguishing areas of concern are related to the potential hazard associated with dispersal of the plutonium and the effect on the kinetic behaviour due to the reduced delayed neutron fraction. However, it has been seen elsewhere in Section VI that the proposed system and previous EBWR core are closely related in regard to plutonium content and effective delayed neutron fraction, as well as plant configuration. As a consequence, the response of the proposed system to various abnormal conditions is in most cases similar to that previously expected during the 100 MW operation. The proposed system has distinct advantages over that of the previous system owing to the exclusive use of oxide fuel as compared with the previous metal fuel. These advantages are obvious in the ability of the proposed system to sustain larger step reactivity insertions without damage to the core, the greater corrosion resistance of the oxide vs. metal type fuel, and the greater protection against plutonium dispersal after a fuel rod failure since the plutonium is chemically bound in the present fuel.

Although the various reactivity coefficients for the proposed experiment differ in value from previous loadings, the same qualitative behaviour is to be expected for the various abnormal conditions previously discussed.³ That is to say, events which previously did not lead to serious core damage will again not be damaging for the present core. The response of the system for a given event will be quantitatively different. For example, normal rate of withdrawal of a control rod following a violation of the eight rod shutdown condition would result in a slow increase in power for the present system as was the case for the 100 MW operation. The final steady power will depend upon the conditions involved and will be different for the present system as compared with previous experience owing for example to the different void coefficients of reactivity in the two cases. The qualitative results however are the same. These same general comments hold for all cases involving slow changes in reactivity whether due to changes in control rod position, boric acid content, or water temperature.

Events which previously would have caused core damage may now not be damaging. For example, owing both to the improved inherent safety of the present system and to the decreased fuel element worth as compared with the 100 MW operation, the consequences of dropping a fuel element into the proposed core would not result in serious core damage whereas such an event would exceed the inherent shutdown capabilities of the 100 MW core and would result in melting of the metal fuel. In either case, a loading error or violation of the shutdown margin would have to precede the incident in order for an accident to be possible. As before, dropping of a fuel element would require gross failure of either the fuel element structure or of the fuel handling mechanism. Of course even the improved inherent

safety of the oxide core cannot prevent core damage for arbitrarily large reactivity insertions. Thus, the incident postulated and analyzed for the 100 MW operation which involved very rapid removal of a stuck control rod using the building crane would also lead to serious damage for the present core.

It can be concluded that owing to improvements in the plant and to the fact that an oxide core is involved, together with the benefit of experience and confidence gained during previous EBWR operation, and to the similarity of the proposed and previous system, that the plutonium recycle experiment is as safe as the system previously encountered in EBWR and in fact represents a significantly more reliable reactor-plant configuration.

APPENDIX A

Stability

Stability measurements were made on EBWR as part of initial operation.^{55,56} From data obtained at 5, 20, 30, and 50 MW extrapolations were

made, as shown in Fig. 26, which predicted the onset of instability at a power level of about 66 MW.⁵⁷ The reactor was subsequently operated at 61.7 MW, which was the maximum power level attainable based on feed-water pump capacity.

The 61.7 MW power level was used to extrapolate to 80 MW by increasing the number of fuel elements in the ratio of 147/114.⁴ The 80 MW power was further extrapolated to 100 MW by assuming an increase in core height from 4 to 5 ft.

Stability measurements were repeated as part of the 100 MW program. Data taken at 10, 20, 40, 60, and 71 MW were extrapolated, as shown in Fig. 27, and 120 MW was estimated as an upper limit for stable operation. The reactor was subsequently operated at 100 MW.⁵

Consideration was given to the effect of plutonium on EBWR kinetics and was reported in EBWR Test Report 42B.⁵⁸

The stability of EBWR will be investigated as part of the Plutonium Recycle Program using the same techniques and equipment as for the 100 MW program. Transfer function measurements will be made at several low power levels to ob-

112-3623

RE-7-30574-B

Fig. 26. Reactor Gain and Phase Stability for
Original 4 ft dia EBWR Core

tain sufficient data which can be extrapolated to determine an upper power limit for stable operation. Transfer function measurements will be made at power at selected intervals in time to determine any decrease in stability due to changes in power distribution.

112-2642

Fig. 27. Reactor Gain and Phase Stability for 5 ft dia EBWR Core 1A

APPENDIX B

Radiation Damage in the EBWR Pressure Vessel

The fast neutron exposure rate at the EBWR vessel wall has been determined, and from it, extrapolations have been made of the integrated exposure at the end of the reactor program. These rates were calculated, verified by foil data, and checked against data from actual specimens exposed in EBWR. While not true surveillance specimens (they were located in the reflector and not right at the vessel wall) they were exposed at the same time and to the same temperature conditions.

Data Available on A212B

Figure 28 is a composite of impact test data on A212B carbon steel (similar to EBWR vessel material).^{59,60} The neutron exposure is expressed in RDU's. This unit is explained and defined in ANL-6826.⁶⁰ Fluxes in terms of neutrons >1 MeV would be somewhat less but of the same order of magnitude. Data are from Izod impact bars for irradiations at below 150°C (300°F) in various reactors. The two NRL points are from standard Charpy V-notch impact tests, and they are indicative of a family of NRL data points, some of which show roughly the same slope as the other data.⁶¹ Thus the Izod data can be used along with the more standard Charpy data, although a safety margin will have to be allowed because the Charpy points appear to fall above the rest of the data.

112-3676

Fig. 28. Trend Line for Low Temperature
($<300^{\circ}\text{F}$) Impact Test Data

on a limited number of impact tests, and often two or more choices of the transition temperature look reasonable. For data from other laboratories this uncertainty estimate is not available. These flags indicate a "probable error" range, rather than the maximum possible error.

Uncertainty

The uncertainty in these measurements is larger than one would wish. An effort has been made to express this by using error range marks on data points. On ANL data the exposure uncertainty is well below $\pm 5\%$ so this is covered by the width of the data points on the graph. Because of the methods used to report neutron exposure, a sizeable uncertainty exists when the attempt is made to plot various data together. Note the exposure error band on the NRL data.

The ordinate point is based

EBWR Impact Test Results

The solid black circles indicate results from irradiations performed in EBWR. The samples used were Izods like the other ANL data, but cut transverse to the final rolling direction of the plate. (The other specimens were longitudinal.) The scatter of impact test data is great, as indicated in Table 21. Control specimens in a location in the vessel far from the core

TABLE 21. EBWR Impact Test Data

Exposure, RDU	TT Shift,* °F	Net Shift, °F
~0	22^{+50}_{-5}	0
1.92×10^{17} ($+1.72 \times 10^{17}$)**	28^{+40}_{-5}	6^{+0}_{-6}
2.72×10^{18}	100^{+45}_{-45}	78^{+0}_{-45}
2.81×10^{18} ($+1.38 \times 10^{18}$)**	125^{+40}_{-0}	103
3.38×10^{18}	136^{+40}_{-25}	114^{+0}_{-25}

*Based on unirradiated TT of -100^{+40}_{-0} .

**Maximum possible RDU contribution due to thermal flux.

showed a shift of 20-65°F. There is some evidence that specimens irradiated at 450-500°F develop greater transition temperature shifts than at room temperature. To check the EBWR irradiation results with the others, the 20-65°F increment was subtracted, the data added to Fig. 28, and satisfactory agreement was obtained.

Two of the capsules were in borated steel shrouds, and hence saw no thermal flux. The others have an error tail representing the possible contribution of thermal neutron capture recoil to the total damaging exposure. Because of self-shielding the estimation of the thermal flux is difficult.

EBWR Vessel Exposure and TT Shift

Izod specimens and dosimeters were irradiated in tubes located in the reflector about 9 in. closer to the core than the inner wall of the vessel. The spectra were calculated. It was determined that:

- 1) The ratio of RDU/sec to the Mn^{54} activation rate was 9.783.
- 2) The ratio of RDU rates at the dosimeter and specimens to the rate at the pressure vessel wall is 13.4.

The measured activation rate at the specimens for the fast neutron reaction $\text{Fe}^{54} (n, p) \text{Mn}^{54}$ is 1.12×10^9 act/sec/ 10^{24} Fe^{54} atoms/MW. Therefore:

Exposure rate at dosimeter and specimens = 1.09×10^{10} RDU/MW-sec

Exposure rate at pressure vessel wall = 8.17×10^8 RDU/MW-sec.

In Table 22 the operating history of the reactor is summarized. These milestones can be located on a plot of transition temperature shift vs. exposure. In Fig. 29 the line has been drawn about 60° above the composite line from Fig. 28 to allow for thermal history effects and possible greater shifts of Charpy tests as compared to Izods. This conservative limit gives a total shift of about 130°F at the worst point in the vessel at the end of the entire proposed reactor life.

TABLE 22. EBWR Operating History

	Integrated Power, MW-sec	Maximum Integrated Exposure at Reactor Vessel Wall, RDU
July 1959		
100 MW Conversion	7.61×10^8	6.22×10^{17}
December 1962		
Pu Recycle Conversion	1.06×10^9	8.65×10^{17}
Pu Recycle Operation		
20 MW-yr	1.69×10^9	1.38×10^{18}
40 MW-yr	2.32×10^9	1.90×10^{18}
80 MW-yr	3.58×10^9	2.93×10^{18}

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Fig. 29 Predicted Transition Temperature Shift for EBWR Pressure Vessel Wall

Discussion

Safety Implications

The entire analysis above deals with the worst point in the reactor system. It considers the belt region at the inner wall of the reactor vessel where the flux is the maximum. No credit is being taken for the fact that the flux is attenuated exponentially through the vessel wall itself. At the outer wall the exposure is down to about 65% of that at the inner wall.⁶² The attenuation means that the stated transition temperature shift is always conservative.

The significance of the measured brittle-ductile transition temperature in regard to the brittle behavior of a structure is complex and not

well settled today. According to Pellini et al.⁶³ a structure is always safe at NDT + 60°F, although in the absence of flaws in the material, the structure would be safe even at NDT.⁶⁴

The measured NDT for EBWR steel by Charpy V-notch was originally about -20°F and tests of material removed from the EBWR vessel wall in 1959 (low irradiation exposure regions) revealed that the NDT of the vessel wall material was about 30-40°F. The change from the -20°F measured for the original A212B material is due to fabrication and accumulated operation history without including the effects of radiation. Thus, since from Fig. 29 the radiation induced shift to date is about 30°F, NDT + 60 is now no greater than +130°F at the worst point.

The transition temperature shift data from the Izod specimens are considered applicable because good correlations appear in the data, and other tests show that the observed shift does not differ greatly for different testing methods. (The Charpy V and NRL Drop Weight Test gave comparable shifts.) The transition temperature given by the Izods is about -150°F, but this cannot have any safety implications.

Schnadt coheracy tests on unirradiated A212B show the steel to be capable of arresting a running crack above 75°F and capable of resisting the initiation of a brittle crack below about 15°F.

Clearly no danger of brittle failure is anticipated at reactor operating temperatures. However, during hydrostatic tests, higher stresses are involved. At some point in the program, it may become desirable to perform hydrostatic tests above room temperature. This is easily done with existing facilities. The test temperature will be established based on reactor operating history and the latest available experimental data on radiation damage.

Considering again the worst possible point, and conservatively saying that NDT + 60 has now reached 130°F there, after 80 MW-yr of operation this temperature would be calculated to be 230°F. By that time data should be available which would permit more realistic appraisal of the situation. Some of this information will be forthcoming from samples irradiated in the tubes used previously in EBWR.

Due to the conservative nature of the above calculations, it is believed that no temperatures above about 180°F will be required for safe hydrostatic testing throughout the course of the experiment, and that initially such tests may be conducted at room temperature.

APPENDIX C

Characteristics of Special Experimental Fuel Rods for the Plutonium ZoneGeneral Comments

The special fuel rods to be used are:

- 1) Natural Uranium Dioxide.
- 2) Mixed Plutonium-Uranium Dioxides with varying Pu^{240} concentrations.
- 3) Plutonium-Aluminum Alloy of varying Pu^{240} concentrations.

All fuel rods will be of the same nominal dimensions as the regular $\text{PuO}_2\text{-UO}_2$ EBWR rods, and clad in the same manner with Zircaloy-2. These rods are to be inserted in various positions in the plutonium zone. The total number of such rods is small (~24) compared to the total loading (1296) and their effect on the total reactivity of the mixed oxide zone is very small. The compositions have been chosen to match the composition (atom ratios) present in standard rods at the various stages of burnup, and the total fuel concentration will have heat generation rates equal to or less than that of the standard elements. This is accomplished by matching the thermal values of ηf in the experimental rods to those of the standard rods. The specifications of the special rods are given in Table 23.

TABLE 23. Specifications of Special Experimental Rods

No. Rods	Type	Enrichment	% Pu^{240}
6	$\text{PuAl}^{(a)}$	3.35 w/o	8
6	$\text{PuAl}^{(a)}$	3.35 w/o	26
4	$\text{UO}_2^{(b)}$	Natural	-
6	$\text{PuO}_2\text{-UO}_2^{(b)}$	1.5 w/o PuO_2 in Depleted UO_2	~20
6	$\text{PuO}_2\text{UO}_2^{(b)}$	1.5 w/o PuO_2 in Depleted UO_2	~26

(a) Density = 2.7 gm/cc. Rods are 4 ft long. Clearance gap of ~0.003 in. between fuel and clad.

(b) 88% theoretical density. Rods are normal EBWR length (~48.5 in.) and have specifications as per the reactor grade mixed oxides.

Specific Comments

Natural Uranium Oxide Elements

These elements have heat generation rates which are less than the standard elements and so they produce no hot spot. They contribute a small gain in negative Doppler coefficient owing to the higher U^{238} concentration than for the standard rods.

Mixed Uranium-Plutonium Oxide Elements

The matching of heat generation rates with those of the standard fuel rods assures the absence of hot spots. These elements will contain higher concentrations of Pu^{240} which will contribute to the negative Doppler coefficient.

Plutonium-Aluminum Alloy Elements

These elements are sufficiently different from the standard loading to require special consideration. The compositions are chosen such that the heat generation rates are comparable to the standard loading initially and the absence of fertile material will cause the elements to become "cold" elements as burnup proceeds.

The possibility of rupturing the Zircaloy-2 cladding due to "ratcheting" under thermal cycling of the rods can be negated by a proper gap between the fuel alloy and clad. Experience with the PRTR (Plutonium Recycle Test Reactor) has shown⁶⁵ that for the proposed elements under similar operating conditions a gap of from 0.004 to 0.007 in. is adequate. As shown in Table 23, a gap of ~0.003 in. has been selected for the PuAl rods for EBWR. A heat generation calculation shows the center temperature of the alloy will be about 700°F in EBWR.

Under excursion conditions it has been shown⁶⁶ that for cores loaded with PuAl alloy elements in light water and having a negative moderator void coefficient, the very rapid heat transfer of the alloy results in boiling at the fuel surface which tends to shut off the excursion. However, since these rods are driven by the remainder of the core, the possibility exists of their power level being forced to such levels as to produce fuel melting and/or clad rupture with subsequent metal-water reactions occurring. Previous experimental studies⁶⁷ of water-metal reactions indicate that it is unlikely that the reaction would go beyond 10% completion. However, even if we assume that such a reaction goes to 100% completion, the additional energy would be only a small fraction of the total energy which would be generated by nuclear reactions during a severe transient. (See Section III.G.5) There are only 924 gm of aluminum alloy in the 4 rods proposed

(~34 moles). For the reaction $2\text{Al} + 3\text{H}_2\text{O} \rightarrow \text{Al}_2\text{O}_3 + 3\text{H}_2 + \Delta\text{H}$, the heat release is 880 Btu/mole and hence only about 30,000 Btu (~32 MW-sec) would be released if the reaction goes to completion.

In summary, the special rods present no special hazards, unique to themselves, to the safe operation of the EBWR.

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REFERENCES

1. J. M. West *et al.*, *Hazard Summary Report on the Experimental Boiling Water Reactor (EBWR)*, ANL-5781.
2. J. M. Harrer *et al.*, *Performance Evaluation of Direct-cycle Boiling Water Nuclear Power Plants Based on Recent EBWR-Borax Data*, Proc. Second UN Intn'l. Conf. on Peaceful Uses of Atomic Energy, Geneva, Switzerland, 9, 264 (1958).
3. E. A. Wimunc and J. M. Harrer, *Hazards Evaluation Report Associated with the Operation of EBWR at 100 MW*, ANL-5781 Addendum (Revision 1) (Oct 1960).
4. R. Avery *et al.*, *EBWR Core 1A Physics Analysis*, ANL-6305 (Feb 1961).
5. E. A. Wimunc *et al.*, *Performance Characteristics of EBWR from 0-100 MWt*, ANL-6775 (Sept 1963).
6. V. Kolba, *EBWR Test Reports (Report No. 103)*, ANL-6229 (Nov 1960).
7. *EBWR Test Reports (100 MWt Operation)* ANL-6703, Report 100-I (Jan 1964).
8. *The EBWR Experimental Boiling Water Reactor*, ANL-5607 (May 1957).
9. *Modification of the Experimental Boiling Water Reactor (EBWR) for Higher Power Operation*, ANL-6552 (April 1962).
10. R. Avery and C. N. Kelber, *Physics Analysis of Proposals for EBWR Core 2*, ANL-6306 (1961).
11. April 1963 Monthly Report to the IAEA.
12. R. C. Daniel *et al.*, *Effects of High Burnup on Zircaloy-clad Bulk UO_2 , Plate Fuel Element Samples*, WAPD-263 (Sept 1962).
13. J. B. Melehan *et al.*, *The Effect of Nuclear Radiation on Ceramic Reactor-fuel Materials*, REIC-27 (June 30, 1963).
14. C. D. Joanou and J. S. Dudek, *GAM-1: A Consistent P1 Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants*, GA-1850 (June 28, 1961).
15. H. Amster and R. Suarez, *The Calculation of Thermal Constants Averaged over a Wigner-Wilkins Flux Spectrum: Description of the SOFOCATE Code* WAPD-TM-39 (Jan 1957).
16. D. A. Klopp and W. E. Zagotta, *Resonance Integral Determination*, Progress Report No. A-1918-8, I.I.T. Research Institute, Chicago, Ill. (March 1962).
17. B. Carlson *et al.*, *The DSN and TDC Neutron Transport Codes*, LAMS-2346 (Feb 1960).
18. H. Honeck, *Some Methods for Improving the Cylindrical Reflecting Boundary Condition in Cell Calculations of the Thermal Neutron Flux*, Trans. Am. Nuclear Soc. 5, 350 (Nov 1962).
19. M. Butler and J. M. Cook, *Univac Programs for the Solution of One-dimensional Multigroup Reactor Equations*, ANL-5437 (Oct 1956).
20. B. J. Toppel *et al.*, *CYCLE and COST, Codes for Fast Reactor Fuel Cycle Analysis and Related Cost Evaluation*, Trans. Am. Nuclear Soc. 5, 92 (June 1962).

21. G. Bilodeau *et al.*, *PDQ, an IBM-704 Code to Solve the Two-dimensional Few Group Neutron Diffusion Equations*, WAPD-TM-70 (August 1957).
22. W. Cadwell, *PDQ-3, a Program for the Solution of the Neutron Diffusion Equations in Two Dimensions on the IBM-704*, WAPD-TM-179 (March 1960).
23. M. L. Tobias and T. B. Fowler, *The Twenty Grand Program for the Numerical Solution of Few-group Neutron Diffusion Equations in Two Dimensions*, ORNL-3200 (Feb. 21, 1962).
24. Unpublished ANL Code for the IBM-704.
25. B. Alder *et al.*, *Methods in Computational Physics, Volume I, Statistical Physics*, Academic Press, N. Y., 1963.
26. Reference 6, page 256.
27. J. L. Russel, Jr., *et al.*, *Absorber Burn-up Experiment Final Report*, GEAP 4017 (April 1962).
28. L. C. Kvitek, *Multigroup Calculations of Effective Delayed-neutron Fractions, Prompt-neutron Lifetime, and Related Kinetics Parameters*, ANL-6511 (March 1962).
29. Unpublished ANL code for the IBM-704.
30. *Reactor Physics Constants*, ANL-5800 Second Edition, 815 (July 1963).
31. D. F. Schoeberle *et al.*, *A Method of Calculating Transient Temperatures in a Multi-region, Axisymmetric, Cylindrical Configuration*, ANL-6654 (Nov 1963).
32. G. R. Keepin *et al.*, *Delayed Neutrons from Fissionable Isotopes of Uranium, Plutonium and Thorium*, Phys. Rev. 107, 1044 (1957).
33. R. Batchelor and H. R. McK. Hyder, *The Energy of Delayed Neutrons from Fission*, J. Nuclear Energy 3, 7 (1956).
34. Reference 30, page 11.
35. A. H. Spano *et al.*, *Self-limiting Power Excursion Tests of a Water-moderated Low-enrichment UO_2 Core in SPERT I*, IDO-16751 (Feb. 28, 1962).
36. R. Scott, Jr., *et al.*, *Transient Tests of the SPERT I Low-enrichment UO_2 Core: Data Summary Report*, IDO-16752 (Sept. 3, 1963).
37. J. E. Houghtaling *et al.*, *Calculation and Measurement of the Transient Temperature in a Low-enrichment UO_2 Fuel Rod during Large Power Excursions*, IDO-16773 (May 18, 1962).
38. C. R. Tipton, Jr., *Reactor Handbook Volume 1 Materials*, 2nd Edition, Interscience Publishers Inc., New York, 1960, p 293.
39. Letter from B. I. Spinrad to D. R. Muller dated March 7, 1961.
40. T. R. Bump and R. W. Seidensticker, *Analysis of Temperatures and Excursions Resulting from Exponential Power Changes in a Reactor*, Nuclear Science and Eng. 4, 44 (1958).
41. S. Timoshenko and J. N. Goodier, *Theory of Elasticity*, McGraw Hill Book Co., 2nd Ed. (1951).
42. M. Petrick and E. A. Spleha, *Thermal Hydraulic Performance Characteristics of EBWR (0 to 100 MW)*, ANL-6693 (May 1963).

43. Reference 7, Test Report 117 B.
44. *Nuclear News* 12, 30 (Dec 1963).
45. R. E. Peterson, *Time Delay in the Doppler Effect Resulting from Inhomogeneities in Mixed PuO_2 - UO_2 Fuels*, HW-78711 (Nov. 11, 1963).
46. B. Lustman, *Zirconium-Metal Reactions*, WAPD-137 (Dec. 1, 1955).
47. J. I. Owens et al., *Metal-Water Reactions: VIII. Preliminary Consideration of the Effects of a Zircaloy-Water Reaction during a Loss of Coolant Accident in a Nuclear Reactor*, GEAP-3279 (Sept. 30, 1959).
48. R. C. Liimatainen et al., *Studies of Metal-Water Reactions at High Temperatures. II. TREAT Experiments Status Report on Results with Aluminum, Stainless Steel-304, Uranium, and Zircaloy-2*, ANL-6250 (Jan 1962).
49. L. Baker, Jr., and L. C. Just, *Studies of Metal-Water Reactions at High Temperatures. III. Experimental and Theoretical Studies of the Zirconium-Water Reaction*, ANL-6548 (May 1962).
50. G. E. Creek et al., *Experiments on the Release of Fission Products from Molten Reactor Fuels*, ORNL-2616 (July 22, 1959).
51. G. W. Parker et al., *Fission Product Release from UO_2 by High Temperature Diffusion and Melting in Helium and Air*, ORNL-CF-60-12-14 (Feb 1961).
52. G. W. Parker et al., *Release of Fission Products on Out-of-pile Melting of Reactor Fuels*, ORNL-3401, pp 5-27 (Feb. 25, 1963).
53. W. E. Browning, Jr., et al., *Release of Fission Products during In-pile Melting of UO_2* , Nucl. Sci. Eng. 18, 151 (1964).
54. J. J. DiNunno et al., *Calculation of Distance Factors for Power and Test Reactor Sites*, TID-14844 (March 23, 1962).
55. J. A. DeShong, Jr., *Power Transfer Functions of the EBWR Obtained Using a Sinusoidal Reactivity Driving Function*, ANL-5798 (Jan 1958).
56. E. S. Beckjord, *Dynamic Analysis of Natural Circulation Boiling Water Power Reactors*, ANL-5799 (March 1958).
57. J. A. DeShong, Jr., and W. C. Lipinski, *Analyses of Experimental Power-Reactivity Feedback Transfer Functions for a Natural Circulation Boiling Water Reactor*, ANL-5850 (July 1958).
58. Reference 6, Test Report 42 B.
59. *Reactor Development Program Progress Report--May 1963*, ANL-6739, pp. 35-41.
60. A. D. Rossin, *Dosimetry for Radiation Damage Studies*, ANL-6826, (March 1964).
61. R. Hawthorne and L. Steele, *Irradiation Effects on Structural Materials Quarterly Progress Report--August 15, 1963*, NRL-1448.
62. K. Shure, *Neutron Spectral Effects on Interpretation of Radiation Damage Exposure*, WAPD-T-1662 (Sept 1963).
63. W. S. Pellini and P. P. Puzak, *Fracture Analysis Diagram Procedures for the Fracture-safe Engineering Design of Steel Structures*, NRL-5920 (March 15, 1963).

64. W. S. Pellini and P. P. Puzak, *Practical Considerations in Applying Laboratory Fracture Test Criteria to the Fracture-safe Design of Pressure Vessels*, NRL-6030 (Nov. 5, 1963).
65. Private Communications with Ralph Sharp, General Electric Co., Hanford, Washington.
66. W. K. Winegardner and N. G. Wittenbrock, *Plutonium Recycle Critical Facility Final Safeguard Analysis, Supplement 4, Analysis of Light Water Moderated Core Loading*, HW-69168 Suppl. 4 (Nov 1963).
67. H. M. Higgins and R. D. Schultz, *The Reaction of Metals with Water and Oxidizing Gases at High Temperature*, IDO-28000 (April 1957).