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**SUMMARY OF THE NUCLEAR DESIGN
AND PERFORMANCE OF THE LIGHT
WATER BREEDER REACTOR (LWBR)
(LWBR Development Program)**

JUNE 1979

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**BETTIS ATOMIC POWER LABORATORY
WEST MIFFLIN, PENNSYLVANIA**

Operated for the U. S. Department of Energy by
WESTINGHOUSE ELECTRIC CORPORATION

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SUMMARY OF THE NUCLEAR DESIGN AND PERFORMANCE OF
THE LIGHT WATER BREEDER REACTOR (LWBR)
(LWBR Development Program)

H. C. Hecker

June 1979

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FOREWORD

The Shippingport Atomic Power Station located in Shippingport, Pennsylvania was the first large-scale, central-station nuclear power plant in the United States and the first plant of such size in the world operated solely to produce electric power. This program was started in 1953 to confirm the practical application of nuclear power for large-scale electric power generation. It has provided much of the technology being used for design and operation of the commercial, central-station nuclear power plants now in use.

Subsequent to development and successful operation of the Pressurized Water Reactor in the DOE-owned reactor plant at the Shippingport Atomic Power Station, the Atomic Energy Commission in 1965 undertook a research and development program to design and build a Light Water Breeder Reactor core for operation in the Shippingport Station.

The objective of the Light Water Breeder Reactor (LWBR) program has been to develop a technology that would significantly improve the utilization of the nation's nuclear fuel resources employing the well-established water reactor technology. To achieve this objective, work has been directed toward analysis, design, component tests, and fabrication of a water-cooled, thorium oxide fuel cycle breeder reactor for installation and operation at the Shippingport Station. The LWBR core started operation in the Shippingport Station in the Fall of 1977 and is expected to be operated for about 3 to 4 years. At the end of this period, the core will be removed and the spent fuel shipped to the Naval Reactors Expanded Core Facility for a detailed examination to verify core performance including an evaluation of breeding characteristics.

In 1976, with fabrication of the Shippingport LWBR core nearing completion, the Energy Research and Development Administration established the Advanced Water Breeder Applications (AWBA) program to develop and disseminate technical information which would assist U. S. industry in evaluating the LWBR concept for commercial-scale applications. The program will explore some of the problems that would be faced by industry in adapting technology confirmed in the LWBR program. Information to be developed includes concepts for commercial-scale prebreeder cores which would produce uranium-233 for light water breeder cores while producing electric power, improvements for breeder cores based on the technology developed to fabricate and operate the Shippingport LWBR core, and other information and technology to aid in evaluating commercial-scale application of the LWBR concept.

FOREWORD (Cont)

All three development programs (Pressurized Water Reactor, Light Water Breeder Reactor, and Advanced Water Breeder Applications) have been administered by the Division of Naval Reactors with the goal of developing practical improvements in the utilization of nuclear fuel resources for generation of electrical energy using water-cooled nuclear reactors.

Technical information developed under the Shippingport, LWBR, and AWBA programs has been and will continue to be published in technical memoranda, one of which is this present report.

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This report presents a summary of the nuclear design and expected nuclear performance of the Light Water Breeder Reactor during operation at the Shippingport Atomic Power Station. Performance predictions are presented for core lifetime, breeding margin, power distributions and performance, kinetic and stability parameters, and for core shutdown and reactivity control capability. Also included is a summary of as-built dimensions of core components and of development of breeding parameter equations and sensitivities.

SUMMARY OF THE NUCLEAR DESIGN AND PERFORMANCE OF THE LIGHT WATER BREEDER REACTOR (LWBR)

H. C. Hecker

I. INTRODUCTION

This report presents a summary of the nuclear design and the expected nuclear performance of the Light Water Breeder Reactor (LWBR) during operation at the Shippingport Atomic Power Station. Core manufacturing data were input to the nuclear analysis calculations to provide the performance predictions for the as-built core prior to reactor operation. A series of periodic tests are being run during core operation to confirm these performance predictions. Comparisons of calculated core nuclear parameters with the results of nuclear tests, performed at beginning-of-life on the LWBR, are presented in Reference I-1.

Section II of this report provides a brief description of the LWBR core and fuel loading distribution. Principal differences between LWBR and other light water cores are also identified. Section III provides a summary of the analysis methods and model qualification. Also presented are design assurance practices and estimated uncertainties in reactivity and breeding predictions.

Performance predictions for the LWBR core presented in Section IV include data on lifetime and reactivity, breeding margin, power distributions and power performance, kinetic and stability parameters, and core shutdown and reactivity control capability. Section V presents conclusions concerning the best estimate for nuclear performance predictions.

Two appendices are included with this report. Appendix A contains a summary of as-built data for fuel rods, grids, and modules in the LWBR core. Development of breeding parameter equations and the effects of various core parameters on the magnitude of breeding are presented in Appendix B.

REFERENCE

- I-1. W. K. Sarber, "Results of Initial Nuclear Tests on LWBR," WAPD-TM-1336, June 1979.

II. DESCRIPTION OF LWBR CORE AND FUEL SYSTEM

A. Core Description

To achieve breeding so as to significantly improve fuel utilization, the LWBR core has been designed to minimize parasitic neutron absorption in core and structural materials.

Among the core design features which contribute to improved neutron economy in the LWBR are:

1. Use of movable fuel to control core reactivity, rather than conventional poison control rods, soluble poison, or burnable poison
2. Use of peripheral radial and axial thorium reflector blanket regions to reduce neutron leakage from the core
3. Use of Zircaloy with a low hafnium content (<40 ppm) for fuel rod cladding and for all structures in the active fuel region except the fuel rod support grids
4. Use of stainless steel (AM-350) rather than Inconel for fuel rod support grids.

Figure II-1 presents a cross-sectional view of the LWBR core showing the orientation of the basic types of modules which comprise the core configuration. As shown, the core contains the following principal types of fuel modules and fuel assemblies:

1. 12 axially movable seed fuel assemblies
2. 12 stationary blanket fuel assemblies composed of standard blanket and power-flattening blanket regions
3. 15 reflector blanket fuel modules.

The seed and blanket assemblies contain both fissile uranium-233 and fertile thorium-232, while the reflector blanket fuel modules contain only thorium. A hexagonal seed assembly and its surrounding annular blanket assembly form a seed/blanket module.

The three central fuel modules of the LWBR core are identical and symmetrical. They were designed in so far as practical to represent modules which could be used in a large central station reactor plant. The nine modules surrounding these central modules have a slightly larger outer blanket region that is fueled with

a higher uranium-233 content and smaller diameter rods than the inner blanket regions. Use of this more highly loaded blanket region, with a larger water volume fraction, produces a more uniform power distribution within the interior of the core, thereby better simulating the environment of a typical large core. This power flattening increases the uranium-233 loading required for the small core used at Shippingport. In a larger breeder core, power flattening could be accomplished with little or no increase in loading in exterior modules.

Surrounding the 12 hexagonal fuel modules in LWBR is an annular region, about 9 inches thick, made up of 15 reflector blanket modules. These serve to reduce neutron leakage from the relatively small Shippingport core to a value representative of a larger LWBR core in the 300- to 400-MW(e) range. Surrounding the reflector blanket fuel modules are 15 stainless steel, nonfuel filler units whose purpose is to limit core flow leakage by filling the space between the reflector modules and the core barrel.

To eliminate the parasitic loss of neutrons to conventional control rod poison materials, core reactivity control is achieved by varying the geometric relationship between the axially movable seed fuel assemblies and the associated stationary annular blanket assemblies which surround each seed assembly. For LWBR operation, this reactivity control is achieved by positioning the 12 movable seed fuel assemblies in a uniform bank using individual control drive mechanisms.

Changing the axial position of the seed assembly relative to the blanket assembly changes the relative amounts of neutron absorptions in the fissile (uranium-233) and fertile (thorium) fuel materials. The LWBR nuclear design is such that the more highly loaded seed has a k_{∞} greater than one and the lower loaded blanket has a k_{∞} less than one. Reactivity is controlled by varying the leakage of neutrons from the high k_{∞} seed regions into the low k_{∞} blanket regions. The reactivity worth of the movable seed is increased by using lengths of natural thorium in some of the seed and blanket rods; this concept is shown in Figure II-2. When the reactor is shut down, the seed assemblies are positioned below the rest of fuel in the core as shown in Figure II-2. To start up the reactor, the operator raises the seed assemblies in a uniform bank. By moving the seed fuel up, more nearly into alignment with the rest of the core, the operator brings the uranium-233 bearing parts of the fuel closer together.

The height of the fuel is shown in Figure II-4, which shows a Type I module with the seed aligned with the blanket. The height of individual fuel steps shown is 14 inches. At the beginning of core life, hot critical operation occurs with the tops of the seeds about 24 inches lower than the top of the stationary blanket. As the core operates, the seed assemblies are moved gradually upward until a position about 24 inches higher than the blanket is reached at the end of core life. From an operator's point of view, the movable fuel control system is similar to that of the movable poison rod control.

A typical fuel module cross section is shown in Figure II-3. Cylindrical fuel pellets are stacked inside hollow Zircaloy-4 cladding tubes and welded at both ends to solid end plugs to form the fuel rods. The fuel pellets are thorium (ThO_2) or binary (ThO_2 - $^{233}\text{UO}_2$) ceramic and are approximately 98 percent of their maximum theoretical density. A plenum region is provided in all rods to limit internal gas pressure buildup resulting from fission gases and volatiles released from the pellets during operation. A helical coil spring is installed in the plenum region at the top of the fuel stack to maintain the position of the pellet stack. The fuel zoning for a Type I module is shown in Figure II-4. The three central modules in the LWBR core are Type I. All 12 seeds are of identical design. The nine outer modules are Type II if there are two sides of power flattening blanket, and Type III if there are three such sides (Figure II-1). The fuel zoning in the 10-rod row power-flattening blanket region, shown in Figure II-5, is similar to the regular blanket for the inner five rows, and has high zone fuel in the outer five-rod rows.

The fuel rods are supported by a system of grids, which are made of AM-350 stainless steel, axially positioned in each fuel assembly. Each assembly contains a Zircaloy shell which provides both a structural member and a defining boundary. The seed and reflector shells surround the outer perimeter of these components. The blanket shell (guide tube) is located on the inside of the annular blanket assembly and provides a path for movement of the movable fuel assembly. Support posts for grids in each assembly are located along the outer boundary of the fuel rod lattice, as shown in Figure II-3. For seeds and reflector blankets, these posts are just inside the shells and are connected to the shells.

There are structural metal pieces above and below the fuel assemblies which provide support, and form regions having a different metal/water ratio. These regions are important in nuclear analysis due to their relative positions as the seed fuel assemblies are moved axially to positions above or below the stationary blanket assemblies (Figure II-2).

Details of the fuel lattice geometry and fuel loading distribution are presented below in Section B. A summary of as-built data for fuel rods, grids, and modules in the LWBR core is presented in Appendix A. A more comprehensive description of the LWBR core is given in Reference II-1.

B. Fuel System and Core Loading Distribution

The LWBR core is fueled with thorium-232 and uranium-233. Capture of neutrons in thorium-232 followed by the β decay of thorium-233 and of protactinium-233 results in the production of uranium-233. An objective, therefore, was to maximize neutron captures in thorium-232 so that the production rate of fissile uranium would exceed the fissile fuel destruction rate. The modular design, with a central seed region containing a relatively high uranium-233 weight percent surrounded by a blanket region with a somewhat lower uranium-233 weight percent, was developed as a means of providing a movable fuel reactivity control method which does not require the use of control poisons.

Selection of the uranium-233 loading distribution between seed and blanket included other considerations. Excess reactivity was balanced between seed and blanket to yield adequate lifetime with a critical movable fuel position at beginning-of-life not too far below the aligned position. This is desirable since the neutron losses increase with seed-blanket axial fuel displacement from the aligned position (Figure II-2). At the same time, the movable fuel reactivity worth as a function of position was designed to provide adequate shutdown capability.

Specification of uranium-233 loading variation in radial zones within seed and blanket assemblies is designed to reduce power peaking in the vicinity of metal/water channels.

Average as-built fuel lattice characteristics for the four primary fuel regions (seed, standard blanket, power-flattening blanket, and reflector blanket)

are presented in Table II-1. Two main differences between these fuel regions are the fuel rod diameters and rod spacings which yield four different metal/water volume ratios.

Table II-1 lists 6, 6.5, or 7.5 for the number of grids in the fuel height. There are nine grids in the seed assembly and eight in the blanket assembly. The reason the table shows fewer grids in the fuel height is that one grid is entirely above the fuel and half of a grid is below the fuel in each assembly. In calculating the grid fraction per level, the support post connectors and exterior boundary straps were subtracted from the grid. These connectors and straps were subtracted because they are exterior to the fuel lattice region. The grid volume per fuel rod data given in Table II-1 accounts for only the number of grids present over the fuel height and only the fraction per level actually present in the fuel lattice, as defined above. Remaining grid volumes are contained in metal-water regions exterior to the fuel lattice regions.

The maximum height of the binary fuel ($\text{U-233 O}_2 + \text{ThO}_2$) region of both seed and blanket is 84 inches with about 10 inches of ThO_2 added to each end for the purpose of reducing axial neutron leakage from the core. This makes the overall fuel height in each fuel rod approximately 104 inches. A schematic of the axial fuel distribution for a Type I module is shown in Figure II-4. The fuel in the movable seed region is composed of two zones of binary fuel. The arrangement is such that the outer four rows of seed rods consist of varying lengths of binary fuel, more lightly loaded than the inner 11 rows. The inner 11 rows of seed rods contain higher loaded binary fuel over the full 84-inch length. In each seed rod, there is only one type of binary fuel. Similarly in the blanket and power-flattening blanket regions, the first four rows of rods next to the seed-blanket metal/water channel have partial height binary fuel with the remainder being ThO_2 . There are three binary loading densities in each blanket region, but only one binary composition in any single blanket rod. This combination of radial and axial fuel zones serves the dual purpose of achieving an acceptable peak-to-average power ratio and providing adequate movable fuel reactivity worths. Average as-built core fuel characteristics for all 12 LWBR fuel compositions are presented in Table II-2.

The total initial fissile loading (U-233 plus U-235) of the LWBR core is 501 kilograms. Fissile loading in each module type is given in Table II-3. Because of the small size of the LWBR core and the desire to flatten the power distribution in Type I modules, loadings in Type II and III modules are larger than for Type I due to the power-flattening blanket. The average loading per module is about 30 percent larger than the loading for a large LWBR core which could consist of Type I modules only.

Total core fissile loading throughout core lifetime will remain within 2 percent of the initial loading. As a function of time in core life, U-233 burnup generally decreases the U-233 content in most binary regions, but neutron absorption in Th-232 results in a buildup of U-233 in thorium regions.

Plutonium buildup in LWBR is negligible since the U-238 content in the initial fuel averages less than 0.4 weight percent of the uranium and less than 0.01 weight percent relative to the thorium. No credit is taken for plutonium in breeding performance predictions.

Principal differences between LWBR and other light water cores are as follows:

1. The LWBR uses uranium-233 and thorium fuel which results in:
 - a. Breeding, that is, producing more fissile fuel than the fuel burned to generate power
 - b. Negligible plutonium and transplutonium actinide production
 - c. More pronounced negative fuel Doppler Coefficient of Reactivity (which gives enhanced core reactivity feedback to curtail an unexpected power increase)
 - d. Smaller delayed neutron fraction (which causes a faster rate of core power increase for LWBR for a given amount of reactivity inserted)
 - e. Stability against xenon oscillations with a very small maximum xenon transient
 - f. Slow reactivity change with time due to buildup, or decay after core shutdown, of protactinium-233 which has a half-life of 27 days.
2. The LWBR reactivity control is achieved by axial positioning of movable fuel assemblies to enhance breeding. This method of control varies the

geometry and the neutron absorption rate in U-233 and thoria throughout operational life so that parasitic loss of neutrons to conventional neutron control poisons is eliminated.

3. The LWBR has close spaced fuel rods to minimize neutron loss by absorption in water.
4. The LWBR has low average power density due to the large unenriched thoria regions.

REFERENCE

- II-1. D. R. Connors, S. Milani, J. A. Fest, and R. Atherton, "Design of the Shippingport Light Water Breeder Reactor," WAPD-TM-1208, January 1979.

TABLE II-1. AVERAGE AS-BUILT LWBR FUEL LATTICE CHARACTERISTICS

	<u>Seed</u>	<u>Standard Blanket</u>	<u>Power Flattening Blanket</u>	<u>Reflector Blanket</u>
Rod center - center spacing (in.)	0.3686	0.6304	0.6304	0.9005
Rod outer diameter (in.)	0.3063	0.5717	0.5274	0.8323
Rod surface-surface spacing (in.)	0.0623	0.0587	0.1030	0.0682
Clad thickness (in.)	0.02217	0.02808	0.02642	0.0419
Clad thickness/diameter ratio	0.072	0.049	0.050	0.050
Number of grid levels	9	8	8	6
Number of grids in fuel height	7.5	6.5	6.5	6
Grid fraction/level, in fuel lattice	0.846	0.79	0.79	0.80
Grid volume/fuel rod (in. ³)*	0.130	0.211	0.211	0.422
Metal/water volume ratio†	1.740	2.981	1.764	3.486
Total number of fuel rods	7428	3234	3581	3047
Number of flux-well rods	None	3	4	1
Total fissile loading (kg)	198.6	116.3	186.1	None
Total Th-232 loading (kg)	5206.5	9487.1	8788.3	18574.2

*Volume in fuel rod lattice based on number of grids in fuel height and the grid fraction per level in the fuel lattice.

†Under nominal hot conditions and with grid volume per fuel rod homogenized throughout the fuel regions.

TABLE II-2. AVERAGE AS-BUILT LWBR CORE FUEL CHARACTERISTICS

	Pellet OD (in.)	Pellet Length (in.)	Percent of Theoretical Density	U-Fissile (w/o)*	U-Fissile (grams/in.)	Fissile Loading (kg)	Loading Th-232 (kg)
<u>Seed</u>							
Thoria	0.2556	0.530	98.01	None	None	None	1846.6
Low zoned	0.2520	0.444	97.71	4.337	0.3416	61.28	1179.5
High zoned	0.2520	0.615	97.55	5.202	0.4114	137.31	2180.4
<u>Standard Blanket</u>							
Thoria	0.5106	0.616	97.80	None	None	None	3670.0
Low zoned	0.5105	0.531	98.61	1.214	0.3920	15.99	1141.6
Medium zoned	0.5105	0.868	98.22	1.668	0.5421	42.68	2205.9
High zoned	0.5105	0.785	98.11	2.005	0.6498	57.67	2469.6
<u>Power Flattening Blanket</u>							
Thoria	0.4696	0.447	98.06	None	None	None	2632.4
Low zoned	0.4695	0.870	98.03	1.654	0.4537	10.24	533.5
Medium zoned	0.4695	0.786	98.04	2.009	0.5509	13.58	580.1
High zoned	0.4696	0.701	97.91	2.739	0.7492	162.29	5042.3
<u>Radial Reflector Blanket</u>	0.7417	0.741	97.28	None	None	None	18574.2
TOTAL						501.04	42056.1

$$*U\text{-Fissile (w/o)} = \frac{U\text{-233} + U\text{-235}}{UO_2 + ThO_2} \times 100$$

U Isotopic Composition

U-232	<0.001 w/o
U-233	98.23
U-234	1.29
U-235	0.09
U-236	0.02
U-238	0.37

TABLE II-3. AVERAGE AS-BUILT LWBR LOADING BY MODULE TYPE

<u>Module Regions</u>	<u>Fissile Loading (kg)</u>		
	<u>Type I Module</u>	<u>Type II Module</u>	<u>Type III Module</u>
Seed†	16.53	16.55	16.56
Standard blanket	16.18	9.34	6.63
Power flattening blanket	None	15.66*	23.22*
Total blanket	16.18	25.00*	29.85*
Module total	32.71	41.55*	46.41*

*Two Type II and two Type III modules have 0.06 kg less loading due to flux well locations.

†A 12-seed average of 16.55 kg was used for all seeds in the calculations.

Figure II-1. LWBR Cross Section Module Identification

Figure II-2. LWBR Movable Fuel Control

Figure II-3. Type I Module Cross Section

Figure II-4. R Z Schematic of Type I Module

*** EXCEPT FOR 8 MEDIUM ZONE RODS PER MODULE IN ROW SIX (4 ON EACH SIDE, ALONG INTERFACE BETWEEN MODULES).**

Figure II-5. R-Z Schematic of Power Flattening Sides of Type II and Type III Modules

III. BASIS FOR LWBR NUCLEAR PERFORMANCE ESTIMATES

The calculational model used in the analysis of the nuclear performance of the LWBR as-built core was qualified with the aid of many comparisons of calculations to experiments and with comparisons to calculational standards. This extensive program has provided confidence in the fundamental data available for U-233 and Th-232 and has verified the ability to calculate the nuclear characteristics of the LWBR core.

A detailed description of the calculational model and its qualification is presented in Reference III-1. This section, therefore, will present only a summary of the analysis methods and model qualification. In addition, estimated uncertainties in reactivity and breeding predictions and a discussion of design assurance practices are presented.

A. Analysis Methods and Core Representation

1. General Features of the Analysis Model

Performance characteristics for the LWBR core were calculated using four group diffusion theory in two- and three-dimensional PDQ (Reference III-2) calculations, with few-group constants obtained from the energy spectrum program PAX (References III-3 and III-4). These included R-Z calculations for a central (Type I) module and three-dimensional calculations for either one-sixth or one-half of the core, depending upon symmetry of the case being analyzed. The R-Z calculations were used to obtain reactivity lifetime and breeding sensitivities to changes in core dimensions or changes in the calculational model. Most performance calculations were done with a three-dimensional one-sixth core geometry to represent the full core since LWBR has sixth core symmetry (see Figure II-1) under normal conditions. A half-core geometric model was used for abnormal half-core symmetric configurations. Calculational biases, obtained from critical experimental results and from three-dimensional sixth core Monte Carlo neutron transport theory calculations (RCP program, Reference III-5), were used to correct LWBR core calculational results.

Features of the calculational model, described in detail in Reference III-1, include:

- a. An automated system of processing manufacturing data for direct input to the PAX and PDQ analysis programs

- b. An extensive evaluation of nuclear data for a Th-²³³U system
- c. An accurate deterministic resonance integral calculation which accounts for resonance self- and mutual-shielding without the use of correction factors
- d. The application of space-energy corrections to infinite media cross sections to account for the finite size of each fuel zone
- e. The use of the three-dimensional diffusion-depletion calculation to treat fuel and fission product isotopes and handle movable fuel
- f. A simplified P-3 transport calculation for high energy neutron group
- g. The use of coolant and fuel temperature feedback to the neutron distribution, which also takes account of clad diametral shrinkage and fuel pellet growth
- h. The use of xenon feedback to compute the spatial flux distribution associated with equilibrium nuclide concentrations of iodine and xenon.

The PDQ nuclear design model has four energy groups, one of which is thermal. The highest energy group uses a simplified-P₃ approximation (Reference III-6) and fast advantage factors. The lower three groups use diffusion theory. Group constants are obtained with the PAX program for use in PDQ. Group constants in the two highest energy groups are infinite medium constants with one-dimensional radial space-energy correction factors based on a P₁ multigroup approximation. The third group covers the resonance range, for which an accurate deterministic resonance integral calculation is performed (Reference III-1, Section IV.D). The third group also includes the heterogeneous and Doppler broadening effects in the resonance treatment.

Homogenized thermal group constants, which utilizes thermal disadvantage factors, were derived from a refinement of the extensively used mixed number density method. The spectrum, from which the diffusion constant is derived, has been modified to improve the agreement in water hole peaks between the calculational model and experiments.

2. Multi-Region Core Representation

Various PDQ calculations were done with different three-dimensional geometric models. Mesh spacings depended on the degree of detail required and the computer space available. Most calculations for the LWBR core were done with a one-sixth

core geometry. A half-core model, with larger transverse and axial mesh spacings, was used for a few half-core symmetric configurations (for example, misalignment of one movable fuel assembly from the aligned-bank position and simulation of nonsymmetrical inlet temperature conditions resulting from a steam line rupture in one of the four coolant loops).

For the one-sixth core geometry, the sketch in Figure III-1 shows both the transverse mesh, used in the PDQ spatial flux solution, and the coarser "depletion block" grid in one-sixth of a Type I module (see Figure II-3). The transverse mesh lines, shown on the periphery of the module in Figure III-1, extend through the module in both directions. Those mesh lines which bound depletion blocks are drawn through the module in Figure III-1. Depletion blocks are subregions in which nuclide concentrations are spatially constant over the block.

The transverse mesh line spacing and depletion block sizes assigned to the Type II and Type III modules in the PDQ geometry are very similar to those shown in Figure III-1 for the Type I module. Comparable spacings and block sizes are also used for the larger diameter rods in the radial reflector region (module Types IV and V). In addition, radially and axially homogenized metal-water regions beyond the thorium reflector blankets are included in the one-sixth core PDQ geometry. The one-sixth core transverse geometry contains 7055 mesh points and 1472 transverse blocks, where the material may vary from one block to another. Of these, 1390 are depletion blocks in fuel regions representing 2883 fuel rod locations; the other 82 blocks are in metal-water regions and are used to define regions of uniform material properties.

Forty-five planes, with a uniform mesh of 3.5 inches, are used to describe the core in the axial direction. Thirty-one of these planes contain fuel, with the upper and lower thorium reflectors represented by 10.5 inches (3 planes of 3.5 inches) rather than the actual 10 inches (see Figure II-4). The other 14 planes contain axial metal-water regions. Note that a uniform axial mesh must be used in PDQ with the LWBR movable fuel requirement. Each axial plane serves as a depletion block boundary, such that depletion blocks, as shown in Figure III-1, each have a height of 3.5 inches. There are, therefore, 41,700 three-dimensional fuel depletion blocks in the one-sixth core geometry.

Fuel rod support grids have a height of 1.4 inches in seed and blanket assemblies and 1.8 inches in the reflector blanket modules. The axial grid levels cannot be represented explicitly in PDQ with a 3.5-inch axial mesh spacing. Therefore, the grid volume per fuel rod, presented in Section II-B and Table II-1, was homogenized uniformly over the axial height of each fuel region in PDQ. The flux peaking occurring between rod support grids was accounted for using a multiplicative extra peaking factor.

Fuel movement for LWBR is accomplished in the PDQ program by specifying a list of transverse depletion blocks and specifying the upper and lower planes which constitute the region to be moved. Several axial metal-water regions above and below the fuel are moved along with the fuel such that these regions remain in the correct position relative to the seed and blanket fuel.

A PDQ eigenvalue calculation at each fixed time gives the core reactivity. The depletion begins after an initial (time zero) spatial flux and power calculation is performed, which includes two recalculations of fuel and coolant temperature distributions. Subsequent calculations, referred to as "timesteps," consist of:

1. Depletion from time t_1 to t_2 using the spatial flux distribution from the previous timestep at t_1 and holding it constant in time
2. Upward motion of the movable fuel to a new estimated critical fuel position
3. Spatial flux calculations including, for a constant power depletion, one recalculation of temperature and xenon distributions.

Since the axial mesh spacing is 3.5 inches, each spatial flux calculation is done with the movable fuel position no more than 1.75 inches from the nominal critical position. In some cases, the timestep lengths Δt during the PDQ depletion calculations are adjusted to further improve the proximity of the fuel position to the estimated critical position.

Depletion calculations described above, to obtain core performance predictions prior to core operation, are done assuming continuous operation at 100 percent of full power, with equilibrium levels of xenon and protactinium. Additional calculations are done at various times in life, with xenon and/or protactinium removed and the movable fuel position adjusted to the critical seed position, to calculate off-equilibrium power distributions.

Timestep lengths (Δt) of 100, 900, and 2000 effective full power hours (EFPH) are used for the initial 3000 EFPH during which time xenon, protactinium, and samarium are building up to near equilibrium levels.

During the middle portion of lifetime, a Δt of 2500 EFPH is used since the movable seed bank is close to being axially aligned with the blanket (see Figure II-2), and the fuel motion is relatively slow.

A reduction in Δt to 1500 or 1000 EFPH is made for the period of more rapid fuel motion late in core lifetime so that the change in movable seed critical position between PDQ timesteps does not exceed 7 inches.

B. Analysis Model Qualification

A series of critical experiments including partial mockup experiments was performed to determine the degree of accuracy of the calculational model. Calculations of these experiments were done throughout the evaluation of the core design and the nuclear model. The partial core mockup consisted of a single Type II module surrounded by a uranium-fueled driver region and a thorium reflector. The seed assembly was movable in the mockup, as in the core, permitting the study of mockup properties as a function of fuel position including movable fuel reactivity worth measurements.

The connection between the critical experiments and the full core was made by developing a highly accurate analytical standard which could calculate both the experiments and the full core. The analytical standard is a Monte Carlo solution to the neutron transport equation with a particular set of neutron cross section libraries. It is embodied in a computer program called RCP (Reference III-5). The RCP program produces an "essentially exact" solution of the steady state transport equation in a core. This means that the core is represented in great geometric detail, that energy and angular dependencies of the neutron cross sections are shown in substantial detail, and that minimal calculational approximations are used.

Calculations of 10 experiments with the analytical standard provided an RCP-Experiment reactivity bias of 0.0011 (Table VIII-1 in Reference III-1). The analytical standard, with a consistent set of nuclear data libraries, was also used to calculate the full core at beginning-of-life. Although thermal-hydraulic temperature feedback is not part of the RCP program, fuel temperatures characteristic of high power operation were used in the RCP calculations of the full core.

Thus, the Doppler broadening of thorium (and other) resonances at high power was described. In this way, the analytical standard together with the critical experiments provide a standard of truth for reactivity, which extends the conditions covered beyond those which a mockup alone could provide. This extension is important for LWBR since about 2 percent in core multiplication factor is tied up in the "power defect", that is, in the fuel temperature rise from essentially zero power, with the moderator at operating temperature, to full power. This power defect cannot be measured in "zero power" critical experiments.

A special PDQ calculation, done consistently with the RCP full core calculation at beginning-of-life, provided a bias between the nuclear analysis model and the analytical standard. The PDQ-RCP reactivity bias of 0.0014, together with the RCP-Experiment bias of 0.0011, yielded a net (PDQ-Experiment) reactivity bias of 0.0025. The full core PDQ and RCP calculations also provide a bias in breeding predictions, where the PDQ model is shown to predict a peak fissile inventory ratio (see Section IV.B) about 0.001 higher than RCP.

Since the analytical standard makes use of a Monte Carlo program, it is not possible to examine the power at points or in fuel pellet-sized volumes throughout a core. Three-dimensional power distributions were measured for the partial core mockup in sufficient detail to construct radial and axial power maps for various seed positions. Direct PDQ calculations of these experimental power distributions yielded experiment-to-calculation (E/C) correction factors or allowances for the different configurations of the partial mockup. Since the partial core mockup was much smaller than the full core, it was not appropriate to take a local E/C factor from a particular location and seed position, and apply it directly to a "corresponding" location in the full core. The environment of any point in the partial mockup is not the same as the environment in the full core. Rather, the totality of E/C data from all seed positions was examined together. Regional factors were chosen to correct the PDQ power distributions of the full core and account for the whole variety of experimental data for the region. These factors are discussed further in Section IV.C.

In addition to the overall qualification procedure of the nuclear model summarized above, individual parts of the model were tested against more exact calculations or against experiments during model development. Details of these tests and qualification procedures are presented in Reference III-1.

C. Evaluation of Uncertainties

Predictions of nominal nuclear performance for the LWBR core, that is, power distribution, reactivity, and breeding, are best estimates based on the analytical models, nuclear data, and physical and chemical description of the core. Associated with the analytical models, nuclear data, and core description used in the calculations are approximations and uncertainties which introduce uncertainties in the predictions of the core performance. The effects of these approximations and uncertainties, as well as core operational uncertainties, on the performance predictions have been evaluated.

The uncertainty in the predicted power distribution is accounted for by applying the necessary factors directly to the predicted power distribution. These factors and their derivation are discussed in Section IV.C.

The uncertainties in reactivity and breeding predictions are presented in Tables III-1 and III-2. Because the available methods for evaluating the effect of the uncertainties in nuclear data on the reactivity and breeding predictions are different, the effects are presented as a combined uncertainty (RCP vs experiment) for reactivity and as individual uncertainties (cross sections) for breeding. Uncertainties are quoted as one standard deviation and have been combined statistically to determine the net deviation. The effect of impurities on breeding is included as a conservative bias in the core breeding prediction and therefore is not included as an uncertainty in breeding in Table III-2.

As discussed in Section III.B, the reactivity differences between RCP and PDQ for full-power conditions are included as a bias in the LWBR core reactivity prediction. The RCP statistics are included as an uncertainty in Table III-1 along with an RCP versus experiment term that characterizes other uncertainties (such as manufacturing uncertainties of the experiments). The 95-percent confidence interval in core reactivity uncertainty of 0.0086 is less than the ± 1 -percent off-nominal allowance assumed for the core analysis (Section IV.A).

D. Design Assurance

Vital to any discussion of core performance is the question of design assurance practices followed in arriving at the current LWBR core performance estimates. A summary of the analysis methods and model qualification has been presented in

Sections III.A and III.B. Design assurance however requires the correct implementation of the calculational model and accurate representation of the core components (fuel loading, metal/water ratios, etc.) in the calculations. Several of the checks on design analyses and accurate knowledge of the core fuel, metal, and water content are discussed below.

1. Design Analyses Assurance

One of the most important LWR design assurance characteristics is the high degree of automation of the nuclear design process. Computer programs, having been written and verified, greatly reduce the amount of data handling done by individuals. These programs are presented in Reference III-1 (Section III.B and Figure III-1). A most useful and important one of these codes, and one which will be used as an illustrative example of the type of automation provided, is the verified PREP1 program. The PREP1 program accepts input data from engineering drawings or manufacturing data and produces input for both of the major nuclear design codes PAX and PDQ. The data generated by PREP1 (consisting of nuclide concentrations, fuel pellet conductivities, flow areas, heat transfer areas, fast advantage factors, and mesh) may be transmitted to the nuclear design codes internal to the computer, thus avoiding transcription errors.

The LWR nuclear design assurance is also provided by independent check calculations and by calculations of progressive designs, which were checked to ensure that they gave expected changes. Some performance characteristics are calculated by both two- and three-dimensional calculations. The results are compared using known sensitivities to calculational differences between the two types of calculations.

Assurance is provided for the calculation-to-experiment comparison discussed in Reference III-1 through the performance of independent PDQ calculations of the same experimental core. Additional confidence in these results is provided by the general agreement, with certain predictable differences, of PDQ and RCP calculations of many of the experimental cores. These analytical models (PDQ and RCP) were constructed independently, and the programs are totally different in their approaches to the problem solution.

2. Core Loading Assurance

Average data for fuel pellets, fuel rods, support grids and modules (given in Table II-1, II-2, and II-3) were obtained from the as-built data collected during the course of core fabrication. These data are summarized in Appendix A. Numerous core loading assurance checks were performed during core manufacturing to ensure correct loading for the 14,243 fuel rods in the 12 seed and blanket modules. These modules contain 21 different fuel rod types (14 types for radial and axial zones, shown in Figures II-4 and II-5, and seven types for top-mounted versus bottom-mounted assembly).

Full content (total uranium and uranium isotopic mix) for each of the eight binary pellet types was obtained from a random sample of pellets from each of more than 800 blends. The total uranium content was obtained by chemical analyses and by nondestructive assay using the Delayed Neutron Gage (Reference III-7). The uranium isotopic mix was obtained by mass spectroscopic analyses.

For each rod type, the fuel pellet lengths were significantly different for each fuel zone and the pellets had a hemispherical dish void on each end. Also, each pellet type had unique markings stamped on the ends of the pellets which were used for visual identification during loading of each rod. Following completion of fuel rod assembly, in-motion radiographic film data were recorded for each fuel rod and examined to verify that all pellets in a rod were of the same diameter and to measure the pellet lengths using the end dish voids and the Pellet Length Gage technique (Reference III-8). This procedure provided a check that each fuel rod contained the proper pellet types and that the fuel zone pellet stack lengths were correct.

Correct fuel rod loading in the 12 seed and 12 blanket assemblies was assured by several procedures. Each fuel rod had a unique rod serial number stamped on both ends, which identified the type of fuel rod. Different rod end closure stem lengths and identification hole locations were used for each fuel rod type. A computer list was prepared for each assembly specifying the assigned rod, by rod identification number, for each grid cell location in the assembly. After rod installation per the above assignment, visual inspection, photography, and metallic gages were used to confirm the correct pattern of end cap stem lengths.

Measured data on rod support grid weights, on cell pitch, and on fuel rod cladding diameter and thickness were recorded for all fuel regions. These data were used in the calculations to correctly represent the volumes of Zircaloy and AM-350 and to specify the fuel rod spacings such that the water volume in each fuel region was correctly represented.

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TABLE III-1. UNCERTAINTIES IN REACTIVITY PREDICTIONS

<u>Source</u>	<u>Δk (one standard deviation)</u>
1. Nuclear design model	
RCP vs experiment	± 0.0036
RCP statistics	± 0.0009
Doppler effect on cross sections	± 0.0011
Fission products [†] (except xenon)	$\pm 0.0014^*$
Xenon	$\pm 0.0006^*$
Other isotopes [‡]	$\pm 0.0007^*$
Depletion cross section variation	$\pm 0.0008^*$
Temperature feedback model	± 0.0006
2. As-built data	± 0.0002
Net Standard Deviation	± 0.0043

*These items are zero at beginning-of-life, so the net standard deviation is slightly smaller then.

[†]End-of-Life value. Fission product uncertainty varies linearly with time in life.

[‡]Isotopes that build up during life and are not fully taken into account in the RCP vs. experiment comparisons.

TABLE III-2. UNCERTAINTIES IN BREEDING PREDICTIONS

<u>Source</u>	<u>ΔFIR (one standard deviation)</u>
1. Cross Sections	
^{233}U	± 0.00196
Thorium capture	± 0.00165
Thorium fission	± 0.00100
Fission products (except xenon)	± 0.00064
Xenon	± 0.00049
^{233}Pa	± 0.00036
Zircaloy-4	± 0.00033
Stainless steel and Inconel	± 0.00015
Other elements	± 0.00056
2. PDQ vs RCP	± 0.00041
3. Depletion model	
Cross section variation	± 0.00018
Temperature feedback model	± 0.00007
4. As-built data	± 0.00022
	<hr/>
Net Standard Deviation	± 0.00300

NOTE: The traverse mesh lines shown on the periphery of the module extend through the module in both directions: only those mesh lines which bound the depletion blocks are drawn through the module in the figure.

Figure III-1. Mesh Lines and Depletion Block Sizes in PDQ for One-Sixth of Type I Module

IV. DESCRIPTION OF LWBR NUCLEAR PERFORMANCE

Nuclear design performance data presented in this section for the LWBR core include lifetime and reactivity data in Section A, breeding performance in Section B, power distribution data in Section C, kinetics parameters and stability in Section D, and shutdown and reactivity control capability in Section E.

A. Nominal and Off-Nominal Lifetime and Reactivity Performance

The lifetime of the LWBR core has been calculated using the design calculational model described in Section III. The LWBR core has a nominal (best estimate) reactivity lifetime of 21,800 hours at 236.6 thermal megawatts (60 MW(e)net) and 520F inlet temperature. The predicted movable fuel bank position versus time for the nominal core, assuming continuous full power operation, is shown in Figure IV-1. Movable fuel positions are defined relative to the full shutdown fuel position (Figure II-2) being 0 inches. At a fuel position of 0 inches, the top of the seed assembly is 60 inches below the top of the blanket assembly. End of reactivity life is defined as the time when the movable fuel critical position is at the 7-foot upstroke position, where the top of the seed is 24 inches above the top of the blanket. The mechanisms are designed to stop fuel motion at the 7-foot upstroke position. The full power movable fuel critical position is within 8 inches of the seed and blanket axial aligned position (60 inches) for about half (from 8500 to 18,500 full power hours) of the core lifetime. This is desirable to minimize the conversion ratio losses to axial water and structure which become significant for large displacements of movable fuel from the aligned position.

Movable fuel position for criticality is a function of both the fuel worth (Δk /inch) and the rate of change in reactivity. Buildup of xenon, samarium, and protactinium early-in-life causes a relatively large rate of decrease in reactivity and thus the fuel motion is relatively rapid for the first 2000 to 3000 hours. Keeping the movable fuel nominally close to the 60-inch (seed and blanket axially aligned) position for a large fraction of the core lifetime requires a large fuel worth. The LWBR core has natural thorium "step" regions in the top half of seeds and in the bottom half of blankets (see Figure II-4) close to the seed-blanket interface. The fuel worth achieved by this fuel distribution is largest for lower positions of the movable fuel and smallest for higher positions of the movable fuel. As the core depletes, U-233 builds up in the thorium step regions,

lowering movable fuel worth. The fuel worth, as a function of time in core life at the full power critical position for the LWBR core (shown in Figure IV-2), decreases from about 0.0042 Δk /inch near beginning-of-life to around 0.0005 Δk /inch at 21,800 hours. The low fuel worth for higher positions of the movable fuel is the major contributor to the rapid rise in the movable fuel position toward end-of-life.

Predicted nominal core fuel position versus time in core life, shown in Figure IV-1, is based on as-built dimensions, for all fuel rod and module parameters, and the best estimate of calculated reactivity. Off-nominal calculations have been done assuming that the reactivity may differ from its predicted value, due to uncertainties discussed in Section III.C. The variation in calculated lifetime has been estimated to be about ± 2500 hours for a uniform ± 1 -percent reactivity uncertainty, also shown in Figure IV-1.

Predicted fuel position versus time for various temperature, xenon and protactinium conditions of the nominal core are shown in Figure IV-3. The early-in-life decrease in reactivity from cold zero power conditions to hot full power, with equilibrium xenon and protactinium, is shown by the fuel position bands at the left of Figure IV-3. Core reactivity increases with reduced power, requiring lowering the movable fuel. The data in Figure IV-3 show the predicted critical fuel positions for five different equilibrium conditions as functions of time in core life:

1. Hot full power, with xenon and protactinium at full power equilibrium levels
2. Hot, power level reduced to zero, xenon and protactinium concentrations at the full power equilibrium levels
3. Hot, power level at zero, xenon concentration decayed to zero, protactinium at the full power equilibrium level
4. Hot, power level at zero, xenon and protactinium concentration decayed to zero
5. Cold (68F) zero power with zero xenon and protactinium concentrations.

Critical fuel positions shown by Curve A in Figure IV-3 are maximum fuel positions which would be predicted for continuous operation at 100-percent power. A rare exception would occur due to maximum xenon, when returning to full power operation, after a shutdown of about 4 hours. In this instance, the critical

fuel position would increase slightly to compensate for about 0.2-percent reduction in core reactivity. Operation at reduced power levels will generally result in fuel positions between Curves A and B, shown in Figure IV-3, where Curve B shows the critical position at zero power with full power equilibrium concentrations of protactinium and xenon.

Following core shutdown periods of about two days, the reactivity increase due to xenon decay from the full power equilibrium level is predicted to be about 1.8 percent and the hot zero power critical position is expected to be close to Curve C in Figure IV-3. Lower hot critical fuel positions, between Curves C and D, are expected following extended periods of core shutdown due to decay of protactinium (Pa-233), which has a half-life of 27 days, to U-233. The total calculated reactivity increase due to Pa decay from the full power equilibrium value is in the range of 2.0 to 2.5 percent due to the reduced neutron absorption by Pa-233 and to the increase in U-233 content in the core.

Fuel positions shown by Curve E in Figure IV-3 are calculated minimum critical positions achievable for the nominal core under cold (68F) plant conditions and assuming sufficient shutdown time to decay all xenon and protactinium.

B. Breeding Performance

This section will discuss the fuel chain transformation of fertile material into fissile fuel, the definition of the conversion ratio (CR) and fissile inventory ratio (FIR), and the variation of CR and FIR versus time for the LWBR core. Calculated fissile fuel distributions in the core as functions of time and at end-of-life will be compared with initial fuel distributions. In addition, possible effects of core operating history on the magnitude of the best estimate peak FIR and peak FIR lifetime will be presented.

1. Definition of Breeding Parameters

The LWBR calculational model predicts breeding performance by calculating conversion ratio and fissile inventory ratio. The following discussion presents the definitions of these quantities and their relationship to nuclear reactions. Derivation of equations, explanations of their practical use, and the relationships between conversion ratio and fissile inventory ratio are presented in Appendix B. Also included in Appendix B are breeding sensitivities to changes in core parameters. Finally, Appendix B discusses the methods used in specifying impurity limits for fuel, clad, and grid materials.

To facilitate an understanding of conversion ratio and fissile inventory ratio, a discussion is given below of the interactions which occur between neutrons and the nuclides of the U-233-thorium fuel system. These reactions include those of prime importance to breeding, that is, those of neutron capture in fertile materials. When neutrons are captured in the fertile materials (Pa-231, Th-232, U-232 and U-234), the fissile materials (U-232, U-233, and U-235) are produced. These same fissile materials are destroyed, however, as they fission to preserve criticality and produce power. Breeding occurs when the rate of conversion to fissile material exceeds the rate of fissile destruction.

Processes of fertile captures, fissile destruction, and other neutron reactions occurring in the LWBR fuel system are shown in Figure IV-4, but can best be represented by defining "chains" of events which the fuel nuclides undergo upon reaction with neutrons. Three such chains are used in the calculational model, shown in Figure IV-5, since the PDQ depletion model treats only linear chains. Extremely short-lived radioactive nuclides are omitted from these chains. The short-lived nuclides include Th-231, Th-233, Th-234, Pa-232, and Pa-234. Half-lives given in Figure IV-5 for Th-232 and the "counter" are used to avoid roundoff difficulties with extremely small numbers in the computer.

The third fuel chain, shown in Figure IV-5, is used primarily to measure the U-232 buildup for health physics studies. Its significance is minor in normal design calculations. For convenience, the Pa-231 concentration is treated as a direct fission neutron yield from Th-232 rather than as an $n,2n$ product, to avoid carrying $n,2n$ cross sections in PDQ. This fake yield is a ratio of thorium cross sections averaged over the first few-group: $\sigma_{n,2n}/\sigma_f$.

Note that the same nuclide appears in more than one chain. In nature only one fuel chain exists (shown in Figure IV-4), but the calculational model requires linearization into the three components given above. This procedure requires no approximations in describing the chains. The theoretical basis for this approach has been given in Reference IV-1.

Uranium fuel produced for LWBR fuel elements contains a small amount (less than one-half percent core average) of U-238. The U-238 is not treated explicitly in a depletion chain, but its effect on the atom balance (for breeding analysis) is accounted for by a post-PDQ correction (see Section IV.B.2).

The role of Pa-233 in these chains is important. It costs one neutron to convert Th-232 into Pa-233. If the Pa-233 decays to fissile U-233, the neutron has been well spent. If, however, the Pa-233 captures a neutron, Pa-234, which decays to fertile U-234, is created. Thus, neutron capture in Pa-233 represents a double detriment to breeding since it cost one neutron to convert fertile thorium to Pa-233 and a second neutron to convert Pa-233 to Pa-234 (which decays to fertile U-234).

Conversion ratio (CR), at time t, is defined by

$$CR(t) = \frac{\text{instantaneous production rate of fissile atoms at time } t}{\text{instantaneous destruction rate of fissile atoms at time } t} \quad (1)$$

Substituting the fertile and fissile nuclides, Equation (1) becomes

$$CR(t) = \frac{C^{Th-232}(t) + C^{U-232}(t) + C^{U-234}(t) + A^{Pa-231}(t) \left(\frac{F^{U-232}(t)}{A^{U-232}(t)} \right) - A^{Pa-233}(t)}{F^{U-232}(t) + A^{U-233}(t) + A^{U-235}(t)} \quad (2)$$

where

$C^i(t) = A^i(t) - F^i(t)$ denotes capture rate in nuclide i at time t,

$A^i(t)$ denotes absorption rate of nuclide i at time t,

$F^i(t)$ denotes fission rate of nuclide i at time t.

The nuclides Th-232, U-232, Pa-231, U-234 are considered to be "fertile material" since upon neutron capture they are converted to fissile material. The fraction of U-232 which fissions is treated as fissile fuel. The assumption that Pa-231 does not fission is implicit in Equation (2). Available nuclear data for Pa-231 support this assumption.

Fissile inventory ratio (FIR), at time t, is defined in Equation (3).

$$FIR(t) = \frac{\text{number of fissile atoms at time } t}{\text{initial number } (N_0) \text{ of fissile atoms}} \quad (3)$$

In terms of the fissile nuclides, this may be expanded to:

$$FIR(t) = \frac{\text{atoms of U-233, U-235, Pa-233 and the fissioning fraction of U-232 at time } t}{\text{atoms of U-233, U-235 and the fissioning fraction of U-232 initially}} \quad (4)$$

where all Pa-233 atoms will decay to U-233 upon core shutdown.

As shown in Equations (1) through (4), conversion ratio (CR) is an instantaneous measure of breeding while fissile inventory ratio (FIR) is an overall measure of breeding. Although the actual measure of breeding is defined above, neither Equation (3) nor (4) is used to actually calculate FIR. Instead, FIR is obtained by integrating the quantity (CR-1) over time, as shown by Equation (5):

$$\text{FIR}(t) = 1 + \frac{1}{N_0} \int_0^t [\text{CR}(t) - 1] A_{(t)}^{\text{fissile}} dt \quad (5)$$

where $A_{(t)}^{\text{fissile}}$ is the neutron absorption rate in fissile material at time t . The reason for integrating CR rather than using Equations (3) and (4) is that the nuclide concentrations available from PDQ calculational data lead to slightly inaccurate FIR's. The inaccuracy arises because of approximations made within the calculation. Nuclide concentrations are determined by depletion equations which use constant fluxes over a relatively long timestep length rather than time dependent fluxes. This depletion procedure is sufficiently accurate for reactivity and power calculations, but for FIR calculations the integration of conversion ratio has been shown to be more accurate than the use of nuclide concentrations, and to be equivalent to the use of very small timesteps. The integration process is performed, using equations presented in Appendix B, as part of the post-PDQ analysis.

The instantaneous production rate of fissile atoms, given by the numerator of Equation (2), is gradually diminished due to neutron losses to Pa-233, Xe-135 and other fission products. At the point in time when the conversion ratio is reduced to 1.0, the FIR, given by Equation (5), is maximized. Thus, this point in time is defined as the peak FIR lifetime.

2. Conversion Ratio and FIR Versus Time

The best estimate calculated conversion ratio and fissile inventory ratio of the LWBR core is shown in Figure IV-6. A peak FIR of 1.0134 is predicted to occur at 19,000 EFPH, the time at which the conversion ratio has decreased to a value of unity. The breeding performance shown in Figure IV-6 corresponds to the best estimate for a nominal reactivity core operated continuously at full (100-percent) power and does not include allowances for the uncertainties in

breeding predictions presented in Section III.C. Actual operation at a time averaged power level below 100 percent and periodic core shutdowns will increase the breeding performance, as discussed below in Section IV.B.4.

To show the importance of minimizing structure in the breeder core, the conversion ratio (CR = neutron production - losses) calculated for the LWBR core at 3000 hours and near end-of-life (20,500 EFPH), is shown in Table IV-1. The core absorption fractions for all elements in the core are ordered by magnitude in Table IV-2, at about 8000 EFPH.

Entries in Item 1 of Table IV-1 are those included in the nuclear depletion model. A comparison of PDQ diffusion theory and RCP Monte Carlo calculations for the as-built core at beginning-of-life indicates that the PDQ calculation overpredicts breeding by 0.0027 in conversion ratio, shown in item 2 of Table IV-1. Additional conversion ratio losses given in item 3 of Table IV-1 are effects which are not included explicitly in the calculational model because of complexity or computer size and time limitations.

The presence of 0.373-percent U-238 in uranium is not treated explicitly in a depletion chain. Fuel impurities and AM-350 grid material impurities are not included in the material descriptions in PDQ. Instrumentation consisting of eight flux well locations, in place of eight fuel rods, have been calculated to reduce CR by 0.0002. The buildup of crud deposits in the core have been estimated to reduce CR over lifetime by an average of 0.0010.

Coolant flow for the exterior row of seed rods and the interior and exterior rows of blanket rods is greater than the coolant flow interior to the fuel rod lattice. This radial variation in coolant flow distribution is not included in the PDQ calculation. This effect is +0.00005 in CR.

Due to computer time and axial mesh limitations, depletion time increments on the order of 2500 hours are used. Smaller time increments would more accurately describe the behavior of conversion ratio with time. An additional correction of -0.00025 in CR is needed to yield the results of a very small time-increment calculation.

The depletion chains used assume direct decay of Th-233 (half-life = 22 minutes) to Pa-233, but a small loss in breeding (0.00005 in CR) results from neutron absorption in Th-233.

Note that all of these corrections to the calculated CR losses, listed in Item 3 of Table IV-1, are relatively small and in total account for only about 1 percent of the total CR losses. Nevertheless, breeding performance predictions are corrected for all of these losses.

The small fraction of neutrons absorbed in water and structure (5.25 percent shown in Table IV-2 at 8000 EFPH) is achieved in LWBR by several important design features. The initial reactivity and thorium wedge design keeps the movable seed fuel nominally close to the seed-blanket aligned position over a large fraction of core lifetime. This, plus the 10-inch ThO_2 end blanket above and below the 84-inch binary height, helps breeding by minimizing neutron losses to water and structure above and below the core fuel, as shown in Figure IV-7. Displacement of the seed fuel above the seed-and-blanket aligned position (60 inches) produces less reflector absorption than downward displacement of equal magnitude because of the ThO_2 wedge regions in the top half of the seed and bottom half of the blanket.

Radial neutron leakage from the core is minimized by the thorium reflector modules (see Figure II-1). Data presented in Figure IV-8 show that fuel in the radial reflector region absorbs about 85 percent of the neutrons escaping the 12 binary fueled core modules and thus produces about 27 kg of U-233 by end of core life (Table IV-3).

3. Fissile Fuel Distribution as a Function of Time

Total core fissile loading throughout core lifetime remains within 2 percent of the initial loading. As a function of time, U-233 burnup generally decreases the U-233 content in most binary regions, but neutron absorption in Th-232 results in a buildup of U-233 in thorium regions. A comparison of the beginning- and end-of-life fissile loadings in all 12 fuel compositions, is given in Table IV-3. There is also a shift with time in U-233 spatial density between seed regions and blanket regions. Figure IV-9 shows the total seed, blanket, power-flattening blanket, and reflector region calculated fissile loadings, summed over the entire core, as functions of time. The approximately 15 percent reduction in total seed region fissile loading is due to burnup resulting from the large excess reactivity required in the seed region to achieve adequate lifetime from the batch-loaded LWBR core. However, the net increase in blanket region fissile loading more than compensates for the reduction in seed region loading.

A comparison is presented in Table IV-4 showing calculated fissile loadings in seed, standard blanket and power-flattening blanket regions in each module type both at beginning- and near end-of-life.

Local variations in U-233 spatial density occur slowly due to the simultaneous burnup of U-233 and the production of U-233 by neutron capture in Th-232. The largest net changes in U-233 loading per rod are in the high-zone seed rods, about 20 percent on the average, and in the high-zone power flattening blanket rods, about 10 percent in an average rod. High-zone standard blanket rods have a smaller change, about 4 percent in an average rod. Calculated axial distributions of U-233 at end-of-life are shown in Figure IV-10 for the binary length of highly depleted seed, standard blanket, and power-flattening blanket high-zone rods.

The net change in U-233 loading within the low-zone seed (23 percent) is about the same as in the high-zone seed, but production of U-233 in the thoria portion of these rods (see Table IV-3) results in average loading per rod about 6 percent below the initial loading. A comparison of the calculated axial distribution of U-233 at beginning-of-life and end-of-life in a typically 42-inch binary seed rod (rod Rows 14 and 15 shown in Figure II-4) is shown in Figure IV-11. Calculated end-of-life U-233 loadings for the blanket and power-flattening blanket rods closest to the seed assemblies, low zone and medium zone in Rows 1 to 3 (see Figures II-4 and II-5), exceed the initial loading. A comparison of the axial distribution of U-233 at beginning- and end-of-life for an average 42-inch binary standard blanket rod in the Type I module, given in Figure IV-12, shows a buildup of U-233 in both the binary and thoria segments of the rod.

Buildup of U-233 in the reflector blanket regions, module Types IV and V (see Figure II-1) is larger in the inner rod rows than in the outer rod rows. At 20,500 EFPH, the total U-233 content in 3047 rods is calculated to be about 27.6 kg, or an average of about 9 grams per rod. The calculated U-233 content for 516 rods in the first two rows of rods is about 10 kg, or about 20 grams per rod. Thus, the first two rows of rods, which contain 17 percent of the total reflector rods, contain 36 percent of the U-233 content in the reflector. The largest U-233 content is calculated to be about 32 grams in the reflector rod located at the apex of the Type IV module. The calculated U-233 axial distribution in this rod at end-of-life is shown in Figure IV-13.

4. Possible Effects of Operating History on Breeding

The peak FIR of 1.0134 calculated to occur at 19,000 EFPH corresponds to the best estimate for a nominal core operated continuously at full power. Both xenon-135 and protactinium-233 concentrations are dependent on the core power level. Periods of core operation at less than full (100-percent) power will result in reduced Xe-135 and Pa-233 concentrations. The reduced neutron absorption by Xe-135 and Pa-233 at power levels below full power will increase the breeding CR as a function of time; hence, the peak FIR and the lifetime at which the peak occurs will both be increased by periods of operation at less than full power and by shutdown periods.

Data given in Table IV-1 show that at 3000 EFPH, at which time the Pa-233 has reached its equilibrium concentration, the calculated neutron losses to Pa-233 plus Xe-135 account for about 26 percent of the total neutron losses. This percentage decreases, as a function of time due to the increased losses to the buildup of other fission products; but even near end-of-life (20,500 EFPH data in Table IV-1) the Pa-233 plus Xe-135 losses are calculated to account for about 18 percent of the total neutron losses.

Specified operating procedures for the LWBR plant require periods of swing load, where power is reduced significantly for some portion of each day, to simulate operating conditions which could be similar to commercial plant operations. Also, shutdown periods of about 3 to 4 weeks are scheduled twice a year for routine maintenance and to perform various tests. The tests are designed to monitor core characteristics and to provide data for comparison with calculated performance predictions.

Xenon-135 is introduced into the core by fission of U-233 (yield = 1.2 percent) and indirectly by β decay (half-life = 6.7 hours) of iodine-135 (produced by 4.8-percent yield from U-233 fission). Xe-135, which depletes by neutron capture and by β decay, with a half-life of 9.2 hours, will typically reach an equilibrium concentration in about 40 hours of core operation. Relative to the Xe-135 concentration for continuous full power operation, a reduced average power level will decrease the equilibrium concentration which Xe-135 attains. Core shutdowns will allow the Xe-135 to decay and to build up to equilibrium at each restart.

Protactinium-233 is produced in the core by neutron capture in Th-232 and is removed by β decay and neutron absorption, as shown in Figure IV-4. The Pa-233 decay constant for a 27 day half-life is $0.297 \times 10^{-6} \text{ sec}^{-1}$. A typical value of the Pa-233 specific absorption rate in the LWBR core is $0.50 \times 10^{-8} \text{ sec}^{-1}$. Therefore, the β decay of Pa-233 to U-233 is calculated to account for about 98.3 percent of Pa-233 removed and neutron absorption accounts for about 1.7 percent. Core operation at an average power level below 100 percent will proportionately reduce both the Pa-233 equilibrium concentration and the Pa-233 neutron absorption. As discussed previously (Section IV.B.1), reduced Pa-233 neutron captures are important to breeding since a Pa-233 capture represents a double neutron loss, one to convert Th-232 to Pa-233 and a second to convert Pa-233 to Pa-234.

PDQ depletion calculations at a power level below 100 percent, including intervals at zero power, provide estimates of breeding increases which would be obtained from a more realistic power history. For example, operation at an average power level of 85 percent has been estimated to increase peak FIR by about 0.0018, and shutdowns of 4 weeks twice a year have been estimated to increase peak FIR by an additional 0.0012. This particular power history would thus increase the best estimate peak FIR by 0.0030 and, in addition, would increase the time in life at which the peak FIR occurs by about 1000 EFPH. Therefore, this power history would yield a best estimate peak FIR of 1.0164 at about 20,000 EFPH compared with the peak FIR of 1.0134 at 19,000 EFPH for continuous full power operation.

C. Power Distribution and Power Performance Data

Radial and axial power distributions have been predicted throughout core life using the explicit three-dimensional symmetric six-core block depletion PDQ calculation for a range of possible fuel positions. These fuel positions might occur due to nominal or off-nominal reactivity and due to off-equilibrium xenon and protactinium conditions. These calculations include directly the feedback effects of three-dimensional distributions of fuel and moderator temperature in the core and effects of fuel rod clad shrinkage.

The locations and magnitudes of fast flux and power peaks are dependent on movable fuel position, power level, and depletion. Power, fast fluence and fuel depletion distribution data were calculated for the +1 and -1 percent off-nominal

reactivity movable fuel positions, discussed in Section IV.A.1 and shown in Figure IV-1, as well as for nominal reactivity fuel positions. These data were used to perform detailed thermal-hydraulic, structural and fuel element performance analyses calculations at the most limiting times in life using the most limiting power peaks, the local fast fluence, and the depletion values from the nominal and off-nominal distributions. These performance analyses will be the subject of separate reports. Data provided for the performance analyses of the LWBR core contain allowances and factors to account for uncertainties in the nuclear model in the form of experiment to calculation biases, manufacturing tolerances, and operational variations. These factors and allowances are discussed in detail in Section IV.C.3.

1. Gross Radial and Axial Power Distributions

Table IV-5 gives calculated power distributions in gross regions of the core at several times in life for a nominal core operated continuously at full power. The percentage of total core power generated in each of the four fuel types (seed, standard blanket, power-flattening blanket, and reflector blanket) is given in Table IV-5 in addition to the fractional contribution from each module type (location of each module type is shown in Figure II-1).

Upward positioning of the movable seed fuel assemblies as a function of life-time is required to maintain criticality in a manner similar to control rod motion in other reactor plants. Greater changes in fuel position are required early in life, shown in Figure IV-1, to compensate for the initial buildup of xenon, protactinium, and samarium. This results in a small increase in the percent of power generated in the seed. Upward positioning of the movable seed assemblies above the aligned position later in core life, as well as depletion effects, results in a small decrease in seed power fraction.

The LWBR core, with a pressure vessel inner diameter of 109 inches, is much smaller than the core in a large central station commercial reactor plant. A small diameter core generally has large radial neutron leakage and large power peaking at the center of the core. The presence of the reflector blanket modules, to reduce radial neutron leakage from the core, leaves an effective diameter for the 12-module power producing region of the core of approximately 68 inches. The LWBR core was therefore designed with a peripheral "power-flattening blanket" region, shown in Figure II-1, to reduce power peaking in the central modules.

These design features for the small LWBR core better simulate the breeding environment of a large core. This power-flattening blanket region contains 10 rows of rods, compared with 6 rows in the inner "standard blanket". It has a larger U-233 content in the fuel and a smaller metal/water ratio than the standard blanket region.

Calculated power distribution data presented in Table IV-5 show that both the three central Type I modules and the three Type II modules each produce about 8.5 percent of core power while each of the six Type III modules contribute about 8 percent of core power. The power density however is somewhat larger toward the core center. For example, the 10,500 hour calculated data in Table IV-5 show that the ratio of power in a Type I seed to power in a Type II seed is 1.14 and the ratio of power in a Type II seed to power per Type III seed is also 1.14.

For conditions of zero xenon at anytime in core life, the movable seed assemblies will have a lower critical position than for equilibrium xenon conditions. The decay of protactinium-233 to U-233, with a half-life of 27 days, will result in an increase in core reactivity during an extended period of core shutdown. The full power equilibrium reactivity worth for all Pa-233 is only about 20 percent greater than the equilibrium Xe-135 reactivity worth. Thus, following a long shutdown period, a gradual approach to full power will restore the xenon and will yield a movable fuel position without protactinium roughly comparable to the fuel position for equilibrium protactinium but zero xenon.

Neither of the above conditions, zero xenon or zero protactinium, defined as "single off-equilibrium" requires any modification in core power operation. Under "double off-equilibrium" conditions of zero xenon and zero protactinium, because of higher power peaking in the core, a gradual approach to full power is specified to achieve nearly a single off-equilibrium movable fuel position prior to full power operation. Note however that, following the initial core startup, a double off-equilibrium condition is not expected very often due to the long shutdown required to decay out the Pa-233 (half-life = 27 days).

Table IV-6 gives the magnitude of the calculated difference in gross region power distribution with off-equilibrium conditions relative to equilibrium conditions. The early-in-life data in Table IV-6 actually represent double off-equilibrium since the Pa-233 concentration is negligible early in core life.

The calculated axial power distribution is skewed toward the top of the movable seed assemblies early in core life and toward the bottom late in core life. This effect is shown by the data presented in Figure IV-14 for four 21-inch axial zones in the 84-inch binary fuel height. The variation in axial power distribution with lifetime is much less pronounced in the blanket regions than in the seed assemblies. Data in Figure IV-14 show that the central 42 inches (Zones 2 plus 3) in both the blanket and power-flattening blanket (PFB) account for about 68 to 77 percent of the region power. Note that the core conditions for axial power data, in Figure IV-14, are exactly the same as for the power data by module and regions, presented in Table IV-5.

Differences in axial power distributions between off-equilibrium conditions and equilibrium xenon and protactinium are lifetime dependent due to the variation in movable fuel worth with lifetime (see Section IV.E.3). Early in core life, the zero-xenon critical position is only about 5 inches lower than the equilibrium-xenon critical position and the gross axial power variation is small. The difference in critical position between equilibrium and single off-equilibrium conditions increases with time to about 12 to 15 inches late in core life. The calculated variations in gross axial power distribution between equilibrium and off-equilibrium conditions, at middle-of-life and end-of-life, are shown in Figure IV-15. Movable fuel positions for the 10,500-hour data in Figure IV-15 are comparable to a 1-percent overreactive core while those for the 18,000-hour data are comparable to a 1-percent underreactive core (see Figure IV-1). The lower movable fuel position for off-equilibrium conditions results in a shift in power toward the top of the seed assemblies relative to the equilibrium fuel position power distribution.

Calculated power distribution data presented in Figures IV-14 and IV-15 show that the gross axial power distributions are primarily dependent on the axial position of the movable seed relative to the stationary blanket and less dependent on the core conditions of depletion time and off-equilibrium xenon or protactinium. Comparison of the nominal core power data for 10,500 EFPH in Figure IV-14 with the +1 percent off-nominal core power data for 10,500 EFPH in Figure IV-15 shows that the axial power distributions and the seed, blanket, and PFB power split are comparable for nearly equal movable seed positions (-7 inch and -10.5 inch, respectively).

The nominal core power distributions at 22,000 EFPH, given in Figure IV-14, with a 17.5-inch upward seed position relative to the blanket, are roughly comparable to the 18,000 EFPH power distributions for the -1 percent off-nominal core, at a +14-inch seed position, given in Table IV-15. The small effect on gross axial power distribution from changes in fissile fuel distribution due to depletion is also shown by the data in Figures IV-14 and IV-15 where the seed fuel position is -17.5 inches relative to the aligned blanket position. The 100-hour data in Figure IV-14 are for the core with full power equilibrium xenon but very little protactinium. Data in Figure IV-15 (Set 2) are for comparable core conditions but after 10,500 EFPH. The 10,500-EFPH data show only a small shift in axial power toward the bottom of both seed and blanket regions, relative to the 100-EFPH distribution.

Calculated heat flux (Btu/hr-ft^2), linear power density (kw/ft) and fuel burnup data (fissions/cm^3 and megawatt days per metric ton (MWD/MT) of thorium plus uranium) are presented in Table IV-7. Note that the maximum levels given in Table IV-7 include all of the allowances and uncertainty factors discussed below in Section IV.C.3. The region average and the local maximum of heat fluxes and of linear power densities are presented in Table IV-7 for seed, blanket, and power-flattening blanket at 100 EFPH and for the reflector blanket at 18,000 EFPH, at which time significant buildup of U-233 has occurred in the reflector blanket. Region average and local maximum fuel burnup data are given in Table IV-7 for both 18,000 and 24,000 EFPH of core operation. Local heat flux, local fission density, and core locations where the maxima occur are discussed in the following section.

2. Local Power Distributions

The LWBR core is designed to be operated and controlled with all seed assemblies positioned in a uniform bank. Acceptable power distributions have been designed into the core by means of the radial and axial fuel zones and the specified U-233 content in each region of the core, as described in Section II. As a result, no operational procedures are required to shape or alter radial or axial power distributions by control element programming during full-power operation. At less than full-power operation, such as the swing load operation discussed in Section IV.B.4, the effects of spatial xenon variation are smaller than the effects on power distribution of operating in the xenon-free condition. Power

peaking calculations (described below) include the maximum effect of xenon variation and require that the core be operated at less than 100-percent power whenever the core is both xenon and protactinium free. The stability of the core against xenon-induced oscillations is discussed in Section IV.D.4.

Representative and calculated limiting local power distributions are presented in Figures IV-17 through IV-28 for various locations in the core. Figure IV-16 shows a sketch of a symmetric sixth-core slice through the 12 binary modules, identifies the locations of fuel rod rows (for which axial sketches are shown in Figures II-4 and II-5), and shows the locations of the rods and tracks for which axial and radial local power distributions are provided. Normalized power density data for each fuel rod type in each module type are presented in Figures IV-17, -18, and -19 for the LWBR core near beginning-, middle-, and end-of-life, respectively. These data have been normalized such that the 12-module average is unity. Data are not shown for the reflector blanket modules since their total power output is less than 3 percent of the total core power (see Table IV-5).

Axial power distributions are shown in Figures IV-20 through IV-25 where heat flux, in units of Btu/hr/ft^2 , is plotted for the entire length of the fuel rods. The data represent the averages of pointwise power calculated at each axial PDQ plane (spaced at 3.5-inch increments) for the core at 100-percent power and include the allowance and uncertainty factors presented in Section IV.C.3. Data for the limiting high-zone seed rod, in rod row 11 of the Type I module, are given in Figure IV-20 for three times in core life: (1) near beginning-of-life, with equilibrium xenon, when the seed assemblies are displaced downward relative to the blanket assemblies, (2) middle-of-life when seed and blanket assemblies are approximately axially aligned, and (3) near end-of-life when the seed assemblies are displaced upward relative to the blanket assemblies.

Axial power shapes for a 42-inch binary rod in seed rod Row 15, which contains the maximum local peak heat flux late in core life, are shown in Figure IV-21. Data for the limiting standard blanket and power-flattening blanket rods are presented in Figures IV-22 and IV-23, respectively. In Figures IV-24 and IV-25, axial power distributions are presented for a typical high-zone seed and high-zone blanket rod in the Type I module at times near beginning-of-life and end-of-life.

Calculated transverse (radial) power distributions are shown in Figures IV-26 and IV-27 for a track through the Type I module seed (Track A, identified in Figure IV-16). Data in Figure IV-26 are given for an axial location 80 inches above the bottom of the seed fuel (at the binary - ThO_2 interface of seed rod Row 12, shown in Figure II-4). The peak heat flux in the seed is at this elevation early in core life. Data in Figure IV-27 are given at an axial location 52 inches above the bottom of the seed fuel, through the axial mid-plane, which contains the peak heat flux around middle of core life. Similar calculated transverse power distributions are shown in Figure IV-28 for a track through the Type I module blanket (Track B, identified in Figure IV-16) at the axial mid-plane of the blanket rods (see Figure II-4).

As shown in Figures IV-20 and IV-24, the seed power peaks shift with life-time from the top to the bottom of the seed assembly. Calculated local heat flux (Btu/hr/ft^2) data for axial segments within the seed rods are shown in Figure IV-29 as a function of fission density. The data include the fuel pellets with maximum heat flux at beginning-, middle-, and end-of-life.

Blanket peak power as a function of time remains close to the axial mid-plane of the blanket fuel (see Figures IV-22 and IV-23). Calculated local heat flux data as a function of fission density are shown in Figure IV-30 for the blanket, power-flattening blanket, and reflector blanket rod locations with maximum pellet heat flux. The heat flux data shown in Figure IV-30 were obtained for movable fuel positions corresponding to the +1 percent off-nominal core (see Figure IV-1). This position yields the largest early-in-life peak heat fluxes for the blanket and power-flattening blanket fuel rods.

Data given on Figures IV-29 and IV-30 show that the maximum local heat fluxes in the seed, standard blanket, and power-flattening blanket regions are largest at beginning-of-life. The heat fluxes in these regions decrease sharply during early-in-life full-power operation. The LWBR core local power peaking factors are, therefore, largest at beginning-of-life. Peaking factors of local heat flux and hot rod power at beginning-of-life and burnup peaking factors near end-of-life are presented in Table IV-8 for the seed, standard blanket, and power-flattening blanket regions. These peaking factors include the allowances and factors presented in Section IV.C.3.

3. Allowances in Power Performance Data

Local pointwise power data and rod power data obtained from the three-dimensional PDQ calculations for LWBR were adjusted by several power allowance factors. These include allowances for uncertainties in the nuclear analysis model, for manufacturing tolerances including fuel misalignment, and for variations anticipated during core operation, which include instrumentation errors. Following is a discussion of the factors used to account for these parameters. The allowances in power performance data are presented in Table IV-9.

Experiment to calculation allowance factors were generated to account for uncertainties and approximations, such as axial homogenization rather than explicit grid levels, in the nuclear analysis mode. These factors are based on detailed comparisons of power distributions calculated using the LWBR model with those power distributions measured in a variety of experimental configurations. The experimental configurations, discussed in detail in References IV-2 and IV-3, contained U-233-O₂ ThO₂ loaded fuel rods but were smaller cores than the LWBR core. Experiment to calculation factors were based on comparisons of gross power sharing, row-by-row power sharing within the gross regions, individual rod-to-rod power shapes, and detailed axial shapes.

Fuel loading factors account for the variations in fissile loading in as-built fuel pellets and rods. Composition averaged loadings were used in the PDQ calculations. The fuel loading factors assigned for local heat flux, given in Table IV-9, are equal to the ratio of maximum measured grams fissile per inch to the average grams fissile per inch. The rod power fuel loading factor was set equal to the 95/95 tolerance interval; that is, two standard deviations from the mean.

The steady-state overpower allowance factors account for possible overpower operation at steady state. The steady-state overpower allowance accounts for:

1. The possibility of operating at 3-percent overpower due to instrument inaccuracy
2. The possibility that operators may allow core power to drift by 3 percent around nominal
3. The possibility of power peaking due to misalignments of a seed or blanket assembly, fuel rods, or binary-ThO₂ axial interfaces.

The misalignment contributions to the steady-state overpower allowances, given in Table IV-9, are based on as-built core measurements of Type I module misalignments since the limiting power seed and blanket rods are located in the Type I modules.

Early in core life the fission power in regions containing ThO_2 pellets is very low and a significant contribution to the heat in these regions is due to gamma energy. As a function of time, the buildup of U-233 in the ThO_2 regions results in an increase in fission power and a reduction in the relative fraction of heat from gamma energy. To correct for this effect, gamma heating factors are applied to the depletion calculated (fissioning) power data in the ThO_2 reflector blanket rods.

Note that the total power allowance factors, given in Table IV-9, are a conservative multiplicative combination of all individual factors without any statistical weighting.

Factors for off-nominal core effects are not included in the power allowances given in Table IV-9, because PDQ depletion calculations were done for off-nominal fuel positions. One set of power data versus time was calculated with a PDQ depletion which followed the +1 percent off-nominal fuel position versus time curve, shown in Figure IV-1. A second set of power data versus time was calculated by a PDQ depletion which used a combination of the +1 percent and -1 percent off-nominal fuel position curves, shown in Figure IV-1. Power data were calculated on the +1 percent path out to 8000 EFPH; two transition calculations at 10,500 and 13,000 EFPH were done at +1/2 and -1/2 percent off-nominal positions; then the -1 percent path was followed to end-of-life (~19,000 EFPH). Both of these sets of power distribution data, with the power allowance factors from Table IV-9 applied, were used in LWBR core performance analysis.

Allowances for off-equilibrium xenon or protactinium operation were obtained in a similar manner, that is by three-dimensional PDQ calculations of power distributions at equilibrium and off-equilibrium fuel positions, such as shown in Figure IV-15.

D. Kinetic Parameters and Stability

The nuclear parameters important to determining the kinetic behavior of the LWBR and associated power plant are evaluated using the nuclear calculational model described in Section III. The important core parameters include temperature and density coefficients of reactivity, effective delayed neutron fraction,

and prompt neutron lifetime. These parameters are used to study typical transients or hypothetical accidents. In the latter case, the best estimate of the parameter is modified to allow for possible uncertainties. A summary of the methods used for kinetic calculations of LWBR is contained in Reference IV-4 (Section 15.1.0.6).

1. Temperature and Density Coefficients of Reactivity

Reactivity coefficients reflect the changes in neutron multiplication due to varying plant conditions such as fuel or moderator temperatures, core power level, or less significantly changes in pressure or void conditions.

In the power operating range, reactivity feedback effects, which tend to compensate for a rapid increase in reactivity, are provided by the negative fuel temperature effect (Doppler effect) and by the negative moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is about twice as large as in a low-enrichment uranium fueled plant due to the $^{238}\text{ThO}_2$ fuel in the LWBR core. Reduction in the moderator density, as a function of increasing moderator temperature, results in a negative reactivity feedback in LWBR.

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per change in fuel temperature and is a measure of the Doppler broadening of Th-232 resonance absorption peaks. Doppler broadening of other isotopes such as Zr, Pa-233, U-233, etc., is also considered, but their contributions to the Doppler effect are small. An increase in fuel temperature increases the effective resonance absorption cross sections of the fuel and produces a corresponding reduction in reactivity.

The moderator temperature coefficient is defined as the change in reactivity per degree change in moderator temperature. An increase in moderator temperature causes a decrease in moderator density resulting in less moderation and a hardened neutron spectrum which causes a decrease in reactivity.

Three-dimensional PDQ calculations, using the LWBR nuclear calculational model, are utilized for determining at-power and zero-power coefficients. Calculated reactivity coefficients are presented in Table IV-10.

The at-power coefficients are determined from a set of temperature feedback PDQ calculations, all at the movable seed bank position and core lifetime of

interest. The at-power Doppler coefficient gives the effect of fuel temperature and fuel temperature distribution changes on reactivity with the constraint of an unchanged moderator temperature and moderator temperature distribution. This effect is calculated by varying both the power level and flow rate so that the ratio of power level-to-flow rate (and moderator temperature) is held constant. The magnitude of the Doppler coefficient is generally smaller later in life than at beginning-of-life. This is a result of changes in the neutron energy spectrum and the relatively smaller absorption in Th-232 as the fission product level increases.

The at-power moderator coefficient gives the reactivity effect of changes in moderator temperature and moderator temperature distribution. This is calculated by starting from a steady-state critical condition and changing both the power level and inlet temperature so that another steady-state critical condition is obtained. This procedure changes both the fuel and moderator temperatures and temperature distributions, and requires using the Doppler coefficient to isolate the effect due to moderator changes. Since the LWBR core does not use either poison rods or burnable poison for control purposes, the variation in moderator temperature coefficient with lifetime is due primarily to changes in the neutron energy spectrum. Two sets of calculated at-power moderator temperature coefficients are presented in Table IV-10. The first set, for fast transients, accounts for changes in moderator temperature in the fuel rod lattice, but assumes no change in water temperature in metal-water seed-blanket and blanket-blanket channels. The second set, for slow transients, assumes sufficient mixing time to achieve comparable changes in both fuel region and channel water temperatures.

Zero-power (uniform) temperature coefficients are determined by performing two complete sets of PAX-PDQ calculations with uniform temperatures and no temperature feedback which is equivalent to zero power conditions. The two sets of calculations are identical except for the perturbation of a single core parameter such as moderator temperature and/or fuel temperature.

The uniform temperature coefficient at ambient temperature ($-0.66 \times 10^{-4} \Delta\rho/^\circ\text{F}$ given in Table IV-10) was calculated for a 10 F change in both fuel and moderator temperature (150 to 160 F) and therefore constitutes a total coefficient. The magnitude of the calculated ambient temperature coefficient is about one-third of

the hot temperature coefficient due to the much smaller change in water (moderator) density per degree F at ambient temperature. The value of $-0.66 \times 10^{-4} \Delta\rho/^\circ\text{F}$, given in Table IV-10, is actually the sum of two effects: (1) the effect of fuel and moderator temperature change, without any change in fuel rod spacing ($=-0.70 \times 10^{-4} \Delta\rho/^\circ\text{F}$, and (2) the effect of increasing the fuel rod spacing due to grid thermal expansion ($=+0.04 \times 10^{-4} \Delta\rho/^\circ\text{F}$).

Hot, zero power uniform temperature coefficients were calculated with a 531 F reference temperature and a 540 F perturbed temperature. The calculated uniform Doppler temperature coefficient of $-0.37 \times 10^{-4} \Delta\rho/^\circ\text{F}$ is smaller in magnitude than the calculated "at-power" coefficient of $-0.62 \times 10^{-4} \Delta\rho/^\circ\text{F}$. However, the at-power coefficient was defined for use in kinetics calculations as the change in reactivity per change in fuel temperature averaged over the entire 12 binary modules, including the ThO_2 step regions and axial end blankets. The relative temperature changes are about 0.02 F in ThO_2 regions and 1.53 F in binary regions. Thus, the at-power Doppler coefficient defined in terms of binary fuel temperature change would be $-0.62 \times 10^{-4} / 1.53$ or $-0.41 \times 10^{-4} \Delta\rho/^\circ\text{F}$, which is in reasonable agreement with the uniform coefficient of $-0.37 \times 10^{-4} \Delta\rho/^\circ\text{F}$.

The calculated pressure coefficient of reactivity, given in Table IV-10, is of much less significance in comparison with the moderator temperature coefficient. A pressure change of 50 to 100 psi has approximately the same effect on reactivity as a 1-degree change in moderator temperature. The calculated value for pressure coefficient of reactivity, $+2.44 \times 10^{-6} \Delta\rho/\text{psig}$, is composed of contributions from both coolant density increase ($1.91 \times 10^{-6} \Delta\rho/\text{psig}$) and fuel rod clad elastic shrinkage ($0.53 \times 10^{-6} \Delta\rho/\text{psig}$) during pressure increases.

2. Power Coefficient of Reactivity

The combined reactivity effect of moderator and fuel temperature changes as core power level changes is called the total power coefficient and is expressed in terms of reactivity change per percent power change.

The total power coefficient; that is, the reactivity change in going from hot, zero power to 100-percent power is calculated to be $-0.018 \Delta\rho$ at beginning-of-life and $-0.015 \Delta\rho$ at end-of-life.

Calculated required reactivity addition as a function of core power level, to compensate for the negative power coefficient of reactivity, is shown in Figure IV-31 for beginning-of-life and end-of-life. The fuel motion required to add reactivity for increasing power from zero to 100 percent, shown in Figure IV-3, varies from about 4 inches at beginning-of-life to over 12 inches late in core life ($\approx 20,000$ EFPH).

3. Delayed Neutron Fraction

The data and procedure for determining the effective fractions of fission neutrons that are delayed are discussed in Sections IV.B.11 and VII.A.2 of Reference IV-6, respectively. The effective delayed fraction for LWBR is small compared to U-235 fueled reactors because the U-233 delayed neutron yield is less than half that of U-235. For LWBR, the effective delayed neutron fraction is calculated to be nearly constant, at about 0.0033, through lifetime. In a conventional core with U-235 enriched fuel, the effective delayed neutron fraction decreases from approximately 0.0075 initially to approximately 0.0045 for depleted fuel. This decrease is due to the reduction in U-235 content and the increase of Pu-239, which also has a low delayed neutron yield similar to U-233. The fissioning of thorium in LWBR contributes approximately 15 percent of the total delayed neutrons although the corresponding fraction of core fissions from thorium is less than 3 percent throughout core life. The third most important fissionable isotope is U-235 which contributes approximately 0.1 and 1.6 percent to the total delayed neutron population at beginning-of-life and end-of-life, respectively.

Delayed neutron fractions, β , based on weighting the fission fractions of U-233, U-235, and Th-232 by their respective delayed yields are shown in Table IV-10 at beginning-of-life and end-of-life for hot and cold conditions. These values are based on PDQ calculations for these various conditions. Also listed in the same table is the effectiveness factor, $\bar{\beta}/\beta$, as determined using the RCP model for beginning-of-life conditions. Since the isotopic fission distribution does not change appreciably with core lifetime, this factor is considered appropriate for all four of the conditions listed in the table. One standard deviation uncertainty on the β values is approximately 4.5 percent

and on the $\bar{\beta}/\beta$ is estimated to be less than 1 percent. The effective core delayed neutron fraction for any of the four core conditions is the product of the appropriate β value from the table and the $\bar{\beta}/\beta$ factor.

4. Prompt Neutron Lifetime

The procedure for determining prompt neutron lifetime is discussed in Section VII.A.3 of Reference IV-6. PDQ calculations of a Type I module were used to determine beginning-of-life and end-of-life values as listed in Table IV-10. The decrease in neutron lifetime at end-of-life relative to beginning-of-life is attributed to spectrum changes which result in increased epithermal and decreased thermal absorption fractions. Thus fewer neutrons reach the thermal energy range from which most of the neutron lifetime is derived.

5. Stability Against Power Oscillations

Power distribution in a reactor can undergo oscillations either in the axial or transverse directions, due to perturbations introduced in the xenon distributions, without changing the total core power. The stability of a core against xenon-induced oscillations can be determined in terms of $\Delta\lambda$, the difference between the fundamental and overtone mode λ -eigenvalues, λ_0 and λ_1 . Equivalently, the dominance ratio, $\bar{\sigma} = \lambda_1/\lambda_0$, can be used in place of $\Delta\lambda$, the relation between them being (Equation II-78 in Reference IV-5):

$$\lambda_0 - \lambda_1 = \Delta\lambda = \lambda_0 (1 - \bar{\sigma}) \quad (1)$$

In addition, the relationship between the conventional μ -mode equation and the dominance ratio (Section II-C, Reference IV-5) can be used as an indicator of core stability. In particular, when

$$\bar{\sigma} \leq \left[1 + \frac{\eta(\gamma_I - \gamma_{Xe})}{(1 + \eta) (1 + \lambda_I/\lambda_{Xe} + \eta) v'} \right]^{-1} \quad (2)$$

the core is necessarily stable. In the above equation

γ_I, γ_{Xe} = direct I-135 and Xe-135 fission yields

λ_I, λ_{Xe} = I-135 and Xe-135 decay constants

$\eta = \bar{\sigma}_{Xe} \phi^{th} / \lambda_{Xe}$, where $\bar{\sigma}_{Xe}$ is the Xe-135 cross section and ϕ^{th} is the steady state thermal flux

$v' = \frac{\langle \phi^*, M\phi \rangle}{\langle \phi^*, Q\phi \rangle}$ where M = fission matrix times v

Q = matrix having only a thermal fission cross section

ϕ^* and ϕ = the dominant overtone adjoint and normal fluxes.

Calculations of the $\bar{\sigma}$ required for LWBR stability were performed using the least conservative values for the parameters in Equation (2). Since Equation (2) is minimized for maximum η and minimum ν , values of $\eta = 1.0$ and $\nu = 1.3$ were used, where possible variations for the LWBR core are $0.6 < \eta < 1.0$ and $1.3 < \nu < 2.1$. Direct fission yields for U-233 of 0.048 to I-135 and 0.012 to Xe-135 (total = 0.060) were used since U-233 provides greater than 96 percent of the power in LWBR throughout lifetime.

Substitution of LWBR parameters into Equation (2) yields a value of $\bar{\sigma} < 0.996$ required for LWBR stability (in the absence of the known stabilizing feedback effects from the temperature coefficients). One of the stabilizing effects with a U-233 fueled core is the fact that about 20 percent of the Xenon-135 is produced directly from fission for U-233 as compared with about 5 percent for U-235. Note however that uncertainty in the distribution of mass 135 fission yields would not greatly affect the calculated results. Applying the total of 0.06 to I-135 (and 0.0 to Xe-135) would yield 0.993 instead of 0.996.

Dominance ratios ($\bar{\sigma} = \lambda_1/\lambda_0$) obtained from three-dimensional PDQ calculations yield the following for LWBR:

$$\bar{\sigma} \leq 0.97 \text{ for the axial mode}$$

$$\bar{\sigma} \approx 0.96 \text{ in the } r\text{-}\phi \text{ plane.}$$

The large margin of stability for the LWBR core can be seen by comparing the value of $(1 - \bar{\sigma})$, which must be exceeded for stability (0.004), with the value of $(1 - \bar{\sigma})$ of the LWBR core (greater than 0.03) which is nearly an order of magnitude larger than required for stability. The negative Doppler and moderator coefficients will further increase the axial and $r\text{-}\phi$ stability, but these additional stabilizing parameters were not required in the above analysis to assure xenon stability for the LWBR.

E. Shutdown Performance and Control Requirements

To eliminate the parasitic loss of neutrons to conventional control rod poison materials, core reactivity control for LWBR is achieved by varying the geometric relationship between the movable seed fuel assemblies and the associated stationary annular blanket assemblies.

Changing the axial position of the seed assembly relative to the blanket assembly changes the relative amounts of neutron absorptions in the fissile

(U-233) and fertile (thorium) fuel materials. The design arrangement of fissionable uranium-233 and thorium is such that when the central region (the seed) of a fuel module is lowered, the neutron absorption in thorium is increased relative to absorption in uranium-233, lowering reactivity. Conversely, raising the seed decreases the absorption in thorium and increases reactivity. The LWBR core has natural thorium "step" regions in the top half of seed rods and in the bottom half of blanket rods (see Figure II-4) close to the seed-blanket interface. The fuel worth achieved by this fuel distribution is largest for lower positions of the movable fuel. As the core depletes, U-233 builds up in the step region, lowering movable fuel worth. The calculated fuel worth as a function of time at the critical position for the LWBR core decreases from approximately $0.0042 \Delta\rho/\text{inch}$ near beginning-of-life to approximately $0.0005 \Delta\rho/\text{inch}$ at end-of-life. The low fuel worth at higher positions of the movable fuel is the major contributor to the large change in the movable fuel positions towards end-of-life shown in Figure IV-1.

Early in core life, a small part of the movable fuel worth is due to the reduction in axial neutron leakage (shown in Figure IV-7) for upward motion of the fuel toward the seed blanket aligned position. Data in Figure IV-7 show that the neutron absorption fraction in axial water and structure (axial leakage) is calculated to decrease from 0.0043 to 0.0020 for an upward fuel motion of 10 inches, from 37 to 47 inches. This yields a change in axial leakage per inch of fuel motion of 2.3×10^{-4} . Calculated movable fuel worths, shown in Figure IV-2, are 42×10^{-4} and $32 \times 10^{-4} \Delta\rho/\text{inch}$ at the 37 and 47 inch fuel positions respectively. The average movable fuel worth between the 37-inch and 47-inch positions is $37 \times 10^{-4} \Delta\rho/\text{inch}$; therefore, the magnitude of the axial leakage change per inch (2.3×10^{-4}) is approximately 6 percent of the movable fuel worth near beginning-of-life. Note also that the decrease in movable fuel worth of $10 \times 10^{-4} \Delta\rho/\text{inch}$ between the 37-inch and 47-inch fuel positions is 4.3 times as large as the change in axial leakage per inch of fuel motion. Thus, the change in axial leakage is not the predominating effect in the change of movable fuel worth as a function of fuel position and lifetime.

Fuel motion above the 60-inch position later in core life results in increased axial leakage (see Figure IV-7). Thus, late in core life the axial leakage has a negative reactivity effect for upward fuel motion but the net reactivity change is positive such that the movable fuel worth is positive (as shown in Figure IV-2) for upward motion of the movable fuel.

The fuel distribution in the core is designed so that over the entire range of fuel movement, the lowering of the seeds always lowers core reactivity. Therefore, from an operational point of view, movable fuel control functions exactly like conventional poison rod control. During normal power operation, all seeds are aligned as a uniform bank and they provide adequate reactivity control and shutdown margin for all anticipated operational requirements of the reactor (that is, xenon poison override, protactinium reactivity changes, lifetime reactivity changes, negative temperature and power reactivity coefficients, hot-to-cold reactivity defect, and cold shutdown).

Shutdown margin is also available for some period of time under abnormal core conditions, such as a movable fuel assembly stuck in the operating position. Long term shutdown margin under abnormal core conditions could require addition of soluble boron to the coolant. A Chemical Shutdown System, discussed in Section 6.5 of Reference IV-4, is provided for the addition of concentrated soluble boron to ensure shutdown margin under abnormal core conditions.

1. Nominal Control and Shutdown Margin

Calculated excess reactivity (all movable fuel 7 feet above the shutdown position) as a function of time is given in Figure IV-32 for cold clean conditions, where xenon-135 and protactinium-233 are assumed to have decayed to cesium and uranium-233, respectively. The calculated excess reactivity is about 23 percent at beginning-of-life and gradually decreases to about 11 percent after 22,000 full-power hours. The calculated reactivity for conditions of hot (zero power) with Xe-135 decayed out is also given in Figure IV-32 for a maximum movable seed position of 7 feet above the shutdown position.

Calculated normal core shutdown reactivity, with movable seed assemblies at the fully-inserted position, is also shown as a function of lifetime in Figure IV-32 for three core conditions. Figure IV-32 shows that the control swing in LWBR diminishes with depletion. Late in core life, in the cold (68 F) condition, the core nominally exhibits approximately 5-percent reactivity shutdown margin with all modules in the fully-inserted position. The shutdown available for earlier times in core life is greater than the end-of-life values. The shutdown quoted is for a condition where the core is both xenon-free and protactinium-free. Xenon decays away in approximately 60 hours while protactinium decays in

about 3000 hours after power shutdown. As Figure IV-32 shows, approximately 1.5 percent additional reactivity shutdown margin is available in the xenon-free but equilibrium-protactinium condition.

Calculated nominal core critical fuel positions versus time for various temperatures, xenon, and protactinium conditions were presented in Section IV.A and are shown in Figure IV-3. Data on the reactivity increases for the conditions shown in Figure IV-3 are given in Table IV-11 for beginning-of-life and end-of-life. The shutdown margin, obtained by lowering the movable assemblies to the minimum position, is also given in Table IV-11. The sum of these reactivity increments yields the reactivity worth of the inserted amount of movable assemblies when positioned at the full-power (operating) critical position.

2. Transient Poison Effects on Fuel Worth

Reactivity worth for all 12 movable assemblies moved in a bank is shown in Figure IV-2, as a function of critical position, for a nominal core operated continuously at full power. An interruption in full-power operation, which causes a change in core reactivity due to the effects of Xe and Pa, results in a change in movable fuel worth due to the change in critical fuel position. Calculated fuel worths (as a function of lifetime) for a nominal core under equilibrium and off-equilibrium conditions are shown in Figure IV-33.

The off-equilibrium effect which decreases reactivity is maximum xenon. A peak xenon condition is achieved after a core shutdown of about 4 hours from a full-power equilibrium condition. Returning the core to full power under peak xenon conditions, at any time in core life, reduces the reactivity by about 0.2 percent relative to full-power equilibrium xenon. The change in fuel worth due to the higher movable fuel critical position varies between $-0.00005 \Delta k/\text{inch}$ early in life and $-0.0001 \Delta k/\text{inch}$ near end-of-life, as shown in curve 2 in Figure IV-33.

Off-equilibrium effects which result in increased reactivity are reductions of power level, of Xe concentration, and of Pa concentration. Both the Xe and Pa concentrations are reduced somewhat if the core is operated at a reduced power level; however, the largest reduction in Xe and Pa occurs due to decay while the core is at zero power. As discussed earlier, the Xe level increases to peak Xe during the first 4 hours after shutdown and the core reactivity decreases by

0.2 percent. For shutdown times beyond 4 hours, core reactivity increases as the Xe level decreases. After about 8 hours at zero power, the Xe level has decreased back to the full-power equilibrium level.

The reactivity worth of protactinium in the core increases from zero to a peak of about 2.5 percent during the initial 3000 full-power hours of operation then decreases to about 2.3 percent late in life (Table IV-11). Note that these protactinium worths account for both the increase in reactivity resulting from an increase in U-233, due to decay of Pa-233 (roughly one-third of the total protactinium worth), and the increase in reactivity due to reduction of neutron losses to Pa-233 absorption.

As a function of time after shutdown, the Pa level decreases much more slowly than the Xe level due to the difference in half-lives (about 9 hours for Xe and 27.0 days for Pa). Thus, a xenon-free condition can exist while most of the Pa is still present. Curve 3 in Figure IV-33 shows the fuel worth as a function of lifetime (with the additional core reactivity and resulting lower movable fuel critical position) for the core at zero power, zero xenon and with half of the Pa decayed to U-233. Similarly, Curve 4 in Figure IV-33 shows the fuel worths following a long period at zero power (>3000 hours) which is sufficient for almost complete decay of protactinium.

The calculated critical fuel positions as functions of lifetime, for ± 1 percent off-nominal reactivities are shown in Figure IV-1. Since off-nominal reactivity affects the time in life at which the movable fuel reaches a particular fuel position, the off-nominal full-worth curves are very similar to those shown in Figure IV-33 for the nominal core. The primary difference is that the fuel reaches the same position about 3000 EFPH earlier for the -1 percent off-nominal core and about 3000 EFPH later for the +1 percent off-nominal core, relative to the nominal core lifetime scale shown in Figure IV-33.

3. Performance with One Movable Fuel Assembly Withdrawn

Calculated core reactivity throughout life with the seed of the most reactive module, a Type III, stuck at the full power (equilibrium Xe and Pa) operating position is shown in Figure IV-34. If a movable fuel assembly should become stuck during normal operations, the core would be successfully shut down under hot, zero power conditions with a shutdown margin of at least 4 percent.

Figure IV-34 shows k_{eff} levels for hot, zero power conditions in the range 0.91 to 0.96 depending on the amount of xenon and protactinium which have decayed and on the lifetime.

After some period of time, if attempts to lower the stuck assembly are not successful, the insertion of soluble boron into the primary coolant would be required. Data given in Figure IV-34 show that criticality could occur during plant cooldown with the reactivity addition resulting from the decay of xenon and protactinium. Provisions have been made for the injection of sufficient chemical poison to ensure shutdown capability under cold, xenon and protactinium free, conditions with the most reactive module stuck. The critical fuel position under a reduced power or off-equilibrium xenon and/or protactinium condition would be lower in the core, and hence, with a module stuck at these positions the shutdown margin would be greater than shown in Figure IV-34. Under off-nominal conditions, the full ± 1 percent off-nominal reactivity allowance is not required in evaluating shutdown with one module stuck at the operating position. At a given time in life, the position of the stuck fuel element will be different for a plus or minus off-nominal reactivity core, but in all cases the shutdown occurs from a critical bank position. Thus, the stuck module reactivity for an off-nominal reactivity core will differ from the nominal core data, shown in Figure IV-34, by less than one percent.

A calculation has also been made to determine the temperature for criticality in the case of one Type III module being stuck at the hot operating position and the other 11 modules inserted. The equilibrium-xenon, equilibrium-protactinium core, with a stuck module near end-of-life, has a hot shutdown margin of about 7-percent reactivity. This would approach criticality during cooldown, as shown in Figure IV-34. The temperature at which criticality is calculated to occur during cooldown, with the most reactive movable assembly stuck at the hot operating position, is shown in Figure IV-35 as a function of core lifetime. The nominal (best estimate) and maximum (best estimate $+2$ percent Δk for reactivity uncertainty) temperatures both include an allowance for the complete decay of xenon prior to cooldown to the given temperature. Allowance for Pa-233 decay is not included because of the very slow rate at which protactinium decays.

Under conditions where the most reactive movable assembly is withdrawn to the maximum 7-foot position and the other movable assemblies are fully inserted,

the core is calculated to be critical under hot (zero power) conditions at the beginning-of-life as shown in Figure IV-36. The cold core excess reactivity would be about 10 percent under these same conditions, also shown in Figure IV-36. Therefore, it is necessary to limit the movable fuel control mechanism testing prior to core operation to less than the maximum 7-foot position. The upper limit to which the movable fuel may be exercised, with a 1-percent shutdown margin at beginning-of-life, is about 71 inches for the hot, nominal reactivity core and 65 inches or 77 inches for the plus or minus 1-percent off-nominal core, respectively. Fuel position as a function of lifetime data, given in Figure IV-1, shows that mechanism testing prior to core operation can be accomplished to cover fuel positions to be encountered during several thousand hours of full-power operation.

The core reactivity with one movable assembly fully withdrawn to the maximum 7-foot position and the other 11 assemblies fully inserted, decreases throughout lifetime. Calculated data in Figure IV-36, for the hot xenon-free core condition, show that relatively early in core life, hot shutdown could be maintained with one module fully withdrawn even for the 1-percent overreactive off-nominal core. Hot mechanism testing to the maximum position of 7 feet can be done with a 1-percent shutdown margin, following about 2000 hours of core operation for a nominal core ($k_{eff} < 0.99$ in Figure IV-36) or about 5000 hours of core operation for a 1-percent overreactive off-nominal core ($k_{eff} < 0.98$ in Figure IV-36).

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TABLE IV-1. LWBR CONVERSION RATIO CALCULATION

1. Depletion Model Calculated Losses	3000 Hours	20,500 Hours
Hydrogen	0.046235	0.040161
Oxygen	0.006723	0.006648
Zirc-4	0.042020	0.039603
Am-350	0.014597	0.012676
Inc-718	0.002763	0.001732
SS-304	0.009638	0.012956
Total Structure	0.121976	0.113777
Pa-233	0.029276	0.023320
U-236	0.000066	0.000201
Xe-135	0.024497	0.023554
Nd-147	0.000011	0.000010
Pm-147	0.001820	0.007053
Pm-148M	0.000396	0.001755
Pm-148	0.000169	0.000602
Pm-149	0.000030	0.000032
Sm-149	0.007365	0.002487
Sm-150	0.000169	0.001330
Sm-151	0.001859	0.001215
Residual fission products	0.011325	0.067222
Fission Products (less Xe-135)	0.023145	0.092444
<u>Total PDQ Calculated Losses</u>	0.198960	0.253296
2. <u>PDQ to RCP Bias</u>	0.0027	0.0027
3. Effects not in PDQ Calculation		
Presence of 0.373% U-238	-0.00038	
Impurities in fuel	-0.00038	
Impurities in AM-350	-0.00015	
Presence of instrumentation	-0.00020	
Crud deposition in core	-0.00100	
Radial coolant flow	+0.00005	
Small depletion time increments	-0.00025	
Depletion of Th-233	-0.00005	
<u>Total Effects Not in PDQ Calculation</u>	0.0024	0.0024
TOTAL CONVERSION RATIO LOSSES	0.204060	0.258396
η	2.256530	2.251609
CR = η - 1 - Losses	1.052470	0.993213

TABLE IV-2. LWBR CORE ABSORPTION FRACTIONS AT 8000 EFPH
ORDERED BY MAGNITUDE

Heavy Elements	0.9165
Clad, Water, and Structure	0.0525
Fission Products	0.0310

Th-232	0.46265
U-233	0.43779
H	0.01947
Zirc-4	0.01795
Xe-135	0.01062
U-234	0.00768
AM-350	0.00612
Pa-233	0.00606
SS-304	0.00465
Sm-149	0.00371
O	0.00291
Pm-147	0.00185
Inc-718	0.00136
Sm-151	0.00134
U-235	0.00130
Pa-231	0.00085
Pm-148M	0.00047
Sm-150	0.00023
Pm-148	0.00017
U-232	0.00010
U-236	0.00004
Pm-149	0.00001
Other Fission Products	<u>0.01267</u>
	1.00000

TABLE IV-3. INITIAL AND END-OF-LIFE FISSILE LOADING DISTRIBUTION

<u>Core Region</u>	<u>Fissile Loading (kg)</u>	
	<u>Beginning -of-Life</u>	<u>20,500 EFPH</u>
<u>Seed</u>		
High Zoned	137.3	110.2
Low Zoned	61.3	47.0
Thoria	<u>0.0</u>	<u>10.5</u>
Total	198.6	167.7
<u>Standard Blanket</u>		
High Zoned	57.7	55.5
Medium Zoned	42.7	43.7
Low Zoned	16.0	18.5
Thoria	<u>0.0</u>	<u>18.2</u>
Total	116.4	135.9
<u>Power Flattening Blanket</u>		
High Zoned	162.3	145.7
Medium Zoned	13.6	12.7
Low Zoned	10.2	10.0
Thoria	<u>0.0</u>	<u>8.1</u>
Total	186.1	176.5
Radial Reflector Blanket	0.0	27.6
Core Total	501.1*	507.7†

*Includes 0.44 kg of U-235.

†Includes 3.9 kg of U-235 and 10.3 kg of Pa-233 which will have decayed to U-233 following core shutdown.

TABLE IV-4. LWBR FISSION LOADING (kg) BY MODULE TYPE

<u>Module Regions</u>	<u>Module I</u>		<u>Module II</u>		<u>Module III</u>	
	<u>BOL†</u>	<u>EOL‡</u>	<u>BOL</u>	<u>EOL</u>	<u>BOL</u>	<u>EOL</u>
Seed	16.55	13.63	16.55	13.91	16.55	14.18
Standard Blanket	16.18	18.95	9.34	10.92	6.63	7.71
Power Flattening Blanket	None	None	15.66*	14.77*	23.22*	22.06*
Total Blanket	16.8	18.95	25.00*	25.69*	29.85*	29.77*
Module Total	32.73	32.58	41.55*	39.60*	46.40	43.95*

*Two Type II and two Type III modules have about 0.06 kg less loading due to flux well locations.

†BOL = Beginning-of-Life

‡EOL = End-of-Life (20,500 EFPH)

TABLE IV-5. DISTRIBUTION OF POWER IN LWBR - PERCENT OF POWER

Nominal, Full Power, Equilibrium Xenon, and Protactinium

<u>Core Region and Fuel Types</u>	<u>100 Hours</u>	<u>1000 Hours</u>	<u>10,500 Hours</u>	<u>22,000 Hours</u>
<u>3 Type I Modules</u>				
Seed	11.46	11.53	11.16	10.36
Blanket	14.17	13.79	14.12	14.30
Total	25.63	25.32	25.28	24.66
<u>3 Type II Modules</u>				
Seed	9.84	10.02	9.78	9.25
Standard Blanket	7.73	7.56	7.75	7.91
Power Flattening Blanket	8.77	8.63	8.12	7.83
Total	26.34	26.21	25.65	24.99
<u>6 Type III Modules</u>				
Seed	16.68	17.23	17.13	16.80
Standard Blanket	10.17	10.04	10.31	10.70
Power Flattening Blanket	21.03	20.97	20.00	19.92
Total	47.88	48.24	47.44	47.42
Total Seed in 12 Modules	37.97	38.78	38.07	36.41
Total Standard Blanket	32.07	31.39	32.17	32.91
Total Power Flattening Blanket	29.81	29.60	28.12	27.75
Total Reflector Blanket	0.15	0.23	1.64	2.93

TABLE IV-6. VARIATION IN POWER DISTRIBUTION DUE TO ZERO XENON OR ZERO PROTACTINIUM-233

Nominal, Full Power, Equilibrium Xenon or Pa-233

<u>Core Region and Fuel Zones</u>		<u>Power for either Xe or Pa Free/Power with Equilibrium Xe and Pa</u>		
		<u>Early in Life</u>	<u>Middle of Life</u>	<u>Late in Life</u>
Type I modules	- Seed	0.976	0.989	1.018
	Blanket	1.014	1.047	1.028
	Total	0.997	1.022	1.024
Type II modules	- Seed	0.974	0.969	1.000
	Standard blanket	1.014	1.037	1.018
	Power flattening blanket	1.031	1.012	0.998
	Total	1.005	1.003	1.005
Type III modules	- Seed	0.965	0.947	0.978
	Standard blanket	1.010	1.023	1.003
	Power flattening blanket	1.021	0.992	0.979
	Total	0.999	0.983	0.984
Total seed		0.971	0.965	0.995
Total standard blanket		1.013	1.037	1.018
Total power flattening blanket		1.024	0.998	0.984
Total reflector blanket		1.017	1.108	1.015

TABLE IV-7. HEAT FLUX, POWER DENSITY AND FUEL BURNUP

<u>Heat Flux (10^3 Btu/hr-ft²)*</u>	<u>Average</u>		<u>Maximum†</u>	
Seed	59.5		375	
Standard blanket	61.5		267	
Power flattening blanket	60.2		285	
Reflector blanket	4.2		77	
<u>Linear Power Density (kw/ft)*</u>	<u>Average</u>		<u>Maximum†</u>	
Seed	1.4		8.8	
Standard blanket	2.7		11.7	
Power flattening blanket	2.4		11.5	
Reflector blanket	0.3		4.7	
<u>Fuel Burnup (10^{20} fissions/cm³)</u>	<u>Average</u>		<u>Maximum†</u>	
	<u>18,000 EFPH</u>	<u>24,000 EFPH</u>	<u>18,000 EFPH</u>	<u>24,000 EFPH</u>
Seed	2.6	3.4	9.7	12.0
Standard blanket	1.3	1.3	4.3	5.6
Power flattening blanket	1.2	1.5	4.6	5.9
Reflector blanket	0.03	0.04	0.6	1.0
<u>Fuel Burnup (10^3 MWD/MT of Th + U)</u>	<u>Average</u>		<u>Maximum†</u>	
	<u>18,000 EFPH</u>	<u>24,000 EFPH</u>	<u>18,000 EFPH</u>	<u>24,000 EFPH</u>
Seed	12.5	16.7	45.3	55.9
Standard blanket	5.9	7.9	19.0	24.5
Power flattening blanket	5.6	7.4	20.5	26.1
Reflector blanket	0.16	0.21	2.8	4.2

*Heat fluxes and linear power densities are for 100 EFPH except for the reflector blanket which are for 18,000 EFPH.

†Maximum values include all allowances discussed in Section IV.C.3 for uncertainties in the nuclear analysis model, off-nominal reactivity, operational variations, and for an experiment-to-calculation bias.

TABLE IV-8. LWBR CORE POWER AND BURNUP PEAKING FACTORS†

<u>Core Region</u>	<u>Peak Local Heat Flux*</u> <u>Region Avg. Heat Flux</u>	<u>Peak Rod Power*</u> <u>Region Power</u>	<u>Peak Local Fissions/cm³†</u> <u>Region Avg. Fissions/cm³</u>
Seed	6.3	2.3	3.7
Standard blanket	4.3	2.0	3.3
Power flattening blanket	4.7	2.2	3.8

*Near beginning-of-life

†Near end-of-life

‡Peak values include all allowances discussed in Section IV.C.3 for uncertainties in the nuclear analysis model, off-nominal reactivity, operational variations, and for an experiment-to-calculation bias.

TABLE IV-9. ALLOWANCES IN POWER PERFORMANCE DATA

A. Seed Region

	<u>Local Heat Flux</u> <u>Fuel Rod Rows</u>			<u>Rod Power</u> <u>All Rod Rows</u>
	<u>1-12</u>	<u>13</u>	<u>14-15</u>	
Experiment/Calc.	1.128	1.180	1.102*	1.072*
Fuel Loading	1.013	1.013	1.013	1.008
Steady-state Overpower	<u>1.084</u>	<u>1.084</u>	<u>1.084</u>	<u>1.080</u>
Total Factor	1.239	1.296	1.210*	1.167*

* For rod Row 15, a 1.066 factor is applied to rods adjacent to support posts.

B. Standard Blanket Region

	<u>Local Heat Flux</u> <u>Fuel Rod Rows</u>				<u>Rod Power</u> <u>Fuel Rod Rows</u>		
	<u>1-2</u>	<u>3</u>	<u>4</u>	<u>5-6</u>	<u>1-3</u>	<u>4</u>	<u>5-6</u>
Experiment/Calc.	1.207	1.176	1.166	1.187*	1.098	1.120	1.140*
Fuel Loading	1.009	1.009	1.009	1.009	1.004	1.004	1.004
Steady-state Overpower	<u>1.071</u>	<u>1.071</u>	<u>1.071</u>	<u>1.071</u>	<u>1.076</u>	<u>1.076</u>	<u>1.076</u>
Total Factor	1.304	1.271	1.260	1.283*	1.186	1.210	1.232*

*Applied a "post factor" of 0.95 to rod in Row 5 adjacent to post at core center; 0.98 to Row 5 rods adjacent to other posts.

TABLE IV-9. ALLOWANCES IN POWER PERFORMANCE DATA (Cont'd)

C. Power Flattening Blanket Region

	<u>Local Heat Flux</u> <u>Fuel Rod Rows</u>			<u>Rod Power</u> <u>All Rod Rows</u>
	<u>1-2</u>	<u>3</u>	<u>4-10</u>	
Experiment/Calc.	1.235	1.193	1.160	1.114
Fuel Loading	1.009	1.009	1.009	1.004
Steady-state Overpower	<u>1.071</u>	<u>1.071</u>	<u>1.071</u>	<u>1.076</u>
Total Factor	1.335	1.289	1.254	1.203

D. Reflector Blanket Region

	<u>Local Heat Flux</u>	<u>Rod Power</u>
Experiment/Calc.	1.187	1.140
Fuel Loading	1.019	1.010
Steady-state Overpower	<u>1.071</u>	<u>1.076</u>
Total Factor	1.295*	1.239*

*Time dependent gamma heating factors were applied in addition to these factors. The variation in gamma heating factors for rod Row 1 is 3.80 to 1.01 and for other rod rows is 2.20 to 1.00.

TABLE IV-10. LWBR KINETIC PARAMETERS

<u>Reactivity Coefficients</u>	<u>BOL</u>	<u>EOL</u>
At-power Doppler coefficient at operating conditions, $10^{-4} \Delta\rho/^{\circ}\text{F}$	-0.62	-0.60
At-power moderator temperature coefficient, fast transients, $10^{-4} \Delta\rho/^{\circ}\text{F}$	-1.3	-1.7
At-power moderator temperature coefficient, slow transient, $10^{-4} \Delta\rho/^{\circ}\text{F}$	-1.6	-2.1
Uniform temperature coefficient, ambient temperature, $10^{-4} \Delta\rho/^{\circ}\text{F}$	-0.66	
Uniform temperature coefficients, hot, zero power, $10^{-4} \Delta\rho/^{\circ}\text{F}$	Doppler	-0.37
	Moderator	<u>-1.85</u>
	Total	-2.22
Pressure coefficient at operating conditions, $10^{-6} \Delta\rho/\text{psig}$	+2.44	
Total power coefficient, $10^{-2} \Delta\rho/100\% \Delta P$	-1.8	-1.5
<u>Delayed Neutron Fractions</u>		
68F Delayed Fractions (β)	0.00325	0.00328
Hot Operating Delayed Fractions (β)	0.00328	0.00331
Effectiveness Factor (β/β)	1.02	1.02
<u>Prompt Neutron Lifetime</u>		
Lifetime (μ seconds)	16.4	12.5

BOL = Beginning-of-Life

EOL = End-of-Life

TABLE IV-11. REACTIVITY CONTROL AT BEGINNING-OF-LIFE (BOL)
AND END-OF-LIFE (EOL)

	BOL <u>Δk</u>	EOL <u>Δk</u>
100% power-to-zero power	0.018	0.015
Xenon worth	0.0	0.017
Protactinium worth	0.0	0.023
Hot zero power to cold	0.043	0.050
Shutdown margin	<u>0.145</u>	<u>0.050</u>
Reactivity worth of inserted amount of movable assemblies at the full-power critical position	0.206	0.155

Figure IV-1. Movable Fuel Position Over Lifetime As-Built LWBR Core

Figure IV-2. Movable Fuel Worth as a Function of Critical Position for a Nominal Core, Hot Full Power

Figure IV-3. Critical Fuel Position for Various Core Conditions in a Nominal LWBR Core

Figure IV-4. Natural ^{233}U , ^{232}Th Chain

Th^{232} Chain for Decay to U^{233}

Th^{232} Chain for Capture to U^{234}

Th^{232} Yielding $\text{Pa}^{231} + \text{U}^{232}$

Figure IV-5. Linear Fuel Chains

Figure IV-6. LWBR Core Breeding for Continuous 60 MWe(net) Operation

Figure IV-7. Neutron Absorption in Axial H_2O and Structure

Figure IV-8. Neutron Absorption External to the Twelve Central Modules in the LWBR Core

Figure IV-9. Region Fissile Loadings

Figure IV-10. Axial U-233 Distribution in 84-Inch Binary Rods Near End-of-Life

Figure IV-11. Axial U-233 Distribution in a 42-Inch Binary Seed Rod Near End-of-Life, Average Rod

Figure IV-12. Axial U-233 Distribution in a 42-Inch Binary Standard Blanket Rod Near End-of-Life, Average Rod

Figure IV-13. Axial U-233 Distribution in a Reflector Blanket
Rod Near End-of-Life, Rod with Largest U-233 Buildup

PFB = Power flattening blanket
 E.B. = ThO_2 end blanket

Figure IV-14. Axial Power Distribution in LWBR - Percent of Power, Nominal Full Power, Equilibrium Xenon and Protactinium

FIGURE IV -15
AXIAL POWER DISTRIBUTION IN LWBR - PERCENT OF POWER
FULL POWER, WITH AND WITHOUT P_0 - 233

+1%, OFF-NCMINA_CORE

-1% OFF-NOMINAL CORE

18000 HOURS, WITH Pd-233

10500 HOURS, WITH 70-233

10500 HOURS, NO PG - 233

SEED

18000 HOURS, NO Pd - 233

<u>ZONE</u>	<u>SEED</u>	<u>BLKT</u>	<u>PFB</u>	<u>SEED</u>	<u>BLKT</u>	<u>PFB</u>	<u>SEED</u>	<u>BLKT</u>	<u>PFB</u>	<u>SEED</u>	<u>BLKT</u>	<u>PFB</u>
1	7.301	4.670	4.113	9.273	3.334	2.929	1.609	6.414	5.562	3.833	5.683	4.748
2	11.888	11.241	9.597	13.130	12.115	9.949	7.876	11.893	10.039	9.200	10.752	8.712
3	13.610	10.793	9.457	11.454	11.999	10.036	14.599	11.229	9.263	13.932	11.327	9.133
4	4.846	5.378	5.023	2.354	5.818	5.230	13.332	2.986	2.567	10.257	5.286	4.391
E.B.	0.213	0.115	0.045	0.319	0.129	0.049	0.068	0.055	0.026	0.085	0.100	0.042
TOTAL	37.859	32.197	28.245	36.530	33.394	28.194	37.484	32.577	27.457	37.307	33.148	27.026

PFB = Power flattening Blanket
E.B. = ThO₂ end blanket

Figure IV-15. Axial Power Distribution in LWBR - Percent of Power, Full Power, with and without Pa-233

Figure IV-16. Symmetric Sixth - Core Slice Through 12 Modules

Figure IV-17. Normalized Power Densities Near Beginning-of-Life, Hot Full Power, Equilibrium Xenon

Figure IV-18. Normalized Power Densities Near Middle-of-Life, Hot Full Power, Equilibrium Xenon

Figure IV-19. Normalized Power Densities Near End-of-Life, Hot Full Power, Equilibrium Xenon

Figure IV-20. Axial Power Shapes in "Limiting" High Zone Seed Rod
Rod Location No. 1 in Figure IV-16.

Figure IV-21. Axial Power Shapes in "Limiting" Low Zone Seed Rod
Rod Location No. 2 in Figure IV-16

Figure IV-22. Axial Power Shapes in a "Limiting" Standard Blanket Rod
Rod Location No. 3 in Figure IV-16

Figure IV-23. Axial Power Shapes in a "Limiting" Power Flattening Blanket Rod
Rod Location No. 4 in Figure IV-15

Figure IV-24. Axial Power Shapes in a "Typical" High Zone Seed Rod
Rod Location No. 5 in Figure IV-16

Figure IV-25. Axial Power Shapes in a "Typical" Standard Blanket Rod
Rod Location No. 6 in Figure IV-16

Figure IV-26. Radial Power Shapes Near Top of Type I Module Seed Assembly

Figure IV-27. Radial Power Shapes at Midplane of Type I Module Seed Assembly

Figure IV-28. Radial Power Shapes at Midplane of Type I Module Blanket Assembly

Figure IV-29. Seed Local Heat Flux as a Function of Fission Density

Figure IV-30. Blanket Local Heat Flux as a Function of Fission Density

Figure IV-31. Reactivity Addition to Compensate Negative Power Coefficient of Reactivity

Figure IV-32. Excess Reactivity and Shutdown Margin

Figure IV-33. Fuel Worths for Nominal Core Versus Lifetime

Figure IV-34. Reactivity With a Stuck Movable Assembly

MOVABLE ASSEMBLY OF MOST REACTIVE MODULE AT
THE FULL POWER CRITICAL POSITION
OTHER II ASSEMBLIES AT 0 FEET
NOMINAL CORE

Figure IV-35. Critical Temperature with a Stuck Assembly

Figure IV-36. Reactivity With a Movable Assembly at the Maximum Position

V. CONCLUSIONS

Performance predictions for the Light Water Breeder Reactor (LWBR) are presented in Section IV. The item of primary importance is the breeding performance (peak FIR) which is estimated to be in the range of 1.013 to 1.016, depending on the power operation history. This increase of about 1.5 percent in fissile loading, relative to the initial loading, is projected to occur following about 19,000 equivalent full power hours of operation, where full power of 60 MW(e) net is obtained from 236.6 thermal megawatts. The reactivity lifetime, which will be reached when the top of the movable seeds are 24 inches above the top of the blankets, is estimated to be 21,800 equivalent full power hours for the nominal LWBR core.

The negative fuel temperature (Doppler) coefficient of reactivity is large for LWBR due to the Doppler broadening of the Th-232 resonance absorption peaks. The uniform Doppler coefficient of $-0.37 \times 10^{-4} \Delta\rho/^\circ\text{F}$, calculated at the hot moderator (zero power) temperature, increases in magnitude to about $-0.6 \times 10^{-4} \Delta\rho/^\circ\text{F}$ for fuel temperatures at full-power operating conditions. Thus, the LWBR Doppler coefficient is more than twice as large as in a low-enrichment uranium fueled plant. The LWBR calculated moderator temperature coefficient of reactivity is in the range -1.6 to $-2.1 \times 10^{-4} \Delta\rho/^\circ\text{F}$. The effective delayed neutron fraction for LWBR is small (0.0033) relative to a U-235 fueled core (≈ 0.0075 initially to 0.0045 for depleted fuel) due to the U-233 yield being less than half that of U-235. The reactivity feedback effects, which tend to compensate for a rapid increase in reactivity, have been used to determine core response during abnormal or accidental transients. Details of safe operation of the core under these conditions is discussed in Chapter 15 of Reference V-1. The LWBR core has a large margin of stability against xenon-induced power oscillations and a small (0.2-percent) peak xenon reactivity transient.

Calculations for the movable fuel assembly reactivity control system show that adequate reactivity control and shutdown margin is provided for all anticipated operational requirements of the reactor. Late in core life, in the cold condition, the core nominally exhibits approximately 5-percent reactivity shutdown margin with all modules in the fully-inserted position and with all xenon and Pa-233 decayed. The shutdown available for earlier times in core life is

greater (~15 percent initially) than at end-of-life. Shutdown margin is also available for some period of time under abnormal core conditions, and a system has been provided to insert concentrated soluble boron to ensure complete shutdown under abnormal core conditions.

Confidence in the calculational methods, used to arrive at the above conclusions, is provided by the many comparisons of calculations to experiments and to calculational standards, as discussed in Section III.

REFERENCE

V-1. Safety Analysis Report for the Light Water Breeder Reactor, 1975.

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APPENDIX A

PHYSICAL DIMENSIONS OF FUEL RODS, GRIDS AND MODULES IN THE LWBR CORE

As-built data for the fuel rods, support grids, and modules in the LWBR core are summarized in Tables A-1 through A-17. Excluded from the tables are data for spare components and assemblies. The data summarized were collected (Reference A-1) during the course of rod, grid, and module fabrication and filed in the computer. The computer files contain fuel stack lengths and weights, binary pellet blend identities, clad outer diameter and thickness, and module cell position for each fuel rod in the core. Binary pellet blend data were filed as blends were being processed through the Delayed Neutron Gage (Reference A-2) to determine uranium content. Also included were the results of the uranium isotopic inspections for each blend. Inspection data on pellet diameter and density were also available for ThO_2 pellet blends.

The rod and blend data were used to compute the uranium and thorium content of fuel rods. The fuel impurity data from the mass spectroscopic evaluations on approximately every tenth pellet blend were filed in the computer and used to compute the Neutron Poison Equivalence (NPE) of the fuel. The data filed for grids included grid weight, weight of bolts and nuts supporting the grids, mean grid cell pitch and module location. The parameters mentioned above, all of which influence breeding and reactivity, are summarized in the accompanying tables.

Tables A-1 through A-8 summarize important fuel rod properties by rod type. In each of the eight tables the mean, standard deviation, minimum and maximum of the as-built rod properties are recorded for each fuel rod type where the types (see Figures II-4 and II-5) are defined as follows:

- low seed - 01, 02, 03, 04, 07 and 08
- high seed - 05 and 06
- low standard blanket - 11 and 12
- medium standard blanket - 14 and 15
- high standard blanket - 13 and 16
- low power-flattening blanket - 21 and 22
- medium power-flattening blanket - 24 and 27

high power-flattening blanket - 23, 25 and 26
reflector blanket - 31 and 32.

The fuel rod properties include:

1. binary stack length
2. position of binary/thoria interface which is the combined length of binary and bottom thoria stacks in seed rods and the length of the bottom thoria stack in blanket rods
3. overall fuel length
4. fissile weight
5. fissile linear density (grams/inch)
6. thorium weight
7. clad outer diameter
8. clad thickness.

Of the eight properties tabulated, six have specification requirements; the two which do not are fissile weight (controlled by fissile linear density and pellet density specifications) and thorium weight (controlled by pellet density specifications).

Table A-9 presents histograms for each of the eight binary composition rod types of fissile linear density (grams per inch) and fissile loading precision. The fissile loading precision is equal to twice the calculated standard deviation (see Appendix B in Reference A-1) of the rod fissile loading uncertainty in relative percent. The data in Table A-9 demonstrate that each binary fuel rod met the requirements on fissile linear density, and the requirement that the computed rod fissile loading precision be no greater than 0.25 percent at the 95-percent confidence level.

Average fuel pellet densities and void fractions (fraction of "right circular cylinder" pellet missing due to end-dishes, champhers, and pellet chips) for the 12 fuel compositions are presented in Table A-10. The "% T.D." is the average value for sampled pellets in blends used in the core, where 20 pellets were sampled per binary blend and 15 pellets were sampled per thoria blend. The sampled pellets were weighed and measured and a geometric density (gm/cm^3) was computed for each pellet, assuming a nominal pellet void fraction for pellet and dishes and tapers. The geometric density was divided by the theoretical density for the fuel type and multiplied by 100 to produce percent T.D. The percent T.D.

thus calculated provides the measure of as-built fuel density or porosity but does not precisely account for as-built fuel pellet stack voids. Since the binary and thorium fuel stacks of each fuel rod were weighed, and since the stack weights reflect the as-built conditions of each pellet (dimensions, chips, and density), the recorded stack weights and lengths were used along with the average pellet diameter and density to compute an effective pellet void fraction for each of the 12 compositions.

Table A-11 lists the average uranium isotopic weight percents for the eight binary fuel compositions. The as-built average U-238 content for the entire core (total uranium loadings are presented in Table A-14) is 0.37 percent. Some fuel which had a high U-238 content was used in the high zone power-flattening blanket resulting in an average U-238 content of 0.59 percent for this composition compared with an average of 0.27 percent for the other binary compositions.

Table A-12 lists average as-built fuel rod properties for the total complement of each rod type, that is, seed, standard blanket, power-flattening blanket, and reflector blanket. Included in Table A-12 are binary and thorium pellet outer diameter and percent theoretical density, and clad outer diameter and thickness. The nominal design specifications are given parenthetically.

Table A-13 summarizes the mean grid cell pitch values and the mean grid weights for the six types of grids in the core. The tabulated weights include the weights of bolts and nuts which support the grids as well as the weights of the assembled grids. A fraction of the as-built grid material is braze consisting of nickel (60 percent), chromium (30 percent), and silicon (10 percent). Evaluations of as-built grids have indicated that the seed grids contain 6.08-percent braze by weight and that the blanket grids contain 6.26-percent braze.

Table A-14 summarizes the as-built uranium and thorium loadings, the mean clad dimensions and the mean grid pitch and weight for each seed and blanket module. These data demonstrate the relative uniformity of the modules. Table A-15 summarizes the thorium loading, mean clad dimensions, and mean grid pitch and weight for the reflector modules. From Tables A-14 and A-15, we see that the LWBR core contains 509.6 kilograms of uranium, of which 501.0 kilograms are fissile ($^{233}\text{U} + ^{235}\text{U}$), and 42,056.1 kilograms of thorium.

Table A-16 lists the fissile loading and relative precision (previously defined above) for each of the eight binary compositions. These data demonstrate that the calculated precision for each composition meets the requirement that the loading precision be no larger than 0.1 percent for any composition.

Table A-17 lists the results of the mass spectroscopic analyses performed on approximately every tenth blend to determine impurity levels in the fuel pellets. The mass spectroscopic data are used to obtain the best estimate neutron poison equivalent (NPE) level (defined in Appendix B) for breeding calculations rather than the chemistry data taken on every blend. The mass spectroscopic data indicate impurity levels below the chemistry limits of detectability for many elements. While the mass spectroscopic impurity evaluations were done on only a tenth of the fuel blends, the NPE values determined from the mass spectroscopic data are judged to provide the best overall estimate of NPE.

Tables A-1 through A-17 provide a summary of the LWBR as-built data. Detailed data from which the tables were constructed were recorded throughout manufacturing stages and filed in the computer. Detailed data edits for each fuel rod, grid, and binary fuel blend can be produced by the computer programs (Reference A-1) which generated the summary tables; however, such voluminous edits are beyond the scope of this report.

REFERENCES

- A-1. B. R. Beaudoin, et al, "ASBLT - A System of Datatran Modules which Process Core Fuel Loading for Use in Calculations," WAPD-TM-1316, February 1979.
- A-2. S. Milani and C. J. Emert, "The Delayed Neutron Gage," WAPD-TM-1088, November 1974.

TABLE A-1. BINARY STACK LENGTH (INCHES)

Type	Rods	Mean	Std Dev	Min	Max
01	360	42.052	0.168	41.739	42.247
02	1008	41.946	0.138	41.769	42.245
03	864	55.973	0.105	55.776	56.247
04	792	70.018	0.144	69.771	70.243
05	2172	84.023	0.199	83.548	84.333
06	1800	84.049	0.197	83.541	84.369
07	360	41.980	0.159	41.719	42.242
08	72	42.031	0.180	41.784	42.238
11	471	41.968	0.032	41.858	42.062
12	501	41.965	0.029	41.871	42.041
13	558	69.979	0.052	69.812	70.122
14	531	55.966	0.404	55.518	56.462
15	582	84.227	0.069	83.956	84.395
16	591	84.120	0.064	83.951	84.305
21	249	41.787	0.058	41.603	41.927
22	291	41.779	0.058	41.593	41.958
23	375	70.118	0.114	69.661	70.334
24	333	55.826	0.042	55.737	55.962
25	1100	84.174	0.127	83.659	84.401
26	1161	84.176	0.134	83.658	84.371
27	72	84.087	0.080	83.903	84.261

TABLE A-2. BINARY/THORIA INTERFACE (INCHES)

Type	Rods	Mean	Std Dev	Min	Max
01	360	51.793	0.010	51.694	51.810
02	1008	50.933	0.010	50.817	50.950
03	864	65.791	0.014	65.678	65.900
04	792	78.932	0.009	78.910	78.950
05	2172	93.794	0.013	93.667	93.840
06	1800	92.934	0.011	92.849	92.960
07	360	51.793	0.008	51.770	51.810
08	72	50.931	0.011	50.911	50.950
11	471	53.514	0.009	53.492	53.532
12	501	51.512	0.009	51.489	51.532
13	558	25.507	0.008	25.455	25.526
14	531	37.510	0.008	37.489	37.529
15	582	11.503	0.008	11.484	11.523
16	591	9.503	0.008	9.482	9.522
21	249	53.499	0.009	53.480	53.520
22	291	51.499	0.008	51.482	51.520
23	375	25.499	0.009	25.481	25.520
24	333	37.502	0.009	37.480	37.520
25	1100	11.502	0.008	11.458	11.520
26	1161	9.502	0.008	9.480	9.520
27	72	11.502	0.008	11.481	11.520

TABLE A-3. OVERALL FUEL LENGTH (INCHES)

Type	Rods	Mean	Std Dev	Min	Max
01	360	103.845	0.019	103.807	103.920
02	1008	103.844	0.020	103.717	103.927
03	864	103.844	0.025	103.743	103.945
04	792	103.841	0.018	103.744	103.924
05	2172	103.863	0.025	103.737	103.936
06	1800	103.839	0.013	103.718	103.911
07	360	103.849	0.024	103.801	103.962
08	72	103.840	0.016	103.749	103.870
11	471	103.801	0.009	103.717	103.825
12	501	104.852	0.008	104.829	104.878
13	558	103.805	0.009	103.696	103.841
14	531	104.849	0.010	104.785	104.874
15	582	103.803	0.010	103.711	103.826
16	591	104.852	0.009	104.823	104.913
21	249	103.811	0.008	103.787	103.828
22	291	104.856	0.008	104.836	104.878
23	375	103.811	0.009	103.786	103.834
24	333	104.858	0.008	104.838	104.883
25	1100	103.808	0.008	103.779	103.835
26	1161	104.859	0.009	104.826	104.881
27	72	103.809	0.007	103.796	103.827
31	1526	101.062	0.011	100.972	101.085
32	1521	103.061	0.009	102.978	103.082

TABLE A-4. FISSILE WEIGHT (GRAMS)

Type	Rods	Mean	Std Dev	Min	Max
01	360	14.3678	0.0915	14.2029	14.5161
02	1008	14.3233	0.0823	14.1261	14.5194
03	864	19.1387	0.0632	18.8946	19.3064
04	792	23.9207	0.1087	23.6047	24.1894
05	2172	34.5915	0.1329	34.1602	35.0367
06	1800	34.5434	0.1687	34.1136	34.9492
07	360	14.3170	0.0812	14.1045	14.4785
08	72	14.3607	0.1077	14.1172	14.5217
11	471	16.4502	0.0351	16.3611	16.5213
12	501	16.4555	0.0289	16.3742	16.5424
13	558	45.4714	0.0759	45.2081	45.6801
14	531	30.3369	0.2433	29.9553	30.8527
15	582	45.6599	0.0944	45.4096	45.9900
16	591	54.6562	0.1141	54.3986	55.0453
21	249	18.9572	0.0506	18.7008	19.0714
22	291	18.9538	0.0512	18.8494	19.0814
23	375	52.5558	0.1140	52.1338	52.8363
24	333	30.7406	0.0567	30.5963	30.8610
25	1100	63.0616	0.1503	62.3354	63.4513
26	1161	63.0587	0.1708	62.2453	63.4485
27	72	46.4020	0.1221	46.1643	46.5843

TABLE A-5. FISSILE LINEAR DENSITY (GRAMS/INCH)

Type	Rods	Mean	Std Dev	Min	Max
01	360	0.34167	0.00116	0.33885	0.34409
02	1008	0.34147	0.00137	0.33717	0.34517
03	864	0.34193	0.00114	0.33836	0.34414
04	792	0.34163	0.00127	0.33754	0.34438
05	2172	0.41169	0.00139	0.40637	0.41588
06	1800	0.41099	0.00165	0.40630	0.41518
07	360	0.34104	0.00096	0.33675	0.34307
08	72	0.34167	0.00174	0.33680	0.34399
11	471	0.39197	0.00076	0.38976	0.39367
12	501	0.39212	0.00063	0.39052	0.39397
13	558	0.64979	0.00105	0.64550	0.65241
14	531	0.54206	0.00146	0.53810	0.54725
15	582	0.54211	0.00114	0.53911	0.54586
16	591	0.64974	0.00125	0.64621	0.65302
21	249	0.45366	0.00095	0.44848	0.45591
22	291	0.45366	0.00117	0.45130	0.45591
23	375	0.74953	0.00132	0.74430	0.75287
24	333	0.55065	0.00112	0.54815	0.55285
25	1100	0.74918	0.00155	0.74410	0.75464
26	1161	0.74913	0.00164	0.74380	0.75361
27	72	0.55184	0.00149	0.54913	0.55394

TABLE A-6. THORIUM WEIGHT (GRAMS)

Type	Rods	Mean	Std Dev	Min	Max
01	360	719.3	1.4	714.4	722.0
02	1008	718.4	1.7	713.0	721.8
03	864	711.1	1.3	706.9	714.0
04	792	702.4	2.0	696.7	707.1
05	2172	690.7	1.5	684.2	695.6
06	1800	690.2	1.5	684.4	694.0
07	360	717.7	1.6	713.1	721.3
08	72	719.4	1.7	714.1	721.5
11	471	2935.3	4.6	2923.7	2947.9
12	501	2965.2	4.6	2953.0	2977.2
13	558	2907.6	3.8	2894.7	2916.8
14	531	2957.1	3.5	2943.5	2972.5
15	582	2913.2	4.0	2902.7	2923.5
16	591	2928.7	3.7	2915.3	2937.9
21	249	2473.9	5.4	2447.0	2483.6
22	291	2499.1	5.7	2471.5	2508.7
23	375	2438.8	3.5	2427.1	2445.3
24	333	2491.9	4.1	2468.3	2498.5
25	1100	2429.8	4.1	2414.9	2445.5
26	1161	2455.8	3.9	2442.9	2465.4
27	72	2453.5	3.8	2445.9	2460.3
31	1526	6034.7	18.8	5975.3	6096.8
32	1521	6157.3	16.3	6093.7	6216.2

TABLE A-7. ROD OUTER DIAMETER (INCHES)

Type	Rods	Mean	Std Dev	Min	Max
01	360	0.30624	0.00045	0.30450	0.30770
02	1008	0.30627	0.00042	0.30480	0.30760
03	864	0.30627	0.00042	0.30500	0.30740
04	792	0.30625	0.00048	0.30480	0.30770
05	2172	0.30630	0.00050	0.30450	0.30810
06	1800	0.30617	0.00049	0.30450	0.30790
07	360	0.30622	0.00045	0.30500	0.30740
08	72	0.30628	0.00042	0.30520	0.30750
11	471	0.57168	0.00047	0.57030	0.57300
12	501	0.57160	0.00055	0.57010	0.57340
13	558	0.57152	0.00048	0.57000	0.57300
14	531	0.57181	0.00047	0.57000	0.57360
15	582	0.57183	0.00053	0.57020	0.57360
16	591	0.57164	0.00054	0.57040	0.57310
21	249	0.52738	0.00038	0.52630	0.52860
22	291	0.52741	0.00036	0.52630	0.52830
23	375	0.52743	0.00042	0.52610	0.52860
24	333	0.52742	0.00041	0.52640	0.52860
25	1100	0.52749	0.00047	0.52570	0.52900
26	1161	0.52742	0.00047	0.52620	0.52910
27	72	0.52741	0.00040	0.52620	0.52820
31	1526	0.83231	0.00055	0.82960	0.83390
32	1521	0.83235	0.00049	0.83030	0.83380

TABLE A-8. CLAD THICKNESS (INCHES)

<u>Type</u>	<u>Rods</u>	<u>Mean</u>	<u>Std Dev</u>	<u>Min</u>	<u>Max</u>
01	360	0.02217	0.00029	0.02130	0.02300
02	1008	0.02216	0.00031	0.02110	0.02300
03	864	0.02221	0.00029	0.02130	0.02390
04	792	0.02220	0.00030	0.02140	0.02300
05	2172	0.02217	0.00032	0.02080	0.02330
06	1800	0.02216	0.00032	0.02120	0.02340
07	360	0.02213	0.00031	0.02130	0.02290
08	72	0.02216	0.00032	0.02140	0.02290
11	471	0.02797	0.00037	0.02700	0.02940
12	501	0.02799	0.00038	0.02700	0.02920
13	558	0.02811	0.00028	0.02740	0.02910
14	531	0.02812	0.00032	0.02710	0.02920
15	582	0.02809	0.00036	0.02720	0.02920
16	591	0.02820	0.00034	0.02720	0.02910
21	249	0.02661	0.00030	0.02560	0.02740
22	291	0.02650	0.00033	0.02560	0.02730
23	375	0.02642	0.00029	0.02550	0.02720
24	333	0.02635	0.00026	0.02560	0.02710
25	1100	0.02643	0.00028	0.02540	0.02730
26	1161	0.02636	0.00028	0.02540	0.02770
27	72	0.02633	0.00028	0.02580	0.02690
31	1526	0.04189	0.00033	0.04030	0.04270
32	1521	0.04187	0.00032	0.04050	0.04300

TABLE A-9. FUEL ROD FISSILE LOADING AND PRECISION HISTOGRAMS

ZONE 1 (LOW SEED)

SPEC RANGE FOR FISSILE GM/IN IS 0.3298 TO 0.3460 (NOM= 0.3379, TOL= 0.0081)

GRAMS FISSILE PER INCH HISTOGRAM				ROD LOADING PRECISION HISTOGRAM			
1	0.33600 -- 0.33699	4		1	0.130 -- 0.139	1	
2	0.33700 -- 0.33799	29		2	0.140 -- 0.149	0	
3	0.33800 -- 0.33899	84		3	0.150 -- 0.159	43	
4	0.33900 -- 0.33999	224		4	0.160 -- 0.169	21	
5	0.34000 -- 0.34099	701		5	0.170 -- 0.179	14	
6	0.34100 -- 0.34199	965		6	0.180 -- 0.189	2	
7	0.34200 -- 0.34299	1015		7	0.190 -- 0.199	192	
8	0.34300 -- 0.34399	402		8	0.200 -- 0.209	595	
9	0.34400 -- 0.34499	28		9	0.210 -- 0.219	1219	
10	0.34500 -- 0.34599	4		10	0.220 -- 0.229	485	
11	0.34600 -- 0.34699	0		11	0.230 -- 0.239	609	
12	0.34700 -- 0.34799	0		12	0.240 -- 0.249	275	
		3456				3456	

Mean = 0.34160 Standard Deviation = 0.00127

ZONE 2 (HIGH SEED)

SPEC RANGE FOR FISSILE GM/IN IS 0.3980 TO 0.4176 (NOM= 0.4078, TOL= 0.0098)

GRAMS FISSILE PER INCH HISTOGRAM				ROD LOADING PRECISION HISTOGRAM			
1	0.40600 -- 0.40699	38		1	0.120 -- 0.129	1	
2	0.40700 -- 0.40799	52		2	0.130 -- 0.139	0	
3	0.40800 -- 0.40899	131		3	0.140 -- 0.149	25	
4	0.40900 -- 0.40999	510		4	0.150 -- 0.159	49	
5	0.41000 -- 0.41099	846		5	0.160 -- 0.169	64	
6	0.41100 -- 0.41199	915		6	0.170 -- 0.179	26	
7	0.41200 -- 0.41299	897		7	0.180 -- 0.189	225	
8	0.41300 -- 0.41399	462		8	0.190 -- 0.199	827	
9	0.41400 -- 0.41499	86		9	0.200 -- 0.209	1245	
10	0.41500 -- 0.41599	35		10	0.210 -- 0.219	687	
11	0.41600 -- 0.41699	0		11	0.220 -- 0.229	384	
12	0.41700 -- 0.41799	0		12	0.230 -- 0.239	375	
13	0.41800 -- 0.41899	0		13	0.240 -- 0.249	64	
		3972				3972	

Mean= 0.41137 Standard Deviation = 0.00155

TABLE A-9. FUEL ROD FISSILE LOADING AND PRECISION HISTOGRAMS (Cont'd)

ZONE 3 (LOW STANDARD BLANKET)

SPEC RANGE FOR FISSILE GM/IN IS 0.3728 TO 0.3958 (NOM= 0.3843, TOL= 0.0115)

GRAMS FISSILE PER INCH HISTOGRAM				ROD LOADING PRECISION HISTOGRAM			
1	0.38900 -- 0.38999	1		1	0.120 -- 0.129	1	
2	0.39000 -- 0.39099	85		2	0.130 -- 0.139	12	
3	0.39100 -- 0.39199	359		3	0.140 -- 0.149	18	
4	0.39200 -- 0.39299	450		4	0.150 -- 0.159	11	
5	0.39300 -- 0.39399	77		5	0.160 -- 0.169	40	
6	0.39400 -- 0.39499	0		6	0.170 -- 0.179	168	
7	0.39500 -- 0.39599	0		7	0.180 -- 0.189	213	
8	0.39600 -- 0.39699	0		8	0.190 -- 0.199	260	
9	0.39700 -- 0.39799	0		9	0.200 -- 0.209	137	
10	0.39800 -- 0.39899	0		10	0.210 -- 0.219	52	
11	0.39900 -- 0.39999	0		11	0.220 -- 0.229	31	
12	0.40000 -- 0.40099	0		12	0.230 -- 0.239	29	
		972				972	

Mean= 0.39205 Standard Deviation= 0.00070

ZONE 4 (MEDIUM STANDARD BLANKET)

SPEC RANGE FOR FISSILE GM/IN IS 0.5165 TO 0.5485 (NOM= 0.5325, TOL= 0.160)

GRAMS FISSILE PER INCH HISTOGRAM				ROD LOADING PRECISION HISTOGRAM			
1	0.53800 -- 0.53899	1		1	0.110 -- 0.119	12	
2	0.53900 -- 0.53999	46		2	0.120 -- 0.129	25	
3	0.54000 -- 0.54099	114		3	0.130 -- 0.139	24	
4	0.54100 -- 0.54199	435		4	0.140 -- 0.149	28	
5	0.54200 -- 0.54299	262		5	0.150 -- 0.159	130	
6	0.54300 -- 0.54399	162		6	0.160 -- 0.169	167	
7	0.54400 -- 0.54499	63		7	0.170 -- 0.179	134	
8	0.54500 -- 0.54599	22		8	0.180 -- 0.189	197	
9	0.54600 -- 0.54699	6		9	0.190 -- 0.199	202	
10	0.54700 -- 0.54799	2		10	0.200 -- 0.209	99	
11	0.54800 -- 0.54899	0		11	0.210 -- 0.219	49	
12	0.54900 -- 0.54999	0		12	0.220 -- 0.229	36	
13	0.55000 -- 0.55099	0		13	0.230 -- 0.239	10	
		1113				1113	

Mean= 0.54208 Standard Deviation= 0.00130

TABLE A-9. FUEL ROD FISSILE LOADING AND PRECISION HISTOGRAMS (CONT.)

ZONE 5 (HIGH STANDARD BLANKET)

SPEC RANGE FOR FISSILE GM/IN IS 0.6232 TO 0.6618 (NOM= 0.6425, TOL= 0.0193)

GRAMS FISSILE PER INCH HISTOGRAM			ROD LOADING PRECISION HISTOGRAM		
1	0.64500 -- 0.64599	1	1	0.110 -- 0.119	11
2	0.64600 -- 0.64699	15	2	0.120 -- 0.129	32
3	0.64700 -- 0.64799	30	3	0.130 -- 0.139	29
4	0.64800 -- 0.64899	241	4	0.140 -- 0.149	46
5	0.64900 -- 0.64999	417	5	0.150 -- 0.159	207
6	0.65000 -- 0.65099	273	6	0.160 -- 0.169	465
7	0.65100 -- 0.65199	116	7	0.170 -- 0.179	125
8	0.65200 -- 0.65299	55	8	0.180 -- 0.189	61
9	0.65300 -- 0.65399	1	9	0.190 -- 0.199	112
10	0.65400 -- 0.65499	0	10	0.200 -- 0.209	42
11	0.65500 -- 0.65599	0	11	0.210 -- 0.219	10
12	0.65600 -- 0.65699	0	12	0.220 -- 0.229	0
13	0.65700 -- 0.65799	0	13	0.230 -- 0.239	0
14	0.65800 -- 0.65899	0	14	0.240 -- 0.249	1
		1149			1149

Mean= 0.64977 Standard Deviation= 0.00116

ZONE 6 (LOW POWER FLATTENING BLANKET)

SPEC RANGE FOR FISSILE GM/IN IS 0.4329 TO 0.4597 (NOM= 0.4463, TOL= 0.0134)

GRAMS FISSILE PER INCH HISTOGRAM			ROD LOADING PRECISION HISTOGRAM		
1	0.44800 -- 0.44899	1	1	0.120 -- 0.129	6
2	0.44900 -- 0.44999	0	2	0.130 -- 0.139	5
3	0.45000 -- 0.45099	1	3	0.140 -- 0.149	4
4	0.45100 -- 0.45199	53	4	0.150 -- 0.159	32
5	0.45200 -- 0.45299	60	5	0.160 -- 0.169	71
6	0.45300 -- 0.45399	229	6	0.170 -- 0.179	73
7	0.45400 -- 0.45499	133	7	0.180 -- 0.189	128
8	0.45500 -- 0.45599	63	8	0.190 -- 0.199	90
9	0.45600 -- 0.45699	0	9	0.200 -- 0.209	0
10	0.45700 -- 0.45799	0	10	0.210 -- 0.219	28
11	0.45800 -- 0.45899	0	11	0.220 -- 0.229	103
		540			540

Mean= 0.45366 Standard Deviation= 0.00107

TABLE A-9. FUEL ROD FISSILE LOADING AND PRECISION HISTOGRAMS (Cont'd)

ZONE 7 (MEDIUM POWER FLATTENING BLANKET)

SPEC RANGE FOR FISSILE GM/IN IS 0.5255 TO 0.5581 (NOM= 0.5418, TOL= 0.0163)

GRAMS FISSILE PER INCH HISTOGRAM				ROD LOADING PRECISION HISTOGRAM			
1	0.54800	-- 0.54899	36	1	0.130	-- 0.139	1
2	0.54900	-- 0.54999	79	2	0.140	-- 0.149	2
3	0.55000	-- 0.55099	96	3	0.150	-- 0.159	5
4	0.55100	-- 0.55199	132	4	0.160	-- 0.169	5
5	0.55200	-- 0.55299	34	5	0.170	-- 0.179	44
6	0.55300	-- 0.55399	28	6	0.180	-- 0.189	47
7	0.55400	-- 0.55499	0	7	0.190	-- 0.199	85
8	0.55500	-- 0.55599	0	8	0.200	-- 0.209	103
9	0.55600	-- 0.55699	0	9	0.210	-- 0.219	113
			405				405

Mean= 0.55086 Standard Deviation= 0.00128

ZONE 8 (HIGH POWER FLATTENING BLANKET)

SPEC RANGE FOR FISSILE GM/IN IS 0.7200 TO 0.7646 (NOM= 0.7423, TOL= 0.0223)

GRAMS FISSILE PER INCH HISTOGRAM				ROD LOADING PRECISION HISTOGRAM			
1	0.74300	-- 0.74399	2	1	0.130	-- 0.139	4
2	0.74400	-- 0.74499	14	2	0.140	-- 0.149	24
3	0.74500	-- 0.74599	34	3	0.150	-- 0.159	50
4	0.74600	-- 0.74699	162	4	0.160	-- 0.169	53
5	0.74700	-- 0.74799	382	5	0.170	-- 0.179	24
6	0.74800	-- 0.74899	543	6	0.180	-- 0.189	142
7	0.74900	-- 0.74999	686	7	0.190	-- 0.199	448
8	0.75000	-- 0.75099	530	8	0.200	-- 0.209	976
9	0.75100	-- 0.75199	171	9	0.210	-- 0.219	418
10	0.75200	-- 0.75299	76	10	0.220	-- 0.229	336
11	0.75300	-- 0.75399	35	11	0.230	-- 0.239	107
12	0.75400	-- 0.75499	1	12	0.240	-- 0.249	54
			2636				2636

Mean= 0.74921 Standard Deviation= 0.00157

TABLE A-10. AVERAGE AS-BUILT PELLET DENSITY AND VOID FRACTION

	<u>Percent Theoretical Density</u>	<u>Theoretical Density (gm/cm³)</u>	<u>Void Fraction</u>
<u>Seed</u>			
Thoria	98.013	9.999	0.01253
Low zoned	97.712	10.035	0.01704
High zoned	97.554	10.042	0.01172
<u>Standard Blanket</u>			
Thoria	97.796	9.999	0.01399
Low zoned	98.608	10.009	0.02494
Medium zoned	98.224	10.013	0.01335
High zoned	98.115	10.016	0.01600
<u>Power Flattening Blanket</u>			
Thoria	98.057	9.999	0.01966
Low zoned	98.034	10.013	0.01998
Medium zoned	98.041	10.016	0.01753
High zoned	97.906	10.022	0.01578
<u>Reflector Blanket</u>	97.282	9.999	0.01317

TABLE A-11. URANIUM ISOTOPIC WEIGHT PERCENT BY FUEL COMPOSITION

	<u>U-232</u>	<u>U-233</u>	<u>U-234</u>	<u>U-235</u>	<u>U-236</u>	<u>U-238</u>
<u>Seed</u>						
Low zoned	0.00075	98.3088	1.2899	0.07947	0.01775	0.30333
High zoned	0.00070	98.3679	1.2742	0.06711	0.01397	0.27610
<u>Standard Blanket</u>						
Low zoned	0.00084	98.3037	1.3289	0.08805	0.02411	0.25442
Medium zoned	0.00082	98.3218	1.3144	0.08078	0.02102	0.26118
High zoned	0.00082	98.2597	1.3504	0.10466	0.03032	0.25412
<u>Power-Flattening Blanket</u>						
Low zoned	0.00082	98.3074	1.3193	0.08665	0.02258	0.26323
Medium zoned	0.00079	98.2260	1.3620	0.11257	0.03247	0.26613
High zoned	0.00070	98.0419	1.2433	0.09690	0.02648	0.59072

TABLE A-12. AVERAGE AS-BUILT PROPERTIES OF LWBR FUEL RODS
(NOMINALS IN PARENTHESES)

	<u>Seed</u>	<u>Std. Blanket</u>	<u>P.F. Blanket</u>	<u>Reflector Blanket</u>
Number of rods	7428	3234	3581	3047
Binary pellet OD	0.2520 (0.2520)	0.5105 (0.5105)	0.4696 (0.4695)	----
Thoria pellet OD	0.2556 (0.2555)	0.5106 (0.5105)	0.4696 (0.4695)	0.7417 (0.7415)
Binary fuel % TD	97.61 (97.00)	98.25 (97.00)	97.93 (97.00)	----
Thoria fuel % TD	98.01 (97.00)	97.80 (97.00)	98.06 (97.00)	97.28 (97.00)
Fuel rod outer diameter	0.30625 (0.3060)	0.57168 (0.5715)	0.52744 (0.5275)	0.83233 (0.8320)
Fuel rod clad thickness	0.02217 (0.0220)	0.02808 (0.02775)	0.02642 (0.02625)	0.04188 (0.0420)

TABLE A-13. AVERAGE AS-BUILT PROPERTIES OF LWBR GRIDS
(NOMINALS IN PARENTHESES)

	<u>No. Grids</u>	<u>Pitch Inches</u>	<u>Weight Grams*</u>
Seed	108	0.36857 (0.3690)	1612.0 (1665.6)
Type I Blanket	24	0.630111 (0.6300)	2302.0 (2216.9)
Type II Blanket	24	0.630119 (0.6300)	2952.6 (2816.2)
Type III Blanket	48	0.63024 (0.6300)	3284.3 (3161.5)
Type IV Reflector	54	0.90059 (0.9020)	2480.0 (2397.6)
Type V Reflector	36	0.90014 (0.9020)	1878.4 (1872.0)

* Includes bolts and nuts which support the grids in the modules.

TABLE A-14. SEED AND BLANKET MODULE AS-BUILT PROPERTIES

Type	Module S/N	Rods	U-232 Grams	U-233 Grams	U-234 Grams	U-235 Grams	U-236 Grams	U-238 Grams	U-Fiss Grams	Thorium KGS	Mean Clad		Mean Grid	
											OD Inches	Wall Inches	Pitch Inches	Wgt Grams
S E D	I-1 L-BB01-04	619	0.1199	16512.0	215.2	13.0	2.7	48.9	16525.0	433.607	0.30626	0.02221	0.36851	1533.
	I-2 L-BB01-05	619	0.1182	16513.9	218.1	15.0	3.2	48.7	16528.8	433.601	0.30620	0.02218	0.36861	1533.
	I-3 L-BB01-06	619	0.1196	16522.5	215.8	12.2	2.7	48.8	16534.7	433.919	0.30630	0.02222	0.36850	1531.
	II-1 L-BB01-09	619	0.1189	16528.6	215.2	11.8	2.6	48.6	16540.4	433.890	0.30636	0.02226	0.36857	1539.
	II-2 L-BB01-10	619	0.1188	16528.7	216.0	12.5	2.7	47.9	16541.2	433.660	0.30636	0.02219	0.36852	1549.
	II-3 L-BB01-13	619	0.1201	16567.1	215.2	11.5	2.5	45.9	16578.6	434.097	0.30626	0.02215	0.36863	1542.
	III-1 L-BB01-07	619	0.1207	16511.4	214.2	12.3	2.5	49.1	16523.6	433.571	0.30616	0.02215	0.36862	1529.
	III-2 L-BB01-08	619	0.1208	16543.6	214.1	11.0	2.3	47.4	16554.6	433.882	0.30621	0.02214	0.36857	1541.
	III-3 L-BB01-12	619	0.1212	16554.8	214.1	10.9	2.3	47.5	16565.7	434.082	0.30621	0.02212	0.36859	1542.
	III-4 L-BB01-11	619	0.1196	16550.2	214.0	10.9	2.3	47.3	16561.1	434.090	0.30627	0.02218	0.36850	1550.
	III-5 L-BB01-14	619	0.1197	16560.2	213.9	10.7	2.2	46.7	16570.9	434.114	0.30620	0.02213	0.36863	1552.
	III-6 L-BB01-16	619	0.1205	16554.4	214.9	11.4	2.5	47.4	16565.8	434.047	0.30621	0.02214	0.36857	1557.
	Seed Totals	7428	1.4380	198447.4	2580.7	143.2	30.5	574.2	198590.4	5206.560				
B L A N K E T	I-1 L-GU52-01	443	0.1353	16171.3	220.1	15.7	4.4	42.1	16187.0	1299.455	0.57171	0.02808	0.63049	2239.
	I-2 L-GU52-02	443	0.1360	16169.1	218.6	14.9	4.1	42.2	16184.0	1299.371	0.57166	0.02806	0.63040	2230.
	I-3 L-GU51-01	443	0.1365	16166.6	217.1	14.0	3.7	42.3	16180.5	1299.300	0.57168	0.02811	0.63044	2228.
	II-1 L-GV51-01	261	0.0784	9328.5	125.5	8.3	2.2	24.6	9336.8	765.649	0.57172	0.02809	0.63053	1314.
	II-2 L-GS22-01	261	0.0778	9326.7	126.9	9.1	2.5	24.4	9335.8	765.910	0.57176	0.02817	0.63044	1313.
	II-3 L-GV52-01	261	0.0785	9331.8	126.2	8.6	2.4	24.3	9340.3	765.716	0.57165	0.02806	0.63051	1310.
	III-1 L-GW51-01	187	0.0554	6622.3	90.1	6.4	1.8	17.2	6628.8	548.505	0.57166	0.02807	0.63029	930.
	III-2 L-GW52-01	187	0.0552	6625.5	90.4	6.6	1.9	17.3	6632.1	548.695	0.57156	0.02805	0.63020	935.
	III-3 L-GW53-01	187	0.0551	6621.1	90.5	6.7	1.9	17.2	6627.8	548.624	0.57172	0.02806	0.63024	933.
	III-4 L-GT22-01	187	0.0550	6626.2	90.2	6.5	1.8	17.3	6632.7	548.692	0.57172	0.02812	0.63021	935.
	III-5 L-GT22-02	187	0.0554	6628.6	90.3	6.5	1.8	17.3	6635.1	548.720	0.57166	0.02811	0.63021	932.
	III-6 L-GT22-03	187	0.0547	6622.1	91.3	7.1	2.1	17.2	6629.2	548.463	0.57163	0.02800	0.63028	930.
	Blkt Totals	3234	0.9733	116239.8	1577.2	110.4	30.6	303.4	116350.1	9487.100				
P E R I O D I C	II-1 L-GV51-01	302	0.1148	15588.5	202.4	16.2	4.7	77.9	15604.7	741.395	0.52755	0.02643	0.63053	1525.
	II-2 L-GS22-01	303	0.1145	15643.5	198.7	14.8	3.9	90.5	15658.3	743.518	0.52745	0.02646	0.63044	1525.
	II-3 L-GS52-01	302	0.1137	15586.2	192.9	13.3	3.4	95.5	15599.5	741.370	0.52742	0.02644	0.63051	1521.
	III-1 L-GW51-01	445	0.1696	23153.1	291.5	21.1	5.7	132.6	23174.2	1092.258	0.52743	0.02640	0.63029	2217.
	III-2 L-GW52-01	445	0.1687	23129.3	306.0	26.3	7.6	117.1	23155.6	1092.029	0.52740	0.02638	0.63020	2230.
	III-3 L-GW53-01	446	0.1688	23209.8	289.3	20.5	5.3	141.3	23230.3	1094.481	0.52744	0.02643	0.63024	2224.
	III-4 L-GT22-01	446	0.1790	23195.6	310.2	27.3	7.9	112.0	23222.9	1094.364	0.52746	0.02641	0.63021	2229.
	III-5 L-GT22-02	446	0.1698	23199.2	303.6	25.2	7.2	120.0	23224.4	1094.561	0.52741	0.02643	0.63021	2222.
	III-6 L-GT22-03	446	0.1693	23208.2	295.4	22.6	6.0	136.1	23230.8	1094.307	0.52744	0.02640	0.63028	2218.
	PERK Totals	3581	1.3522	185913.4	2390.0	187.3	51.7	1023.0	186100.7	8788.283				
	Core Totals	14243	3.7795	506690.6	6547.9	440.9	112.8	1900.6	501041.2	23481.943*				

* Excluding reflector blanket thorium

TABLE A-15. REFLECTOR BLANKET MODULE AS-BUILT PROPERTIES

Module		Rods	Thorium KGS	Mean Clad		Mean Grid	
Type	S/N			OD Inches	Wall Inches	Pitch Inches	Wgt Grams
IV-3	L-RA01-10	228	1389.561	0.83231	0.04185	0.90050	2413.
IV-1	L-RA01-06	228	1390.115	0.83222	0.04182	0.90065	2401
IV-5	L-RA01-07	228	1390.059	0.83235	0.04187	0.90063	2436.
IV-8	L-RA01-08	228	1389.856	0.83250	0.04196	0.90062	2405.
IV-4	L-RA01-09	228	1389.518	0.83231	0.04189	0.90070	2400.
IV-2	L-RA01-02	228	1389.600	0.83248	0.04193	0.90065	2413.
IV-7	L-RA01-05	227	1383.872	0.83240	0.04191	0.90070	2402.
IV-9	L-RA01-03	228	1389.644	0.83237	0.04190	0.90047	2408.
IV-6	L-RA01-04	228	1389.333	0.83223	0.04181	0.90035	2407.
V-5	L-RB01-03	166	1012.642	0.83235	0.04191	0.90010	1810.
V-3	L-RB01-06	166	1011.876	0.83232	0.04189	0.90017	1822.
V-4	L-RB01-08	166	1012.338	0.83225	0.04186	0.90030	1823.
V-2	L-RB01-04	166	1012.393	0.83229	0.04188	0.89988	1810.
V-1	L-RB01-04	166	1012.034	0.83232	0.04186	0.90028	1832.
V-6	L-RB01-05	166	1011.370	0.83222	0.04183	0.90008	1822.
Reflector Totals		3047	18574.211				

TABLE A-16. COMPOSITION FISSILE LOADING AND PRECISION

Composition	Fissile Grams	Precision Relative%
Low seed	61279.4	0.040
High seed	137311.0	0.047
Low standard blanket	15992.2	0.048
Medium standard blanket	42683.0	0.040
High standard blanket	57674.8	0.045
Low power flattening blanket	10235.9	0.051
Medium power flattening blanket	13577.6	0.055
High power flattening blanket	162287.3	0.036
Total	501041.2	

TABLE A-17. NEUTRON POISON EQUIVALENCE (NPE)

<u>Composition</u>	<u>Blends Sampled</u>	<u>Average NPE</u>
Low seed	13	14.3
High seed	23	14.2
Low standard blanket	3	17.1
Medium standard blanket	21	13.2
High standard blanket	9	17.7
Low power flattening blanket	2	15.8
Medium power flattening blanket	2	14.3
High power flattening blanket	5	19.3
All binary	78	14.8
Seed thoria	5	15.2
Standard blanket thoria	18	14.7
Power flattening blanket thoria	4	6.8
Reflector blanket thoria	36	28.7
All thoria	63	22.2
All binary and thoria	141	18.2

APPENDIX B

BREEDING PARAMETERS AND SENSITIVITIES

Parameters of importance to LWBR breeding will be presented in this appendix in three sections. Section I will develop the breeding formulas and the relationship between conversion ratio and fissile inventory ratio. Section II will define the neutron poison equivalence and discuss its use in specifying fuel pellet impurity limits. Section III will present the magnitude of breeding improvement or breeding loss calculated for certain changes in core design parameters or operation specifications.

I. DEFINITIONS AND DERIVATIONS OF LWBR BREEDING PARAMETERS

The LWBR calculational model predicts breeding performance by calculating conversion ratio and fissile inventory ratio. The following discussion presents the definitions of these quantities and the derivation of equations which express their meaning mathematically. Further, the relationships between conversion ratio and fissile inventory ratio are derived, and explanations of their practical use in thermal breeder analysis are presented.

1. Neutron Reactions with Nuclides of the LWBR Fuel

To facilitate an understanding of conversion ratio and fissile inventory ratio, a discussion is given here of the interactions which occur between neutrons and the nuclides of the U-233-thorium fuel system. These reactions include those of prime importance to breeding, that is, those of neutron capture in fertile materials. The subset of fissionable nuclides which can fission following the absorption of a zero kinetic energy neutron (that is, absorption of a thermal as opposed to a fast neutron) are defined as "fissile" nuclides. This set of fissile nuclides includes U-232, U-233, and U-235. When neutrons are captured in the fertile materials Pa-231, Th-232, U-232, and U-234, the fissile materials U-232, U-233, and U-235 are produced. These same fissile materials are destroyed, however, as they fission to preserve criticality and produce power. When the rate of conversion to fissile material exceeds the rate of fissile destruction, breeding occurs.

The processes of fertile captures, fissile destruction, and all other neutron reactions occurring in the LWBR fuel system are represented by defining "chains" of events which the fuel nuclides undergo upon reaction with neutrons.

These chains have previously been described in Section IV.B.1 for the U-233-thorium fuel system. Note that the nuclide U-232 has both fissile and fertile characteristics and in the following breeding analysis will be allowed this dual nature.

2. Definition of Fissile Inventory Ratio (FIR)

The definition of FIR is:

$$FIR(t) = \frac{\text{fissile atoms present at time } t}{\text{initial fissile atoms present}} \quad (1)$$

With one exception, fissile materials are those materials which fission epithermally and thermally and have epithermal and thermal η values greater than unity; these materials are U-232, U-233, and U-235.

The one material in addition to the above nuclides which must be treated as fissile material is Pa-233. Pa-233 beta decays to U-233 with a half-life of 27 days, and upon core shutdown all Pa-233 present will decay to U-233. Thus, Pa-233 must be included in a fissile material inventory.

A basic assumption in the above definition of FIR is that any fissile isotope is just as worthy as another fissile isotope, that is, U-233, U-232 or U-235 are not preferentially weighted relative to each other. Accordingly, Equation (1) above may be expanded to:

$$FIR(t) = \frac{\gamma N^{U-232}(t) + N^{U-233}(t) + N^{U-235}(t) + N^{Pa-233}(t)}{N^{U-232}(0) + N^{U-233}(0) + N^{U-235}(0)} \quad (2)$$

where:

N^i = the nuclide concentration of nuclide i

t = time

0 = zero time

γ = the average fission to absorption ratio for U-232

3. Calculation of Fissile Inventory Ratio

Although the actual measure of breeding is defined above, neither Equation (1) nor Equation (2) is used to actually calculate FIR. Instead, FIR is

determined by integrating the average conversion ratio (CR) as a function of a time. The reason for integrating CR rather than using Equations (1) and (2) is that the nuclide concentrations available from calculated data lead to slightly inaccurate FIR's. The inaccuracy arises because of approximations made within the calculation. Nuclide concentrations are determined by depletion equations which use constant fluxes over a relatively long timestep rather than time dependent fluxes. This depletion procedure is sufficiently accurate for reactivity and power calculations, but for FIR calculations the integration of conversion ratio has been shown to be more accurate than the use of nuclide concentrations, and equivalent to the use of very small timesteps. The integration process and definition of conversion ratio are given below.

4. Definition of Conversion Ratio

Conversion ratio is defined as follows:

$$CR = \frac{\text{instantaneous production rate of fissile atoms}}{\text{instantaneous destruction rate of fissile atoms}} \quad (3)$$

Substituting the fissile nuclides, Equation (3) becomes

$$CR(t) = \frac{C^{Th-232}(t) + C^{U-232}(t) + C^{U-234}(t) + A^{Pa-231}(t) \left(\frac{F^{U-232}(t)}{A^{U-232}(t)} \right) - A^{Pa-233}(t)}{F^{U-232}(t) + A^{U-233}(t) + A^{U-235}(t)} \quad (4)$$

where:

$C^i(t) = A^i(t) - F^i(t)$ = capture rate in nuclide i at time t

$A^i(t)$ = absorption rate of nuclide i at time t

$F^i(t)$ = fission rate of nuclide i at time t

The nuclides Th-232, U-232, Pa-231, and U-234 are considered to be "fertile material" since upon neutron capture they are converted to fissile material. The assumption that Pa-231 does not fission is implicit in Equation (4). Available nuclear data for Pa-231 support this assumption.

5. Integration of Conversion Ratio

The definition of conversion ratio given above shows the reaction rates which determine the amount of fissile fuel produced and destroyed in an

instantaneous manner. Classifying these same reaction rates as those which are fertile and those which are fissile, new expressions may be written which describe the fissile fuel produced (FFP) and fissile fuel destroyed (FFD) for the period of time t_n to t_{n+1} . These expressions are:

$$FFP(t_n \rightarrow t_{n+1}) = \left(A_n^{\text{fertile}} - F_n^{\text{fertile}} - A_n^{\text{Pa-233}} \right) \cdot (t_n - t_{n+1}) \quad (5)$$

$$FFD(t_n \rightarrow t_{n+1}) = \left(A_n^{\text{fissile}} \right) \cdot (t_{n+1} - t_n) \quad (6)$$

where:

A_n^i = the absorption rate of nuclide i at the beginning of the time period n

F_n^i = the fission rate of nuclide i at the beginning of time period n

Combining the expression above to determine the atoms of fissile fuel present in the core at time t_{n+1} , $FF(t_{n+1})$:

$$FF(t_{n+1}) = FF(t_n) + FFP(t_n \rightarrow t_{n+1}) - FFD(t_n \rightarrow t_{n+1}) \quad (7)$$

Now, the average conversion ratio over the time interval is defined as:

$$\overline{CR}(t_n) = \frac{FFP(t_n \rightarrow t_{n+1})}{FFD(t_n \rightarrow t_{n+1})} \quad (8)$$

Dividing Equations (5) and (6) and then substituting Equation (7) into the result yields

$$FFP(t_n \rightarrow t_{n+1}) = [\overline{CR}(t_n)] [FFD(t_n \rightarrow t_{n+1})] \quad (9)$$

Equation (7) can now be rewritten using Equation (9):

$$\begin{aligned} FF(t_{n+1}) &= FF(t_n) + [\overline{CR}(t_n)] [FFD(t_n \rightarrow t_{n+1})] - FFD(t_n \rightarrow t_{n+1}) \\ &= FF(t_n) + [\overline{CR}(t_n) - 1] [FFD(t_n \rightarrow t_{n+1})] \end{aligned} \quad (10)$$

FIR is defined as:

$$FIR = \frac{FF(t_{n+1})}{FF(t_0)} \quad (11)$$

Combining this expression with Equation (10):

$$FIR(t_{n+1}) = \frac{FF(t_n) + [\overline{CR}(t_n) - 1] [FFD(t_n \rightarrow t_{n+1})]}{FF(t_0)} \quad (12)$$

Rearranging and observing that $FF(t_n)/FF(t_0) = FIR(t_n)$, the equation above becomes

$$FIR(t_{n+1}) = FIR(t_n) + \frac{1}{FF(t_0)} [\overline{CR}(t_n) - 1] [FFD(t_n \rightarrow t_{n+1})] \quad (13)$$

Total FIR is a cumulative sum over n time periods and can be expressed

$$FIR(t) = 1 + \frac{1}{FF(t_0)} \sum_{p=1}^n [\overline{CR}(t_p) - 1] [FFD(t_p \rightarrow t_{p+1})] \quad (14)$$

where:

$$p = 1, 2, 3 \dots$$

$$n = \text{the time period index}$$

$$FIR(t_0) = 1$$

Equation (14) is an expression for FIR over n discrete time intervals. In the limit, as the time interval approaches zero, Equation (14) becomes:

$$FIR(t) = 1 + \frac{1}{FF(0)} \int_0^t [CR(t) - 1] [A^{\text{fissile}}(t)] dt \quad (15)$$

where $CR(t)$ and A^{fissile} are instantaneous and not time-averaged quantities.

In actual practice, Equation (14) is a numerical integration over the time periods (or time steps) of a depletion calculation. For example

$$\Delta FIR(t_n) = \Delta FIR(t_n \rightarrow t_{n+1}) = \frac{\frac{(A(t_{n+1})^{\text{fissile}} + A(t_n)^{\text{fissile}})}{2}}{FF_0} \left[\left(\frac{CR(t_{n+1}) + CR(t_n)}{2} \right) - 1 \right] \Delta t \quad (16)$$

and,

$$FIR = 1.0 + \sum \Delta FIR(t),$$

or,

$$= 1.0 + \sum \frac{\overline{A}^{\text{fissile}}}{FF_0} (\overline{CR} - 1) \Delta t \quad (17)$$

Differentiation of this expression with respect to time yields

$$\Delta \text{FIR}(t) = (\overline{\text{CR}} - 1) \frac{\overline{A}^{\text{fissile}}}{N_0} \Delta t \quad (18)$$

This relationship is particularly useful in determining lifetime breeding effects based upon observed changes in conversion ratio at beginning-of-life.

6. Other Parameters of Breeding

Another definition can be given for conversion ratio that leads to the derivation for $\eta\epsilon$, η , and ϵ . These parameters are important in analyzing neutron spectrum effects on breeding. This second definition of conversion ratio is

$$\text{CR}(t) = \eta\epsilon(t) = 1 - \text{losses}(t) \quad (19)$$

The total neutron production ($\eta\epsilon$) minus all neutron losses and minus the neutron required to preserve criticality is a measure of the number of neutrons available for conversion of fertile material to fissile material. The above definition of conversion ratio may be combined with previous definitions to derive expressions for $\eta\epsilon$, η , and ϵ . These derivations are provided below.

7. $\eta\epsilon$, η , and ϵ

In the steady-state reactor calculation, the neutrons absorbed must equal the neutrons produced, that is,

$$A^{\text{total}}(t) = \frac{\{\nu F(t)\}^{\text{total}}}{\lambda} \quad (20)$$

where:

$$\begin{aligned} A^{\text{total}}(t) = & A^{\text{Th-232}}(t) + A^{\text{U-231}}(t) + A^{\text{Pa-231}}(t) \\ & + A^{\text{U-233}}(t) + A^{\text{U-235}}(t) + A^{\text{U-232}}(t) + A^{\text{Pa-233}}(t) \\ & + A^{\text{other}}(t) + A^{9998}(t), \end{aligned}$$

or

$$\begin{aligned} A^{\text{Th-232}} + A^{\text{U-232}} + A^{\text{U-234}} = & [(\nu F)^{\text{total}}/\lambda] - A^{9998} - A^{\text{U-233}} \\ & - A^{\text{U-235}} - A^{\text{Pa-233}} - A^{\text{Pa-231}} - A^{\text{other}} \end{aligned} \quad (21)$$

where:

λ = the k_{eff} of the calculation

$A^{other}(t)$ = absorption rate in H, O, Zr, Xe, fission products and other parasitic material

A^{9998} = an artificial absorption rate to properly account for the thorium n, 2n reaction

Rearranging this equation and substituting into Equation (4) yields

$$CR = \frac{[(\nu F)^{total}/\lambda] - A^{9998} - (F^{Th-232} + F^{U-234}) - A^{Pa-231} \frac{C^{U-232}}{A^{U-232}}}{D^{fiss}} - 1 - \frac{A^{other} + 2 A^{Pa-233}}{D^{fiss}} \quad (22)$$

where:

D^{fiss} = fissile destruction rate
 $= F^{U-232} + A^{U-233} + A^{U-235}$

When the form of Equation (22) is compared to that of Equation (19),

$$\eta_{eff} = \frac{[(\nu F)^{total}/\lambda] - A^{9998} - A^{Pa-231} \frac{C^{U-232}}{A^{U-232}} - (F^{Th-232} + F^{U-234})}{D^{fiss}} \quad (23)$$

$$\text{and losses} = \frac{A^{other} + 2 A^{Pa-233}}{D^{fiss}} \quad (24)$$

If we let

$$\eta = \frac{\text{rate of neutron production by fissile}}{\text{rate of neutron destruction of fissile}} \quad (25)$$

$$= \frac{(\nu F)^{fiss}/\lambda}{D^{fiss}} \quad (26)$$

then by division

$$\epsilon = 1 + \frac{[(\nu F)^{\text{fert}}/\lambda] - (F^{\text{Th-232}} + F^{\text{U-234}})}{(\nu F)^{\text{fiss}}/\lambda} + \frac{(\nu F)^{\text{other}}/\lambda}{(\nu F)^{\text{fiss}}/\lambda} + \frac{A^{9998}}{(\nu F)^{\text{fiss}}/\lambda} - \frac{A^{\text{Pa-231}} \frac{C^{\text{U-232}}}{A^{\text{U-232}}}}{(\nu F)^{\text{fiss}}/\lambda} \quad (27)$$

In order to write the conversion ratio in terms of production of neutrons minus losses, it is necessary to define a production and a loss term. We call the production term $(\eta\epsilon - 1)$ where $\eta\epsilon$ is defined by Equation (23). $\eta\epsilon$ represents all the non-fertile plus the net fertile production of neutrons per neutron destruction of a fissile atom. The quantity $(\eta\epsilon - 1)$ is the net production of neutrons per fissile atom destroyed. The loss term counts those neutrons which are absorbed in materials which are not beneficial to the breeding process.

The $\eta\epsilon$ breeding parameters can be written in a slightly different form. By dividing Equation (23) by A^{total} and using Equation (20):

$$\eta\epsilon = \frac{1 - \frac{A^{9998}}{A^{\text{total}}} - \frac{F^{\text{fert}}}{A^{\text{total}}} - \frac{A^{\text{Pa-231}}}{A^{\text{total}}} \times \frac{C^{\text{U-232}}}{A^{\text{U-232}}}}{(\nu F)^{\text{fiss}}/A^{\text{total}}} \quad (28)$$

Now, the steady state equilibrium conditions are

$$(\nu F)^{\text{total}}/\lambda = A^{\text{total}} \quad (20)$$

$$\text{and production} = \text{absorption} + \text{leakage} \quad (29)$$

that is:

$$(\nu F)^{\text{total}} = A^{\text{total}} + L \quad (30)$$

Simple algebra using Equations (20) and (30) results in

$$L_f = L/A^{\text{total}} = \lambda - 1 \quad (31)$$

This equation defines the leakage fraction, L_f .

A neutron balance demands that

$$L_f + \sum_i A_f^i = 1 \quad (32)$$

where the A_f^i are the isotopic absorption fractions. This equation can be written

$$\sum_i [A_f^i / (1 - L_f)] = 1 \quad (33)$$

Further, by definition

$$\sum_i \frac{A^i}{A_{\text{total}}} = 1 \quad (34)$$

and therefore

$$A_f^i = (1 - L_f) \frac{A^i}{A_{\text{total}}} \quad (35)$$

Equation (28) can be rewritten as

$$\eta \epsilon = \frac{\left(1 - \frac{F^{\text{fert}}}{A_{\text{total}}}\right) (1 - L_f) - \left(A_f^{9998} + A_f^{\text{Pa-231}} \frac{C^{\text{U-232}}}{A^{\text{U-232}}}\right)}{\left(A_f^{\text{U-233}} + A_f^{\text{U-235}}\right) + \frac{F^{\text{U-232}}}{A_{\text{total}}} (1 - L_f)} \quad (36)$$

This equation is sometimes more useful than Equation (23).

II. NEUTRON POISON EQUIVALENCE (NPE) METHOD FOR SETTING IMPURITY LIMITS

1. Definition of Neutron Poison Equivalence

The need for a concise, accurate method for use in setting pellet impurity limits prompted the development of neutron poison equivalences. The reduction in conversion ratio caused by impurities in the fuel would depend on the amount of each impurity present and on the neutron absorption properties of each element. Elements with large neutron absorption cross sections could be tolerated at only low levels, whereas elements with moderate cross sections could be tolerated at higher levels. Impurities with extremely low cross sections might not need to be controlled at all from the point of view of breeding. Nuclides with extremely high cross sections deplete rapidly and may affect breeding less adversely late in life than early in life, depending on their daughter nuclides. The cross sections of some nuclides are strongly dependent on spectrum, and their effect on breeding would differ by region. A system was needed which would correlate all these facts and assist in evaluating impurity levels.

The Neutron Poison Equivalence (NPE) method of impurity analysis quantizes the neutron poisoning capability of any given element or group of elements and evaluates this poisoning relative to that of any other element or group of elements. Specifically, NPE's are a set of numbers which compare the total number of neutrons absorbed by any element over a depletion history to the number of neutrons absorbed by a reference isotope on an equal weight basis over the same period of time. In the LWBR NPE system, the element cobalt was chosen as the reference isotope with a standard concentration of one part per million by weight of thorium. The neutron absorption capability of one part cobalt per million parts of thorium was assigned a reference value of 1.0. Cobalt was chosen since its absorption rate is relatively insensitive to neutron spectrum and lifetime. Mathematically, the NPE of any element, i , may be expressed by:

$$NPE_i(t) = \int_0^T A_i(t) dt / \int_0^T A_{\text{cobalt}}(t) dt, \quad (1)$$

where $A_i(t)$ is the neutron absorption rate of element i . Details of how these depletion effects are calculated are explained below.

The breeding reduction caused by an impurity depends upon the amount of the impurity present, the neutron absorption cross section for that impurity, and the neutron environment including depletion. The depletion effects are important, since certain elements with a high absorption cross-section tend to burn out over lifetime while other elements with a low absorption cross-section do not. Further, daughter products are formed from the impurity initially present. These daughter products form and deplete at different rates and may cause an increase in the integrated absorption attributable to the initial impurity isotopes. A modified version of the CINDER program (Reference B-1) was adapted to correlate all these factors and provide a consistent and total evaluation of the neutron absorption of any element over lifetime. The program requires as input data power or flux levels, group flux ratios, time step duration, nuclide cross sections and self-shielding factors, initial atom densities including concentrations of 1 ppm of each impurity, radioactive decay and fission yield fractions, and nuclide chain coupling information. From these input data, the time integrated neutron absorption for each nuclide is computed. As shown in Equation (1), the ratio of these integrated neutron absorptions for each element to that of cobalt is defined as the NPE for that element.

2. NPE and Conversion Ratio

To assess the effect on breeding of varying amounts of impurities the term "total neutron poison equivalency" is useful. This term, NPEY, is defined to be:

$$\text{NPEY} = \sum_i (C_i) (\text{NPE}_i) \quad , \quad (2)$$

where:

C_i = the concentration of element i present initially in ppm (grams of i per million grams of thorium)

NPE_i = the neutron poison equivalence of element i .

From the definition of NPE, it follows that an effective cobalt concentration may be defined as:

$$C_{\text{cobalt}}^{\text{eff}} = \text{NPEY}$$

Now, let us investigate the effect on conversion ratio of this "equivalent" concentration of cobalt. The change in conversion ratio, ΔCR , caused by an absorption fraction in cobalt, ΔF_{cobalt} , is written as:

$$\Delta CR = \frac{1}{AF_{\text{fissile}}} \cdot AF_{\text{cobalt}}$$

where AF_{fissile} is the absorption fraction of fissile fuel. The cobalt absorption fraction is computed as follows:

$$AF_{\text{cobalt}} = C_{\text{cobalt}} \cdot \left(\frac{A_{\text{cobalt}}}{A_{\text{thorium}}} \right)_{1 \text{ ppm}} \cdot AF_{\text{thorium}}$$

where:

C_{cobalt} = the concentration of cobalt in ppm

$A_{\text{cobalt}}/A_{\text{thorium}}$ = the ratio of the cobalt absorptions to the thorium absorptions for an initial cobalt concentration of 1 ppm

AF_{thorium} = the absorption fraction of thorium.

The expression for change in CR becomes:

$$\Delta CR = \frac{1}{AF_{\text{fissile}}} \cdot C_{\text{cobalt}} \left(\frac{A_{\text{cobalt}}}{A_{\text{thorium}}} \right)_{1 \text{ ppm}} \cdot AF_{\text{thorium}}$$

A typical computation of this kind is the following:

$$\frac{1}{AF_{\text{fissile}}} = 2.26$$

$$\left(\frac{A_{\text{cobalt}}}{A_{\text{thorium}}} \right)_{1 \text{ ppm}} = 2.0 \times 10^{-5}$$

$$AF_{\text{thorium}} = 0.47$$

$$\Delta CR = -0.000021 \cdot C_{\text{cobalt}}$$

Given the total neutron poison equivalency, NPEY, of the impurities, the effect on conversion ratio would be predicted to be

$$\Delta CR = -0.000021 (NPEY_{total})$$

A very useful application of the above relationship is the determination of a maximum allowable value of NPEY. By limiting total conversion ratio losses to a specified amount, the total allowable NPEY can be calculated. This total can then be distributed among the various impurities to establish limits on individual impurity concentrations. Obviously, this total limit must be consistent with manufacturing process capability. This is the case for LWBR limits. To assure meeting the neutron poison equivalence requirement, it has been necessary to require a special purification step to the thorium nitrate feed material to reduce losses to rare earth impurities.

3. Breeding Comparison of Two Alloys

NPE's can also be used to compare the poisoning effects of different alloys. For example, a comparison between SS-304 and AM-350 stainless steel could be made by computing a quantity defined as "poison indicator," PI. The poison indicator for alloy a, PI_a , is computed as follows:

$$PI_a = \rho_a \sum_i W_i \cdot NPE_i$$

where:

ρ_a = the density of alloy a

W_i = the weight percent of element i in alloy a

NPE_i = the neutron poison equivalence of element i

By comparing the poison indicators of SS-304 (1.618) and AM-350 (1.496), the designer can tell which alloy is more poisonous to neutrons. The alloy with the larger PI value will absorb the most neutrons. Since SS-304 is more poisonous than AM-350, the ratio ($PI_{SS-304}/PI_{AM-350} = 1.082$) indicates the fractional increase in, for example, grid losses which would be caused by fabricating the grids of SS-304 instead of AM-350 provided the grid volume is not changed.

4. Fuel Impurities

Returning now to a discussion of the various categories of impurities, consider first the impurities contained in the fuel-pellets. These pellet impurities are controlled by imposing (1) maximum allowable limit on the poisoning effects of individual impurities and (2) a maximum allowable limit on the total poisoning effects of all impurities present. The individual impurity limits allow the manufacturer some flexibility for the type and amount of impurities introduced locally while assuring adequate physical properties of the fuel, but the total impurity limit ensures against accumulation of excessive amounts of any one or any group of impurities. Both types of fuel impurity limits are specified in units of NPE in the manufacturing specifications for UO_2 - ThO_2 pellets, ThO_2 pellets, and the feed materials, thorium nitrate, and uranyl nitrate. Different specifications are required for the pellets and the feed materials since some impurities not present in the feed materials may be introduced during the pellet manufacturing process. The probable fuel impurities are identified in Tables B-1, -2, -3, and -4. Table B-1 lists the controlled impurities and their allowable limits for the UO_2 - ThO_2 fuel pellets; Table B-2 does likewise for the ThO_2 pellets. The ThO_2 pellet impurity limits are identical to those of the UO_2 - ThO_2 pellets with the exceptions that uranium is included as an impurity in the ThO_2 specification and Pu is included as an impurity in the binary specification. Note that the specifications for both pellet types require that the average impurity content shall not exceed a total NPE of 50.

Tables B-3 and B-4 list the controlled impurities and their allowable limits for the thorium and uranyl nitrate feed material. The tables show maximum individual impurity limits that apply to individual lots of material. However, these maxima do not apply over the entire composition, but rather the overall impurity content in a composition shall not exceed a total NPE of 50. This procedure gives control over the total impurity content in a composition while allowing some variability in content of individual impurities. The neutron poison equivalence factors presented in Tables B-1, -2, -3, and -4 represent conservative choices over a wide range of neutron spectra and lifetimes. Rare earths are controlled in thorium nitrate because this is the plausible source of rare earth contamination since thorium is often found with rare earths in natural ores. Restrictive limits on maximum content of these elements with high absorption

cross-section could, if present in large enough quantities, cause an initial reactivity bias. Such a bias would rapidly change with depletion, confusing physics test results. The thorium nitrate specification on rare earth limits also include a provision which states that the average content of specified rare earths in the ThO_2 material shall not exceed a total NPE of 10. In addition, the remainder of the average impurity content of the thorium nitrate material shall not exceed an NPE of 25. Note, too, that the uranyl nitrate specifications are not as tightly controlled as those of the thorium feed materials. These uranyl nitrate limits are relaxed such that the total average impurity content shall not exceed 260. This relatively relaxed limit is acceptable since the percent uranium in the fuel is of the order of 5 weight percent or less.

5. Zr-4 Impurities

The LWBR core contains many components fabricated of Zr-4 material which influence breeding. Included in these components are Zr-4 rod cladding, full length shells between seed and blanket, and the posts used for grid attachments. Impurities in these zirconium alloy components augment the breeding penalties caused by the base metal, and they are, therefore, of interest when assessing breeding performance.

Table B-5 identifies the basic constituents of the Zr-4 alloy and the alloy impurities. These constituents and impurities were identified in the material specification for Zircaloy-4 wrought product. Note from the impurity table that maximum allowable concentrations are specified for each individual impurity in units of "parts per million parts zircaloy" by weight.

The breeding effects of Zr-4 impurities were explicitly calculated in all nuclear design calculations. This explicit treatment was accomplished by modifying the Zr-4 neutron cross sections to include the neutron cross sections of the impurities. In the LWBR calculations, therefore, the breeding losses to the impurities are included in the breeding losses calculated for Zr-4.

6. AM-350 Impurities

Table B-6 identifies constituent elements and probable impurities for the AM-350 stainless steel rod support system. Further, the maximum allowable individual impurity levels are reproduced from the AM-350 stainless steel material specification.

In a manner similar to that employed when representing the Zr-4 alloy, nuclear design calculations include the effect of AM-350 impurities explicitly by modifying the base AM-350 neutron cross sections.

III. BREEDING SENSITIVITIES TO DESIGN CHANGES

During the development and design of the LWBR, it was often necessary to provide estimates of the breeding effects of changes in design specifications. These changes included variations in clad thickness, fuel pellet configuration, rod size and pitch, grid volumes, and other parameters. To determine the nuclear effects of a particular design change, data from at least two calculations are required. The first of these calculations must be a controlled or reference problem which describes a reference core design. The second problem is designed to be an exact duplicate of the first except for the particular design change of interest. This exact duplication avoids the problem of confusing two different effects in one set of calculational results.

As a general rule, the design change sensitivities were estimated based on data from beginning-of-life (BOL) calculations. In most cases, BOL data are sufficient, because methods have been developed to estimate many of the well known depletion effects, including predicting changes in peak FIR based on beginning-of-life conversion ratio differences. (These methods are discussed in Section I of this Appendix.) Depletion calculations are performed only when very complicated depletion effects are expected to result from the design change.

The change in core breeding performance is analyzed in two parts. The first is the change in breeding caused by substituting one core material for another; for example, Zr-4 clad for thorium pellet. The second is the change in breeding caused by the changes in core reactivity characteristics. Breeding is coupled to reactivity since movable seed motion determines the amount of axial neutron leakage from the core. As the movable seed is positioned away from the aligned location, neutron leakage to the reflector regions increases and breeding performance deteriorates. Conversely, some gain in breeding may be realized if the movable seeds are positioned closer to the aligned location as a result of decreased reactivity. Thus, breeding performance is influenced by (1) exchanging core materials and (2) system reactivity perturbations which influence the conversion ratio losses to axial reflectors. To determine the magnitude of these effects, the necessary breeding data comparisons are those of (1) comparing conversion ratios from each problem that is corrected for reflector loss differences, and (2) comparing initial critical positions which are indicative of conversion ratio losses to reflector regions over lifetime. The process of correcting

conversion ratios consists simply of normalizing the two calculated conversion ratios to the same reflector losses. This normalization to equivalent reflector losses removes the influence of reactivity on conversion ratio and allows a determination of the effects of material exchanges alone. The process of comparing the initial critical positions provides an estimate of fuel position differences over lifetime. Based on this comparison and upon existing correlations of reflector losses with fuel position, an estimate can be made of the expected change in conversion ratio losses to the reflector. The total conversion ratio effect is the sum of material and reflector loss influence. Of these two breeding influences, the effect of exchanging core material is generally the major effect, for only severe reactivity perturbations significantly affect reflector losses.

Where reactivity changes are small, changes in peak FIR are inferred from the beginning-of-life conversion ratio data through the relationship $\Delta \text{Peak FIR} = k \Delta \text{CR}$, where k has a value of 0.425 for the LWBR core with a peak FIR lifetime of 19,000 equivalent full power hours. This relationship is derived and explained in Section I of this Appendix. When reactivity changes are large, peak FIR effects are estimated by using the above relationship augmented by a term, $k \Delta \text{CR}_{\text{refl losses}}$, to account for reflector loss increases or decreases.

The beginning-of-life procedures outlined above are used only when the design change of interest is not expected to influence breeding losses to depletion products. If the design change is expected to significantly affect Pa-233 or fission product behavior or significantly perturb the neutron spectrum, then full depletion calculations are required to assess such effects.

Breeding sensitivities developed from calculations like those described above are presented in Table B-7. The ΔFIR values given in Table B-7 were not calculated for the as-built LWBR core presented in this report but were calculated for earlier quite similar designs. In addition, because of the large expense of the three dimensional core calculations, many of the sensitivities were calculated using a two dimensional representation of the Type I module. However, subsequent three dimensional core calculations for these earlier designs, with several design changes included simultaneously, indicate that the full core sensitivities are roughly equivalent to those calculated for the Type I module. Therefore, the breeding sensitivities presented in Table B-7 should be considered as reasonable order of magnitude sensitivities for the as-built LWBR core.

REFERENCE

- B-1. T. R. England, "CINDER - A One-Point Depletion and Fission Product Program," WAPD-TM-334 (Revised), June 1964. Also, T. R. England, R. Wilczynski, and N. L. Whitmore, "CINDER-7: An Interim Report for Users," LA-5885-MS, April 1975.

TABLE B-1. IMPURITY LIMITS FOR UO_2 - ThO_2 PELLETS

<u>Impurity</u>	<u>Neutron Poison Equivalence Factor</u>	<u>Max. Individual Impurity Limit (ppm)</u>
Al	0.01	500
B	30.0	1
C	----	200
Ca	0.07	70
Cl & Br	1.2	15
Co	1.0	10
Cr	0.07	100
Cu	0.09	40
F	----	20
Fe	0.06	300
Mg	0.03	100
Mn	0.34	10
Mo	0.11	100
N	0.11	50
Ni	0.11	200
Si	0.02	300
Ti	0.16	20
V	0.13	25
Hg	1.10	1.0

Note 1 - The total Neutron Impurity Equivalence is determined by calculating the summation for all impurities of the product of (ppm of impurity) times (Neutron Poison Equivalence factor of the impurity). Limits of detection shall be used in calculating the Neutron Poison Equivalence when results of analysis are below those limits.

Note 2 - Other impurities specified in fuel ordering information are as follows:

Pu 0.94 Ufissile	--	2.0
Pu 1.512 Ufissile	--	3.0
Pu 1.890 Ufissile	--	4.0
Pu 2.508 Ufissile	--	5.0
Pu 3.729 Ufissile	--	8.0
Pu 5.476 Ufissile	--	10.0

TABLE B-2. IMPURITY LIMITS FOR ThO₂ PELLETS

<u>Impurity</u>	<u>Neutron Poison Equivalence Factor</u>	<u>Max. Individual Impurity Limit (ppm)</u>
Al	0.01	500
B	30.0	1
C	----	200
Ca	0.07	70
Cl & Br	1.2	15
Co	1.0	10
Cr	0.07	100
Cu	0.09	40
F	----	20
Fe	0.06	300
Mg	0.03	100
Mn	0.34	10
Mo	0.11	100
N	0.11	50
Ni	0.11	200
Si	0.02	300
Ti	0.16	20
V	0.13	25
U	----	25
Hg	1.10	1.0

Note 1 - The total Neutron Impurity Equivalence is determined by calculating the summation for all impurities of the product of (ppm of impurity) times (Neutron Poison Equivalence factor of the impurity). Limits of detection shall be used in calculating the Neutron Poison Equivalence when results of analysis are below those limits.

Note 2 - Total order quantity Uranium impurity shall not exceed 10 ppm.

TABLE B-3. THORIUM FEED MATERIAL IMPURITIES

<u>Impurity</u>	<u>Neutron Poison Equivalence Factors</u>	<u>Max. Individual Impurity Limit (ppm)</u>
<u>Rare Earths</u>		
Dy	6.2	2
Eu	14.0	1
Cd	3.2	2
Sm	4.1	5
<u>Others</u>		
Al	0.01	50
B	30.0	1
Ca	0.07	40
Cl & Br	1.2	10
Cr	0.07	25
Hg	1.10	25
Co	1.00	10
Cu	0.09	10
F	----	10
Fe	0.06	75
Mg	0.03	25
Mn	0.34	5
Mo	0.11	25
Ni	0.11	50
P	----	150
Si	0.02	50
Ti	0.16	10
U	----	10
V	0.13	5

TABLE B-4. URANYL NITRATE FEED MATERIAL IMPURITIES

<u>Impurity</u>	<u>Neutron Poison Equivalence Factors</u>	<u>Max. Individual Impurity Limit (ppm)</u>
Al	0.01	300
B	30.0	3
Ca	0.07	100
Cd	2.3	5
Cl & Br	1.2	15
Co	1.0	25
Cr	0.07	100
Cu	0.09	150
Fe	0.06	1000
K	0.06	200
Hg	1.10	250
Mg	0.03	150
Mn	0.34	50
Mo	0.11	100
Na	0.02	300
Ni	0.11	200
Np	2.4	100
P	0.03	150
Pu (total)	----	150
Pu-238	1.05	25
Si	0.02	300
Th	----	1000
Ti	0.16	75
V	0.13	100
W	0.50	20
Zn	0.02	100

TABLE B-5. Zr-4 CONSTITUENT ELEMENTS AND IMPURITIES

Constituents

<u>Element</u>	<u>Weight Percent</u>
Sn	1.20 - 1.70
Fe	0.18 - 0.24
Cr	0.07 - 0.13
O	0.09 - 0.16
Zr	Remainder

Impurities

<u>Element</u>	<u>ppm Maximum</u>	<u>Element</u>	<u>ppm Maximum</u>
Al	75	N	80
B	0.5	Nb	100
Cd	0.5	Ni	70
C	270	Si	120
Co	20		
Cu	50	Ta	200
Hf	35	Ti	50
H	25	W	100
Mn	50		
Mo	50	Th	7
P	100	Gd	5
Pb	100	U (total)	3.5
Zr	100	U-235	0.025
Sm	10	V	50

TABLE B-6. AM-350 CONSTITUENT ELEMENTS AND IMPURITIES

<u>Constituent Element</u>	<u>w/o</u>
C	0.06 - 0.10
Mn	0.60 - 1.0
Cr	16.4 - 17.0
Ni	4.0 - 4.5
Mo	2.5 - 3.0
N	0.07 - 0.11
Fe	74.79 - 73.64

<u>Impurity Element</u>	<u>Maximum w/o</u>
Si	0.50
P	0.04
S	0.03
B	0.01
Co	0.10

TABLE B-7. BREEDING SENSITIVITIES

<u>Fuel Rod and Pellet Dimensions</u>	<u>ΔFIR</u>
1. Increase all clad thickness 1 mil Preserve pellet OD Increase rod OD 1.0 mil Decrease gap size 0.5 mil	-0.00059
2. Decrease seed clad thickness 1 mil Preserve rod OD Preserve fuel-to-clad gap Increase pellet OD 2 mils	+0.00041
3. Decrease blanket clad thickness 1 mil Preserve rod OD Preserve fuel-to-clad gap Increase pellet OD 2 mils	+0.00041
4. Decrease all clad thickness 1 mil Preserve pellet OD Preserve fuel-to-clad gap Decrease rod OD 2 mils	+0.00005
5. Seed clad shrinkage of 1 mil in Seed I, 0.8 mil in Seed II, III Preserve pellet OD Preserve clad thickness	-0.00028
6. Blanket clad shrinkage of 1 mil on binary, 1.33 mils on ThO ₂ Preserve pellet OD Preserve clad thickness	-0.00024
<u>Grid Volume and Material</u>	
1. Decrease seed grid volume 10 percent	+0.00023
2. Decrease blanket grid volume 10 percent	+0.00027
3. Substitute Inconel-718 for AM-350 on equal volume basis	-0.0027
4. Substitute Zr-4 for AM-350 on equal volume basis	+0.0038

TABLE B-7. BREEDING SENSITIVITIES (Cont'd)

Metal and Water Sensitivities

1. Add a seed center post, remove seven fuel rods	-0.00022
2. Increase seed rod spacing 1 mil Preserve rod OD	+0.00008
3. Increase blanket rod spacing 1 mil Preserve rod OD	-0.00035
4. Increase T-inlet by 10F	+0.00032
5. Decrease system pressure 100 psi	+0.00005

Fuel Content

1. Increase seed percent theoretical density from 97 to 99 percent Preserve Uranium loading	+0.00017
2. Increase blanket percent theoretical density from 97 to 99 percent Preserve Uranium loading	+0.00042
3. Increase reflector blanket percent theoretical density from 97 to 99 percent	+0.00005
4. Pellet chips, seed Preserve nominal w/o Decrease volume of all seed pellets 0.5 percent	-0.00002
5. Pellet chips, blanket Preserve nominal w/o Decrease volume of all blanket pellets 0.5 percent	-0.00012

Core Operation

1. Increase T-inlet by 10F	+0.00032
2. Decrease system pressure 100 psi	+0.00005
3. Reduce thermal power 1.5 percent	+0.0018
4. Four week shutdown twice a year	+0.0012