**THREE CORE CONCEPTS FOR
PRODUCING URANIUM-233 IN
COMMERCIAL PRESSURIZED LIGHT
WATER REACTORS FOR POSSIBLE
USE IN WATER-COOLED BREEDER
REACTORS**

(AWBA Development Program)

MASTER

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BETTIS ATOMIC POWER LABORATORY
WEST MIFFLIN, PENNSYLVANIA

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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 Atomic Energy Commission (now Department of Energy, 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-uranium 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 4 to 5 years. At the end of this period, the core will be removed and the spent fuel shipped to the Naval Reactors Expended 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, now DOE, 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 is exploring some of the problems that would be faced by industry in adapting technology confirmed in the LWBR program. Information being 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.

All three development programs (Pressurized Water Reactor, Light Water Breeder Reactor, and Advanced Water Breeder Applications) are under the technical direction of the Division of Naval Reactors of DOE. They have 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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The water-cooled-breeder fuel cycle, based on the use of thorium and uranium-233, would consist of two phases:

- Prebreeder phase: Uranium-235 or other fissile material would be used to fuel light-water reactors that would also contain thorium; operation of these reactors would produce fissile uranium-233 from the thorium while providing power for generating electricity.
- Breeder phase: Uranium-233 produced in the prebreeder reactors would be used to fuel breeder reactors that are expected to be self-sustaining; that is, they would produce enough fissile fuel to replace the fuel used in generating electricity and require no mining or enrichment of uranium for fissile-fuel makeup.

This report describes selected prebreeder core concepts which could be backfit into a reference light water reactor similar to current commercial reactors, and produce uranium-233 for use in water-cooled breeder reactors. The prebreeder concepts were selected on the basis of minimizing fuel system development and reactor changes required to permit a backfit. The fuel assemblies for the prebreeder core concepts discussed in this report would occupy the same space envelope as those in the reference core but contain a 19 by 19 array of fuel rods instead of the reference 17 by 17 array. An instrument well and 28 guide tubes for control rods have been allocated to each prebreeder fuel assembly in a pattern similar to that for the reference fuel assemblies. Backfit of these prebreeder concepts into the reference reactor would require changes only to the upper core support structure while providing flexibility for alternatives in the type of fuel used.

Three alternative fuel systems were considered. The first fuel alternative evaluated was one in which thorium rods were distributed among slightly enriched uranium rods; uranium-233 was assumed to be retrieved from the thorium rods for use in water-cooled breeders, and the uranium recovered from the spent uranium fuel was assumed to be re-enriched and supplemented with fresh enriched uranium and recycled in prebreeders. A second alternative assumed that the plutonium produced in the uranium was recycled into the thorium rods, thereby burning nearly all of the plutonium produced. This would also permit use of more thorium rods by reason of thermal design considerations, and thereby would achieve higher uranium-233 production. A third alternative assumed substitution of rods containing thorium and highly enriched uranium for those containing slightly enriched uranium. This case would take advantage of the more favorable nuclear characteristics of uranium-233 and would produce minimal amounts of plutonium.

This report identifies changes to the reference reactor, licensability considerations, other technical considerations, and anticipated testing requirements. From the analysis of these concepts, it appears practical to design prebreeder cores that could be backfit into existing commercial reactors without loss in power capability while producing uranium-233 for future use in water-cooled breeders.

THREE CORE CONCEPTS FOR PRODUCING URANIUM-233 IN COMMERCIAL PRESSURIZED LIGHT WATER REACTORS FOR POSSIBLE USE IN WATER-COOLED BREEDER REACTORS (AWBA Development Program)

I. INTRODUCTION

Production of uranium-233 for possible later use in a water-cooled breeder reactor can be accomplished in a pressurized light water reactor which is operating to provide power for generating electricity. The reactor core, which would be

fueled primarily with thorium and enriched uranium, is referred to as a prebreeder core. No prebreeder cores have been built or operated; they exist today only as concepts. A study has been made to establish the feasibility of replacing a conventional core in a pressurized light water reactor by a backfit prebreeder core; that is, a prebreeder core which could be installed and operated in an existing plant without major modifications to the plant, and with no loss in power capability. A constraint in this study was that the prebreeder core concepts would require minimum fuel system development work for commercialization.

The B&W-241 reactor was chosen as a reference because at the time the study was initiated it was an example of a large commercial design having a somewhat lower power density and potentially more flexibility for modification than the other large pressurized water reactor commercial designs. The B&W-241 system is no longer offered and no B&W-241 plants were or are being built. The essential difference between a B&W-241 core and a B&W-205 core is that the fuel assembly and fuel rod power ratings of the 205 are higher and the fuel assembly flow is also higher to provide the same total core power and flow in 205 fuel assemblies as would have been provided in 241 fuel assemblies. It is believed that similar changes could be made to the prebreeder core concepts discussed here and that useful quantities of uranium-233 could be produced in the B&W-205 plant and in other high power density pressurized water reactor plants.

No representation is made or intended either that the particular core concepts presented here would be satisfactory with respect to safety and licensability without further refinements in the core details, or that all of the problems associated with backfit application have been considered.

It should be recognized that the Babcock & Wilcox Company (B&W) was not consulted on this study and therefore bears no responsibility for the way in which

their safety analysis report data (Reference 1) were used, or for the conclusions of this study. In addition, this work should not be construed as an endorsement of the reference B&W Standard-241 core or of its expected performance capability.

A. Background

Because the uranium-233 needed for water-cooled breeder reactors does not occur in nature, use of this fuel cycle would require the production of uranium-233 by some practical means, such as production as a by-product of fission reactors operating to provide electrical energy. These dual purpose reactors, which today exist only as concepts, are referred to as prebreeders. An essential common feature of all prebreeder core concepts discussed here is that they would be fueled with both uranium and thorium. Under neutron irradiation some of the thorium would be converted to uranium-233. Prebreeder core concepts have been studied briefly as part of the initial evaluation of the Light Water Breeder Reactor (LWBR) concept and more recently as part of the environmental assessment for the Shippingport LWBR program (Reference 2). The LWBR studies of prebreeder core concepts included a wide range of alternatives. The Department of Energy (DOE) Advanced Water Breeder Applications (AWBA) program, under the technical direction of the Naval Reactors Division, is developing more detailed studies of commercial-scale prebreeder core concepts to help provide industry with the technical information necessary to evaluate these alternatives. This report describes part of that work.

Among the prebreeder alternatives, including those which would not be constrained to fit into existing PWR plants without loss in power capability, probably the most contrast would exist between core concepts controlled with movable fuel and core concepts controlled by the current commercial practice of varying the concentration of boric acid in the coolant. The movable fuel control

core concepts would be similar to the LWBR at Shippingport in mechanical details and would have relatively high uranium-233 production and relatively small net consumption of uranium ore per unit of uranium-233 produced. Characteristically, they would have lower average power density and hence lower reactor power for a given reactor vessel size, and larger fissile fuel inventory than current commercial cores. The boron control core concepts would be similar to current commercial cores in mechanical details. They would have comparable power density and somewhat larger fissile fuel inventory than current commercial cores. They would have a smaller uranium-233 production rate and a larger net uranium ore consumption per unit of uranium-233 produced than the movable fuel control core concepts.

If a commitment were made to commercialize the LWBR fuel cycle, resource and timing considerations would be important in judging the relative merits of various prebreeder core concepts. The objective of developing the largest number of LWBR cores from available uranium resources might favor use of movable fuel control prebreeder cores which would probably have application primarily in new plants. On the other hand, the objective of earliest commercial operation of prebreeders would favor core concepts requiring minimum changes relative to current commercial core designs. Core concepts satisfying this requirement would be boron control prebreeder cores having the capability to backfit into current commercial reactors. The developmental aspects of the fuel rod and fuel rod support technology needed for these core concepts are identified in this report.

These prebreeder core concepts are all constrained to have the same overall core dimensions and fuel assembly dimensions as the reference core and to have the same power capability. However, in order to maximize the production rate of

uranium-233 for subsequent use in breeders, the number of fuel rods in the prebreeder core concepts was increased relative to the number of fuel rods in the reference core. This resulted in about a 10-percent increase in heavy metal (uranium plus thorium) loading relative to the heavy metal (all uranium) loading in the reference core.

Studies on the use of thorium in a backfit core for a pressurized water reactor have been reported previously in Reference 3. The purpose of those studies was to examine the improvement in fuel utilization (defined as the ratio of energy produced by the reactor to the short tons of natural uranium from milling required to fuel the reactor) attainable by the use of thorium in a specific fashion. The studies considered a core with both a fuel rod lattice and fuel assembly dimensions which were identical to those of a commercial design core. All uranium-233 produced in the thorium was assumed to be recycled into the core. It was concluded that a modest improvement in fuel utilization was attainable, relative to the fuel utilization in the commercial design core.

The goal of the backfit prebreeder core concepts discussed in this report would be to produce uranium-233, containing minimum amounts of other uranium isotopes, for use in a water-cooled breeder. Hence it would be desirable to keep the uranium-233, produced in some or all of the thorium fuel, separate from the uranium isotopes (primarily uranium-235, -236, and -238) present in the uranium fuel. The concepts considered herein would maintain the required separation through the use of two types of fuel rods, referred to as seed rods and blanket rods. Fresh (unirradiated) blanket rods would contain only thorium (and possibly plutonium). Any uranium present in fresh fuel would be in the seed rods only.

Three alternative fuel arrangements are considered in this report. The first alternative (prebreeder core Concept 1) would use slightly enriched uranium seed rods and thorium blanket rods. The second alternative (prebreeder core Concept 2) would use slightly enriched uranium seed rods and blanket rods containing a mixture of plutonium and thorium. The third alternative (prebreeder core Concept 3) would use seed rods containing a mixture of highly enriched uranium diluted with thorium and blanket rods containing only thorium. In core Concept 3 only the uranium-233 produced in the thorium blanket rods would be used in breeders; the uranium-233 produced in the seed rods would be recycled into fresh seed rods.

For prebreeder core Concepts 1 and 3 to have the same power capability as the reference core would require that the total number of prebreeder fuel rods be greater than the total number of fuel rods in the reference core. This is because the thorium blanket rods in the prebreeder would produce little power during the early stages of their irradiation. For prebreeder core Concept 2, where additional power capability early in the life of the blanket rods would be available because of the addition of plutonium to the blanket rods, increasing the number of fuel rods relative to the number in the reference core would increase the total amount of thorium which could be loaded into the core, and thereby would increase the uranium-233 production rate.

In developing the core concepts reported here, consideration was given to increasing the 17 by 17 array of fuel rods in the reference fuel assembly to a 19 by 19 or a 21 by 21 array in the same space envelope. A 19 by 19 array was chosen to provide a reasonable balance among nuclear, thermal, hydraulic, and mechanical considerations. The power capability of each prebreeder core concept would depend on variables such as fuel rod pitch and diameter. The values of the

variables presented here were chosen on the basis of making the calculated power capability for a prebreeder core concept greater than or equal to the calculated power capability for the reference core, where all calculations are made using the same methods. Further optimization studies on these prebreeder core concepts could be made, for example, to improve the production rate of uranium-233. Improvements also could result from the relaxation of the constraints on variables such as core size and fuel assembly size. Such additional studies would go beyond the scope intended for this report since the intention here is to identify core concepts which would require a minimum modification to the reference nuclear steam system. The core technology developments necessary for these core concepts are identified in this report.

In summary, this report presents prebreeder core concepts which could be backfit into the B&W Standard 241 nuclear steam supply system. The fuel assemblies would contain enriched uranium-bearing seed rods in a 19 by 19 rod array and would employ, to the maximum extent possible, fuel rod and fuel rod support technology similar to that currently being developed by U. S. industry. By varying the fuel compositions and the relative number of seed and blanket rods, considerable flexibility is obtained within the specified constraints.

B. Report Content

Descriptions of the physical and operational features of the prebreeder core concepts follow immediately after a statement of the conclusions. Section III (Results) includes a brief summary of the mechanical features, followed by a 70-point comparison with the reference core, summary statements of nuclear, fuel rod, and thermal-hydraulic performance estimates, and a summary of resource requirements and environmental considerations. The sensitivity of performance to changes in the variables is also presented.

Section IV (Considerations for Further Development) discusses differences in mechanical and fuel system features between the prebreeder core concepts and the reference core with respect to licensing and technical considerations for further development of the prebreeder core concepts. Section V identifies some of the development and testing which would be necessary to bring these core concepts to operating status. Details of analysis models and results are given in the appendices:

- A. MECHANICAL CONCEPTS
- B. NUCLEAR CONCEPTS
- C. FUEL ROD CONCEPTS
- D. THERMAL AND HYDRAULIC CONCEPTS
- E. RESOURCE AND SERVICE REQUIREMENTS AND ENVIRONMENTAL CONSIDERATIONS.

II. CONCLUSIONS

Three prebreeder core concepts have been evaluated for backfit application in the B&W Standard-241 nuclear steam supply system (the reference system). These core concepts use fuel assemblies with a 19 by 19 rod array; the fuel assemblies would fit within the same space envelope as those of the reference core. The three prebreeder core concepts differ from each other only with respect to the fuel system and seed rod diameter. Core Concept 1 would contain enriched uranium seed rods and thorium blanket rods. Core Concept 2 would contain enriched uranium seed rods and thorium-plutonium blanket rods. Core Concept 3 would contain thorium blanket rods and thorium plus highly enriched uranium seed rods. All fuel systems would require minimal fuel rod development effort. These evaluations have led to the following conclusions:

1. Annual uranium-233 production of 161, 255 and 198 kg would be possible for core concepts 1, 2, and 3, respectively, with operation at the reference system power level of 3800 MWth. Operation of five prebreeders for 5 to 8 years would supply approximately 6200 kg of uranium-233, which would be sufficient to supply the in-core plus out-of-core fissile loading of a 1000 MWe light water-cooled breeder, assuming a one year out of core time for fuel reprocessing and refabrication.
2. For all prebreeder core concepts, limiting the power to 3230 MWth, 85 percent of the reference value, would permit a decrease in the number of seed rods and a corresponding increase in the number of blanket rods. This would increase the annual uranium-233 production to 204, 268 and 224 kg for core concepts 1, 2 and 3, respectively.

3. Annual uranium requirements for operation at 3800 MWth range from 9 percent greater to 17 percent less for the prebreeder core concepts than for the reference core, assuming the reference core recycles uranium only.
4. Annual separative work requirements for operation at 3800 MWth range from 8 to 23 percent greater for the prebreeder core concepts than for the reference core, again assuming that the reference core recycles uranium only.
5. All three prebreeder core concepts are calculated to have thermal-hydraulic and fuel element performance comparable with that of the reference core.
6. The control rod assemblies of the prebreeder core concepts would differ from those of the reference core. This would require redesign of the plenum upper assembly.
7. A developmental testing program as discussed in Section V would be required to confirm the nuclear and mechanical performance, the fuel irradiation performance, and the thermal and hydraulic performance along with the reflood characteristics of these prebreeder core concepts. Critical assembly experiments would be desirable to confirm reactivity predictions. It is anticipated that satisfactory completion of such a test program would permit licensing of these prebreeder core concepts.

III. RESULTS

A. Summary Description of Prebreeder Core Concepts

The backfit prebreeder core concepts described in this report would be based on modifications of the 17 by 17 fuel assembly used in the reference reactor system reported in Reference 1. Testing of this reference fuel assembly was initiated in 1972. By use of the reference mechanical design features to the maximum extent possible, the changes required to backfit a prebreeder core into the reference plant would be minimized. Additionally, the developmental testing program required to demonstrate structural adequacy and to fully qualify the mechanical design of the prebreeder core concepts would also be minimized. Much of the testing and analysis from the reference system would be applicable to the prebreeder core concepts.

Appendix A includes a description with illustrations that define conceptual mechanical arrangements of prebreeder core concepts for backfit application into the reference plant. The specific areas described are major components that would require a redesign, relative to the reference core, for the prebreeder core concepts. The major components would be the fuel assemblies, the control assemblies, and the plenum assembly. In addition, Appendix A identifies the major technical considerations due to the mechanical rearrangement, and the testing program which would be required prior to licensing the core concepts.

As shown in Figures 1 and 2 of this section, the number of fuel assemblies and overall fuel assembly size would be identical to those of the reference core. The prebreeder core concepts would be sized to fit the reference core support assembly including the core basket and lower grid assembly. In addition, the prebreeder core concepts would use the same in-core instrumentation arrangement

*THESE COMPONENTS WOULD BE NEW FOR THE BACKFIT APPLICATION ALL OTHERS WOULD BE EXISTING COMPONENTS

Figure 1. Prebreeder Reactor Assembly

- 1 FUEL ASSEMBLY
Number per core
Pitch, in
Cross section size, in
Length overall, in
- 2 VESSEL
Outside dia., in
Inside dia., in
Wall thickness, in
- 3 CORE SUPPORT ASSEMBLY
Outside dia., in
Inside dia., in

B & W SHOWN STD - 241	PRE BREEDER SHOWN FOR BACKFIT
241	241
8 587	8 587
8 536	8 536
176 6	176 6
214 5	214 5
195 0	195 0
9 75	9 75
173 5	173 5
167 5	167 5

Figure 2. Cross Section Through Reactor Vessel
Internals and Core

and reactor vessel head assembly, including the control rod drive system. The arrangement of rods in each fuel assembly would be changed from a 17 by 17 array as used in the reference core to a 19 by 19 array. Separate seed and blanket rods would be located on a generally alternating pattern. Each fuel assembly would contain 28 control rod guide tubes and one instrumentation tube assembly. The reference core uses 24 control rod guide tubes. All fuel assemblies within a given prebreeder would be identical in mechanical detail and would be mechanically interchangeable in any core location. The major components of the reference core that would require redesign for the prebreeder core concepts are the fuel assemblies, plenum assembly, control rod guide tubes, control rod assemblies, axial power shaping rod assemblies, burnable poison rod assemblies, and orifice rod assemblies.

The fuel assembly shown in Figure 3 would consist of 332 fuel rods, 28 control rod guide tubes, one instrumentation tube assembly, ten spacer grids, and two end fittings. The guide tubes, spacer grids, and end fittings would form the basic structure to support the seed and blanket rods in a 19 by 19 array. The guide tubes would provide a continuous guide for each control rod through the full length of travel; that is, the control rod could not be withdrawn from the guide tube during operation. The fuel rods would be supported within the spacer grids by contact points (springs and dimples) integral with the grid for each cell. The guide tubes would be rigidly attached to the upper and lower end fittings. A comparison of the fuel assembly components, materials, and dimensions of the prebreeder core concepts and the reference core is presented in Table 1.

Locations of seed and blanket rods within a fuel assembly for the three prebreeder core concepts are shown in Figure 4, Sheets 1, 2, and 3. Both seed and blanket fuel rods shown in Figure 5 would be internally pressurized with helium

Figure 3. Prebreeder Fuel Assembly

TABLE 1. FUEL ASSEMBLY MECHANICAL PARAMETERS

	<u>Reference Core</u>	<u>Prebreeder Core Concept</u>
Fuel Assembly (FA) Type	Canless with finger control	Canless with finger control
Number of FA's per Core	241	241
Pitch, inches	8.587	8.587
Control Rod Guide Tube Number/Material Size OD by ID, in.	24 Zircaloy-4 0.465 by 0.430	28 Zircaloy-4 0.426 by 0.391
Instrument Tube Number/Material Size OD by ID, in.	1 Zircaloy-4 0.420 by 0.390	1 Zircaloy-4 0.390 by 0.360
Instrument Tube Spacer Sleeve Number/Material Size OD by ID, in.	7 Zircaloy-4 per tube 0.480 by 0.428	9 Zircaloy-4 per tube 0.428 square by 0.397
Grids Number/Material Type of Construction	8 Inconel-718 TIG welded "egg-crate"	10 Inconel-718 TIG welded "egg-crate"
Fuel Rods, Number per FA	264	See Figure 4 for number of seed and blanket rods.
Cladding Material Dimensions OD by ID, in.	Zircaloy-4 0.379 by 0.332	Zircaloy-4 See Figure 5 for dimensions of seed and blanket rods.

to enhance fuel rod reliability by reducing fuel-cladding interaction and to provide improved heat transfer within the rods, which would result in lower fuel temperatures. The fuel would be in the form of sintered and ground pellets of

Figure 4, Sheet 1. Prebreeder Fuel Assembly 19 by 19 Rod Array, No Plutonium Recycle

Figure 4, Sheet 2. Prebreeder Fuel Assembly 19 by 19 Rod Array, Plutonium Recycle

Figure 4, Sheet 3. Prebreeder Fuel Assembly 19 by 19 Rod Array with Highly Enriched Urania, No Plutonium Recycle

Figure 5. Fuel Rods

low-enriched uranium dioxide or of thorium dioxide. In core Concept 2 the thorium dioxide would also contain plutonium dioxide, and in core Concept 3 the thorium dioxide seed rods would contain highly enriched uranium dioxide. As in some current commercial designs, pellet ends would be dished to minimize differential thermal expansion between the fuel and cladding and chamfered to minimize interaction with the cladding at fuel pellet interfaces.

Also as in current commercial designs, springs would be located above and below the fuel column to maintain the fuel column in place during shipping and handling and to permit axial differential growth and thermal expansion between the fuel and the cladding. Fuel fission gas release would be contained within voids in the pellets, the axial gap between pellets, the radial gap between the pellets and cladding, and the gas plenums at the ends of the fuel rods.

Typical of commercial cores, the spacer grids would be made from strips slotted and fitted together to form an "egg-crate" arrangement. The arrangement of strips would be permanently assembled by a tungsten-inert-gas (TIG) weld at the upper and lower surface at each intersection formed by the strips. The square cells formed by the strips would provide support for fuel rods in two perpendicular directions through contact points on each of the four cell walls: two springs on two adjacent walls, and four dimples on the two opposite walls, with different size springs and dimples for seed rods and blanket rods.

The instrumentation tube assembly would function to guide, position, and contain the in-core instrumentation guide thimble within the fuel assembly. The instrumentation tube would be located on the axial centerline of the fuel assembly. The spacer sleeves would fit around the instrumentation tube and would be located

between spacer grids. The spacer sleeves would provide a positive spacing between grids to prevent excessive axial movement of the grids due to primary coolant flow through the fuel assembly.

A control assembly would be located in each fuel assembly. Each fuel assembly would have either a moveable control rod assembly (CRA), a moveable axial power shaping rod assembly (APSRA), a fixed burnable poison rod assembly (BPRA), or a fixed orifice rod assembly (ORA). Because of mechanical and geometric identity, these control assemblies would be designed to be interchangeable among fuel assemblies. The reactivity of the prebreeder core concepts under operating conditions would be controlled by boric acid solution in the coolant, by 76 CRA's and by 8 APSRA's, (the same numbers of each as in the reference core). In the fuel assemblies which do not contain a CRA or APSRA, a fixed BPRA or fixed ORA would be used.

Each fuel assembly would be installed and seated on the lower grid of the core support assembly. Alignment would be achieved by index pins located on the lower grid engaging holes in the lower fitting of the fuel assembly. The fuel assemblies would be restrained axially by the plenum assembly located above the fuel assemblies. Index pins on the plenum assembly would align and provide lateral support at the upper end of each fuel assembly. The pins would also interact with the spring retainer plate located in the upper end fitting of the fuel assembly to load the helical springs, thereby providing a positive clamping force to each fuel assembly.

B. Comparison with Reference Core

The prebreeder core concepts were selected to backfit into the plant used for the reference core. A comparison of important parameters for these prebreeder core concepts and for the reference core is shown in Table 2.

TABLE 2. REACTOR COMPARISON

		Prebreeder Core Concepts		
		#1 UO ₂ Seed ThO ₂ Blanket	#2 UO ₂ Seed ThO ₂ -PuO ₂ Blanket	#3 UHEO ₂ -ThO ₂ Seed ThO ₂ Blanket
<u>Reference Reactor</u> ⁽¹⁾				
<u>Thermal and Hydraulic Parameters</u>				
Reactor Core Heat Output, MWth	3800	3800	3800	3800
Electrical Output, MWe (net) ⁽²⁾	1322	1322	1322	1322
Reactor Core Heat Output, 10 ⁶ Btu/hr	12,966	12,966	12,966	12,966
Heat Generated in Fuel Rod, %	97.3	97.5	97.4	97.5
System Pressure, Nominal, psia	2250	2250	2250	2250
System Pressure, Min. Steady-State, psia	2205*	2205	2205	2205
DNB Correlation	BAW-2	BAW-2	BAW-2	BAW-2
Minimum Allowable DNBR	1.32	1.32	1.32	1.32
Design Overpower, % Design Power	111 ⁽³⁾	114	117	112
Number of Pumps/RV Inlets/RV Outlets	4/4/2	4/4/2	4/4/2	4/4/2
<u>Coolant Flow</u>				
Total Thermal Flow Rate, 10 ⁶ lb/hr	159.0	154.0	154.2	153.5

TABLE 2 (Cont)

		Prebreeder Core Concepts		
		#1 U ²³⁵ O ₂ Seed ThO ₂ Blanket	#2 U ²³⁵ O ₂ Seed ThO ₂ -PuO ₂ Blanket	#3 U ²³⁵ O ₂ -ThO ₂ Seed ThO ₂ Blanket
<u>Reference Reactor</u> ⁽¹⁾				
<u>Thermal and Hydraulic Parameters (Cont)</u>				
<u>Coolant Flow (Cont)</u>				
Effective Flow Rate for Heat Transfer, 10 ⁶ lb/hr	150.3	145.6	145.8	145.1
Effective Flow Area for Heat Transfer, ft ²	66.4	61.0	61.1	60.2
Average Velocity Along Fuel Rods, ft/sec	14.6	15.4	15.4	15.6
Average Mass Velocity, 10 ⁶ lb/hr-ft ²	2.26*	2.39	2.39	2.41
Core Unrecoverable Pressure Drop, psia	11.6	18.6	18.6	19.5
Pump Head, ft	316	~326	~326	~326
Power/Flow, MWt/10 ⁶ lb/hr	23.9	24.7	24.6	24.8
<u>Coolant Temperature</u>				
Nominal Inlet, °F	573.5	572.9	572.9	572.9
Average Rise in Vessel, °F	56.2	57.9	57.9	58.1
Average Rise in Core, °F	59.0	60.8	60.6	60.9
Average in Core, °F	603.0	603.3	603.2	603.3
Average in Vessel, °F	601.6	601.9	601.9	601.9
Nominal outlet Temp, °F	629.7	630.8	630.8	631.0

TABLE 2 (Cont)

		Prebreeder Core Concepts		
		#1 UO ₂ Seed ThO ₂ Blanket	#2 UO ₂ Seed ThO ₂ -PuO ₂ Blanket	#3 UHEO ₂ -ThO ₂ Seed ThO ₂ Blanket
<u>Reference Reactor⁽¹⁾</u>				
<u>Thermal and Hydraulic Parameters (Cont)</u>				
<u>Heat Transfer</u>				
Active Heat Transfer, Surface Area, ft ²	75,229	89,012	88,831	89,643
Seed Rods	-	65,076	48,937	57,728
Blanket Rods	-	23,936	39,894	31,915
Average Heat Flux, Btu/hr-ft ²	167,700	142,000	142,200	141,000
Seed Rods	-	177,700	187,800	194,900
Blanket Rods	-	44,900	86,100	43,700
Maximum Heat Flux for Normal Operation, Btu/hr-ft ²	445,000	-	-	-
Seed Rods	-	449,800	436,600	488,600
Blanket Rods	-	150,900	231,700	158,100
Average Thermal Output, kw/ft	4.88	3.89	3.88	3.89
Seed Rods	-	4.76	4.89	5.22
Blanket Rods	-	1.31	2.51	1.27
Maximum Thermal Output for Normal Operation, kw/ft	12.95*	-	-	-
Seed Rods	-	12.3	11.9	13.4
Blanket Rods	-	4.1	6.3	4.4
Maximum Thermal Output at 112% Power, kw/ft	14.5	-	-	-
Seed Rods	-	13.7	13.3	15.0
Blanket Rods	-	4.6	7.1	4.9

TABLE 2 (Cont)

	Prebreeder Core Concepts			
	<u>Reference Reactor⁽¹⁾</u>	<u>#1 UO₂ Seed ThO₂ Blanket</u>	<u>#2 UO₂ Seed ThO₂-PuO₂ Blanket</u>	<u>#3 UO₂-ThO₂ Seed ThO₂ Blanket</u>
<u>Thermal and Hydraulic Parameters (Cont)</u>				
<u>Heat Transfer (Cont)</u>				
Nuclear Heat Flux Power	2.59	3.09	3.00	3.38
Peaking Factor				
Nuclear Enthalpy Rise	1.55	1.90	1.87	2.03
Power Peaking Factor				
Engineering Heat Flux Hot Channel Factor	1.025	1.025	1.025	1.025
Engineering Enthalpy Rise Hot Channel Factor	1.014	1.014	1.014	1.014
<u>Fuel Central Temperature</u>				
Peak at 100% Power, °F	3560	3340	3280	3420
Peak at 112% Power, °F	3850*	3649	3592	3748
<u>Core Mechanical Parameters</u>				
<u>Fuel Assemblies</u>				
Design	Canless with finger control	Canless with finger control	Canless with finger control	Canless with finger control
Number of Fuel Assemblies	241	241	241	241
UO ₂ Rods per Assembly	264	248	192	220
Blanket Rods per Assembly	-	84	140	112
Rod Pitch, in.	0.501	0.449	0.449	0.449

TABLE 2 (Cont)

		Prebreeder Core Concepts		
		#1 UO ₂ Seed ThO ₂ Blanket	#2 UO ₂ Seed ThO ₂ -PuO ₂ Blanket	#3 UHEO ₂ -ThO ₂ Seed ThO ₂ Blanket
<u>Reference Reactor</u> ⁽¹⁾				
<u>Core Mechanical Parameters (Cont)</u>				
<u>Fuel Assemblies (Cont)</u>				
Rod-Rod Spacing, in.	0.122*	-	-	-
Seed-Seed	-	0.100	0.110	0.100
Seed-Blanket	-	0.085	0.090	0.085
Seed-Control Rod Guide Tube	0.079*	0.0615	0.0665	0.0615
Fuel Assembly Pitch, inches	8.587	8.587	8.587	8.587
Number of Grids per Assembly	8	10	10	10
Grids				
Lattice Material	Square Inconel-718 strips	Square Inconel-718 strips	Square Inconel-718 strips	Square Inconel-718 strips
Type of Construction	Welded "egg-crate"	Welded "egg-crate"	Welded "egg-crate"	Welded "egg-crate"
Type of Spring	Leaf with both ends fixed	Leaf with both ends fixed	Leaf with both ends fixed	Leaf with both ends fixed
<u>Fuel Rods</u>				
Number	63,624	80,012	80,012	80,012
Seed Rods	-	59,768	46,272	53,020
Blanket Rods	-	20,244	33,740	26,992
Outside Diameter, in.	0.379	-	-	-
Seed Rods	-	0.349	0.339	0.349
Blanket Rods	-	0.379	0.379	0.379

TABLE 2 (Cont)

		Prebreeder Core Concepts		
		#1 UO ₂ Seed ThO ₂ Blanket	#2 UO ₂ Seed ThO ₂ -PuO ₂ Blanket	#3 UO ₂ -ThO ₂ Seed ThO ₂ Blanket
<u>Reference Reactor (1)</u>				
<u>Core Mechanical Parameters (Cont)</u>				
<u>Fuel Rods (Cont)</u>				
Diametral Gap, in.	0.008	-	-	-
Seed Rods	-	0.007	0.007	0.007
Blanket Rods	-	0.004	0.004	0.004
Clad Thickness, in.	0.0235	-	-	-
Seed Rods	-	0.0235	0.0235	0.0235
Blanket Rods	-	0.020	0.020	0.020
Clad Material	Cold-worked Zircaloy-4	Annealed Zircaloy-4	Annealed Zircaloy-4	Annealed Zircaloy-4
<u>Fuel Pellets</u>				
Material	UO ₂	-	-	-
Seed Rods	-	UO ₂	UO ₂	UO ₂ + ThO ₂
Blanket Rods	-	ThO ₂	ThO ₂ + PuO ₂	ThO ₂
Density, % of Theoretical	93.5	-	-	-
Seed Pellets	-	94.5	94.5	98.0
Blanket Pellets	-	98.0	98.0	98.0
Diameter, in.	0.324	-	-	-
Seed Rods	-	0.295	0.285	0.295
Blanket Rods	-	0.335	0.335	0.335
Length, in.	0.375	-	-	-
Seed Rods	-	0.375	0.375	0.375
Blanket Rods	-	0.375	0.375	0.375

TABLE 2 (Cont)

		Prebreeder Core Concepts		
		#1 UO ₂ Seed ThO ₂ Blanket	#2 UO ₂ Seed ThO ₂ -PuO ₂ Blanket	#3 UHEO ₂ -ThO ₂ Seed ThO ₂ Blanket
	Reference Reactor ⁽¹⁾			
<u>Core Mechanical Parameters (Cont)</u>				
<u>Control Rod Assemblies</u>				
Neutron Absorber Cladding Material	Ag-In-Cd Cold-worked 304 SS	Ag-In-Cd Cold-worked 304 SS	Ag-In-Cd Cold-worked 304 SS	Ag-In-Cd Cold-worked 304 SS
Clad Thickness, in.	0.019	0.017	0.017	0.017
Number of Clusters, Full/Part Length	76/8	76/8	76/8	76/8
Number of Absorber Rods per Assembly	24	28	28	28
Absorber Rod Diameter, in.	0.390	0.351	0.351	0.351
<u>Core Structure</u>				
Core Support Assembly ID/OD, inches	167.5/173.5	167.5/173.5	167.5/173.5	167.5/173.5
Pressure Vessel ID/OD, inches	195/214.5	195/214.5	195/214.5	195/214.5
<u>Nuclear Parameters</u>				
<u>Nuclear Characteristics</u>				
Core Diameter, inches (equivalent)	150.4	150.4	150.4	150.4
Core Average Active Fuel Height, inches	143	143	143	143

TABLE 2 (Cont)

		Prebreeder Core Concepts		
		#1 U ^E O ₂ Seed ThO ₂ Blanket	#2 U ^E O ₂ Seed ThO ₂ -PuO ₂ Blanket	#3 U ^H E ₂ O ₂ -ThO ₂ Seed ThO ₂ Blanket
	Reference Reactor ⁽¹⁾			
<u>Nuclear Parameters (Cont)</u>				
<u>Nuclear Characteristics (Cont)</u>				
H ₂ O/Fuel, Molecular Ratio Fuel Assembly (Cold) ⁽⁴⁾	2.92	2.44	2.42	2.39
Fuel-to-Coolant Volume Ratio	0.475	0.573	0.585	0.609
<u>Core Fuel Enrichment in Fresh Fuel, w/o</u>				
	3.13 ⁽⁶⁾	5.10	6.65	49.5
	Average 2.79 ⁽⁷⁾			
<u>U₃O₈⁽⁵⁾ Requirements</u>				
With fuel Recycle ST/year	209 (U recycle)	228 (U recycle)	195 (U, Pu recycle)	175 (U recycle)
<u>Separative Work Requirements</u>				
MT SWU/year ⁽⁸⁾	160 ⁽⁹⁾ (U recycle)	197 (U recycle)	175 (U, Pu recycle)	172 (U recycle)
<u>U-233 Production</u>				
kg/year after 1 percent total losses (1/2 percent to reprocessing and 1/2 percent to fabrication)	-	161	255	198

TABLE 2 (Cont)

		Prebreeder Core Concepts		
		#1	#2	#3
		UO ₂ Seed	UO ₂ Seed	UHEO ₂ -ThO ₂ Seed
		ThO ₂ Blanket	ThO ₂ -PuO ₂ Blanket	ThO ₂ Blanket
<u>Reference Reactor</u> ⁽¹⁾				
<u>Nuclear Parameters (Cont)</u>				
Seed Rod Peak	10.2	11.9	9.4	11.4
Depletion; EOL, 10 ²⁰ f/cc				

*Value inferred from Reference 1.

- (1) Values were obtained from Reference 1. U₃O₈ and separative work requirements were obtained from Bettis calculations.
- (2) Will vary slightly, depending upon particular plant.
- (3) Calculated for this report. See Sections I and II of Appendix D for explanation.
- (4) Fuel is total uranium, thorium, and plutonium.
- (5) Assumes 0.8 load factor, 0.2-percent enrichment tails. Resource requirements are discussed in Section III.C.5.
- (6) Calculated for this report; not stated in Reference 1.
- (7) First core value cited in Reference 1.
- (8) SWU = Separative Work Units; see Section III.C of Appendix E.
- (9) Calculated for this report; see Section II.B.3 of Appendix E.

Nominal power and pressure of 3800 MWth and 2250 psia, the same as those quoted for the reference core, were assumed for the prebreeder core concepts. In developing these core concepts, one objective was to achieve a calculated CHF (Critical Heat Flux) steady state overpower capability at least as high as that calculated for the reference core, using the same method of calculation for all cores. This objective was met for all three prebreeder core concepts. Another objective was to achieve calculated fuel rod performance comparable with that calculated for the reference core, again using the same method of calculation for all cores. This objective was also met for all three prebreeder core concepts.

Relative to the reference core, the prebreeder core concepts would have the following hydraulic characteristics: 8 to 10 percent lower flow area (60 to 61 ft² versus 66.4 ft²) and more spacer grids (10 versus 8) resulting in 60 to 70 percent higher unrecoverable pressure drop; about 3 percent lower flow, resulting in 3 percent higher power-to-flow ratio; 5 to 7 percent higher fluid linear velocity and mass velocity; and the main coolant pumps would operate at about a 3 percent higher head. The differing flow areas in the three prebreeder core concepts are the result of differing rod diameters and relative numbers of seed and blanket rods (with number of seed-plus-blanket rods being the same for all prebreeder core concepts).

The average core temperature in the prebreeder core concepts would be the same as that in the reference core. Since the flow rate would be slightly lower relative to values for the reference core, the nominal inlet temperature would be slightly lower and the nominal exit temperature slightly higher (1°F). These values would be within the normal range of operation of the reference core.

During steady state operation, there would be significant differences between the prebreeder core concepts and the reference core in peak heat flux, peak thermal output (kw/ft of fuel rod) and in peak fuel centerline temperature. These peak values would be important with respect to steady state thermal capability, fuel rod integrity and LOCA performance. In the prebreeder core concepts, all of these peak values would occur in the seed rods. Relative to values for the reference core, the peak heat flux would be about the same for core Concepts 1 and 2, and would be about 10 percent greater for core Concept 3; the maximum thermal output would be 5 to 8 percent lower for core Concepts 1 and 2 and would be about 3 percent higher for core Concept 3; the peak fuel temperatures would be lower and the margin to melting would be higher for all three core concepts. The calculated peak fuel temperature and margin to melting for core Concept 3 are functions of the assumed thermal conductivity and melting temperature of thorium as discussed in Section IV.C of Appendix C.

The fuel assemblies in the prebreeder core concepts would be similar to those in the reference core. The number of fuel assemblies would be the same as in the reference core and they would be located on the same pitch. The fuel assemblies would contain: 248 uranium rods and 84 thorium rods for core Concept 1, 192 uranium rods and 140 plutonium-thorium rods for core Concept 2, and 220 uranium-thorium rods and 112 thorium rods for core Concept 3. These values compare with 264 uranium rods in the reference core. Because of the increased number of rods in the same space, the rod pitch would be reduced from 0.501 inch to 0.449 inch. The nominal minimum rod spacing in the prebreeder core concepts would be smaller than the nominal rod-to-rod spacing in the reference core (0.085 inch for core Concepts 1 and 3 and 0.090 inch for core Concept 2 versus 0.122 inch for the reference core). The seed rod diameter for core Concept 2 would be 10 mils less than for core Concepts 1

and 3 (with a corresponding 5-mil increase in seed-to-blanket rod spacing). The number of grids in the prebreeder would be increased to ten (from eight in the reference core) to provide the added support required to maintain adequate clearances with the smaller diameter seed rods and the more closely spaced fuel rods. The prebreeder core concept grids were assumed to be mechanically similar to the reference grid.

The number of fuel rods in the prebreeder core concepts would be 26 percent greater than in the reference core. The seed rods would be smaller in diameter (0.339 and 0.349 inch versus 0.379 inch) in order to obtain acceptable thermal performance and rod spacing. The blanket rods would be the same diameter as the reference core rods since this would be about the largest rod which could be accommodated without reducing rod spacings below acceptable limits. The diametral fuel-to-clad gap in the prebreeder rods would be slightly less than in the reference core. The fuel rod cladding assumed in the prebreeder core concepts is recrystallized annealed (RXA) Zircaloy-4, chosen because of the lower creep rate relative to cold-worked (CW) Zircaloy-4 used in the reference core. The seed rod clad thickness would be the same as in the reference core while the blanket rod clad thickness would be 0.0035 inch less. The fuel pellets in the rods would have dimensions comparable to those in the reference core. The thorium pellets would have a slightly higher percent of theoretical density than the uranium pellets.

The control rod assemblies in the prebreeder core concepts would be in the same locations as those in the reference core. The number of control rods per assembly would be increased from 24 to 28 to offset the reduction in control rod effectiveness resulting from the smaller diameter (0.351 inch versus 0.390 inch)

dictated by the increased number of rods in the assembly. The prebreeder core concepts would use the same pressure vessel, thermal shield, core barrel, and filler pieces as the reference system.

The prebreeder core concepts would have average enrichments greater than the average enrichment of the reference core, they would be refueled annually in one-third core increments as is the reference core, and fresh fuel assemblies would be located at the periphery of the core. The annual new fuel and separative work requirements are listed in Table 2. More details on the resource requirements are given in Section C.5.

C. Performance Estimates

The sections below provide estimates of the nuclear, fuel, thermal, and hydraulic performance of these prebreeder core concepts and the resource requirements and environmental impacts associated with each.

1. Nuclear Performance

Nuclear performance was estimated by four-group diffusion theory depletion calculations using the PDQ08 computer code (Reference 4). Few group diffusion theory nuclear cross sections were generated by Monte Carlo calculations using the RCP01 computer code (Reference 5). A detailed description of the nuclear analysis is given in Appendix B.

The RCP01 model represented a radial slice of a quarter fuel assembly. The fuel rods, control rods, guide tubes, instrument tube, and spacer sleeve were explicitly represented in the RCP01 model for each core.

Mass flow, radial power distributions, reactivity coefficients, and shutdown margin were estimated from quarter-core radial slice PDQ08 calculations. The PDQ08 model represented homogenized quarter fuel assemblies in the quarter core.

Depletion calculations were run at the full power rating of 3800 MWth and utilized fuel management strategies to reduce radial power peaking. A capacity factor of 0.8 was obtained by assuming a 0.2-year shutdown each year. All fuel management calculations were made for annual refuelings of one-third core increments with unirradiated fuel in fuel management Zone 1, and once-irradiated and twice-irradiated fuel in Zones 2 and 3, respectively. These zones are defined in Section III.A of Appendix B and are identifiable from Tables B-10, B-16, B-17 and B-18. Local radial peaking factors were obtained from cell calculations and applied to the quarter-core results. Axial power peaking factors were estimated from three-dimensional, quarter-core calculations performed for a Westinghouse PWR core (Reference 6). Grid material was homogenized into the moderator to give the impact on core reactivity but not on local power peaking. A 10-percent increase was applied to the calculated axial power peaking to account for the effect of grids and for temperature feedback. Axial factors were determined based on the Westinghouse core rather than the reference core because these calculations were available from earlier work. Since the Westinghouse core has nearly the same active core length (144 inches versus 143 inches for the reference core) and a fuel assembly design similar to that of the reference core, it was judged not to be necessary to repeat the three-dimensional calculations.

a. Mass Flows - Tables 3 through 5 present mass flows for the prebreeder core concepts for the case of one-third core annual refueling, full recycle of uranium, and one year out of core for reprocessing plus refabrication. These mass flows would be representative of the fresh fuel input at the start, and the spent fuel discharged at the end, of the fifteenth year of operation. This would be

TABLE 3. MASS FLOWS FOR YEAR 15 OF PREBREEDER CORE CONCEPT 1

<u>Nuclide</u>	<u>Rod Type</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
Th-232	Blanket	11,851.0	11,570.2
U-232	Blanket	0.0	0.57
U-233	Blanket	0.0	162.6
U-234	Blanket	0.0	15.5
U-235	Seed	1463.8	574.3
U-236	Seed	326.0	396.3
U-238	Seed	26,888.8	26,247.0
Pu-239	Seed	0.0	191.7
Pu-240	Seed	0.0	54.9
Pu-241	Seed	0.0	40.9
Pu-242	Seed	0.0	8.8
Total Fissile	Seed and Blanket	1463.8	969.5
Fissile Pu	Seed	0.0	232.6
Total Fertile	Seed and Blanket	38,739.8	37,887.6

approximately the midpoint in the life of a nuclear power plant, and at this point the uranium-236 from recycled uranium would be close to an equilibrium level because of uranium-236 removal during re-enrichment.

The presence of uranium-236 in recycle uranium would have a negligible effect on uranium-233 production. For prebreeder core Concept 2, the calculated annual uranium-233 production increased by 1.5 percent (4 kg/year) when uranium-236 was removed completely from the fresh fuel in all of the annual refuelings.

TABLE 4. MASS FLOWS FOR YEAR 15 OF PREBREEDER CORE CONCEPT 2

<u>Nuclide</u>	<u>Rod Type</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
Th-232	Blanket	19,370.1	18,950.1
U-232	Blanket	0.0	1.01
U-233	Blanket	0.0	257.6
U-234	Blanket	0.0	21.6
U-235	Seed	1377.5	618.6
U-236	Seed	335.7	388.0
U-238	Seed	19,001.0	18,534.6
Pu-239	Seed	0.0	152.2
Pu-240	Seed	0.0	43.5
Pu-241	Seed	0.0	28.3
Pu-242	Seed	0.0	5.3
B1 Pu-239	Blanket	150.5	12.2
B1 Pu-240	Blanket	42.4	14.5
B1 Pu-241	Blanket	26.6	25.8
B1 Pu-242	Blanket	5.1	17.0
B2 Pu-239	Blanket	12.5	1.00
B2 Pu-240	Blanket	14.6	2.08
B2 Pu-241	Blanket	23.8	6.19
B2 Pu-242	Blanket	16.1	14.8
Fissile Pu	Seed and Blanket	213.4	225.7 (7.2 to be sent to waste manage- ment)*

TABLE 4 (Cont)

<u>Nuclide</u>	<u>Rod Type</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
Total Fissile	Seed and Blanket	1590.9	1101.9
Total Pu	Seed and Blanket	291.6	322.9 (24.1 to be sent to waste manage- ment)*
Total Fertile	Seed and Blanket	38,428.2	37,568.4

B1 - Plutonium which is being recycled for the first time.

B2 - Plutonium which is being recycled for the second time.

*After the second recycle plutonium would have low fissile content and would be unsuitable for further use in prebreeder core concept.

Prebreeder core Concept 1 would require an annual input of 1464 kg uranium-235 to produce 163 kg uranium-233. Assuming 1-percent total losses for reprocessing and fabrication of breeder fuel rods, the net production rate would be 161 kg of uranium-233 per year. In addition to the 163 kg uranium-233, the fuel assemblies removed from the core at the end of an annual cycle would contain 574 kg uranium-235, 233 kg of fissile plutonium, and 396 kg of uranium-236.

In prebreeder core Concept 2, plutonium would be recycled into the blanket rods to improve performance. The nuclides in Table 4 with identifications preceded by B1 (for example, B1 Pu-239) would have been recycled from the seed rods of a previous cycle into the blanket rods. The nuclides with identifications preceded by B2 would have been recycled from the blanket rods of a previous cycle back into blanket rods a second time. At the end of the second cycle of irradiation of the recycled plutonium, the remaining plutonium, which would be heavily contaminated by plutonium-242, would be sent to waste management. Prebreeder core

TABLE 5. MASS FLOWS FOR YEAR 15 OF PREBREEDER CORE CONCEPT 3

<u>Nuclide</u>	<u>Rod Type</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
Th-232	Seed	21,656.5	21,271.1
U-232	Seed	2.50	2.76
U-233	Seed	407.6	416.5
U-234	Seed	107.5	108.3
U-235	Seed	1154.7	474.9
U-236	Seed	431.2	435.3
U-238	Seed	228.7	178.2
Pu-239	Seed	0.0	8.0
Pu-240	Seed	0.0	2.0
Pu-241	Seed	0.0	2.4
Pu-242	Seed	0.0	0.6
Th-232	Blanket	15,801.4	15,450.7
U-232	Blanket	0.0	0.82
U-233	Blanket	0.0	199.6
U-234	Blanket	0.0	19.0
U-235	Blanket	0.0	2.7
U-236	Blanket	0.0	0.2
Seed Fissile		1562.3	901.8
Total Fissile		1562.3	1104.1
Total Fertile		37,794.1	37,029.3

Concept 2 would require an input fissile loading of 1378 kg uranium-235 and 213 kg fissile plutonium to produce 258 kg (255 kg net) of uranium-233 per year. Each year, at the end of a depletion cycle, 619 kg of uranium-235 and 226 kg of fissile plutonium (of which 219 kg would be recycled) would remain in the fuel assemblies to be removed from the core that year.

For prebreeder core Concept 3, as with prebreeder core Concept 1, unirradiated blanket rod fuel would not be spiked with fissile material. Prebreeder core Concept 3 would require an input fissile loading of 1155 kg uranium-235 and 408 kg of recycled uranium-233 per year to produce 200 kg (198 kg net) of breeder-grade uranium-233 per year from the blanket rods. Since highly enriched (93 percent) uranium would be used to fabricate the fuel in the seed rods, very little fissile plutonium would be produced by this prebreeder core concept.

b. Best Estimate Power Peaking - Best estimate radial and axial power peaking factors were determined by diffusion theory depletion calculations. The maximum radial power peaking factor is defined in Appendix B, Section III.A as the ratio of the radial peak rod power to core radial average (over seed and blanket rods) rod power.

The maximum best estimate radial peaking factors for prebreeder core Concept 1 would be 1.62 for the seed rods, occurring at 1000 to 3000 EFPH in Zone 1, and 0.65 for the blanket rods, occurring at the end of the cycle (EOC) in Zone 3. During depletion there would be a gradual shift in power from the seed rods to the blanket rods as uranium-233 would be produced in the blanket rods and uranium would be depleted from the seed rods.

The maximum best estimate radial peaking factors for prebreeder core Concept 2 would be 1.61 for the seed and 0.82 for the blanket. The seed rod peak

would occur at 1000 EFPH in Zone 2 while the blanket rod peak would occur at the beginning of a cycle (BOC) in Zone 1. Since the blanket rods would be spiked with plutonium in the newly loaded fuel assemblies, there would be only a slight shift in power from the seed rods to the blanket rods during depletion.

The maximum best estimate radial peaking factors for prebreeder core Concept 3 would be 1.75 for the seed rods, occurring at BOC in Zone 1, and 0.68 for the blanket rods, occurring at EOC in Zone 3. The power would shift gradually with operation from the seed rods to the blanket rods as observed in prebreeder core Concept 1.

The maximum axial power peaking factor from the calculation on the Westinghouse PWR core was 1.32 at BOC in Zone 1 fuel assemblies. With a 10-percent factor to account for grid and temperature feedback effects, the maximum best estimate axial factor would be 1.45. The calculated axial peaking factor decreased steadily within each management zone with depletion. At any time in core life the maximum total peaking factor in each fuel management zone was taken as the product of the maximum axial and radial factors for that zone at that time in life. This may introduce a small but unknown conservatism since the maximum axial and radial factors may not occur at the same spatial location.

c. Moderator Temperature Coefficients and Doppler Deficits - Moderator temperature coefficients were calculated for the three prebreeder core concepts using a moderator temperature change from 601°F (average hot operating temperature) to 515°F (minimum temperature in a steam line break accident). Calculations were made at 1000 EFPH intervals throughout the fifteenth year of operation. As noted in Subsection a above, the uranium-236 would be close to an equilibrium level by this cycle. Negative moderator temperature coefficients were obtained for all

three prebreeder core concepts throughout this cycle. In addition, moderator temperature coefficients were estimated for the three prebreeder core concepts and for the reference core using $\pm 10^{\circ}\text{F}$ variations from the average operating temperature of 601°F . The results of these calculations indicated that the coefficients for core Concepts 1 and 2 would be slightly more negative and for core Concept 3 would be slightly less negative throughout the cycle than those for the reference core.

Doppler deficits for the prebreeder core concepts were calculated at intervals during the fifteenth year of operation using a power change from 100 percent to 0 percent. The Doppler deficits for core Concepts 1 and 3 varied through the cycle from 1.2 to 1.7 percent $\Delta\rho$ and from 1.0 to 1.1 percent $\Delta\rho$, respectively. Core Concept 2 had a constant 1.4 percent Doppler deficit. For comparison, Reference 1 (Section 4.3) reports a Doppler deficit for the reference core of 1.1 percent $\Delta\rho$ at beginning of life and 1.3 percent $\Delta\rho$ at the end of cycle 1. It is concluded that there is no substantial difference between the Doppler deficit values going from zero to full power in the reference core and the prebreeder core concepts.

d. Reactivity Shutdown Margins - Excess reactivity in a commercial PWR core is controlled by the use of soluble boron in the primary coolant and control rod assemblies interspersed throughout the core. Below some minimum hot shutdown temperature (556°F for the reference core), soluble boron is added to the primary coolant to ensure subcriticality. Control rod assemblies are used to ensure that sufficient negative reactivity is rapidly available in the core hot operating condition and under various accident conditions. To ensure that the prebreeder

core concepts would have acceptable shutdown reactivity at 556°F relative to the reference core, shutdown calculations were performed for all three prebreeder core concepts and for the reference core at intervals during the fifteenth year of operation. Results of these calculations presented in Appendix B indicate that at 556°F all prebreeder core concepts would be less reactive or have more shutdown margin, than the reference core.

Calculations of shutdown margin were also made at 515°F for all four cores. As discussed in Appendix B, this temperature can occur following a steam line break. These calculations also indicate that the prebreeder core concepts would have more shutdown margin than the reference core.

As discussed in Appendix B, the greater shutdown margins in the prebreeder core concepts are attributable to relative number, distribution, and surface area of the control rods.

2. Fuel Rod Performance

The performance of the prebreeder core concepts was evaluated with the procedures described in Appendix C. For each prebreeder core concept, the seed rod was selected for detailed study since the blanket rod would be expected to be significantly less limiting by virtue of the higher fuel melting point, lower fission gas release, and smaller linear heat rate. Analyses were performed using the CYGRO computer program (Reference 7) and the procedures qualified for the Shippingport Light Water Breeder Reactor, with the exception of gas release and thermal conductivity, which were modeled after the procedures used in Reference 1. Calculations were made using the maximum condition fuel rod power histories which are shown in Figure 6. The depletion dependent maximum condition axial and radial power peaks used to determine these fuel rod power histories are discussed in

Figure 6. Seed Rod Power History

Sections II.C and III.B of Appendix D. The fuel rods were assumed to deplete at an average of 90-percent power and to experience up-power transients at the time in life when the highest power peaks occur and also at end of fuel rod life.

To determine the capabilities of a prebreeder core concept, the performance was compared to performance indicators calculated for the reference core. The performance indicators selected are considered the most significant for initial concept selection. Included in this group are pellet average and pellet maximum temperature and clad stresses and strains. Performance indicators for the reference core were also obtained with the procedures described in Appendix C, using fuel rod data from Reference 1 and a fuel rod power history developed from rod peaking factors obtained from the nuclear analysis of the reference core performed by the authors. These performance indicators for the reference core, as calculated by the authors, constitute standards to which the performance indicators of the prebreeder concepts are compared. This technique is considered to provide a reasonable performance assessment of fuel rods for purposes of evaluating backfit concepts.

Peak values of the performance indicators calculated using the CYGRO program are compared in Table 6. (Some of the reference core values in Table 6, which are from the performance calculations made by the authors, differ slightly from the reference core values in Table 2 which were obtained directly or inferred from data in Reference 1.) The values of center temperature can be used to compare the proximity to melting, while maximum fuel pellet average temperature is an index of the ability of a concept to meet the clad temperature and oxidation limits for a loss-of-coolant accident (LOCA). The clad stresses listed in Table 6 are measures of transient performance capability related to stress corrosion

TABLE 6. FUEL ROD PERFORMANCE COMPARISONS

Parameter	Reference Core	Prebreeder Core Concept 1	Prebreeder Core Concept 2	Prebreeder Core Concept 3
Center Temperature, °F, 112% Power (BOL)	3808	3649	3592	3748
Fuel Average Temperature, °F, 100% Power (BOL)	2143	2080	2074	2160
Clad Circumferential Stress, psi, 100% Power (EOL)	20,000	24,100	-3100*	18,500
Clad Generalized Stress, psi, 112% Power (EOL)	24,100	27,600	**	21,000
Clad Axial Strain, % (EOL)	1.77	0.64	0.50	0.60
Clad Plastic Radial Strain	~0.0	~0.0	~0.0	~0.0
Fuel Pellet-to-Clad Diametral Gap, in.	0.008	0.007	0.007	0.007
Clad Material	70% CW	RXA	RXA	RXA
Max. Design Power, kw/ft	12.8	12.3	11.9	13.4
Peak Rod Depletion, 10^{20} f/cc	10.2	11.9	9.4	11.4

*Gap exists between fuel and clad

**Fuel clad interaction initiated but stress is <1000 psi.

cracking and clad strain instability performance. Peak rod depletion is not considered as a specific performance indicator, but rod depletion is part of the input into the performance calculations and in this way depletion differences are accounted for.

Except for the clad stress in core Concept 1, the prebreeder core concepts were calculated to have better performance indicators than the reference core. Prebreeder core Concept 1 would have higher stresses but a lower center and average temperature. Core Concept 3 would have significantly more margin to melting (based on center temperature) than the reference core since the melting point of the uranium dioxide-thorium dioxide fuel would be $\sim 700^{\circ}\text{F}$ higher than that of the uranium dioxide fuel in the reference core. The fuel average temperature for prebreeder core Concept 3 would be slightly higher (17 degrees) than for the reference core, but it is judged that it would be acceptable. Alternatives such as changes in the pellet-to-clad gap or higher density fuel in combination with using cold worked (CW) Zircaloy cladding could be considered to improve the clad performance and if necessary to reduce clad temperatures and increase the margin to the cladding temperature and oxidation limits during a LOCA. In the case of core Concept 1, if the thermal margin were traded off for improvements in clad stress by increasing the fuel-to-clad initial diametral gap from 7 to 8 mils, the clad stresses could be reduced to levels comparable to those for the reference core.

The lifetime variations of the peak pellet center temperature at steady-state power (90-percent power) for prebreeder core Concept 1 and the reference core as calculated by the authors are shown in Figure 7. The lifetime maximum would occur for both cores at beginning of cycle (0 EFPH). The prebreeder rod would have a lower lifetime maximum, but at times would operate at somewhat higher temperatures.

Figure 7. Pellet Temperature History, No Plutonium Recycle (Core Concept 1)

The comparison is similar for prebreeder core Concepts 2 and 3 and is presented in Appendix C. For performance criteria that are time and temperature dependent, prebreeder core Concepts 1 and 2 could have less margin than the reference core. The significantly higher melting point of the uranium dioxide-thorium dioxide fuel system in prebreeder core Concept 3 would be expected to improve the overall fuel pellet attributes and compensate for higher temperatures throughout life.

Comparative profiles of the lifetime variation in clad stresses for prebreeder core Concept 1 and for the reference core are shown in Figure 8. Included are the steady-state stresses and the stress levels for the up-power transients. Throughout the first 90 percent of operating history, the prebreeder clad stresses (recrystallized annealed (RXA) Zircaloy) would be below those for the reference core (cold worked (CW) Zircaloy) because of differences in the creep characteristics of the materials. The prebreeder clad-pellet interaction would occur later in life but, because of the differences in clad creep rate, the clad stresses would reach a higher level. In core Concept 2, fuel clad interaction would not occur by the time the fuel assembly was removed for reprocessing and in core Concept 3, the clad stresses would be in the transition region and below the reference core levels. The stresses for core Concepts 2 and 3 are presented in Appendix C.

With the proper selection of fuel rod parameters, prebreeder core concepts could be developed using annealed Zircaloy cladding in which the clad circumferential stress (an indicator of the susceptibility to iodine stress corrosion cracking) and the generalized stress (a clad plastic stability performance indicator) would be calculated to be below the reference core values. Because of the

Figure 8. Clad Stress History, No Plutonium Recycle (Core Concept 1)

lower creep rate with annealed material, the clad axial strain would be much lower, thereby reducing the potential both for axial gaps to develop and for short tube collapse to occur. In the prebreeder core concepts, the option would exist to use either annealed or cold worked cladding. The final selection would be made factoring in the latest operational and test results.

Additional considerations not explicitly evaluated here are discussed in Appendix C. These considerations include rod grid spring follow, plenum details, corrosion, and rod bowing. By using engineering approximations and comparative engineering it was judged that, for the prebreeder core concepts being considered, the performance in these areas would be comparable with that in the reference core.

Thus, a comparison of peak values of selected performance indicators has indicated that it would be possible to develop prebreeder core concepts having acceptable fuel rod performance. When compared to the reference core, the prebreeder seed rods might operate at a higher temperature throughout a significant part of the rod life. For certain performance criteria that are time and temperature dependent, less margin might exist in the prebreeder applications, but further detailed analysis would be expected to demonstrate that fuel rod stresses and temperatures would be below the criteria established for acceptability. Options identifying performance trade-offs, including the use of 70-percent cold worked clad in the prebreeder seed rods, were investigated and are described in Section D.2 and also in Appendix C.

3. Thermal and Hydraulic Performance

Steady state thermal and hydraulic calculations were made for the prebreeder core concepts and for the reference core using the HOTROD three-dimensional

thermal-hydraulic computer program (Reference 8), the BAW-2 CHF (Critical Heat Flux) correlation (Reference 1, Section 4.4) and engineering factors (such as flow area reductions) consistent with those used for the reference core analysis reported in Reference 1 (Section 4.4). The calculated steady state overpower capability presented in Table 7 was 111 percent for the reference core, and 114 percent, 117 percent, and 112 percent, respectively for prebreeder core Concepts 1, 2, and 3.

TABLE 7. STEADY STATE CHF OVERPOWER CAPABILITY PREDICTED BY HOTROD

	Overpower (% of 3800 MW)	<u>Prebreeder Overpower Capability</u> <u>Reference Core Overpower Capability</u>
Reference Core	111	-
Prebreeder Core Concept 1	114	1.03
2	117	1.06
3	112	1.01

It was determined that the HOTROD model with the BAW-2 CHF correlation conservatively predicts thermal capability. This was established by comparing the steady state thermal capability of the reference core calculated by the authors with that reported in Reference 1 (Section 4.4) and by analysis of numerous critical heat flux experiments.

Some of the important HOTROD input parameters are listed in Table 8. Relative to the reference core, the prebreeder core concepts would have 3 percent less flow and higher unrecoverable core pressure drops (18.6 to 19.5 psi versus 11.6 psi); these differences would occur because the prebreeder core concepts

TABLE 8. HOTROD INPUT PARAMETERS

<u>Parameter</u>	<u>Common Parameters*</u>	<u>Unique Parameters</u>	
		<u>Reference Core</u>	<u>Prebreeder Core Concepts</u>
Core Power, MW(th)	3800	-	-
Total Flow Rate, 10 ⁶ lb/hr	-	159.0	154.0 (Concept 1) 154.2 (Concept 2) 153.5 (Concept 3)
Percent Flow Reduction Due to			
Leakage	5.5	-	-
Maldistribution	1.0	-	-
Core Pressure Drop, psi			
Total	-	15.5	22.5 (Concepts 1 and 2) 23.4 (Concept 3)
Unrecoverable	-	11.6	18.6 (Concepts 1 and 2) 19.5 (Concept 3)
Inlet Temperature, °F	-	576.5	575.9
System Pressure, Minimum Steady-State, psia	2205	-	-
Grids			
Number of Grids	-	8	10
Average Loss	-	0.75	1.20 (Concepts 1 and 2)
Coefficient per Grid			1.23 (Concept 3)
Percent Reduction in Hot Subchannel Flow Area			
Interior Subchannel	2	-	-
Peripheral Subchannel	3	-	-
Percent Reduction in Hot Fuel Assembly Flow Area	2	-	-

*Reference core and prebreeders

TABLE 8. (Cont)

<u>Parameter</u>	<u>Common Parameters*</u>	<u>Unique Parameters</u>	
		<u>Reference Core</u>	<u>Prebreeder Core Concepts</u>
Mechanical Hot Channel Factor			
Enthalpy Rise	1.014	-	-
Local Heat Flux	1.025	-	-
Radial Power Peaking Factor Seed (Blanket) at the Most Limiting Core Location	-	1.55	1.85(0.03)** (Concept 1) 1.75(0.92) (Concept 2) 1.82(0.03)**(Concept 3)
Axial Power Peaking Factor at Limiting Location	-	1.67	1.67

*Reference core and prebreeders

**The most limiting conditions would occur at beginning of life when blanket rods produce little power.

would have more grids and smaller hydraulic diameters for individual flow channels resulting in higher grid loss coefficients and core flow resistance. To account for leakage around the core, the flow through the core was assumed to be 5.5 percent less than the total reactor coolant flowrate; this is based on Section 1.3 of Reference 1. Except for the peaking factors, the remaining data in Table 8 are all from Section 4.4 of Reference 1. The flow in the hot bundle position was assumed to be 1 percent less than average. The maximum control band errors on core pressure and inlet temperature were applied in the most conservative manner. Nominal inlet temperature was increased by 3°F, e.g. from 573.5°F to 576.5°F. Nominal system pressure of 2250 psia was reduced by 45 psi to 2205 psia. The flow area for the most limiting subchannel was the nominal area reduced 2 percent (3 percent for a peripheral subchannel) to account for channel closure. The flow area

of subchannels at the intermodular gap was reduced to represent the conservative situation where the adjacent fuel assemblies are bowed toward the hot fuel assemblies. This reduced the total flow area of the hot fuel assembly by 2 percent. A hot channel factor of 1.014 on enthalpy rise and a local hot spot factor of 1.025 on heat flux were applied. The most limiting core locations, for which the axial and radial power peaking factors are given, would all occur in Zone 1 at BOC.

Additional details of the calculational model are presented in Appendix D.

In summary, the HOTROD results, presented in Table 7, indicate that the steady state CHF overpower capability for all three prebreeder core concepts would be somewhat greater than that of the reference core. As discussed in Section I of Appendix D, these results are taken to indicate that during normal and accident transients, the prebreeder core concepts would also have acceptable CHF power capability.

4. Loss-of-Coolant Accident Performance

The performance of the three prebreeder core concepts during a hypothetical loss-of-coolant accident (LOCA) was evaluated by comparing their calculated behavior and margins to design criteria limits with those of the reference core calculated in the same manner. The FLASH-6 computer program (Reference 12) was used to calculate the first ten seconds of the blowdown portion of a postulated LOCA from a double-ended guillotine break at a coolant pump discharge.

The first 10 seconds of blowdown were found to be very similar for all four cores. Flow rates out of the hot fuel assembly were found to be similar. Calculated cladding temperatures for the hot spot fuel location in all four cases exhibited the same overall behavior and the peak temperatures calculated were

very nearly the same, lying in the band between 1630°F and 1810°F. These results were taken to indicate that during the remainder of the blowdown phase, the behavior of the prebreeder core concepts would be similar to that of the reference core and that similar conditions would exist in all four cores at the start of the reflood portion of the transient when the blowdown is complete. Since the plant would be identical in all cases and the initial temperatures would be nearly the same, it is expected that the reflood behavior of the four cores would be very similar also.

Comparative calculations of reflood characteristics during emergency core cooling were not made and would be of high priority in further consideration of these concepts.

Additional details of the LOCA analysis are presented in Appendix D.

5. Resource and Service Requirements and Environmental Considerations

This section presents a summary of the calculated principal resource and service requirements and environmental considerations associated with the support of the operation of the prebreeder core concepts and the reference core. The results presented here do not include the actual operation of a reactor plant with the various core types installed because the requirements for and effects of this operation are assumed to be nearly independent of the type of core installed.

The details of the methods used for these calculations, the basic assumptions, and a more detailed discussion of the principal results are presented in Appendix E.

The principal conclusion drawn from these calculations is that the annual requirements of resources and services and the annual environmental considerations

would be approximately the same for all four types of cores. However, there would be significant differences in the amounts of uranium and separative work required by the three prebreeder core concepts to produce sufficient uranium-233 to support a water-cooled breeder.

a. Resource and Service Requirements - The principal resource requirements would be uranium and thorium from mining. The principal service requirements, in addition to the mining and milling of uranium and thorium ore, would be enrichment, fuel fabrication, and fuel reprocessing.

The annual requirements for each of the prebreeder core concepts and for the reference core are presented in Table 9. Also included in this table are the total amounts of U_3O_8 , ThO_2 , and separative work which would be required to produce 6200 kg of uranium-233. (This amount of uranium-233 would be sufficient to supply the fissile requirements of a light water-cooled breeder with a rating of 3000 MWth or 1000 MWe. The fissile requirement includes both the in-core and the out-of-core inventory of a breeder assuming a one year out of core time for fuel reprocessing and refabrication.)

It is observed that the annual requirements for both U_3O_8 and separative work would fall into relatively narrow ranges between 175 to 228 ST U_3O_8 and between 160 to 197 MT SWU. The annual ThO_2 requirement (assuming no recycle of thorium into prebreeders) would range from zero for the reference core to 47 short tons for prebreeder core Concept 3. The annual number of fuel rods fabricated would be higher for the prebreeders because of their 19 by 19 array versus the 17 by 17 array for the reference core. It is important to note that for prebreeder core Concept 3 remote fabrication would be required for the seed rods (which would comprise about 65 percent of the total number of fuel rods) because they would contain uranium-233 and associated uranium-232. For prebreeder core Concept 2

TABLE 9. RESOURCE AND SERVICE REQUIREMENTS
AND U-233 PRODUCTION

<u>Item</u>	<u>Reference Core</u>	<u>Prebreeder Core Concepts</u>		
		<u>1</u>	<u>2</u>	<u>3</u>
		<u>Annual Requirements</u>		
ST U_3O_8	209	228	195	175
ST ThO_2	0	15	24	47
Uranium Ore Mined (1000 ST)*	105	114	98	88
Thorium Ore Mined (1000 ST)*	0	3.0	4.8	9.4
Enrichment, MT SWU	160	197	175	172
Fuel Fabrication, kilorods	21	27	27	27
Reprocessing, MT HM	36	39	39	39
Uranium-233 Production (Kg) (after 1 percent loss to reprocessing plus fabrication)	0	161	255	198
		<u>Requirement to Produce 6200 kg U-233</u>		
ST U_3O_8	--	8800	4700	5500
ST ThO_2	--	580	580	1500
Enrichment, MT SWU	--	7600	4300	5400

*Uranium ore assumed to contain 0.2 percent U_3O_8 . Thorium ore assumed to contain 0.5 percent ThO_2 . No losses assumed for milling. No recycle of thorium.

remote fabrication might also be required for the blanket rods which would contain plutonium. The annual requirements for fuel reprocessing in terms of mass throughput would be very similar for all four core types.

The total requirements of U_3O_8 , ThO_2 , and enrichment to provide 6200 kg of uranium-233 would have a much wider range than the annual requirements; the

U_3O_8 requirements would range from 4700 short tons to 8800 short tons, the ThO_2 requirements would range from 580 to 1500 short tons and the separative work requirements would range from 4300 metric tons to 7600 metric tons. Prebreeder core Concept 2 would have the smallest requirements, partly because the plutonium would be recycled and partly because it would be the only one of the three prebreeders in which fresh blanket fuel would contain fissile material (the recycled plutonium). This second feature would permit prebreeder core Concept 2 to have the largest number of blanket rods and the highest annual output of breeder usable uranium-233.

The significance of these total requirements is that following the production of sufficient uranium-233 from one or more prebreeders, a breeder could be started up. No additional mining or separative work would be required to support the breeder operation. The irradiated thorium stock pile resulting from the operation of prebreeders to produce the above amount of uranium-233 would be more than sufficient to provide the requirements for the complete fuel cycle of one light water-cooled breeder. This would include thorium undergoing irradiation in the core plus thorium undergoing reprocessing or refabrication. This requirement would be equivalent to about 524 ST ThO_2 . Any excess ThO_2 remaining from prebreeder operation would be used as makeup thorium (to replace reprocessing and fabrication losses and conversion to uranium-233) during breeder operation.

An additional factor relevant to breeder operation is that a water-cooled breeder would be expected to have a higher annual mass throughput than the prebreeders and the reference core. Hence, following the replacement of a prebreeder with a water-cooled breeder both the fuel fabrication and fuel reprocessing annual requirements would be expected to increase relative to the values presented in the table.

b. Environmental Considerations - Environmental considerations on an annual basis are presented in Table 10. The data in Table 10 are based on the Shippingport LWBR Environmental Statement (Reference 2) and the method of calculation is presented in Appendix E. It is observed that these considerations would be comparable for all four types of reactor cores. Again, it should be recognized that following replacement of a prebreeder core with a breeder core, the annual requirements for mining and enrichment would go to zero, while the annual requirements for fabrication and reprocessing would probably increase.

D. Sensitivities

Sensitivity to various changes including fuel assembly mechanical changes, power level changes, fuel rod changes, and sensitivity of uranium requirements to fuel-cycle parameter changes are discussed below.

1. Mechanical, Nuclear, and Thermal Sensitivities

A simplified analysis model was utilized to determine the sensitivity of prebreeder mechanical, nuclear, and thermal performance and characteristics to changes in various core parameters. The simplified procedure provided a comparative analysis of prebreeder core concepts and the reference core. For each parameter change, prebreeder performance was predicted relative to that of the reference core. It was assumed that approximating or neglecting many variables and phenomena in the simplified analysis procedure would affect the predicted performance of both the prebreeder and reference cores equally.

Changes were made in the following parameters: average reactor plant temperature, thoria rods located in unrodded control rod guide tubes, number of grids, core power level, seed and blanket rod diameters, number of seed rods, number of control rods, core lifetime, and the addition of uranium-235 to the

TABLE 10. CALCULATED ANNUAL ENVIRONMENTAL CONSIDERATIONS FOR
SUPPORT OF A 1322 MWE REACTOR

Consideration	ENVIRONMENTAL CONSIDERATION FOR EACH FUEL CYCLE COMPONENT						Total Environ- mental Consider- ation
	Reactor ⁽¹⁾	Uranium Production ⁽²⁾	Thorium ⁽³⁾ Production	Uranium Enrichment	Fuel Fabrication	Reprocessing ⁽⁴⁾	
Land permanently committed (acres per facility life)	PBR-1	80	1	0	0	2	83
	2	68	2	0	0	2	72
	3	61	4	0	0	2	67
	REF LWR	73	0	0	0	2	75
Land temporarily committed (acre years per reactor year)	PBR-1	583	17	90	19	77	786
	2	498	27	80	19	77	701
	3	447	55	78	40	77	697
	REF LWR	536	0	73	17	71	697
Total water (10 ⁶ gallons)	PBR-1	188	3	167	2	9	369
	2	161	6	149	2	9	327
	3	144	12	146	1	9	312
	REF LWR	172	0	136	2	8	318
Petroleum and other liquid fuels (10 ³ gallons) (150 standard cubic feet of natural gas = 1 gal- lon petroleum)	PBR-1	676	13	12	0	293	994
	2	578	22	11	0	293	904
	3	519	43	11	0	293	866
	REF LWR	620	0	10	0	270	900
Coal (MT)	PBR-1	1520	18	111,100	632	464	114,000
	2	1300	29	98,700	632	464	101,000
	3	1160	58	97,000	507	428	99,200
	REF LWR	1390	0	90,200	583	428	92,600
Chemicals (MT) to air and water	PBR-1	113	1	1220	11	30	1380
	2	97	1	1090	11	30	1230
	3	87	2	1070	11	30	1200
	REF LWR	103	0	992	11	28	1130
Radiological dose ⁽⁶⁾ commitment (man rem)	PBR-1	634	n ⁽⁵⁾	n	n	38	672
	2	542	n	n	n	38	580
	3	487	n	n	n	38	525
	REF LWR	582	0	n	n	34	616
Thermal (10 ⁹ BTU)	PBR-1	142	3	4530	27	85	4790
	2	122	4	4030	27	85	4270
	3	110	8	3960	27	79	4180
	REF LWR	131	0	3680	25	79	3920
Electrical energy (Megawatt years)	PBR-1	0.7	n ⁽⁵⁾	55.0	0.3	0.2	56.2
	2	0.6	n	48.8	0.3	0.2	49.9
	3	0.6	n	48.0	0.3	0.2	49.1
	REF LWR	0.7	n	44.6	0.3	0.2	45.8

(1) PBR-1 = prebreeder core concept 1 etc. REF LWR = Reference light water reactor

(2) Includes mining, milling and conversion to UF₆

(3) Includes mining and milling

(4) Includes storage and waste management

(5) n signifies negligible compared with considerations from other fuel cycle components

(6) World population 70 year total body dose commitment.

blanket pellets. The results, shown in Tables 11, 12 and 13 for prebreeder core Concepts 1, 2 and 3, respectively, include the performance indicators and characteristics considered to be of most interest. The tables show the percent change from the base values caused by variations in each parameter.

Reducing the plant average temperature was examined as a method for improving thermal capability. A reduction in T_{ave} from 603°F to 598°F was assumed since this was considered to be about the maximum allowable temperature reduction which would still permit selected steam conditions to be maintained in the reference plant. The results were that a 5°F reduction in T_{ave} would cause a 1-percent improvement in thermal capability. The uranium-233 production and the peak linear power rating would increase slightly due to a reduction in plant thermal efficiency and a corresponding increase in thermal power to maintain constant electrical output.

Adding thorium rods to the unrodded control rod guide tubes was examined as a method for increasing uranium-233 production. About 65 percent of the guide tubes would be unrodded and thus potentially available for this purpose. Thorium rods were placed in all of the unrodded locations and also in just one-half of the unrodded locations. The results were that uranium-233 production could be increased by 10 to 18 percent (depending on the particular core concept) by utilizing all of the unrodded control rod guide tube sites and by 5 to 9 percent if only one-half of the sites were used. The thermal capability would be relatively unchanged. Note that the addition of thorium rods in available guide tube sites would necessarily affect the fuel and poison management strategy.

Changes were made to the grid loss coefficient. These would be of interest for example in the event that addition or modification of grids was required for

TABLE 11. SENSITIVITY STUDIES FOR PREBREEDER CORE CONCEPT 1

Value of Parameter	Percent Change from Prebreeder Core Concept 1 Base Value						
	Annual U-233 Production	Annual Uranium Requirement	Annual Separative Work Requirement	Fuel-to Coolant Ratio	U-235 Loading per Fuel Assembly	Peak kw/ft (BOC, Zone 1)	Relative Thermal Capability
<u>Average Plant Temperature (Base Value is 603°F)</u>							
598°F	+ 1.1	+ 2.2	+ 2.4	0.0	+ 1.4	+ 2.5	+ 1.0
<u>Thoria in Guide Tubes (65% of the Guide Tube Locations would be Available, None Used for Thoria in Base Concept)</u>							
All Available Guide Tubes	+18.0	+ 2.6	+ 4.1	+10.8	+ 7.8	+ 0.5	+ 0.3
1/2 Available Guide Tubes	+ 9.0	+ 1.3	+ 2.0	+ 5.3	+ 3.8	+ 0.1	+ 0.2
<u>Grid Loss Coefficient</u>							
+50% Loss Coefficient	0.0	0.0	0.0	0.0	0.0	0.0	- 1.8
<u>Power Level/Lifetime/Capacity Factor (Base Values are 100%/3.0 yrs/0.8)</u>							
95%/3.16 yrs/0.805	- 5.0	- 4.4	- 4.6	0.0	- 2.6	- 5.4	+ 4.6
90%/3.33 yrs/0.810	-10.0	- 8.8	- 9.2	0.0	- 5.2	-10.8	+ 9.8
85%/3.53 yrs/0.815	-15.0	-13.2	-13.8	0.0	- 7.8	-16.2	+15.9
<u>Seed Rod Diameter (Base Value would be 0.349 in.)</u>							
0.300 in.	+ 0.1	- 3.4	- 3.7	-29.2	-18.9	- 2.6	- 3.4
0.379 in.	- 0.3	+ 4.7	+ 5.8	+26.0	+19.9	+ 1.3	+ 0.4
<u>Blanket Rod Diameter (Base Value would be 0.379 in.)</u>							
0.300 in.	-40.6	- 4.8	- 8.1	-20.0	-13.7	+ 2.4	+ 1.0
<u>Number of Seed Rods (Base Concept would have 248 Seed Rods)</u>							
228	+23.1	+ 1.6	+ 3.1	+ 2.8	+ 1.9	+ 6.9	- 1.7
208	+45.9	+ 3.1	+ 6.2	+ 5.6	+ 3.8	+15.2	- 3.6
188	+68.4	+ 4.7	+ 9.2	+ 8.5	+ 5.7	+25.3	- 5.6
168	+90.7	+ 6.1	+12.2	+11.4	+ 7.7	+38.0	- 8.8

TABLE 11 (Cont)

Percent Change from Prebreeder Core Concept 1 Base Value							
Parameter	Annual U-233 Production	Annual Uranium Requirement	Annual Separative Work Requirement	Fuel-to Coolant Ratio	U-235 Loading per Fuel Assembly	Peak kw/ft (BOC, Zone 1)	Relative Thermal Capability
<u>Number of Control Rods per Assembly (Base Concept would have 28 Control Rods, 248 Seed Rods)</u>							
32 (248 Seed Rods)	- 4.6	- 0.6	- 1.0	- 2.4	- 1.8	+ 0.2	- 0.3
32 (244 Seed Rods)	+ 0.1	- 0.4	- 0.4	- 1.8	- 1.4	+ 1.6	- 0.6
<u>Core Lifetime/Capacity Factor (Base Values are 3.0 yrs/0.80)</u>							
2.5 yrs/0.780	- 1.1	- 2.2	- 2.3	0.0	- 1.3	+ 0.1	0.0
3.5 yrs/0.814	+ 0.8	+ 1.6	+ 1.7	0.0	+ 0.9	0.0	0.0
<u>Add Uranium to Blanket (Base Concept would have No Fissile Added to Blanket Rods)</u>							
U-235 = 0.37%, U-238 = 0.03%	- 0.2	+ 1.8	+ 1.6	0.0	- 3.4	- 2.4	+ 0.7
U-235 = 0.09%, U-238 = 0.01%	- 0.1	+ 0.4	+ 0.4	0.0	- 0.8	- 0.5	0.0

TABLE 12. SENSITIVITY STUDIES FOR PREBREEDER CORE CONCEPT 2

Value of Parameter	Percent Change from Prebreeder Core Concept 2 Base Value						
	Annual U-233 Production	Annual Uranium Requirement	Annual Separative Work Requirement	Fuel-to- Coolant Ratio	U-235 Loading per Fuel Assembly	Peak kw/ft (BOC, Zone 1)	Relative Thermal Capability
<u>Average Plant Temperature (Base Value is 603°F)</u>							
598°F	+ 1.2	+ 2.1	+ 2.1	0.0	+ 1.0	+ 2.5	+ 1.0
<u>Thoria in Guide Tubes (65% of the Guide Tube Locations would be Available, None Used for Thoria in Base Concept)</u>							
All Available Guide Tubes	+10.3	+ 3.3	+ 4.5	+10.7	+ 9.6	+ 2.7	+ 1.1
1/2 Available Guide Tubes	+ 5.2	+ 1.6	+ 2.2	+ 5.2	+ 4.6	+ 1.4	+ 0.7
<u>Grid Loss Coefficient</u>							
+50% Loss Coefficient	0.0	0.0	0.0	0.0	0.0	0.0	- 1.8
<u>Power Level/Lifetime/Capacity Factor (Base Values are 100%/3.0 yrs/0.8)</u>							
95%/3.16 yrs/0.805	- 5.0	- 4.1	- 4.2	0.0	- 1.8	- 5.4	+ 4.6
90%/3.33 yrs/0.810	-10.0	- 8.2	- 8.4	0.0	- 3.6	-10.7	+ 9.7
85%/3.53 yrs/0.815	-15.0	-12.3	-12.6	0.0	- 5.4	-15.9	+15.4
<u>Seed Rod Diameter (Base Value would be 0.339 in.)</u>							
0.300 in.	- 1.5	+ 8.0	+10.3	-18.4	- 0.9	+ 5.5	- 2.3
0.379 in.	+ 2.1	-11.3	-12.9	+26.1	+ 3.2	- 7.4	- 1.3
<u>Blanket Rod Diameter (Base Value would be 0.379 in.)</u>							
0.300 in.	-40.0	- 8.6	-13.6	-31.3	-25.0	+ 4.3	0.0
<u>Number of Seed Rods (Base Concept would have 192 Seed Rods)</u>							
172	+12.5	+ 7.1	+ 9.5	+ 3.4	+10.5	+14.1	- 1.0
152	+24.8	+13.9	+18.6	+ 6.8	+21.2	+31.8	- 3.3
132	+36.8	+20.2	+27.1	+10.4	+31.8	+54.7	- 6.9
112	+48.7	+26.2	+35.1	+14.0	+42.4	+85.7	-12.6

TABLE 12 (Cont)

Value of Parameter	Percent Change from Prebreeder Core Concept 2 Base Value						
	Annual U-233 Production	Annual Uranium Requirement	Annual Separative Work Requirement	Fuel-to- Coolant Ratio	U-235 Loading per Fuel Assembly	Peak kw/ft (BOC, Zone 1)	Relative Thermal Capability
<u>Number of Control Rods per Assembly (Base Concept would have 28 Control Rods, 192 Seed Rods)</u>							
32 (192 Seed Rods)	- 1.3	- 0.6	- 0.7	- 1.2	- 1.2	0.0	- 0.2
32 (188 Seed Rods)	+ 1.2	+ 1.0	+ 1.4	- 0.5	+ 1.0	+ 2.6	- 0.6
<u>Core Lifetime/Capacity Factor (Base Values are 3.0 yrs/0.80)</u>							
2.5 yrs/0.780	- 1.2	- 2.0	- 2.1	0.0	- 0.9	+ 0.2	0.0
3.5 yrs/0.814	+ 0.8	+ 1.4	+ 1.5	0.0	+ 0.7	0.0	0.0

TABLE 13. SENSITIVITY STUDIES FOR PREBREEDER CORE CONCEPT 3

Value of Parameter	Percent Change from Prebreeder Core Concept 3 Base Value						
	Annual U-233 Production	Annual Uranium Requirement	Annual Separative Work Requirement	Fuel-to- Coolant Ratio	U-235 Loading per Fuel Assembly	Peak kw/ft (BOC, Zone 1)	Relative Thermal Capability
<u>Average Plant Temperature (Base Value is 603°F)</u>							
598°F	+ 1.1	+ 2.4	+ 2.4	0.0	+ 1.8	+ 2.5	+ 1.3
<u>Thoria in Guide Tubes (65% of the Guide Tube Locations would be Available, None Used for Thoria in Base Concept)</u>							
All Available Guide Tubes	+13.5	+ 1.2	+ 1.2	+10.5	+ 6.8	+ 0.6	+ 0.2
1/2 Available Guide Tubes	+ 6.8	+ 0.6	+ 0.6	+ 5.2	+ 3.4	+ 0.3	+ 0.1
<u>Grid Loss Coefficient</u>							
+50% Loss Coefficient	0.0	0.0	0.0	0.0	0.0	0.0	- 1.8
<u>Power Level/Lifetime/Capacity Factor (Base Values are 100%/3.0 yrs/0.8)</u>							
95%/3.16 yrs/0.805	- 5.0	- 4.7	- 4.7	0.0	- 3.5	- 5.5	+ 4.6
90%/3.33 yrs/0.810	-10.0	- 9.5	- 9.4	0.0	- 7.0	-11.0	+ 9.8
85%/3.53 yrs/0.815	-15.0	-14.2	-14.1	0.0	-10.6	-16.3	+15.5
<u>Seed Rod Diameter (Base Value would be 0.349 in.)</u>							
0.300 in.	- 0.1	- 2.8	- 3.1	-26.2	-16.1	- 2.2	- 4.4
0.379 in.	+ 0.1	+ 1.4	+ 1.6	+22.7	+12.8	+ 1.0	+ 1.0
<u>Blanket Rod Diameter (Base Value would be 0.379 in.)</u>							
0.300 in.	-40.3	- 4.5	- 4.4	-25.4	-18.0	+ 2.3	+ 1.7
<u>Number of Seed Rods (Base Concept would have 220 Seed Rods)</u>							
200	+17.1	+ 0.8	+ 0.7	+ 2.5	+ 2.4	+ 8.5	- 2.2
180	+34.0	+ 1.5	+ 1.3	+ 5.1	+ 4.8	+19.0	- 4.8
160	+50.7	+ 2.2	+ 1.9	+ 7.7	+ 7.3	+32.8	- 8.1
140	+67.1	+ 2.8	+ 2.4	+10.4	+ 9.8	+49.8	-12.7

TABLE 13 (Cont)

Value of Parameter	Percent Change from Prebreeder Core Concept 3 Base Value						
	Annual U-233 Production	Annual Uranium Requirement	Annual Separative Work Requirement	Fuel-to- Coolant Ratio	U-235 Loading per Fuel Assembly	Peak kw/ft (BOC, Zone 1)	Relative Thermal Capability
<u>Number of Control Rods per Assembly (Base Concept would have 28 Control Rods, 220 Seed Rods)</u>							
32 (220 Seed Rods)	- 3.5	- 0.4	- 0.4	- 2.3	- 1.7	+ 0.1	- 0.3
32 (216 Seed Rods)	0.0	- 0.2	- 0.2	- 1.8	- 1.2	+ 1.7	- 0.5
<u>Core Lifetime/Capacity Factor (Base Values are 3.0 yrs/0.80)</u>							
2.5 yrs/0.780	- 1.1	- 2.4	- 2.4	0.0	- 1.8	0.0	0.0
3.5 yrs/0.814	+ 0.7	+ 1.6	+ 1.6	0.0	+ 1.2	- 0.1	0.0
<u>Add Uranium to Blanket (Base Concept would have No Fissile Added to Blanket Rods)</u>							
U-235 = 0.37%, U-238 = 0.03%	- 0.1	+ 4.1	+ 4.2	0.0	- 5.5	- 3.3	+ 1.0
U-235 = 0.09%, U-238 = 0.01%	0.0	+ 1.0	+ 1.0	0.0	- 1.4	- 0.8	+ 0.3

structural considerations. In the calculations, the grid loss coefficient was uniformly distributed along the entire core length so that an X percent increase in the number of grids would have the same effect as an X percent increase in loss coefficient. The result was that a 50 percent increase in grid loss coefficient would cause 2 percent reduction in thermal capability.

Reducing the core power level to values less than 3800 MWth was examined as a method of improving core thermal capability and fuel rod linear power (kw/ft) performance. An increase in core lifetime consistent with the reduction in power level was assumed. In this case the relative thermal capability increased and the peak linear power decreased by about the same ratio as the reduction in power level. For example, a 15-percent reduction in power level would cause a 16-percent decrease in peak linear power (kw/ft) and a 15- to 16-percent increase in relative thermal capability. A 15-percent reduction in power level would cause a 15-percent reduction in yearly uranium-233 production if no changes were made to the number of blanket rods. The total uranium-233 production for the entire core life would be the same as that for the base cases. For prebreeder core Concept 1 the 15 percent power reduction could be combined with a reduction in the number of seed rods (determined by peak linear power (kw/ft) and thermal capability) and an increase in the number of blanket rods to achieve an overall increase in annual uranium-233 production of about 27 percent. For prebreeder core Concepts 2 and 3 similar changes would yield increases of 5 percent and 13 percent, respectively, in annual uranium-233 production.

Changes were also made in the seed and blanket rod diameters. In all cases, uranium-235 enrichment was adjusted to maintain a one-third core annual refueling

schedule at the base capacity factor. With the blanket rod diameter fixed at the base value of 0.379 inch, reducing the seed rod diameter from 0.379 inch to 0.300 inch would decrease thermal capability by 4 to 5 percent for prebreeder core Concepts 1 and 3, and 1 percent for prebreeder core Concept 2, and would have only a minor effect on uranium-233 production.

Reducing the blanket rod diameter from 0.379 inch to 0.300 inch would reduce uranium-233 production by 40 percent for all three prebreeder concepts due to the significant reduction in blanket rod thorium content. The thermal capability and peak linear power (kw/ft) would be relatively unchanged.

For each prebreeder core concept, the number of seed rods was varied from the base value to a value 80 seed rods below the base (that is, down to 168 seed rods for prebreeder core Concept 1, 112 seed rods for prebreeder core Concept 2, and 140 seed rods for prebreeder core Concept 3). Uranium-235 enrichment was adjusted to maintain a one-third core annual refueling schedule at the base capacity factor. Reducing the number of seed rods would cause a corresponding increase in the number of blanket rods and a substantial increase in uranium-233 production. A reduction of 80 seed rods would cause increases in uranium-233 production of 90 percent for prebreeder core Concept 1, 48 percent for prebreeder core Concept 2, and 67 percent for prebreeder core Concept 3. At the same time there would be modest increases in uranium and separative work requirements for prebreeder core Concepts 1 and 3 due to increased uranium enrichment, and substantial increases for prebreeder core Concept 2 to compensate for the decreased plutonium production inherent with reduced seed fuel volume. However, for a reduction of 80 seed rods, the peak linear power (kw/ft) would be increased by 38 to 85 percent and the thermal capability would be reduced by 9 to 13 percent

depending on the concept. These changes in thermal capability and peak linear power (kw/ft) essentially would restrict the seed rod count to about the present base values for the three prebreeder core concepts, if the reference core power capability were to be maintained.

The number of control rod guide tubes per fuel assembly was also changed. This change would be of interest if requirements on reactivity control or shut-down margin were changed. The base prebreeder fuel assemblies would each have 28 control rod guide tubes and this number was increased to 32. First, the number of seed rods was held constant and number of blanket rods was reduced by four. This would cause a reduction in uranium-233 production of between 1 and 5 percent depending on the prebreeder core concept, and a negligible reduction in thermal capability. Next, the number of blanket rods was held constant and the number of seed rods was reduced by four. This would cause no change in uranium-233 production for prebreeder core Concepts 1 and 3, a 1-percent increase for prebreeder core Concept 2, and the thermal capability would be reduced by about one-half percent for all three prebreeder core concepts.

Changes were also made in the cycle lifetime. The base core concepts would have a cycle life of 3 years, which means that fuel rods would remain in the core for 3 years. One-third of the core would be refueled every year. Cycle times of 2.5 years and 3.5 years were considered while maintaining one-third core refueling. Uranium-235 loadings were varied to provide the desired lifetime. The results were that modest increases in uranium-233 production would be attained as the cycle time is increased. Increasing the cycle time from 2.5 to 3.5 years would increase the annual uranium-233 production by 2 percent. At the same time, however, the uranium and separative work requirements would be increased approximately 4 percent. There would be only a negligible effect on the thermal capability and peak kw/ft.

The final change was adding fissile uranium to the blanket rods of prebreeder core Concepts 1 and 3. Two levels of highly enriched uranium addition were considered: 0.4 percent total uranium (0.37-percent uranium-235, 0.03-percent uranium-238) and 0.1 percent total uranium (0.09-percent uranium-235, 0.01-percent uranium-238). Without a change in the number of seed and blanket rods, the effect on uranium-233 production for both prebreeder core concepts would be negligible. At 0.4-percent uranium addition, the peak kw/ft would be reduced by 2 to 3 percent, depending on the prebreeder core concept. The thermal capability would be increased by about 1 percent. For prebreeder core Concept 1 this would permit a reduction in the number of seed rods to 236, with an accompanying 14-percent increase in uranium-233 production. However all of the uranium-233 produced would contain uranium-235, 236, and 238 (thus reducing its effectiveness for application in a water cooled breeder) unless some way were available to keep the bred uranium-233 separated from the uranium initially loaded into the blanket rods.

2. Fuel Sensitivities

A limited sensitivity study was performed to identify ways of improving or trading off specific performance indicators. The fuel parameters changed were initial fuel-clad gap size, fuel density, and clad material. Table 14 presents performance indicators for the parameter values investigated for prebreeder core Concept 1. Similar indicators for prebreeder core Concepts 2 and 3 are presented in Appendix C. Similar trends were observed for all three prebreeder core concepts.

By comparing the second and fourth columns of performance data in Table 14, it can be seen that a 1-mil increase in initial diametral gap from the nominal 7 mils would increase the maximum center temperature 135°F and the maximum average temperature 93°F. A similar result is obtained for prebreeder core Concept 2. A larger sensitivity to initial gap change is observed for uranium

TABLE 14. FUEL ROD PERFORMANCE, NO PLUTONIUM RECYCLE (CORE CONCEPT 1)

Performance Indicator	Reference Core	Prebreeder Core Concept 1	Variation from Prebreeder Core Concept 1			
			Increased Fuel Pellet Density By 2%	0.008-in. Gap	CW Clad 0.007-in. Gap	CW Clad 0.008-in. Gap
Center Temperature, °F, 112% Power (BOC)	3808	3649	3362	3784	3687	3827
Fuel Average Temperature, °F, 100% Power (BOC)	2143	2080	1992	2173	2094	2186
Clad Circumferential Stress psi, 100% Power (EOC)	20,000	24,100	20,800	15,200	21,100	21,200
Clad Generalized Stress, psi, 112% Power (EOC)	24,100	27,600	24,700	19,900	24,100	24,500
Clad Axial Strain, % (EOC)	1.77	0.64	0.63	0.56	1.86	1.85
Clad Plastic Radial Strain, % (EOC)	~0.0	~0.0	~0.0	~0.0	~0.07	~0.0
Fuel Pellet-to-Clad Diametral Gap, in.	0.008	0.007	0.007	0.008	0.007	0.008
Clad Material	70% CW	RXA	RXA	RXA	CW	CW

dioxide-thorium dioxide fuel (prebreeder core Concept 3) where a 1-mil increase would produce a 160°F increase in center temperature and a 116°F increase in average fuel temperature.

The impact of changes in initial gap size on clad stress would be dependent on the base case fuel-clad interaction state at the time the maximum stress occurs, generally at the end of life. If in the base case a fuel-clad gap still existed at the end of life, an increase in initial gap size would have little effect on clad stress. If in the base case contact occurred early in life between fuel and clad and the two reached a "steady-state" tensile stress condition (see reference core in Figure 8), then increasing or decreasing the initial gap size would possibly have little or no effect other than to delay or advance the time to reach a "steady-state" tensile stress condition without reducing clad stresses. This situation would exist in the alternatives with cold-worked (CW) clad listed in the fifth and sixth data columns of Table 14. Changing the initial gap size can alter the fuel-clad interaction state and cause large changes in stress. This is illustrated in Table 14 (data columns 2 and 4) where increasing the initial gap by 1-mil to 0.008 inches is shown to reduce the clad circumferential stress from 24,100 psi to 15,200 psi.

Performance indicators with an increase in fuel density are in the third data column of Table 14. A higher fuel density would reduce pellet densification and improve fuel conductivity; both of these effects would lower fuel temperatures. A 2-percent increase in the reference density (from 95 percent to 97 percent of theoretical density) would reduce the center temperature approximately 287°F and also would reduce the clad stresses. Based on these data, it may be possible, by using higher density fuel, to improve both thermal and stress-strain performance.

Performance indicators with 70-percent CW cladding are, in the last two columns of Table 14. In general, the performance of the fuel rods using cold worked cladding would be very similar to that of the fuel rods using annealed cladding, except for the clad axial strain which would be approximately 3 times higher due to the higher in-pile creep rate of cold worked cladding. The results also demonstrate that fuel rods using cold worked cladding could be developed to obtain the values of selected performance indicators achieved by the reference core. This could be significant if lower resistance to stress corrosion cracking or poorer strain instability performance was encountered with annealed cladding.

Based on these limited sensitivity studies, it is judged that sufficient flexibility exists to accommodate minor changes from the base fuel rod parameters which might be required to obtain performance equivalent to that of the reference core.

IV. CONSIDERATIONS FOR FURTHER DEVELOPMENT

The core concepts defined in this report have evolved with the intention of minimizing the changes required to the reference core as defined in Reference 1 and of making maximum use of the reference core mechanical features. The development program required to demonstrate structural adequacy and to fully qualify the mechanical design of the prebreeder core concepts likewise would be minimized. Some of the testing and analysis done for the reference core would be applicable to the prebreeder core concepts.

A. Changes Required to the Reference System

The major components that would require a redesign for the prebreeder core concepts are the fuel assemblies, the control assemblies, and the plenum assembly. Specific areas for each of the major components that would require changes are as follows:

1. Fuel assemblies
 - a. Seed and blanket fuel rods
 - b. Spacer grids
 - c. Upper and lower end fittings
 - d. Control rod guide tubes and orifice plug
 - e. Instrumentation tubes and spacer sleeves.
2. Control assemblies
 - a. Control Rod Assembly including control rods, spider, and the threaded fastener
 - b. Axial Power Shaping Rod Assembly including control rods, spider, and threaded fastener

- c. Burnable Poison Rod Assembly including poison rod, spider, locking plate, and spring
 - d. Orifice Rod Assembly including orifice rod, spider, locking plate, and spring.
3. Plenum assembly
- a. Control rod guide structure
 - b. Plenum cover
 - c. Upper grid plate.

B. Licensability

The three prebreeder core concepts would be similar to conventional pressurized water cores. They would use the same moderator and coolant (light water), have similar mechanical features and employ established fuel rod technology. The same requirements would have to be fulfilled for prebreeder licensing as for conventional cores. Detailed core analyses would be necessary to demonstrate satisfactory performance at the desired power level under both normal operating conditions and potential accident conditions. Technical development work and testing programs as outlined in Section V would have to be accomplished to qualify the performance prediction procedures for the prebreeder geometry and the ranges of variables involved.

For prebreeder operation to be meaningful, there would have to be an acceptance of long-range fuel recycle, without which breeder concepts cannot exist. Prebreeder core Concept 1 would not require fuel recycle for its implementation; however, without uranium-235 recycle it would require appreciably more U_3O_8 and separative work than the reference core. Prebreeder core Concepts 2 and 3, on the other hand, would require recycle of prebreeder fuel.

The current emergency core cooling system acceptance criteria being used by the Nuclear Regulatory Commission (NRC) for uranium based fuel would need to be extended to the thorium based fuel systems used in the prebreeders. Since thorium has a higher melting point than uranium and may have improved thermal conductivity, this extension is thought to be straightforward.

These prebreeder core concepts, which would produce uranium-233 for future use in water-cooled breeder reactors, characteristically would have some higher resource and service requirements than conventional light water reactors. These short-term costs would be incurred in order to obtain the long-term benefit of having water-cooled breeders which would operate without consuming naturally occurring uranium-235.

C. Additional Considerations

During the development of these core concepts, areas were identified where further evaluation would be required to optimize the fuel rod and mechanical details, and the fuel loading.

1. Nuclear

The nuclear analyses reported here explicitly represented an average radial slice of the core being calculated. Three-dimensional nuclear and thermal effects on power peaking and on core reactivity and shutdown were inferred. Determination of optimal core features would require explicit analytical treatment of these effects. Three-dimensional analysis would need to include the effects of thermal feedback and of spacer grid geometry on local power peaking. The depletion dependence of core axial leakage and its effect on reactivity shutdown would need to be investigated.

Since these core concepts have previously untested mixed lattices of uranium and thorium-bearing fuel rods as well as rods containing thorium and plutonium, it would be desirable to have a critical assembly experiment to confirm the analytical predictions.

2. Fuel Technology

a. Uranium Dioxide Seed Rods - The uranium dioxide seed rods would have features similar in geometry and materials to current commercial cores and also to fuel rods in the Shippingport LWBR design (Reference 2). As a result, the geometric arrangement and materials selection would introduce no new technological concerns. Although the pellet diameters would be smaller than in commercial core designs (0.285 to 0.295 versus 0.324 inch), they still would be larger than the Shippingport LWBR seed pellet (0.252 inch). Use of pressurized fuel rods is standard practice in current commercial cores, and in these studies, a helium fill pressure consistent with commercial practice was assumed. However, an optimum choice of pressure would require a detailed evaluation of core performance for both normal operation and for the loss-of-coolant accident. Higher initial fill pressure would benefit normal operating performance but would have an unfavorable effect on loss-of-coolant accident performance.

b. Thorium Dioxide-Plutonium Dioxide Blanket Rod - The geometric and mechanical features of this blanket rod would be essentially identical to those typical of commercial fuel rods, but the blanket rod power would be considerably lower (5 kw/ft versus 13 kw/ft). Thus, no unique technological concerns would be expected for these rods. However, little information is available for fuel pellets composed of thorium dioxide and plutonium dioxide. Although the small amounts of

plutonium dioxide in such pellets would not be expected to significantly alter the characteristics of the pellet, this has not yet been demonstrated. In addition, the same considerations for the initial helium fill pressure described for the seed rods would apply to the blanket rods.

c. Uranium Dioxide-Thorium Dioxide Seed Rod - The geometric parameters of the seed rod would be similar to those of the standard commercial rod designs. Seed rod pellets would consist of uranium-235 dioxide and thorium. The uranium-thorium seed rod is used in the Shippingport LWBR core with uranium-233 as the fissile material. Irradiation test rods, in support of LWBR, used uranium-235 in place of uranium-233. Thus, much of the required technology is available from the LWBR core design and operating experience, LWBR irradiation test rods, and commercial core experience. In commercial application it might be possible to operate the fuel rod at a maximum power of approximately 16 kw/ft, taking advantage of the higher melting temperature of thorium, which is about 700°F higher than that of uranium in today's commercial cores. Measurements of the thermal conductivity of uranium-thorium fuel at high temperatures, and fuel rod irradiation tests would be required to confirm power capability.

3. Thermal and Hydraulic

a. CHF Characteristics - There have been no known CHF tests on geometries like those of the prebreeder core concepts which contain intermingled rods of two diameters. The closest known test geometry is that of the LWBR blanket test (Reference 9), where the test bundle had two areas with a different rod size (0.571 inch and a 0.520 inch diameter) in each area. The test data were predicted acceptably by available CHF correlations. However, previous experience has shown

that the accuracy of the CHF correlation can be affected by changes in geometry. Experimental CHF data would be required to develop a correlation specifically applicable to the geometry of the prebreeder core concept.

The effect on CHF of both turbulent fluid mixing and diversion fluid mixing at the grids is another consideration. Indications are that in commercial cores fluid mixing increases the power capability by 5 to 10 percent. The prebreeders would have smaller rod-to-rod spacings than commercial cores; this would tend to reduce mixing, but the prebreeders would have more grids which would tend to increase mixing. In addition, the diversion mixing would be strongly grid-design dependent. CHF tests conducted with prototype grids would serve to qualify, as a package, the CHF correlation, turbulent mixing model, and grid mixing model for the particular geometry and radial heat flux distribution tested.

b. Core Pressure Drop - In each prebreeder core concept, the fuel assemblies would contain seed and blanket rods with differing diameters, and this would produce several different types of hydraulic subchannels. Grid loss coefficients would depend on both grid geometry and the subchannel geometry. Each subchannel type would therefore have different flow resistance. The uncertainty in calculated pressure drop is judged to be on the order of 10 percent; the related uncertainty in calculated core flow and thermal capability is less than 1 percent. Flow measurements with the exact fuel assembly geometry would be required to confirm or refine the calculational method.

c. Rod Bow - The rods in the prebreeder fuel assemblies would have less rod-to-rod clearance and the seed rods would be smaller than the rods in a commercial core. The smaller rods would tend to exhibit an increased amount of

bow while the closer spacing would increase the severity of a given amount of bow. Use of more grids would tend to offset these effects, but would increase pressure losses. It is believed that the number of grids used in the prebreeder core concepts would be adequate to prevent unacceptable rod bow, but additional analysis and testing would be required to verify this. Special CHF tests would be required to experimentally verify the effect of the rod bow on the CHF power capability.

d. Reactivity Shutdown Margin Following Steam Line Break - The accident analyses presented in Reference 1 (Section 15.1) indicate that following a steam line break, the primary coolant temperature could drop briefly to 515°F, at which condition the reactivity shutdown margin would be a minimum; however, the core would remain shut down even with the most reactive control rod assembly (CRA) stuck in the fully-withdrawn position.

The nuclear calculations presented in Appendix B indicate that the prebreeder core concepts would have more shutdown margin at 515°F than would the reference core. However, a detailed analysis would be required to ensure sufficient negative reactivity in the control rods to maintain the reactor in a subcritical condition during a steam line break accident under all operating conditions and at all times in life. Additional discussion of shutdown margin is provided in Appendix B.

e. Loss-of-Coolant Accident - The loss-of-coolant accident (LOCA) is a design basis accident in current commercial power plants. Comparative calculations of the first 10 seconds of blowdown for a postulated LOCA in the reference plant have shown that the performance of the prebreeder core concepts would be very similar to that of the reference core. The results are taken to indicate that

reflood performance of the prebreeder core concepts also would be comparable to that of the reference core. However, detailed calculations of this portion of the accident would have to be performed. Since relative to the reference core the prebreeder core concepts would have rods of different sizes with different amounts of stored energy, it might be necessary to conduct tests to verify the reflood heat transfer characteristics of the prebreeder core concepts.

4. Mechanical

a. Reduced Diameter Control Rods and Control Rod Guide Tubes - These changes would affect significantly the hydraulic forces acting on the control rods and the hydraulically induced vibration characteristics of the rods, which in turn would affect rod wear and control rod scram rate. These effects would depend on dimensions, tolerances, alignment, and pressure losses. Conservative allowances are believed to have been made in choosing the control rod and guide tube parameters, but additional analyses would be necessary to confirm proper mechanical performance.

b. Grid Features - Relative to the reference core, there would be a larger number of welds in each of the grids for the 19 by 19 fuel assembly. Fabrication feasibility, including a procedure to control distortion during welding, would need to be demonstrated.

The two sizes of springs and dimples to accommodate the seed and blanket rods, and the method of installing rods in the grids, would also require development and testing.

c. Fuel Rod Bow - Considerations of fuel rod bow were mentioned in subsection 3 above in relation to thermal and hydraulic considerations. In the

area of mechanical considerations, a detailed bow analysis would be required including such factors as rod straightness, grid misalignments, grid spring characteristics at the beginning and end of life, relaxation, irradiation and thermal growth, and rod thrust loads.

d. Instrumentation Tube and Spacer Sleeve with Decreased Diameters -

The reduced inside diameter of the instrumentation tube would reduce the clearance to the instrument thimble. The thimble would be the pressure boundary for housing the flux sensing instrument which traverses the core length. The thimble would be retracted from the core during refueling. Ability to install and remove the thimble would depend on dimensions, tolerances, and alignment. Further evaluation would be required and might dictate changes in the instrument and thimble.

Spacer sleeves around the instrument tube would provide a positive spacing of the grids during normal operation. During shipment, supports might be required to avoid exceeding stress limits.

e. Structural - Structural evaluations of all reactor components would be required. The structural evaluations would include static and dynamic stress, deflection, and vibration analysis for shipping and handling loads, normal operating conditions, and plant transients, seismic loading, and LOCA's.

5. Fuel Cycle

Some special developments would be required in the areas of fabrication and reprocessing. The required developments would depend on the prebreeder core concept selected. The reprocessing requirements could also impact the fuel assembly mechanical design.

a. Fuel Fabrication - The seed fuel for prebreeder core Concept 3 would contain uranium-233 and uranium-232. Two of the daughter products

of uranium-232 (thallium-208 and bismuth-212) emit high energy gamma rays. The gamma radiation associated with this seed fuel would require remote fabrication in heavily shielded facilities. These facilities would be very similar to, if not identical with, the facilities which would be required for the fabrication of water-cooled breeder fuel. The blanket fuel for prebreeder core Concept 2 would contain plutonium. This might also require remote fabrication. Urania or oxide fuel with enrichment greater than that in current commercial cores would be required, and this might require modification to criticality control procedures. The highly enriched urania required for prebreeder core Concept 3 would require appropriate safeguard measures.

- b. Fuel Reprocessing - All three prebreeder core concepts are based on the requirement that the uranium-233 produced in the blanket fuel can be kept separate from the uranium in the seed fuel. For all three core concepts, one alternative would be to physically separate the seed rods from the blanket rods at the head end of the reprocessing operation. This would require compatibility between the fuel assembly mechanical design and the design of the fuel assembly disassembly stage of the reprocessing operation. For prebreeder core Concepts 1 and 2, an alternative to mechanical separation might be differential dissolution of the fuel. In this process the seed and blanket rods would be chopped simultaneously and charged into

a two-stage dissolver. The first stage would dissolve only one type of fuel; then the second stage would dissolve the other type. Developmental work would be required to determine if this is practical.

V. DEVELOPMENT AND TESTING REQUIREMENTS

To maximize the utility of these prebreeder core concepts as backfits, the need for new development was purposely minimized, but specific testing would be needed in some cases to confirm the performance prediction. Examples of the development and testing which would be anticipated are given below.

A. Mechanical Testing Requirements

The required mechanical test program for the prebreeder core concept using the 19 by 19 fuel assembly would be expected to be comparable to the test program that B&W planned and initiated in support of the 17 by 17 fuel assembly design as reported in Reference 10. The following is an outline of a series of tests that would be required to confirm the analyses of the prebreeder core concepts:

1. Upper end fitting
 - a. Axial and lateral load-deflection tests
 - b. Holddown spring load-deflection test
2. Grid
 - a. Load-deflection tests of springs, dimples, guide tube/grid support, and instrumentation tube/grid support
 - b. Tensile test of TIG weld
 - c. Load-deflection tests of intermediate and end grids
 - d. Fatigue tests of intermediate and end grids
 - e. Slip load tests of intermediate and end grids
 - f. Buckling test of end grid
 - g. Impact test on all grids.
 - h. Cam and key test

- i. Development of intersecting joint designs, fabrication procedures, and manufacture of feasibility mockups using welded and/or brazed joints for Inconel-718, or possibly AM-350 or Zircaloy materials
 - j. Static load testing of grid joints identified in Item i above
 - k. Fatigue testing of grid joints identified in Item i above
 - l. Corrosion testing of grid joints identified in Item i above
 - m. Irradiation testing of grid joints identified in Item i above.
- 3. Control rod guide tube and instrument tube joints
 - a. Load-deflection test
 - b. Load-to-failure test.
- 4. Instrument tube spacer sleeve
 - a. Load-deflection test including end effects of grid support
 - b. Axial buckling load including end effects of grid support.
- 5. Fuel assembly - Two fuel assemblies would be required, one with spring set at BOL conditions and one set at EOL conditions.
 - a. Lateral load-deflection test with restraints at both end fittings for each fuel assembly
 - b. Frequency and damping test with restraints at both end fittings for each fuel assembly
 - c. Lateral load-deflection test with restraint at lower end fitting for each fuel assembly and load applied at upper end fitting
 - d. Torsional stiffness test with restraint at lower end fitting for each fuel assembly and load applied at upper end fitting
 - e. Axial stiffness test with restraints at both end fittings for fuel assembly with beginning of life conditions.

6. Fuel assembly with control rod assembly (CRA) - Fuel assembly, plenum control rod drive structure, control rod assembly, and control rod drive mechanism. Testing to be performed at room temperature and at reactor operating conditions.
 - a. Scram insertion rate test
 - b. Control rod wear test at nominal and maximum misaligned conditions for design rod travel
 - c. Wear and fretting test of nominal and maximum misaligned conditions for design lifetime
 - d. Vibration test with various flow rates at various temperatures.
7. Fuel assembly with axial power shaping rod assembly (APSRA) - Same tests are required as Item 6 using an axial power shaping rod assembly in place of a control rod assembly.
8. Fuel assembly with burnable poison rod assembly (BPRA) and with orifice rod assembly (ORA) - Same tests are required for each of these assemblies as Items 6.b and 6.c.
9. Upper plenum with control rod guide structure
 - a. Fabrication development to establish fabrication processes and procedures to minimize distortions
 - b. Hydraulic testing of plenum area with octagonal guide tubes to evaluate effect of cross flows.

B. Nuclear Testing Requirements

Confirmation of nuclear performance results would require low-power critical experiments on core mockups. Although many critical experiments involving each of the fuel materials used in the prebreeder core concepts have been performed

and documented, there are few, and in some cases, no experiments for the combinations of fuel (e.g., plutonium in thorium or uranium-238 and thorium) used in the prebreeder core concepts. Comparisons of calculated and experimental results from the critical experiments would be used to confirm the relative worth of uranium-235, uranium-233, and fissile plutonium as well as uranium-238 and thorium in these lattices and thereby assure the needed accuracy of core lifetime and conversion ratio predictions. In addition, these tests would be used to provide calculational-to-experimental correction factors to be applied to core power distributions, shutdown reactivities, and reactivity coefficients.

C. Fuel Testing Requirements

The fuel rods would be generally similar to existing designs (Shippingport LWBR or commercial rods) in both the mechanical aspects and the materials being considered. The fuel pellet temperature in the uranium dioxide-thorium dioxide seed rods of prebreeder core Concept 3 would be higher than those in the fuel rods of the Shippingport LWBR core, and there are limited data available for the 2 w/o plutonium dioxide-thorium dioxide fuel pellet of core Concept 2. Except for these areas the necessary fuel technology has been used in operating reactors. Pressurized fuel rods are in use in the commercial nuclear industry and therefore should not be a development item; however, the test program would need to include fuel swelling and burst tests to expand the data base for estimating LOCA performance. In addition, depletion of a fuel assembly might be required to acquire: (1) irradiation fuel rod proof tests covering the possible range of manufacturing tolerances and expected operating conditions, (2) fuel rod grid interaction tests (rod bowing confirmation tests), and possibly (3) fuel rod clad material testing including both annealed and cold-worked materials.

In the case of the thorium dioxide-plutonium dioxide pellet, only a limited amount of data covering material properties and irradiation performance is available. This is also true for the thorium dioxide-uranium dioxide pellet for operation at temperatures greatly above those in the Shippingport LWBR fuel. Tests to confirm basic material properties such as fuel swelling, gas release, and additional data on thermal conductivity and thermal expansion would be needed. The limited data that are available indicate that thorium dioxide-plutonium dioxide is expected to perform comparably to thorium dioxide-uranium dioxide.

D. Thermal and Hydraulic Testing Requirements

The thermal and hydraulic testing program which would be required for the prebreeders is similar to that which was necessary for the reference core. That program is presented in Reference 10. The principal required tests would be:

1. Reflood heat transfer test - The purpose of this test would be to obtain heat transfer and liquid carry-over data that could be used to develop and qualify correlations for predicting the reflood portion of the LOCA. This would be a full length emergency cooling heat transfer test (FLECHT) with the geometry of the prebreeder fuel assemblies. The two different rod sizes and decay heat power levels (seed and blanket) would need to be included in the geometrically prototypical test bundle.
2. CHF test - The purpose of this test would be to qualify the CHF correlation for the specific prebreeder geometry and grid concept selected. The rod bundle size would be in the range of 16 to 25 rods and be prototypical as far as radial power distribution, grids, and geometry. More than one test assembly might be required to adequately

test all of the cell types (for example, 3 seed - 1 blanket, 2 seeds - 2 blankets, 1 seed - 3 blankets, 3 seeds - 1 guide tube) in the fuel assembly. The testing might include rods in contact if analyses show this to be a reasonable limiting case. The ranges of variables would be:

Pressure	1500 to 2400 psia
Inlet Temperature	420 to 630°F
Mass Velocity	$0.5 \text{ to } 3.0 \times 10^6 \text{ lb/hr-ft}^2$

3. CHF with rod bowing test - The purpose of this test would be to investigate the effect that rod bowing has on the CHF capability of the prebreeder core concepts. An analysis of the CHF margin and the amount of bowing anticipated in each cell type would indicate which cell types should be tested. It might be necessary to test more than one assembly.
4. Core pressure drop test - The purpose of this test would be to provide basic pressure drop data on a full-sized fuel assembly. The data would be used to calculate form loss coefficients and confirm the basic hydraulic calculations used to analyze normal reactor operation as well as accident conditions.
5. Fuel assembly lift test - The purpose of this test would be to determine the flow rate that would lift the fuel assemblies off the lower end fitting. The data would be used to verify the holddown spring design and to provide input information for the postulated LOCA analysis.
6. Upper plenum arrangement pressure drop test - The purpose of this test would be to determine how the various geometric arrangements in the upper plenum affect the core pressure drop. All possible outlet geometries

with and without control rods would need to be tested at core operating pressure and temperature. The data would provide the information needed to calculate the flow split in the core among the various fuel assemblies.

7. Control rod wear test - The purpose of this test would be to verify that the design of the control rod assembly, the control rod guide structure, and the fuel assembly control rod guide tubes is adequate. The control rods in one fuel assembly would be cycled through one fuel assembly design lifetime of normal rod withdrawals, normal rod insertions, and emergency scrams at operating pressure and temperature.
8. Assembly wear and life test - The purpose of this test would be to demonstrate overall fuel assembly acceptability. A complete prototype fuel assembly could be flow-tested at core operating pressure and temperature for at least 4000 hours. At about 1000-hour intervals, the fuel assembly would be inspected for wear and fretting. During the final inspection all components would be examined in detail and wear measurements made.
9. Rod vibration test - The purpose of this test would be to determine the amount of vibration of the fuel rods. A prototypical fuel assembly would be flow tested over a wide range of flow rates and temperatures. Several instrumented rods in the assembly would be monitored for vibrations. The results would be used in the flow-induced vibration and fretting analysis. Dynamic stiffness and damping information obtained would be used in the seismic analysis.
10. Visual evaluation - The purpose of this test would be to confirm vibration and flow characteristics. A prototypical fuel assembly would be tested

in a low-pressure and low-temperature loop over a wide range of flow rates. High speed photography and visual observations would be used to determine the vibratory response of the various components in the fuel and control rod assemblies. The test results would be used as input to the flow-induced vibration and fretting analysis.

E. Materials Testing Requirements

Other than fuel, only one material not in present use in commercial PWR's has been identified. It is possible that AM-350 would be advantageous for a grid material since the tighter fuel rod spacing might require a material with better stress relaxation properties than Inconel-718. AM-350 was used for grids in the Shippingport LWBR and intensive testing has been completed in characterizing its material properties.

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APPENDIX A
MECHANICAL CONCEPTS

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APPENDIX A

MECHANICAL CONCEPTS

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APPENDIX A

MECHANICAL CONCEPTS

I. INTRODUCTION

The backfit prebreeder mechanical concepts as defined in this appendix are based on modifications of the Babcock & Wilcox (B&W) Standard 241 17 by 17 fuel assembly designed for use in the reactor system reported in Reference A-1. Design, development, analysis, and testing of this fuel assembly by B&W was initiated in 1972. The B&W-241 was chosen as a reference because at the time the study was initiated it was an example of a large commercial design having a somewhat lower power density and potentially more flexibility for modification than the other large pressurized water reactor commercial designs. The B&W-241 system is no longer offered, and no B&W-241 plants were or are being built. The B&W-205 reactor also utilizes a 17 by 17 fuel assembly, which is similar to that of the B&W-241 except for an increase in length due to the larger plenum at the top of each fuel rod. The essential difference between a B&W-241 core and a B&W-205 core is that the fuel assembly and fuel rod power ratings of the 205 are higher and the fuel assembly flow is also higher to provide the same total core power and flow in 205 fuel assemblies as in 241 fuel assemblies.

In developing the 19 by 19 prebreeder fuel assembly concepts, use of the 17 by 17 fuel assembly design features to the maximum extent possible would minimize the changes required to backfit the 19 by 19 prebreeder fuel assembly concept into a plant designed to accept the 17 by 17 fuel assembly. The developmental testing program required to demonstrate structural adequacy and to fully qualify the mechanical performance of the prebreeder core concepts would also be minimized. Much of the testing and analysis from the reference system would be applicable to the prebreeder core concepts.

This appendix includes a description with illustrations that define the conceptual mechanical arrangement of prebreeder core concepts for backfit application into the reference system. The specific areas described include the major components that would be redesigned for the prebreeder core concepts. The major components would be the fuel assemblies, control assemblies, and the plenum assembly. In addition, this appendix identifies the major areas for further development and the testing program that would be required prior to licensing the prebreeder core concepts.

II. CORE ARRANGEMENT

As shown in Figures A-1, A-2, and A-3, the number of fuel assemblies and overall fuel assembly size of the prebreeder core concepts would be identical to those of the reference core. The prebreeder core concepts would be sized to fit the reference core support assembly including the core basket and lower grid assembly. In addition, the same in-core instrumentation arrangement and reactor vessel head assembly including the control rod drive system would be used without any changes. The arrangement of rods in each fuel assembly would be increased from a 17 by 17 array as used in the reference core and shown in Figure A-4, to a 19 by 19 array with separate seed and blanket rods located on a generally alternating pattern as shown in Figure A-5. Figure A-5, Sheets 1, 2, and 3 illustrate three fuel assembly cross sections that were evaluated for this report. Each fuel assembly would contain 28 control rod guide tubes and one instrumentation tube assembly. The reference core uses 24 control rod guide tubes. All fuel assemblies would be identical in mechanical design and would be mechanically interchangeable in any core location. The major components that would have to be redesigned for the prebreeder core concepts are the fuel assembly, plenum

*THESE COMPONENTS WOULD BE NEW FOR THE BACKFIT APPLICATION ALL OTHERS WOULD BE EXISTING COMPONENTS

Figure A-1. Prebreeder Reactor Assembly

Figure A-2. Cross Section Through Prebreeder Reactor Vessel Internals and Core

Figure A-3. Cross Section Through Prebreeder Reactor Vessel Nozzles

Figure A-4. B&W Fuel Assembly Section, 17 by 17 Rod Array

Figure A-5, Sheet 1. Prebreeder Fuel Assembly 19 by 19 Rod Array, Core Concept 1

Figure A-5, Sheet 2. Prebreeder Fuel Assembly 19 by 19 Rod Array, Core Concept 2

Figure A-5, Sheet 3. Prebreeder Fuel Assembly 19 by 19 Rod Array, Core Concept 3

assembly, control rod guide tubes, control rod assemblies, axial power shaping rod assemblies, burnable poison rod assemblies, and orifice rod assembly.

III. FUEL ASSEMBLIES

The complete core for the prebreeder core concepts would have 241 fuel assemblies arranged in a square lattice, the same as in the reference B&W core, as shown in Figure A-2. Each fuel assembly would accept any control assembly (control rod assembly, axial power shaping rod assembly, burnable poison rod assembly, or orifice rod assembly). As in the reference core, the prebreeder fuel assembly serial number, located on a top edge of the upper end fitting, would be used with a core location map to ensure proper location and orientation of the fuel assembly.

As shown in Figures A-5 and A-6, the prebreeder fuel assembly would consist of 332 fuel rods, 28 control rod guide tubes, one instrumentation tube assembly, ten spacer grids, and two end fittings. The guide tubes, spacer grids, and end fittings would form the basic structure to support the seed and blanket rods in the 19 by 19 array. The fuel rods would be supported within the spacer grids by contact surfaces, springs and dimples, which would be integral with the grid for each cell. The guide tubes would be rigidly attached to the upper and lower end fittings. A comparison of the fuel assembly components, materials, and dimensions of the prebreeder and the reference B&W core are listed in Table A-1.

Each fuel assembly would be installed and seated on the lower grid of the core support assembly as shown in Figure A-1. Alignment would be ensured by index pins located on the lower grid engaging holes in the lower fitting of the fuel assembly. The fuel assemblies would be restrained axially by the plenum assembly located above the fuel assemblies. Index pins on the plenum assembly would align and provide lateral support at the upper end of each fuel assembly. The

Figure A-6. Prebreeder Fuel Assembly

TABLE A-1. CORE MECHANICAL PARAMETERS .

	<u>Reference Core</u>	<u>Prebreeder. Core Concepts</u>
Fuel Assembly (FA)		
Structural Design and Control Rod Design	Canless with finger control	Canless with finger control
Number of FA's per Core	241	241
Pitch, in.	8.587	8.587
Length, in.	176.6 long	176.6 long
Cross section, in.	8.536 square	8.536 square
Control Rod Guide Tube		
Number per FA	24	28
Material	Zircaloy-4	Zircaloy-4
Size OD by ID, in.	0.465 by 0.430	0.426 by 0.391
Instrument Tube		
Number per FA	1	1
Material	Zircaloy-4	Zircaloy-4
Size OD by ID, in.	0.420 by 0.390	0.390 by 0.360
Instrument Tube Spacer Sleeve		
Number per Instrument Tube	7	9
Material	Zircaloy-4	Zircaloy-4
Internal diameter, in.	0.428	0.397
External shape	Circular	Square
External dimensions, in.	0.480 diameter	0.428 by 0.428
Grids		
Number per FA	8	10
Material	Inconel-718	Inconel-718

TABLE A-1 (Cont)

	<u>Reference Core</u>	<u>Prebreeder Core Concepts</u>
Strip Material Thickness, in.	0.014 interior 0.018 exterior	0.014 interior 0.018 exterior
Lattice	Square	Square
Type of Construction	"Egg-crate"	"Egg-crate"
Welding method	TIG	TIG
Type of Grid Contact with Fuel Rod	Leaf spring and dimples	Leaf spring and dimples
Fuel Rods		
Number per FA	264	332-See Figure A-5 for number of seed and blanket rods.
Cladding Material	Zircaloy-4	Zircaloy-4
Dimensions		
OD by ID, in. length, in.	0.379 by 0.332 158.125	-
Seed OD by ID, length	-	See Figure A-7
Blanket OD by ID, in. length, in.	-	0.379 by 0.339 158.125
Pitch, in.	0.501	0.449
Active Fuel Length, in.	143	143
Control Rod Assembly (CRA)		
Number of CRA's per Core	76	76
Number of Control Rods per CRA	24	28
Control Rod OD, in.	0.390	0.351
Cladding Thickness, in.	0.019	0.017
Cladding Material	304 SS cold-worked	304 SS cold-worked

TABLE A-1 (Cont)

	<u>Reference Core</u>	<u>Prebreeder Core Concepts</u>
Poison Material, %	80 Ag, 15 In, 5 Cd	80 Ag, 15 In, 5 Cd
Poison Section Length, in.	134	134
Control Rod Stroke, in.	139	139
Axial Power Shaping Rod Assembly (APSRA)		
Number of APSRA's per Core	8	8
Number of APSR's per APSRA	24	28
APSR OD, in.	0.390	0.351
Cladding Thickness, in.	0.019	0.017
Cladding Material	304 SS cold-worked	304 SS cold-worked
Poison Material, %	80 Ag, 15 In, 5 Cd	80 Ag; 15 In, 5 Cd
Poison Section Length, in.	36	36
APSR Stroke, in.	139	139
Burnable Poison Rod Assembly (BPRA)		
Number of BPRA's	148 (first cycle)	0*
Number of BPR's per BPRA	24	28**
BPR Dimension OD, in.	0.380	0.341
Cladding Thickness, in.	0.032	0.029
Cladding Material	Zircaloy-4 cold-worked	Zircaloy-4 cold-worked
Orifice Rod Assembly (ORA)		
Number of ORA's per Core	9 (first cycle)	157***
Number of OR's per ORA	24	28

*Section III.A of Appendix B

**Only if BPRA's were to be required

***Section IV.D of this appendix

TABLE A-1 (Cont)

	<u>Reference Core</u>	<u>Prebreeder Core Concepts</u>
Orifice Rod OD, in.	0.410	0.371
Orifice Rod Material	304 SS annealed	304 SS annealed

pins would also interact with the spring retainer plate located in the upper end fitting of the fuel assembly to load the helical springs, thereby providing a positive downward clamping force to the top of each fuel assembly.

Clearance between fuel assemblies for installation and removal would be nominally 0.051 inch, the same as used in the reference core. It should be noted that the Westinghouse commercial reactors are designed with a smaller nominal clearance of 0.040 inch which could be considered for use in the prebreeder core concepts. Both B&W and Westinghouse have completed assembly testing of their respective designs. In addition, both have successfully completed some refueling operations for fuel assemblies with such clearances. From a recent Westinghouse refueling operation after one cycle, as reported in Reference A-2, results of dimensional measurement program indicated the maximum fuel assembly bow was much greater than the nominal clearance between fuel assemblies and, therefore, grid-to-grid contact in the assembled core would occur. However, no problems were experienced in reseating the fuel assemblies back into the core. Reference A-2 concluded that, due to the inherent flexibility of the assembly, the fuel assembly may tend to straighten when hanging freely from a crane. Post-irradiation examination after one cycle of operation for the B&W Mark B 15 x 15 fuel assembly design, as reported in Reference A-3, also indicated a maximum fuel assembly bow much greater than the nominal clearance between fuel assemblies. These fuel assemblies were also reinstalled in the core, apparently without experiencing any problems.

Results of additional refuelings have not yet been published. Although the fuel assembly design used in the B&W Standard 241 is the Mark D 17 x 17, similar to the Mark C 17 x 17 used in the more recent B&W 205 reactors, the clearance between fuel assemblies for the Mark D and C is the same as the Mark B design and successful refueling experience with the Mark B design provides some assurance that the clearance used in the Mark D design is adequate.

Based on the assembly testing program and the successful refueling and operating experience compiled by B&W and Westinghouse, it would appear that a nominal clearance as small as 40 mils between fuel assemblies could be technically justified due to the flexibility of the fuel assembly. In the event a problem is encountered in the refueling operation, a spreader could be used to increase clearances around any one fuel assembly. On this basis, it is judged that the 51-mil nominal clearance for the prebreeder core concepts would be adequate.

A. Fuel Rods

Both seed and blanket fuel rods shown schematically in Figure A-7 would be internally pressurized with helium to enhance fuel rod reliability by reducing fuel-cladding interaction and to provide improved heat transfer within the rods, resulting in lower fuel temperatures. The fuel would be in the form of sintered and ground pellets in both seed and blanket rods. Pellet ends would be dished to minimize differential thermal expansion between the fuel and cladding and would be chamfered to minimize ridging of the cladding at fuel pellet interfaces.

A spring would be located above the fuel column to separate the fuel from the fuel rod upper end cap. The spring would maintain the fuel column in place during shipping and handling. In operation, the spring would permit axial differential growth and thermal expansion between the fuel and the cladding. A spring would also be located below each fuel column to axially locate the

	B & W STD. 241	SHIPPING PORT LWBR	PRE-BREEDER CORE CONCEPTS		
			#1 NO Pu RECYCLE	#2 WITH Pu RECYCLE	#3 WITH HE URANIA
CLAD OD	0.379	—	—	—	—
SEED	—	0.306	0.349	0.339	0.349
BLANKET	—	—	0.379	0.379	0.379
CLAD ID	0.332	—	—	—	—
SEED	—	0.267	0.302	0.292	0.302
BLANKET	—	—	0.339	0.339	0.339
CLAD MAT'L	ZIRC 4	ZIRC 4	ZIRC 4	ZIRC 4	ZIRC 4

FOR PELLET DETAILS SEE TABLE 2

Figure A-7. Fuel Rods

bottom of the fuel column and separate the fuel from the lower fuel rod end cap. Rigid spacers would be used both above and below the fuel column between the spring and the fuel column to provide insulation for the spring from the heat of the fuel pellets.

Fuel fission gas release would be contained within voids in the pellets, the radial gap between the pellets and cladding, and the void around the springs at both ends of the fuel rods. The cladding on fuel rods would be subjected to external hydrostatic pressure, gradually increasing internal pressure, thermal stresses, vibration, and the effects of differential expansion of the fuel and cladding caused by thermal expansions and by fuel growth due to irradiation effects. In addition, the properties of the cladding would be influenced by thermal and irradiation effects. The final design details of the rods, such as end plugs, weld configuration and procedures, material specifications, spring, spacers, and dimensional requirements would be selected based on evaluation of existing and proven technology from the reference core, from the Shippingport Light Water Breeder Reactor (LWBR) program and from additional testing identified in Section V of the main text. The prebreeder seed rod would have a slightly larger diameter than the LWBR seed rods, while the blanket rod would be essentially the same (pellet-to-clad gap is less) as the rods in the reference B&W-241 core.

B. Upper End Fitting

The details of the upper end fitting for the prebreeder fuel assemblies would be made comparable to those in the reference core to ensure compatibility with the reference core coupling and handling equipment. Differences would be limited to the penetrations required for attachment of the 28 control rod guide tubes and the instrumentation tube. As noted in Table A-1 these tubes would be slightly smaller in diameter than those in the reference core. In addition, the

plan view location of the tubes would be different for the 19 by 19 array. The outer control rods would be located farther away from the fuel assembly center and the inner control rods would be located closer to the fuel assembly center. The maximum change in location would be 0.267 inch for the outermost control rods that would be located near the index pin alignment holes. The alignment holes would be used to position the upper end of the fuel assembly in the upper core grid plate, an integral part of the plenum assembly. Each upper end fitting would include holddown springs and a spring retainer plate to provide a positive holddown force to oppose the hydraulic force and thereby clamp the fuel assembly between the lower grid assembly and the plenum assembly. The spring sizing would differ from that in the reference core, and would be based on the hydraulic forces for the prebreeder concepts. The lower plate of the upper end fitting would be a grid-type structure with penetrations for coolant flow and for attaching guide tubes.

C. Lower End Fitting

The lower end fitting for the prebreeder concepts would be designed to (1) position the fuel assembly on the lower grid plate of the core support assembly, (2) provide a guide path for in-core instrumentation to the instrumentation tube, and (3) permit coolant flow directly into the bottom of the fuel assembly. Positioning of the fuel assembly would be accomplished by alignment holes in the lower end fitting which fit onto index pins on the lower grid plate. Positioning of the fuel assembly would therefore align the instrumentation penetration with the instrument guide tubes in the core support assembly.

The details of the lower end fitting would be similar to those in the reference core to ensure compatibility with the lower grid assembly and the in-core instrument guide tubes. These components are integral with the core support

assembly which would be reused for the prebreeder core concepts. Changes from the reference core would include relocation of the guide tubes, sizing the penetrations for the guide tubes and instrumentation tube, and sizing of the openings and ribs to provide a support surface for the bottom end of the fuel rods for the 19 by 19 array.

D. Spacer Grids

Typical of commercial reactors, the spacer grids for the prebreeder concepts would be made from strips which are slotted and fitted together to form an "eggcrate" arrangement as shown in Figure A-8. The arrangement of strips would be permanently assembled as for the reference core by a tungsten-inert-gas (TIG) weld at the upper and lower surface at each intersection formed by the strips. Each strip would contain springs and support dimples and, if required, coolant mixing vanes to ensure adequate flow mixing between flow channels and between fuel assemblies. Peripheral strips would aid in guiding the fuel assembly by providing a lead-in to avoid interference with adjacent assemblies during handling operations. The square cells formed by the strips would provide support for fuel rods in two perpendicular directions through contact surfaces on each of the four cell walls. These contact surfaces would include two springs on two adjacent walls and four dimples on two opposite walls. Springs and dimples would be sized to accommodate seed or blanket rods. The springs, dimples, and mixing vanes, if required, would be integrally stamped into each strip. All grids would be identical except for the two end grids which would have the peripheral strips extended to provide a skirt that would be mechanically attached to the upper and lower end fittings.

Features of the reference B&W grid design include TIG welded intersections, provision to "key" the springs open during fuel rod assembly, and no attachments

Figure A-8. Prebreeder Grid

between guide tubes and grid. The TIG welded intersections have been used in other B&W reactor designs and have been used in the loss of fluid test (LOFT) program. This feature has been extensively tested by B&W and in the LOFT program, as reported in References A-4 and A-5. The provision to "key" the springs open during fuel rod assembly permits fuel rod insertion into the grids without frictional restraint, thus minimizing installation stresses and damage to the fuel rod clad. After the rods are installed, the "keying" is released and springs contact and secure the fuel rods. As discussed in the following section, the grids are spaced axially by spacer sleeves located between grids and fitting around the instrumentation tube. The guide tubes are not clamped to the grids. As a result, the grids are free to accommodate differential thermal expansion and irradiation growth between the fuel rods and guide tubes. As discussed in Reference A-6, post-irradiation examination of fuel through the end of one cycle from one reactor plant indicates rod bow to be less than 22-percent closure for 99.95 percent of the data and less than 40-percent closure for all data. As stated in Reference A-3, considerably more inspection data at the end of two cycles has been obtained and is being evaluated. These data indicate fuel rod bow to be small and the resistance of the fuel to bowing is partially attributed to the grid design.

The prebreeder grid concept based on the reference design is shown in Figure A-8. Each grid would consist of 40 strips with 20 perpendicular to 20, to form the 19 by 19 lattice of square cells. The size of the springs and dimples would be different for seed and blanket fuel rods to accommodate the difference in rod diameters. The minimum spring force requirements on the fuel rod at the beginning of life (BOL) would be set high enough to withstand shipping and handling loads without rod shifting, and to minimize possible fretting. The

maximum spring force requirements at BOL would be limited to avoid exceeding stress limits of the fuel rod cladding at contact surfaces and to allow thermal expansion of the fuel rods without excessive distortion or buckling. The end of life (EOL) spring force would be set to be adequate to preclude rod lift-off from the dimples, considering the rod dimensional changes and spring relaxation.

E. Guide Tubes, Instrumentation Tube, and Spacer Sleeves

The mechanical arrangement of the guide tubes and instrumentation tubes as shown in Figure A-6 for the prebreeder concepts would be the same as for the reference core except for the differences required to accommodate the 19 by 19 array with 28 control rods. These differences would result in small changes in plan view location, slightly smaller diameter tubing, and a greater number of guide tubes.

The guide tubes would provide a continuous guide for each control rod through the full length of travel. That is, the control rod could not be withdrawn from the guide tube during operation. The guide tubes would also function as the structural member for the fuel assembly. Flanged and threaded sleeves would be welded to each end of the guide tubes and the guide tubes would be mechanically attached to each end fitting by lock-welded nuts to complete the basic structure. The spacer grids would provide transverse positioning of the guide tubes. In addition to the small changes in plan view location of the guide tubes, the outside diameter of the guide tubes would be decreased from 0.465 to 0.426 inch and the number of guide tubes would be increased from 24 to 28. Since the guide tubes would be used for structural supports, the total cross sectional area of the prebreeder guide tubes would be approximately 6 percent greater than in the reference core to accommodate the additional fuel assembly weight which would be increased by approximately 5 percent.

The instrumentation tube assembly for the prebreeder concepts would be designed to guide, position, and contain the in-core instrumentation guide thimble within the fuel assembly. The instrumentation tube would be located on the axial centerline of the fuel assembly and would be retained axially at the lower end fitting by a retainer sleeve. The upper end of the tube would be laterally restrained by a small clearance fit into the upper end fitting. Transverse location of the instrument tube would be provided by the spacer grids. The decreased pitch of the 19 by 19 array would result in decreasing the outside diameter of the instrumentation tube from 0.420 to 0.390 inch and decreasing the inside diameter from 0.390 to 0.360 inch. It is judged that the decreased inside diameter change could be accommodated by decreasing the clearance between the thimble and inside diameter of the instrumentation tube, decreasing the thimble wall thickness, decreasing the clearance between the instrument and thimble, or decreasing the outside diameter of the instrument.

The spacer sleeves for the prebreeder concepts would be designed to fit around the instrument tube and would be located between spacer grids. The spacer sleeves would provide a positive spacing between grids to prevent axial movement of the grids during primary coolant flow through the fuel assembly. Changes relative to the reference core that would be required for the prebreeder concepts include an increased number of spacers because of the increased number of grids and sizing to accommodate the decreased fuel rod pitch of the 19 by 19 array. The outside dimension of the spacer sleeves would be decreased from 0.480 inch OD as used in the reference core to 0.428 inch square. The inside diameter would be decreased from 0.428 to 0.397 inch.

IV. CONTROL ASSEMBLIES

Since all fuel assemblies for the prebreeder concepts would be mechanically identical and each would contain 28 guide tubes, a control-type assembly would be located in each fuel assembly. Each fuel assembly would contain either a moveable control rod assembly (CRA), a moveable axial power shaping rod assembly (APSRA), a fixed burnable poison rod assembly (BPRA), or a fixed orifice rod assembly (ORA). These assemblies would be designed to be interchangeable and could be installed into any fuel assembly. The reactivity of the core under operating conditions would be controlled by 76 CRA's, 8 APSRA's, and soluble boron. In the fuel assemblies which would not contain a CRA or APSRA, a fixed BPRA or fixed ORA would be used. Relative to the reference core, changes in the control assemblies that would be required for the prebreeder core concepts include plan view relocation of rods, slightly smaller diameter rods as required by the 19 by 19 rod array, and a greater number of control rods per assembly. A description of each of the assemblies follows.

A. Control Rod Assemblies

Each CRA (Figure A-9) for the prebreeder concepts would have 28 control rods, a stainless steel spider, and a female coupling. The 28 control rods would be attached to the spider by a nut threaded to the upper shank of each rod. After assembly, all nuts would be lock-welded. The control rod drive would be coupled to the CRA by a bayonet connection. Full-length guidance for the CRA would be provided by the control rod guide tube located in the upper plenum assembly and by the fuel assembly guide tubes. The CRA and guide tubes would be designed with adequate flexibility and clearances to permit freedom of motion within the fuel assembly guide tubes throughout the stroke. As stated in Reference A-6, the diametral clearance of 0.04 inch between the fuel assembly guide

Figure A-9. Prebreeder Control Rod Assembly

tubes and rods for the 17 x 17 Mark C fuel assembly design is considered adequate to guide the rods without adversely affecting the scram insertion time. Reference A-6 also states that B&W plans to test the Mark C clearance at simulated reactor conditions. Use of the same clearance for the prebreeder concepts would also require testing to qualify the scram insertion rate.

The CRA guide paths would be designed to ensure that these assemblies would not disengage from the fuel assembly guide tubes during operation. The guidance would be provided by close tolerance indexing of the fuel assembly's upper end fitting with the plenum assembly's grid plate. Each control rod would contain a section of neutron absorber material, an alloy of silver-indium-cadmium (Ag-In-Cd), which would be clad in type-304 stainless steel tubing. Stainless steel end pieces would be welded to the tubing to form a water- and pressure-tight container for the absorber material. The stainless steel tubing would provide structural strength for the control rods and would prevent corrosion of the absorber material. A spring spacer similar to that in the fuel rod would be used to prevent absorber motion within the cladding during shipping and handling and to permit differential expansion in service.

The control rods would be designed to withstand all operating loads, including those resulting from hydraulic forces, thermal gradients, and pressure differential effects. The Ag-In-Cd alloy poison does not yield a gaseous product under irradiation and does not increase the internal pressure of the control rod during life. The rod cladding would be designed to be freestanding for its operational life when subjected to the effects of external pressure, temperature, and irradiation. The cladding-to-absorber gap would be sized to accommodate swelling of the absorber material without stressing the cladding beyond yield.

B. Axial Power Shaping Rod Assemblies

The APSRA (Figure A-10) for the prebreeder concepts would be identical in physical configuration to the full-length absorber CRA, but would have absorber in the lower portion of the rod only. The section of tubing above the absorber would be vented so that it would always be filled with water; therefore, the pressure differential across the tube wall would be negligible. The female couplings of the APSRA and the CRA would have slight dimensional differences to ensure that each type of rod could be coupled only to the correct type of drive mechanism.

C. Burnable Poison Rod Assemblies

The BPRA's (Figure A-11) for the prebreeder concepts would be designed to be installed in particular fuel assemblies to provide some reactivity control. Like the CRA and APSRA, each BPRA would have 28 poison rods, a stainless steel spider, and a female coupling. The 28 poison rods would be mechanically attached to the spider by lock-welded threaded fasteners. The hub of the spider would contain a helical coil spring and a cruciform-shaped locking plate. After the installation of the BPRA in the fuel assembly, it would be clamped in place by the grid plate of the upper plenum assembly by applying a load to the cruciform-shaped locking plate, thereby compressing the coil spring. The diametral clearance between the burnable poison rods and guide tubes would provide adequate flow for cooling the rods. The burnable poison rod would be designed to withstand all operating life condition loads including hydraulic forces, thermal gradients, and pressure differentials.

Figure A-10. Prebreeder Axial Power Shaping Rod Assembly

Figure A-11. Prebreeder Burnable Poison Rod Assembly

D. Orifice Rod Assembly

Each ORA (Figure A-12) for the prebreeder concepts would have 28 solid stainless steel rods mechanically attached to a spider as previously described for the CRA. The hub of the spider and method of clamping the assembly would be identical to the arrangement described for the BPRA. The rod would be extremely short and engage only the upper section of the guide tubes above the active core region. These assemblies would be used to control bypass leakage in those fuel assemblies which do not contain a CRA, APSRA, or BPRA.

V. PLENUM ASSEMBLY

The plenum assembly (Figures A-13 and A-14) for the prebreeder concepts would be located directly above the reactor core and would be removed as a single component before refueling. The assembly would consist of a plenum cover, upper grid, CRA guide tube structures, and a flanged plenum cylinder with openings for reactor outlet flow. The plenum cover would be a forging with holes to accept the CRA guide tube structures. The forging would be supported from the core support upper flange. The plenum cylinder would be attached to and supported from the plenum cover.

The CRA guide tube structures would be welded to the plenum cover and bolted to the upper grid. The guide tube structures would be designed to provide CRA guidance, protect the CRA from the effects of coolant crossflow, and provide structural attachment of the grid assembly to the plenum cover.

Each guide tube structure would consist of a mounting flange, 16 perforated slotted tubes, and 12 sets of tube segments which would be oriented and attached to a series of spacers to provide continuous guidance for the CRA full-stroke travel (Figure A-14). Clearances would be designed to accommodate misalignment

*EXCEPT FOR DIMENSIONS AFFECTED BY ROD
DIAMETER CHANGES, ALL OTHER DIMENSIONS
WOULD BE THE SAME AS IN THE B&W STD 241 CORE

Figure A-12. Prebreeder Orifice Rod Assembly

Figure A-13. Prebreeder Plenum Assembly

Figure A-14. Prebreeder Plenum Assembly (Typical Bottom View of Upper Grid, Drawn Approximately to Scale)

between the guide tube structure and the fuel assemblies. An octagonal guide tube structure would be required to accommodate the relocation of the four outermost control rods farther away from the fuel assembly centerline. Fabrication of the octagonal tube with a longitudinal weld seam would require the development of a fabrication process and procedures to minimize distortions.

The plenum cylinder would be a large cylindrical section having flanges on both ends to connect the cylinder to the plenum cover and the upper grid. Holes in the plenum cylinder would provide a flow path for coolant water. The upper grid would be a perforated plate which would locate the lower end of the individual CRA guide tube structure relative to the upper end of a corresponding fuel assembly. The grid would be bolted to the lower flange of the plenum cylinder. Locating keyways in the cover forging of the plenum assembly would be designed to engage the locating keyways in the reactor vessel flange to align the plenum assembly with the reactor vessel, the reactor closure head control rod drive penetrations, and the core support assembly. The bottom of the plenum assembly would be guided by the inside surface of the core support assembly.

The major changes in the plenum assembly required for the prebreeder concepts would be in the CRA guide tube structure as a result of the smaller diameter control rods, greater number of control rods, and the plan view relocation of the control rods. These changes would have a significant effect on the hydraulic forces acting on the control rods, hydraulically induced vibration characteristics of the rods, and the resultant wear on the rods. These changes could also affect the control rod scram insertion rate. These effects have not been evaluated and constitute areas requiring further development.

VI. CORE SUPPORT ASSEMBLY

For the backfit prebreeder concepts, it would not be necessary to make any changes to the core support assembly. This section describing the core support assembly (Section VI of this Appendix), is quoted from Reference A-1 and is included to provide a description of all reactor components.

"The core support assembly consists of the upper core support weldment, lower core support weldment, lower grid assembly, flow distributor, in-core instrument guide tubes, and surveillance specimen holder tubes. Static loads from the assembled components and fuel assemblies and dynamic loads from CRA trip, hydraulic flow, thermal expansion, seismic disturbances, and LOCA loads are all carried by the core support assembly. The core support assembly components are as follows:

A. Upper Core Support Weldment

"This weldment is a flanged cylinder that mates with the reactor vessel opening. Its forged top flange rests on a circumferential ledge in the reactor vessel's closure flange. The cylindrical wall has two nozzle openings which seal to the reactor vessel's outlet nozzles by the differential thermal expansion between the stainless steel upper core support and the carbon steel reactor vessel, and openings for eight core support vent valves. The outlet nozzle seal surfaces are finished and fit to a predetermined cold gap providing clearance for installation and removal of the core support assembly. At reactor operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the reactor vessel or the internals.

B. Lower Core Support Weldment

"This weldment consists of rolled plate and a forged ring which is welded to form one large cylinder whose top is welded to the upper core support weldment. The core basket, which rests on the lower grid, fits inside the lower core support weldment. The lower core support weldment is flanged at the lower end, and the lower flange is bolted to the lower grid assembly. All bolts are captured and locked after final assembly. Coolant flow is downward along the outside of the core support assembly and upward through the fuel assemblies. Small portions of the coolant flow upward through the space between the lower core support cylinder and the core barrel, and between the core barrel and the vertical plates forming the core basket assembly.

C. Core Basket Assembly

"This assembly consists of vertical plates held by horizontal former plates which fit inside the core barrel. The fuel assemblies, core basket assembly, lower grid assembly, and flow distributor are supported by the core support assembly.

D. Lower Grid Assembly

"The lower grid assembly provides alignment and support for the fuel assemblies, supports the core basket and flow distributor, and aligns the in-core instrument guide tubes with the fuel assembly instrument tubes. The lower grid consists of two grid structures which are separated by short columns and surrounded by a forged cylinder. The upper structure is a perforated plate and the lower structure is a machined forging. This forging rests on and is bolted to the lower flange of the core support assembly. A perforated flat plate midway between the two grid structures properly distributes coolant before it enters the core. The fuel assemblies are aligned by pins bolted to the lower grid perforated plate.

E. Flow Distributor Assembly

"The flow distributor assembly is a perforated dished head with an internal perforated flat plate which is bolted to the bottom flange of the core support assembly. This assembly supports the in-core instrument guide tubes and distributes the inlet coolant entering the bottom of the core.

F. Thermal Shielding

"The core basket assembly and the core support cylinder provide thermal shielding to prevent excessive neutron exposure of the reactor vessel wall and to reduce gamma heat generation in the reactor vessel wall.

G. In-Core Instrument Guide Tube Assembly

"These assemblies guide the in-core instrument assemblies from the instrument penetrations in the reactor vessel's bottom head to the instrument tubes in the fuel assemblies. Horizontal clearances are provided between the reactor vessel's instrument penetrations and the instrument guide tubes in the flow distributor to accommodate misalignment.

H. Core Support Vent Valves

"The core support vent valves are located in the upper core support weldment just above the outlet nozzle area. These vent valves establish direct communication between the upper core region and the inlet annulus should the pressure differential across the core support (from the inlet annulus to the core) become negative. Each vent valve consists of a valve body and a disc which is hung closed in its normal position. The face of the valve body is inclined 5 degrees from the vertical to ensure a positive seal of the disc. A positive pressure differential across the disc during normal operation further ensures the positive seal of the disc. The vent valve is bolted on a forged reinforcing ring which is welded to the upper core support weldment."

VII. ASSESSMENT OF CHANGES

A. Increased Fuel Assembly Weight

Total weight of the fuel assemblies for the prebreeder core concepts would be approximately 5 percent greater than for the reference core. The increased weight would affect the axial and lateral loads and load carrying capability of the control rod guide tubes and threaded fasteners attaching the guide tubes to the end fittings. Since the outside diameter of the guide tubes would be decreased from 0.465 to 0.426 inch and the number of guide tubes would be increased from 24 to 28, the inside diameter of the prebreeder guide tubes have been sized to provide approximately 6 percent more cross sectional area than in the reference core.

B. Increased Core Pressure Drop

Due to the increased number of grids and the reduced flow area in the prebreeder core concepts, the total core pressure drop would increase by 59 percent compared to the reference core. Core pressure drop is discussed in Section III of Appendix D. The increased loading on the fuel assembly would result in the following changes.

1. Instrumentation tube spacer sleeves - Since the external space available for the spacer sleeves would be reduced due to the decreased cell pitch and it would be desirable to minimize the decrease of the inside diameter of the instrumentation tube, the spacer sleeve for the prebreeder core concepts would have a square rather than a circular outer perimeter to achieve increased cross sectional area. The dimensions of the prebreeder spacer sleeve given in Table A-1 would provide approximately 60 percent more cross sectional area than in the reference core to

accommodate the increased core pressure drop. Standard methods for axially loaded columns were used to calculate the margin to buckling (ratio of minimum stress which causes buckling to maximum stress expected) for both the prebreeder and the reference core spacer sleeves. The calculations accounted for the increased core pressure drop, the increased spacer sleeve cross section area and the reduced spacing between grids. The calculated margin to buckling of the prebreeder spacer sleeves was approximately 18 percent greater than for the reference core spacer sleeves.

2. Grids -. The increased pressure drop would result in additional loading on the grids. Even though the grid thickness would be the same as in the reference design and the number of grid strips would be increased for the 19 by 19 array, the height of the grid should be increased by approximately 20 percent to maintain the same design margin to bending for the increased pressure loading.
3. Fuel assembly holddown springs - The increased pressure drop would result in additional loading on the fuel assembly holddown springs. To maintain the same margin in the holddown capability during operating conditions, it would be necessary to increase the preload from the holddown springs. This would require a new spring design. It is noted that the B&W B-SAR-205 design reported in Reference A-7 for a 17 by 17 rod array has a total unrecoverable core pressure drop of 18.5 psi, comparable to the prebreeder core concepts. Amendment 2 of Reference A-7 indicates that the holddown springs provide an adequate downward force on the fuel assembly at design system flow rates. Since the prebreeder core

concepts would have more springs than the B-SAR-205 design, it would appear that a new spring could be designed for the prebreeders within the available space.

C. Spacer Grid Spring Design

The decreased cell pitch for the 19 by 19 rod array in the prebreeder core concepts would result in decreasing the space available for the spring and the opening between springs used in the reference B&W design for a camming key to hold the springs open for rod installation. If it were assumed that the same spring force would be used in the prebreeder core concepts as used by B&W (same spring design), then the space available for designing the camming key would be decreased. This would impose a constraint on the ability to design an adequate camming key. In addition, the use of two rod sizes (seed and blanket) would impose a further constraint. A significant effort would be required to design a new spring compatible with the camming key feature for two rod sizes. If these constraints would preclude this feature for the prebreeder core concepts, then the rods could be installed by overcoming the frictional restraint of the grid springs as done in other commercial reactors and in the Shippingport LWBR.

D. Grid TIG Welds

To assess the adequacy of the TIG welded grid for the prebreeder core concepts, detailed features of the B&W grid would be required. For example, if the total reaction from the spring load were applied to the weld as shear and bending moments, a rather large minimum weld penetration depth would be required. This would necessitate stringent requirements on the TIG welding process and would require testing to evaluate the effects on the spring characteristics. However, if locating dimples that provided an additional reaction to the spring force were used, a much smaller weld penetration depth would be required. These details

for the B&W designs, apparently proprietary, are not available. Based on the acceptable operating performance of the TIG welded grid design used for the B&W 15 x 15 fuel assembly and the fact that it is the reference design for the B&W 17 x 17 fuel assembly, it is judged that the TIG welded grid could be designed for use in the prebreeder core concepts.

E. Increased Fuel Rod Loading on Grids

For an earlier 15 x 15 fuel assembly design, Reference A-4 presents the fuel assembly stress and deflection analysis for a postulated loss of coolant accident and for seismic loading. These results indicate a margin of 13 percent in the grid welds as a result of the total maximum possible fuel rod slip load. Since the prebreeder concepts would have a larger number of fuel rods and grids, a detailed fuel assembly stress and deflection analysis for LOCA and seismic loading would be required to justify adequacy of the grids.

A preliminary assessment of the effects of the increased number of fuel rods and grids for the prebreeder core concepts indicates that the fuel rod slip load and therefore the grid weld loads would be greater for the prebreeder core concepts than for the reference core. The method used for this assessment is similar to that used for the Shippingport LWBR grid load calculations. Assuming the same spring force would be required in the prebreeder core concepts as used in the reference core and using ratios of the number of rods, the number of grids, and the number of grid panels, the weld force for the prebreeder core concepts would be expected to be 42 percent greater than for the reference core. To maintain the same stress in the weld, it would be necessary to increase the prebreeder grid height by approximately 20 percent since the grid thickness would be the same as used in the reference core.

F. Abnormal Loading on Guide Tubes

The results presented in Reference A-4 indicate that the control rod guide tubes in the 15 x 15 fuel assembly design have a 27 percent margin to critical buckling during postulated loss of coolant accidents or seismic loading. An assessment of guide tube loading based on taking ratios of the changes in fuel assembly weight, size and number of control rod guide tubes, and the spacing between grids indicates the margin to critical buckling would be increased to 55 percent for the prebreeder core concepts when compared to the 15 x 15 fuel assembly designs. Again, a detailed analysis of fuel assembly stresses and deflections for hypothetical loss of coolant accidents and seismic loading would be required to justify the adequacy of the prebreeder guide tubes.

G. Spacing Between Grids

The use of reduced fuel rod diameters and reduced spacing between fuel rods for the prebreeder core concepts would require more grids to ensure that vibratory fretting wear and fuel rod bow would be maintained at acceptable levels. Relative to vibration and fretting wear at the contact surfaces between fuel rods and springs/dimples, the natural frequency of the reduced diameter prebreeder rods with nine grids would be approximately 15 percent higher than that for the reference core, and for ten grids would be approximately 90 percent higher than that for the reference core. The higher natural frequency would reduce the effect of hydraulically induced vibrations and thereby minimize wear. Seed rods for the Shippingport LWBR have a natural frequency that is approximately 60 percent higher than that for the reference core.

The potential for rod bowing resulting in rod-to-rod or rod-to-structure contact for the prebreeder core concepts having the reduced diameter fuel rods

and reduced spacing between fuel rods would be minimized by increasing the number of grids from eight to ten, thus decreasing the span between grids. In order to analyze bowing in the prebreeders, a relationship between grid span, rod diameter, and rod spacing was developed assuming that the primary bowing mechanism is due to the radial energy gradient and that bow is augmented by the thrust load which is proportional to the number of grids. This relationship included normalizing the energy gradient for commercial reactors, assuming bow as a specified fraction of the spacing, and fitting the parameters to the commercial design. Use of this relationship indicates that ten grids would compensate for the reduced rod diameter.

VIII. MAJOR AREAS FOR FURTHER DEVELOPMENT

Major areas for further development in extending the technology developed for the 17 by 17 fuel assembly to the 19 by 19 fuel assembly for the prebreeder core concepts are outlined below.

A. Reduced Diameter Control Rods and Control Rod Guide Tubes

These changes would significantly affect the hydraulic forces acting on the control rods and the hydraulically induced vibration characteristics of the rods, which in turn would affect rod wear and control rod scram rate. These effects would depend on dimensions, tolerances, alignment, and pressure losses. Additional analyses would be required to confirm proper mechanical performance.

B. Grid Features

Relative to the reference core there would be a larger number of welds in each of the grids for the 19 by 19 fuel assembly. Fabrication feasibility, including a procedure to control distortion during welding, would need to be demonstrated.

The two sizes of springs and dimples to accommodate the seed and blanket rods would also require development and testing.

C. Fuel Rod Bow

The reduced rod diameter and the reduced spacing between rods required for the prebreeder core concepts would increase the concern of rods touching; however, the use of more grids would tend to offset this concern. The bow analysis is dependent on rod straightness, grid misalignments, grid spring characteristics at beginning and end of life, relaxation, irradiation, thermal growth, rod thrust loads, and power distributions. A detailed bow analysis would be required as the core concepts were developed.

D. Instrumentation Tube and Spacer Sleeve with Decreased Diameters

The reduced inside diameter of the instrumentation tube would reduce the clearance to the instrument thimble in the prebreeder core concepts. This thimble is the pressure boundary housing the flux sensing instrument which traverses the core length and it would be retracted from the core during refueling. Ability to install and remove the thimble would depend on dimensions, tolerances, and misalignment. Further evaluation would be required and might dictate the need for changes in the instrument and the thimble.

E. Others

Other areas of concern that would require technical justification to ensure the adequacy of the prebreeder core concepts are structural evaluation of all reactor components, fuel assembly bow, and hydriding. The structural evaluation would include a static and dynamic stress, deflection, and vibration analysis for shipping and handling loads, normal operating conditions and plant transients, seismic loading, and hypothetical accidents.

Post-irradiation fuel assembly bow has been measured and is reported in Reference A-3 for the B&W 15 by 15 fuel assembly. This reference states that all parameters measured showed changes that were either as designed or conservative compared to analytical predictions. Fuel assembly bow is primarily due to thermal gradients and end moments imposed on the fuel assembly which are further augmented by axial loads. For the 19 by 19 prebreeder fuel assembly, the primary bowing mechanism (thermal gradients and end moments due to the support system) should be comparable to that in the B&W 15 by 15 or 17 by 17 cores. The augmented bow (increased axial loading due to core pressure drop of 18.6 psi compared to 11.6 psi) in the prebreeder would be greater than in the reference 17 by 17 core but should be comparable to that in the 17 by 17 core described in Reference A-7 (core pressure drop of 18.5 psi). Therefore, it would appear that the prebreeder fuel assembly bow would be acceptable and would not be any greater than in the B&W B-SAR-205 core. However, a fuel assembly bow analysis for the prebreeder core concepts would be required as part of the fuel assembly technical justification as the concepts were developed.

The potential for hydriding exists in a number of locations in the prebreeder core concepts. Several locations of particular interest would be: (1) mechanical connections of the Zircaloy control rod guide tube to end fittings, (2) the Zircaloy instrumentation tube spacer sleeve/instrument tube to the grid, (3) the stainless steel control rod cladding which moves within the Zircaloy guide tubes, and (4) the contact surfaces between the Zircaloy fuel rod cladding and the grids. These locations are not unique to the prebreeder core concepts but also occur in the reference core. As identified in Reference A-1, B&W has

1 taken steps to minimize the possibility of hydriding and it is assumed that B&W has satisfactorily demonstrated adequate resistance to hydriding in all areas of concern. A similar program to ensure satisfactory hydriding performance would be required for the prebreeder core concepts.

IX. TEST PROGRAM

The mechanical test program for the prebreeder core concepts using the 19 by 19 fuel assembly would be comparable to the test program that B&W has planned and is performing in support of the 17 by 17 fuel assembly design of References A-7 and A-8.

The following is an outline of a series of tests typically performed to confirm the analyses and to demonstrate the adequacy of the core features.

1. Upper end fitting
 - a. Axial and lateral load-deflection tests
 - b. Holddown spring load-deflection test
2. Grid
 - a. Load-deflection tests of springs, dimples, guide tube/grid support, and instrumentation tube/grid support
 - b. Tensile test of TIG weld
 - c. Load-deflection tests of intermediate and end grids
 - d. Fatigue tests of intermediate and end grids
 - e. Slip load test of intermediate and end grids
 - f. Buckling test of end grid
 - g. Impact test on all grids
 - h. Cam and key test

- i. Development of intersecting joint designs, fabrication procedures, and manufacture of feasibility mockups using welded and/or brazed joints for Inconel-718, or possibly AM-350 or Zircaloy materials
 - j. Static load testing of grid joints identified in Item i above
 - k. Fatigue testing of grid joints identified in Item i above.
 - l. Corrosion testing of grid joints identified in Item i above.
 - m. Irradiation testing of grid joints identified in Item i above.
3. Control rod guide tube and instrument tube joints
- a. Load-deflection test
 - b. Load-to-failure test.
4. Instrument tube spacer sleeve
- a. Load-deflection test including end effects of grid support
 - b. Axial buckling load including end effects of grid support.
5. Fuel assembly - Two fuel assemblies are required, one with a set of springs at beginning of life conditions and one with a set at end of life conditions.
- a. Lateral load-deflection test with restraints at both end fittings for each fuel assembly
 - b. Frequency and damping test with restraints at both end fittings for each fuel assembly
 - c. Lateral load-deflection test with restraint at lower end fitting for each fuel assembly and load applied at upper end fitting
 - d. Torsional stiffness test with restraint at lower end fitting for each fuel assembly and load applied at upper end fitting

- e. Axial stiffness test with restraints at both end fittings for fuel assembly with beginning of life conditions.
- 6. Fuel assembly with control rod assembly - Fuel assembly, plenum control rod guide structure, control rod assembly, and control rod drive mechanism. Testing to be performed at room temperature and at reactor operating conditions.
 - a. Scram insertion rate test
 - b. Control rod wear test at nominal and maximum misaligned conditions for design rod travel
 - c. Wear and fretting test of nominal and maximum misaligned conditions for design lifetime
 - d. Vibration test with various flow rates at various temperatures.
- 7. Fuel assembly with axial power shaping rod assembly (APSRA) - Same tests are required as Item 6 using APSRA in place of a control rod assembly.
- 8. Fuel assembly with burnable poison rod assembly and with orifice rod assembly - Same tests are required for each of these assemblies as Items 6.c and 6.d.
- 9. Upper plenum with control rod guide structure
 - a. Fabrication development to establish fabrication process and procedures to minimize distortions
 - b. Hydraulic testing of plenum area with octagon shaped guide tubes to evaluate effect of cross flows.

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APPENDIX B
NUCLEAR CONCEPTS

E. G. Johnson

APPENDIX B

NUCLEAR CONCEPTS

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APPENDIX B

NUCLEAR CONCEPTS

I. INTRODUCTION

This appendix describes the nuclear calculations and results for the three prebreeder core concepts and for the reference core. It includes a detailed description of the calculational models used to develop nuclear cross sections and to calculate core performance, and presents calculated power distributions, neutron balances, isotopic mass flows, and kinetics parameters. All calculations were made using nominal fuel pellet and cladding compositions and dimensions. Detailed nuclear calculations for all three prebreeder core concepts and for the reference core (described in Reference B-1) were made using the diffusion theory-depletion computer program PDQ08 (Reference B-2).

Section II of this appendix describes the model and the nuclear data used to generate few group cross sections which were used in the PDQ08 calculations. The cross section generation was performed entirely with Monte Carlo calculations of quarter fuel assemblies.

Section III describes the diffusion theory model used to obtain nuclear performance estimates. Quarter-core radial slice diffusion theory calculations were made using a geometric model in which each quarter fuel assembly was represented as a separate region. This model was used to determine the core fissile loading required for a desired core lifetime, to define a fuel management strategy which would result in acceptable power peaking performance, to estimate core temperature coefficients and stuck control rod assembly shutdown reactivity margins, and to determine core kinetics parameters.

Section IV describes the depletion model, the depletion chains, and depletion periods used in the calculations. Section V describes the calculation of the production rate of uranium-232 whose daughters are responsible for the high energy gamma radiation associated with uranium-233 and with irradiated thorium.

Section VI describes nuclear calculations performed by the authors for the reference core. Pertinent core parameters were obtained from Reference B-1. These calculations provided both a basis for qualifying the calculational model through comparison of calculated nuclear performance indicators with those reported in Reference B-1, and a reference point for evaluation of nuclear performance of the prebreeder core concepts.

Section VII presents the nuclear performance results for the three prebreeder core concepts. Mass flows, uranium-233 production rates, kinetics parameters, and power peaking factors are presented. All three prebreeder core concepts would have negative temperature coefficients throughout core life, and would have acceptable stuck control rod assembly shutdown reactivity.

II. CROSS SECTION MODEL

A. Basic Nuclear Data

A summary of the basic data used in generating few group cross sections for the prebreeder core concepts and for the reference core is given below.

1. Atomic masses - The atomic masses are given in Table B-1. These masses were obtained from Reference B-3.
2. Densities - The maximum theoretical densities used for slightly enriched uranium dioxide, highly enriched uranium dioxide, and thorium dioxide are 10.96 gm/cc, 10.85 gm/cc, and 10.0 gm/cc, respectively. The values for uranium dioxide are based on Reference B-4 and the value for thorium dioxide is based on Reference B-5.

TABLE B-1. ATOMIC MASSES

<u>Isotope</u>	<u>Atomic Mass</u>
Th-232	232.04
Pa-231	231.04
Pa-233	233.04
U-232	232.04
U-233	233.04
U-234	234.04
U-235	235.04
U-236	236.05
U-238	238.05
Pu-238	238.05
Pu-239	239.05
Pu-240	240.05
Pu-241	241.06
Pu-242	242.06
Hydrogen	1.0079
B-10	10.013
Oxygen	15.999
Neutron	1.0087

-
3. Avogadro's number - The value of Avogadro's number used was 0.60220×10^{24} atoms/gm-mole (Reference B-6).
 4. Control rod composition - The control rod composition used in the prebreeder analyses was the same as that specified for the reference core. This composition is 80 percent silver, 15 percent indium, and 5 percent cadmium (Reference B-1).
 5. Cross section libraries - Nuclide cross section libraries used in the calculations are shown in Table B-2. Whenever possible, these cross sections were taken from the evaluated nuclear data file (ENDF/B) maintained by the National Nuclear Data Center at Brookhaven National Laboratory. Table B-2 identifies the ENDF/B material number of each set of nuclide properties taken directly from ENDF/B, the characterization of materials that were constructed from nuclides

available from ENDF/B, and the characterization of data used for nuclides not available from ENDF/B (protactinium-231, uranium-232, and neptunium-239). The metallurgical composition of alloys represented in the calculations is also given.

B. The Monte Carlo Model

Major details of the nuclear cross section model are given in Table B-3 and are summarized below.

1. The Few-Group Model

Monte Carlo calculations were used to provide spectrum averaged few group cross sections for use in diffusion theory calculations. A four energy group model similar to that used for the Shippingport Light Water Breeder Reactor (LWBR) calculations at Bettis was used for all diffusion theory calculations described. This model represented all neutrons in the reactor with cross sections appropriate to three fast energy groups and one thermal energy group. The three fast groups covered the energy ranges 10 MeV to 0.82 MeV, 0.82 MeV to 5.53 keV, and 5.53 keV to 0.625 eV. The thermal group covered the range from 0.625 eV to 0.0 eV. The 0.625 eV to 5.53 keV energy group was selected to cover the energy range where the major portion of resonance absorption occurs. The energy group above 0.82 MeV was chosen to provide appropriate representation of fast fission and of the neutron slowing down source distribution.

2. Thermal Hydrogen Scattering Kernels

The thermal scattering kernels were 25 energy group Haywood kernels generated for the temperature of interest.

TABLE B-2. CROSS SECTION DATA FILES

I. Data Files Obtained from ENDF/B

<u>Nuclide</u>	<u>Alphanumeric ID</u>	<u>ENDF/B Material No.</u>
Thorium-232	Th-232	1296
Protactinium-233	Pa-233	1119
Uranium-233	U-233	1260
Uranium-234	U-234	1043
Uranium-235	U-235	1261
Uranium-236	U-236	1163
Uranium-238	U-238	1262
Plutonium-238	Pu-238	1050
Plutonium-239	Pu-239	1159
Plutonium-240	Pu-240	1105
Plutonium-241	Pu-241	1106
Plutonium-242	Pu-242	1161
Hydrogen	H1	1148
Oxygen	O-16	1276
Boron-10	B-10	1273
Iron	Fe	1192
Nickel	Ni	1123
Chromium	Cr	1121
Niobium	Nb	1189
Aluminum	AL	1193
Cadmium	Cd	1281
Xenon-135	Xe-135	1026
Neodymium-147	Nd-147	1237
Promethium-147	Pm-147	1238
Promethium-148	Pm-148	1239
Promethium-149	Pm-149	1240
Samarium-149	Sm-149	1027
Samarium-150	Sm-150	1244
Samarium-151	Sm-151	1245

TABLE B-2 (Cont)

II. Characterization of Constructed Data Files

<u>Nuclide</u>	<u>Alphanumeric ID</u>	<u>Resonance Integral⁽¹⁾ (barns)</u>	<u>Thermal (barns) 2200 m/sec</u>
Protactinium-231	Pa-231	0.43314 + 3	0.20414 + 3
Uranium-232	U-232	0.53395 + 3	0.15907 + 3
Neptunium-239	Np-239	0.47765 + 3	0.45269 + 2
Silver (Natural)	Ag	0.75665 + 3	0.63443 + 2
Indium (Natural)	In	0.31106 + 4	0.19415 + 3
Zircaloy-4	Zirc-4	0.12689 + 1	0.20835 + 0
Steel			
AM-350		0.25970 + 1	0.47929 + 1
SS-304		0.17404 + 1	0.29328 + 1
Inconel-718		0.24085 + 1	0.36375 + 1

III. Constituents of Alloys

<u>Constituents</u>	<u>Weight % of Constituent in</u>			
	<u>AM-350</u>	<u>SS-304</u>	<u>Inc-718</u>	<u>Zirc-4</u>
Cr	16.7	18.5	20.0	0.1
Ni	4.3	9.5	53.0	-
Fe	75.2	70.5	21.5	0.21
Nb	-	-	5.0	-
Al	-	-	0.5	-
Mn	0.8	1.0	-	-
Si	0.24	0.5	-	-
B	0.01	-	-	-
Mo	2.75	-	-	-
Zr	-	-	-	98.1075
O	-	-	-	0.125
Hf	-	-	-	0.0075
Sn	-	-	-	1.45

(1) Data are infinitely dilute integral values.

TABLE B-3. MAJOR FEATURES OF CROSS SECTION
CALCULATIONAL MODEL

<u>Item</u>	<u>Description and Comments</u>
Number of Fast Energy Groups in Diffusion Theory Model	Three fast energy groups. Group 1 covers the range above 0.82 MeV; group 2 covers the range from 5.53 keV to 0.82 MeV; and group 3 covers the range from 0.625 eV to 5.53 keV.
Number of Thermal Groups in Diffusion Theory Model	One thermal group covering the range from 0.0 eV to 0.625 eV. Thermal cross sections used in design calculations were determined using Monte Carlo spatially dependent spectrum model.
Fission Source Spectra	U-235: ENDF/B material 1261. U-233: ENDF/B material 1260.
Thermal Hydrogen Scattering Kernels	25 energy group Haywood kernels, temperature dependent.
Resonance Shielding and Thermal Self-Shielding	Accounted for internal to RCP01 calculations. No shielding factors are applied externally to nuclear cross sections.

3. Monte Carlo Calculations

The basic computer program used in the calculation of prebreeder and reference core few group nuclear cross sections was the RCP01 Monte Carlo program (Reference B-7). Input consists of physical and geometric data for all components within the region of interest. Output consists of nuclide reaction rates and neutron fluxes over the various compositions in the solution region. Nuclear data were prepared for input into RCP01 by RCPL1 (Reference B-8). RCPL1 uses resonance parameters obtained from the ENDF/B library, calculates the Doppler broadened resonance cross sections, and stores this data along with the multi-group cross section data in a job library suitable for use in RCP01. RCP01 has

the capability to perform either fixed source or eigenvalue calculations using Monte Carlo techniques. To determine the relative reaction rates over the energy range of interest (0.0 eV to 10 MeV), a large number of neutrons are followed from their birth by fission to their destruction by absorption or their leakage from the system. Average values of important reaction rates as well as statistical estimates concerning the accuracy of each are calculated.

Effective few group microscopic cross sections for all of the nuclides appearing in Table B-2 were generated using RCP01. The RCP01 model used 31 energy intervals to describe the energy range from 0 eV to 10 MeV with these intervals being further subdivided into as many as 1000 subintervals to allow accurate representation of resonance cross sections. The cross section libraries used in RCPL1 are listed in Table B-2.

Resonance shielding factors are not needed for the diffusion theory cross sections since resonance self-shielding and mutual shielding effects are treated exactly in the Monte Carlo calculations, and these effects are therefore implicitly included in the few group cross sections obtained from the Monte Carlo calculation.

The RCP01 neutron transport calculations determine the space and energy dependent neutron flux spectrum. In these calculations $\Sigma^C(E)$, the energy dependent macroscopic capture cross section, is normally calculated (for each composition) internally by the program, using the atom densities and energy dependent microscopic cross sections for each nuclide in the composition. $\Sigma^C(E)$ is used, together with macroscopic scattering and fission cross sections, to determine neutron collision points. Because of the program limit of 10 nuclides with resonance

cross sections, $\Sigma^C(E)$ was calculated externally (using the RCPL1 program) for seed and blanket pellets. Separate calculations were made for heavy metal captures and for fission product captures. Each resulting $\Sigma^C(E)$ was entered into the program as a microscopic capture cross section assigned to an artificial or "composite" nuclide with unit atom density. Through this artifice the effect of all nuclides on the neutron spectrum was treated exactly, despite the program limits.

Following the calculation of the space and energy dependent neutron flux spectrum, editing was done for each nuclide individually to obtain spectrum averaged few group cross sections for the nuclide.

RCP01 calculations were made for quarter fuel assembly cells with explicit representation of the location, size, and composition of seed and blanket rod cladding and fuel pellets, guide tubes, and water. To account for the dependence of few group cross sections on depletion, these cell calculations were made using nuclide inventories appropriate to each fuel management zone of each core. Average nuclide inventories for each fuel management zone at the middle of a cycle (i.e., 3500 EFPH after the start of a cycle) were obtained from preliminary PDQ08 depletion calculations for the appropriate core, made using depletion independent few group cross sections. Tables B-4 and B-5 identify the nuclides employed in the various calculations. A separate set of nuclides was used in RCP01 calculations for prebreeder core Concept 3 to permit determination of separate few group cross sections in the seed and blanket for nuclides appearing in both locations. As shown in Tables B-4 and B-5, the fission cross sections of all important fissile nuclides were explicitly included in the neutron transport calculation.

TABLE B-4. NUCLIDES USED IN RCP01 CALCULATIONS
OF PREBREEDER CORE CONCEPTS 1 AND 2

Nuclide Number	Neutron Transport RCP01	RCP01's for Cross Section Edits ⁽⁵⁾		
		First	Second	Third
1	Th-232 Fission	Pu-238 Fission	U-238 Capture	Nd-147
2	U-233 Fission	Pu-240 Fission	Np-239	Pm-147
3	U-235 Fission	Th-232 Capture	Pu-238 Capture	Pm-148M
4	U-238 Fission	Pa-231	Pu-239 Capture	Pm-148
5	Pu-239 Fission	Pa-233	Pu-240 Capture	Pm-149
6	Pu-241 Fission	U-232	Pu-241 Capture	Sm-149
7	Seed Pellet ⁽¹⁾	U-233 Capture	Pu-242	Sm-150
8	Blkt Pellet ⁽²⁾	U-234	Inconel-718	Sm-151
9	Fission Products ⁽³⁾	U-235 Capture	Stainless Steel 304	Residual fission ⁽⁴⁾ product
10	Zircaloy-4	U-236	Zircaloy-4	Zircaloy-4
11	Hydrogen	Hydrogen	Hydrogen	Hydrogen
12	Oxygen	Oxygen	Oxygen	Oxygen
13	Boron-10	Boron-10	Boron-10	Boron-10
14	AM-350	AM-350	AM-350	AM-350
15	Xe-135	Xe-135	Xe-135	Xe-135

(1) Composite nuclide represents capture cross sections of all nuclides in seed pellet.

(2) Composite nuclide represents capture cross sections of all nuclides in blanket pellet.

(3) Composite nuclide represents absorption cross sections of all fission product isotopes.

(4) Composite nuclide represents all fission product nuclides not represented explicitly.

(5) The edit RCP01 calculations determine reaction rates for the various nuclides using the neutron spectrum generated in the neutron transport RCP01.

TABLE B-5. NUCLIDES USED IN RCP01 CALCULATIONS
OF PREBREEDER CORE CONCEPT 3

Nuclide Number	Neutron Transport RCP01	RCP01's for Cross Section Edits ⁽⁶⁾			
		First	Second	Third	Fourth
1	Th-232 Fission	Pu-238 Fission	U-234	B Th-232 Capture	Nd-147
2	U-233 Fission	Pu-240 Fission	U-235 Capture	B Pa-231	Pm-147
3	U-235 Fission	Pu-241 Fission	U-236	B Pa-233	Pm-148M
4	Pu-239 Fission	U-238 Fission	U-238	B U-232	Pm-148
5	B Th-232 Fission ⁽¹⁾	B U-235 Fission	Pu-238 Capture	B U-233 Capture	Pm-149
6	B U-233 Fission	Th-232 Capture	Pu-239 Capture	B U-234	Sm-149
7	Seed Pellet ⁽²⁾	Pa-231	Pu-240 Capture	B U-235 Capture	Sm-150
8	Blanket Pellet ⁽³⁾	Pa-233	Pu-241 Capture	B U-236	Sm-151
9	Fission Products ⁽⁴⁾	U-232	Pu-242	Stainless Steel	Residual Fission Product ⁽⁵⁾
10	Zircaloy-4	U-233 Capture	Np-239	Inconel-718	Zircaloy-4
11	Hydrogen	Hydrogen	Hydrogen	Hydrogen	Hydrogen
12	Oxygen	Oxygen	Oxygen	Oxygen	Oxygen
13	Boron-10	Boron-10	Boron-10	Boron-10	Boron-10

TABLE 5 (Cont)

<u>Nuclide Number</u>	<u>Neutron Transport RCP01</u>	<u>RCP01's for Cross Section Edits⁽⁶⁾</u>			
		<u>First</u>	<u>Second</u>	<u>Third</u>	<u>Fourth</u>
14	AM-350	AM-350	AM-350	AM-350	AM-350
15	Xe-135	Xe-135	Xe-135	Xe-135	Xe-135

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- (1) Nuclide identifications preceded by a "B" indicate that this nuclide appears only in the blanket rods.
- (2) Composite nuclide represents capture cross sections of all nuclides in the seed pellet.
- (3) Composite nuclide represents capture cross sections of all nuclides in blanket pellet.
- (4) Composite nuclide represents absorption cross sections of all fission product nuclides.
- (5) Composite nuclide represents all fission product nuclides not represented explicitly.
- (6) The edit RCP01 calculations determine reaction rates for the various nuclides using the neutron spectrum generated in the neutron transport RCP01.

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The neutron transport RCP01 calculations which determined the neutron flux spectrum were eigenvalue calculations with symmetry boundary conditions. The calculations typically processed approximately 86,000 neutron histories. The initial source was assumed to be spatially flat within the fuel pellets and was given a uranium-235 fission source energy distribution. Reaction rates were edited in the four group energy scheme described above.

Output from RCP01 consists of isotopic reaction rates, total slowing down densities, total neutron fluxes, and diffusion coefficients in each of the four energy groups for each of the compositions specified in the problem, as well as for the entire problem solution region. The data output from RCP01 was translated into few group nuclear cross sections in a tableset form suitable for use in PDQ08. The tablesets generated for use in the quarter core radial calculations contained cross sections averaged over an entire quarter-fuel assembly cell. The method used for generating the few group cross sections was as follows:

1. Diffusion coefficients obtained from RCP01 were adjusted to preserve core leakage and age to thermal energy.
2. Macroscopic removal cross sections were determined by dividing the groupwise total slowing down density in the appropriate region by the groupwise flux in that region.
3. Microscopic few group cross sections (σ_c , σ_f , and $\nu\sigma_f$) were calculated from the reaction rates for the nuclides of interest and from region fluxes. The capture and fission cross sections were added to obtain absorption cross sections.

The resulting few group cross sections were combined in tables containing macroscopic transport and removal cross sections and microscopic absorption

and fission cross sections for all nuclides of interest. For convenience in calculating moderator and temperature coefficients, the removal and transport cross sections were treated as effective microscopic cross sections for hydrogen. Three cross section tables were generated for each prebreeder core concept, one for each cycle of depletion. Cross sections for use in radial reflector regions were generated using a composite cell RCP01 calculation which represented a Zone 1 fuel assembly adjacent to the reflector (including core basket, thermal shield and a portion of the pressure vessel). Cross sections appropriate for two separate reflector regions were determined. The first reflector region was approximately two inches thick and included the core basket and associated water; the second region extended outward through the pressure vessel.

Few group cross sections for the burnable poison rods in the reference core were obtained from an RCP01 calculation incorporating such rods. The cross sections were assumed to remain constant throughout the cycle. Hence, the reference core calculation with burnable poison rods inserted is correct at beginning-of-cycle but does not accurately deplete poison rods. This effect is considered small since poison rods were included in only two fuel assemblies within the quarter core.

RCP01 neutron transport calculations were made at moderator temperatures of 601°F, 556°F, and 515°F. The 601°F calculations and one set of 515°F calculations used a Doppler broadening temperature of 1400°F in the seed pellets and Doppler broadening temperatures of either 930°F (spiked with plutonium) or 750°F (unspiked) in the blanket pellets. The 556°F calculations and the other set of 515°F calculations used Doppler broadening temperatures of 556°F to represent a zero power condition. The PDQ08 core depletion calculations used

(601°F cross sections. The 556°F cross sections were used in calculations of hot core shutdown reactivity, and the 515°F cross sections were used in calculations of the core reactivity for the shutdown conditions associated with a steam line break and for calculation of temperature and Doppler deficits. The significance of these temperatures is discussed in more detail in subsection F.

C. Fission Product Model

The fission product model had the following features:

1. All fission products appearing in the samarium-149 fission product chains (neodymium-147; promethium-147, -148, -149, -150, -151) as well as xenon-135 were represented explicitly in the PDQ08 calculations. The composite fission product nuclide in the RCP01 neutron transport calculations consisted of the samarium chain nuclides plus a fission product residual representing all the fission products which were not treated explicitly. The relative nuclide inventories in each fuel management zone were obtained from preliminary PDQ08 depletions. Few group cross sections for explicitly treated nuclides in PDQ08 were obtained from the RCP01 edit calculations.
2. The fission product residual cross section representing all other fission product nuclides was calculated for prebreeder core Concept 1 using the technique discussed below.
3. The fission product residual thermal cross section calculated for prebreeder core Concept 1 was corrected by the ratio of oxygen cross section in the core concept of interest to that in core Concept 1 for use in the other concepts. Determination of the residual thermal cross section in this manner assumes that the cross section is nearly $1/v$.

4. The fission product residual epithermal cross section for prebreeder core Concept 1 was adjusted for use in other concepts by multiplying the Concept 1 value by the ratio of RCP01 calculated fission product residual cross section in the concept of interest to that in core Concept 1.

The fission product residual few group cross section for prebreeder core Concept 1 was calculated by performing CINDER (Referent B-9) calculations simulating the depletion of a fuel assembly. Few-group fluxes were obtained from a preliminary PDQ depletion at 500 EFPH intervals for each fuel management zone. The fuel nuclide cross sections were obtained from a preliminary RCP01 neutron transport calculation and were properly spectrum weighted for the CINDER calculation. In this RCP01 calculation xenon-135 was represented explicitly and twenty-five of the most important fission products (see Table B-6) were represented as a composite nuclide to accurately account for their effects on the neutron flux spectrum. All nuclear data for these isotopes were derived from ENDF/B data files. Few group microscopic cross sections were obtained from RCP01 edits for ten* of the most important epithermal absorbers and three* of the most important thermal absorbers for input to CINDER. Epithermal and thermal cross section correction factors were defined for the explicit absorbers as the ratio of RCP01 cross section to cross section in the then current version of CINDER. The remaining CINDER fission product epithermal cross sections were weighted by the average correction applied to the ten explicit epithermal absorbers (0.82). The remaining fission product thermal cross sections were weighted by the average thermal shielding factor (0.88) of the 25 nuclides represented in the RCP01 calculation.

*Designated in Table B-6.

TABLE B-6. EXPLICIT FISSION PRODUCTS REPRESENTED IN RCP01

Xe-135 ⁽¹⁾	Sm-152 ⁽²⁾
Nd-147	Eu-153 ⁽²⁾
Pm-147 ⁽²⁾	Kr-83
Pm-148(G)	Ru-101
Pm-148(M)	Eu-155
Pm-149	Pr-143
Sm-149 ⁽¹⁾	Pr-141
Sm-150	Xe-131 ⁽²⁾
Sm-151 ⁽¹⁾	Cs-133 ⁽²⁾
Nd-143 ⁽²⁾	Mo-95 ⁽²⁾
Tc-99 ⁽²⁾	Cs-135
Rh-103 ⁽²⁾	Sm-147
Nd-145 ⁽²⁾	Eu-154

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- (1) Spectrum dependent Group 4 cross sections from RCP01 used in CINDER.
(2) Spectrum dependent Group 3 cross sections from RCP01 used in CINDER.

Since CINDER cross section data are based on a mixed number density (MND) flux spectrum, and RCP01 provides a Monte Carlo flux spectrum, the thermal fluxes and cross sections were appropriately modified prior to input to CINDER. The spectrum weighting factor was calculated as the ratio of the hydrogen thermal absorption cross section from RCP01, and the hydrogen 2200 m/s cross section of 0.332 barn.

The CINDER depletion was started using fresh fuel assembly loadings. The PDQ08 fluxes from fuel management Zones 1, 2, and 3 were used for the first, second, and third years of fuel assembly depletion, respectively. One year of

depletion was assumed to be 7000 hours at full power (7000 EFPH), which corresponds to a capacity factor of 0.8. At the end of each year, a 650-hour zero power period simulated reactor shutdown for fuel management and refueling.

III. DIFFUSION THEORY (PDQ08) MODEL

This section describes the diffusion theory model used for the performance assessment of the prebreeder core concepts and the reference core. All diffusion theory calculations were made with the PDQ08 computer program. Since the pre-breeder core concepts would be quarter-symmetric with respect to the core radial centerlines, quarter-core radial slice calculations were used to determine radial performance. Three-dimensional quarter-core calculations were used to assess axial peaking factors. The basic radial geometry is shown in Figure B-1. In Figure B-1, the solid lines represent fuel assembly boundaries within the quarter core and the numbers at the upper left of each fuel assembly are the fuel assembly location numbers. Each fuel assembly in the core was divided into quarter sections for the purpose of fuel management as will be discussed later. The thick line near the edge of the core represents the boundary between freshly fueled and depleted fuel assemblies.

A. Radial PDQ08 Model

The PDQ08 radial model described a quarter core (60.25 fuel assemblies) using a 38 by 38 mesh. Uniform mesh intervals of 2.14675 inches (one-quarter of the fuel assembly pitch of 8.587 inches) were used to describe the core. Beyond the outermost row of fuel assemblies there were two mesh intervals of 2.15 inches each followed by two mesh intervals of 4.11 inches each. Each quarter fuel assembly in the radial model was defined to be a depletion block; that is, neutron flux levels and nuclide inventories were calculated for each of these quarter fuel assemblies throughout core depletion. All materials within a quarter

Figure B-1. Quarter-Core PDQ Description

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fuel assembly were homogenized to provide average nuclide inventories. The grid material (Inconel-718) was assumed to be homogenized over the entire active length of the core so that the effect of grids on core reactivity was accounted for. This results in a slight overestimate of the poisoning effect of the grids. However, since the grids account for only about 0.5 percent of the total neutron captures, this effect is negligible.

All freshly fueled fuel assemblies within each prebreeder core concept would have identical nuclide inventories except for the central fuel assembly in the core which is assumed to contain no recycled fuel and have a fissile loading in the seed rods which is approximately 50 percent of that in the other freshly fueled fuel assemblies. The unique center fuel assembly features are the result of the fuel management plan in which the center fuel assembly plus a third of the remaining fuel assemblies are replaced each cycle. The loading in the center fuel assembly would be adjusted to provide acceptable peaking near the center of the core. The nuclide inventories in the depleted quarter fuel assemblies at the beginning of a depletion cycle (BOC) were obtained from PDQ08 nuclide concentration files generated during the previous cycle.

Diffusion theory depletion calculations were performed for each prebreeder core concept and for the reference core. Results from these calculations were used to determine the amount of fissile material which would have to be added to the cores at the start of each cycle, to determine the mass flows (input and output masses) from the cores, to determine fuel management strategies and the resulting power peaking performance, and to determine temperature coefficients and stuck control rod assembly shutdown reactivity levels. All depletion calculations were made assuming that no control rods were present in the core and

(that reactivity control was accomplished with soluble boron. A poison search to criticality was performed with PDQ08 to determine the required boron content.

Zone dependent axial buckling values were used to account for the effect of axial depletion profiles on core leakage. These buckling values were determined by varying the axial buckling in a two-dimensional quarter fuel assembly depletion calculation of core Concept 1 to provide a cell K_{eff} equal to that in a three-dimensional calculation for the same quarter fuel assembly. Results of these calculations indicated that if the geometric buckling (including 7 cm reflector savings) were used throughout the depletion, the two-dimensional cell k_{eff} would be equal to the three-dimensional K_{eff} at beginning of cycle, but would be approximately 0.6 percent higher at 7000 EFPH. This 0.6 percent difference in K_{eff} would then remain approximately constant through the rest of the depletion (to 21,000 EFPH for a given fuel assembly). These results are in good agreement with results of previous cylindrical geometry calculations performed by Bettis for a pressurized water reactor (PWR) fuel assembly. These calculations indicated that a 0.7 percent higher eigenvalue would be obtained at 7000 EFPH, and this difference would persist through 21,000 EFPH. Axial buckling values which provided agreement between two-dimensional and three-dimensional calculations at 3500 EFPH, 10,500 EFPH, and 17,500 EFPH were used for zones 1, 2, and 3, respectively.

In determining the required fissile loading for the cores the following assumptions were made:

1. For prebreeder core Concepts 1 and 2, (and for the reference core), the uranium remaining in the seed rods (or the reference fuel rods) removed from the core at the end of each cycle would be re-enriched and used

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to fuel fresh seed rods (or reference fuel rods) in a subsequent cycle. Recycle losses of 1.2 percent were assumed. The uranium-236 content in the recycled fuel was determined using equations for the behavior of uranium-236 in an enrichment plant having one or two recycle uranium feed streams, one or two product streams, and natural uranium feed (Reference B-11). For continuous recycle (i.e., the seed rod or reference rod fuel is assumed not to be discarded over the lifetime of the core) the uranium-236 would increase to significant levels.

2. For prebreeder core Concept 2, plutonium bred in the uranium seed rods would be recycled into the thorium blanket rods. The plutonium isotopes were assumed to decay for 1 year (the time assumed required for recycle) before being re-inserted into the core. Plutonium removed from the thorium rods after one cycle of irradiation would be recycled for another cycle. Following the second cycle of irradiation, the plutonium would be discarded due to its contamination by plutonium-242, which decreases its worth as a fissile material.
3. For prebreeder core Concept 3, all uranium remaining in the seed rods would be recycled back into the seed rods in a subsequent cycle without re-enrichment. This assumption results in maximum non-fissile uranium content in the core (Uranium-232 and Uranium-236) and is thus conservative in terms of fissile makeup requirements. Recycle losses of 1.2 percent were assumed for all nuclides, but no further decrease in uranium-236 content was assumed since the fuel was not assumed to go through the re-enrichment process. Highly enriched uranium (93 percent uranium-235) would be added to the recycled fuel to provide the required fissile content.

To provide a consistent basis for comparing the prebreeder and reference core nuclear performance, the depletion of each core was calculated for 14 years with relatively long time steps (Section IV), with fuel management, and with uranium recycle. Following the fourteenth year of depletion a more refined fuel management was performed and short time step calculations were performed for one additional year (year 15) to provide final results.

Fuel management defines a strategy for determining the location of depleted fuel assemblies within the core such that power peaking factors are minimized. Approximately one-third of the fuel assemblies in the core would be replaced at the end of each cycle. (In the present work, 81 of 241 fuel assemblies would be replaced since the central fuel assembly was assumed to be replaced after each cycle.) Hence, about two-thirds of the core would consist of partially depleted fuel assemblies at the beginning of each cycle, with one-third of the fuel assemblies having been depleted for one cycle and one-third for two cycles. In general, the fresh fuel assemblies would be placed on the periphery of a prebreeder core and the depleted fuel assembly locations would be determined on an individual basis to minimize variations in the fuel assembly relative power densities (RPD's). In addition to fuel management, burnable poison rods were used in the reference core calculations to further reduce power peaking. Burnable poison rods were not required in the prebreeder core concepts.

Each core is subdivided conceptually into three fuel management zones. Zone 1 is defined as the aggregate of locations where fresh fuel assemblies would be installed. Zones 2 and 3, respectively, are defined similarly in terms of locations where fuel assemblies having experienced one cycle or two cycles of depletion would be installed. Fuel management details are presented in Sections VI and VII.C of this appendix.

Fuel management was implemented in PDQ08 by defining each quarter fuel assembly in the core as a separate figure (called a final figure). Nuclide inventories for each of these final figures were stored in PDQ08 concentration files at the end of each depletion timestep. An option is available in PDQ08 to allow movement of the final figures from one location to another within the problem solution area. Use of quarter fuel assemblies would allow rotation of the fuel assembly in the fuel management strategy; however, no rotation was used in the prebreeder or reference core calculations reported here. An additional one-or-two-percent power peaking reduction might be possible using fuel assembly rotation.

After determining near-optimum fuel management strategies, best estimate power peaking results were obtained by combining radial quarter-core Relative Power Density (RPD) results with local power peaking factors obtained from quarter-fuel assembly cell calculations and axial power peaking factors obtained from previous three-dimensional calculations of a commercial PWR core. Maximum radial local peaking factors were obtained for each rod type (seed or blanket) in each management zone of the core using the following equation:

$$F_r = \frac{(RPD)(F_{r,j})(f_{j,z})N_r}{N_j} \quad (1)$$

where:

F_r = Local radial peaking factor,

RPD = Fuel assembly power density relative to the core average power density determined from the radial quarter-core calculations,

$F_{r,j}$ = Local power peaking factor for rods of type j (either seed or blanket) determined from cell calculations, defined for example as follows:

$$F_{r,\text{seed rods}} = \frac{\text{Peak seed rod power in cell}}{\text{Average seed rod power in cell}}$$

$f_{j,z}$ = Fraction of power produced in rods of type j in management zone z,

N_r = Total number of rods (seed and blanket) in fuel assembly (equals 332 for the prebreeders), and

N_j = Number of rods of type j in a fuel assembly.

The values of F_r calculated in this manner are equivalent to the following:

$$F_{r,\text{seed rods}} = \frac{\text{radial peak power in seed rod}}{\text{radial average power in all (seed + blanket) rods}}$$

with a similar equivalence for blanket rods.

Moderator temperature coefficients and Doppler deficits were calculated using the radial model described above and nuclear cross sections generated as described in Section II.B.3 for moderator temperatures of 601°F and 515°F with Doppler broadening temperatures appropriate to both full core power and zero power. For each core concept PDQ08 calculations were performed using these cross sections and nuclide concentrations generated by the depletion calculations for year 15. These calculations provided core reactivities at both temperatures and both power levels. The difference between the calculated reactivity at 515°F, 100 percent power and the calculated reactivity at 601°F, 100 percent power, was divided by -86°F to provide the moderator temperature coefficient. The difference between the calculated reactivity at 515°F, 100 percent power and 515°F, zero power was defined as the Doppler deficit.

The kinetics parameters $\bar{\beta}$ (the effective delayed neutron fraction) and Λ^* (the mean prompt neutron generation time) were calculated using the radial geometry described above and core inventories determined from the year 15 depletion calculations. A perturbed eigenvalue λ' was calculated for each of the pre-breeder core concepts and for the reference core using the same cross sections as in the base calculations. The fission source was augmented by the factor $(1 + \beta_{\text{avg}})$, and the extra neutrons added by the term β_{avg} had a few group spectrum characteristic of the delayed neutron spectrum. The value and spectrum for β_{avg} were determined by fission fraction weighting values of β (the delayed neutron fraction for a single nuclear species) for the fissionable isotopes in the core (thorium-232, uranium-233, uranium-235, plutonium-239, and plutonium-241). The value of $\bar{\beta}$ was determined using the equation

$$\bar{\beta} = \frac{\lambda' - \lambda^T}{\lambda^T} \quad (2)$$

where λ^T is the unperturbed eigenvalue from the base (year 15 depletion) calculation.

The value of Λ^* is shown in Reference B-12 to be identical to the reactivity change resulting from a uniform perturbation of the absorption cross sections throughout the entire core by an amount $\Delta\Sigma_a(E) = 1 \text{ sec}^{-1}/V(E)$, where $V(E)$ is the neutron speed in cm/sec at energy E . This perturbation was implemented in PDQ08 by adding equal traces of boron-10 throughout all regions of the calculations.

The quarter core radial model described above and the core inventories determined from year 15 depletion calculations were also used for calculations of prebreeder and reference core shutdown reactivities. Shutdown calculations were run for average coolant temperatures of 556°F (the hot shutdown temperature)

and 515°F (the minimum temperature during steam line break accident). Calculations were made for the core with all control rod assemblies (CRA's) inserted to determine which fuel assemblies with control rods inserted contained the highest relative power densities. The three highest power fuel assemblies were in all cases located near the core periphery. The CRA was removed from one of these locations and the quarter core calculation was rerun to assess the core reactivity with one control rod assembly (CRA) stuck in the withdrawn position. This was done for each of the three locations to determine the minimum shutdown. One control rod assembly stuck calculations were made for core Concept 1 investigating all of the rodded fuel assemblies in the quarter core. These calculations confirmed that fuel assemblies selected for one control rod assembly stuck calculations using the power method above did give minimum shutdown.

Because quarter-core radial calculations were used, stuck control rod assembly calculations actually represented rods removed from either two or four fuel assemblies in the cores. Since such isolated quarter-core symmetric locations experience minimal nuclear coupling, it was assumed that results of such calculations directly provided the one control rod assembly stuck reactivity. To test this assumption, comparisons were made between calculations for one control rod assembly stuck and two control rod assemblies stuck within the quarter core. It was determined from these comparisons that two control rod assemblies stuck (control rod assemblies in fuel assemblies 3 and 59 of Figure B-1) yielded essentially the same calculated reactivity as one control rod assembly stuck (control rod assembly in fuel assembly 3). The difference in reactivity between the two calculations was only 0.004% $\Delta\rho$.

B. Axial PDQ08 Model

Axial power peaking information was obtained from results of three-dimensional quarter-core calculations made for the Westinghouse RESAR-3S PWR core (Reference B-10). These calculations were available from early studies of prebreeder core concepts. The RESAR core contains 193 fuel assemblies vs 241 for the reference core. Since the core height and fuel assembly physical characteristics are essentially the same for the RESAR-3S and the reference and prebreeder cores, it was judged that axial peaking factors obtained from RESAR-3S calculations would be valid for both the prebreeder core concept and reference core calculations.

The radial geometry used for the three-dimensional calculations was similar to that used in the prebreeder core concept radial calculations discussed above. Axially, the core was described using 30 mesh intervals. Six mesh intervals were used to represent the upper and lower reflectors and 24 uniform mesh intervals (6 inches each) were used to describe the active core region. The grids were homogenized over the entire active core region to give the effect of these grids on core reactivity but not on local power peaking. Upper and lower reflector regions were represented as homogeneous mixtures of the materials actually appearing in those regions.

A fuel management strategy for the RESAR-3S core developed from radial calculations was used as input to the three-dimensional PDQ08 calculations. The three-dimensional calculation was performed for four cycles using this fuel management strategy to reach an approximate equilibrium depletion cycle condition.

The beginning of cycle axial peaking factor in the freshly fueled fuel assemblies was determined from the fourth cycle calculation to be 1.32, which is 10 percent below the 1.45 values reported in Reference B-10. This difference may

be a result of not incorporating temperature feedback or explicit grid representation into the axial model. Consequently, to account for these effects all axial factors obtained from the three-dimensional PDQ08 calculations were multiplied by a factor of 1.10 to obtain best estimate factors. Best estimate axial peaking factors as a function of core average depletion level and fuel management zone were determined from these three-dimensional calculations.

IV. DEPLETION MODEL

Depletion calculations were performed at a constant power level corresponding to full power for 7000 hours, simulating one year of operation at an 80 percent capacity factor. An initial timestep length of 100 hours at full power was used to build xenon to its approximate equilibrium level. For the calculations representing the first 14 years of depletion, the 100-hour period was followed by a 900-hour period at full power and three 2000-hour periods at full power to reach 7000 EFPH. At 7000 EFPH, the power level was reduced to essentially zero power and a 1760-hour period at zero power was used to decay all isotopes (of which protactinium-233 and xenon-135 are particularly important) prior to the start of the next cycle. The total cycle length was 8760 hours (365 days). For the fifteenth year of depletion, the initial 100-hour depletion at full power was followed by a 150-hour period and twenty-seven 250-hour periods at full power to reach 7000 EFPH. The use of short (250-hour) full power time periods eliminated the need to use xenon and protactinium feedback techniques during depletion. It is recognized that in actual core operation, the decay of isotopes would probably not occur in a 1760-hour time period at the end of the cycle. Much of the decay would probably take place during short shutdown periods scattered throughout the year. The actual decay time at end of cycle would be approximately 650 hours.

However, use of a more exact representation would have a negligible effect on core performance, and in particular would have less than a 1 percent effect on annual uranium-233 production.

The depletion of all important nuclides was treated explicitly. All fission products appearing in the samarium-149 fission product chains (that is, neodymium-147; promethium-147, -148, -148M, -149; samarium-149, -150, and -151) as well as iodine-135 and xenon-135 were represented explicitly. The remaining fission products were combined to form a lumped residual fission product, as described in the discussion of the cross section model.

Depletion chain descriptions are provided in Table B-7. Seven explicit chains were used to describe the depletion behavior of all important heavy nuclides and fission products. Chains 1 and 2 described the depletion of thorium-232 and subsequent buildup of the uranium isotopes. Chain 1 accounted for protactinium-233 decay into uranium-233. Chain 2 treated the formation of uranium-234 from capture of a neutron by protactinium-233. Chain 3, the uranium-232 production chain, is described in Section V.

Chain 4 in Table B-7 describes the buildup of the plutonium isotopes from the depletion of uranium-238. Chains 5, 6, and 7 describe the fission products which have been explicitly incorporated into the model.

Half-lives and decay constants are given in Table B-8. The yield fractions for the explicitly represented fission products are listed in Table B-9.

V. PRODUCTION OF URANIUM-232

Chain 3 of Table B-7 was used to assess the buildup of uranium-232. Protactinium-231 formed as a result of the (n, 2n) reaction in thorium-232 may capture a neutron to form protactinium-232 which decays to uranium-232.

TABLE B-7. NUCLIDE CHAINS

Chain	Nuclide	Preceding Nuclide In Chain	Coupling from Preceding Nuclide(1)	Comments
1	Th-232	NA	NA	First nuclide in chain.
	Pa-233	Th-232	C + γ	Decay assumed instantaneous so that Th-233 is not shown.
	U-233	Pa-233	γ	
	U-234	U-233	C	
	U-235	U-234	C	
	U-236	U-235	C	
	U-237	U-236	C	Capture in U-237 is ignored.
2	Th-232	NA	NA	First nuclide in chain, duplicate of Th-232 in Chain 1.
	Pa-233	Th-232	C + γ	Duplicate of Pa-233 in Chain 1.
	U-234	Pa-233	C	Added to U-234 built up from Chain 1.
	U-235	U-234	C	Added to U-235 built up from Chain 1.
	U-236	U-235	C	Added to U-236 built up from Chain 1.
	U-237	U-236	C	Added to U-237 built up from Chain 1.
3	Pa-231	NA	NA	First nuclide in chain. ⁽²⁾
	U-232	Pa-231	C	Assumes that each capture in Pa-231 results in U-232 formation.
	U-233	U-232	C	Because of low nuclide inventory, chain artificially stops at U-233.
4	U-238	NA	NA	First nuclide in chain.
	Np-239	U-238	C	Decay assumed instantaneous so U-239 is not shown.
	Pu-239	Np-239	γ	
	Pu-240	Pu-239	C	
	Pu-241	Pu-240	C	
	Pu-242	Pu-241	C	
	Pu-243	Pu-242	C	

(1) NA = not applicable, C = neutron capture, γ = decay.

(2) Pa-231 is formed by an (n, 2n) reaction in Th-232. This is treated as a fission product of Th-232 with a yield of 0.266161.

TABLE B-7 (Cont)

Chain	Nuclide	Preceding Nuclide In Chain	Coupling from Preceding Nuclide(1)	Comments
5	I-135	NA	NA	Fission product chain, fission product yields given in Table B-9.
	Xe-135	I-135	γ	
6	Nd-147	NA	NA	Fission product chain.
	Pm-147	Nd-147	γ	47.4% of Pm-147 captures result in Pm-148M.
	Pm-148M	Pm-147	C	
	Pm-149	Pm-148M	C	
	Sm-149	Pm-149	γ	
	Sm-150	Sm-149	C	
	Sm-151	Sm-150	C	
7	Nd-147	NA	NA	Duplicate of Nd-147 in Chain 6.
	Pm-147	Nd-147	γ	Duplicate of Pm-147 in Chain 6.
	Pm-148	Pm-147	C	52.6% of Pm-147 captures result in Pm-148.
	Pm-149	Pm-148	C	Contributions from this chain added to contribution from Chain 6.
	Sm-149	Pm-149	γ	
	Sm-150	Sm-149	C	
	Sm-151	Sm-150	C	

(1) NA = not applicable, C = neutron capture, γ = decay.

TABLE B-8. DECAY CONSTANTS

<u>Nuclide</u>	<u>Half-Life</u>	<u>Decay Constant (sec⁻¹)</u>
Th-232	1.41×10^{10} years	0.16×10^{-17}
Pa-231	3.24×10^4 years	0.676×10^{-12}
Pa-233	27.0 days	0.2961×10^{-6}
U-232	72 years	0.305×10^{-9}
U-233	1.55×10^5 years	0.1414×10^{-12}
U-234	2.45×10^5 years	0.899×10^{-13}
U-235	7.14×10^8 years	0.308×10^{-16}
U-236	2.34×10^7 years	0.9378×10^{-15}
U-237	6.75 days	0.119×10^{-5}
U-238	4.51×10^9 years	0.487×10^{-17}
Np-239	2.35 days	0.341×10^{-5}
Pu-239	2.44×10^4 years	0.9005×10^{-12}
Pu-240	6600 years	0.3358×10^{-11}
Pu-241	15 years	0.146×10^{-8}
Pu-242	3.87×10^5 years	0.568×10^{-13}
Pu-243	4.96 hours	0.388×10^{-4}
I-135	6.7 hours	0.2874×10^{-4}
Xe-135	9.16 hours	0.2102×10^{-4}
Nd-147	11.06 days	0.7254×10^{-6}
Pm-147	2.625 years	0.8373×10^{-8}
Pm-148M	41.5 days	0.1933×10^{-6}
Pm-148	5.4 days	0.1485×10^{-5}
Pm-149	53.1 hours	0.3626×10^{-5}

TABLE B-8 (Cont)

<u>Nuclide</u>	<u>Half-Life</u>	<u>Decay Constant (sec⁻¹)</u>
Sm-149	1000 years	0.22×10^{-10}
Sm-150	1465 years	0.15×10^{-10}
Sm-151	93 years	0.2362×10^{-9}

The production of protactinium-231 from the thorium-232 (n, 2n) reaction was represented as a direct yield from thorium-232 fission. The yield fraction value of 0.266 (the ratio of the thorium-232 (n, 2n) reaction rate to the thorium-232 fission rate) determined for the Shippingport LWBR was used in the calculations.

Due to the importance of the uranium-232 concentration in irradiated thorium and uranium-233 fuel from the standpoint of shielding and from the standpoint of non-proliferation, uranium-232 production rates calculated using the model described above (the base model) were compared with the results of a more detailed analysis. Two additional sources of uranium-232 production not considered in the LWBR model were also examined. These are the production from the uranium-233 (n, 2n) reaction and from the Th-230 (n, β) Pa-231 (n, β) U-232 reactions.

Prebreeder core Concept 1 was chosen for these comparisons. Monte Carlo calculations were performed to obtain effective fission yield fractions for both thorium-232 (n, 2n) and uranium-233 (n, 2n) reactions. These yield fractions and a chain for the U-233 (n, 2n) U-232 reaction were incorporated into the prebreeder core Concept 1 depletion model and calculations were performed to an equilibrium cycle. These calculations showed an increase in uranium-232 production of 9 percent relative to the values obtained when the uranium-233 (n, 2n) reaction was ignored. Therefore, the uranium-232 production rates calculated using the base model were increased by 9 percent to account for production from uranium-233.

TABLE B-9. FISSION PRODUCT YIELD FRACTIONS

Fission Product Nuclide	Fractional Yield from fission of					
	<u>Th-232</u>	<u>U-233</u>	<u>U-235</u>	<u>U-238</u>	<u>Pu-239</u>	<u>Pu-241</u>
I-135	0.049636	0.048	0.06147	0.05778	0.06934	0.06260
Xe-135	0.46153×10^{-3}	0.012	0.002829	0.00222	0.00266	0.002405
Nd-147	0.0293	0.0178	0.0212	0.0280	0.0205	0.0220
Pm-147	0.0	0.360×10^{-6}	0.0	0.0	0.0	0.0
Pm-148	0.0	0.1895×10^{-5}	0.1×10^{-7}	0.0	0.0	0.0
Pm-148M	0.0	0.189×10^{-5}	0.1×10^{-7}	0.0	0.0	0.0
Pm-149	0.0145	0.00773	0.01	0.021	0.013	0.012
Sm-149	0.0	0.0	0.0	0.0	0.0	0.0
Sm-150	0.763×10^{-5}	0.1232×10^{-3}	0.415×10^{-5}	0.0	0.0	0.0
Sm-151	0.0042	0.00365	0.00408	0.009	0.008	0.005

Thorium-230 is a product of uranium-234 α -decay and may be present in mined thorium up to about 80 ppm in the Th-232 when uranium and thorium are found in the same ore body. On the other hand, there are mineral deposits where thorium occurs relatively free of uranium and the thorium-230 content is as low as 2 ppm in thorium-232. To assess the contribution to the uranium-232 production rate from this source, a range of from 0 to 100 ppm thorium-230 in thorium was assumed. Thermal and epithermal (few groups 4 and 3 respectively) cross sections of 12.2 barns and 111 barns were used as the thorium-230 capture cross sections. Depletion calculations were performed for prebreeder core Concept 1 using 10 and 100 ppm thorium-230 in thorium. Equilibrium cycle calculations for these cases indicated an increase in uranium-232 production rate of 0.40 percent per ppm thorium-230. This result was used to obtain the formula for estimating the actual uranium-232 content:

$$\text{U-232 mass} = (\text{calculated U-232 mass}) \times (1.0 + 0.004(\text{PPM Th-230 in Th-232}))$$

Thorium-230 concentrations in the order of 10 ppm would contribute about a 4 percent increase in uranium-232 production rate.

VI. REFERENCE CORE CALCULATIONS

Nuclear calculations were performed for the reference core to provide both a basis for qualifying the calculational model through comparison of calculated nuclear performance indicators with those reported in Reference B-1, and a reference point for evaluation of nuclear performance of the prebreeder core concepts. The reference core calculations were performed using the model described previously. An 80-percent capacity factor was assumed and continuous recycle of uranium (as described above for prebreeder core Concepts 1 and 2) was used. The

required uranium enrichment using continuous uranium recycle and annual refueling of one-third of the core was determined to be 3.13 w/o uranium-235 in uranium.

Fuel management calculations were performed for the reference core in the manner described in Section III. In addition to selecting the location of depleted fuel assemblies in the reference core, it was necessary to insert lumped burnable poison rods (B_4C) in fuel assemblies 20 and 63 (Figure B-1) to reduce radial power peaking to a level comparable with that reported for the reference core. This is a commercially accepted method of reducing power peaking, and Reference B-1 reports that in the first loading of the reference core burnable poison rods would be used in 148 of the 241 fuel assemblies.

The resulting fuel management strategy for the reference core in year 15 is given in Table B-10. This table shows where the fuel assembly in each location in a given cycle will be located in subsequent cycles. The fuel assembly location numbers given in the tables correspond to location numbers in Figure B-1. To demonstrate the use of the Table B-10, consider the fuel assembly in location 11. From Figure B-1, it can be seen that this fuel assembly contains fresh fuel (it lies outside the thick line near the edge of the core). From Table B-10, it is seen that in the next cycle the fuel assembly will be in location 14 and in the following cycle in location 12. After three cycles of irradiation it will be removed from the core. Note that some fuel assemblies are put into two locations at the end of a cycle (for example, the fuel assembly in location 29 is put into locations 4 and 8). This results because locations on the radial symmetry lines of the core are actually only half fuel assemblies in the quarter core (see Figure B-1). In practice, the full fuel assembly (fuel assembly 29) would

TABLE B-10. FUEL MANAGEMENT STRATEGY FOR THE REFERENCE CORE IN YEAR 15
(Fuel Assembly Numbers Refer to Locations in Figure B-1.)

<u>Fuel Assembly Location</u>	<u>Fuel Assembly Location in Next Cycle</u>	<u>Fuel Assembly Location</u>	<u>Fuel Assembly Location in Next Cycle</u>	<u>Fuel Assembly Location</u>	<u>Fuel Assembly Location in Next Cycle</u>
1	6	24	30	47	Removed
2	Removed	25	Removed	48	55
3	Removed	26	51	49	Removed
4	2	27	Removed	50	Removed
5	3	28	17	51	Removed
6	25 (1/2)	29	4 and 8	52	65
7	Removed	30	Removed	53	5 and 44
8	7	31	Removed	54	26
9	Removed	32	31	55	Removed
10	23	33	Removed	56	Removed
11	14	34	56	57	Removed
12	Removed	35	57	58	Removed
13	Removed	36	25 (1/2)	59	Removed
14	12	37	32	60	48
15	21	38	Removed	61	41
16	13	39	38	62	18 and 52
17	33	40	47	63*	35
18	27	41	49	64	43
19	24	42	50	65	Removed
20*	15	43	58	66	40
21	Removed	44	59	67	34
22	Removed	45	39	68	42
23	22	46	16	69	36

*Contains 24 lumped burnable poison rods each containing approximately 0.2 w/o B₄C.

go into one of these locations and the fuel assembly located in the symmetric location of another quarter of the core would go into the other. Fuel assemblies would always move in such a manner as to maintain the quarter-core symmetry.

The calculated maximum radial peaking factor (F_r) for the reference core using the above management strategy was 1.41, which is approximately 4 percent higher than the value of ~1.35 identified for this core in Section 4.3 of

Reference B-1. The lower value could be achieved by refining the fuel management strategy and by inserting burnable poison rods (with various boron loadings) in additional fuel assemblies

VII. NUCLEAR PERFORMANCE

All results reported in this section were obtained using the diffusion theory depletion model described in Sections II and III.

A. Uranium Enrichment Requirement

To determine the uranium enrichment required in fresh prebreeder fuel assemblies, it was assumed that the cores would be operated between partial refuelings for a 1-year cycle of 7000 EFPH. At the end of the fifteenth year, the prebreeder core concepts were constrained to have a homogeneous boron content of the moderator less than 100 ppm in order to be comparable to commercial cores which typically have end of cycle boron concentrations of 30-50 ppm.

Each prebreeder core concept and the reference core were depleted with uranium recycle and plutonium recycle (prebreeder core Concept 2 only) utilizing fuel management. At the start of the fifteenth year the fresh fuel (a mixture of virgin and recycled uranium) would require the following fissile content to provide the desired 1-year cycle length: 5.10 w/o and 6.65 w/o uranium-235 enrichment for prebreeder core Concepts 1 and 2, respectively; and 49.5 w/o uranium-235 and 17.5 w/o uranium-233 for prebreeder core Concept 3. The enrichments in prebreeder core Concepts 1 and 2 would be higher than that in the reference core because of the addition of thorium and the reduction in the number of fresh rods containing uranium in the prebreeder core concepts.

B. Mass Flows and Neutron Economy

Tables B-11 through B-14 present mass flows and beginning and end of cycle homogeneous boron concentrations for prebreeder core Concepts 1, 2, and 3, and for the reference core, respectively. Masses in these tables are representative of the fifteenth year of operation. The Uranium-233 output mass flows include all of the protactinium-233 which would be present in irradiated fuel at the time of discharge.

Table B-11 presents the mass flows calculated for prebreeder core Concept 1. This core concept would require an input fissile content equivalent to 1464 kg uranium-235 per year to produce 163 kg uranium-233 per year. The fuel assemblies removed from the core would also yield 574 kg uranium-235 per year plus an additional 233 kg of fissile plutonium per year.

Table B-12 presents mass flows calculated for prebreeder core Concept 2. The nuclides in Table B-12 with identifications preceded by B1 (for example, B1 Pu-239) would be recycled from the seed rods of a previous cycle into the blanket rods. The nuclides with identifications preceded by B2 would be recycled from the blanket rods of a previous cycle back into some blanket rods. At the end of the second cycle of irradiation of the recycled plutonium, it was assumed that the remaining plutonium, which would be heavily contaminated by plutonium-242, would be discarded because of the reduced worth of the plutonium as a fissile material. Prebreeder core Concept 2 would require an input fissile loading of 1591 kg (1378 kg of uranium-235 and 213 kg of fissile plutonium) to produce 258 kg of uranium-233 per year. The fuel assemblies removed from the core would contain 619 kg of uranium-235 and 226 kg of fissile plutonium (of which 218 kg is to be recycled).

TABLE B-11. MASS FLOWS FOR YEAR 15 OF PREBREEDER CORE CONCEPT 1

<u>Nuclide</u>	<u>Rod Type</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
Th-232	Blanket	11,851.0	11,570.2
U-232	Blanket	0.0	0.62
U-233	Blanket	0.0	162.6
U-234	Blanket	0.0	15.5
U-235	Seed	1463.8	574.3
U-236	Seed	326.0	396.3
U-238	Seed	26,888.8	26,247.8
Pu-239	Seed	0.0	191.7
Pu-240	Seed	0.0	54.9
Pu-241	Seed	0.0	40.9
Pu-242	Seed	0.0	8.8
Total Fissile	Seed + Blanket	1463.8	969.5
Fissile Pu	Seed	0.0	232.6
Boron Concentration (ppm)			
BOC - 1407			
EOC - 24			

Table B-13 presents mass flows calculated for prebreeder core Concept 3. As with prebreeder core Concept 1, the blanket rod fuel pellets would not be spiked with any fissile material at BOC. This core concept would require an input fissile loading of 1562 kg (1155 kg of uranium-235 and 408 kg of uranium-233) per year to produce 200 kg uranium-233 per year in the blanket rods.

TABLE B-12. MASS FLOWS FOR YEAR 15 OF PREBREEDER CORE CONCEPT 2

<u>Nuclide</u>	<u>Rod Type</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
Th-232	Blanket	19,370.1	18,950.1
U-232	Blanket	0.0	1.01
U-233	Blanket	0.0	257.6
U-234	Blanket	0.0	21.6
U-235	Seed	1377.5	618.6
U-236	Seed	335.7	388.0
U-238	Seed	19,001.0	18,534.6
Pu-239	Seed	0.0	152.2
Pu-240	Seed	0.0	43.5
Pu-241	Seed	0.0	28.3
Pu-242	Seed	0.0	5.3
B1 Pu-239*	Blanket	150.5	12.2
B1 Pu-240	Blanket	42.4	14.5
B1 Pu-241	Blanket	26.6	25.8
B1 Pu-242	Blanket	5.1	17.0
B2 Pu-239	Blanket	12.5	1.0
B2 Pu-240	Blanket	14.6	2.1
B2 Pu-241	Blanket	23.8	6.2
B2 Pu-242	Blanket	16.1	14.8
Fissile Pu	Seed + Blanket	213.4	225.7 (7.2 to be discarded)

*Nuclides with identification preceded by B1 would be recycled from the seed rods of a previous cycle into blanket rods. Nuclides with identifications preceded by B2 would be recycled from blanket rods of a previous cycle back into blanket rods.

TABLE B-12 (Cont)

<u>Nuclide</u>	<u>Rod Type</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
Total Fissile	Seed + Blanket	1590.9	1101.9
Total Pu	Seed + Blanket	291.6	322.9 (24.1 to be discarded)

Boron Concentration
(ppm)

BOC - 1355

EOC - 55

An additional 417 kg uranium-233 per year would be produced in the seed rods and would be recycled into the seed rods of a subsequent cycle. Since the seed rod fuel would initially contain a mixture of thorium, highly enriched (93 percent) uranium and recycled fuel, little fissile plutonium would be produced in prebreeder core Concept 3.

Table B-14 presents mass flows calculated for the reference core. This core would require an input fissile loading of 1180 kg uranium-235 each year. The fuel assemblies removed from the core would contain 363 kg of uranium-235 and 267 kg of fissile plutonium.

Table B-15 compares the neutron absorption fractions in the various core materials for the three prebreeder core concepts and for the reference core. The data were calculated for the middle (3500 EFPH) of the fifteenth year and indicate the relative neutron balance. During any shutdown period following operation at power there would be a small increase in fissile loading due to decay of protactinium-233. Based on the data in Table B-15, it has been calculated that

TABLE B-13. MASS FLOWS FOR YEAR 15 OF PREBREEDER CORE CONCEPT 3

<u>Nuclide</u>	<u>Rod Type</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
Th-232	Seed	21,656.5	21,271.1
U-232	Seed	2.50	2.76
U-233	Seed	407.6	416.5
U-234	Seed	107.5	108.3
U-235	Seed	1154.7	474.9
U-236	Seed	431.2	435.3
U-238	Seed	228.7	178.2
Pu-239	Seed	0.0	8.0
Pu-240	Seed	0.0	2.0
Pu-241	Seed	0.0	2.4
Pu-242	Seed	0.0	0.6
Th-232	Blanket	15,801.4	15,450.7
U-232	Blanket	0.0	0.82
U-233	Blanket	0.0	199.6
U-234	Blanket	0.0	19.0
U-235	Blanket	0.0	2.7
U-236	Blanket	0.0	0.2
Seed Fissile		1562.3	901.8
Total Fissile		1562.3	1104.1
Boron Concentration (ppm)			
BOC - 1580			
EOC - 50			

TABLE B-14. MASS FLOWS FOR YEAR 15 OF THE REFERENCE CORE

<u>Nuclide</u>	<u>Input Mass (kg/yr)</u>	<u>Output Mass (kg/yr)</u>
U-235	1179.6	362.7
U-236	148.7	263.8
U-238	36,353.0	35,488.8
Pu-239	0.0	216.1
Pu-240	0.0	75.9
Pu-241	0.0	51.3
Pu-242	0.0	15.0
Total Fissile	1179.6	630.1
Fissile Pu	0.0	267.4
Boron Concentration (ppm)		
BOC - 1371		
EOC - 97		

if all the protactinium-233 at the end of a cycle decayed to uranium-233, the effect would be to increase the core fissile loadings by 0.5 percent for core Concept 1, 0.7 percent for core Concept 2, and 1.4 percent for core Concept 3. Associated reactivity increases might be as large as 0.7 percent $\Delta\rho$. This would pose no new operational problems since much larger reactivity due to xenon-135 decay must already be accommodated in conventional PWR cores.

C. Power Peaking

As stated earlier, fuel management strategies were developed to minimize radial power peaking in the prebreeder core concepts and the reference core. The management strategy for the reference core was described in Table B-10. The

TABLE B-15. CORE NEUTRON ABSORPTION FRACTIONS
AT MIDDLE OF CYCLE

<u>Nuclide</u>	<u>Reference Core</u>	<u>Core Concept 1</u>	<u>Core Concept 2</u>	<u>Core Concept 3</u>
H-1(1)	0.0389	0.0282	0.0269	0.0321
O-16(1)	0.0035	0.0036	0.0035	0.0035
SS 304(1)	0.0142	0.0155	0.0106	0.0113
Zircaloy-4	0.0097	0.0107	0.0098	0.0108
Xe-135	0.0186	0.0171	0.0158	0.0171
Nd-147	0.0001	0.0001	0.0001	0.0001
Pm-147	0.0029	0.0028	0.0027	0.0029
Pm-148M	0.0011	0.0010	0.0009	0.0010
Pm-148	0.0003	0.0003	0.0003	0.0003
Pm-149	0.0001	0.0001	0.0001	0.0001
Sm-149	0.0058	0.0055	0.0054	0.0050
Sm-150	0.0007	0.0006	0.0006	0.0006
Sm-151	0.0027	0.0024	0.0023	0.0020
Th-232	--	0.0957	0.1418	0.1741
Pa-231	--	0.0002	0.0004	0.0005
Pa-233	--	0.0023	0.0033	0.0039
U-232	--	0.0001	0.0001	0.0004
U-233	--	0.0361	0.0485	0.1619
U-234	--	0.0010	0.0013	0.0159
U-235	0.2675	0.2997	0.2546	0.2566
U-236	0.0135	0.0219	0.0209	0.0267
U-238	0.2804	0.2132	0.1525	0.0103
Np-239	0.0004	0.0003	0.0002	0.0001
Pu-238	0.0001	0.0001	0.0001	0.0001
Pu-239	0.1851	0.1297	0.0891	0.0074
Pu-240	0.0407	0.0283	0.0178	0.0019
Pu-241	0.0226	0.0137	0.0079	0.0010
Pu-242	0.0010	0.0006	0.0002	0.0001
Pu-243	0.0001	0.0001	0.0001	0.0001
Fission Product	0.0328	0.0293	0.0275	0.0313
Residual				
B-10	0.0441	0.0258	0.0211	0.0283
Inconel-718	0.0049	0.0057	0.0052	0.0061
(Grids)				
Axial Leakage(2)	0.0084	0.0087	0.0087	0.0091
			B1 Pu-239(3)	B Th-232(4)
			0.0445	0.1246
			B1 Pu-240	B Pa-231
			0.0258	0.0004
			B1 Pu-241	B Pa-233
			0.0250	0.0029
			B1 Pu-242	B U-232
			0.0022	0.0001
			B2 Pu-239	B U-233
			0.0036	0.0483
			B2 Pu-240	B U-234
			0.0052	0.0013
			B2 Pu-241	B U-235
			0.0106	0.0003
			B2 Pu-242	B U-236
			0.0031	0.0001

(1) Includes radial leakage from active core.

(2) Axial buckling values as a function of fuel assembly depletion level were input to all radial calculations to account for axial leakage.

(3) Nuclides with identifications preceeded by B1 and B2 would be recycled plutonium

(4) Nuclides with identifications preceeded by B would be in blanket pellets.

strategies for prebreeder core Concepts 1, 2, and 3 for year 15 are shown in Tables B-16, B-17, and B-18, respectively. Burnable poison rods would not be required to reduce power peaking to acceptable levels in the prebreeder core concepts.

Table B-19 presents the best estimate radial power peaking for the reference core and for each prebreeder core concept. (Section III.B of Appendix D presents a discussion of the factors applied to the best estimate radial power peaks to obtain the maximum condition power peaks used for the thermal-hydraulic and fuel

TABLE B-16. FUEL MANAGEMENT STRATEGY FOR PREBREEDER
CORE CONCEPT 1 IN YEAR 15

(Fuel Assembly numbers refer to locations in Figure B-1.)

Fuel Assembly Location	Fuel Assembly Location in Next Cycle	Fuel Assembly Location	Fuel Assembly Location in Next Cycle	Fuel Assembly Location	Fuel Assembly Location in Next Cycle
1	2	24	21	47	17
2	40	25	Removed	48	41
3	8	26	56	49	43
4	Removed	27	36	50	Removed
5	4	28	39	51	Removed
6	Removed	29	24	52	Removed
7	6	30	Removed	53	5 and 44
8	Removed	31	14	54	49
9	Removed	32	Removed	55	58
10	47	33	25	56	Removed
11	23	34	57	57	Removed
12	Removed	35	51	58	Removed
13	Removed	36	Removed	59	18
14	Removed	37	3 and 7	60	26
15	13	38	12	61	27 and 59
16	30	39	32	62	34
17	Removed	40	Removed	63	35
18	Removed	41	Removed	64	42
19	55	42	50	65	40
20	15	43	Removed	66	48
21	Removed	44	52	67	38
22	Removed	45	16	68	33
23	22	46	31	69	65

TABLE B-17. FUEL MANAGEMENT STRATEGY FOR PREBREEDER
CORE CONCEPT 2 IN YEAR 15

(Fuel Assembly numbers refer to locations in Figure B-1.)

Fuel Assembly Location	Fuel Assembly Location in Next Cycle	Fuel Assembly Location	Fuel Assembly Location in Next Cycle	Fuel Assembly Location	Fuel Assembly Location in Next Cycle
1	3	24	Removed	47	50
2	Removed	25	Removed	48	40
3	4	26	Removed	49	Removed
4	Removed	27	36	50	Removed
5	2	28	38	51	Removed
6	Removed	29	39	52	Removed
7	6	30	21	53	7 and 27
8	33	31	Removed	54	41
9	Removed	32	13	55	26
10	23	33	Removed	56	57
11	15	34	Removed	57	Removed
12	Removed	35	34	58	Removed
13	Removed	36	Removed	59	52
14	12	37	5 and 8	60	56
15	24	38	16	61	18 and 44
16	Removed	39	25	62	48
17	22	40	Removed	63	43
18	33	41	51	64	35
19	17	42	49	65	Removed
20	14	43	58	66	55
21	Removed	44	65	67	47
22	Removed	45	30	68	42
23	31	46	32	69	59

rod analyses.) Maximum radial power peaking factors (defined as the ratio of radial peak rod power to core radial average rod power) are given for both the urania and the thorium rods within each fuel management zone. (Fuel management zones are defined in Section III.A.) The maximum seed radial peaking factors would occur in Zone 1 for prebreeder core Concepts 1 and 3 and for the reference core. The prebreeder core Concept 2 peak would occur in Zone 2. For prebreeder core Concept 1, a peak of 1.62 would occur at 1000 to 3000 EFPH; for prebreeder

TABLE B-18. FUEL MANAGEMENT STRATEGY FOR PREBREEDER
CORE CONCEPT 3 IN YEAR 15

(Fuel Assembly numbers refer to locations in Figure B-1.)

Fuel Assembly Location	Fuel Assembly Location in Next Cycle	Fuel Assembly Location	Fuel Assembly Location in Next Cycle	Fuel Assembly Location	Fuel Assembly Location in Next Cycle
1	2	24	Removed	47	40
2	25	25	Removed	48	26
3	4	26	Removed	49	Removed
4	Removed	27	18	50	Removed
5	6	28	17	51	Removed
6	Removed	29	15	52	Removed
7	8	30	12	53	7 and 27
8	Removed	31	Removed	54	41
9	Removed	32	24	55	50
10	23	33	Removed	56	58
11	39	34	Removed	57	Removed
12	Removed	35	49	58	Removed
13	Removed	36	Removed	59	52
14	13	37	3 and 5	60	55
15	31	38	22	61	44 and 59
16	Removed	39	16	62	35
17	33	40	Removed	63	43
18	Removed	41	34	64	48
19	56	42	57	65	25
20	14	43	51	66	47
21	Removed	44	36	67	30
22	Removed	45	38	68	42
23	21	46	32	69	65

core Concept 2, the peak of 1.61 would occur at 1000 EFPH; for prebreeder core Concept 3, the peak would be 1.75 at beginning of cycle; and for the reference core the peak of 1.41 would be at 2000 EFPH. The maximum blanket peaking factor for prebreeder core Concepts 1 and 3 would occur at end of cycle in Zone 3 and for prebreeder core Concept 2 at beginning of cycle in Zone 1. Within a given fuel management zone of the core, the seed rod peaking factor in prebreeder core Concepts 1 and 3 would tend to decrease with core depletion while the blanket

TABLE B-19. BEST ESTIMATE RADIAL POWER PEAKING (F_r) IN YEAR 15

Core Concept 1

Time	Zone 1		Zone 2		Zone 3	
	Seed	Blanket	Seed	Blanket	Seed	Blanket
BOC	1.60	0.02	1.59	0.34	1.39	0.58
100 EFPH	1.61	0.02	1.56	0.34	1.36	0.57
1000 EFPH	1.62	0.05	1.54	0.35	1.31	0.55
3000 EFPH	1.62	0.13	1.48	0.42	1.24	0.56
5000 EFPH	1.57	0.22	1.43	0.49	1.20	0.60
7000 EFPH	1.51	0.30	1.38	0.55	1.20	0.65

Core Concept 2

BOC	1.55	0.82	1.53	0.75	1.38	0.75
100 EFPH	1.50	0.79	1.56	0.76	1.40	0.76
1000 EFPH	1.46	0.74	1.61	0.78	1.43	0.77
3000 EFPH	1.45	0.70	1.61	0.79	1.41	0.77
5000 EFPH	1.44	0.69	1.57	0.79	1.37	0.78
7000 EFPH	1.44	0.69	1.52	0.80	1.33	0.78

Core Concept 3

BOC	1.75	0.02	1.73	0.29	1.49	0.54
100 EFPH	1.73	0.02	1.74	0.29	1.48	0.53
1000 EFPH	1.72	0.04	1.70	0.30	1.45	0.54
3000 EFPH	1.70	0.11	1.61	0.37	1.39	0.58
5000 EFPH	1.66	0.18	1.54	0.45	1.34	0.63
7000 EFPH	1.63	0.26	1.49	0.52	1.29	0.68

Reference Core

BOC	1.33	--	1.32	--	1.07	--
100 EFPH	1.29	--	1.33	--	1.08	--
1000 EFPH	1.38*	--	1.20	--	1.04	--
3000 EFPH	1.39	--	1.16	--	1.03	--
5000 EFPH	1.30	--	1.16	--	1.02	--
7000 EFPH	1.24	--	1.15	--	1.01	--

*The maximum value of F_r is 1.41 and occurs in zone 1 at 2000 EFPH.

(

peaking factor would increase. This would result from the shift in power from the seed to the blanket as uranium-233 is produced and uranium-235 is depleted. In prebreeder core Concept 2, the peaking factor would remain relatively constant within a zone as a result of the spiking of the blanket rods with plutonium. In this case, there would be no significant shift in power from the seed rods to the blanket rods as the cycle progresses because the uranium-233 produced in the blanket would tend to offset the depletion of fissile plutonium in those rods.

Table B-20 presents the best estimate axial power peaking in each fuel management zone as a function of time during the cycle. As described in Section III.B of this appendix, these results were derived from an equilibrium cycle three-dimensional quarter-core calculation for the Westinghouse RESAR-3S core. The calculation represented both axial and radial reflectors as well as the full active length of the core. Grid material was homogenized into the moderator to yield the effect on core reactivity but not on local power peaking. The axial power peaking values were increased approximately 10 percent from the calculated values to account for effects of grids and thermal feedback. This yielded a beginning of cycle maximum best estimate axial power peaking factor in the prebreeder cores of 1.45 in management Zone 1. The 1.45 value is the same as that quoted by commercial vendors for PWR cores. The peaking factor would decrease steadily within each management zone with depletion. Axial peaking factors for Zones 2 and 3 differed by less than 1 percent at all times in core life and were therefore assumed to be equal. These zones would have equivalent axial peaking factors because they would be interspersed throughout the internal portion of the core. Section II.C of Appendix D presents a

TABLE B-20. BEST ESTIMATE AXIAL POWER PEAKING* (Fa(max)) FOR EQUILIBRIUM CYCLE

Time	<u>Zone 1</u>	<u>Zone 2</u>	<u>Zone 3</u>
BOC	1.45	1.34	1.34
100 EFPH	1.32	1.27	1.27
1000 EFPH	1.27	1.24	1.24
3000 EFPH	1.22	1.21	1.21
5000 EFPH	1.20	1.20	1.20
7000 EFPH	1.20	1.20	1.20

*Values listed are values from three dimensional calculation multiplied by 1.10.

discussion of the factors applied to the best estimate axial power peaks to obtain the maximum condition power peaks used for the thermal-hydraulic and fuel rod analyses.

D. Moderator Temperature Coefficients and Doppler Deficits

Core moderator temperature coefficients were estimated based on an 86°F temperature change (from 601°F to 515°F). All calculations were performed for fuel temperatures corresponding to full power operation, and with all control rods withdrawn from the core. The calculations established that negative temperature coefficients would exist in each core concept throughout core depletion. As moderator temperature increases, the density of boron in the core decreases at the same rate as the water density, so the positive reactivity change due to the decrease in boron density tends to offset the negative reactivity change associated with the decrease in water density. Commercial PWR cores are generally designed such that the moderator temperature coefficient is nearly zero at beginning of cycle (the moderator temperature coefficient of the reference core is reported as $+0.08 \times 10^{-4} \Delta\rho/^\circ\text{F}$ at beginning of life in Section 4.3 of Reference B-3). As seen from Table B-21, the moderator temperature coefficients for all three prebreeder core concepts are calculated to be negative throughout the cycle.

TABLE B-21. PREBREEDER CORE CONCEPT MODERATOR
TEMPERATURE COEFFICIENTS FOR YEAR 15

<u>Time</u>	<u>Temperature Coefficient ($\delta\rho/\delta T$)</u>		
	<u>Core Concept 1</u>	<u>Core Concept 2</u>	<u>Core Concept 3</u>
B0C	-0.87×10^{-4}	-1.70×10^{-4}	-0.04×10^{-4}
100 EFPH	-1.21×10^{-4}	-2.03×10^{-4}	-0.28×10^{-4}
1000 EFPH	-1.37×10^{-4}	-2.08×10^{-4}	-0.62×10^{-4}
3000 EFPH	-1.55×10^{-4}	-2.28×10^{-4}	-1.00×10^{-4}
5000 EFPH	-1.71×10^{-4}	-2.47×10^{-4}	-1.27×10^{-4}
7000 EFPH	-1.91×10^{-4}	-2.67×10^{-4}	-1.56×10^{-4}

To ensure that the relatively large temperature range (86°F) used to determine the moderator temperature coefficients did not provide unrealistically large negative values, calculations were run for all prebreeder core concepts and for the reference core using $\pm 10^\circ\text{F}$ temperature variations from the operating temperature of 601°F. These calculations represented only the changes in hydrogen, oxygen, and boron concentrations resulting from the moderator density change. Results of these calculations indicate that the moderator temperature coefficients at 601°F would be approximately 0.7×10^{-4} to 1.5×10^{-4} more negative than the values given in Table B-21 at all times in the cycle.

Doppler deficits for the prebreeder concept cores were calculated at a moderator temperature of 515°F. The results are shown in Table B-22. The variation in the Doppler deficit in core Concept 1 results from the increasing plutonium content of this concept throughout the cycle.

TABLE B-22. DOPPLER DEFICITS FOR YEAR 15

Doppler Deficit - $\Delta\rho$ (100% power to 0% power)

<u>Time</u>	<u>Core Concept 1</u>	<u>Core Concept 2</u>	<u>Core Concept 3</u>	<u>Reference^(a)</u>
BOC	0.012	0.014	0.010	--
100 EFPH	0.012	0.014	0.011	--
1000 EFPH	0.012	0.014	0.011	--
3000 EFPH	0.014	0.014	0.011	--
5000 EFPH	0.015	0.014	0.011	--
7000 EFPH	0.017	0.014	0.011	--

(a) Not calculated for reference core. Table 4.7.10 of Reference B-1 gives beginning and end of first cycle Doppler deficits of 0.011 and 0.013, respectively.

E. Kinetics Parameters

Tables B-23 and B-24 show the calculated values of $\bar{\beta}$ and Λ^* for the pre-breeders and the reference core. The trend observed in Table B-23 is that the effective delayed neutron fraction decreases as the fraction of fissions produced by plutonium-239 and/or uranium-233 increases. This occurs because the delayed neutron fraction from fission of plutonium-239 or uranium-233 is significantly

TABLE B-23. EFFECTIVE DELAYED NEUTRON FRACTIONS, $\bar{\beta}$, FOR YEAR 15

<u>Time</u>	<u>$\bar{\beta}$</u>			
	<u>Core Concept 1</u>	<u>Core Concept 2</u>	<u>Core Concept 3</u>	<u>Reference^(a)</u>
BOC	0.0062	0.0057	0.0053	
100 EFPH	0.0062	0.0057	0.0053	0.0062
1000 EFPH	0.0061	0.0057	0.0053	0.0061
3000 EFPH	0.0059	0.0056	0.0052	0.0058
5000 EFPH	0.0057	0.0055	0.0051	0.0056
7000 EFPH	0.0056	0.0054	0.0049	0.0054

(a) Calculated using same method as for prebreeder core concepts.

TABLE B-22. MEAN PROMPT NEUTRON GENERATION TIME, Λ^* , FOR YEAR 15

Time	$\Lambda^* (\times 10^{-6} \text{ sec.})$			
	Core Concept 1	Core Concept 2	Core Concept 3	Reference ^(a)
BOC	17.6	14.9	16.6	21.5
100 EFPH	17.6	15.0	16.7	21.6
1000 EFPH	17.8	15.3	17.4	21.7
3000 EFPH	18.0	15.8	18.2	22.2
5000 EFPH	18.2	16.2	18.9	22.9
7000 EFPH	18.5	16.7	19.6	22.8

(a) Calculated using same method as for prebreeder core concepts.

smaller than that from fission of uranium-235. Also, the mean prompt neutron generation time in all prebreeder core concepts would be significantly less than in the reference core primarily because of the higher fissile loadings in the prebreeder core concepts. For prebreeder core Concept 2 the relatively large amount of plutonium-239 present further reduces the mean prompt neutron generation time because plutonium has a larger neutron absorption probability per atom than either uranium-235 or uranium-233.

F. Reactivity Shutdown Margins

During normal core operation at full power, the control rods would be fully withdrawn and core reactivity would be controlled with soluble boron. In Reference B-1 (Section 4.3) the design basis for shutdown margin is that sufficient CRA worth is available to shut down the reactor with at least a 1% $\Delta k/k$ subcritical margin in the hot condition at any time during the life cycle with the most reactive CRA stuck in the fully withdrawn position. A shutdown reactivity analysis is presented in Reference B-1 for the beginning and end of the first cycle assuming a hot shutdown temperature of 556°F. (At temperatures below 556°F soluble boron normally would be used in the primary coolant of the reference

plant to provide the necessary shutdown. For any of the prebreeder core concepts, soluble boron could be added to the primary coolant in sufficient quantity to provide shutdown at any temperature below 556°F.) A minimum shutdown margin of 3.2% $\Delta k/k$ was demonstrated. To assure that the prebreeder core concepts would have minimum shutdown margin comparable with that of the reference core, the shutdown margin was calculated at the middle of the fifteenth year for all four cores. The results of these calculations, presented in Table B-25, are that in the hot shutdown condition the prebreeder core concepts would be less reactive, or have more shutdown margin, than the reference core.

Calculations of shutdown margin were also made at 515°F. Based on data in Reference B-1 (Section 15.1), the accident which would result in minimum shutdown margin following reactor trip is a double-ended break of a 29.5 inch steam line between the steam generator and the main steam line isolation valves. Following reactor trip, the primary coolant temperature would drop briefly to about 515°F, at which condition the shutdown margin would be a minimum. The results of the present calculations, presented in Table B-26, are that again the prebreeder core concepts would be less reactive, or have more shutdown margin, than the reference core.

The greater shutdown margins in the prebreeder core concepts are attributable to relative number, distribution and surface area of the control rods. The reference core has 24 control rods per CRA and a control rod diameter of 0.390 inch. The prebreeder core concepts would have 28 control rods per CRA and a control rod diameter of 0.351 inch; relative to the reference core, the control rods in a prebreeder CRA would be distributed over a larger radial area of the fuel assembly and would have about 5 percent greater surface area.

TABLE B-25. ONE CRA STUCK SHUTDOWN REACTIVITIES* FOR THE PREBREEDER CORE CONCEPTS AND THE REFERENCE CORE IN THE HOT (556°F) SHUTDOWN CONDITION FOR YEAR 15

<u>Time</u>	<u>Core Reactivities (β ρ)</u>			
	<u>Core Concept 1</u>	<u>Core Concept 2</u>	<u>Core Concept 3</u>	<u>Reference</u>
BOC	-3.7	-3.4	-4.6	-3.3
1000 EFPH	-3.6	-3.6	-4.5	-3.0
3000 EFPH	-3.6	-3.5	-4.2	-2.8
5000 EFPH	-3.8	-3.4	-4.0	-2.7
7000 EFPH	-4.1	-3.3	-3.7	-2.7

*Limiting control rod assembly is in fuel assembly 03 for core Concepts 1 and 3 and in fuel assembly 29 for core Concept 2 and the reference core (see Figure B-1). All calculations including reference core were performed using the same methods.

TABLE B-26. ONE CRA STUCK REACTIVITIES* FOR THE PREBREEDER CORE CONCEPTS AND THE REFERENCE CORE AT 515°F FOR YEAR 15

<u>Time</u>	<u>Core Reactivities (β ρ)</u>			
	<u>Core Concept 1</u>	<u>Core Concept 2</u>	<u>Core Concept 3</u>	<u>Reference</u>
BOC	-2.7	-2.4	-4.3	-2.2
1000 EFPH	-2.2	-2.3	-3.8	-1.3
3000 EFPH	-1.7	-2.2	-3.4	-1.4
5000 EFPH	-1.4	-2.0	-3.0	-1.2
7000 EFPH	-1.1	-1.7	-2.6	-0.9

*Limiting control rod assembly is in fuel assembly 03 for core Concepts 1 and 3 and in fuel assembly 29 for core Concept 2 and the reference core (see Figure B-1). All calculations including reference core were performed using the same methods.

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APPENDIX C
FUEL ROD CONCEPTS

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APPENDIX C
FUEL ROD CONCEPTS

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APPENDIX C

FUEL ROD CONCEPTS

I. INTRODUCTION

Evaluations of fuel rod performance were made to assess demands on the pre-breeder rods relative to the demands on the rods in the reference Babcock & Wilcox Standard 241 core (Reference C-1). Through the use of sensitivity studies, margins to tentative performance criteria were identified, along with potential performance trade-offs. A description of this effort is contained in this appendix which is composed of: (1) a brief description of the model, (2) calculation method and tentative performance criteria, (3) prebreeder seed rod fuel performance evaluations, and (4) potential improvements. The detailed evaluations were limited to the seed rod. The blanket rod would experience significantly less power (~ 5 kw/ft) resulting in lower temperatures and more margin to melting, less gas release, and lower fuel swelling and clad creep. Therefore the blanket rods were considered to be less limiting, and detailed evaluations were not made.

II. FUEL MODEL

The performance evaluation of the seed rod for the three different core concepts was made using the latest version of the CYGRO program (Reference C-2) and the modeling techniques and input parameters qualified for analysis of the Shippingport Light Water Breeder Reactor (LWBR) core (Reference C-3) except as noted in subsequent discussions. Details of the fuel rods are found in Figure C-1. All calculations were made using nominal fuel pellet and cladding dimensions. Some of the important features of the prebreeder rods include pressurization (the analysis assumed a 500 psi cold pressurization) and use of

Figure C-1. Fuel Rods

(

recrystallized and annealed clad (RXA) as the reference clad material. In the uranium dioxide fuel rods, the amount of fission gas assumed to be released is based on data used for commercial core design and presented in Reference C-1. The internal gas pressure was increased during life of the fuel rod to account for the added pressure from the fission gas. For the uranium dioxide seed rod 50 percent release of fission gas was assumed, and for the fuel system containing a homogeneous mixture of uranium dioxide-thorium dioxide, 25 percent release of fission gas was assumed. The uranium dioxide-thorium dioxide value is based on a gas release assessment in support of LHB. The uranium dioxide thermal conductivity representation used in the commercial industry (Reference C-1) was simulated and assigned to the fuel regions of the uranium dioxide seed rods. In the case of the uranium dioxide-thorium dioxide fuel, the commercial uranium dioxide representation was adjusted for the basic differences in material properties.

Because of the lack of data on the thermal conductivity of thorium dioxide-uranium dioxide fuel at high temperature (above $\sim 2900^{\circ}\text{F}$) two representations were assumed for this variable. In the base conductivity case, it was assumed that the conductivity of thorium dioxide-uranium dioxide fuel is 15 percent greater than that of uranium dioxide fuel. This implies that the conductivity of thorium dioxide-uranium dioxide has an upswing at high temperatures (starting at $\sim 2900^{\circ}\text{F}$) as does uranium dioxide fuel. In the variation-from-base conductivity case the conductivity of thorium dioxide-uranium dioxide fuel was assumed to be the same as that in the base case up to $\sim 2900^{\circ}\text{F}$ and then to decrease monotonically with increasing temperature, being approximately proportional to the reciprocal of absolute temperature.

Annealed clad (RXA), as used in the LWBR seed rods, was selected as the clad material for the initial seed rod studies because of the lower creep rate relative to cold-worked and stress-relieved material (CW). The possibility is recognized that RXA material might have lower resistance to stress corrosion cracking (SCC) and plastic instability than does CW material. Investigations into the SCC performance of both RXA and CW material have been made in support of the LWBR program (Reference C-4). In the reference cited it was concluded that, for material having the same final surface conditions, prior cold working actually has a small effect on increasing the susceptibility to SCC. However, potential influence of fast neutron flux on iodine stress corrosion cracking was acknowledged as an unknown. In the event that RXA cladding would prove to be unacceptable, CW cladding would be an acceptable alternative, based on its application in the reference core and in the blanket rods of the Shippingport LWBR core. A final selection of cladding material could be made at a later date and would consider other aspects such as corrosion behavior and fracture resistance.

Fuel rod performance throughout life was determined using maximum condition power peaks which included the effect of fuel management cycles. The depletion dependent maximum condition axial and radial power peaks are discussed in Sections II.C and III.B of Appendix D and are presented in Tables D-1 and D-5. The rods were assumed to deplete at 90-percent power and to experience up-power transients at the time in life when the highest power peaks would occur and also at end of fuel rod life. These peaks would result during a xenon-free condition and for a particular rod would occur at initial insertion and subsequent to each fuel management cycle startup. The power transients consisted of a power swing from 90-percent power to 100-percent power to simulate an operational transient and also from 90 percent to 112 percent to simulate an accident transient. These

transients were assumed only to provide a basis for comparing fuel rod performance for the various cores. In practice, the allowable operational transients and the peak power allowed by reactor protection system set points would be based on limiting fuel clad stresses to acceptable values in a much larger range of cases. The calculated peak rod power history at 100-percent core power is shown in Figure C-2 for the prebreeder core concepts and the reference core. As indicated in Section V, some flexibility would exist for making changes in the geometry or materials of the prebreeder fuel rods to improve performance if required.

III. FUEL ROD CRITERIA

To determine the capabilities of the prebreeder core concepts, the performance was compared to performance indicators calculated for the reference core. The performance indicators were selected by reviewing the LWBR design considerations and are considered the most significant for initial concepts evaluation. These indicators are also typical of the considerations used in commercial core design. Included in this group are pellet average and pellet maximum temperature, and clad stresses and strains. Fuel rod depletion was not specifically addressed as a performance consideration for this study but depletion differences would be reflected in the calculated clad stress and strain. Table C-1 lists the performance considerations and the significance of each.

Performance indicators for the standard reference core were obtained from a CYGRO analysis using fuel rod data from Reference C-1 and a fuel rod power history developed from rod peaking factors obtained from the nuclear analysis of the reference core performed by the authors of this report. These performance indicators of the reference core, as calculated by the authors, constitute standards to which the performance indicators of the prebreeder core concepts are compared. This technique is considered to provide a reasonable performance

Figure C-2. Seed Rod Power History

TABLE C-1. PERFORMANCE CONSIDERATIONS FOR INITIAL EVALUATION

1. Maximum fuel temperature, which provides comparative data on proximity to melting.
2. Pellet average temperature, which provides a measure of margin to the fuel rod cladding temperature below which Zircaloy-water reaction would be avoided during a loss-of-coolant accident (LOCA).
3. Clad generalized stress calculated during a reactivity insertion transient, which is a measure of the margin to clad plastic instability.
4. Circumferential clad stress during a normal maneuvering transient, which compares the margin to an iodine stress corrosion cracking condition.
5. Clad thinning strain, which compares the ability to avoid appreciable localized thinning strain.
6. Clad axial strain, which is an indicator of the potential for axial gap formations and short tube collapse.

assessment of fuel rods for purposes of evaluating backfit core concepts.

Indirectly it accounts for local conditions (such as pellet chips) and considerations that are outside the scope of this study. Also, this allows for a reasonable determination of fuel rod capability without performing the many different analyses that may be required to provide a "worst case" for each particular performance indicator. A comparison of the prebreeder fuel rod parameters with those of the reference core is found in Table C-2. Prebreeder core Concept 1 would have no plutonium recycle; prebreeder core Concept 2 would include plutonium recycle; and prebreeder core Concept 3 would use highly enriched urania in a thorium diluent.

IV. PERFORMANCE EVALUATIONS

Results of the analysis comparing performance indicators considered to be significant in assessing fuel rod capability are listed in Tables C-3, C-4, and C-5. (In these tables some of the reference core values, which are from the performance calculations made by the authors, differ slightly from the reference

TABLE C-2. FUEL ROD PARAMETERS

Parameter	Core Concept 1		Core Concept 2		Core Concept 3		Reference Core
	Seed	Blanket	Seed	Blanket	Seed	Blanket	
Rod OD, in.	0.349	0.379	0.339	0.379	0.349	0.379	0.379
Pellet OD, in.	0.295	0.335	0.285	0.335	0.295	0.335	0.324
Initial Cold Diametral Gap, in.	0.007	0.004	0.007	0.004	0.007	0.004	0.008
Clad Thickness, in.	0.0235	0.020	0.0235	0.020	0.0235	0.020	0.0235
Max. Design Power, kw/ft	12.3	~4	11.9	~5	13.4	~4	12.8
Peak Depletion, 10^{20} f/cc	11.9	--	9.4	--	11.4	--	10.2

core values in Table 2 which were obtained directly or inferred from data in Reference C-1.) The values of clad average and fuel center temperature can be compared directly to the reference core results. However, no allowance has been made for the possible difference in fracture resistance of RXA tubing (possibly 10 percent lower than that of CW tubing) used in the prebreeder core concepts compared to the CW tubing used for the reference core. To maintain the same margin to the limits as achieved in the reference core, the RXA clad stresses (both circumferential and generalized) may have to be below the values calculated for the reference core; this would require further evaluations.

A. Uranium Dioxide Seed, Thorium Dioxide Blanket, No Plutonium Recycle (Prebreeder Core Concept 1)

A comparison of the performance of the prebreeder without plutonium recycle (core Concept 1) to that of the reference core is presented in Table C-3. Data

TABLE C-3. FUEL ROD PERFORMANCE, NO PLUTONIUM RECYCLE (CORE CONCEPT 1)

Performance Indicator	Reference Core	Prebreeder Core Concept 1				
		7 mil Gap(1) RXA Clad	7 mil Gap RXA Clad Fuel Pellet Density Increased By 2%(2)	8 mil Gap RXA Clad	7 mil Gap CW Clad	8 mil Gap CW Clad
Center Temperature, °F, 112% Power (BOL)	3808	3649	3362	3784	3687	3827
Margin to Melting, °F	1272	1431	1718	1296	1393	1253
Fuel Average Temperature, °F, 100% Power (BOL)	2143	2080	1992	2173	2094	2186
Clad Circumferential Stress, psi, 100% Power (EOL)	20,000	24,100	20,800	15,200	21,100	21,200
Clad Generalized Stress, psi, 112% Power (EOL)	24,100	27,600	24,700	19,900	24,100	24,500
Clad Axial Strain, % (EOL)	1.77	0.64	0.63	0.56	1.86	1.85
Clad Plastic Radial Strain, % (EOL)	~0.0	~0.0	~0.0	~0.0	0.07	~0.0

NOTE: (1) Gap is the initial cold fuel pellet to clad diametral gap.

(2) Reference fuel pellet density assumed to be 95% of theoretical.

TABLE C-4. FUEL ROD PERFORMANCE, PLUTONIUM RECYCLE (CORE CONCEPT 2)

Performance Indicator	Reference Core	Prebreeder Core Concept 2			
		7 mil Gap(1) RXA Clad	8 mil Gap RXA Clad	7 mil Gap CW Clad	8 mil Gap CW Clad
Center Temperature, °F, 112% Power (BOL)	3808	3592	3727	3664	3808
Margin to Melting, °F	1272	1488	1353	1416	1272
Fuel Average Temperature, °F, 100% Power (BOL)	2143	2074	2156	2095	2178
Clad Circumferential Stress, psi, 100% Power (EOL)	20,000	-3100*	-3100*	20,800	20,100
Clad Generalized Stress, psi, 112% Power (EOL)	24,100	**	*	24,400	23,700
Clad Axial Strain, % (EOL)	1.77	0.50	0.50	1.81	1.85
Clad Plastic Radial Strain, % (EOL)	~0.0	~0.0	~0.0	~0.0	~0.0

*Gap exists between fuel and clad.

**Fuel-clad interaction initiated but stress is <1000 psi.

NOTE: (1) Gap is the initial cold fuel pellet to clad diametral gap.

TABLE C-5. FUEL ROD PERFORMANCE, HIGHLY ENRICHED URANIUM
DIOXIDE-THORIUM DIOXIDE SEED (CORE CONCEPT 3)

Performance Indicator	Reference Core	Prebreeder Core Concept 3				
		7 mil Gap(1) RXA Clad	6 mil Gap RXA Clad	8 mil Gap RXA Clad	7 mil Gap CW Clad	6 mil Gap CW Clad
Center Temperature, °F, 112% Power (BOL)	3808	3748 (4150) ^a	3587 (3958)	3908 (4400)	3798 (4200)	3629 (4000)
Margin to Melting, °F	1272	2012 (1610) ^a	2173 (1712)	1852 (1360)	1962 (1560)	2131 (1670)
Fuel Average Temperature, °F, 100% Power (BOL)	2143	2160 (2361) ^a	2063 (2249)	2276 (2522)	2173 (2374)	2074 (2260)
Clad Circumferential Stress, psi, 100% Power (EOL)	20,000	18,500	22,400	-2800**	18,200	25,400*
Clad Generalized Stress, psi, 112% Power (EOL)	24,100	21,000	24,900	4900	20,400	27,000*
Clad Axial Strain, % (EOL)	1.77	0.60	0.68	0.49	1.81	1.81
Clad Plastic Radial Strain, % (EOL)	~0.0	~0.0	~0.0	~0.0	~0.0	0.1

*Limited at beginning of Cycle 3. In all other core concepts and options studied, the maximum stress occurred at EOL.

**Gap exists between fuel and clad.

NOTES: (1) Gap is the initial cold fuel pellet to clad diametral gap.

^aValues in () are for the variation from base case (reduced) conductivity of ThO₂-UO₂.

column 2 of this table gives the performance indicators for the base case fuel parameters of core Concept 1. Transient data in Figures C-3 and C-4 and the discussion in subsections 1 and 2 below are based on the data in columns 1 and 2.

1. Thermal Performance

For both this prebreeder core concept and the reference core, the lifetime peak fuel center temperature at 112 percent power and the lifetime peak average fuel temperature at 100 percent power would occur in fresh fuel rods at the beginning of a cycle (BOC) with the core in a xenon free condition. These lifetime peak fuel temperatures would be lower in the prebreeder core concept than in the reference core. These peak temperatures would occur at BOC because of high power peaks; these power and temperature peaks would diminish rapidly as xenon built up (see Figures C-2 and C-3). For the prebreeder core concept, the initial power would peak at 12.3 kw/ft (at 100-percent core power), but in about 100 hours the power would be reduced to 11.3 kw/ft. At the 11.3 kw/ft local linear power, the fuel center and average temperatures would be reduced 190°F and 70°F, from 3649°F and 2080°F, respectively. Therefore, for this concept, and also for the reference core, reductions in lifetime peak temperature could be achieved if, for the first 100 hours after a fuel management cycle, the core power were limited to something less than full power.

The calculated lifetime variation of the fuel center temperature at steady-state power (90-percent power) for both cores is shown in Figure C-3. In general, the prebreeder rod would operate at a higher temperature. Thus, performance criteria that are time-temperature dependent would have less margin in the prebreeder core concept. Items that can be affected include the amount of fission gas released and behavior of defected fuel rods.

Figure C-3. Pellet Temperature History, No Plutonium Recycle (Core Concept 1)

Figure C-4. Clad Stress History, No Plutonium Recycle (Core Concept 1)

2. Clad Stress-Strain Performance

The clad circumferential stress (an indicator of the susceptibility to iodine SCC as described in Reference C-4) and the generalized stress (a clad plastic stability performance indicator) are calculated to be higher than the reference core values near end of life (EOL) for prebreeder core Concept 1. Because of the lower creep rate with RXA material, the clad axial strain would be much lower, thereby reducing the possibility of axial gaps developing and the propensity for short tube collapse to occur. However, this would not be a major concern in a highly pressurized rod.

A profile of the lifetime variation in clad stresses for prebreeder core Concept 1 and also the reference core is shown in Figure C-4. Included are the steady-state stresses and the stress levels for the up-power transients. Except for the last 10 percent of rod life, the prebreeder clad stresses would be below or equivalent to those for the reference core. In addition, because of the lower creep rate, the clad-pellet interaction would occur later in life. However, after pellet-clad interaction occurs, the lower creep rate would cause the tensile stress to peak at a higher "steady-state" level than in the reference core. The peak stress for the prebreeder would be enough higher (20%) than that in the reference core that if this concept were to be pursued, alternatives and performance trade-offs would be considered to reduce pellet-clad interaction stresses.

3. Performance Trade-Offs

A number of alternatives would be available for improving specific performance indicators, several of which were investigated in detail and are discussed below. Improvements in performance would be possible by taking advantage of the characteristics of high density fuel pellets. Pellet shrinkage would be reduced

with higher density fuel, and the pellet-to-clad gap would be smaller, leading to improved gap conductivity early in life. Also, with fewer pores due to the higher density, the conductivity of the fuel would be increased. Performance gains that would result from an increase of 2 percent in pellet density (from 95 to 97 percent theoretical density) can be determined by comparing data columns 2 and 3 of Table C-3. The high density fuel concept would have increased margin with respect to the peak values of the performance indicators presented. At the higher density, the rod would operate not only at a lower temperature but the resulting reduced thermal expansion would delay fuel-clad interaction. The clad stresses would be reduced to levels comparable to but slightly above those of the reference core. However, the temperatures throughout life, in general, would still be higher than calculated for the reference core. Manufacturing experience with fuel densities in this range is available from the Shippingport L'IBR program where the seed size pellets had a nominal density of 97.5 percent.

Clad stresses are dependent on clad-pellet interaction which normally occurs late in life for a typical initial clad-pellet gap and a rod power history that decreases with life. Due to the lower rod power late in life and the closing of the clad-to-pellet gap over life, the maximum center temperatures would occur at or near the beginning of life (BOL). In some instances, significant improvements in clad stresses would be possible if interaction between the clad and pellet were delayed by increasing the initial gap size. This would, however, result in higher beginning of life fuel temperatures. These trends were assessed by evaluating a concept with the gap size increased from 0.007 to 0.008 inch (data column 5 of Table C-3). The calculated fuel average temperature increased to 30°F above that in the reference

core; however, the calculated clad circumferential stress and the generalized stress were reduced to well below the reference values (15,200 psi compared to 20,000 psi for the circumferential stress and 19,900 psi compared to 24,100 psi for the generalized stress).

The use of the same type of cladding used in the reference core (70-percent CW) was also evaluated. In this evaluation the same material properties were used as in the LWBR blanket rod evaluations. Concept arrangements having a cladding outer diameter-to-thickness ratio of 16 (the same as that in the reference core) and an initial gap of 7 or 8 mils were evaluated. The results are tabulated in data columns 5 and 6 of Table C-3. Acceptable temperatures and stresses were calculated for both concepts. However, the calculated maximum values of the clad stresses were essentially unchanged when going from a 7- to an 8-mil gap, the only effect of increasing the gap being to delay the time of clad-pellet interaction until later in life. The delaying of fuel-clad interaction to later in life would constitute an improvement in fuel performance since the length of time the clad and pellet would be in contact would be lessened, thus minimizing the vulnerability to clad failure.

4. Summary

Prebreeder core Concept 1 would be marginal with respect to criteria determined from peak values of the performance indicators evaluated. However, flexibility would exist to improve the clad stresses and develop acceptable performance using RXA material by increasing the initial clad-to-pellet gap or by using higher density fuel pellets. In addition, based on the comparison of performance indicators, if the cladding material (CW) used in the reference core were selected for the prebreeder application, a concept could be developed with fuel element performance indicators comparable to those of the reference core.

B. Uranium Dioxide Seed, Thorium Dioxide-Plutonium Dioxide Blanket, Plutonium Recycle (Prebreeder Core Concept 2)

A performance comparison, similar to that made for core Concept 1 was also made for the seed rods of the plutonium recycle prebreeder (core Concept 2). This comparison is presented in Table C-4. Data column 2 of this table gives the performance indicators for the base case fuel parameters. Transient data in Figures C-5 and C-6 and the discussion in Sections 1 and 2 below are based on the data in columns 1 and 2.

1. Thermal Performance

For this core concept, the lifetime peak fuel center temperature at 112 percent power and the lifetime peak average fuel temperature at 100 percent power would also occur in fresh fuel rods at BOC with the core in a xenon free condition. These peak temperatures would be lower than those in both prebreeder core Concept 1 and the reference core; this is consistent with the relative fuel rod power at BOC (Figure C-2) and the similarity of the fuel rods (Table C-2). Reductions in lifetime peak temperature could also be achieved for this core concept by restricting the power level for approximately the first 100 hours after a fuel management cycle startup.

The lifetime variation in temperature shown in Figure C-5 would be similar to that of prebreeder core Concept 1. Although the fuel temperatures would be lower than in the reference core during most of the first fuel management cycle, they would be higher for the second and third fuel management cycles. The same considerations with respect to time-temperature effects that were discussed for core Concept 1, would also apply to core Concept 2.

Figure C-5. Pellet Temperature History, Plutonium Recycle (Core Concept 2)

2. Clad Stress-Strain Performance

The lower rod power would delay pellet-clad interaction to beyond end-of-life for steady-state conditions, providing significant margin to the values calculated for the reference core. Peak stress values are tabulated in data column 2 of Table C-4 and the lifetime variation of the circumferential stress is displayed in Figure C-6. Only during an up power accident transient at end of life would fuel-clad interaction be expected to occur, but at this operating condition the clad stress is calculated to be less than 1000 psi.

3. Performance Trade-Offs

Alternatives investigated for prebreeder core Concept 2 are included in Table C-4. The thermal margin could be traded for a delay in fuel-clad interaction. A fuel rod with an initial gap of 8 mils (data column 3 of Table C-4) would have margin to peak center temperature, would have acceptable average temperature, and fuel-clad interaction would be delayed beyond the end of life for all conditions evaluated including the accident power transient to 112-percent power at end-of-life. Acceptable performance indicators are also calculated for a core concept using CW cladding material. Although the axial strain would be slightly higher in the core concept with CW cladding than in the reference core, it is judged that this slight increase would not be significant with respect to short tube clad collapse due to the formation of axial gaps; this would be especially true for a pressurized fuel rod.

4. Summary

Because of the lower peak rod power, the fuel rods of core Concept 2 are calculated to have more margin to the thermal and stress limits than fuel rods of the reference core. The lower clad stresses would be expected to compensate

Figure C-6. Clad Stress History, Plutonium Recycle (Core Concept 2)

for any decrement in performance attributed to RXA cladding. Flexibility also would exist to further improve the performance by varying gap size and changing fuel density. Concepts using CW cladding would also be an acceptable alternative.

C. Highly Enriched Uranium Dioxide-Thorium Dioxide Seed, Thorium Dioxide Blanket, No Plutonium Recycle (Prebreeder Core Concept 3)

A performance comparison, similar to those made for prebreeder core Concepts 1 and 2, was also made for the highly enriched uranium dioxide-thorium dioxide seed rod of prebreeder core Concept 3. This comparison is presented in Table C-5. Data column 2 of this table gives the performance indicators for the base case fuel parameters. Transient data in Figures C-7 and C-8 and the discussion in subsections 1 and 2 below are based on the data in columns 1 and 2. In the analysis of this core concept the differences in the fuel pellet attributes relative to those of uranium dioxide, such as a modified fuel thermal conductivity, lower thermal coefficient of expansion, and higher fission gas release temperature were accounted for in the CYGRO model. When comparing performance indicators related to melting temperature (pellet center temperature), allowances for the significantly higher melting temperature of the uranium dioxide-thorium dioxide ($\sim 700^{\circ}\text{F}$ higher) must be considered.

1. Thermal Performance

For this core concept, the lifetime peak fuel center temperature at 112 percent power and the lifetime peak average fuel temperature at 100 percent power would also occur in fresh fuel rods at BOC.

For the base conductivity case the lifetime peak fuel center temperature would be slightly lower than that of the reference core, despite this prebreeder core concept having a higher rod power (kw/ft) than that in the reference core.

Figure C-7. Pellet Temperature History, Highly Enriched Uranium Dioxide-Thorium Dioxide Seed (Core Concept 3)

Figure C-8. Clad Stress History, Highly Enriched Uranium Dioxide-Thorium Dioxide Seed (Core Concept 3)

Due to the higher melting temperature of thorium dioxide-uranium dioxide fuel relative to that of uranium dioxide fuel, the margin to melting in this core concept would be over 700°F greater than that in the reference core.

Relative to the reference core, the lifetime peak average fuel temperature would be 17°F higher; however, the higher fission gas release temperature associated with uranium dioxide-thorium dioxide fuel would reduce fuel rod deformations during a LOCA. The lower coefficient of thermal expansion of the thorium dioxide-uranium dioxide fuel would be an advantage with respect to clad stresses during up power transients.

The lifetime variation in fuel center temperature is compared to that of the reference core in Figure C-7. Although the prebreeder is calculated to operate at a slightly higher temperature for approximately two thirds of fuel rod life, this would be offset by the improved characteristics of the fuel used in the prebreeder. In particular, less gas release and better capability for operating with a defected rod would be expected.

Fuel temperatures calculated using the variation-from-base conductivity for thorium dioxide-uranium dioxide are presented in parentheses in Table C-3. Relative to the results for the calculations made using the base conductivity, the predicted fuel center temperature is about 400°F higher and the predicted fuel average temperature is about 200°F higher. The margin to melting would still be higher, by about 300°F, than that in the reference core. The fuel rod deformation during a LOCA would still be expected to be below that in the reference core.

2. Clad Stress-Strain Performance

For the calculations using the base conductivity, the clad circumferential and generalized stresses for prebreeder core Concept 3 would be about the same as those for the reference core. The variation in clad circumferential stresses during steady-state operation and stresses that would occur during the postulated transient conditions are shown in Figure C-8. It is apparent from Figure C-8 that fuel-clad interaction would be delayed until late in life in this prebreeder core concept. With the use of RXA material as the cladding in the prebreeder core concept, the clad axial strain is calculated to be approximately one third that of the reference core and should minimize the potential for the formation of axial gaps in the fuel stack. In both the reference core and the prebreeder core concept, the clad plastic radial tensile strains would be negligible.

Clad stress was not calculated using the variation-from-base conductivity. However, it is expected that the peak tensile stress calculated for either steady state or transient operation using this conductivity would not differ significantly from the peak stress calculated using the base conductivity. This expectation is based on the data appearing in Figures C-7 and C-8 (both of which are based on the base conductivity). Figure C-8 indicates that the peak tensile stress would occur at end of fuel rod life (defined to be 21,000 effective full power hours). Figure C-7 indicates that the maximum pellet center temperature at this time in life would be about 2200°F. But below temperatures of about 2900°F the conductivity is the same for both representations; hence, calculated temperatures and clad tensile stresses would also be the same.

3. Performance Trade-Offs

Alternatives investigated for prebreeder core Concept 3 are included in Table C-5. The discussion which follows is based on the calculations using the

base conductivity for thorium dioxide-uranium dioxide fuel. A prebreeder core concept having a smaller initial gap which would increase the stresses but reduce fuel temperatures was evaluated along with a core concept having a 1-mil larger gap. These results are shown in data columns 3 and 4 of Table C-5. A 1-mil reduction in initial gap (column 3) would improve the thermal performance; the fuel center temperature and the fuel average temperature would be, respectively, 220°F and 80°F below those of the reference core values. However, the clad stresses would exceed the levels calculated for the reference core. A 1-mil increase in initial gap would delay fuel-clad interaction enough that interaction would not occur except during a postulated overpower accident at the end of life. Clad stresses would be low, but both the fuel center temperature and average temperature would be higher than those calculated for the reference core. The optimum core concept would be expected to have a gap somewhere between 7 and 8 mils.

Results for core concepts having 70-percent CII cladding, listed in data columns 5 and 6 of Table C-5, show that prebreeder core concepts that would meet the desired temperature and stress-strain levels using this cladding material could likely be developed. Relative to the reference core, the core concept with a 7 mil initial gap (data column 5) would have a slightly higher peak average fuel temperature and slightly lower peak fuel center temperature and peak cladding stresses; the core concept with a 6 mil initial gap (data column 6) would have substantially lower peak fuel center temperatures and substantially higher peak cladding stresses. A core concept with CII cladding and an initial gap between 6 and 7 mils could have peak fuel temperatures and cladding stresses comparable with those of the reference core.

For the core concept with CH cladding and a 6 mil initial gap, fuel clad contact and the peak cladding stress would occur at the start of the third cycle of fuel rod irradiation. (For all other cases and core concepts considered, the peak cladding stresses would occur at end of fuel rod life.) This early occurrence of fuel-clad contact would be due to the 6 mil initial gap combined with the creep rate of CH cladding, which is higher than that of RXA cladding. The peak cladding stress at this time would be caused by the peak in the fuel rod power (Figure C-7) at the start of the cycle.

4. Summary

For prebreeder core Concept 3 there would be a number of alternatives that would have acceptable performance indicators while operating at an initial rod power 5 percent higher than in the reference core. Two representations of the thermal conductivity of thorium dioxide-uranium dioxide were considered in the calculation of performance indicators. The representation with lower fuel conductivity leads to predicted peak and average fuel pellet temperatures which would be above those of the reference core. However, due to the higher melting point and higher gas release temperature of uranium dioxide-thorium dioxide fuel relative to those of uranium oxide fuel, adequate margin in center temperature would be expected to exist for this core concept during steady-state and transient operation. Clad stresses would also be expected to be acceptable.

D. Other Considerations

Although only a select number of performance indicators were explicitly evaluated, those evaluated are judged most important for initial core concept

selection. The following is a partial list of other considerations along with some discussion as to the adequacy of the prebreeder core concepts with respect to these items.

1. Fuel rod gas plenum - The fuel rod gas plenum would be expected to be very similar to that of the reference fuel rods in size (length) and mechanical arrangement. On this basis, it is judged that a plenum could be developed that would have adequate volume and mechanical characteristics.
2. Clad corrosion - The prebreeder core concepts would not have any unique features affecting clad corrosion. The rods are not expected to be exposed to any environmental condition or clad burden requirements that would cause a special problem related to clad corrosion or hydrogen pickup. In fact, RXA clad might be somewhat more corrosion resistant than C/W clad.
3. Rod bowing - Detailed analysis involving CYGRO and ROBOT (rod bowing analysis, Reference C-5) calculations to determine the magnitude of the expected bow would require extensive nuclear data defining fuel rod transverse and axial power distributions that are unavailable at this stage of development of the core concept. However, to account for changes relative to the reference core, a relationship was developed to describe the interaction of axial grid spacing with rod diameter and rod spacing. This relation, discussed in more detail in Appendix A, identified a need for a reduction in axial grid spacing to maintain rod bow equivalent to that in the reference core. Accordingly, additional grids were included in the prebreeder core concepts. Additional discussion of rod bow and grid details is found in Appendix A.

4. Clad collapse into axial pellet stack gap - Short tube buckling stability performance, which is a measure of the ability to resist axial collapse into a pellet gap, is not anticipated to be a problem since pressurized rods would be used. Nevertheless, for further development of these core concepts the fuel rod stability to short tube buckling would have to be assessed.
5. Loss-of-Coolant Accident adequacy - Current Nuclear Regulatory Commission regulations require that during a Loss of Coolant Accident (LOCA) the clad temperature be less than 2200°F, total clad oxidation be less than 17 percent, and a coolable core geometry be maintained. Clad temperature, and to a first approximation, the clad oxidation would be affected primarily by fuel pellet average temperature. Based on calculations reported in Appendix D, the clad temperature limit, and therefore the clad oxidation limit, would be met by all prebreeder core concepts. Use of a pressurized rod rather than an unpressurized rod would be expected to have an adverse effect on maintaining coolable geometry; however, pressurization of fuel rods to the approximate magnitudes considered for the prebreeder core concepts is standard practice in the commercial industry.

The capability for maintaining a coolable geometry in the prebreeder core concepts was assessed assuming: (1) the clad strain in the seed rods would be less than or equal to that expected in the reference core fuel rods and (2) if the fraction of flow area remaining between rods for a postulated rod strain is the same as for the reference core, then the extent to which a coolable geometry exists would also be the same.

In addition, if the fraction of flow area is the same, then any increase in temperature due to partial closing of the flow channel would be assumed to be no higher than for the reference core.

For various values of clad strain, the change in flow area for a flow region bounded by four rods was calculated for the reference core and also for the prebreeder core concepts. In the prebreeder core concepts, a region bounded by three seed rods and a blanket rod is considered representative of most regions. A flow region bounded by 4 seed rods would occur only between fuel assemblies where the rod spacing would be approximately the same as in the reference core, and rod power would be slightly lower. Therefore, these regions are considered less limiting in maintaining a coolable geometry. Because of the low power and minimal gas release expected in the blanket rods, the clad strain during a LOCA was assumed negligible relative to that in the seed rods. The results of this assessment are found in Figure C-9. At any particular strain, the fraction of flow channel area remaining is calculated to be higher in the prebreeder core concepts.

Therefore the ability to maintain a coolable geometry for the reflood stage would be comparable for the prebreeder core concepts and the reference core if the cladding strains were comparable. The LOCA evaluation in Appendix D indicates that during the blowdown phase, the cladding temperature would be comparable for all four cores. This is taken as an indication that the cladding strains would also be comparable.

Figure C-9. Flow Area Change with Clad Strain

6. Defect corrosion performance - Clad rupture from fuel-clad interaction, with subsequent corrosion of the fuel (defect corrosion), is most probable near end of life in both the prebreeder core concepts and the reference core. The UO_2 in the seed rods would have corrosion resistance similar to that of the UO_2 in the reference core because of the similarity in fuel systems. Based on the good corrosion resistance shown by UO_2 - ThO_2 fuel and ThO_2 pellets in the Shippingport LWBR program, the UO_2 - ThO_2 seed rods and the prebreeder blanket rods would be expected to have somewhat better defect corrosion performance than the rods in the reference core.
7. Grid spring follow - The rod diameter change with life of the prebreeder core concepts was compared to that of the reference core to determine if adverse effects on spring follow would be introduced with changes in geometry and rod power requirements. Figure C-10 shows this comparison for prebreeder core Concept 1 with both RXA and CW cladding. The predicted diameter change with the CW cladding would closely follow that of the reference core, while very little diameter change with life would be predicted with the RXA cladding. A similar situation would exist for prebreeder core Concepts 2 and 3. Based on a comparison of rod diameter changes, rod spring follow in the prebreeder core concepts would not be expected to be of any greater concern than in the reference core. However, if the grid spring force is lost by about 12,000 hours, the prebreeder core concepts using the reference CW cladding would have an advantage.

Figure C-10. Fuel Rod Diameter Change

V. COMPARISON OF PREBREEDER CORE CONCEPTS WITH REFERENCE CORE

The prebreeder studies have shown that core concepts could be developed for each prebreeder fuel system to give values for selected thermal and stress-strain performance indicators comparable with those for the reference core. A comparison of the performance indicators for the prebreeder core concepts and the reference core that was used as a basis for determining acceptability is shown in Table C-6. A relative ranking of each performance indicator is tabulated in Table C-7.

Based on this comparison, prebreeder core Concept 3 would be considered to be the preferred core concept if fuel melting is the primary fuel rod performance concern, because of the higher melting temperature of the fuel used in this core concept. Core Concept 2 would have greater margin with respect to the other performance indicators because of the low peak rod power and fuel depletion. The clad stresses in core Concept 1 would exceed those of the reference core; however, all core concepts would have the flexibility to produce improved specific performance indicators by varying geometry or material selections. In the case of core Concept 1, increasing the gap to 8 mils would cause the clad stresses to become less than the reference values and fuel temperatures would still be acceptable. This flexibility is illustrated in the options listed in Tables C-3, C-4, and C-5. Although prebreeder core Concept 2 is judged to have the greatest overall margin, it is expected that a workable design could be achieved for all three core concepts.

VI. AREAS FOR FURTHER DEVELOPMENT

In the preceding sections the acceptability of the prebreeder core concepts has been based on peak values of specific performance indicators; and therefore, history-dependent phenomena were not always explicitly accounted for. Peak temperatures could be altered to a desired level by adjusting the gap size, but the

TABLE C-6. FUEL ROD PERFORMANCE COMPARISONS

<u>Performance Indicator</u>	<u>Reference Core</u>	<u>Prebreeder Core Concept 1</u>	<u>Prebreeder Core Concept 2</u>	<u>Prebreeder Core Concept 3</u>
Center Temperature, °F, 112% Power (BOL)	3808	3649	3592	3748
Margin to Melting, °F	1272	1431	1488	2012
Fuel Average Temperature, °F, 100% Power (BOL)	2143	2080	2074	2160
Clad Circumferential Stress, psi, 100% Power (EOL)	20,000	24,100	-3100*	18,500
Clad Generalized Stress, psi, 112% Power (EOL)	24,100	27,600	**	21,000
Clad Axial Strain, % (EOL)	1.77	0.64	0.50	0.60
Clad Plastic Radial Strain, % (EOL)	~0.0	~0.0	~0.0	~0.0
Fuel Pellet-to-Clad Diametral Gap, in.	0.008	0.007	0.007	0.007
Clad Material	70% CW	RXA	RXA	RXA
Max. Power, kw/ft	12.8	12.3	11.9	13.4
Peak Rod Depletion, 10^{20} f/cc	10.2	11.8	9.4	11.4

*Gap exists between fuel and clad.

**Fuel-clad interaction initiated but stress is <1000 psi.

TABLE C-7. RANKING OF PERFORMANCE INDICATORS

Criteria	Relative Ranking ⁽¹⁾		
	Prebreeder Core Concept 1	Prebreeder Core Concept 2	Prebreeder Core Concept 3
Center Temperature, Margin to Fuel Melting	3	2	1
Fuel Average Temperature,			
LOCA Clad Temperature Index	2	1	3 ⁽²⁾
Gas Release and Pressure	3	2	1
Clad Circumferential Stress, Index of Susceptibility to SCC	3	1	2
Clad Generalized Stress, Index to Clad Strain Instability	3	1	2
Clad Axial Strain, Potential for Forming Axial Gaps	3	1	2
Clad Radial Plastic Strain, Potential to Avoid Localized Thinning	Even	Even	Even

(1) A value of 1 denotes the most margin.

(2) Detailed LOCA studies could show this core concept to be better because of its better heat transfer characteristics.

)

profile throughout life would be dependent on the rod power history. This is illustrated in Figures C-2, C-3, C-5, and C-7 where the calculated rod power history and center temperature history for the prebreeder core concepts and the reference core are shown. The prebreeder pellet center temperatures would be below that of the reference core for periods in life but, in general, the prebreeders would operate at slightly higher pellet center temperatures. Further development of these core concepts would require evaluation of performance margins relative to time-temperature dependent criteria. However, prebreeder core Concept 3 would have $\text{UO}_2\text{-ThO}_2$ fuel which is expected to have better performance characteristics than the reference UO_2 fuel. Therefore, for this core concept it is expected that the performance margins relative to time-temperature dependent criteria would be at least as good as those of the reference core.

VII. CONCLUSIONS

The conclusions here are based on comparing selected performance indicators for the prebreeders with those for the reference core using the same analysis methods for all cores. Based on a simple comparison of fuel rod linear power (Figure C-2) and fuel rod geometries (Table C-2), it was expected that the performance indicators for the prebreeders would be similar to those for the reference core. Detailed studies using the CYGRO program confirmed that for all three prebreeder core concepts it would be possible to obtain performance indicators comparable with those of the reference core (Table C-6). If required, sufficient flexibility would exist to trade performance in one area to improve performance in another. This flexibility could be used to accommodate non-conservatisms which might be due to the use of nominal dimensions and material properties and a limited number of performance indicators for determining

acceptability (Section VI). Based on a relative ranking of performance indicators (Table C-7), prebreeder core Concept 2 is judged to possess the greatest overall fuel rod performance margin. Core Concept 3 would have the greatest margin to fuel melting because of the higher melting temperature of the fuel. Acceptable stress-strain performance would exist for all three prebreeder core concepts. Though not all concerns were directly addressed, sufficient margin is considered to exist to overcome any additional concerns that would be identified in a more detailed analysis so that prebreeder fuel rods would be expected to perform comparably to commercial fuel rods.

VIII. REFERENCES

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APPENDIX D
THERMAL AND HYDRAULIC CONCEPTS

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APPENDIX D

THERMAL AND HYDRAULIC CONCEPTS

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APPENDIX D

THERMAL AND HYDRAULIC CONCEPTS

I. INTRODUCTION AND SUMMARY

This appendix describes the thermal and hydraulic analysis models and procedures that have been used for estimating the performance of the prebreeder core concepts. A detailed thermal and hydraulic steady state analysis of the reference core (Reference D-1) and of each prebreeder core concept was performed with the HOTROD computer program (Reference D-2). The thermal performance was calculated utilizing the Babcock and Wilcox BAW-2 critical heat flux (CHF) correlation (Reference D-1) with minimum DNBR (Departure from Nucleate Boiling heat flux Ratio) of 1.32. By comparing the steady state thermal capability of the reference core calculated by the authors with that reported in Section 4.4 of Reference D-1, and by analysis of numerous CHF experiments performed at Bettis or reported in the literature, it was determined that the HOTROD model conservatively predicts thermal capability.

Acceptable performance of the prebreeder core concepts was defined as the capability for steady state operation at a minimum of 111 percent of design power (design power = 3800 MWth) with a minimum DNBR (Departure from Nucleate Boiling heat flux Ratio) of 1.32. The basis for these criteria is presented below.

Adequate thermal performance of a core is normally demonstrated for a Safety Analysis Report by performing a detailed analysis for transients resulting from a broad spectrum of accidents, such as rod withdrawal accidents and steam line rupture accidents, and demonstrating that no fuel damage will occur. This type of demonstration is beyond the scope of the present work. However, the

kinetics parameters and reactivity coefficients for the prebreeder core concepts would be similar to those for the reference core, and protective system setpoints could be adjusted to accommodate some differences in core behavior during transients. Therefore during transients the limiting conditions reached in the prebreeder core concepts would be expected to be similar to those reached in the reference core. (Comparisons are presented in Appendix B for moderator temperature coefficients, Doppler deficits, control rod worths, prompt neutron lifetimes and effective delayed neutron fractions.)

On the basis of the above it was judged that if the prebreeder core concepts had a calculated steady state thermal overpower capability equal to or exceeding that of the reference core using the same minimum DNBR (1.32), they would also have acceptable transient power capability. A steady-state overpower capability of 111 percent of design power was calculated by the authors for the reference core using the same methods that were used for calculations on the prebreeder core concepts. Therefore 111 percent steady state overpower capability was chosen as the acceptance criterion for the prebreeder core concepts.

Using the analytical methods of this report, the steady-state overpower capability of prebreeder core Concepts 1, 2, and 3 is respectively 114 percent, 117 percent and 112 percent of full power with a minimum DNBR of 1.32.

II. QUALIFICATION OF HOTROD MODEL

A. General Features of HOTROD Model

The HOTROD analysis utilized the Moody single phase friction factor, the Martinelli-Nelson two-phase friction factor, and the void fraction and vapor generation models described in Reference D-2. The high pressure void fraction model assumes homogeneous flow and the nonequilibrium vapor generation model

from the work of Griffith, Clark and Rohsenow, Reference D-3. The HOTROD formulation of this model is closely analogous to that developed by Maurer, Reference D-4. A two-phase mixing correlation developed for the HOTROD program was also used. Additional information on these models can be found in References D-2, D-3 and D-4. The BAW-2 CHF correlation was used (Section 4.4 of Reference D-1) with a minimum DNBR of 1.32 indicating adequate margin to CHF.

B. Analysis of CHF Tests

Results of CHF testing on rod bundles has been reported in References D-5 through D-8. From these references, a total of 165 sets of test conditions were chosen for analysis using HOTROD. These conditions covered a wide range of variables; channel hydraulic diameters from 0.11 to 0.63 inch, system pressure from 2000 to 2314 psia, inlet temperatures from 488 to 607°F, inlet mass velocities from 1.5 to 4.0×10^6 lb/hr-ft², channel average heat fluxes up to 856,000 BTU/hr-ft², local heat fluxes up to 1.188×10^6 BTU/hr-ft², uniform and varying transverse and axial heat fluxes, and heated channel lengths from 54 to 96 inches. Although in both the prebreeder core concepts and in the reference core the heated channel length is 144 inches, the development of models based on test data for shorter heated lengths is standard practice. Thus, the range of test variables includes the values characteristic of the prebreeder core concepts, presented in Table 2 of the main text and as part of Tables D-2 and D-7 which are discussed later.

The HOTROD analyses indicated that inception of CHF was conservatively predicted for all cases using a DNBR (Departure from Nucleate Boiling heat flux Ratio) of 1.32 with the BAW-2 CHF correlation, Equations 4.4-16 and 4.4-17 of Reference D-1. This is the minimum allowable DNBR used by B&W for the reference core, as reported in Section 4.4 of Reference D-1. Although the data base of

165 points is small compared with the data base normally used for commercial core designs, the above results indicate that the HOTROD model, using the BAW-2 CHF correlation and a minimum DNBR of 1.32, is adequate for the present purpose of indicating the possibility of developing backfit prebreeders with adequate thermal capability. The final qualification of a model for analyzing a prebreeder would require additional critical heat flux testing for specific geometry and grid details.

C. Reference Core Analysis

Some of the details of the calculational model utilized in the HOTROD analysis of the reference core are shown in Figure D-1. The HOTROD model represented one-fourth of a fuel assembly.

The radial power peaking factors used in HOTROD for individual fuel rods are indicated by the F_r values in Figure D-1 and are based on the beginning-of-life best estimate radial power shape from the nuclear calculations made for the reference core by the authors as described in Sections III.A, VI and VII.C of Appendix B. The best estimate shape was multiplied by a normalizing factor to make the peak F_r equal to 1.55. 1.55 is the maximum radial power peaking factor specified in Section 4.4 of Reference D-1 as one of the maximum design conditions used in the thermal-hydraulic analysis of the reference core as reported in Reference D-1.

Axial power shapes were based on the best estimate calculations reported in Section III.B of Appendix B. These shapes were modified so that for each fuel management zone the best estimate axial peak was increased by 15 percent (i.e., by a factor of 1.15). The maximum axial peak, occurring in fuel management zone 1, was then 1.67, which is the same as the maximum axial factor specified in Section 4.4 of Reference D-1 as one of the maximum design conditions used in

- 264 FUEL RODS PER FUEL ASSY, DIAMETER = .379"
- 1 INSTRUMENT TUBE SLEEVE PER FUEL ASSY, DIAMETER = .480"
- ▨ 24 CONTROL ROD GUIDE TUBES PER FUEL ASSY, DIAMETER = .465"
- RADIAL PEAKING FACTOR, $\bar{F}_R$
 HOTROD SUBCHANNEL NUMBER

Figure D-1. 17 by 17 Array Reference Core
 (HOTROD Model, 1/4 Fuel
 Assembly)

the thermal-hydraulic analysis of the reference core. The dependence of axial peaking on fuel management zone and irradiation time is displayed in Table D-1. The axial shape factors are discussed in more detail in Section VII.C of Appendix B.

HOTROD input parameters are presented in Table D-2. One hundred percent power capability was set equal to 3800 MWth and a total reactor coolant flow rate of 159×10^6 lb/hr was used. These values are from Section 4.1 of Reference D-1. To account for leakage around the core, the flow through the core was assumed to be 5.5 percent less than the total reactor coolant flow rate. This reduction is based on data in Section 1.3 of Reference D-1.

The effect of spacer grids on pressure drop and fluid mixing between subchannels was represented by a loss coefficient, (K), which was applied continuously over the entire core length. This representation, chosen for conservatism, eliminated abrupt K losses at grid locations and thereby minimized the calculated amount of mixing between subchannels. The loss coefficient in a given type of subchannel was made dependent on the ratio of flow area in the spacer grid to unobstructed flow area in the subchannel. K values assumed were 0.70 per grid for subchannels with only fuel rods on the boundary, and 0.87 per grid for subchannels with a control rod guide tube (which has a larger diameter than a fuel rod) on the boundary. These values gave an unrecoverable core pressure drop consistent with that specified in Reference D-1.

The core pressure drop was used as the forcing pressure drop for the HOTROD hot fuel assembly of Figure D-1. The core pressure drop, which depends slightly on core power, was obtained by running a HOTROD calculation for a fuel assembly with average flow, and with average power (represented by setting $\bar{F}_r$ equal to

TABLE D-1. MAXIMUM CONDITION AXIAL POWER PEAKING FACTORS*

<u>Time in Cycle</u>	<u>Zone 1</u>	<u>Zones 2 and 3</u>
BOC	1.67	1.54
100 EFPH	1.52	1.46
1000 EFPH	1.46	1.43
2000 EFPH	1.44	1.41
3000 EFPH	1.40	1.39
4000 EFPH	1.39	1.39
5000 EFPH	1.38	1.38
6000 EFPH	1.38	1.38
7000 EFPH	1.38	1.38

*These values include a 1.15 factor multiplier as discussed in the text.

TABLE D-2. HOTROD INPUT PARAMETERS

<u>Parameter</u>	<u>Common Parameters*</u>	<u>Unique Parameters</u>	
		<u>Reference Core</u>	<u>Prebreeder Core Concepts</u>
Core Power, MW(th)	3800	-	-
Total Flow Rate, 10 ⁶ lb/hr	-	159.0	154.0 (Concept 1) 154.2 (Concept 2) 153.5 (Concept 3)
Percent Flow Reduction Due to Leakage	5.5	-	-
Maldistribution	1.0	-	-
Core Pressure Drop, psi			
Total	-	15.5	22.5 (Concepts 1 and 2) 23.4 (Concept 3)
Unrecoverable	-	11.6	18.6 (Concepts 1 and 2) 19.5 (Concept 3)
Inlet Temperature, °F	-	576.5	575.9
System Pressure, Minimum Steady-State, psia	2205	-	-

*Reference core and prebreeders

TABLE D-2 (Cont)

Parameter	Common Parameters*	Unique Parameters	
		Reference Core	Prebreeder Core Concepts
Grids			
Number of Grids	-	8	10
Average Loss Coefficient per Grid	-	0.75	1.20 (Concepts 1 and 2) 1.23 (Concept 3)
Percent Reduction in Hot Subchannel Flow Area			
Interior Subchannel	2	-	-
Peripheral Subchannel	3	-	-
Percent Reduction in Hot Fuel Assembly Flow Area	2	-	-
Mechanical Hot Channel Factor			
Enthalpy Rise	1.014	-	-
Local Heat Flux	1.025	-	-

*Reference core and prebreeders

1.0 for all fuel rods). The calculated pressure drop was 15.5 psi. Subtracting the elevation pressure drop of 3.9 psi gives an unrecoverable pressure drop of 11.6 psi, the same as that specified in Section 4.4 of Reference D-1.

Additional assumptions used in the analysis are those used in Section 4.4 of Reference D-1 for the "maximum design condition", which represents the most severe conditions. These are as follows. The maximum control band errors on core pressure and inlet temperature were applied in the most conservative manner. Nominal inlet temperature of 573.5°F was increased by 3°F to 576.5°F. Nominal system pressure of 2250 psia was reduced by 45 psi to 2205 psia. The flow area for the most limiting subchannel was the nominal area reduced 2 percent (3 percent for a peripheral subchannel) to account for channel closure. The flow area

of subchannels at the intermodular gap was reduced to represent the conservative situation where the adjacent fuel assemblies are bowed toward the hot fuel assemblies. This reduced the total flow area of the hot fuel assembly by 2 percent. The flow in the hot bundle position was assumed to be 1 percent less than average. A hot channel factor of 1.014 on enthalpy rise and a local hot spot factor of 1.025 on heat flux were applied.

The CHF calculations were performed using the Babcock and Wilcox BAW-2 CHF correlation (Equation 4.4-16 of Reference D-1). Because the correlation programmed in HOTROD was applicable only to an axially uniform heat flux problem, it was necessary to decrement the departure from nucleate boiling ratio (DNBR) calculated by HOTROD in order to account for nonuniform axial power shapes. Equation 4.4-17 of Reference D-1 defines the appropriate factor for decrementing the DNBR heat flux calculated using the BAW-2 CHF correlation. The decremented DNBR was obtained by dividing the HOTROD DNBR by the calculated factor.

The results of both the B&W and HOTROD analyses using the BAW-2 correlation are presented in Table D-3. The DNBR's shown there are departures from nucleate boiling heat flux ratios, and are not power ratios. A 5 percent difference in heat flux ratio is approximately equivalent to a 1 percent difference in power ratio. Therefore, the differences between the Reference D-1 and the HOTROD calculations are equivalent to 4 percent in power capability for the unit cell and only 1 percent in power capability for the control rod cell. The key point from the data in Table D-3 is that, as was the case for the calculations discussed in Section B, the HOTROD calculations are conservative, i.e., they predict a low power capability relative to a reference; this time the reference is the B&W

calculations, which use a model based on a very extensive data base. The HOTROD calculations would give a DNBR of 1.32 for the unit cell at a core power of 111 percent of 3800 MWth.

TABLE D-3. DNBR CALCULATIONS⁽¹⁾ FOR THE REFERENCE CORE

	B&W Analysis Using BAW-2 Correlation		HOTROD Analysis Using BAW-2 Correlation
	Calculated for Equilibrium Cycle	Minimum Acceptable	Calculated for Year 15
Unit Cell ⁽²⁾	1.49	1.32 ⁽⁴⁾	1.27
Control Rod Cell ⁽³⁾	1.54	1.32 ⁽⁴⁾	1.47

(1) Minimum DNBR calculated at power level of 112% of 3800 MWth.

(2) A unit cell is bounded by four fuel rods.

(3) A control rod cell is bounded by three fuel rods and a control rod guide tube.

(4) Specified in Section 4.4 of Reference D-1.

The difference between the DNBR values calculated using HOTROD and those calculated by B&W using their program may be due to the fact that in the HOTROD analysis fluid mixing between the hot subchannel and its surroundings was minimized by smearing the grid loss coefficients over the entire core length. Further, the beneficial effect of mixing from colder fuel assemblies was also not included. These modeling differences would affect calculated mass velocity, quality and DNBR.

For the limiting unit cell the mass velocity calculated by HOTROD was 1.83×10^6 lb/hr-ft², approximately 4 percent less than the reported mass velocity of 1.91×10^6 lb/hr-ft², while the exit quality calculated by HOTROD was 18.0 percent compared with the reported value of 15.3 percent. (The reported

values appear in Section 4.4 of Reference D-1.) These differences in mass velocity and exit quality would account for the difference in the DNBR values. For the limiting control rod cell, the mass velocity calculated using HOTROD was 1.73×10^6 lb/hr-ft² which agreed with the reported value. Exit quality is not reported for this cell in Reference D-1. However, because of the equality of the mass velocities for this cell, the DNBR value calculated using HOTROD should be closer to the value calculated by B&W than it was for the limiting unit cell. This is indeed the case, with the HOTROD and B&W DNBR values being respectively 1.47 and 1.54.

The above differences in calculated mass flow velocities, qualities and DNBR values are not to be considered to imply any shortcoming in the B&W calculational methods which, as noted above, have been qualified by means of comparisons with numerous CHF experiments on their fuel rod geometry. Rather, the reference core results reported in Reference D-1 are taken as part of the basis for qualifying the HOTROD calculational method for this type of geometry.

III. PREBREEDER ANALYSIS

A. Introduction

Overpower thermal capability of the prebreeder core concepts is defined as the power which leads to a DNBR of 1.32 in the most limiting subchannel. As noted in the introduction to this appendix, acceptable thermal capability is the capability to operate at a minimum of 111 percent of 3800 MWth. Based on the results presented in Sections II.B and C, HOTROD with the BAW-2 correlation and a DNBR requirement of 1.32 conservatively predicts the critical heat flux for the reference core and a wide range of experiments where the range of test variables includes the values characteristic of the prebreeder core concepts.

B. General Features of HOTROD Model for Prebreeder Analysis

Fuel assembly parameters for the prebreeder core concepts are presented in Table D-4. The HOTROD modeling was identical in most respects with that used for the analysis of the reference core as described in Sections II.A and C. The HOTROD input parameters are presented in Table D-2. Only the differences between the reference core and prebreeder input parameters are discussed here.

The total thermal design flow for each prebreeder core concept was calculated externally to HOTROD using the reference plant pump head characteristics and the nominal flow resistance (based on nominal dimensions for the prebreeder core concept) at the core average temperature.

The effect of spacer grids was again represented by a loss coefficient which was applied continuously over the entire core length. Loss coefficients for individual subchannels ranged from 0.87 to 1.41. Core average grid loss coefficients would be greater for the prebreeder core concepts than for the reference core because the prebreeder subchannels would have smaller hydraulic diameters than those of the reference core. The higher grid loss coefficients would lead to a greater core pressure drop.

The core average temperature for the prebreeder core concepts was taken to be the same as for the reference core. However, the inlet temperature for the prebreeder core concepts would be slightly lower than that of the reference core because of a slightly lower total flow.

The radial and axial power peaking factors used in the thermal analysis of these prebreeder core concepts are shown in Tables D-1 and D-5. The radial peaking factors were obtained from the best-estimate radial peaking factors given in Section VII.C of Appendix B by multiplying by a factor of 1.16 to obtain the

TABLE D-4. PREBREEDER FUEL ASSEMBLY PARAMETERS

	Core Concept 1 $U^{E}O_2$ Seed ThO_2 Blanket	Core Concept 2 $U^{E}O_2$ Seed ThO_2 - PuO_2 Blanket	Core Concept 3 $U^{HE}O_2$ - ThO_2 Seed ThO_2 Blanket
Number of Seed Rods	248	192	220
Number of Blanket Rods	84	140	112
Number of Control Rods	28	28	28
Number of Instrument Tubes	1	1	1
Seed Rod			
Clad OD, in.	0.349	0.339	0.349
Clad ID, in.	0.302	0.292	0.302
Pellet OD, in.	0.295	0.285	0.295
Blanket Rod			
Clad OD, in.	0.379	0.379	0.379
Clad ID, in.	0.339	0.339	0.339
Pellet OD, in.	0.335	0.335	0.335
Control Rod			
Guide Tube OD, in.	0.426	0.426	0.426
Guide Tube ID, in.	0.391	0.391	0.391
Control Rod OD, in.	0.351	0.351	0.351
Instrument Tube			
Guide Tube OD, in.	0.390	0.390	0.390
Guide Tube ID, in.	0.360	0.360	0.360
Spacer Sleeve OD, in.	0.428 square	0.428 square	0.428 square
Rod Spacing			
Seed-Seed, in.	0.100	0.110	0.100
Seed-Blanket, in.	0.085	0.090	0.085
Seed-Control Rod, in.	0.0615	0.0665	0.0615
Seed-Instrument Tube, in.	0.0605	0.0655	0.0605

TABLE D-5. PREBREEDER MAXIMUM CONDITION RADIAL POWER PEAKING FACTORS, $F_r^{(1)}$,
IN YEAR 15

Time in Life	Zone 1 ⁽²⁾		Zone 2 ⁽²⁾		Zone 3 ⁽²⁾	
	Seed	Blanket	Seed	Blanket	Seed	Blanket
Core Concept 1	(UO ₂ Seed, ThO ₂ Blanket, No Plutonium Recycle)					
BOC	1.85	0.03	1.85	0.40	1.61	0.67
100 EFPH	1.87	0.03	1.81	0.39	1.58	0.66
1000 EFPH	1.88	0.05	1.79	0.40	1.52	0.64
2000 EFPH	1.90	0.10	1.76	0.44	1.47	0.64
3000 EFPH	1.88	0.16	1.72	0.48	1.43	0.65
4000 EFPH	1.85	0.21	1.69	0.52	1.41	0.68
5000 EFPH	1.82	0.26	1.66	0.56	1.40	0.70
6000 EFPH	1.79	0.30	1.63	0.60	1.39	0.72
7000 EFPH	1.75	0.35	1.60	0.64	1.39	0.75
Core Concept 2	(UO ₂ Seed, ThO ₂ -PuO ₂ Blanket, Plutonium Recycle)					
BOC	1.80	0.95	1.77	0.87	1.59	0.87
100 EFPH	1.74	0.92	1.81	0.89	1.62	0.88
1000 EFPH	1.69	0.85	1.87	0.90	1.66	0.89
2000 EFPH	1.68	0.83	1.87	0.90	1.65	0.89
3000 EFPH	1.68	0.81	1.86	0.91	1.63	0.89
4000 EFPH	1.67	0.80	1.85	0.92	1.64	0.90
5000 EFPH	1.67	0.79	1.82	0.92	1.59	0.90
6000 EFPH	1.67	0.79	1.80	0.92	1.56	0.90
7000 EFPH	1.67	0.80	1.76	0.92	1.54	0.91
Core Concept 3	(U ^{HE} O ₂ -ThO ₂ Seed, ThO ₂ Blanket, No Plutonium Recycle)					
BOC	2.03	0.03	2.01	0.34	1.72	0.63
100 EFPH	2.00	0.03	2.01	0.34	1.71	0.63
1000 EFPH	2.00	0.05	1.96	0.36	1.68	0.63
2000 EFPH	1.99	0.08	1.91	0.39	1.64	0.66
3000 EFPH	1.97	0.13	1.86	0.44	1.61	0.68
4000 EFPH	1.95	0.17	1.81	0.48	1.58	0.71
5000 EFPH	1.93	0.21	1.78	0.53	1.55	0.74
6000 EFPH	1.90	0.26	1.75	0.57	1.52	0.76
7000 EFPH	1.88	0.30	1.72	0.61	1.49	0.79

(1) As discussed in the text, these factors include a 1.16 multiplier on the best estimate factors.

(2) Fuel management zones are defined in Section III.A of Appendix B.

equivalent of a maximum design condition. This factor was based on Reference D-1 where the best estimate radial factor of 1.34 (given in Section 4.3) was increased by a factor of 1.16 to obtain the factor of 1.55 (given in Section 4.4) for the maximum design condition. The radial power shape within the hot fuel assembly was assumed to be the same as that for the reference core hot fuel assembly shown in Figure D-1. For each prebreeder core concept, once the peak radial power in the hot fuel assembly was defined, the relative radial power among individual fuel rods was determined using the shape from Figure D-1. The axial power peaking factors were the same as those used for the reference core analysis described above in Section II.C.

C. Thermal Performance of Prebreeder Core Concept 1

The arrangement of seed and blanket rods for prebreeder core Concept 1 is shown in Figure D-2. No blanket rods would be placed immediately adjacent to each other, thus eliminating any 70-mil rod spacings which would have resulted. In addition, no blanket rods would be placed adjacent to control rod guide tubes, eliminating any 46.5-mil rod spacings which would have resulted. Additional seed rods, which would produce more power than blanket rods, would be placed in the outer row of the fuel assembly where the radial peaking factors would generally be lower due to the absence of control rod guide tubes. The location of the control rod guide tubes, which would be the same for all three prebreeder core concepts, was established to meet mechanical and arrangement requirements. No subchannel, except those on the outer edge between two fuel assemblies, would contain more than three seed rods since this would reduce the thermal capability of the concept.

For this core concept, Zone 1 at beginning of cycle and at 2000 effective full power hours (EFPH) and Zone 2 at beginning of cycle were analyzed with

Figure D-2. 19 by 19 Array Prebreeder Core Concept 1
(HOTROD Model, 1/8 Fuel Assembly)

HOTROD, since it was clear from examining Tables D-1 and D-5 that these would be the limiting times and locations with a maximum seed radial peaking factor in Zone 1 of 1.85 at beginning of cycle and 1.90 at 2000 EFPH and in Zone 2 of 1.85 at beginning of cycle. Based on the radial shape of Figure D-1, the peak seed rod value of 1.85 (Zone 1, BOC) is assumed to occur near Channel 27 of Figure D-2, at the same radial location (to within the nearest fuel rod) as the peak value of 1.55 in the reference core shown in Figure D-1.

The results of the analysis are summarized in Tables D-6 and D-7. The minimum overpower capability of prebreeder core Concept 1 would be 114 percent of the nominal power rating. The overpower capability in Zones 1 and 2 at beginning of cycle would be equal, while Zone 1 at 2000 EFPH would have 4 percent greater overpower capability. For all other zones and times in life the overpower capability would be greater than 114 percent. All values are based on the power level which would produce a minimum DNBR of 1.32 using the BAW-2 critical heat flux correlation.

The most limiting subchannel was found to be Channel 27 (Figure D-2). The next most limiting subchannels would be Channels 22 and 28, which would have 1 to 2 percent greater overpower capability than Channel 27. Additional thermal and hydraulic information concerning the limiting channel is provided in Table D-7.

D. Thermal Performance of Prebreeder Core Concept 2

The arrangement of seed and blanket rods for this prebreeder core concept is shown in Figure D-3. The arrangement was established using basically the same guidelines as for prebreeder core Concept 1.

TABLE D-6. CHF OVERPOWER CAPABILITY PREDICTED BY HOTROD*

	Overpower (% of 3800 MW)	<u>Prebreeder Overpower Capability</u> <u>Reference Core Overpower Capability</u>
Reference Core	111	-
Prebreeder Core Concept 1	114	1.03
2	117	1.06
3	112	1.01

*CHF capability is based on a minimum DNBR of 1.32 with the BAW-2 CHF correlation.

The results of the analysis are summarized in Tables D-6 and D-7. The minimum overpower capability would be 117 percent of the nominal power rating. The overpower capability would be limited by Zone 1 at the beginning of cycle where the maximum seed radial peaking factor would be 1.80, the blanket radial peaking factor would be 0.95, and the maximum axial peaking factor would be 1.67. The limiting subchannel would be Channel 20. Zone 2 at 1000 EFPH was also analyzed to ensure that it would be less limiting than Zone 1 at the beginning of cycle, and was shown to have approximately 121 percent overpower capability. For all other zones and times in life, the overpower capability would be equal to or greater than 117 percent.

Several other channels were found to be nearly as limiting as Channel 20. For example, Channels 4, 8, 9, 12, 13, 17, 18, 19, 21, 23, and 26 would all be within 2 percent of Channel 20 in overpower capability. Additional thermal and hydraulic information for the limiting channel is shown in Table D-7.

TABLE D-7. HOTROD ANALYSIS RESULTS

	Reference Core	Prebreeder Core Concepts		
		Concept 1 UO ₂ Seed ThO ₂ Blanket (No Pu Recycle)	Concept 2 UO ₂ Seed ThO ₂ -PuO ₂ Blanket (Pu Recycle)	Concept 3 UHEO ₂ -ThO ₂ Seed ThO ₂ Blanket (No Pu Recycle)
Power Level for Minimum DNBR = 1.32 (% of 3800 MW)	111.1	113.8	117.2	112.5
Minimum DNBR at 112% of 3800 MW	1.27	1.42	1.58	1.35
Limiting Subchannel	17 (Fig. D-1)	27 (Fig. D-2)	20 (Fig. D-3)	37 (Fig. D-4)
Limiting Zone and Time in Life		Zone 1, BOC	Zone 1, BOC	Zone 1, BOC
Thermal Conditions for Limiting Subchannel (at Power level Where min. DNBR = 1.32)				
Mass Velocity*, (10 ⁶ lb/hr-ft ²)				
Average	1.84	1.81	1.95	2.16
Minimum	1.72	1.66	1.86	1.83
Exit Quality, %	17.5	18.4	17.5	17.1
Exit Enthalpy, BTU/lb	770	774	770	768
Distance from Fuel Bottom for Minimum DNBR, inches	104	107	104	107
Radial Power Peaking Factor				
Seed	1.55	1.85	1.80	1.82
Blanket	--	0.03	0.95	0.03
Maximum Axial Power Peaking Factor	1.67	1.67	1.67	1.67

*The mass velocities listed here are for the limiting subchannel in each core. The core average mass velocities for the three prebreeder core concepts would be nearly identical as shown in Table 2 of the main text of this report. The large variation in mass velocities shown for the limiting subchannel is due to the large variation in flow area. Prebreeder core Concept 3 would be limiting in Channel 37 which would have a large flow area (and hydraulic diameter) due to the gap between fuel assemblies, and thus would have a large mass velocity.

Figure D-3. 19 by 19 Array Prebreeder Core Concept 2
(HOTROD Model, 1/8 Fuel Assembly)

E. Thermal Performance of Prebreeder Core Concept 3

The arrangement of seed and blanket rods for this prebreeder core concept is shown in Figure D-4. The arrangement was established using basically the same guidelines as for prebreeder core Concept 1.

The results of the analyses are summarized in Tables D-6 and D-7. The minimum overpower capability would be 112 percent of the nominal power rating. As was the case for prebreeder core Concepts 1 and 2, the limiting condition would occur in zone 1 at the beginning of a cycle. However, although the radial seed rod peak F_r of 2.03 would occur near Channels 20 and 27, the most limiting condition would occur in Channel 37 (at the fuel assembly edge) where the value of F_r would be 1.82 but the channel would be entirely surrounded by seed rods. The interior channels would not be limiting because, of the four bounding rods on each channel, only the two seed rods would produce high power. For all other zones and times in life, the overpower capability would be equal to or greater than 112 percent. Channels 38 and 39 were predicted to have overpower capability within 2 percent of Channel 37. Additional thermal and hydraulic information for the limiting channel is shown in Table D-7.

IV. LOSS-OF-COOLANT-ACCIDENT PERFORMANCE EVALUATION

The performance of the three prebreeder concepts during a hypothetical loss-of-coolant-accident (LOCA) was evaluated by comparing their calculated behavior and margins to design criteria limits with those of the reference core calculated in the same manner. The FLASH-6 computer program (Reference D-9) was used to calculate the first ten seconds of the blowdown portion of the LOCA from a double-ended guillotine break at a coolant pump discharge, combined with sufficient pipe displacement to ensure that the leak flow from the two ends of the pipe do not interact. Section 6.3 of Reference D-1 states that this accident results

Figure D-4. 19 by 19 Array Prebreeder Core Concept 3
(HOTROD Model, 1/8 Fuel Assembly)

in the highest fuel rod clad temperatures, and imposes the most severe requirements on the ECCS (Emergency Core Cooling System). The blowdown period is defined as the time between the inception of the accident and the time when the primary system has depressurized to the point where there is zero leak flow. Calculations for the reference core, reported in Section 6.3 of Reference D-1, indicate that end of blowdown would occur 24 seconds after the start of the accident. Present calculations were made for only the first ten seconds of the blowdown period because the calculations reported in Reference D-1 indicated that the peak fuel temperature occurred during this period; further, the behavior of all four cores during this period was similar, and at the end of this period the conditions in all four cores were similar. These results were taken to indicate that all four cores would have similar behavior during the remainder of the LOCA.

The FLASH-6 program models the reactor plant and all of the important physical processes involved during the LOCA including reactor power, fuel temperature, fission and decay heating, fuel/clad gap dimensions and conductance, cladding swelling and flow area reduction, cladding oxidation, zirconium-water reaction rates, and heat transfer and coolant flow over the entire range of possible liquid-vapor conditions. The hot fuel assembly flow and hot spot fuel rod temperatures as calculated by FLASH-6 were chosen as the indicators of core performance during the LOCA. The reference plant has two primary coolant loops, each split into two pump branches in the cold-leg. Each loop contains two centrifugal pumps and a downflow vertical steam generator, and a pressurizer is connected to one of the loops. These components were explicitly represented in the FLASH calculations. The ECCS was not included in the modeling because

it would not be activated during the first ten seconds of the transient. Section 6.3 of Reference D-1 notes that lag times of at least 35 seconds are assumed between receipt of the activation signal and injection of ECCS water.

Fluid enthalpy change during the LOCA was calculated for both the hottest fuel assembly and the remainder of the core. Hot spot fuel rod temperature was calculated by representing the hottest seed rod in contact with the fluid in the hottest fuel assembly. This is consistent with the assumption made under similar conditions in the safety analysis for the Shippingport LWBR Core (Reference D-10).

In the calculations for prebreeder core Concept 3, the thermal conductivity of the thoria-urania fuel was taken to be the same as that of the urania fuel in the other cores. The plant model, the axial power distributions and the power decay behavior were assumed to be the same for the prebreeder core concepts and for the reference core because all cores would be capable of operating in the reference plant with the reference control system.

The blowdown for the reference core and the three prebreeder core concepts would be very similar, as shown by the hot fuel assembly flow rates in Figure D-5. After the LOCA is initiated, the fuel clad temperature would rise steadily to the peak value as core power decayed to the point that decay heat would become the primary contributor to the core heat burden; the temperature would then drop rapidly as a result of a liquid water surge dropping down through the core. Figure D-6 shows that the cladding temperatures for the hot spot fuel location in all four cases would exhibit the same overall behavior and the peak temperatures reached would be very nearly the same, lying in the band between 1630°F and 1810°F. These results were taken to indicate that during the remainder of the blowdown phase, the behavior of the prebreeder core concepts would be similar

Figure D-5. Flow Rate in Hottest Fuel Assembly during Loss-of-Coolant Accident for Reference Core and Prebreeder Core Concepts

Figure D-6. Peak Cladding Temperature during Loss-of-Coolant Accident for Reference Core and Prebreeder Core Concepts

to that of the reference core and that similar conditions would exist in all four cores at the start of the reflood portion of the transient when the blowdown is complete. Since the plant would be identical in all cases and the initial temperatures would be nearly the same, it is expected that the reflood behavior of the four cores would be very similar also.

V. CONCLUSIONS

Explicit steady state thermal and hydraulic calculations were performed for the three prebreeder core concepts using the HOTROD computer program, the BAW-2 critical heat flux correlation and engineering factors (such as flow area reductions) consistent with those used for the reference core analysis reported in Reference D-1. The HOTROD model was qualified, and shown to predict thermal capability conservatively by calculations on the reference core and on a substantial number of CHF tests.

The calculated steady state overpower capability (percent of 3800 MWth) for the three prebreeder core concepts was 114 percent for Concept 1, 117 percent for Concept 2, and 112 percent for Concept 3. The overpower capability calculated by the authors for the reference core was 111 percent. Based on this result and the similarity of kinetics parameters between the prebreeder core concepts and the reference core, it was judged that a minimum steady state overpower capability of 111 percent would indicate acceptable prebreeder performance. For all three prebreeder core concepts, the overpower capability would be limited by conditions in Zone 1 at beginning of cycle; at all other times in life the overpower capability would be equal to or greater than this initial capability.

Explicit calculations of the first 10 seconds of the hypothetical LOCA demonstrated that the overall blowdown behavior of the four cores would be very

similar and that during blowdown the maximum cladding temperatures would all be in the range 1630°F to 1810°F, well below the established limit of 2200°F. Stable temperatures reached would also be very similar, indicating that the initial conditions for the reflood portion of the LOCA would be similar.

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APPENDIX E
RESOURCE AND SERVICE REQUIREMENTS AND
ENVIRONMENTAL CONSIDERATIONS

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APPENDIX E

RESOURCE AND SERVICE REQUIREMENTS AND
ENVIRONMENTAL CONSIDERATIONS

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APPENDIX E
RESOURCE AND SERVICE REQUIREMENTS AND
ENVIRONMENTAL CONSIDERATIONS

I. INTRODUCTION

Resource and service requirements and environmental effects were estimated for each of the prebreeder core concepts and for the reference core using a three-step process.

1. The nuclear calculations provided the mass flow rates of fissile and fertile heavy metal in fresh fuel and in spent fuel.
2. The mass flows from Step 1 determined the mass flows through each of the facilities in the fuel cycle from mining through waste management.
3. For each facility in the fuel cycle the environmental effects in each of nine categories were calculated as the product of the mass flow and an environmental effect factor. The environmental effect factors, defined as an environmental effect per unit mass flow, were based on data in Reference E-1.

This appendix presents a discussion of Steps 2 and 3 above, the data used to define the environmental effect factors, and the results of the calculations. The principal conclusion is that all three prebreeder core concepts and the reference core would have approximately the same requirements and effects per year of operation.

II. DISCUSSION

A. Background

The nuclear fuel cycle consists of all the activities needed to extract energy from nuclear resources. Current light water nuclear reactors are fueled with uranium which has been enriched in uranium-235, and fabricated into fuel

elements for the reactor core. Spent fuel from these plants is currently stored because of the delay of reprocessing operations. However, for purposes of estimating the environmental impacts of a closed fuel cycle it has been assumed here that spent fuel would be reprocessed and that some or all of the recovered fissile and fertile nuclides would be recycled.

In the Shippingport LWR Environmental Statement (Reference E-1), a comprehensive assessment of a conceptual thorium-uranium fuel cycle is presented. The discussion and calculations presented here are based largely upon that assessment.

The current uranium fuel cycle, and a conceptual thorium-uranium fuel cycle are represented by the schematic diagram in Figure E-1 which shows the principal operations in the nuclear fuel cycle and the streams of nuclear material.

The present calculations assume the same operating characteristics for the conceptual fuel cycle facilities as were assumed for the calculations of Reference E-1. In addition, a second assumption is that the magnitudes of the environmental effects from each conceptual facility of Reference E-1 would be directly proportional to the mass flow rate input to that facility. This second assumption is the basis for calculating the environmental effect factors discussed below. This assumption would yield accurate results if the isotopic composition of the fuel materials from the present prebreeder core concepts were identical with those from the prebreeders of Reference E-1, or if the effects from a given facility were independent of the isotopic composition of the material input. This second assumption is expected to yield reasonably accurate results both because the isotopic compositions of the present prebreeder materials are similar or identical to those of the prebreeders of Reference E-1, and because many of the environmental effects are independent of the isotopic content of the materials.

Figure E-1. Nuclear Fuel Cycle (with Fuel Recycle Assumed)

B. Fuel Recycle Assumptions

This section describes briefly some of the assumptions which were made relative to the materials to be recycled. These assumptions affect some of the processes in the conceptual fuel cycle.

1. Reprocessing

Reprocessing would separate the valuable nuclides (to be recycled) in spent fuel from waste materials such as fission products. The spent fuel from a prebreeder is assumed to be separated into five streams,* as follows:

- a. Thorium.
- b. Uranium bred from thorium in blanket rods. This would be primarily uranium-233, with small amounts of uranium-232, 234, 235, and 236. The uranium-232 and high enrichment would present special handling requirements.
- c. Uranium from seed rods. For prebreeder core Concepts 1 and 2, this would be primarily uranium-235 and uranium-238, with a uranium-235 enrichment of about 2 to 3 percent. For prebreeder core Concept 3, this would contain primarily uranium isotopes 233, 235, and 238, with a combined 233 plus 235 enrichment of about 60 percent. The uranium-232 and high enrichment would present special handling requirements.
- d. Plutonium bred from uranium-238 in seed rods, or unburned once recycled plutonium from the blanket rods of prebreeder core Concept 2.**

*For the reference core, only the last three streams would be present.

**Refer to Section III.C.1 of the main text for details of assumed plutonium recycle in this core.

- e. Fission products and waste materials, including cladding hulls, fuel assembly structure and unburned twice recycled plutonium from the blanket rods of prebreeder core Concept 2.*

The first four streams would contain nuclides which could be recycled. The fifth stream, which will be referred to simply as the waste stream, would actually consist of several streams containing different types of wastes, but for present purposes it is sufficient to treat these as a single stream. The waste stream would also include small quantities of the nuclides present in the first four streams. These small quantities of valuable nuclides are referred to as reprocessing losses because even though their physical location would be known, they could not be economically recovered from the waste stream and therefore would be "lost" with respect to recycle of fuel materials. In Reference E-1, it was assumed that these losses would be a maximum of 0.5 percent in reprocessing, i.e., considering just the nuclides to be recycled, of each type of nuclide input to reprocessing, less than 0.5 percent would be lost to the waste stream. For present calculations, it was assumed that losses are 0.5 percent. Such small losses are of small consequence to the resource or environmental considerations of the present work.**

Reprocessing would consist conceptually of chopping spent fuel rods into short segments, placing the segments into a chemical solution to dissolve the fuel material, and chemically processing the resulting mixture to obtain the desired product streams. One of the primary requirements for the reprocessing of prebreeder fuel would be to recover the relatively pure uranium-233 bred in

*Refer to Section III.C.1 of the main text for details of assumed plutonium recycle in this core.

**This is in contrast to the situation which would occur in a breeder fuel cycle where such losses of the fissile nuclides, uranium-233 and uranium-235, could have significant consequences with respect to long term consumption of uranium.

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the blanket rods while keeping this bred uranium separate from uranium in the seed rods. Two possible processes for making this separation would be as follows:

- a. At the head end of the reprocessing operation, prior to chopping of the fuel rods, remove them intact from the fuel assemblies and mechanically separate the seed rods from the blanket rods. The two types of rods could then be processed separately.
- b. Chop both seed and blanket rods together and use a differential dissolution process. Conceptually, this would require a two stage dissolver; the first stage would dissolve only one type of fuel (seed or blanket), then the second stage would dissolve the other. Such a process would not be applicable to prebreeder core Concept 3 because both seed and blanket fuel would be primarily thorium dioxide.

In estimating the environmental considerations for reprocessing, it has been assumed that they would be the same for the above two processes.

2. Fuel Fabrication

The material used for fuel fabrication would include fresh uranium, fresh thorium and recycled uranium from seed rods. For prebreeder core Concept 2 plutonium from seed rods and blanket rods would also be used.

In the case of prebreeder core Concepts 1 and 2 and the reference core, the recycled uranium would contain only the isotopes 234, 235, 236, and 238,* and it is assumed here that this recycled uranium could be processed in the same way as natural uranium.

*It is recognized that this uranium would contain traces of uranium-232 from alpha decay of plutonium-236. However, it is assumed here that the uranium-232 concentration would be sufficiently low that the recycled uranium could still be handled the same way as natural uranium.

The uranium-233 used in prebreeder core Concept 3 would contain a significant amount of uranium-232, which has a 72-year half life and energetic gamma-emitting decay products, primarily thallium-208. Although when freshly separated from its decay products, uranium-233 (containing uranium-232) would contain no energetic gamma emitting decay products, the decay product concentrations would rapidly build up to significant levels. Hence, it is assumed here that the seed fuel for prebreeder core Concept 3 would require remote fabrication in a facility similar to that assumed in Reference E-1 for the fabrication of fuel for a light water cooled breeder.

Spent thorium from any of the prebreeders would contain more than the naturally occurring amount of thorium-228 because of alpha decay of uranium-232. Thorium-228, which has a half life of 1.9 years, also has thallium-208 as a decay product. The use of recycled thorium in prebreeders would require either remote fabrication of thorium fuel, or a long storage time, on the order of 20 or more years prior to reuse, to allow decay of the thorium-228. Because of this, one of the assumptions concerning fuel recycle is that only fresh thorium would be used in fresh fuel for prebreeder core Concepts 1 and 2. For prebreeder core Concept 3, where fresh seed fuel would contain uranium-232 from the recycled uranium, there would probably be no reason not to use recycled thorium for the seed rods since remote fabrication would be required anyway. Nevertheless it has been assumed that only fresh thorium is used in prebreeder core Concept 3 to put the thorium requirements for all three prebreeders on a consistent basis.

In the fabrication process, a small amount of fuel material would wind up in waste materials such as rubber gloves, wipes, and discarded equipment. This

is referred to as a "loss" of fuel material (just as in reprocessing) because even though the physical location of the material would be known, it could not be recovered economically and therefore would be lost with respect to use in the reactor. In Reference E-1, these fabrication losses were assumed to be less than 0.5 percent and for the present calculations these losses were also assumed to be 0.5 percent. Again, these small losses are of no consequence to the environmental or resource considerations presented here.

3. Uranium Enrichment

The enrichment of a mass of uranium is defined in terms of the concentration of uranium-235 in the uranium. In the enrichment process, a uranium feed stream of some enrichment is processed into a product stream (of higher enrichment than that of the feed stream) and a tails stream (of lower enrichment than that of the feed stream). The amount of processing which has to be performed to produce a given amount of product at the desired enrichment is measured in terms of separative work units (SWU). The enrichment of the tails stream is sufficiently low that the uranium in this stream is not used as fuel in light water reactors, and a large amount of separative work would be required (using currently available enrichment technology) to remove any of the uranium-235 from the tails stream; thus, the uranium-235 in the tails stream is assumed to be unavailable for use in light water reactors.

For given feed and product enrichments, the enrichment of the tails stream would determine both the amount of feed required and the amount of separative work required to produce a unit mass of product. As the tails enrichment decreases, the amount of feed decreases, but the amount of separative work increases. For

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the calculations made here, a tails enrichment of 0.2 percent was chosen to provide a reasonable balance between uranium feed required and separative work required.

It was assumed that all uranium recovered from the reference core or from spent seed fuel from prebreeder core Concepts 1 and 2 would eventually be reenriched and reused in seed fuel for the prebreeders. This includes the implicit assumption that the uranium-232 content of the recovered uranium would be below the maximum permissible level for enrichment plants, currently 0.11 parts uranium-232 per million parts uranium-235. The uranium recovered from the spent seed fuel of prebreeder core Concept 3 would be approximately 0.15 percent uranium-232, and consequently would probably be precluded from reenrichment in either gaseous diffusion or centrifuge enrichment facilities. Therefore, it was assumed that uranium recovered from prebreeder core Concept 3, containing about 29 percent uranium-235, 26 percent uranium-233, 27 percent uranium-236 and 11 percent uranium-238 would be mixed with highly enriched uranium (about 93 percent enrichment) to obtain uranium for fresh fuel containing approximately 50 percent uranium-235, 17 percent uranium-233, 18 percent uranium-236 and 10 percent uranium-238.

The uranium recovered from the reference core and from spent seed fuel of prebreeder core Concepts 1 and 2 would also contain significant quantities of uranium-236 (produced by non-fissioning capture in uranium-235) in addition to uranium-235 and uranium-238. The enrichment calculations showed that more than half of the uranium-236 present in the feed stream passed into the product stream, and the fissile uranium loading of the fresh fuel used in the nuclear calculations accounted for this amount of uranium-236.

It has also been assumed that no loss of uranium would occur during the enrichment process.

C. Resources, Services, and Environmental Effects

The resource and service requirements and environmental effects for the prebreeder core concepts and for the reference core are presented here. The requirements and effects considered here include those from operation of all of the major fuel cycle facilities except the operation of the nuclear plants to produce electricity (and in the case of the prebreeders to produce uranium-233 as well). The operation of the nuclear plants is not included because this operation, and therefore the requirements for and effects of this operation, are assumed to be independent of the type of core installed. The requirements and effects considered here also do not include the materials required for construction of the fuel cycle facilities, nor do they include the environmental effects during construction of the facilities.

It is important to recognize that for the prebreeder core concepts the requirements and environmental considerations would apply only during the time that the prebreeders would be operated to produce uranium-233 for breeders. Subsequent replacement of prebreeders by breeders would eliminate the requirements for enrichment, eliminate or greatly reduce the requirements for mining, and would probably increase the requirements for fuel fabrication and reprocessing. The effects of water cooled breeder operation were considered in detail in Reference E-1, but such considerations are outside the scope of the present work.

1. Nuclear Resource Requirements

a. Uranium From Mining - The annual requirements for uranium from mining are presented in Table E-1.

TABLE E-1 CALCULATED ANNUAL RESOURCE AND SERVICE REQUIREMENTS TO SUPPORT OPERATION OF A 1322 MWE REACTOR

	Reference Core	Prebreeder Concepts		
		1	2	3
Uranium				
Overburden moved* - 10 ⁶ MT	2.77	3.03	2.59	2.35
Ore mined 10 ³ MT	95	104	89	80
U ₃ O ₈ from mining - ST	209	228	195	175
Thorium ⁽¹⁾				
Overburden moved - 10 ⁶ MT	0	0.01	0.01	0.03
Ore mined 10 ³ MT	0	2.7	4.4	8.6
ThO ₂ from mining - ST	0	15	24	47
Separative work MTS ¹ U**	160	197	175	172
Fuel fabrication - thousands of fuel rods	21	27	27	27
Reprocessing MTHM ⁽²⁾	36	39	39	39 ⁽³⁾

(1) Assumes no recycle of thorium

(2) Metric Tons Heavy Metal. The values also represent very nearly the mass of heavy metal fabricated into fuel rods

(3) 23 MT of seed fuel and 16 MT of blanket fuel

*Refer to Section II.C.3.c of this Appendix

**Refer to Section III.C of this Appendix

The first assumption affecting uranium requirements is that the uranium recovered from spent seed fuel rods would be eventually recycled to the prebreeders as described in Section B.3. Another assumption which affects the uranium mining requirement is that plutonium would not be recycled in prebreeder core Concept 1, but would be recycled in prebreeder core Concept 2 in the thorium blanket rods. The final assumption is that there would be no losses in extracting uranium from the ore sent through the milling facility. This assumption has no impact on the relative amounts of uranium from mining required for the different core types, or on the relative inputs of services and environmental effects

associated with uranium mining and milling. This assumption was made in Reference E-1 and would be consistent with the objective of operating prebreeders in order to eventually conserve scarce uranium resources. Milling losses in 1977 were approximately 8.2 percent.

It is observed that the prebreeder annual uranium mining requirements would not be substantially different from those of the reference core. The annual requirements for prebreeder core Concepts 2 and 3 would be lower than for prebreeder core Concept 1, since all seed fissile material in prebreeder core Concepts 2 and 3 would be recycled but plutonium from prebreeder core Concept 1 would not be recycled. The annual requirement for prebreeder core Concept 3 would be lower than for prebreeder core Concept 2, because all of the fertile material in prebreeder core Concept 3 would be thorium, so that the recycled fissile fuel would contain a large proportion of uranium-233, which would yield better fuel utilization in light water reactors than either uranium-235 or plutonium.

b. Thorium From Mining - The annual requirements for thorium from mining are also presented in Table E-1. Here again it was assumed that there were no losses in extracting thorium from the ore sent through the milling facility. Since there is presently no large scale thorium mining and milling industry, there are no data on milling losses for such an industry. It was also assumed that all of the thorium discharged from prebreeders would be stored for eventual use in breeders; any thorium in fresh prebreeder fuel would be fresh thorium. For prebreeder core Concepts 1 and 2 the basis for this assumption is that with the use of only fresh thorium, the prebreeder fuel would contain little or no uranium-232 and therefore would probably not require relatively expensive remote fabrication. For prebreeder core Concept 3, where uranium-233 containing significant amounts of uranium-232 would be recycled and remote fabrication of seed

fuel would be required even if fresh thorium were used in the seed fuel, the assumption is the same as for prebreeder core Concepts 1 and 2 simply for the sake of consistent treatment of thorium requirements.

It is observed that the annual thorium mining requirements would be nearly the same for prebreeder core Concepts 2 and 3, and would be about twice as large as those for prebreeder core Concept 1. This difference occurs because prebreeder core Concept 1 would have no fissile fuel in the fresh thorium fuel; fresh thorium fuel would therefore produce almost no power so the core volume which could be made available for thorium would be quite limited.

2. Services

The services considered here are mining, enrichment, fuel fabrication, and reprocessing (including waste storage). Requirements for these are presented in Table E-1. The amounts of other services which would be required are not presented because they are approximately proportional to these four. These other services include milling (in which U_3O_8 and ThO_2 are extracted from their respective ores), which is proportional to the amount of ore mined; conversion of U_3O_8 to UF_6 , which is proportional to the amount of U_3O_8 obtained from mining; transportation between facilities, which is proportional to the amount of material input to the facilities; and management of radioactive wastes from other than reprocessing facilities, which is proportional to the throughput for each facility.

a. Uranium and Thorium Ore Mining - The amount of ore mined would depend on both the amount of U_3O_8 and ThO_2 required from mining and on the grade of the ore, that is, on the concentration of U_3O_8 in uranium ore and on the

concentration of ThO_2 in thorium ore. The base data presented in Reference E-1 assumed concentrations of 0.2 percent U_3O_8 in uranium ore and 0.5 percent ThO_2 in thorium ore. Reference E-1 also assumed that both types of ore would be produced from open pit mines, with a ratio of overburden moved to ore mined of about 30 to 1 for uranium mining and about 3 to 1 for thorium mining; the latter figure is based on the geological characteristics of known thorium deposits, since no open pit thorium mines are currently in operation.

b. Enrichment - It is observed that the annual enrichment requirements for the three prebreeders would be very similar to those for the reference core. Of the four cores prebreeder core Concept 3 would have the smallest uranium mining requirements, but it would not have the smallest enrichment requirements because of the high (93 percent) enrichment required for the prebreeder core Concept 3 uranium.

c. Fuel Fabrication and Reprocessing - The number of fuel rods fabricated each year would be the same for all three prebreeder core concepts, and the mass of heavy metal fabricated and reprocessed each year would be very nearly the same for each prebreeder core concept. The quantities for the prebreeders would be slightly higher than those for the reference core. The prebreeders would have more rods (19 x 19 array versus 17 x 17 array) and the objective of maximizing the amount of thorium irradiated would account for the higher mass of heavy metal. With respect to fuel fabrication, it should be noted again that the uranium-233 bearing fuel rods of prebreeder core Concept 3, which comprise about 60 percent of the heavy metal loading, would require remote fabrication because of radiation from the daughters of uranium-232.

3. Environmental Considerations

Table E-2 presents a brief summary of environmental considerations for each type of reactor core. This summary lists nine categories and five fuel cycle operations; each of the entries was determined by summing the considerations from the much more detailed listing presented in Reference E-1. The relatively small effects due to transportation have not been included.

The principal conclusion from a review of the data in this table is that the environmental considerations would be essentially the same for all four reactor cores. The only exception is that no thorium mining would be necessary for the reference core. It should be noted that once the prebreeders have produced sufficient uranium-233 for a breeder, the environmental impact would become smaller because no more uranium mining, milling or enrichment would be required. By contrast, the present LWR's would continue expending uranium so long as that unit of nuclear power generation is needed.

It is important to note that these environmental effects are only estimates based on conceptual model facilities that could be operating in the future. Advances in fuel cycle technology or changes in environmental regulations could result in changes in the environmental effects. In any future industry, all releases and discharges would comply with applicable environmental regulations regarding the quality of affected air and water and radiological doses to affected individuals. However, based on the results presented below, it is judged that advances in technology or changes in environmental regulations would approximately equally affect the environmental effects from all four types of cores, so that the relative values of the environmental effects would remain comparable for all four types of cores.

TABLE E-2 CALCULATED ANNUAL ENVIRONMENTAL CONSIDERATIONS
FOR SUPPORT OF A 1322 MWE REACTOR

TABLE 10. CALCULATED ANNUAL ENVIRONMENTAL CONSIDERATIONS FOR
CYCLE COMPONENT

Consideration	ENVIRONMENTAL CONSIDERATION FOR EACH FUEL CYCLE COMPONENT						Total Environ- mental Consider- ation
	Reactor (1)	Uranium Production (2)	Thorium Production (3)	Uranium Enrichment	Fuel Fabrication	Reprocessing (4)	
Land permanently committed (acres per facility life)	PBR-1	80	1	0	0	2	83
	2	68	2	0	0	2	72
	3	61	4	0	0	2	67
	REF LWR	73	0	0	0	2	75
Land temporarily committed (acre years per reactor year)	PBR-1	583	17	90	19	77	786
	2	498	27	80	19	77	701
	3	447	55	78	40	77	697
	REF LWR	536	0	73	17	71	697
Total water (10 ⁶ gallons)	PBR-1	188	3	167	2	9	369
	2	161	6	149	2	9	327
	3	144	12	146	1	9	312
	REF LWR	172	0	136	2	8	318
Petroleum and other liquid fuels (10 ³ gallons) (150 standard cubic feet of natural gas = 1 gal- lon petroleum)	PBR-1	676	13	12	0	293	994
	2	578	22	11	0	293	904
	3	519	43	11	0	293	866
	REF LWR	620	0	10	0	270	900
Coal (MT)	PBR-1	1520	18	111,100	632	464	114,000
	2	1300	29	98,700	632	464	101,000
	3	1160	58	97,000	507	428	99,200
	REF LWR	1390	0	90,200	583	428	92,600
Chemicals (MT) to air and water	PBR-1	113	1	1220	11	30	1380
	2	97	1	1090	11	30	1230
	3	87	2	1070	11	30	1200
	REF LWR	103	0	992	11	28	1130
Radiological dose ⁽⁶⁾ commitment (man rem)	PBR-1	634	n ⁽⁵⁾	n	n	38	672
	2	542	n	n	n	38	587
	3	487	n	n	n	38	525
	REF LWR	582	0	n	n	34	616
Thermal (10 ⁹ BTU)	PBR-1	142	3	4530	27	85	4790
	2	122	4	4030	27	85	4270
	3	110	8	3960	27	79	4180
	REF LWR	131	0	3680	25	79	3920
Electrical energy (Megawatt years)	PBR-1	0.7	n ⁽⁵⁾	55.0	0.3	0.2	56.2
	2	0.6	n	48.8	0.3	0.2	49.9
	3	0.6	n	48.0	0.3	0.2	49.1
	REF LWR	0.7	n	44.6	0.3	0.2	45.8

(1) PBR-1 - prebreeder core concept 1 etc. REF LWR - Reference light water reactor

(2) Includes mining, milling and conversion to UF₆

(3) Includes mining and milling

(4) Includes storage and waste management

(5) n signifies negligible compared with considerations from other fuel cycle components

(6) World population 70 year total body dose commitment.

In the following paragraphs a brief explanation is provided for the phases of each fuel cycle operation which make the principal contribution to each of the categories in the table. Volume 4 of Reference E-1 presents a much more detailed explanation for each of the considerations. No attempt is made to explain the relatively small differences between the entries for the various reactor cores.

a. Land - The land permanently committed due to operation of a facility represents land which, in the event that the operation of the facility were discontinued and the facility were dismantled, could not be returned to normal use, primarily because of the storage requirements for radioactive wastes. Dismantling of a facility could follow replacement by a new facility, or could occur if new forms of power generation eliminated the need for the facility. The permanently committed acreage values in Table E-2 represent all of the land that would remain permanently committed at the end of a normal facility life (typically taken to be about 30 years). This permanent commitment could be made entirely at the start of the construction or operation of the facility, or could be increased to its final value gradually over the life of the facility. The land temporarily committed could be returned to normal use at the end of facility life. This temporary land commitment would be required as long as the reactors continue to operate, and is expressed in acre years per reactor year. The land temporarily committed for uranium mining includes land which could be in the process of being mined, land used for disposal of tailings remaining after U_3O_8 is extracted from the ore, and an exclusion area. The land temporarily committed for reprocessing would include the reprocessing plant and the exclusion area.

b. Water - The water volumes listed in Table E-2 represent water discharged to air, ground or bodies of water, following use or accumulation in fuel

cycle operations. Water discharged from the production of uranium would be primarily in the form of drainage pumped out of open pit uranium mines. In Reference E-1 it was assumed that all of the uranium would be produced by open pit mining, and that the ore contained 0.2 percent by weight of U_3O_8 . The water discharged in the enrichment process would be primarily cooling water associated with the production of the electricity used in the enrichment process. In Reference E-1 it was assumed that all enrichment was by the gaseous diffusion process.

c. Ore Mined and Overburden Moved - The amount of each type of ore mined would be proportional to the amount of U_3O_8 or ThO_2 required and to the grade of the ore, i.e., the fraction of U_3O_8 and ThO_2 in the respective ores. In Reference E-1 the uranium ore was assumed to contain 0.2 percent U_3O_8 and the thorium ore was assumed to contain 0.5 percent ThO_2 . Since only 15 to 47 short tons of ThO_2 would be required annually, while from 175 to 228 short tons of U_3O_8 would be required annually, the above assumptions on ore grade imply that 90 to 97 percent of the ore mined would be uranium ore. In Reference E-1 it was assumed that both uranium and thorium ores would be obtained from open pit mines. The amount of overburden moved would be proportional both to the amount of ore mined and to the ratio of tons of overburden moved to tons of ore obtained. In Reference E-1 it was assumed that this ratio was about 30 to 1 for uranium mining and about 3 to 1 for thorium mining. Therefore, more than 99 percent of the overburden moved would be associated with uranium mining.

d. Petroleum - Petroleum for uranium production would be used primarily to power earth moving equipment. It was noted above that Reference E-1 assumed all uranium to be produced from open pit mining; this would require the removal of large volumes of overburden.

e. Coal - Coal for uranium enrichment would be used for production of electricity used in the enrichment process. As noted above, Reference E-1

assumed that all enrichment was by the gaseous diffusion process. In addition, it was assumed that 70 percent of the electricity used in any fuel cycle facility was produced by coal fired plants.

f. Chemicals - The chemical releases from uranium enrichment would be associated primarily with the production of electricity used in the enrichment process. These chemical releases from production of electricity would be primarily combustion products from burning coal, and chemicals used to treat the cooling water in the electric generating plants. Item e above states relevant assumptions made in Reference E-1.

g. Radiological Effects - The measure chosen to represent the radiological effects of the fuel cycle operation is the approximate 70-year total body dose commitment to the world population.

The principal approximation involved in the present dose commitment calculations is that only five nuclides (radium-226, carbon-14, tritium, radon-222, and krypton-85) were used to compute the dose commitment. This approximation was used because the data presented in Reference E-1 (Table IX-49A) show that, of the approximately 50 nuclides considered, over 90 percent of the total body dose commitment would occur from these five nuclides.

In Reference E-1, the 70-year total body dose commitment to the world population was used to assess the radiological effects of releasing radioactive materials to the environment; this 70-year dose commitment is defined as the total dose accrued by the entire world population over a period of 70 years as the result of radionuclides being released to the environment during one year of normal fuel cycle operation. This one year of fuel cycle operation was assumed to occur during the first year of the 70 year period. The dose to the world population was calculated as the product of "dose to an individual" and "number

of individuals exposed". The population exposed to radium-226 and radon-222 was assumed to be 2×10^8 persons, which is 80 percent of the projected United States population in 1985. The world population exposed to carbon-14, tritium, and krypton-85 was assumed constant at the expected 1985 level of 4.7×10^9 persons. This 70 year dose commitment includes the effects of exposure to the radioactivity as it is being released, exposure to radioactivity which may remain in the environment after the release has stopped, and exposure to radioactivity which may be retained in the body after external sources have disappeared.

Dose commitment factors for the principal radionuclides released from fuel cycle operations were derived from Reference E-1 and are listed in Table E-3. These values were used to estimate the whole body dose commitment for the world population due to operation of these cores.

About 75 percent of the dose commitment from the fuel cycle operation would be due to radium-226 released during uranium milling and radon-222 released during uranium mining and milling. The remainder of the dose commitment would be due primarily to carbon-14, tritium, and krypton-85 released during spent fuel reprocessing. (Small quantities of these last three nuclides would also be released during reactor operation, but none of the environmental effects of reactor operation have been included in the present work.)

Table E-2 shows that the approximate dose commitment would be nearly the same for all four reactor core types. It should be noted that operation of prebreeders would eventually permit breeders to replace prebreeders or conventional light water cores and eliminate the associated uranium mining and milling. This would substantially reduce the dose commitment from the fuel cycle.

III. MODEL FOR CALCULATION OF ENVIRONMENTAL CONSIDERATIONS

A. Definition of Environmental Effects Factors

Reference E-1 is the source of the base data for all environmental effects described in this report. In Reference E-1, each type of fuel cycle facility has a specified throughput (for example, 1000 MT of heavy metal per year through the model reprocessing plant), and tables are provided in Reference E-1 listing the annual environmental effect (for example, curies of krypton-85 released per year) which would be expected for each facility. The tables of effects in Reference E-1 contain many more categories than the nine categories used in the present work because some of the present categories were defined by combining certain categories of Reference E-1 and omitting some categories of Reference E-1 which are not significant for the present evaluation. For each facility and for each of the nine categories of the present work, the total annual effect was divided by the facility annual throughput to define the effect factor. These effect factors are presented in Table E-3. The factors in Part 1 of this table all have dimensions of effect/mass flow. To illustrate the nature of these factors consider the first entry under uranium mining. The factor has a numerical value of 0.0291; the effect is overburden moved in units of 10^6 MT and the unit of mass flow is 10^3 MT uranium ore, so the factor is: 0.0291×10^6 MT overburden moved/ 10^3 MT uranium ore. The annual effect in any category which would be produced by any facility in support of one of the present reactor cores is then defined to be the product of 1) the effect factor for the given facility and given category and 2) the annual throughput for that facility due to the one reactor core.

TABLE E-3 ENVIRONMENTAL EFFECT FACTORS* FOR HYPOTHETICAL
MODEL FUEL CYCLE FACILITIES

PART 1 EFFECTS NOT INCLUDING DOSE COMMITMENTS

Fuel Cycle Operation Unit Mass Flow	Uranium Mining 10 ³ MT Ore	Thorium Mining 10 ³ MT Ore	Uranium Milling and UF ₆ Production 1 MT U ₃ O ₈	Thorium Milling 1 MT ThO ₂	Enrichment 1 MTSWU	Fuel Fabrication (no U-233) 1 MTHM	Fuel Fabrication (with Uranium-233) 1 MTHM	Reprocessing and Waste Management 1 MTHM
<u>EFFECT AND UNIT VALUE</u>								
Overburden moved (10 ⁶ MT)	0.0291	0.0031	-	-	-	-	-	-
Land temporarily committed (acre years per reactor year)	5.40	6.27	0.109	0.0209	0.455	0.486	1.37	1.97
Land permanently committed (acres per facility lifetime)	0.251	0.0063	0.259	0.105	0.0006	0	0	0.0430
Water total (10 ⁶ gallons)	1.35	0.136	0.231	0.247	0.850	0.0541	0.0318	0.220
Petroleum and other liquid fuels (10 ³ gallons)	0.209	0.141	3.16	0.980	0.0618	0	0	7.50
Coal (MT)	0.608	-	7.02	1.35	564	16.2	11.0	11.9
Chemicals to air and water (MT)	0.0709	0.0431	0.513	0.0470	6.20	0.292	0.119	0.776
Thermal to air and water (10 ⁹ BTU)	0.016	-	0.677	0.193	23.0	0.703	0.456	2.19
Electric energy (Megawatt years)	0.0003	-	0.0035	0.0006	0.279	0.0084	0.0057	0.0062
Radium-226 (curies)	-	-	8.23 x 10 ⁻⁵	-	-	-	-	-
Carbon-14 (curies)	-	-	-	-	-	-	-	0.00324
Tritium (curies)	-	-	-	-	-	-	-	351
Radon-222 (curies)**	2.29	-	0.835	-	-	-	-	-
Krypton-85 (curies)	-	-	-	-	-	-	-	75.7

*Effect factors are defined as follows:

$$\text{Factor} = \frac{\text{Total annual effect from hypothetical model facility}}{\text{Total annual mass flow through hypothetical model facility}}$$

Factors are independent of type of core supported by facility

**The release factors do not include the radon-222 daughters

TABLE E-3 (Cont)

PART 2 FACTORS FOR WORLD POPULATION TOTAL BODY DOSE COMMITMENT

<u>Nuclide</u>	<u>Factor in units of man rem per curie released</u>
Radium-226	3.28×10^4
Carbon-14	40.4
Tritium	2.24×10^{-3}
Radon-222	0.183*
Krypton-85	5.46×10^{-4}

*This dose factor includes the effect of the radon-222 daughters which are born following the release of radon-222

The Reference E-1 treatment for land permanently committed deserves special mention. At the end of the operational lifetime of the mines, mills and reprocessing plants there would be a certain amount of land which could not be returned to its original use due to the requirement for proper disposition of radioactive wastes. All of the land permanently committed for these facilities is defined to be committed in the first year of facility operation.

The estimate of the radiological effect required an additional factor to relate curies of a particular nuclide released to world population total body dose commitment for that nuclide. These factors for the five nuclides used in the present calculations were obtained from Table IX-49A of Reference E-1, and are also presented in Table E-3.

All of the facilities discussed in Reference E-1 were hypothetical model facilities and the data were based on a concept of a future industry. The assumptions made in Reference E-1 concerning the details of operation of the model facilities were based where possible on the operation of existing facilities and on the assumption that the facilities would have to meet environmental standards current at the time of publication of Reference E-1. However, it is expected that the design and construction of all future commercial facilities would have to meet requirements for the mitigation of adverse environmental effects existing then, and that the operation of all commercial facilities would be subject to monitoring by the Nuclear Regulatory Commission or a similar agency to ensure continuing compliance with the standards for both radiological and nonradiological releases current at that time. In view of this, the environmental effects data presented in this appendix, as well as the data

presented in Reference E-1, serve only as an approximate indication of the environmental effects which might characterize a future light water reactor industry.

There is no reprocessing of spent commercial reactor fuels in the United States today, but substantial quantities of uranium oxide fuel and thorium oxide fuel (900 tons of thorium were reprocessed to recover 1500 kg of uranium-233) have been successfully reprocessed in the past, and the technology for reprocessing is available. Therefore it is judged that despite the absence of reprocessing on a commercial scale in the United States, it is still possible to make reasonable estimates of the environmental considerations for reprocessing operations which might be conducted in the near future. Recently, relative to the uranium fuel cycle, the ERDA Fuel Cycle Task Force (Reference E-2) concluded that "the remaining questions in closing the fuel cycle are not ones of technical feasibility. Rather, the problems are political and societal questions leading to licensing and regulatory uncertainties which...have delayed or prevented needed actions on the part of industry and the Government in resolving them."

B. Uranium Calculations

This section presents the equations used to estimate the amount of uranium feed which would be required to provide the enriched uranium for fresh fuel. These equations account for uranium which would be recovered from spent seed fuel and returned to the enrichment plant (the case for prebreeder core Concepts 1 and 2) and for uranium which would be recovered from spent seed fuel and mixed with highly enriched uranium to produce fresh seed fuel (the case for prebreeder core Concept 3). These equations also account for uranium losses during reprocessing and during fabrication.

The operational nodes and flowpaths are shown in Figure E-1. Steady state material balances at the process nodes provide the necessary equations, as follows:

1. Uranium balance at enrichment

$$F + R = T + P$$

where:

F = feed mass flow of natural uranium

R = recycle mass flow of uranium (all isotopes) recovered from spent fuel for the case (e.g., prebreeder core Concepts 1 and 2) where this uranium is sent through enrichment

T = tails mass flow

P = product mass flow of enriched uranium

All mass flows are in units of mass per unit time, e.g., metric tons uranium per year.

2. Uranium-235 balance at enrichment

$$XF \cdot F + XR \cdot R = XT \cdot T + XP \cdot P$$

where:

XF = enrichment of natural uranium

XR = enrichment of recycle uranium

XT = enrichment of tails uranium

XP = enrichment of product uranium

All enrichments are expressed as the ratio of mass of uranium-235 to total mass of uranium.

3. Uranium balance at fabrication

$$P = \frac{UI - R}{1 - LF}$$

where:

UI = total uranium mass flow (all isotopes) in fresh fuel installed in the reactor. R is defined above and is used in this equation only for the case (e.g., prebreeder core Concept 3) where uranium recovered from spent seed fuel is reused in seed fuel without being sent through enrichment.

LF = fraction of fissile material lost during fabrication. Assumed to be the same for all uranium-isotopes independent of the source.

4. Uranium fissile balance at fabrication

a. Uranium-235

$$XP \cdot P = \frac{U5I}{1 - LF} - U5R$$

b. Uranium-233

$$U3I = U3R \cdot (1 - LF)$$

where:

U5R or U3R = uranium-235 or uranium-233 mass flow (in units of mass per unit time) from reprocessing for the case (e.g., prebreeder core Concept 3) where uranium recovered from spent seed fuel is reused in seed fuel without being sent through enrichment.

U5I or U3I = uranium-235 or uranium-233 mass flow (in units of mass per unit time) in fresh fuel installed in the reactor.

5. Uranium fissile balance at reprocessing

a. Uranium-235 when recycle uranium is sent to enrichment

$$X_R \cdot R = U_{50} \cdot (1 - LR)$$

b. Uranium-233

$$U_{3R} = U_{30} \cdot (1 - LR)$$

c. Uranium-235 when recycle uranium is not sent to enrichment

$$U_{5R} = U_{50} \cdot (1 - LR)$$

where:

U_{50} or U_{30} = uranium-235 or uranium-233 mass flow (in units of mass per unit time) from discharge of spent fuel

LR = fraction of fissile material lost during reprocessing. Assumed to be the same for uranium-235 and uranium-233.

Of the seventeen variables in these equations, six are specified by the nuclear design calculations: U_I , U_{5I} , X_R , U_{50} , U_{3I} , and U_{30} ; four are specified by the process parameters assumed: X_F , X_T , LF , and LR ; the remaining seven are determined by the equations.

C. Separative Work Calculations

The separative work required to obtain the desired product enrichment is calculated in terms of a value function (which is not directly related to price or cost) defined as follows:

$$V(X) = (2X - 1) \ln (X/(1 - X))$$

where X represents enrichment expressed as a fraction. $V(X)$ has dimensions of SWU (Separative Work Units).

The total separative work done is given by the following equation.

$$S = P \cdot V(XP) + T \cdot V(XT) - F \cdot V(XF) - R \cdot V(XR)$$

R appears in this equation only for the case where recycle uranium is sent to enrichment. When the mass flow rates P, T, etc., are expressed in units of metric tons per year, S has units of metric tons SWU per year.*

IV. REFERENCES

- E-1. "Final Environmental Statement, Light Water Breeder Reactor Program," ERDA 1541, June 1976.
- E-2. "Nuclear Fuel Cycle," ERDA-33, March 1975.

*The separative work equations presented here are approximate in that the function $V(X)$ assumes the presence of only two isotopes (e.g., uranium-235 and uranium-238). For product enrichments below about 90 percent these equations are adequate for uranium containing small amounts of the isotopes 234 and 236 which would be typical of natural uranium or of recycle uranium from prebreeder core Concepts 1 and 2.

GLOSSARY

APSRA	<u>A</u> xial <u>P</u> ower <u>S</u> haping <u>R</u> od <u>A</u> ssembly
Backfit	A prebreeder core concept which could replace a conventional core in a conventional reactor plant with no major changes required to the plant.
Blanket Rods	Prebreeder fuel rods which are primarily thorium and which at the start of irradiation contain between zero and 2 w/o fissile atoms in heavy metal.
BOC	Beginning of fuel irradiation cycle.
B P RA	<u>B</u> urnable <u>P</u> oison <u>R</u> od <u>A</u> ssembly
CHF	<u>C</u> ritical <u>H</u> eat <u>F</u> lux
CRA	<u>C</u> ontrol <u>R</u> od <u>A</u> ssembly
CW	<u>C</u> old <u>W</u> orked stress relief annealed fuel rod cladding.
DNBR	<u>D</u> eparture from <u>N</u> ucleate <u>B</u> oiling <u>R</u> atio
EFPH	<u>E</u> ffective <u>F</u> ull <u>P</u> ower <u>H</u> ours. One EFPH is operation for X hours at (100/X) percent of full power.
EOC	End of fuel irradiation cycle.
LOCA	<u>L</u> oss- <u>o</u> f- <u>C</u> oolant- <u>A</u> ccident
RIT	<u>R</u> eactivity <u>I</u> nsertion <u>T</u> ransient
RPD	<u>R</u> elative <u>P</u> ower <u>D</u> ensity
RWA	<u>R</u> od <u>W</u> ithdrawal <u>A</u> ccident
RXA	Recrystallized Annealed fuel rod cladding.
SCC	<u>S</u> tress <u>C</u> orrosion <u>C</u> racking
Seed Rods	Prebreeder fuel rods which at the start of irradiation contain from 5 percent to 10 percent fissile atoms in heavy metal.
Thermal Capability	Power at which the core can operate without exceeding limits on fuel rod temperature, stress or heat flux.
Thorium	Thorium dioxide - ThO_2 .
Urania	Uranium dioxide - UO_2 .

GLOSSARY (Cont)

U^E

Uranium which has been moderately enriched (enrichment between 2 percent and 10 percent) in the isotope U-235.

U^{HE}

Uranium which has been highly enriched (enrichment above 90 percent) in the isotope U-235.