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VOLUME III

LIGHT-WATER BREEDER REACTORS

January 1980

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM

U.S. DEPARTMENT OF ENERGY
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FOREWORD

The Department of Energy (DOE) Nonproliferation Alternative Systems Assessment Program (NASAP) is a planned program of studies of nuclear power systems, with particular emphasis on identifying and then evaluating alternative nuclear reactor/fuel-cycle systems that have acceptable proliferation-resistance characteristics and that offer practical deployment possibilities domestically and internationally. The NASAP was initiated in 1977, in response to President Carter's April 1977 Nuclear Power Policy Statement.

The NASAP objectives are to (1) identify nuclear systems with high proliferation resistance and commercial potential, (2) identify institutional arrangements to increase proliferation resistance, (3) develop strategies to implement the most promising alternatives, and (4) provide technical support for U.S. participation in the International Nuclear Fuel Cycle Evaluation (INFCE) Program.

The NASAP is not an assessment of all future energy-producing alternatives. Rather, it is an attempt to examine comprehensively existing and potentially available nuclear power systems, thus providing a broader basis for selecting among alternative systems. The assessment and evaluation of the most promising reactor/fuel-cycle systems will consider the following factors: (1) proliferation resistance, (2) resource utilization, (3) economics, (4) technical status and development needs, (5) commercial feasibility and deployment, and (6) environmental impacts, safety, and licensing.

The DOE is coordinating the NASAP activities with the U.S. Nuclear Regulatory Commission (NRC) to ensure that their views are adequately considered at an early stage of the planning. In particular, the NRC is being asked to review and identify licensing issues on systems under serious consideration for future research, development and demonstration. The Preliminary Safety and Environmental Information Document (PSEID) is the vehicle by which the NASAP will provide information to the NRC for its independent assessment. The PSEID contains the safety and environmental assessments of the principal systems. Special safeguards measures will be considered for fuel cycles that use uranium enriched in U-235 to 20% or more, uranium containing U-233 in concentrations of 12% or more, or plutonium. These measures will include the addition of radioactivity to the fuel materials (i.e., spiking), the use of radioactive sleeves in the fresh fuel shipping casks, and other measures. The basis for the safeguards review by the NRC is contained in Appendix A.

The information contained in this PSEID is an overlay of the present safety, environmental, and licensing efforts currently being prepared as part of the NASAP. It is based on new material generated within the NASAP and other reference material to the extent that it exists. The intent of this assessment is to discern and highlight on a consistent basis any safety or environmental issues of the alternative systems that are different from a reference LWR once-through case and may affect their licensing. When issues exist, this document briefly describes the research, development, and demonstration (RD&D) requirements that would help resolve them with the normal engineering development of a reactor/fuel cycle system.

The preparation of this document takes into consideration NRC responses to the DOE preliminary safety and environmental submittal of August 1978. Responses to these initial comments have been, to the extent possible, incorporated into the text. Comments by the NRC on this PSEID were received in mid-August 1979 and, as a result of these comments, some changes were made in this document. Additional comments were incorporated as Appendix B. Comments that are beyond the scope and resources of the NASAP may be addressed in research, development, and demonstration programs on systems selected for additional study. The intent of this document (and the referenced material) is to provide sufficient information on each system so that the NRC can independently ascertain whether the concept is fundamentally licensable.

This PSEID was prepared for the DOE through the cooperative efforts of the Argonne National Laboratory, the Oak Ridge National Laboratory, and NUS Corporation.

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Chapter 1

GENERAL DESCRIPTION

1.1 GENERAL OVERVIEW OF LIGHT-WATER PREBREEDER AND BREEDER SYSTEMS

The water-cooled breeder reactor would be fueled with thorium and uranium dioxide fuel and could use either light- or heavy-water coolant. The concepts considered in this preliminary safety and environmental information document (PSEID) discuss only light-water breeder reactors (LWBRs) because there is more technical information available on the light-water systems than for the heavy-water systems and the light-water breeder reactor concepts are further developed. Thorium is plentiful and has no present major energy-related use. The LWBRs could be operated in existing pressurized-water reactor (PWR) power plants but because the power density would be lower than that of a conventional PWR, the principal application would be in new plants specifically designed to accommodate the breeder concept. The operation of the LWBR core in the Shippingport Atomic Power Station in Shippingport, Pennsylvania, is expected to confirm that breeding can be achieved in a PWR using the thorium/uranium-233 fuel system in a seed-blanket core configuration. This reactor is designed to breed more new uranium-233 from thorium than it consumes to produce electrical energy. This means that, once on the breeding cycle, an LWBR replacement core could be produced without requiring further mining or enrichment of uranium ore. Development work to date indicates that LWBR cores could be used in any PWR; furthermore, although no specific work has been performed, no inherent limitations are known that would prevent application of the LWBR concept to boiling-water reactors (BWRs).

The LWBR program is developing the basic technology required to use thorium as an energy source, basing this development on the well-established technology of LWRs, which is also the basis of the present commercial nuclear industry. At present, thorium cannot be effectively exploited to anywhere near its full potential for power production. To do so requires the capabilities of a breeder reactor.

To operate a breeder reactor on the uranium-233 thorium fuel cycle, it is necessary to have enough uranium-233 for the initial fuel loading. Because uranium-233 is not available in nature, the initial loading must be produced from thorium by irradiating it in a reactor. The LWBR core for the Shippingport reactor was built with uranium-233 produced in the U.S. Department of Energy (DOE) production reactors. This approach was chosen to test directly whether it was possible to breed in an LWR with the thorium/uranium-233 system. Uranium-233 fuel for commercial LWBR breeders, however, would have to be produced in large commercial power-producing reactors fueled initially with thorium and the naturally occurring uranium-235 or with plutonium from other reactors. Such reactors producing uranium-233 have been designated "prebreeders."

Since the LWBR prebreeder and breeder concepts utilize light water as the coolant, it would be possible to backfit either prebreeder or breeder cores based on the LWBR concept into existing PWR plants as well as to operate such cores in a new PWR design optimized for use as a prebreeder or in a new plant intended for eventual use as a breeder reactor. While no studies have been made, there is no known inherent reason why LWBR prebreeders could not also be backfitted into existing BWRs. Prebreeder cores would generate electrical power by the fissioning of uranium-235 (or plutonium) while producing uranium-233. The rate of buildup of uranium-233 in prebreeder cores would depend on the fuel strategy chosen, but calculations indicate that, under optimum conditions in a new plant designed for use as a prebreeder, enough uranium-233 for a commercial-size breeder core might be produced in about 10 years of operation with prebreeder cores.

1.2 SUMMARY OF THE CONCEPTS DISCUSSED IN THIS REPORT

Chapters 2, 3, and 4 discuss three differing design concepts for LWBR prebreeder/breeder systems. These concepts consist of prebreeders and breeders operating in the range of 700-1,300 MWe. The prebreeder would be operated until sufficient uranium-233 has been produced to supply an accompanying breeder.

The first concept (Chapter 2), the prebreeder and breeder with LWBR type I modules, is based on an array of fuel modules that are geometrically identical with the type I modules of the Shippingport LWBR core (Ref. 1), which is now in operation at the Shippingport Atomic Power Station in Shippingport, Pennsylvania. The NRC has completed its review of the safety analysis report for this breeder reactor and has issued its safety evaluation report which concluded that the LWBR core may be operated in the Shippingport Atomic Power Station without undue risk to the health and safety of the public (Ref. 2).

The second concept (Chapter 3), in which a light-water backfit prebreeder would supply an advanced breeder, is based on a prebreeder that could be directly backfitted into an existing commercial reactor; the breeder reactor would be a new plant and would be an advanced version of the Shippingport seed-blanket LWBR. The breeder concept would have less neutron-absorbing structure, a more uniform power distribution, and a higher average power density than does the Shippingport-type LWBR core. In addition, the fuel loading and reactivity control would be modified.

The third concept (Chapter 4), also is based on a prebreeder which could be backfitted into an existing commercial reactor; the breeder would be an advanced seed-blanket concept. In this concept the prebreeder would use a thorium-based fuel containing highly enriched uranium instead of the low-enrichment uranium-dioxide duplex-pellet fuel used in the prebreeders of concepts 1 and 2.

REFERENCES FOR CHAPTER 1

1. Babcock and Wilcox, Reference Safety Analysis Report, BSAR-205, NRC Docket STN 50-561.

2. U.S. Nuclear Regulatory Commission, Safety Evaluation Report, Light Water Breeder Reactor, Shippingport Atomic Power Station, NUREG-0083, Project 561, July 1976.

Chapter 2

PREBREEDER AND BREEDER REACTORS BASED ON LIGHT-WATER-BREEDER (LWBR) TYPE I MODULES

2.1 GENERAL DESCRIPTION OF THE CONCEPT

The prebreeder and breeder concepts discussed here are based on an array of fuel modules that are geometrically identical with the type I modules of the Shippingport LWBR core, except that the seed lattice of the breeder module has been altered to reduce the Zircaloy content.

The Shippingport LWBR core and the principles of its operation are briefly described below. This description also applies to the present prebreeder and breeder concepts with the exceptions noted. Tables 2-1 through 2-4 list the parameters for the breeder and prebreeder concepts.

The Shippingport LWBR core has 12 hexagonal seed-blanket fuel modules surrounded by 15 reflector-blanket modules to form a nearly circular array. The three central modules are of type I. They measure 17 inches across the flats of their hexagonal shape and have an active (i.e., containing uranium-233) fuel length of 84 inches; in addition they have axial reflector blankets of thoria measuring about 10 inches at top and bottom. Each of the 12 seed-blanket modules consists of a central movable seed assembly surrounded by a stationary blanket assembly.

The module arrangement of the present prebreeder and breeder concepts is shown in Figure 2-1, and the arrangement of fuel rods in a type I module is shown in Figure 2-2. The cores would consist of 109 seed-blanket modules, surrounded by 54 reflector-blanket modules to form a nearly circular array. Each of the 109 seed-blanket modules would be geometrically identical with the LWBR type I Shippingport module. The prebreeder and breeder cores would fit into a pressure vessel with a 238-inch inside diameter.

In the Shippingport LWBR seed-blanket modules, the fuel in both the movable seed assemblies and the fixed blanket assemblies consists of natural thorium oxide or a binary mixture of uranium-233 oxide and thorium oxide. The average enrichment (ratio of uranium-233 to total heavy metal) is higher in the movable seed assemblies than in the fixed blanket assemblies. In the present prebreeder concept, the fuel would consist of natural thorium oxide (containing no uranium) and of moderately enriched uranium oxide (performing the same function as the binary mixture of uranium-233 oxide and thorium oxide). The geometric arrangement of the fuel is shown in Figures 2-3 and 2-4.

2.1.1 FUEL DESCRIPTION

The fuel material in the Shippingport LWBR is thorium dioxide with up to 5.9% of uranium-233 dioxide. The fuel-rod support grids are made from AM-350 steel. These are the only materials used in the Shippingport LWBR that are not used in current commercial light-water reactor (LWR) cores. Thorium dioxide and highly enriched uranium dioxide were used successfully in the first Indian Point core, and many in-reactor and out-of-reactor tests were made on the Shippingport LWBR fuel and AM-350 grids before the Shippingport LWBR core went into operation. For the present prebreeder and breeder concepts it is assumed that Zircaloy grids will be used. Although

Zircaloy grids are being used in commercial PWR cores, their use in a close-packed hexagonal array characteristic of a breeder reactor has yet to be proven.

The fuel for the Shippingport LWBR core is in the form of cylindrical ceramic pellets inside Zircaloy-4 rods. The pellets are either a high-density solid solution of uranium-233 oxide in thorium oxide or pure high-density thorium oxide. The fuel for the present breeder concept would be similar to the Shippingport LWBR fuel. The present prebreeder concept also uses Zircaloy-4 rods, but the pellet configuration would be different. All of the thorium would be present in the form of cylindrical pellets of thorium oxide. However, the moderately enriched uranium would be in the form of annular pellets containing a high-density ternary solid solution of uranium oxide, zirconium oxide, and calcium oxide. The core of the annulus would contain a pellet of thorium oxide. Figure 2-5 shows this arrangement, which is called a duplex fuel pellet. The purpose of this pellet design is to permit the separation during fuel reprocessing of uranium-233 bred in thorium from the uranium (primarily 235, 236, and 238) resulting from the initial fissile charge.

2.1.2 REACTIVITY CONTROL

Reactivity control in the Shippingport LWBR is achieved by lifting or lowering the movable seed assemblies (see Figure 2-3). There are no control rods, soluble poison, or fixed burnable poisons for reactivity control during normal operation or shutdown. Changing the axial position of the movable seed assembly relative to the fixed blanket assembly changes the relative rates of neutron absorptions in fissile (uranium-233) and fertile (thorium) material and thereby changes the reactivity. This method of reactivity control eliminates neutron losses to control poison; in the Shippingport LWBR and in the present breeder concept it is necessary if breeding is to be achieved. In the present prebreeder concept where the fissile fuel would be uranium-235 and the fertile fuels would be thorium and uranium-238, this method of reactivity control would lead to a relatively high annual production rate of uranium-233.

2.1.3 ACCOMMODATION IN EXISTING PHYSICAL PLANTS

The Shippingport LWBR core was designed to be backfitted into the existing pressure vessel at the Shippingport Atomic Power Station. A new pressure-vessel head and new control-drive mechanisms were required, but the remainder of the plant needed no modification to accommodate the core. Some features, such as flywheel generators were added to provide ample margin for a loss-of-coolant-flow accident. If a plant were built to accommodate the present concepts, only the reactor vessel, closure head and control-drive mechanisms would be significantly different from those in a conventional PWR plant with the same thermal output as the conceptual prebreeder or breeder. The remaining components would be very similar to those in the reference plant, although the containment, pumps, and pressurizer might have to be larger than those of the reference plant because of the larger core.

2.1.4 FUEL MANAGEMENT AND FUELING ALTERNATIVES

For the present prebreeder concept, the mass flows are based on unloading and reloading the entire core every 3 years. A fuel-management plan similar to current commercial practice would also be possible. For the present breeder concept, a more conventional fuel-management scheme would be used: one-third of the core is assumed to be replaced each year.

2.1.4.1 Prebreeder Concept

A prebreeder fueling alternative that is possible in principle but has not been studied is to refuel one-third of the core annually. This plan might improve the power distribution and permit operation at higher power. Other refueling strategies, such as semiannual refueling, could also be used. However, it is expected that there would be very little effect on resource utilization because with movable-fuel reactivity control there would be no loss of neutrons to control poisons; hence changing the refueling strategy would not change the number of neutrons available for capture in fertile atoms.

2.1.4.2 Breeder Concept

For the breeder concept the possible fueling alternatives would be as follows: reduction in fissile inventory, operation at a higher power density, and operation to a higher fuel burnup. Since any or all of these alternatives would reduce the conversion ratio, breeding may no longer be achieved, in which case the core would operate as a high-gain converter. In comparison to the breeder, a high-gain converter would in some cases reduce short-term (30 years) mining requirements but would in all cases increase long-term (100 years or more) requirements. A reduction in short-term requirements would result from a reduction in fissile loading.

2.1.5 FUEL CYCLES

The prebreeder and breeder concepts would be based on an array of hexagonal fuel modules each of which would be geometrically identical with the Type I modules of the Shippingport LWBR core, except that the seed lattice of the breeder module would be altered to reduce the Zircaloy content. The core modules would be surrounded by reflecting blanket modules. Reactivity control would be achieved by lifting or lowering movable fuel assemblies. The prebreeder and breeder phases would have the same physical arrangement except for the movable fuel assembly.

2.1.5.1 Prebreeder Concept Based on LWBR Type I Modules

The prebreeder would use less than 20% enriched uranium-235 fuel in the form of UO_2-ZrO_2-CaO (ternary) duplex pellets alternating with thorium dioxide pellets in the fuel rods. The duplex pellet would consist of a ternary annulus around a cylindrical thorium dioxide center. The spent fuel would be reprocessed in two stages to recover uranium, plutonium, and thorium. The first stage would recover the uranium-235 remaining in the UO_2-ZrO_2-CaO annulus. The second stage would recover the bred uranium-233 from the thoria. The uranium-235 would be recycled. The plutonium and the bred uranium-233 would be sent to secure storage and the thorium would be sent to 10-year interim storage. A diagram of a typical cycle is shown in Figure 2-6.

2.1.5.2 Breeder Concept Based on LWBR Type I Modules

The breeder would be fueled with a binary solid solution of highly enriched (>84% uranium-233) urania and thorium dioxide in the form of pellets. The spent fuel would be reprocessed to recover the uranium-233 which would be recycled to remote fuel fabrication. The thorium would be separated and also recycled to remote fuel fabrication. A flow diagram for a typical cycle is shown in Figure 2-7.

2.1.5.3 Quantitative Fuel Inventories

Table 2-5 summarizes the overall fuel-management information, including the separative-work requirements typical of this concept. Table 2-6 shows the calculated isotopic makeup of cycle 6 of a typical prebreeder core, Table 2-7 shows the calculated isotopic makeup of the equilibrium cycle for a possible breeder, and Tables 2-8 and 2-9 show the calculated overall isotopic makeup of the typical system over its 30-year history. Figures 2-6 and 2-7 indicate the isotopic mass flows for the prebreeder and breeder concepts, respectively, if scaled to 1,000 MWe. The quantities of isotopes in Tables 2-8 and 2-9 have been multiplied by the factors 1,000/721 and 1,000/711, respectively, to obtain the values shown in Figures 2-6 and 2-7. (The quantities 721 and 711 would be the net electrical powers for the prebreeder and breeder, respectively, as given in Table 2-1.)

2.1.5.4 Fuel Reprocessing and Refabrication

The reprocessing of LWBR fuel would be similar to the reprocessing of LWR fuel in that high-grade fissile material of high toxicity would be generated and handled, although the fission-product concentration in the breeder fuel would be lower than that in the LWR fuel. The uranium-233 recovered from LWBR fuel would contain about 2,500 to 4,500 ppm of uranium-232 whose daughter products emit a more penetrating radiation than do the transuranic isotopes of the LWR fuel cycle and would require more highly automated and shielded fabrication equipment for breeder fuel. Fabrication equipment for prebreeder fuel would differ little, if at all, from that of LWR fuel. Although the penetrating radiation from the uranium-232 accompanying uranium-233 would present difficulties in fabrication, it would also deter diversion.

Table 2-1. Generalized performance specifications: prebreeder and breeder concepts based on LWBR type I modules

Parameter	Prebreeder core	Breeder core[a]
Power plant performance parameters		
Reactor thermal power output, MWt	2,026	2,026
Net electrical power output, MWe	721	711
Plant heat rate, Btu/kW-hr	9,590	9,720
Performance parameters		
Core heat output, MWt	2,026	2,026
Core volume, liters[b]	38,600	38,600
Core loading (first core), kg		
Heavy metal[b]	148,000	164,000
Fissile fuel	4,388	3,528
Conversion ratio (cycle average)[c]	0.72	~1.02
Fissile inventory ratio[c]	0.90	~1.01
Average discharge burnup, MWd/MTHM[d,b]	11,200	10,100
Peak discharge burnup, MWd/MTHM[d,b]	51,100	~46,000
Fuel type	ThO_2 core with $UO_2/ZrO_2/CaO$ annulus	Binary UO_2/ThO_2
Reactor inlet temperature, °F	582	585
Reactor outlet temperature, °F	613	610

[a]Possible technological advances over the Shippingport LWBR design have been factored into the breeder core concept.

[b]Excluding axial and radial reflectors.

[c]Includes fissile plutonium production in prebreeder; breeder values are for equilibrium cycle and do not include plutonium production.

[d]Heavy metal charged.

Table 2-2. Reactor concept data:
prebreeder and breeder concepts based on LWBR type I modules

Parameter	Prebreeder core[a]	Breeder core[a]
Geometric information		
Core height, cm[a]	213	213
Number of core enrichment zones	5	5
Number of assemblies[a]	109	109
Equivalent diameter, cm[a]	478	478
Number of pins per assembly[b]	619/444	325/444
Pin pitch-to-diameter ratio[b]	1.21/1.10	1.234/1.102
Overall assembly length, cm	366	366
Lattice pitch, cm[b]	0.94/1.60	1.313/1.600
Assembly material	Machined Zircaloy-4 grids	Machined Zircaloy-4 grids
Cladding parameters		
Cladding outside diameter, mils[b]	306/571.5	419/571.5
Cladding wall thickness, mils[b,c]	22/27.75	21/28
Cladding material	Zircaloy-4	Zircaloy-4
BOC6/BOEC fissile inventory, kg[c]	4,453	4,610
External fissile inventory, kg[d,e]	4,453/4,453	1,563/3,125
Fissile gain or loss, kg/cycle[d]	471 (loss)	12 (gain)
Specific power, kW/kg fissile[f]	455	574
Specific power, kW/kg HM	13.7	12.3
Power density, kW/liter	52.4	52.4

[a]Excluding axial and radial reflectors.

[b]Pairs of values indicate seed rods/blanket rods.

[c]BOC6 means beginning of the sixth 3-year prebreeder cycle; BOEC means beginning of an equilibrium cycle in the breeder.

[d]Breeder value for equilibrium cycle.

[e]Inventories are for 1 year out-of-core time/2 years out-of-core time.

[f]Breeder value for initial cycle.

Table 2-3. Fuel-assembly volume fractions:
prebreeder and breeder concepts based on LWBR type I modules

Component[a]	Initial-cycle prebreeder core		Equilibrium-cycle breeder core	
	Seed	Blanket	Seed	Blanket
Fuel[b]	0.330	0.546	0.384	0.566
Coolant[c]	0.394	0.285	0.407	0.265
Structure	0.276	0.169	0.209	0.169
Total	1.000	1.000	1.000	1.000

[a]Reactivity control obtained by movable seed assembly.
[b]Includes pellet volume only.
[c]Includes coolant between modules.

Table 2-4. Core-region volume fractions:
prebreeder and breeder concepts based on LWBR type I modules

Component[a]	Initial-cycle prebreeder core[b]	Equilibrium-cycle breeder core[b]
Fuel[c]	0.473	0.500
Coolant[d]	0.315	.317
Structure	0.212	0.183
Total	1.000	1.000

[a]Reactivity control obtained by movable seed assembly.
[b]Excludes radial reflector.
[c]Includes pellet volume only.
[d]Includes coolant between modules.

Table 2-5. Fuel-management information for prebreeder and breeder concepts based on LWBR type I modules

Parameter	Prebreeder core		Breeder core
Average capacity factor, %	75		75
Approximate fraction of core replaced	1.0		1/3
Lag time assumed between fuel discharge and recycle reload	3 yr		1 yr/2 yr
Fissile material loss fractions [(a)]			
Fabrication loss fraction	0.01		
Conversion loss fraction	0.005		
Reprocessing loss fraction	0.01		
Requirements (ST/GWe)	U_3O_8	ThO_2	ThO_2
Initial core	1,556	295	425
Cycle 5 or equilibrium reload requirement	575	8.3	2.6
30-year cumulative requirements[b,c,d]	3,063/3,788	1,288/1,299	(e)
50-year cumulative requirements	3,063/3,788	1,288/1,299	(e)
100-year cumulative requirements	3,063/3,788	1,288/1,299	(e)
Separative-work requirements,[f] 10^3 MTSWU/GWe			
Initial core	1,405		0
Cycle 5 reload	548		0
30-year cumulative requirement[d]	2,910/3,599		0
50-year cumulative requirement	2,910/3,599		0
100-year cumulative requirement	2,910/3,599		0

[a]Fissile losses during recycle are assumed to be 2% for the first 40 years of operation and 1% thereafter, reflecting improved recycle technology in later generation recycle plants.

[b]Fabrication and reprocessing losses are assumed to be 1% each for first 40 years of operation and 0.5% thereafter, reflecting improved recycle technology in later generation recycle plants.

[c]Assumed thorium oxide out-of-core time is 10 years during prebreeder operation and 1 or 2 years for breeder operation. At the end of prebreeder operation nearly all the thorium oxide mined for prebreeder use would be available for recycle into a breeder reactor. Breeder thorium oxide requirements are shown independent of this thorium oxide source.

[d]Prebreeder requirements are given for 1 year out-of-core time/2 years out-of-core time.

[e]The 1,244 short tons of thorium dioxide recovered from prebreeder is sufficient for 200 to 250 years of breeder operation.

[f]Uranium hexafluoride conversion losses are assumed to occur during conversion of uranium dioxide to uranium hexafluoride. No losses are assumed for reconversion from uranium hexafluoride to uranium dioxide.

Table 2-6. Fuel inventory for cycle 6 prebreeder core concept

Isotope	Quantity (kg)	
	BOC[a]	EOC[b]
Thorium-232	126,015.0	124,608.0
Protactinium-233	0.0	74.0
Uranium-232	0.0	2.5
Uranium-233	0.0	843.0
Uranium-234	0.0	67.0
Uranium-235	4,452.8	2,886.7
Uranium-236	403.0	642.0
Uranium-238	17,408.2	17,064.0
Plutonium-239	0.0	155.4
Plutonium-240	0.0	30.7
Plutonium-241	0.0	20.4
Plutonium-242	0.0	2.3

[a]Beginning of the sixth 3-year prebreeder cycle.
[b]End of the sixth 3-year prebreeder cycle.

Table 2-7. Fuel inventory for equilibrium cycle: breeder concept based on LWBR type I modules[a]

Isotope	Quantity (kg)					
	Zone 1		Zone 2		Zone 3	
	BOEC	EOEC	BOEC	EOEC	BOEC	EOEC
Thorium-232	79,961.0	79,751.1	77,595.7	77,399.7	77,399.7	77,210.2
Protactinium-233	0.0	22.8	7.5	22.1	7.5	22.0
Uranium-232	3.1	3.1	3.0	3.0	3.0	3.0
Uranium-233	1,321.5	1,306.6	1,285.9	1,274.4	1,288.9	1,273.7
Uranium-234	597.0	600.6	584.4	585.7	585.7	585.4
Uranium-235	238.0	239.4	233.0	233.5	233.5	233.4
Uranium-236	163.5	164.4	160.0	160.4	160.4	160.3
Fission products	0.0	196.9	191.5	383.1	383.1	574.6

[a]Abbreviations: BOEC, beginning of equilibrium cycle; EOEC, end of equilibrium cycle.

Table 2-8. Reactor charge data

Year	Reactor charge (kg)							
	Th-232	U-232	U-233	U-234	U-235	U-236	U-238	Total
0.0	168,100.0	0.0	0.0	0.0	4,388.3	0.0	17,553.2	190,041.5
3.0	168,100.0	0.0	0.0	0.0	4,388.3	0.0	17,553.2	190,041.5
6.0	168,100.0	0.0	0.0	0.0	4,426.7	239.9	17,466.9	190,233.5
9.0	168,100.0	0.0	0.0	0.0	4,426.7	239.9	17,466.9	190,233.5
12.0	168,100.0	0.0	0.0	0.0	4,452.8	403.0	17,408.2	190,364.0
15.0	168,100.0	0.0	0.0	0.0	4,452.8	403.0	17,408.2	190,364.0
18.0	168,100.0	0.0	0.0	0.0	4,470.5	513.6	17,368.4	190,452.5
19.8	238,496.0	14.3	3,481.3	331.4	47.1	3.5	0.0	242,373.6
20.8	80,957.4	4.9	1,181.7	112.5	16.0	1.2	0.0	82,273.6
21.8	80,957.4	4.9	1,181.7	112.5	16.0	1.2	0.0	82,273.6
22.8	80,931.1	4.8	1,183.8	127.5	24.6	2.0	0.0	82,273.9
23.8	80,905.1	4.7	1,185.9	142.4	33.1	2.9	0.0	82,274.1
24.8	80,879.2	4.7	1,187.9	157.3	41.6	3.7	0.0	82,274.4
25.8	80,879.2	4.7	1,187.9	157.3	41.6	3.7	0.0	82,274.4
26.8	80,879.2	4.7	1,187.9	157.3	41.6	3.7	0.0	82,274.4
27.8	80,851.7	4.6	1,194.3	174.3	45.6	4.2	0.0	82,274.6
28.8	80,824.2	4.6	1,200.6	191.2	49.6	4.7	0.0	82,274.9
29.8	80,796.8	4.5	1,206.9	208.1	53.5	5.2	0.0	82,275.1

Table 2-9. Reactor discharge data

Year	Reactor discharge (kg)													
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Fission products
3.0	166,654.0	75.8	2.6	869.7	68.6	2,823.0	299.9	17,206.1	156.7	31.0	20.6	2.3	188,210.8	1,800
6.0	166,654.0	75.8	2.6	869.7	68.6	2,823.0	299.9	17,206.1	156.7	31.0	20.6	2.3	188,210.8	1,800
9.0	166,654.0	75.8	2.6	869.7	68.6	2,861.4	503.8	17,121.5	155.9	30.8	20.5	2.3	188,367.4	1,800
12.0	166,654.0	75.8	2.6	869.7	68.6	2,861.4	503.8	17,121.5	155.9	30.8	20.5	2.3	188,367.4	1,800
15.0	166,654.0	75.8	2.6	869.7	68.6	2,887.5	642.0	17,064.0	155.4	30.7	20.4	2.3	188,473.5	1,800
18.0	166,654.0	75.8	2.6	869.7	68.6	2,887.5	642.0	17,064.0	155.4	30.7	20.4	2.3	188,473.5	1,800
19.8	167,151.5	77.4	1.5	581.5	35.1	3,447.2	674.6	17,190.8	90.7	17.9	11.9	1.3	189,281.4	1,050
20.8	80,724.5	26.2	4.8	1,167.2	128.5	24.8	2.0	0.0	0.0	0.0	0.0	0.0	82,078.2	196.6
21.8	80,505.1	26.0	4.8	1,172.4	143.6	33.2	2.9	0.0	0.0	0.0	0.0	0.0	81,887.9	387.9
22.8	80,291.2	25.8	4.7	1,172.7	157.9	41.5	3.7	0.0	0.0	0.0	0.0	0.0	81,697.7	579.3
23.8	80,290.5	25.8	4.7	1,172.8	158.3	41.8	3.7	0.0	0.0	0.0	0.0	0.0	81,697.7	579.3
24.8	80,289.8	25.8	4.7	1,172.8	158.7	42.0	3.7	0.0	0.0	0.0	0.0	0.0	81,697.7	579.3
25.8	80,261.9	25.7	4.7	1,179.5	175.8	46.0	4.2	0.0	0.0	0.0	0.0	0.0	81,697.7	579.3
26.8	80,234.8	25.6	4.6	1,186.0	192.5	49.9	4.7	0.0	0.0	0.0	0.0	0.0	81,698.1	579.3
27.8	80,207.7	25.5	4.6	1,192.3	209.2	53.8	5.2	0.0	0.0	0.0	0.0	0.0	81,698.3	579.3
28.8	80,207.0	25.5	4.6	1,192.5	209.6	53.9	5.2	0.0	0.0	0.0	0.0	0.0	81,698.3	579.3
29.8	80,206.3	25.5	4.6	1,192.7	210.1	54.0	5.2	0.0	0.0	0.0	0.0	0.0	81,698.3	579.3
30.8	80,178.6	25.4	4.5	1,199.1	227.3	58.0	5.7	0.0	0.0	0.0	0.0	0.0	81,698.5	579.4

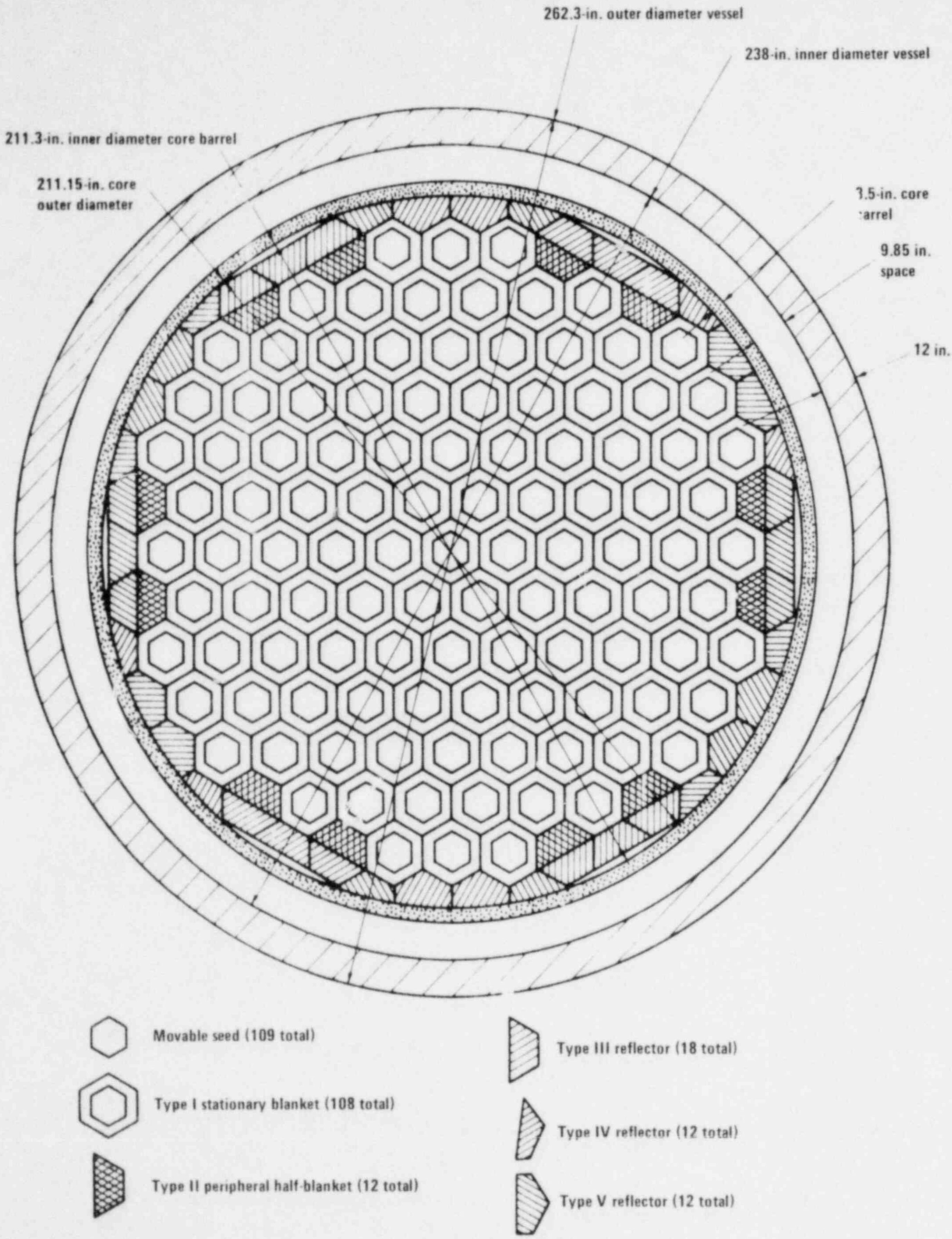


Figure 2-1. Full core cross section.

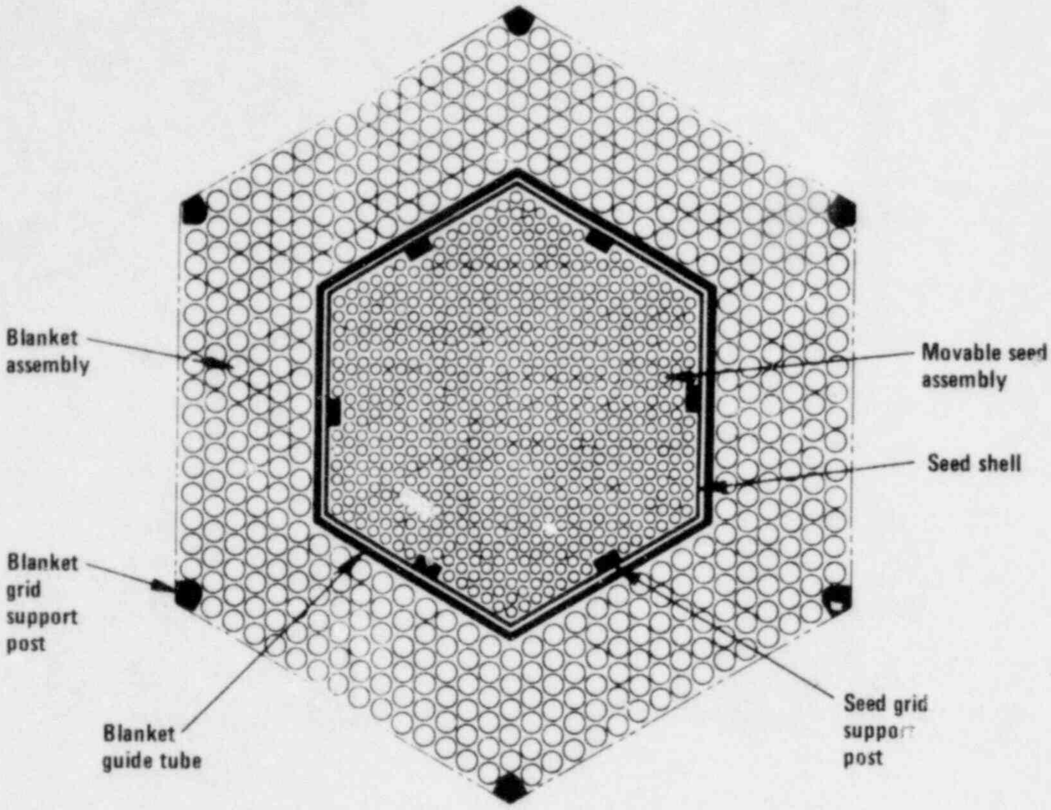


Figure 2-2. Typical seed/blanket module cross section.

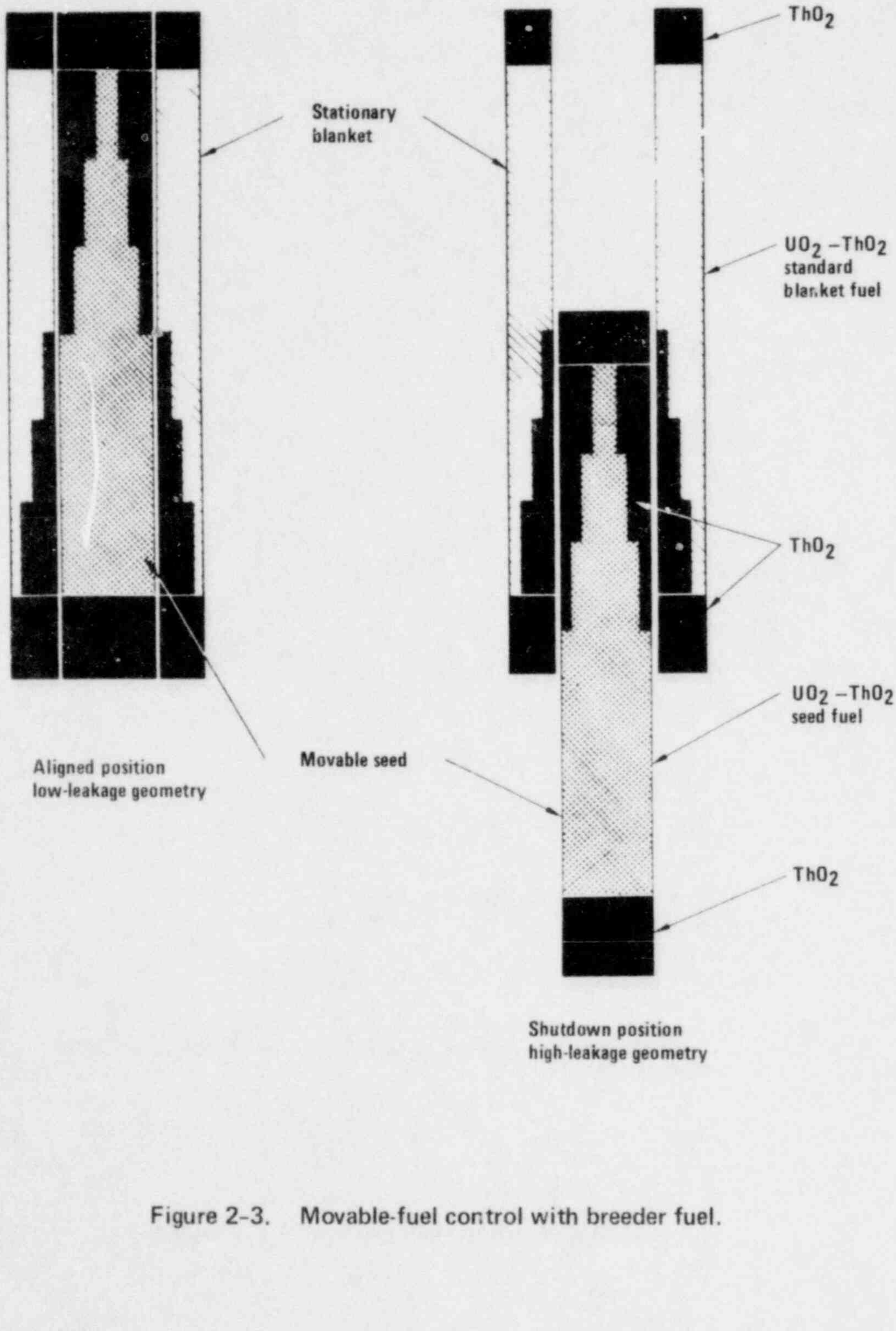


Figure 2-3. Movable-fuel control with breeder fuel.

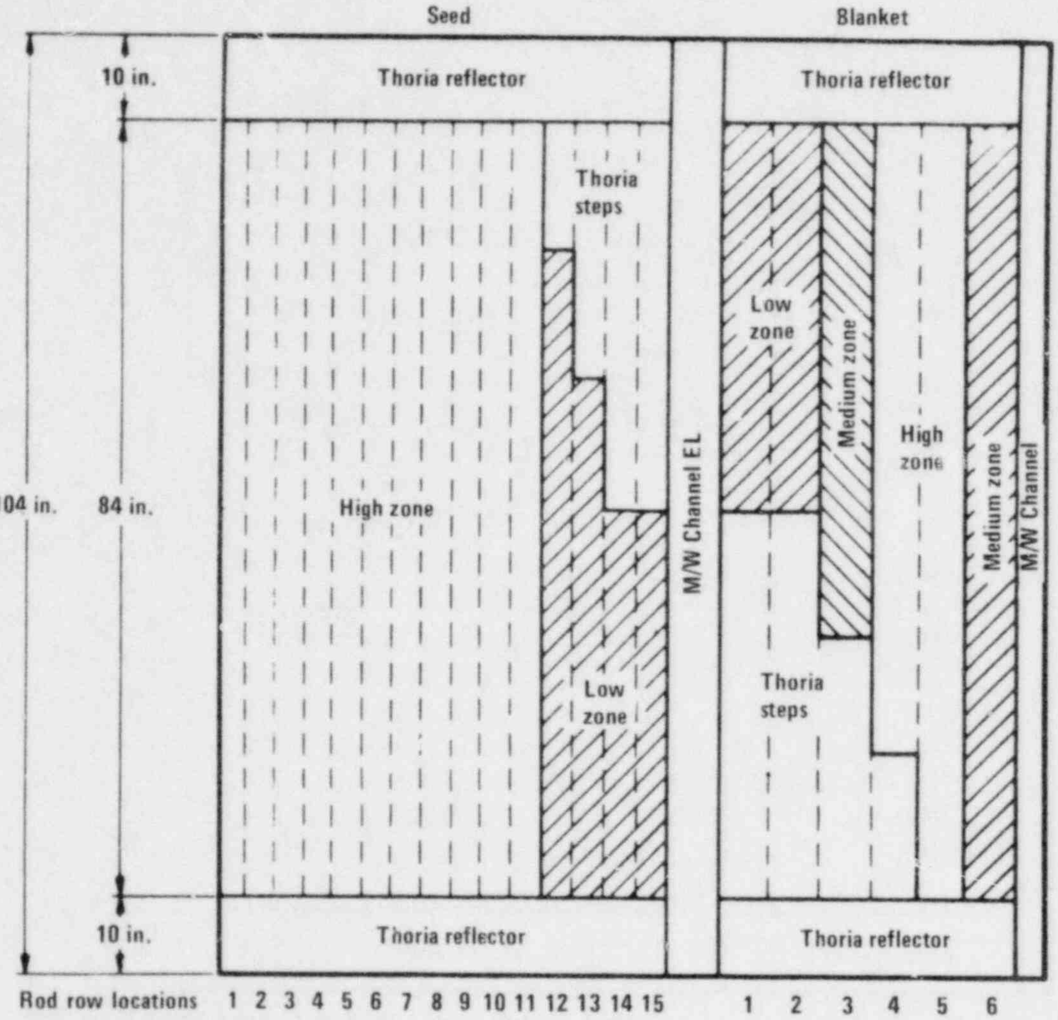


Figure 2-4. R-Z schematic of module.

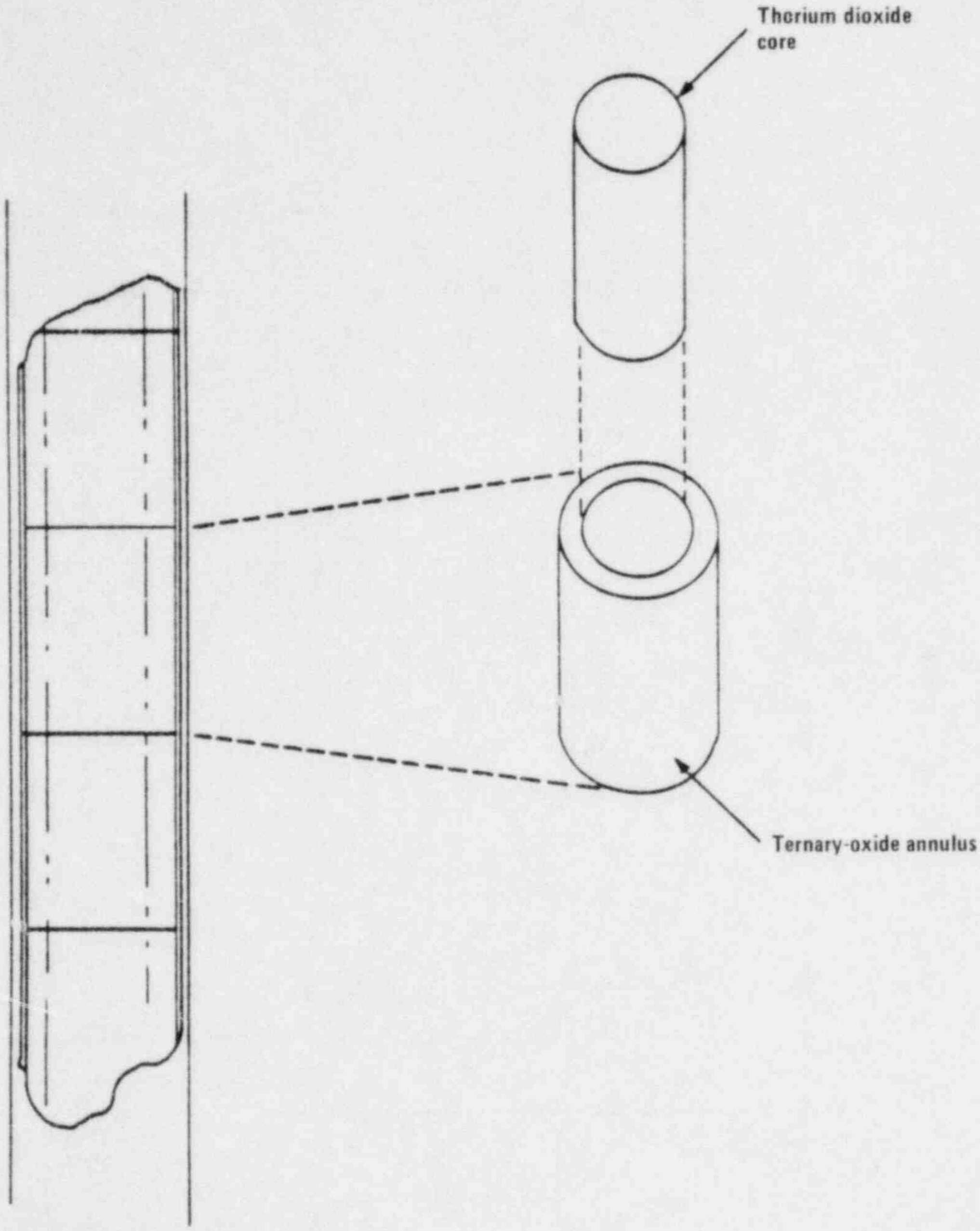
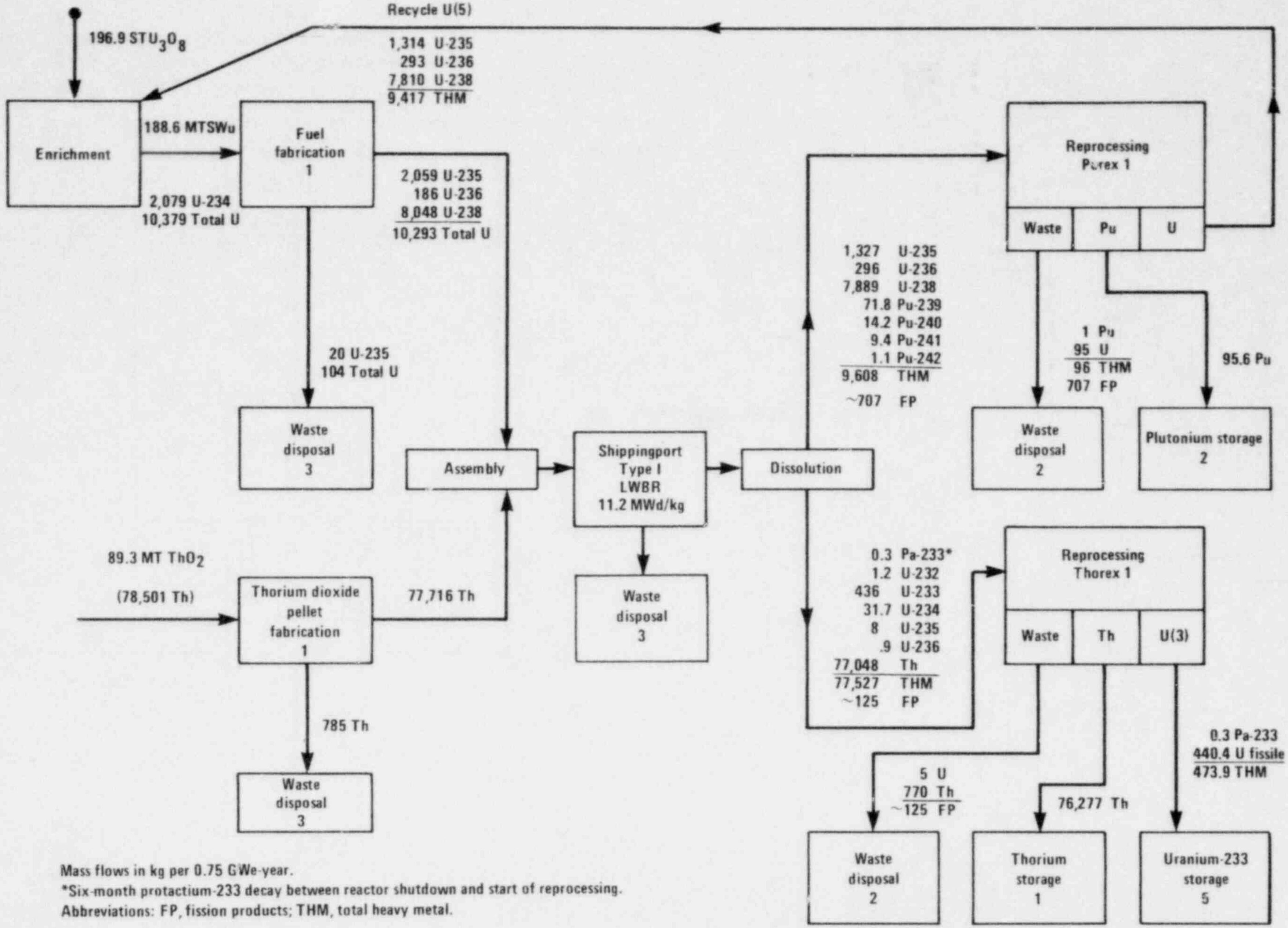
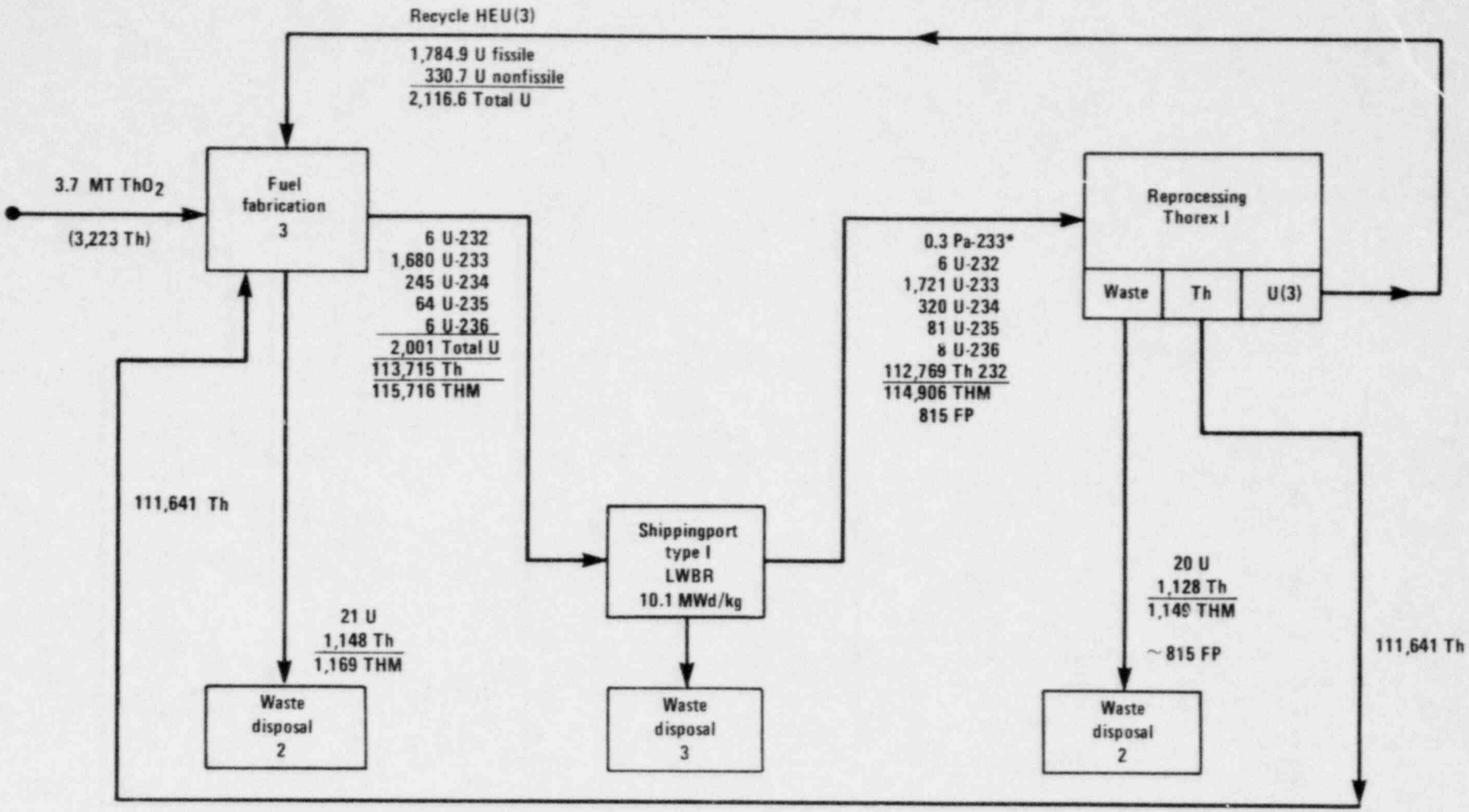


Figure 2-5. Duplex fuel pellets.



Mass flows in kg per 0.75 GWe-year.
*Six-month protactium-233 decay between reactor shutdown and start of reprocessing.
Abbreviations: FP, fission products; THM, total heavy metal.

Figure 2-6. Typical LWBR material-flow diagram for MEU(5), Th prebreeder Shippingport type I concept.



Mass flows in kg per 0.75 GWe-year.

*Six-month Pa-233 decay between reactor shutdown and start of reprocessing.

Abbreviations: FP, fission products; THM, total heavy metal.

Figure 2-7. Typical LWBR material-flow diagram, HEU(3)-Th, Shippingport type I breeder concept.

2.2 FUEL MECHANICAL, NUCLEAR, AND THERMAL-HYDRAULIC CONSIDERATIONS

Since the Shippingport LWBR design was supported by an extensive program of in-reactor and out-of-reactor testing, and the core has operated satisfactorily for over 12,000 hours as of the end of September 1979 effective full-power hours, the mechanical and thermal-hydraulic design features and the material applications constitute proven concepts. The mechanical and thermal-hydraulic design of the present prebreeder and breeder concepts and the materials used (except for the grids) are therefore assumed to be similar to, if not identical with, those of the LWBR.

2.2.1 MECHANICAL CONSIDERATIONS

2.2.1.1 Fuel-Ro[illegible]acing Grids and Subassembly Structural Rigidity

In the Shippingport LWBR type I module, the fuel rods are held in place laterally by a system of AM-350 steel grids. In the present prebreeder and breeder concepts the grids are assumed to be Zircaloy. The technical capability to substitute Zircaloy for AM-350 grids will have to be confirmed by development and testing.

The seed assembly would be contained in a Zircaloy shell. Another Zircaloy shell (the guide tube) would be located inside the annular blanket assembly and provide a path for lifting and lowering the movable seed assembly. Both shells would provide structural rigidity for the assemblies. Zircaloy support posts for the grids would also be present along the outer boundary of the blanket assembly.

2.2.1.2 Module Suspension

In the Shippingport LWBR core, each of the 12 movable seed assemblies is suspended by a lead screw from the control-drive mechanism, which in turn is attached to the pressure-vessel head. The stationary assemblies are also suspended from the head. In the present prebreeder and breeder concepts, the mechanical arrangement for lifting and lowering the movable seed assemblies would be similar to that in the Shippingport LWBR; the support and holddown arrangement for the stationary blanket assemblies in the present concepts has not been specified but could also be similar to that used in the Shippingport LWBR.

2.2.2 VALIDATION OF CALCULATIONS

Nuclear calculations have been performed by Bettis Atomic Power Laboratory (Bettis) to estimate reactivity levels, loading requirements, lifetime, and mass flows for the core. Sufficient calculations were performed to estimate the time-dependent effects of fuel recycle on these parameters. Nuclear cross sections for use in depletion calculations were generated by methods that explicitly represent the space and energy effects on neutron resonance capture. Resource requirements were estimated from the time-dependent mass-flow data. In addition, thermal and hydraulic calculations have been performed by the DOE to estimate core power capability.

2.2.2.1 Nuclear Calculations for the Prebreeder

The nuclear performance of the LWBR type I module prebreeder concept was determined by Bettis from three-dimensional diffusion theory (PDQ) module calculations using four neutron-energy groups with breakpoints at 0.8 MeV, 5.53 keV, and 0.625 eV. In these calculations, the height of the movable-fuel section of the module was adjusted throughout depletion to provide a critical configuration. Neutron cross sections for these calculations were determined by means of the PAX03 computer program.

Microscopic cross sections for all important isotopes were obtained with the PAX03 program, which combines calculations for fast spectrum, resonance effects, thermal spectrum, and self-shielding. Geometric input to PAX03 consists of physical descriptions of the components of the module (e.g., a fuel cell comprised of a fuel pellet, cladding, and associated water, and a metal-water cell comprised of a guide tube and associated water) and the relative volumes of these components within the module. Heterogeneous resonance integrals are determined in PAX03 by a collision-probability method based on the integral Boltzmann equation under the assumption of isotropy in both the laboratory and center-of-mass systems. All scattering and slowing-down sources are assumed to be flat over an individual region of the cell. Thermally, spatial shielding of the cross sections is treated using the Multiple-Sauer method. A point-depletion capability in PAX03 was used to obtain cross-section behavior as a function of time. The depletion model used in PAX03 was the same as that used in PDQ. In this four-energy-group model, equations describing the depletion behavior of all important heavy elements and of the predominant fission products were solved. The treatment of the behavior of fission products used 11 major decay chains (28 fission-product isotopes) and a twelfth fictitious chain to account for the remaining nuclides.

The methods used in calculating the nuclear performance of the conceptual prebreeder are refinements of the methods developed and validated in the Shippingport PWR and LWBR programs, which is the major validation of these methods.

2.2.2.2 Nuclear Calculations for the Breeder

The nuclear performance of the breeder concept was determined by Bettis from point-depletion calculations utilizing four neutron-energy groups with breakpoints at 0.8 MeV, 5.53 keV, and 0.625 eV. The neutron cross sections used in these calculations were obtained from detailed Monte Carlo calculations for representative fuel assemblies. The point-depletion results were used to estimate reactivity levels, lifetime, breeding performance, and mass flow for the core.

Effective few-group microscopic cross sections were generated by means of the RCP01 Monte Carlo program. The Monte Carlo model used 31 energy intervals to describe neutron energies between 0 eV and 10 MeV, with each interval being further divided into as many as 1,000 subintervals to permit accurate representation through all resonances. The primary source of basic cross-section information was the ENDF/B data libraries.

Detailed hexagonal fuel assemblies were represented in the RCP01 calculations, including explicit geometric representations of the fuel pellet, cladding, moderator, and, where appropriate, guide tube for each fuel-bearing and non-fuel-bearing rod in the assembly. The calculated isotopic reaction rates were used to generate highly accurate few-group microscopic cross sections, appropriate for an entire assembly, for use in the point-depletion model. To facilitate the rapid examination of a number of design concepts, several different RCP01 calculations were made to span the range of fuel temperature, moderator temperature, and fuel-to-coolant ratio anticipated for the breeder. In addition, heavy-metal isotopic mixes characteristic both of initial and of equilibrium-cycle loadings were represented.

Few-group microscopic data from the RCP01 calculations were employed in the four-neutron-energy group survey depletion model. In this model, equations describing depletion chains for all important heavy-metal isotopes and the dominant fission-product chains for xenon and samarium were solved. Additional fission-product absorption was incorporated via a residual-fission-product nuclide. The point-depletion results were used to estimate core reactivity levels and lifetimes and to calculate the ratios of heavy-metal isotopes as a function of fuel depletion. In using this model to evaluate breeding performance, appropriate adjustments were made to the calculated conversion ratios to account for leakage and noncritical reactivity levels in the computations. The estimates of breeding performance and isotopic ratios as a function of fuel depletion were then combined to obtain the desired estimates of core mass flow.

The method used in calculating the nuclear performance of the breeder evolved from the methods developed and validated by analysis of the operating LWBR core at Shippingport.

2.2.2.3 Resource Requirements

The requirements for yellowcake, thorium dioxide, and separative work were estimated from the prebreeder and breeder mass flows normalized to 1,000-MWe reactors. The prebreeder is assumed to operate continuously until sufficient uranium-233 has been generated to supply both in-core and out-of-core inventories for the breeder as well as the small amount of makeup required through 100 years of operation.

The resource requirements for the conceptual prebreeder-breeder system were estimated on a cycle-by-cycle basis from the previously calculated mass flows. Integral results for 30, 50, and 100 years of operation were obtained as sums of the cycle-by-cycle results normalized to an integrated-system capacity of 1,000 MWe.

2.2.3 THERMAL-HYDRAULIC CONSIDERATIONS

2.2.3.1 Core-Coolant Flow

A common feature of all seed-blanket reactor concepts is that the power density in the seed assemblies is higher than that in the blanket assemblies. In the Shippingport LWBR core, the available flow is apportioned by the use of flow orifice plates located in the top and the bottom of each blanket assembly. The details of the orificing arrangement were dictated in part by certain specific features of the Shippingport plant, such as pumping power. However, the use of orificing would be desirable in the present prebreeder and breeder concepts to minimize pumping power.

2.2.3.2 Bypass Inlet Flow Balance System

In the Shippingport LWBR core, the bypass inlet flow balance system is used to counteract the upward hydraulic force on each movable fuel assembly and to insure that the weight of the assembly is an effective net downward force on the movable fuel assemblies whenever they are released from the control-drive mechanism. The net downward force is, in turn, required to insure that the movable fuel assemblies move downward to subcritical positions when safety shutdown (scram) is signaled by the plant operating instrumentation and the control-drive mechanism is unlatched from the lead screw.

The bypass inlet flow balance system consists of a hydraulic piston connected to the top of each movable seed assembly and associated piping. The piping transmits fluid from the pressure-vessel inlet at the bottom of the pressure vessel to the hydraulic piston; the pressure on the hydraulic piston is essentially the core inlet pressure.

The present prebreeder and breeder concepts also require a system for balancing the upward hydraulic force on the movable seed assemblies. A system similar to that used for the Shippingport LWBR could be used.

2.2.3.3 Thermal Analysis Parameters

Thermal performance has been analyzed by Bettis with a simplified calculational model. This model has been qualified by performing detailed module calculations allowing for the transfer of two-phase fluid properties in three dimensions to predict local fluid conditions and the critical heat flux. These detailed calculations were made with the computer program HOTROD, which was used for the thermal analysis of the Shippingport LWBR core.

The simplified model relates the steady-state overpower thermal performance of the proposed concept to that of a reference commercial design, such as the Babcock & Wilcox Standard 205 design (Ref. 1). The difference in total reactor flow between the proposed concept and the reference design is determined from changes in parameters that affect the flow, such as the core hydraulic diameter, total core flow area, fuel-rod length, and the number and type of grids. Mass velocity and inlet temperature are calculated from the flow for a specified core-average temperature. The hot-channel critical-heat-flux performance is then determined by factoring in changes in the parameters that affect the critical heat flux, such as mass velocity, hydraulic diameter, inlet temperature, power-peaking factors, and channel length.

Commercial design procedures, methods, hot-channel factors, and critical-heat-flux correlations provided the basis for the analysis. The peak linear power would be maintained at a level that results in acceptable fuel element and loss-of-coolant accident (LOCA) performance.

2.3 ENVIRONMENTAL CONSIDERATIONS

2.3.1 SUMMARY ASSESSMENT

The thermal effluents from the conceptual breeder and prebreeder based on LWBR type I modules would be slightly lower in quantity than those from the reference LWR. The chemical and biocidal releases would be similar. The radiological impacts from normal operation are predicted to be about one-third to two-thirds (depending on which dose component is considered) of those from the reference LWR.

2.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM (R.G. 4.2/3.2)

The prebreeder and breeder concept described in this chapter is one LWBR scheme that may be capable of meeting current regulations, including the requirements of Appendix I to 10 CFR 50. This scheme would consist of a prebreeder and breeder based on a fuel-element geometry identical with that of the Shippingport movable-fuel (type I) LWBR, except that the seed lattice of the breeder is modified to reduce the Zircaloy content. The cores would have the same dimensions, thereby allowing the breeder core to replace the prebreeder core in the same vessel without changes in the mechanical design of the vessel or closure. Typical basic plant parameters are presented in Table 2-10.

2.3.3 STATION LAND USE

There are no specific features of the LWBR plants that would indicate differences in land use from that of LWR plants.

2.3.4 STATION WATER USE (R.G. 4.2/3.3)

The predominant single use of water is for makeup to the heat-dissipation systems. Much smaller amounts are required for the plants (after demineralization) as well as for such uses as laundry, showers, and sanitary facilities (Table 2-11). As shown in Table 2-11 the average annual rate of makeup water required would be 7,900 gpm at 1,000 MWe. In comparison the LWR reference plant requires 8,500 gpm.

2.3.5 HEAT-DISSIPATION SYSTEMS (R.G. 4.2/3.4)

A 1,000-MWe prebreeder or breeder plant would reject approximately 6.2 x 10^9 Btu/hr of waste heat. Any of several types of heat-dissipation systems may be used, depending on site conditions and other factors. One common type is the wet natural-draft cooling tower. That type of system, with freshwater makeup, was assumed for this report. The same type of system has been assumed for the reference LWR.

A typical natural-draft cooling tower for a 1,000-MWe unit has a single shell with a height of about 550 feet and a maximum shell diameter of about 410 feet. Heat is dissipated to the atmosphere by evaporation and by sensible-heat transfer. Evaporation predominates, but the balance between the two depends on air temperature and humidity. The average rate of water use therefore varies from month to month. Blowdown is required to limit the concentration of solids in the circulating water. For the plants discussed here, a maximum concentration factor of 5 is assumed, although other values are frequently found. Data for a typical heat-dissipation system are shown in Table 2-12 for a site in the north-central United States.

Circulating water would be periodically chlorinated to control algae and other slime-forming microorganisms. Typically, chlorine would be added as required to achieve a residual chlorine content of 0.5 to 1.0 ppm for 1 to 2 hours per day. The cooling-tower blowdown may have a small residual chlorine content during periods of chlorination.

2.3.6 RADWASTE SYSTEMS AND SOURCE TERMS (R.G. 4.2/3.5)

This section briefly describes sources of radioactivity, release paths, and typical radioactive-waste-processing systems in commercial PWRs. It also presents estimates of the quantities of radioactivity which might be released. These quantities are based on releases from a typical PWR (Ref. 2); they are normalized to 1,000 MWe and adjusted to account for differences in the PWR and LWBR concepts. No specific attempt has been made to reduce these releases below those for a typical PWR and no specific plant concept has been selected for the LWBR. Since LWBR cores are PWR cores, this presents a representative but not exclusive case for review as part of this effort.

The main differences accounted for are fuel composition, reactivity-control systems, and rates of burnup. The fuel in a typical PWR is a low-enrichment uranium oxide. In the breeder based on the LWBR type I module concept, the fuel would consist of a natural thorium oxide or a binary mixture of uranium-233 oxide and thorium oxide in both the movable seed assemblies and the fixed blanket assemblies, ratio of uranium-233 to total heavy metal (i.e., enrichment) being higher in the seed assemblies. The fuel in the prebreeder, consisting typically of natural thorium oxide (without uranium) and a moderately enriched uranium oxide, would perform the same function as the binary mixture of uranium-233 oxide and thorium oxide in the breeder. The result of these differences would be a slight shift in the percent fission yield as a function of mass number.

The reactivity-control systems for the PWR and LWBR concepts also differ. In the PWR reactivity is controlled by the use of various neutron poisoning methods, including neutron-absorbing control rods, burnable-poison rods, and the addition of boric acid to the reactor coolant. In the prebreeder and breeder based on the LWBR type I module, on the other hand, reactivity would be controlled by lifting or lowering movable seed assemblies. By changing the axial position of the seed assemblies relative to the stationary blanket assemblies, the relative rates of neutron absorption by the fissile uranium-233 (or uranium-235) and the fertile thorium (or uranium-238) would also change, thereby changing the reactivity in the core. The result of this difference would be the elimination of the use of boron in the reactor. Among other things, this would eliminate a main source of tritium production.

The last major difference between the PWR and LWBR systems accounted for in the calculated quantities of radioactivity released is the burnup. The average discharge fuel burnup in the PWR is about 30,000 MWd/MT versus 10,000 MWd/MT for the LWBR concept. This results in a reduction in radioactivity release rates.

2.3.6.1 Source Terms (R.G. 4.2/3.5.1)

The sources of radioactivity in the plants would be fission products and materials in the reactor core and coolant that become activated by neutron irradiation. Small amounts of fission products are released to the reactor coolant through defects in the fuel cladding, while corrosion and wear products of plant and core materials deposit on the core, become activated, and would be released to the coolant. The estimates of radioactivity releases below are based on the same fuel-failure fraction for the LWBR and for the PWR. Operating experience may show significantly lower values for LWBRs than for PWRs. Periodic radiochemical analyses of the coolant have shown no indication of fuel-element failure during the first 10,000 EFPH of operation of the Shippingport LWBR. To establish this, research, development, and demonstration will be necessary, using data from the Shippingport LWBR and possibly other fuel irradiations.

Two isotopes of particular interest are carbon-14 and tritium. Carbon-14 is produced by an (n,p) reaction of nitrogen-14 and by an (n, α) reaction of oxygen-17. Tritium is produced by ternary fissions. Radioactivity would be removed from the reactor coolant by cleanup in the chemical and volume control system and by fluid removal from the system by leakage. Figure 2-8 is a block diagram showing potential paths for the removal of radioactivity from a typical reactor-coolant system. The leakage paths would serve as sources of radioactivity to other plant systems.

Figure 2-9 shows typical steam and power-conversion system components that are most important from the standpoint of radioactivity in the system and releases to the environment. Noble gases and small amounts of iodine that might leak into the steam generator would be carried out with the steam, pass through the turbine and condenser, and be removed from the condenser by the air-removal system. A filter system would remove most of the iodine, leaving the noble gases and a small amount of iodine to be discharged into the atmosphere. Noble gases and iodine also could reach the atmosphere directly in a small amount of steam leakage. Nonvolatile radioactive materials could collect in the steam-generator liquid. They would be removed in the blowdown stream, which might go to a condenser and there mix with the condensate; alternatively, it might be held in a blowdown-collection tank and not fed into the condensate steam. In some plants about 65% of the condensate stream passes through the condensate-polishing demineralizer as it is returned to the steam generator. Thus, nonvolatile radioactive isotopes may be collected in the condensate-polishing demineralizers, or held in a blowdown-collection system.

2.3.6.2 Liquid-Radwaste System (R.G. 4.2/3.5.2)

A miscellaneous-liquid-waste system (Figure 2-10) could process liquid wastes from the sources described above as well as from such other sources as laundries and showers, equipment drains, and floor drains. Laundry and shower wastes and condensate from the containment coolers would be collected and monitored. If there is no significant radioactivity, these wastes would be discharged, with the laundry and shower wastes being filtered before discharge. If they contain significant activity, these streams would be routed to the equipment discussed below for processing.

Waste to be processed would be collected in the waste tanks and passed through particulate and carbon filters to remove oil and other organics. It would then go to an evaporative waste concentrator. The concentrates (bottoms) would be sent to the solid-waste-handling system for solidification and disposal. The distillate would be passed through an ion exchanger and then sorted in a waste condensate tank for monitoring and discharge. Discharges from the miscellaneous-liquid-waste system might be directed to the river, lake, ocean, or other body of water on which the plant is sited. The present analysis is based on this assumption, which is worst-case from a radiation-discharge standpoint.

Approximate quantities of important isotopes making up the typical liquid radioactive source term are shown in Table 2-13.

2.3.6.3 Gaseous-Waste System (R.G. 4.2/3.5.3)

A typical gaseous-waste system is shown in Figure 2-11. Compressed storage would be provided for gases removed from equipment that processes reactor coolant. These gases are hydrogen or nitrogen possibly containing small amounts (volumetrically) of fission products. A recombiner would be provided to allow the removal of hydrogen or oxygen from the stored gases.

Nitrogen cover gas displaced by filling the reactor drain tank would be compressed in the gaseous-waste system. Hydrogen could be removed by recombination and the nitrogen stored for reuse as a cover gas.

In addition to these potential major sources of radioactive gases, there are the potential leakage paths discussed earlier. These would be small leaks from the reactor-coolant system to the containment, small leaks of reactor coolant to the auxiliary building, and small leaks from the reactor-coolant system to the steam and power-conversion system.

A typical containment would be equipped with an internal recirculating filter system containing particulate, absolute, and charcoal filters for the removal of particulates and radioiodines before containment purge. Such a containment would be vented or purged through similar filter systems.

The auxiliary-building ventilation system may also contain particulate, absolute, and charcoal filters. This system would filter air exhausted from areas that might become contaminated by reactor-coolant leakage. Most of the gaseous activity leaking into the steam and power-conversion system would be contained in air removed from the condenser. This effluent would also be filtered by particulate, absolute, and charcoal filters. The calculated gaseous releases of radioactivity are shown in Table 2-14 for this typical PWR type plant system.

2.3.6.4 Solid-Radwaste Systems (R.G. 4.2/3.5.4)

Materials transferred to the solid-radwaste system for disposal would include spent demineralizer (ion-exchange) resins and evaporator concentrates. These would be solidified for offsite disposal. Other solid wastes (contaminated clothing, paper, and filters) would also be sent off the site for disposal. A total of 1,050 drums (capacity 55 gallons) is estimated to be shipped for offsite disposal each year. This is the same estimate used for the reference LWR.

2.3.7 CHEMICAL AND BIOCIDAL WASTES

The primary sources of chemical and biocidal wastes would be the cooling-tower blowdown stream and chemical effluents from the regeneration of demineralizers (ion-exchanger resins) that treat makeup water. The cooling-tower blowdown stream would contain dissolved solids that enter the makeup stream and are concentrated by evaporation during operation of the cooling towers. This stream would also intermittently contain a small chlorine residual from chlorination of the condenser cooling water (Section 2.3.5).

Acid and caustic soda solutions would be used for demineralizer regeneration. These wastes would be held up and neutralized before discharge. They would contain no radioactivity.

2.3.8 EFFECTS OF OPERATION OF THE HEAT-DISSIPATION SYSTEM (R.G. 4.2/5.1)

The quoted thermal efficiency of the prebreeder concept and of the breeder concept is slightly higher than that of the reference LWR. The effects are therefore a fraction of a percent smaller than those for the reference LWR.

2.3.9 RADIOLOGICAL IMPACTS FROM ROUTINE OPERATIONS (R.G. 4.2/5.2)

The radiation-dose contributions from liquid effluents from the typical PWR plant described above are listed by nuclide in Table 2-15; those from noble gases and from radioiodines and particulates are shown on Tables 2-16 and 2-17, respectively. The doses are lower than those for the reference LWR by factors of 0.3 to 0.6. These values are within the guidelines of Appendix I guidelines (applicable to LWRs) and therefore should not represent any problem in licensing.

2.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES (R.G. 4.2/5.3)

The chemical and biocidal discharges would be similar to those from the reference LWR; therefore the effects will also be similar.

2.3.11 OCCUPATIONAL EXPOSURE

Compilations and studies of historical data show that workers in typical commercial PWR plants are exposed to an integrated radiation dose that averages 400 to 500 man-rem/yr-unit. Since this is related to plant design and is not greatly affected by the installed core, a backfit prebreeder/breeder system based on LWBR type I modules installed in a typical commercial PWR plant such as that described here would yield similar exposure rates. Most of this dose would be incurred in maintenance and repair activities; much smaller amounts would be received in reactor operation, waste processing, and refueling. Exposures of this magnitude may be expected for the units discussed here, regardless of the type of core installed.

Table 2-10. Basic parameters describing the prebreeder and breeder concepts based on LWBR type I modules

Parameter	Prebreeder	Breeder
Fuel cycle	U-Th recycle	U-Th breeder
Burnup, MWd/MT	11,000	10,100
Base reactor thermal output, MWt	2,026	2,026
Electrical output, MWe	721	711
Normalized electrical output, MWe	1,000	1,000
Heat rate, Btu/kW-hr	9,590	9,720
Heat dissipation rate at 1,000 MWe, Btu/hr	6.2×10^9	6.3×10^9

Table 2-11. Typical station water use for a 1,000-MWe plant

Use	Flow (gpm)
Makeup to cooling-tower system (maximum)	13,400
Makeup to cooling-tower system (average)	7,900
Input to laundry, hot showers, sanitary and potable water	3
Input to demineralized-water system	140
Demineralized-water-system waste	10

Table 2-12. Typical heat-dissipation-system design data for a wet natural-draft cooling tower for a 1,000-MWe plant

Heat-dissipation rate (maximum, full power), Btu/hr	6.2×10^9
Evaporation and drift (maximum, full power), gpm	10,600
Evaporation and drift (annual average), gpm	6,300
Blowdown (maximum), gpm	2,800
Blowdown (annual average), gpm	1,600

Table 2-13. Liquid radioactive release source terms for a typical commercial PWR plant described here with LWBR-type core installed[a]

Nuclide	(Ci/yr)	Nuclide	(Ci/yr)
Bromine-82	0.00007	Cesium-138	0.00001
Bromine-83	0.0001	Barium-139	0.00002
Rubidium-86	0.00002	Barium-140	0.0001
Strontium-89	0.0002	Lanthanum-140	0.00005
Strontium-91	0.00005	Cerium-141	0.00001
Yttrium-91m	0.00001	Cerium-143	----
Yttrium-91	0.00008	Praseodymium-143	0.00001
Zirconium-95	0.00001	Cerium-144	0.00002
Niobium-95	0.00001	Praseodymium-144	0.00001
Molybdenum-99	0.0001	Neodymium-147	----
Technetium-99m	0.0001	Sodium-24	0.00005
Ruthenium-103	0.00001	Phosphorus-32	0.00001
Rhenium-103m	0.00001	Phosphorus-33	0.00005
Tellurium-125m	----	Chromium-51	0.00007
Tellurium-127m	0.0001	Manganese-54	----
Tellurium-127	0.0002	Manganese-56	0.0003
Tellurium-129m	0.0002	Iron-55	0.00007
Tellurium-129	0.00009	Iron-59	0.00007
Iodine-130	0.0002	Cobalt-58	0.001
Tellurium-131m	0.0002	Cobalt-60	0.00012
Tellurium-131	0.00005	Nickel-65	0.00001
Iodine-131	0.07	Niobium-92	0.00003
Tellurium-132	0.005	Tin-117m	0.00001
Iodine-132	0.005	Tungsten-185	0.00001
Iodine-133	0.04	Tungsten-187	0.0002
Iodine-134	0.00003	Neptunium-239	0.0001
Cesium-134m	0.00002	Protactinium-233	0.00001
Cesium-134	0.002	All others[b]	0.00005
Iodine-135	0.008	Total[c]	0.15
Cesium-136	0.005		
Cesium-137	0.005	Tritium	100
Barium-137m	0.005		

[a]Normalized to 1,000 MWe.

[b]Includes isotopes with discharges of less than 10^{-5} Ci/yr-unit.

[c]Does not include tritium.

Table 2-14. Gaseous radioactive releases for a typical commercial PWR plant described here with LWBR-type core installed[a]

Nuclide	(Ci/yr)	LWR
Krypton-83m	1	1
Krypton-85m	14	11
Krypton-85	380	380
Krypton-87	2	2
Krypton-88	16	14
Krypton-89	1	1
Xenon-131m	23	44
Xenon-133m	40	80
Xenon-133	2,800	1,200
Xenon-135m	1	1
Xenon-135	62	50
Xenon-137	1	1
Xenon-139	1	1
Iodine-131	0.03	0.05
Iodine-133	0.03	0.06
Tritium	175	580
Carbon-14	3.0	6.0
Particulates	0.025	0.05

[a]Normalized to 1,000 MWe.

Table 2-15. Contributions to doses by liquid effluents for a typical commercial PWR plant described here with LWBR-type core installed

Nuclide	Contribution (%) to organ dose: Adult whole body	Contribution (%) to organ dose: Critical organ
Tritium	20	2
Cesium-134	27	1
Cesium-136	9	1
Cesium-137	40	1
Iodine-131	1	88
Iodine-133	1	8
Others	4	2
Ratio of dose to that from reference LWR	0.35	0.49

Table 2-16. Contributions to doses by noble gases for a typical commercial PWR plant described here with LWBR-type core installed

Nuclide	Contribution (%) to organ dose: Whole body	Contribution (%) to organ dose: Skin
Krypton-83m	(a)	(a)
Krypton-85m	1	1
Krypton-85	1	22
Krypton-87	1	1
Krypton-88	22	9
Krypton-89	1	1
Xenon-131m	(a)	2
Xenon-133m	(a)	(a)
Xenon-133	62	55
Xenon-135m	(a)	(a)
Xenon-135	10	9
Xenon-137	(a)	(a)
Ratio of dose to that from reference LWR	0.44	0.46

[a]Less than 1%.

Table 2-17. Contributions to doses by radioiodines and particulates for a typical commercial PWR plant described here with LWBR-type core installed

Nuclide	Contribution (%) to organ dose: Infant thyroid	Contribution (%) to organ dose: Child thyroid
Iodine-131	97	93
Iodine-133	1	1
Carbon-14	1	4
Tritium	(a)	2
Ratio of dose to that from reference LWR	0.60	0.58

[a]Less than 1%.

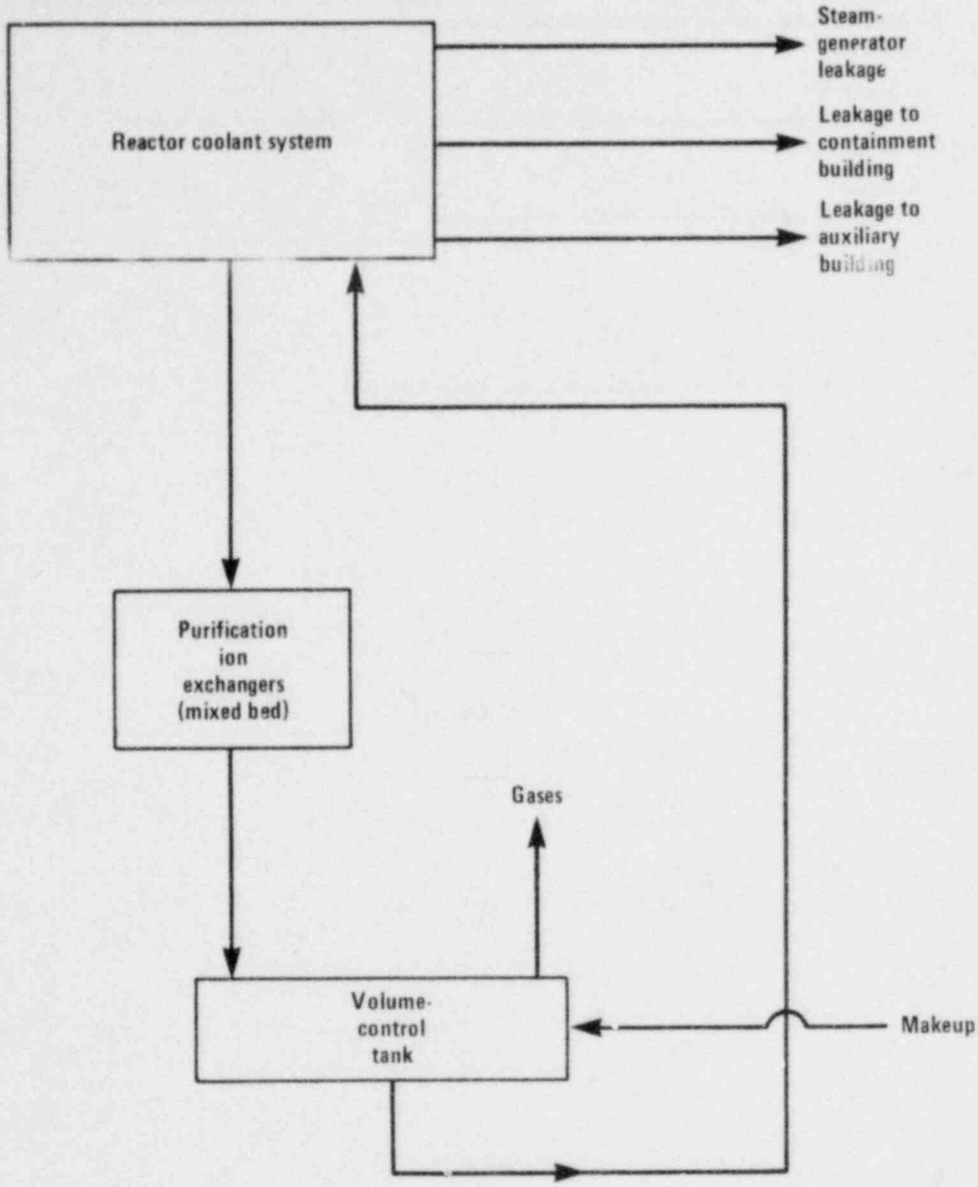
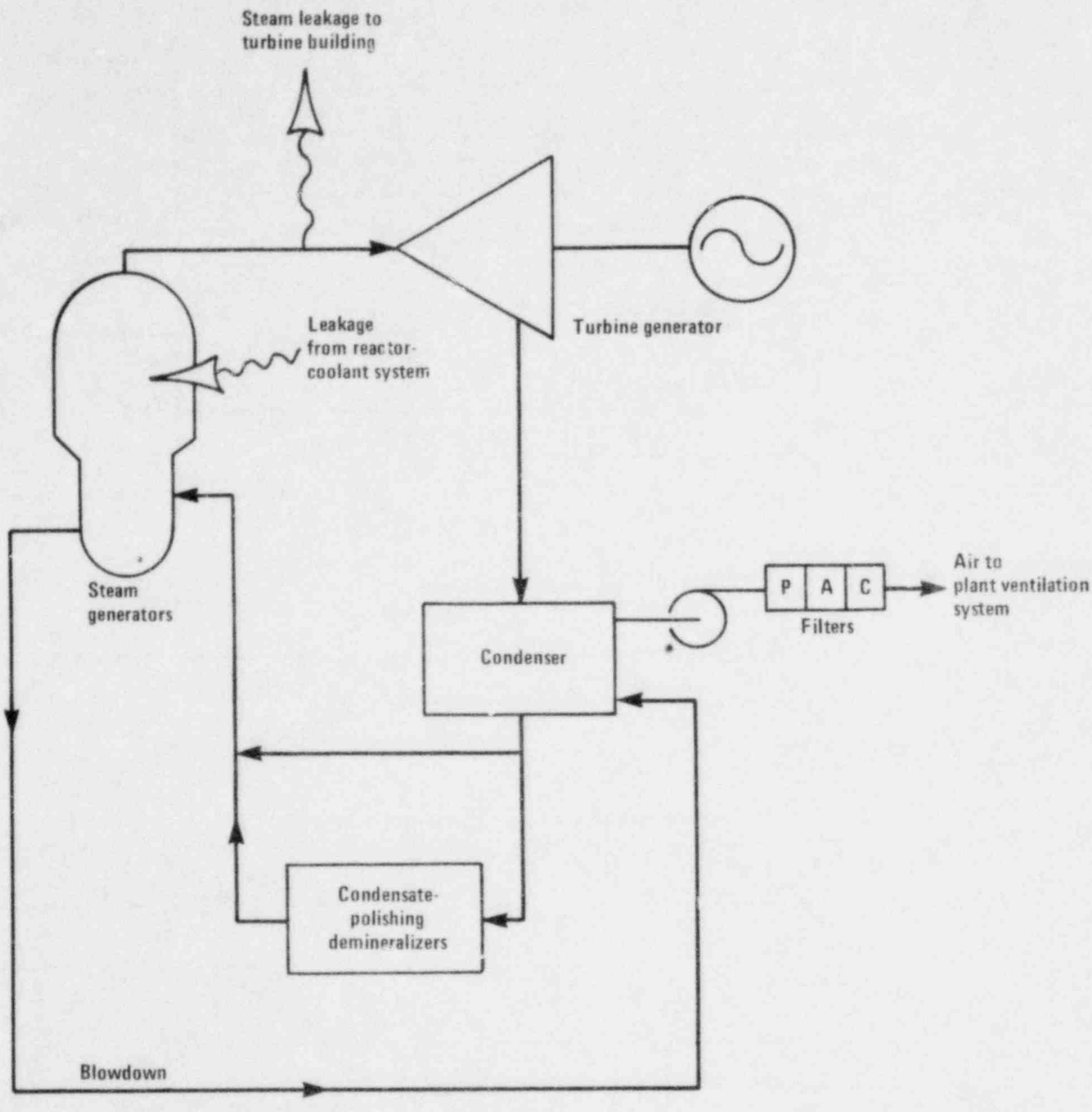


Figure 2-8. Reactor coolant, chemical, and volume-control systems for typical commercial PWR plant described here with LWBR-type core installed.



Abbreviations: P = particulate; A = air; C = carbon.

Figure 2-9. Typical steam and power-conversion systems with sources of radioactivity for typical commercial PWR plant described here with LWBR-type core installed.

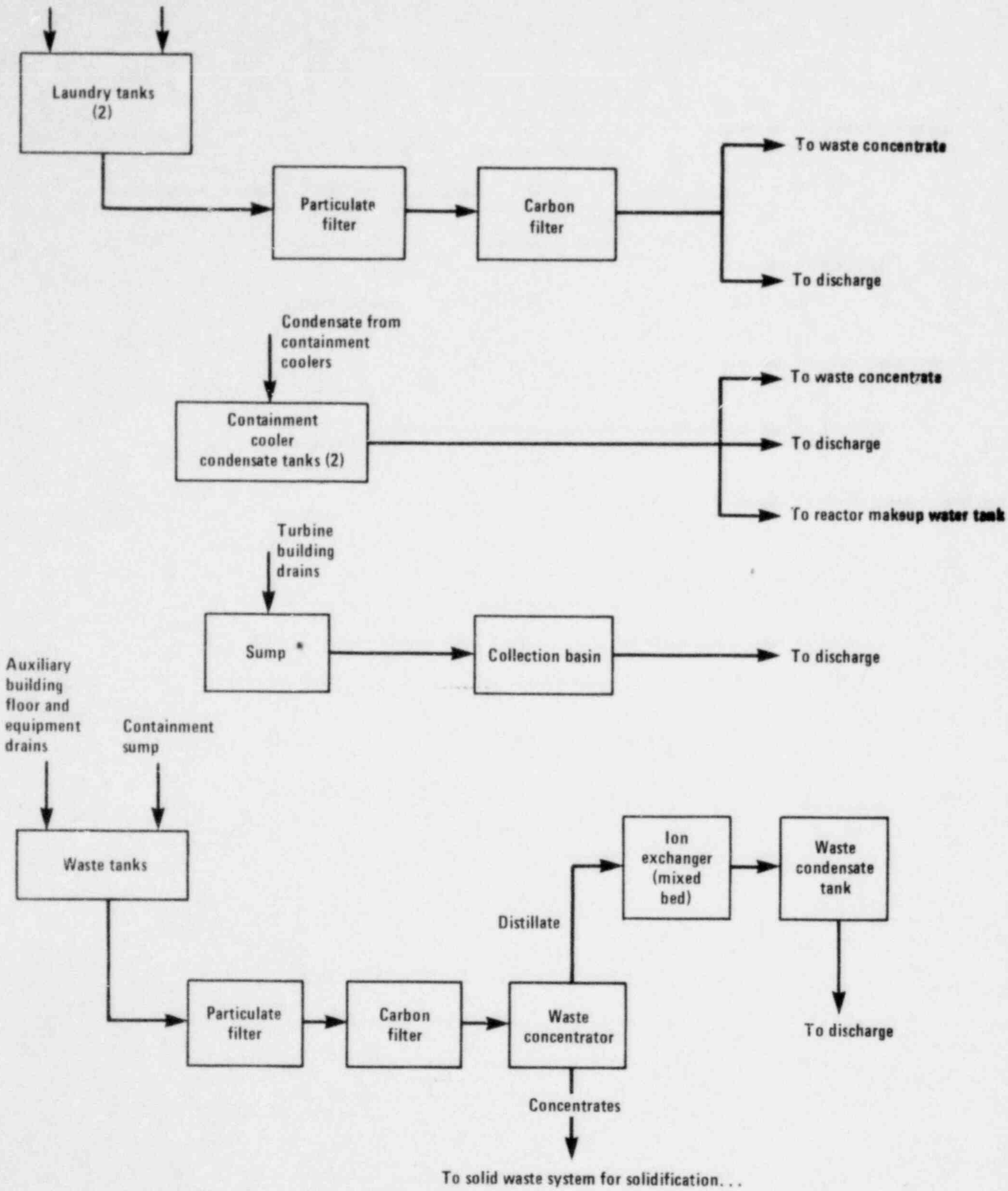
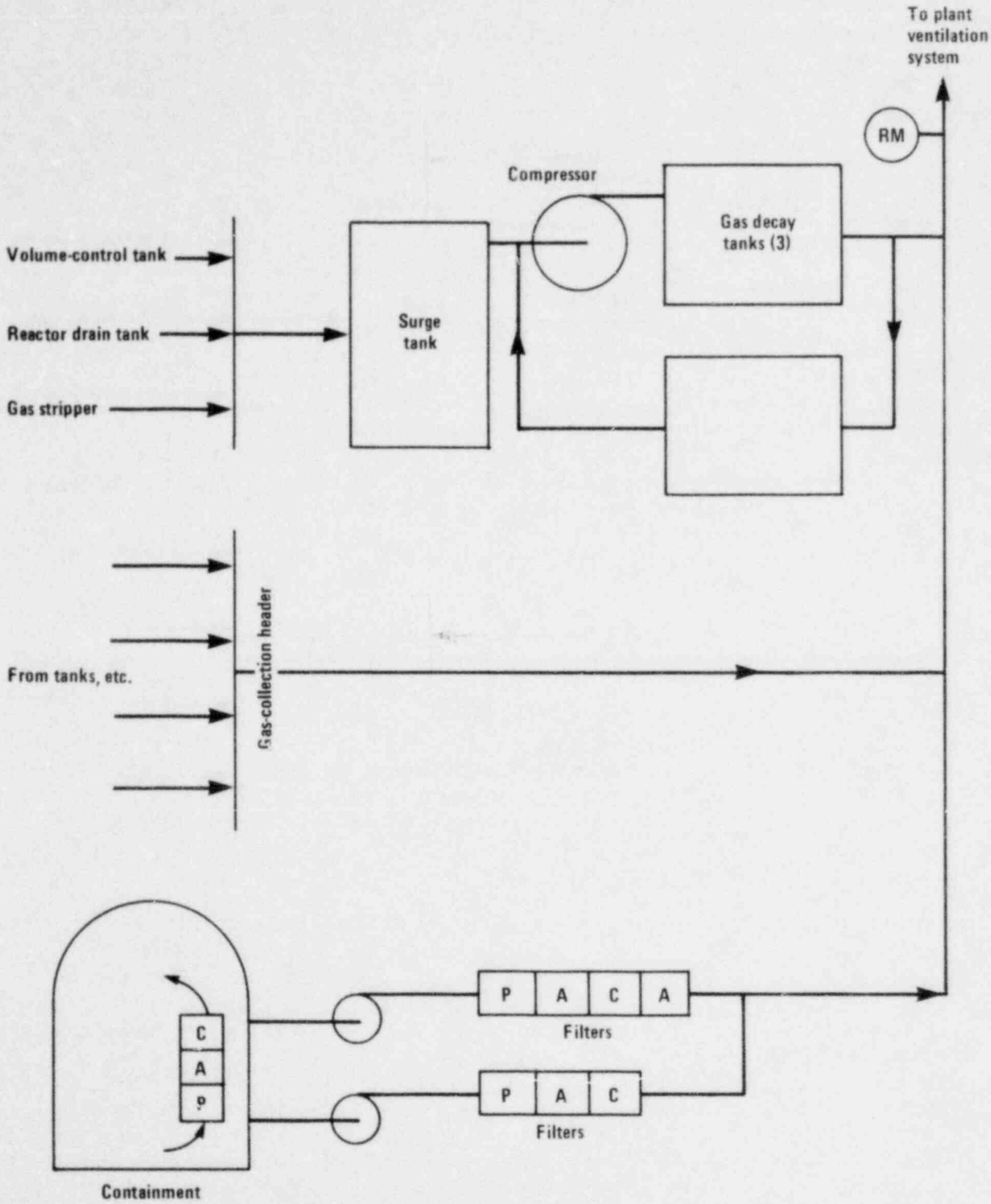


Figure 2-10. Typical miscellaneous-liquid-waste management for typical commercial PWR plant described here with LWBR-type core installed.



Abbreviations: RM = radiation monitor; C = carbon; P = particulate; A = air.

Figure 2-11. Typical gaseous-waste management system for typical commercial PWR plant described here with LWBR-type core installed.

2.4 SAFETY CONSIDERATIONS

2.4.1 UNIQUE SAFETY ASPECTS OF THE LWBR CONCEPT

This section discusses the major unique features of the LWBR concept as compared with other PWRs. These features were evaluated in the Shippingport LWBR safety analysis and were found to be acceptable in the NRC review.

2.4.1.1 Tight Lattice

The rod-to-rod spacing in the LWBR is 60 mils rather than the 120 mils used in commercial PWRs. The close spacing requires a different fuel-element and grid design and different fuel-assembly procedures compared with those used in commercial practice to avoid fuel rod-to-structure and rod-to-rod contact. In setting core operational limits (specifications of set points and allowable power increase) for the Shippingport LWBR it was assumed that rod-to-structure contact does occur. Extensive in-reactor and out-of-reactor tests with rods in contact have been completed. Design assessments of fuel-rod bowing in the Shippingport LWBR core predict that rod-to-rod contact is unlikely but even if it were to occur, it is calculated to be acceptable. Bowing analysis of the blanket has been completed on a worst-case basis, and the rod-to-rod spacing has been determined as a function of reactor lifetime. These data demonstrate that adequate margin is incorporated into the Shippingport LWBR design and indicate that a commercial-scale reactor of this design should be feasible.

2.4.1.2 Core Thermal Margins

Thermal-analysis programs and correlations for the Shippingport LWBR have been proved applicable for thermal-hydraulic analyses under widely ranging axial and radial heat-flux distributions in a close rod array, including a coupled-region interface. The critical-heat-flux correlation for the Shippingport LWBR conservatively predicts the data for the full range of the Shippingport LWBR geometries and heat-flux distributions. Additional critical heat flux testing may be required to confirm the acceptability of the breeder seed lattice modifications mentioned above.

2.4.1.3 Provision for Accident Prevention

The probability of accident initiation for the Shippingport LWBR is comparable to that of any other PWR. The safety and protection of the Shippingport LWBR plant have been designed in accordance with regulatory guidelines and requirements. An emergency core-cooling system appropriate to an LWBR has been incorporated into the Shippingport design. No impediment to providing comparable protection features in a larger plant has been identified.

2.4.1.4 Acceptability of Movable Fuel for Reactivity Control

To date, all reactivity-control functions required by an operating nuclear power plant have been satisfactorily performed in the Shippingport LWBR by means of the movable fuel. These include control functions required for shutdown, plant heatup, power operation, and lifetime reactivity changes.

For large-core applications using fuel management, the amount of control required should be significantly smaller than that available in the LWBR demonstration. However, since the reactivity increase available from one control assembly during power operation is greater in the LWBR than in current LWRs, additional consideration will need to be given to control-mechanism design to assure that control-element ejection is not credible, as was done for the Shippingport LWBR core.

2.4.1.5 Power and Temperature Coefficients

The calculation of temperature and power coefficients using the design model has given good agreement for both critical experiments and the operational configurations of the Shippingport LWBR core. Measured and calculated zero-power temperature coefficients during initial and periodic testing to date have agreed satisfactorily for both hot and cold conditions. These same quantities were compared for single-module experimental configurations and showed good agreement. Measured values of full-power-range reactivity defects during both initial and periodic testing were reproduced with good agreement by the calculational model. In addition, a measured power coefficient near full power was also calculated satisfactorily. Calculations have indicated that the power and temperature coefficients for the present breeder concept would be similarly predictable and comparably negative. It is expected that the power and temperature coefficients for the present prebreeder concept would be in the range between those of current commercial PWR cores and the Shippingport LWBR core.

2.4.1.6 Control Stability and Adequacy

The LWBR core at Shippingport has successfully operated as part of an integrated commercial power network. This operation has included long periods at constant power, several weeks of planned swing-load operations, and controlled startup and shutdown periods from zero to full power. In addition, special tests were run to demonstrate the dynamic characteristics and response of the plant to typical load-change rates common in commercial plants. Performance was satisfactory during both swing-load and steady-state operation and in the special tests. The results indicate that control stability and adequacy can be maintained in a core with movable-fuel reactivity control.

2.4.1.7 Nuclear Stability

Analytical studies have examined the stability of large uranium-233/thorium reactors with high power densities against spatial xenon oscillations and have compared it with the stability of uranium-fueled PWRs of comparable sizes and ratings (Ref. 3). These studies show that uranium-233/thorium systems are inherently significantly more stable, primarily because of lower total xenon yields, a larger fraction of xenon-chain yield direct to xenon, and more negative Doppler coefficients of reactivity. Initial physics testing at the Shippingport LWBR demonstrated that the LWBR module design results in a tightly coupled core and provides confidence that the present breeder concept would have acceptable stability properties. Since the conceptual prebreeder would operate at less than half the specific power (kW/kg fissile) of current PWR cores, it is expected that its stability would be at least as good as that of a commercial PWR.

2.4.1.8 Prebreeder Fuel

The behavior of the fuel, which would be contained in rods identical in outside dimensions to those used in the Shippingport LWBR design and would be composed of thorium dioxide pellets and ternary uranium dioxide/zirconium dioxide/calcium oxide pellets arranged as shown in Figure 2-5, is a potential uncertainty. Specifically, the uncertainty associated with the depletion-dependent behavior of the ternary-oxide annulus remains to be resolved. The concept requires that the ternary-oxide annulus have a peak fuel burnup that is about equal to or slightly higher than that in current commercial practice (about 50,000 MWd/MT). Design questions remain, however, on the structural characteristics of the annular fuel arrangement and on the effect of the high-temperature inner annular surface on defected-rod performance. These matters will have to be investigated before the capabilities of such a system can be established. The prebreeder fuel system is being evaluated for possible further development under the DOE Advanced Water Breeder Applications program, including analytical work to extend the scope of present fuel-element modeling and irradiation testing.

Placing the fissile fuel annulus around the thorium dioxide core, which initially would contain no fissile fuel, would cause most of the fission energy to be generated close to the cladding and therefore would result in lower average fuel-element temperatures than would be obtained with a solid uranium dioxide pellet. The primary purpose of this arrangement is to accommodate a volume of thorium dioxide fuel in the fuel element without derating it and to provide an arrangement whereby selective fuel dissolution during reprocessing would yield relatively pure uranium-233. In addition, preliminary calculations indicate that the arrangement may also have the advantage of reducing the fuel-cladding temperature during a loss-of-coolant accident.

2.4.2 DESIGN-BASIS CRITERIA

The Shippingport LWBR design has been reviewed by the NRC and by the Advisory Committee on Reactor Safeguards. There are no known changes that are required in the design-basis criteria.

2.5 LICENSING STATUS AND CONSIDERATIONS

2.5.1 STATUS

The prebreeder and breeder core concepts would contain modules similar to the type I modules now being used in the Shippingport LWBR. The conceptual prebreeder module would be is identical with the LWBR type I module with the exception of the fuel in the fuel elements. The conceptual breeder module would be identical with the LWBR type I module from a safety standpoint.

Before the Shippingport LWBR started operating, it was reviewed by the NRC, the Advisory Committee on Reactor Safeguards, and an independent group appointed by the Commonwealth of Pennsylvania. The reviews uncovered no safety-related licensing issues for this 60-Mw experimental reactor with a planned 3-year operating period, indicating any concerns significantly different from those of commercial PWRs. The NRC has not yet reviewed a large, commercial-size light-water breeder reactor design, but some issues have been identified. These issues can probably be resolved during the engineering development of commercial plants. The design-basis criteria reviewed by the NRC for the Shippingport light-water breeder reactor are not expected to change for the large commercial light-water breeder design, except as these criteria may be generally changed for all reactors as a result of the ongoing reviews of the Three Mile Island incident. It is believed that the existing LWBR Safety Analysis Report, the approximately 250 technical reports published to date on LWBR development and engineering test work, the numerous government agency and public reviews, and the successful operation of the Shippingport LWBR have greatly minimized the uncertainty in developing and licensing a commercial-size version.

The LWBR core is now operating in the Shippingport Atomic Power Station with a core loading of U-233/Th fuel and more than 12,000 effective full-power hours of operation have been completed as of the end of September 1979. All test and operating results are satisfactory to date with core behavior being well within the uncertainty limits provided for in core-performance predictions.

2.5.2 UNRESOLVED ENVIRONMENTAL AND LICENSING CONSIDERATIONS

The principal environmental and licensing issues remaining with respect to the reactor concepts reported here are those generic to the recycle of fuel from fission reactors and the management of their wastes. The uranium-232 produced by the irradiation of thorium fuel has some advantages and some disadvantages relative to uranium fuel recycle.

2.5.3 PLANT OPERABILITY

As mentioned in Section 2.4.1.6, the Shippingport LWBR has successfully operated as part of an integrated commercial power network. This operation has included long periods at constant power, several weeks of planned swing-load operations, and controlled startup and shutdown periods from zero to full power. In addition, special tests were run to demonstrate the dynamic characteristics and response of the plant to typical load-change rates common in commercial plants. Performance was satisfactory during both swing-load and steady-state operation and in the special tests. The results indicate that plant operability would not be adversely affected by a core with movable-fuel reactivity control.

It was noted that LWBRs are expected to be more stable against spatial xenon instabilities than are current PWRs, partly because of the more negative Doppler coefficient of reactivity produced by the thorium. In the case of the prebreeder, which operates at less than half the specific power of current PWR cores, it is expected that the stability would be at least as good as that of a commercial PWR.

The maintenance of a plant containing either of the present prebreeder or breeder concepts would be essentially the same as that of a plant containing a conventional PWR core.

2.6 TECHNOLOGY STATUS AND RESEARCH, DEVELOPMENT, AND DEMONSTRATION REQUIREMENTS

2.6.1 GENERAL

The LWBR core has operated successfully for more than 10,000 effective full-power hours in the Shippingport Atomic Power Station. Therefore, much of the technology for both the prebreeder and the breeder cores is now available. However, the use of a prebreeder-breeder system requires the recycle of fuel and the research and development necessary to implement reprocessing, refabrication, and waste management.

The characteristically low average power density in LWBR cores (about 50 kW/liter) would require the development of large pressure vessels which would be required for application of the advanced breeder concept in a large (1,000 MWe) plant. Under the DOE Advanced Water Breeder Applications (AWBA) program, a vendor has completed a study of the feasibility of manufacturing reactor vessels larger than those currently operating or planned. However, additional studies and confirmation of manufacturing capabilities for such large reactor vessels will probably be needed.

The main item of reactor research and development for the prebreeder would be to determine the performance of the fuel. This fuel system is being evaluated for possible further development under the DOE AWBA program, including analytical work to extend the scope of present fuel-element modeling and irradiation testing.

Additional items of research, development, and demonstration would be related to the desirability of improving breeder fuel utilization by reducing the amount of Zircaloy in the seed-assembly and blanket-assembly shells and by reoptimizing the seed and blanket lattice. Reducing the Zircaloy structure would require mechanical design development, and modification of the lattice might require additional critical-heat-flux testing.

The research, development, and demonstration requirements for this concept are summarized in Table 2-18.

2.6.2 SPECIFIC CONCERNS NEEDING RESOLUTION

2.6.2.1 Minor Accidents

There is no identified reason to suppose that during LWBR operation the effects of minor accidents and the potential for radioactivity releases and public exposure would be significantly different from conventional PWRs. The conceptual prebreeder and breeder cores and a conventional core would have approximately the same power per module and the same number of fissions per module at refueling.

2.6.2.2 Major Accidents

"Major" accidents include potential fuel-coolant interactions in the seed-assembly region and recriticalities. Based on calculations of a hypothetical loss-of-coolant accident for the Shippingport LWBR, it is expected that satisfactory LOCA performance can be shown for commercial-scale LWBR cores with lattices and power densities similar to those of the Shippingport LWBR. These calculations show that there would be almost no hydrogen generation from the reaction of the cladding with water or steam. The use of fuel rods containing duplex pellets will have to be evaluated; preliminary calculations indicate that the performance of these rods during a LOCA event would be like that of conventional rods containing monolithic oxide pellets. Additional development will be required to confirm this.

Table 2-18. Technological advance requirements

Plant component	Operating experience evaluated	Prototype or production components being manufactured	No new knowledge required	Contemporary technology with modified configuration/application	Modest improvement in performance or size from available systems	Modest improvement in performance or size and modified configuration/application
Nuclear fuel	B[a]	B	B			P[b]
Reactivity-control systems		✓	✓			
Reactor vessel					✓	
Core-support structure					✓	
Reactor-vessel internals, including shielding, ducting, control-rod guides, baffles, etc.	✓	✓	✓	✓		
Primary-coolant pumps and auxiliary systems	✓	✓	✓			
Primary-coolant chemistry/ radiochemistry control	✓	✓	✓			
Primary-system heat exchangers	✓	✓	✓			
Reactor instrumentation	✓	✓	✓			
Emergency core-cooling/safe-shutdown systems	✓	✓	✓			
Containment, containment-cleanup systems, and effluent-control systems	✓	✓	✓			
Other accident-mitigating systems (i.e., plant-protection systems)	✓	✓	✓			
On-site fuel-handling and storage and shipping equipment				✓		
Main turbine	✓	✓	✓			
Other critical components, if any	✓	✓	✓			
Balance-of-plant components	✓	✓	✓			

[a]Operating data being obtained from LWBR.
[b]Irradiation test data presently being obtained for prebreeder fuel.

Abbreviations: B, breeder; P, prebreeder.

REFERENCES FOR CHAPTER 2

1. Babcock & Wilcox, Reference Safety Analysis Report, BSAR-205, NRC Docket STN 50-561.

2. U.S. Nuclear Regulatory Commission, Perkins Final Environmental Statement, NUREG-75/088, October 1975.

3. T. R. England, G. L. Hartfield, and R. K. Deremer, Xenon Spatial Stability in Large Seed-Blanket Reactors, WAPD-TM-606, Bettis Atomic Power Laboratories, April 1967.

Chapter 3

LIGHT-WATER BACKFIT PREBREEDER SUPPLYING ADVANCED BREEDER

3.1 OVERALL DESCRIPTION OF THE CONCEPT

The conceptual prebreeder reactor described in this chapter would use a core that could be backfitted into an existing pressurized water reactor (PWR) vessel without any changes to the plant. The core would operate in essentially the same manner as the PWR core and with a similar fuel-management strategy. However, while operating to produce power for generating electricity, this prebreeder would also generate significant quantities of uranium-233 which could be used in light-water breeder reactor (LWBR). The technology for this concept is essentially the same as that for current PWR cores, with the exception of the fuel system, which would consist of thorium dioxide pellets and uranium dioxide pellets within the same fuel rod. Uranium dioxide fuel is used in PWR cores, and thorium dioxide fuel is used in the LWBR core presently operating at the Shippingport Atomic Power Station.

Tables 3-1 through 3-4 list the parameters for the conceptual light-water prebreeder and the advanced breeder.

The advanced-breeder described here would constitute an advance over the LWBR core presently operating at the Shippingport Atomic Power Station because it would have less neutron-absorbing material in its structure, a more uniform power distribution, and a slightly higher average power density than does the Shippingport LWBR core. However, the operating characteristics would be very similar to those of the Shippingport LWBR in that reactivity during power operation would be controlled by means of movable fuel (lifting and lowering of fuel rods).

Because the power density would be lower than that of a conventional PWR core, the principal application for the advanced breeder described here would be in new plants specifically designed to accommodate the breeder concept. As the technology for this concept is very similar to that for the Shippingport LWBR core, the deployment of this concept would require only modest extensions of LWBR technology and would entail relatively few new licensing, safety, and environmental issues.

The advanced breeder would be self-sustaining. Once the prebreeder has produced sufficient uranium-233 to supply the in-core and out-of-core requirement for the breeder, the breeder would continue to operate without requiring new uranium fuel. An alternative to the present breeder concept would be to operate a breeder-like core as a high-gain converter. The converter core would differ from the breeder core in one or more of the following ways: lower fissile-material loading, higher power density, or higher fuel burnup. A high-gain converter would require small quantities of makeup uranium-233, whereas the breeder discussed here would not.

3.1.1 BACKFIT PREBREEDER

The present backfit prebreeder concept would be a 205-module core identical in mechanical design, hydraulic features, and power rating with a reference commercial PWR plant and core design--here taken to be the Babcock & Wilcox-205 design (Ref. 1). A cross section of the core is shown in Figure 3-1. The fuel, which would be contained in Zircaloy rods identical in outside dimensions to those used in the reference commercial PWR design, would be composed of thorium dioxide pellets alternating with duplex pellets. The thorium dioxide pellets and the duplex pellets would have essentially the same outer dimensions. The duplex pellets, shown in Figure 3-2, would consist of a uranium dioxide annulus with a cylindrical thorium dioxide center. The enrichment of the uranium would be close to 16%, and the uranium-235 content of the core would be about 5% of the total heavy-metal (uranium plus thorium) content. The purpose of this duplex pellet design is to permit the separation of uranium-233 bred in thorium from the uranium (primarily 235, 236, and 238) derived from the initial fissile charge.

The features of the fuel rods would lead to thermal characteristics that would be somewhat different from those in the standard commercial core; this point is discussed later. Reactivity control has been assumed to be the same as for the standard PWR. Soluble boron would be used for reactivity shim during normal operation; poison shutdown rods would also be provided. Calculations have indicated that the concept would operate acceptably as a backfit, with no change required in pumping power. Relative to the reference PWR, some changes might be required in the initial level of soluble boron in the coolant after a refueling and in the rate at which the boron is removed during operation at power, since the reactivity characteristics of the prebreeder concept during depletion would be slightly different from those of the reference PWR. Reactivity (temperature and power) coefficients might also be somewhat different from those of the reference PWR, necessitating minor changes in protection-system set points and operator actions during normal transients and under accident conditions. However, it is judged that all of these differences in core behavior could be accommodated without changes in the reactor plant. The reactor plant would therefore be identical with the reference PWR plant.

3.1.2 ADVANCED BREEDER

The advanced-breeder concept would be similar to the LWBR core in that the fuel would be a high-density, solid solution of uranium-233 dioxide and thorium dioxide, the lattice would be much drier than that of a reference PWR core, and reactivity would be controlled during operation at power by lifting and lowering movable fuel rods. Additional similarities to the Shippingport LWBR core would be the hexagonal modules, thorium dioxide blanket pellets at the top and bottom of the fuel rods, and a thorium dioxide reflecting blanket around the periphery of the core. One of the more significant differences between the present breeder concept and the LWBR core would be the configuration of the movable fuel. In the Shippingport LWBR core, the movable fuel consists of entire assemblies (seed assemblies) that can be lifted and lowered; these are surrounded by annular stationary blanket assemblies. In the present breeder concept, the movable fuel consists of individual rods containing only thorium dioxide and dispersed more or less uniformly throughout the core. These rods, called thorium dioxide fingers, would be arranged much like the fingers of poison rods now being used in commercial PWRs.

Another distinguishing feature of this concept is scram reactivity, which would be supplied by poison-finger shutdown rods similar to those used in the standard PWR. Soluble boron would be used for cold shutdown and to compensate for the initial buildup of fission products after startup. Thus, in the present concept the movable fuel (thorium dioxide fingers) would provide only the reactivity control required during operation at power--the reactivity to follow load changes and to compensate for long-term fission-product buildup. A typical example of the control sequence, starting with the plant cold after a refueling, would be as follows: Initially, the poison-finger rods would be fully inserted and the coolant would contain soluble boron. During plant heatup, most of the soluble boron would be gradually removed from the coolant. With the plant hot, criticality would be attained by complete withdrawal of the poison-finger rods and partial withdrawal of the thorium dioxide fingers. From this point on the movable fuel would be lifted to compensate for the buildup of long-lived fission products.

Features of the module arrangement for the advanced-breeder concept are shown in Figure 3-3, and features of the movable thorium dioxide finger control concept are shown in Figure 3-4. The core would consist of 157 hexagonal modules, approximately 12 inches across flats. This array of modules would be surrounded by 48 reflector blanket modules to form a nearly circular array. Each hexagonal module would contain fixed seed rods, movable thorium dioxide finger rods, and poison-finger shutdown rods. The thorium dioxide finger rods and poison-finger shutdown rods move within Zircaloy-4 guide tubes. The seed rods and the guide tubes would be located on a uniform triangular pitch.

3.1.3 ACCOMMODATION IN EXISTING PHYSICAL PLANTS

The conceptual prebreeder reactor would use a core that could be backfitted into an existing PWR reactor vessel without any changes to the plant.

Because the prebreeder-breeder described here has a lower power density than does a conventional PWR core, its principal application would be in new plants specifically designed to accommodate the breeder concept. In such a plant, however, only the reactor vessel, closure head and mechanisms would be significantly different from those in a reference plant. The containment, pumps, and pressurizer might have to be larger than those of the reference plant because of the larger core.

3.1.4 FUEL MANAGEMENT AND FUELING ALTERNATIVES

Fuel management for this concept would consist of replacing approximately one-third of the core annually (excluding peripheral breeder blanket assemblies). Fresh modules would be installed near the periphery of the core, while the most depleted modules would be removed from near the center of the core.

For the breeder concept, the possible fueling alternatives are as follows: reduction in fissile-material inventory, operation at higher power density, and operation to a higher fuel burnup. Any or all of these alternatives would reduce the conversion ratio, and breeding might no longer be achieved; the core would then operate as a high-gain converter. Relative to the operation of a breeder, a high-gain converter would in some cases reduce short-term (30-year) mining requirements but would in all cases increase long-term (100 years or more) mining requirements. A reduction in short-term requirements would result from a reduction in fissile-material loading.

3.1.5 FUEL CYCLES

The prebreeder would consist of a PWR-type core backfitted into a B&W 205 vessel. The fuel would consist of thorium dioxide pellets alternating with duplex pellets. The duplex pellets would consist of a uranium dioxide annulus around a cylindrical thorium dioxide center. Reactivity control would be the same as that in a standard PWR.

The advanced breeder would use a PWR-type vessel extrapolated to a larger size from the present commercial PWRs. The fuel would consist of a high-density solid solution of uranium and thorium dioxides. Reactivity control would be achieved by movable thorium dioxide control rods. At equilibrium the advanced breeder would produce a small surplus of uranium-233 (U(3)).

3.1.5.1 Backfit Prebreeder MEU(5)-Th, U(5) Recycle

The prebreeder would utilize 15.4% enriched uranium-235 (U(5)) thorium dioxide duplex fuel. The spent duplex fuel would first be reprocessed to recover the uranium-235 from the uranium dioxide annulus and would then be reprocessed to recover the bred uranium-233 from the thorium dioxide. The recovered uranium-235 would be recycled to fuel fabrication, the plutonium and bred uranium-233 would be sent to a secure storage center and the thorium would be sent to 10-year interim storage. The flow diagram is shown in Figure 3-5.

3.1.5.2 Advanced Breeder HEU(3)-Th

The advanced breeder would use a large extrapolated-PWR-type vessel modified for a tight-lattice, hexagonal fuel bundle and thorium dioxide control rods. The fuel would consist of binary solid solution of highly enriched uranium (82%) and thorium dioxides in the form of pellets. The spent fuel would be reprocessed to recover the uranium-233 which would be recycled to remote fuel fabrication. The flow diagram is shown in Figure 3-6.

3.1.5.3 Quantitative Fuel Inventories

Table 3-5 summarizes the overall fuel-management information, including separative-work requirements typical of this concept. Table 3-6 shows the calculated isotopic content of cycle 29 of a typical prebreeder core, Table 3-7 shows the calculated isotopic content of the breeder equilibrium cycle, and Tables 3-8 and 3-9 show the calculated overall isotopic charge/discharge data for a typical system over its assumed 30-year history. Figures 3-5 and 3-6 indicate the isotopic mass flows for this prebreeder and breeder concept if scaled to 1,000 MWe. The isotopic masses of Tables 3-8 and 3-9 have been divided by the factors 1.295 and 1.035, respectively, to obtain the values shown in Figures 3-5 and 3-6. The prebreeder would be rated at 1,295 MWe, and the breeder at 1,035 MWe (Table 3-1).

3.1.5.4 Fuel Reprocessing and Refabrication

The reprocessing of LWBR fuel would be similar to the reprocessing of light-water reactor (LWR) fuel in that high-grade fissile material of high toxicity would be generated and handled, although the fission-product concentration in the breeder fuel would be lower than that in the LWR fuel. The uranium-233 recovered from LWBR fuel would contain about 2,500 to 4,500 ppm of uranium-232 whose daughter products emit a more penetrating radiation than do the transuranic isotopes of the LWR fuel cycle and would require more highly automated and shielded fabrication equipment for breeder fuel. Fabrication equipment for prebreeder fuel would be little different from that of LWR fuel. The penetrating radiation from the uranium-232 accompanying uranium-233 would present difficulty in fabrication, and it would also deter diversion.

Table 3-1. Generalized performance specifications: light-water backfit prebreeder concept supplying advanced breeder concept

Parameter	Backfit prebreeder	Advanced breeder
Power plant performance parameters		
Reactor thermal power output, MWt	3,800	2,900
Net electrical power output, MWe	1,295	1,035
Plant heat rate, Btu/kW-hr	9,990	9,570
Core performance parameters		
Core heat output, MWt	3,800	2,900
Core volume, liters[a]	35,400	52,300
Core loading (first core), kg		
Heavy metal[a]	89,800	235,000
Fissile fuel	3,680	4,498
Conversion ratio, cycle average	0.48[b]	--
Average for initial cycle	--	~1.05
Average for equilibrium cycle	--	~1.02
Fissile inventory ratio	0.68[b]	--
Initial cycle	--	~1.03
Equilibrium cycle	--	~1.01
Average discharge burnup, MWd/MTHM[c,a]	34,800	10,100
Peak discharge burnup MWd/MTHM[c,a]	54,600	~26,000
Fuel type	Alternating ThO_2 and duplex pellets	Binary UO_2ThO_2
Reactor inlet temperature, °F	569	576
Reactor outlet temperature, °F	626	628

[a]Excluding axial and radial reflectors.
[b]Includes fissile plutonium production.
[c]Heavy metal charged.

Table 3-2. Reactor concept data: light-water backfit prebreeder concept supplying advanced breeder concept

Parameter	Backfit prebreeder	Advanced breeder[a]
Geometric information		
Core height, cm	363	366
Number of core enrichment zones	2	1
Number of assemblies	205	157
Equivalent diameter, cm	353	427
Number of pins per assembly[b]	264	288/99
Pin pitch-to-diameter ratio[b]	1.32	1.110/1.196
Overall assembly length, cm	449	497
Lattice pitch, cm	1.28	1.61
Assembly material	Grid material, Inconel; guide tubes, Zircaloy-4	Machined Zircaloy-4 grids
Cladding parameters		
Cladding outside diameter, mils[b]	379	571/530
Cladding wall thickness, mils[b]	23.5	28/35
Cladding material	Annealed Zircaloy-4	Zircaloy-4
Beginning-of-cycle fissile inventory, kg[c]	3,855	5,888
External fissile inventory, kg[d,e]	1,450/2,900	1,981/3,962
Fissile gain or loss, kg/cycle[d]	462 (loss)	24.5 (gain)
Specific power, MW/kg fissile[f]	993	645
Specific power, MW/kg HM	42	12.4
Power density, kW/liter	107	55.5

[a]Excluding axial and radial reflectors.

[b]Pairs of values indicate seed rods/blanket rods.

[c]"Beginning-of-cycle" means beginning of cycle 29 in the prebreeder and beginning of an equilibrium cycle in the breeder.

[d]Prebreeder value includes uranium-235 makeup requirement. Breeder value for equilibrium cycle.

[e]Inventories are for 1-year out-of-core time/2-year out-of-core time.

[f]Breeder value for initial cycle.

Table 3-3. Fuel-assembly volume fractions: light-water backfit prebreeder concept supplying advanced breeder concept

	Backfit prebreeder			Advanced breeder fuel assembly	
	Assembly type I		Assembly type 2	Safety rods	Safety rods
Component	Control out	Control in	Control out	out	in
Fuel[a,b]	0.295	0.2[illegible]	0.295	0.417[c] 0.114[d]	0.417[c] 0.114[d]
Coolant[e]	0.585	0.549	0.585	0.285	0.269
Structure	0.120	0.120	0.120	0.184	0.187
Control	0.0000	0.036	0.0	--	0.013
Total	1.000	1.000	1.000	1.000	1.000

[a]Includes pellet volume only.

[b]Includes coolant between modules.

[c]Seed assembly.

[d]Blanket assembly.

[e]In prebreeder reactivity control obtained via soluble boron and boron shutdown rods. In breeder control obtained via movable thoria fingers.

Table 3-4. Core-region volume fractions: light-water backfit prebreeder concept supplying advanced breeder[a] concept

Component	Backfit prebreeder	Advanced breeder
Fuel[b]	0.295	0.417[c] 0.087[d]
Coolant[e]	0.585	0.321
Structure	0.120	0.175
Total	1.000	1.000

[a]Excluding axial and radial reflectors.

[b]Includes pellet volume only.

[c]Seed assembly.

[d]Blanket assembly.

[e]Includes coolant between modules.

Table 3-5. Fuel-management information: light-water backfit prebreeder concept supplying advanced breeder[a] concept

Parameter	Prebreeder		Breeder
Approximate fraction of core replaced	1/3		1/3
Lag time assumed between fuel discharge and recycle reload[b]	1 yr/2 yr		1 yr/2 yr
Fissile-material loss fractions			
Conversion loss fraction	0.01		(e)
Fabrication loss fraction	0.005		(e)
Reprocessing loss fraction	0.01		(e)
Fuel requirements, ST/GWe	U_3O_8	ThO_2	ThO_2
Initial core	662	61	391
Cycle 15 or equilibrium reload requirement[b,c,d,e]	178	0.95	2.4
30-year cumulative requirements	4,236/5,235	277/283	286/404
50-year cumulative requirements	4,236/5,235	277/283	349/456
100-year cumulative requirements	4,236/5,235	277/283	471/578
Separative-work requirements, 10^3 SWU/GWe[f]			
Initial core	572		0
Equilibrium reload	167		0
30-year cumulative requirement[b,d]	3,972/4,912		0
50-year cumulative requirement	3,972/4,912		0
100-year cumulative requirement	3,972/4,912		0

[a]Average plant capacity factor is assumed to be 75%.

[b]Cumulative time shown starts with initial prebreeder operation for both prebreeder and breeder; i.e., breeder operation is initiated after prebreeder has produced sufficient uranium-233 to fuel the breeder for its initial load and as many reloads as required during out-of-core time.

[c]Assumes thorium dioxide out-of-core time is 10 years during prebreeder operation and 1 or 2 years for breeder operation. At the end of prebreeder operation nearly all the thorium dioxide mined for prebreeder use is available for recycle into a breeder reactor. Breeder thorium dioxide requirements are shown independent of this thorium dioxide source.

[d]Cumulative requirements are for 1-year out-of-core time/2-year out-of-core time.

[e]Fabrication and reprocessing losses are assumed to be 1% each for first 40 years of operation and 0.5% thereafter, reflecting improved recycle technology in later-generation plants.

[f]Uranium hexafluoride conversion losses are assumed to occur during the conversion of uranium dioxide to uranium hexafluoride. No losses are assumed for reconversion from uranium hexafluoride to uranium dioxide.

Table 3-6. Fuel inventory for cycle 29: light-water backfit prebreeder concept[a]

Isotope	Quantity							
	Zone 1		Zone 2		Zone 3		Zone 4	
	BOC	EOC	BOC	EOC	BOC	EOC	BOC	EOC
Thorium-232	20,590.8	20,458.8	20,458.8	20,229.0	20,229.0	20,025.1	302.2	299.4
Protactinium-233	0.0	18.7	1.8	30.6	2.9	27.0	0.0	0.5
Uranium-232	0.0	0.1	0.1	0.5	0.5	1.0	0.0	0.0
Uranium-233	0.0	93.2	110.1	218.2	245.9	296.5	0.0	2.2
Uranium-234	0.0	3.8	3.8	17.5	17.5	32.0	0.0	0.2
Uranium-235	1,439.8	1,148.8	1,148.1	783.2	783.2	564.1	10.9	6.8
Uranium-236	342.7	381.3	377.2	426.4	426.4	445.2	0.0	0.8
Uranium-238	7,476.4	7,412.3	7,412.9	7,299.6	7,299.6	7,196.5	129.9	128.0
Plutonium-239	0.0	38.0	38.1	63.4	63.4	69.4	0.0	0.9
Plutonium-240	0.0	5.0	5.1	15.7	15.7	22.4	0.0	0.2
Plutonium-241	0.0	2.2	2.2	12.2	12.2	20.1	0.0	0.1
Plutonium-242	0.0	0.1	0.1	2.1	2.1	5.5	0.0	0.0

[a]Abbreviations: BOC, beginning of cycle; EOC, end of cycle.

Table 3-7. Light-water advanced breeder concept:[a] fuel inventory for equilibrium cycle

Isotope	Quantity (kg)					
	Zone 1		Zone 2		Zone 3	
	BOEC	EOEC	BOEC	EOEC	BOEC	EOEC
Thorium-232	106,715.7	106,410.8	104,403.0	104,118.6	104,118.6	103,845.9
Protactinium-233	0.0	34.4	11.5	33.6	11.4	33.6
Uranium-232	4.1	4.2	4.1	4.1	4.1	4.1
Uranium-233	1,686.7	1,667.5	1,658.3	1,642.4	1,664.6	1,641.9
Uranium-234	721.9	728.4	714.7	717.4	717.4	717.1
Uranium-235	290.3	292.9	287.4	288.5	288.5	288.3
Uranium-236	196.8	198.6	194.9	195.6	195.6	195.5
Fission products	0.0	280.2	274.9	549.8	549.8	824.8

[a]Abbreviations: BOEC, beginning-of-equilibrium cycle; EOEC, end-of-equilibrium cycle.

Table 3-8. Reactor charge data

Year	Reactor charge (kg)							
	Th-232	U-232	U-233	U-234	U-235	U-236	U-238	Total
0	62,073.3	0.0	0.0	0.0	3,680.0	0.0	24,046.0	89,799.3
1	20,893.0	0.0	0.0	0.0	1,395.9	0.0	7,661.1	29,950.0
2	20,893.0	0.0	0.0	0.0	1,395.9	0.0	7,661.1	29,950.0
3	20,893.0	0.0	0.0	0.0	1,400.6	29.4	7,656.4	29,979.4
4	20,893.0	0.0	0.0	0.0	1,409.1	82.2	7,647.9	30,032.2
5	20,893.0	0.0	0.0	0.0	1,416.2	126.7	7,640.8	30,076.7
6	20,893.0	0.0	0.0	0.0	1,416.2	126.7	7,640.8	30,076.7
7	20,893.0	0.0	0.0	0.0	1,416.2	126.7	7,640.8	30,076.7
8	20,893.0	0.0	0.0	0.0	1,419.4	146.7	7,637.6	30,096.7
9	20,893.0	0.0	0.0	0.0	1,425.1	182.6	7,631.9	30,132.6
10	20,893.0	0.0	0.0	0.0	1,430.0	212.9	7,627.0	30,162.9
11	20,893.0	0.0	0.0	0.0	1,430.0	212.9	7,627.0	30,162.9
12	20,893.0	0.0	0.0	0.0	1,430.0	212.9	7,627.0	30,162.9
13	20,893.0	0.0	0.0	0.0	1,432.1	226.5	7,624.9	30,176.5
14	20,893.0	0.0	0.0	0.0	1,436.0	250.9	7,621.0	30,200.9
15	20,893.0	0.0	0.0	0.0	1,439.3	271.5	7,617.7	30,221.5
16	20,893.0	0.0	0.0	0.0	1,439.3	271.5	7,617.7	30,221.5
17	20,893.0	0.0	0.0	0.0	1,439.3	271.5	7,617.7	30,221.5
18	20,893.0	0.0	0.0	0.0	1,440.8	280.7	7,616.2	30,230.7
19	20,893.0	0.0	0.0	0.0	1,443.5	297.3	7,613.5	30,247.3
20	20,893.0	0.0	0.0	0.0	1,445.7	311.3	7,611.3	30,261.3
21	20,893.0	0.0	0.0	0.0	1,445.7	311.3	7,611.3	30,261.3
22	20,893.0	0.0	0.0	0.0	1,445.7	311.3	7,611.3	30,261.3
23	20,893.0	0.0	0.0	0.0	1,446.7	317.6	7,610.3	30,266.9
24	20,893.0	0.0	0.0	0.0	1,448.5	328.9	7,608.5	30,278.9
25	20,893.0	0.0	0.0	0.0	1,450.0	338.4	7,607.0	30,288.4
26	20,893.0	0.0	0.0	0.0	1,450.0	338.4	7,607.0	30,288.4
27	20,893.0	0.0	0.0	0.0	1,450.0	338.4	7,607.0	30,288.4
28	20,893.0	0.0	0.0	0.0	1,450.7	342.7	7,606.3	30,292.7
29	20,893.0	0.0	0.0	0.0	1,452.0	350.4	7,605.0	30,300.4

Table 3-8. Reactor charge data (continued)

Year	Reactor charge (kg)							
	Th-232	U-232	U-233	U-234	U-235	U-236	U-238	Total
29.3	319,730.0	18.2	4,437.9	422.4	60.0	4.4	0.0	324,673.1
30.3	107,934.3	6.2	1,498.2	142.6	20.3	1.5	0.0	109,603.0
31.3	107,934.3	6.2	1,498.2	142.6	20.3	1.5	0.0	109,603.0
32.3	107,901.1	6.1	1,502.2	159.9	31.4	2.6	0.0	109,603.4
33.3	107,868.1	6.0	1,506.1	177.1	42.6	3.7	0.0	109,603.7
34.3	107,835.4	6.0	1,510.0	194.3	53.6	4.8	0.0	109,604.0
35.3	107,835.4	6.0	1,510.0	194.3	53.6	4.8	0.0	109,604.0
36.3	107,835.4	6.0	1,510.0	194.3	53.6	4.8	0.0	109,604.0
37.3	107,796.6	5.9	1,518.9	218.0	59.4	5.5	0.0	109,604.3
38.3	107,757.8	5.8	1,527.9	241.9	65.0	6.2	0.0	109,604.7
39.3	107,719.0	5.7	1,536.8	265.9	70.6	6.9	0.0	109,605.0
40.3	107,719.0	5.7	1,536.8	265.9	70.6	6.9	0.0	109,605.0
41.3	107,719.0	5.7	1,536.8	265.9	70.6	6.9	0.0	109,605.0
42.3	107,680.3	5.6	1,545.5	290.0	76.3	7.6	0.0	109,605.3
43.3	107,641.7	5.6	1,554.1	314.0	81.9	8.3	0.0	109,605.6
44.3	107,604.5	5.5	1,561.9	337.2	87.8	9.1	0.0	109,606.0
45.3	107,604.5	5.5	1,561.9	337.2	87.8	9.1	0.0	109,606.0
46.3	107,604.5	5.5	1,561.9	337.2	87.8	9.1	0.0	109,606.0
47.3	107,572.8	5.4	1,566.6	356.3	94.5	10.6	0.0	109,606.3
48.3	107,541.1	5.3	1,571.3	375.3	101.3	12.1	0.0	109,606.5
49.3	107,509.8	5.3	1,575.9	394.1	108.0	13.7	0.0	109,606.8

Table 3-9. Reactor discharge data

	Reactor discharge (kg)													
Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Fission products
1	20,758.1	23.3	0.3	90.0	3.9	786.8	36.7	7,938.3	40.9	5.5	2.4	0.2	29,686.1	380
2	20,528.4	25.0	0.7	220.4	17.6	666.6	102.8	7,626.5	66.0	16.3	12.6	2.1	29,285.0	770
3	20,324.5	27.5	1.0	298.7	32.2	516.8	158.4	7,376.6	70.8	22.8	20.3	5.5	28,855.1	1,150
4	20,324.5	27.5	1.0	298.7	32.2	516.8	158.4	7,376.6	70.8	22.8	20.3	5.5	28,855.1	1,150
5	20,324.5	27.5	1.0	298.7	32.2	516.8	158.4	7,376.6	70.8	22.8	20.3	5.5	28,855.1	1,150
6	20,324.5	27.5	1.0	298.7	32.2	521.5	183.4	7,372.0	70.8	22.8	20.3	5.5	28,880.2	1,150
7	20,324.5	27.5	1.0	298.7	32.2	530.0	228.3	7,363.8	70.7	22.8	20.3	5.5	28,925.3	1,150
8	20,324.5	27.5	1.0	298.7	32.2	537.1	266.1	7,357.0	70.6	22.7	20.3	5.5	28,963.2	1,150
9	20,324.5	27.5	1.0	298.7	32.2	537.1	266.1	7,357.0	70.6	22.7	20.3	5.5	28,963.2	1,150
10	20,324.5	27.5	1.0	298.7	32.2	537.1	266.1	7,357.0	70.6	22.7	20.3	5.5	28,963.2	1,150
11	20,324.5	27.5	1.0	298.7	32.2	540.3	283.1	7,353.9	70.6	22.7	20.3	5.5	28,980.3	1,150
12	20,324.5	27.5	1.0	298.7	32.2	546.0	313.6	7,348.4	70.6	22.7	20.3	5.5	29,011.0	1,150
13	20,324.5	27.5	1.0	298.7	32.2	550.9	339.4	7,343.7	70.5	22.7	20.2	5.5	29,036.8	1,150
14	20,324.5	27.5	1.0	298.7	32.2	550.9	339.4	7,343.7	70.5	22.7	20.2	5.5	29,036.8	1,150
15	20,324.5	27.5	1.0	298.7	32.2	550.9	339.4	7,343.7	70.5	22.7	20.2	5.5	29,036.8	1,150
16	20,324.5	27.5	1.0	298.7	32.2	553.0	350.9	7,341.7	70.5	22.7	20.2	5.5	29,048.4	1,150
17	20,324.5	27.5	1.0	298.7	32.2	556.9	371.7	7,337.9	70.5	22.7	20.2	5.5	29,069.3	1,150
18	20,324.5	27.5	1.0	298.7	32.2	560.2	389.2	7,334.8	70.4	22.7	20.2	5.5	29,086.9	1,150
19	20,324.5	27.5	1.0	298.7	32.2	560.2	389.2	7,334.8	70.4	22.7	20.2	5.5	29,086.9	1,150
20	20,324.5	27.5	1.0	298.7	32.2	560.2	389.2	7,334.8	70.4	22.7	20.2	5.5	29,086.9	1,150
21	20,324.5	27.5	1.0	298.7	32.2	561.7	397.0	7,333.3	70.4	22.7	20.2	5.5	29,094.7	1,150
22	20,324.5	27.5	1.0	298.7	32.2	564.4	411.1	7,330.7	70.4	22.7	20.2	5.5	29,108.9	1,150
23	20,324.5	27.5	1.0	298.7	32.2	566.6	423.0	7,328.6	70.4	22.7	20.2	5.5	29,120.9	1,150
24	20,324.5	27.5	1.0	298.7	32.2	566.6	423.0	7,328.6	70.4	22.7	20.2	5.5	29,120.9	1,150
25	20,324.5	27.5	1.0	298.7	32.2	566.6	423.0	7,328.6	70.4	22.7	20.2	5.5	29,120.9	1,150
26	20,324.5	27.5	1.0	298.7	32.2	567.6	428.4	7,327.6	70.4	22.7	20.2	5.5	29,126.3	1,150
27	20,324.5	27.5	1.0	298.7	32.2	569.4	438.0	7,325.9	70.3	22.6	20.2	5.5	29,135.8	1,150
28	20,324.5	27.5	1.0	298.7	32.2	570.9	446.0	7,324.5	70.3	22.6	20.2	5.5	29,143.9	1,150
29	20,324.5	27.5	1.0	298.7	32.2	570.9	446.0	7,324.5	70.3	22.6	20.2	5.5	29,143.9	1,150
29.3	61,711.8	62.6	0.8	408.7	29.1	3,089.7	1,208.0	22,230.1	132.2	27.1	19.9	3.3	88,923.3	750.0
30.3	107,604.8	37.8	6.2	1,480.0	161.6	31.8	2.6	0.0	0.0	0.0	0.0	0.0	109,324.8	279.9
31.3	107,294.1	37.6	6.1	1,488.2	179.1	42.9	3.7	0.0	0.0	0.0	0.0	0.0	109,051.7	554.5
32.3	106,993.7	37.4	6.0	1,487.3	195.5	53.7	4.8	0.0	0.0	0.0	0.0	0.0	108,778.5	829.1
33.3	106,993.1	37.4	6.0	1,487.4	195.9	53.9	4.8	0.0	0.0	0.0	0.0	0.0	108,778.5	829.1
34.3	106,992.4	37.4	6.0	1,487.5	196.2	54.1	4.8	0.0	0.0	0.0	0.0	0.0	108,778.5	829.1
35.3	106,952.9	37.3	5.9	1,496.9	220.2	60.0	5.5	0.0	0.0	0.0	0.0	0.0	108,778.8	829.2
36.3	106,914.1	37.1	5.9	1,506.2	244.0	65.6	6.2	0.0	0.0	0.0	0.0	0.0	108,779.1	829.2
37.3	106,875.4	37.0	5.8	1,515.3	267.8	71.2	6.9	0.0	0.0	0.0	0.0	0.0	108,779.4	829.3
38.3	106,874.6	37.0	5.8	1,515.5	268.2	71.3	7.0	0.0	0.0	0.0	0.0	0.0	108,779.4	829.3
39.3	106,873.9	37.0	5.8	1,515.7	268.7	71.4	7.0	0.0	0.0	0.0	0.0	0.0	108,779.4	829.3
40.3	106,834.6	36.9	5.7	1,524.8	293.0	77.1	7.7	0.0	0.0	0.0	0.0	0.0	108,779.7	829.3
41.3	106,796.1	36.8	5.6	1,533.6	316.9	88.7	8.4	0.0	0.0	0.0	0.0	0.0	108,780.0	829.3
42.3	106,758.4	36.7	5.5	1,542.0	340.0	88.5	9.2	0.0	0.0	0.0	0.0	0.0	108,780.3	829.3
43.3	106,757.7	36.7	5.5	1,542.1	340.4	88.6	9.2	0.0	0.0	0.0	0.0	0.0	108,780.3	829.3
44.3	106,756.9	36.7	5.5	1,542.3	340.9	82.7	9.2	0.0	0.0	0.0	0.0	0.0	108,780.3	829.3
45.3	106,721.7	36.6	5.5	1,549.7	360.7	95.7	10.8	0.0	0.0	0.0	0.0	0.0	108,780.6	829.4
46.3	106,687.1	36.5	5.4	1,556.8	380.2	102.6	12.3	0.0	0.0	0.0	0.0	0.0	108,780.9	829.4
47.3	108,653.7	36.4	5.3	1,563.2	399.3	109.4	13.8	0.0	0.0	0.0	0.0	0.0	108,781.2	829.4
48.3	106,653.1	36.4	5.3	1,563.3	399.7	109.5	13.8	0.0	0.0	0.0	0.0	0.0	108,781.2	829.4
49.3	106,652.5	36.4	5.3	1,563.4	400.0	109.7	13.9	0.0	0.0	0.0	0.0	0.0	108,781.2	829.4
50.3	106,624.0	36.3	5.3	1,567.4	415.8	116.5	16.2	0.0	0.0	0.0	0.0	0.0	108,781.4	829.4

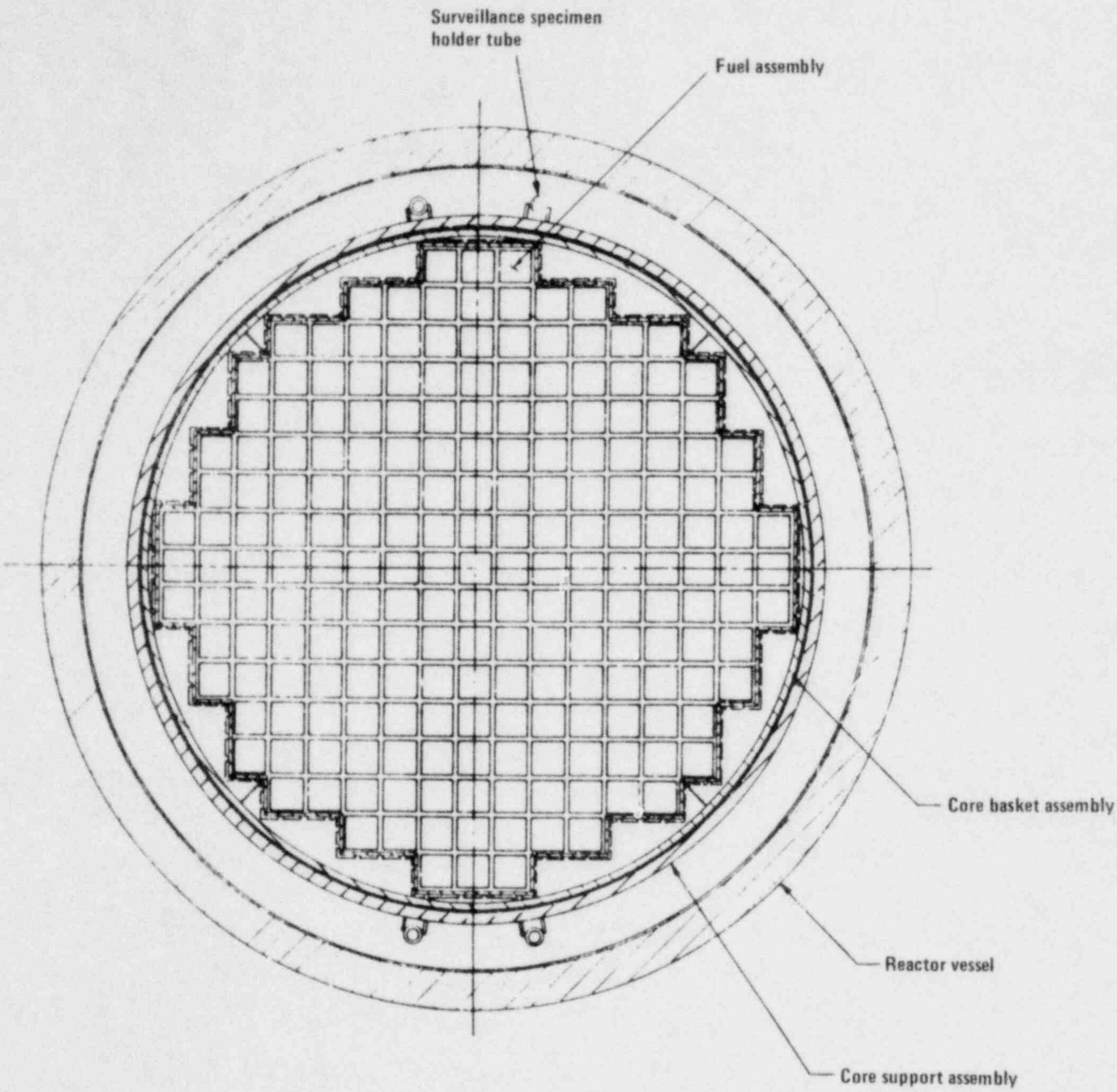


Figure 3-1. Prebreeder backfit reactor vessel and internals cross section.

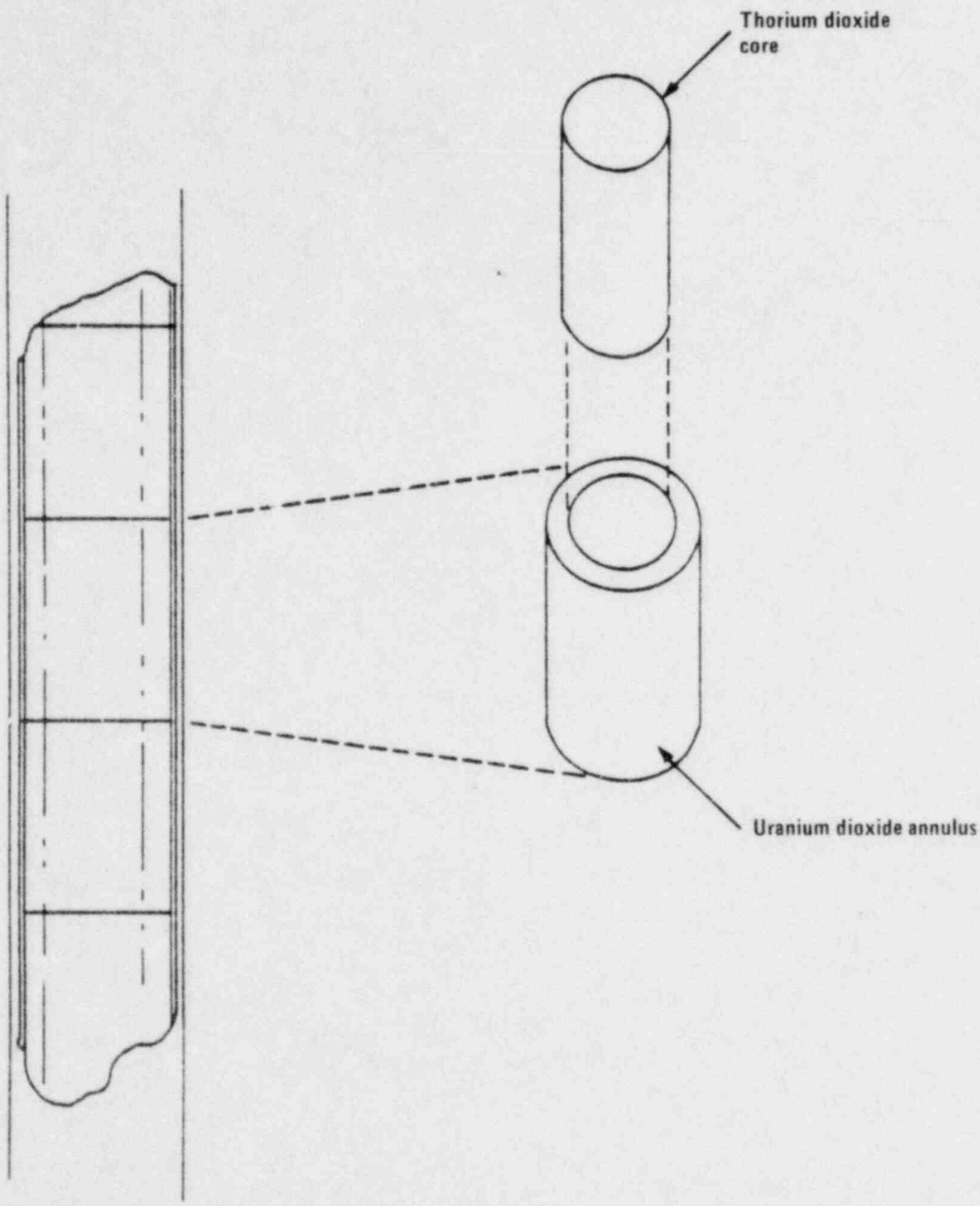


Figure 3-2. Duplex fuel pellets.

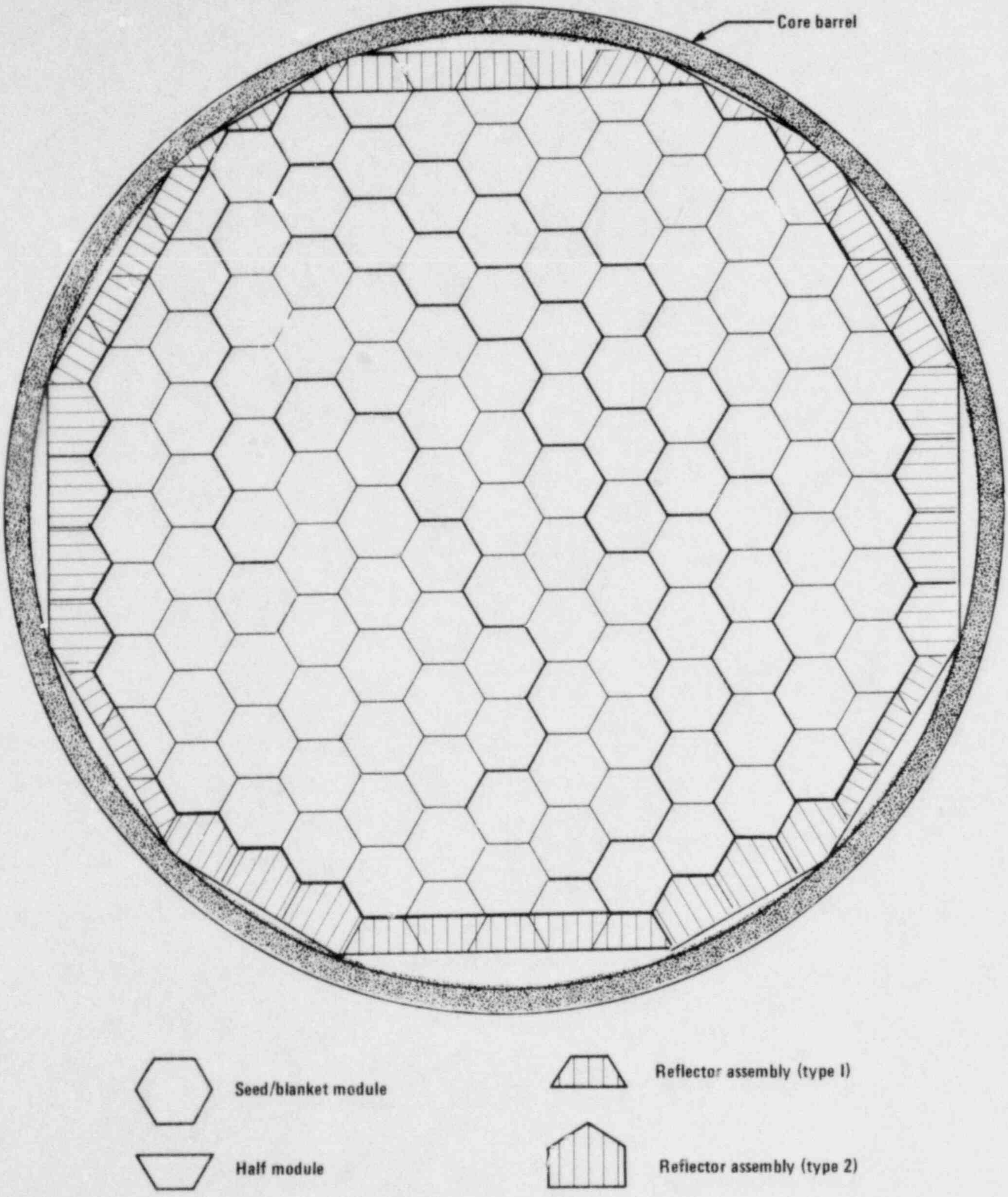


Figure 3-3. Advanced breeder full core cross section.

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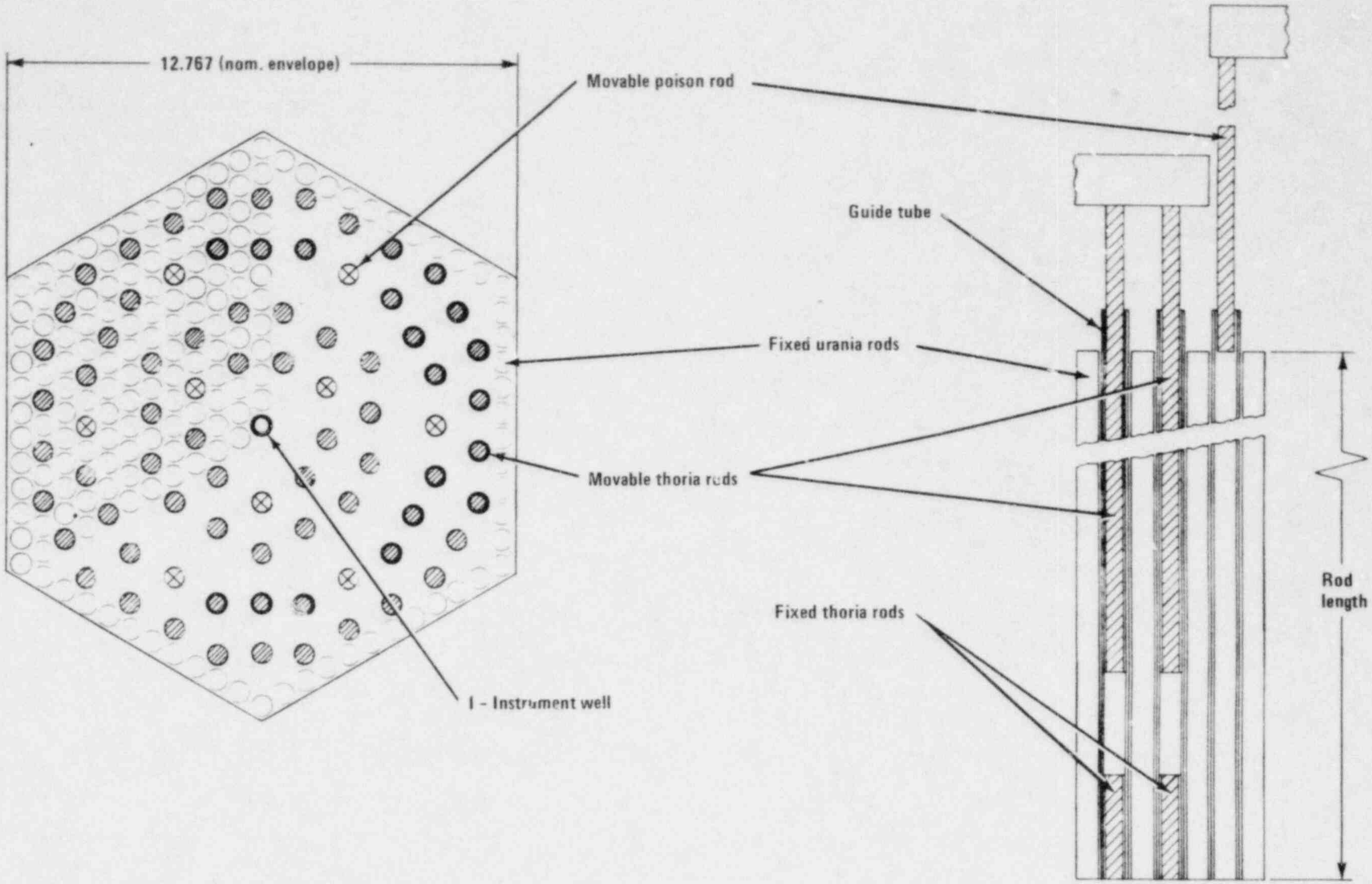
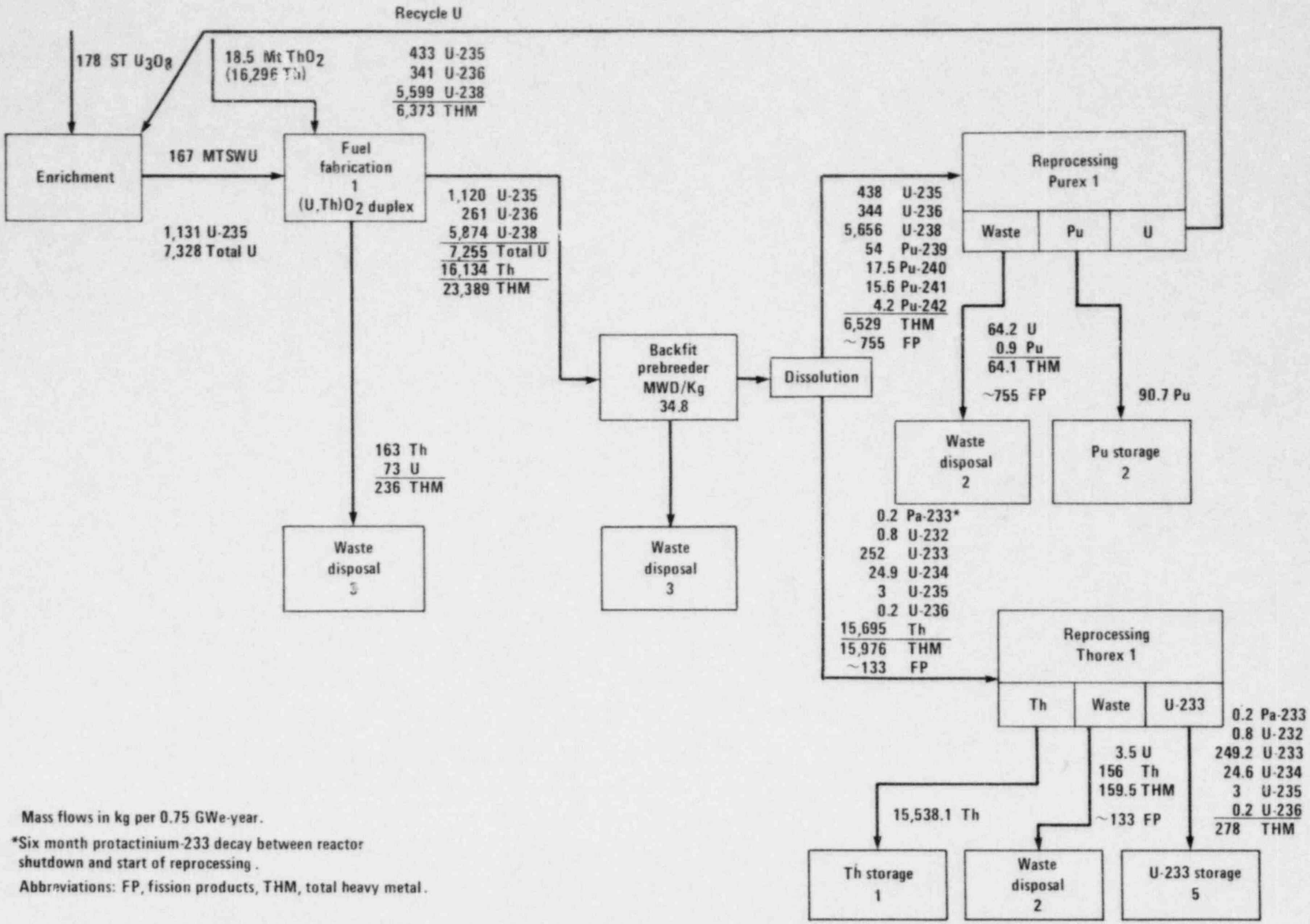


Figure 3-4. Advanced breeder movable-finger control concept.

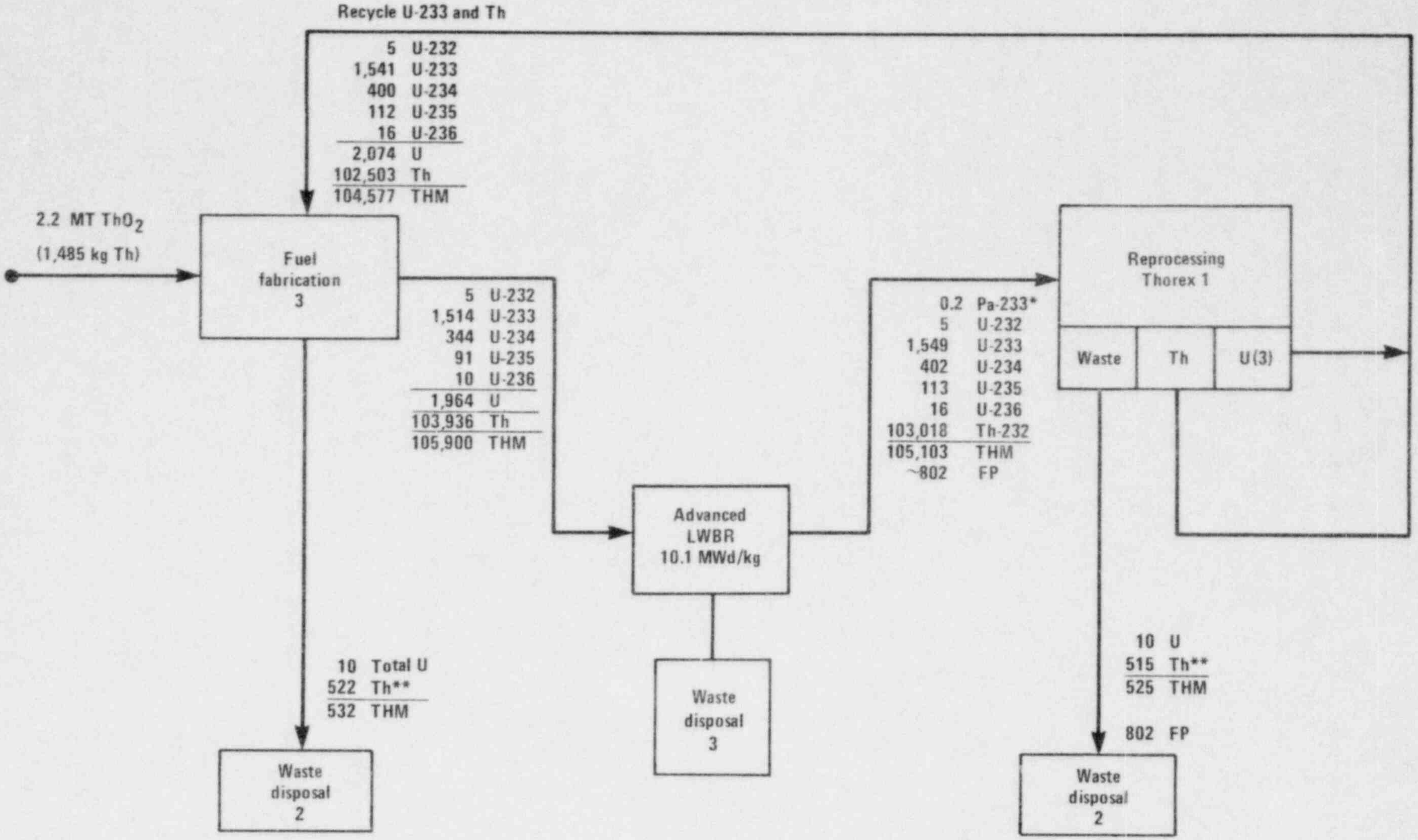


Mass flows in kg per 0.75 GWe-year.

*Six month protactinium-233 decay between reactor shutdown and start of reprocessing.

Abbreviations: FP, fission products, THM, total heavy metal.

Figure 3-5. Typical LWBR material-flow diagram MEU(5)-Th, U(5) recycle, backfit prebreeder (PWR).



Mass flows in kg per 0.75 GWe-year.

*Six-month protactinium-233 decay between reactor shutdown and start of reprocessing.

**Fabrication and reprocessing losses 0.5% each, reflecting improved recycle technology in later generation plants.

Abbreviations: FP, fission products; THM, total heavy metal.

Figure 3-6. Typical LWBR material-flow diagram HEU(3)-Th, recycle U(3) advanced breeder (PWR).

3.2 FUEL MECHANICAL, NUCLEAR, AND THERMAL-HYDRAULIC CONSIDERATIONS

This section identifies the principal unique mechanical, nuclear, and thermal-hydraulic features of the prebreeder and breeder concepts. Preliminary reviews indicate that the design and manufacture of reactors with these features may be feasible with the use of existing design and testing methods, existing manufacturing capabilities, and proved materials. Additional testing, however, may be required.

3.2.1 MECHANICAL CONSIDERATIONS

3.2.1.1 Prebreeder Concept

This concept would be identical with the reference PWR except for the fuel pellets within the fuel rods. Relative to the fuel pellets in the reference PWR, the fuel pellets in the present prebreeder concept could have a different behavior (with respect to thermal expansion, densification, growth, chipping, and cracking) during power cycling and irradiation. In addition, the peak fission density in the uranium dioxide (fissions per cubic centimeter of uranium dioxide) would be higher than it is in the reference PWR. Calculations have indicated it may be possible to choose fuel-pellet parameters (density, edge chamfer, end dishing) and other parameters (pellet-to-cladding gap, internal rod pressure, and startup rate limitations) so as to ensure that the axial and radial forces on the fuel-rod cladding would be acceptable. Extensive irradiation testing and design would be required to determine the acceptability of using this fuel concept; this would require many years. Other than for the specification of those parameters, there are no special mechanical considerations for this concept.

3.2.1.2 Breeder Concept

The breeder concept would have a dry lattice relative to the reference PWR. In order to avoid rod-to-rod contact and insure adequate cooling, the fuel rods must be placed on a triangular pitch, which can be accommodated with a minimum amount of structure in a hexagonal module. Because of the drier lattice, the weight density (weight per square foot of radial area) would be higher, the core pressure drop would be higher, and the hydraulic lifting pressure would be higher than in the reference PWR.

The fuel would be in the form of cylindrical ceramic pellets inside Zircaloy-4 rods. The pellets would be either a high-density binary solid solution of uranium dioxide and thorium dioxide or pure high-density thorium dioxide. For the seed rods, the diameters of the fuel-rod cladding and the pellets would be the same as those for the blanket rods of the Shippingport LWBR core. For the movable thorium dioxide finger rods, the diameters of the fuel-rod cladding and the pellets would be slightly smaller than those of the seed rods; this smaller diameter provides clearance for rod movement and the flow of coolant inside the guide tubes.

There would be two types of movable rods: poison rods and thorium dioxide finger rods. As uranium-233 is bred in the thorium dioxide finger rods, the rods would begin to produce heat. Thus, the thorium dioxide finger guide tubes would be heated while the poison-rod guide tubes would not, and a differential thermal expansion would develop between the guide tubes. The design of the thorium dioxide finger rods, and guide tubes has to account for the fissioning of bred uranium-233 in the thorium dioxide and the effects of thermal expansion and fuel growth. The growth of the thorium dioxide and the resulting stress on fuel rods are expected to be different from those in the reference PWR.

Individual guide tubes are assumed to be used in this assembly to protect and guide the thorium dioxide control rods. The central sheath would be an asterisk-shaped hollow column enclosing the poison-finger shutdown rods and their interconnecting "spider." The guide tubes and sheath would be attached to base plates at both ends to form a sturdy assembly and to protect the control rods from coolant crossflow in the outlet-plenum region and to insure smooth insertion of the control rods into the the fuel assembly.

In addition to the guide tubes for the thorium dioxide finger rods and the poison finger shutdown rods, each fuel assembly would contain one instrument guide tube. The grids, which would be similar to those used in the Shippingport LWBR core, supply lateral support and alignment for the fuel rods and guide tubes, and would be attached to the poison-rod guide tubes and the instrument guide tube. The grids for the present concept would be Zircaloy-4, in contrast with the AM-350 grids in the Shippingport LWBR core. The poison-rod and instrument guide tubes would provide the primary vertical structure of the assembly and support the axial loads occurring during core operation.

The number and wall thickness of the poison-rod guide tubes and the wall thickness of the instrument guide tube are variables that can be adjusted on the basis of the calculated loads generated by the fuel assembly during core operation.

The control-rod-drive mechanisms must provide two functions. The first is complete withdrawal and scram of the poison-finger shutdown rods; the second is lifting and lowering of the thorium dioxide finger rods.

In the advanced breeder concept there would be one control-rod-drive mechanism above each module. In one preliminary concept of the control system, the control-rod-drive mechanism would have independently actuated concentric lead screws. The central lead screw would operate the poison-finger shutdown rods, and the annular lead screw would operate the thorium dioxide finger rods. A control-drive train to achieve this would require development. An alternative to this approach would be to use smaller modules with thorium dioxide finger rods in every second module and poison-finger rods in the remaining modules.

For the concentric-lead-screw concept, the guide assembly is assumed to be located above the fuel assembly in the outlet-plenum region. This contains guide tubes for the thorium dioxide fingers around a central poison cluster sheath.

An alternative concept for a control-rod-drive mechanism would be the conventional single-acting drive with scram capability. This would be used in a core with smaller modules having poison-finger rods in every second module and thorium dioxide finger rods in the remaining modules. This alternative would lead to somewhat higher power peaking factors.

Conceptually, the fuel-assembly supporting structure and the plenum assembly for the core would be similar to those for the standard PWR. The core barrel would be suspended from the reactor-vessel closure flange. The core basket, fuel assemblies, lower support, and flow distributor would be supported by the core barrel. The fuel assemblies would rest on the lower support, which also would provide radial alignment. The flow distributor would contain a fuel-assembly alignment plate, a plenum barrel, and a plenum cover. The primary mechanical consideration for the plenum region would be to design to avoid crossflow (which could produce lateral forces on the poison and thorium dioxide finger guide tubes and cause rod jamming) and vibration.

3.2.1.3 Materials

No new or unproved materials are required. However, the uranium dioxide in the prebreeder duplex pellets would be irradiated to higher burnup than that in the reference PWR. In addition, the use of Zircaloy in grids of a close-packed hexagonal rod array requires confirmed testing and analyses.

3.2.2 NUCLEAR CALCULATIONS

3.2.2.1 Prebreeder Nuclear Calculations

The nuclear performance of the backfit prebreeder was determined by Bettis from slice diffusion-depletion calculations (PDQ) of a 205-module commercial core, using four neutron energy groups with breakpoints at 0.8 MeV, 5.53 keV, and 0.625 eV. Prebreeder fuel pellets were substituted for the commercial fuel, and calculations were run using a reasonable fuel-management strategy to obtain a near-equilibrium cycle. Nuclear cross sections used in these calculations were obtained from infinite-medium point-depletion calculations (PAX) which represented the duplex rods as well as the nonfuel components of the modules. Cross sections generated by PAX were used as input to PDQ as interpolating tables to show the dependence of the cross sections on core depletion. The PDQ calculations were used to estimate core reactivity levels, loading requirements, radial power peaking, and mass flows.

Microscopic cross sections for all important isotopes were obtained by means of the PAX03 program, which combines the calculations of fast spectrum, resonance effects, thermal spectrum, and self-shielding. Geometric input to PAX03 consists of physical descriptions of module components (e.g., a fuel cell made up of a fuel pellet, cladding, and associated water and a metal-water cell comprised of a guide tube and associated water) and the relative volumes of these components within the module. Heterogeneous resonance integrals are determined in PAX03 by a collision probability method based on the integral Boltzmann equation under the assumption of isotropy in both the laboratory and center-of-mass systems. All scattering and slowing-down sources are assumed to be flat over an individual region of the cell. Thermally, spatial shielding of the cross sections was treated by using the Multiple Saver method. A point-depletion capability in PAX03 was used to obtain cross-section behavior as a function of time. The depletion model used in PAX03 was the same as that used in PDQ. In this four-energy-group model, equations describing the depletion behavior of all important heavy elements and of the predominant fission products were solved. The treatment of the behavior of fission products used 11 major decay chains (28 fission-product isotopes) and a twelfth fictitious chain to account for the remaining nuclides.

The methods used in calculating [illegible] nuclear performance are refinements of those developed and confirmed by analysis of the Shippingport PWR and LWBR cores which is the major validation of these methods.

3.2.2.2 Breeder Nuclear Calculations

The nuclear performance of the breeder concept was determined from point-depletion calculations based on four neutron-energy groups with breakpoints at

0.8 MeV, 5.53 keV, and 0.625 eV. The neutron cross sections used in these calculations were obtained from detailed Monte Carlo calculations for representative fuel assemblies. The point-depletion results were used to estimate core reactivity levels, lifetime, breeding performance, and mass flows.

Effective few-group microscopic cross sections were generated by means of the RCP01 Monte Carlo program. The Monte Carlo model used 31 energy intervals to describe neutron energies between 0 eV and 10 MeV, with each interval being further divided into as many as 1,000 subintervals to permit accurate representation through all resonances. The primary source of basic cross-section information was the ENDF/B data libraries.

Detailed hexagonal fuel assemblies were represented in the RCP01 calculations, including explicit geometric representations of the fuel pellet, cladding, moderator, and, where appropriate, guide tube for each fuel-bearing and non-fuel-bearing rod in the assembly. The calculated isotopic reaction rates were used to generate highly accurate few-group microscopic cross sections, appropriate for an entire assembly, for use in the point-depletion model. To facilitate the rapid examination of a number of concepts, several different RCP01 assembly calculations were performed to span the range of fuel temperature, moderator temperature, and fuel-to-coolant ratio anticipated for the breeder. In addition, heavy-metal isotopic mixes characteristic both of initial and of equilibrium-cycle loadings were represented.

Few-group microscopic data from the RCP01 calculations were employed in the four-group survey depletion model. In this model, equations describing depletion chains for all important heavy-metal isotopes and the dominant fission-product chains for xenon and samarium were solved. Additional fission-product absorption was incorporated via a residual-fission-product nuclide. The point-depletion results were used to estimate core reactivity levels and lifetimes and to calculate the ratios of the heavy-metal isotopes as a function of fuel depletion. In using this model to evaluate breeding performance, appropriate adjustments were made to the calculated conversion ratios to account for leakage and noncritical reactivity levels in the computations. The estimates of breeding performance and isotopic ratios as a function of fuel depletion were then combined to obtain the desired estimates of core mass flow. The methods used in calculating the nuclear performance of the breeder evolved from those developed and validated by analysis of the operating LWBR core at Shippingport.

3.2.3 THERMAL-HYDRAULIC CONSIDERATIONS

3.2.3.1 Prebreeder Concept

The alternating pellets in the fuel rods of the prebreeder concept lead to an alternating high and low heat flux, especially in fresh fuel rods, because most of the heat is produced in the annulus of the duplex pellet. The effect has to be included in calculations of the critical heat flux and fuel temperature for normal operation and in calculations of cladding temperature during a loss-of-coolant accident (LOCA).

The ratio of the peak heat flux to the average heat flux depends on the lengths of the alternating pellets. To establish a basis for predicting performance during normal operation, preliminary tests have been performed with electrically heated single rods to simulate the performance of alternating pellets about 0.4 inch long. This preliminary testing has indicated that over the range of expected heat flux and lengths of alternation the critical heat flux is determined primarily by the rod-average heat flux and is not affected measurably by the peak-to-average heat flux. Additional full-length rod tests and rod-bundle tests will have to be performed to confirm this preliminary conclusion.

The peak fuel temperatures during normal operation of the prebreeder concept would be lower than those in the reference PWR because most of the heat would be produced in the thin uranium dioxide annulus of the duplex pellet and also because the thermal conductivity of thorium dioxide is higher than that of uranium dioxide. The volumetric heat capacity of thorium dioxide is also lower than that of uranium dioxide. As a result of these effects, the total heat stored in a fuel rod would be lower for the prebreeder concept than for the reference PWR. During a loss-of-coolant accident, the peak fuel-rod cladding temperature would be affected by the total heat stored in the fuel rods.

3.2.3.2 Breeder Concept

In comparison with the reference PWR, the breeder concept would have a higher pressure drop and would require greater pumping power.

Neutron capture in the thorium dioxide finger rods would produce a substantial heat flux, and the cooling requirements would be greater than for the poison-finger rods used in the reference PWR for power shaping. The plenum design would require limiting crossflow so as to avoid vibration of, and unacceptable forces on, the guide tubes for both the poison-finger shutdown rods and the thorium dioxide finger rods.

3.2.3.4 Thermal Calculations

Thermal performance has been analyzed by Bettis with a simplified calculational model. This model has been qualified by performing detailed module calculations allowing for the transfer of two-phase fluid properties in three dimensions to predict local fluid conditions and the critical heat flux. These detailed calculations were made with the computer program HOTROD, which was used for the thermal analysis of the Shippingport LWBR core.

The simplified model relates the steady-state overpower thermal performance of the proposed concept to that of a reference commercial design, such as the Babcock & Wilcox Standard 205 design. The difference in total reactor flow between the proposed concept and the reference design was determined from changes in parameters that affect the flow, such as the core hydraulic diameter, total core flow area, fuel-rod length, and number and type of grids. Mass velocity and inlet temperature were calculated from the flow for a specified core-average temperature. The hot-channel critical-heat-flux performance was then determined by factoring in changes in the parameters that affect the critical heat flux, such as coolant velocity, hydraulic diameter, inlet temperature, power-peaking factors, and channel length.

Commercial design procedures, methods, hot-channel factors, and correlations provided the basis for the analysis. The peak linear power, in kilowatts per foot, would be maintained at a level that results in acceptable fuel-element and LOCA performance.

3.3 ENVIRONMENTAL CONSIDERATIONS

3.3.1 SUMMARY ASSESSMENT

The thermal, the chemical and biocide, and the radiological releases from this system would be each similar to the corresponding releases from the reference LWR. This system should, therefore, present no environmental licensing problems.

3.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM (R.G. 4.2/3.2)

The light-water backfit prebreeder supplying an advanced breeder is another LWBR scheme that may be capable of meeting current regulations, including Appendix I to 10 CFR 50. This scheme would consist of a conventional PWR vessel and plant with a modified core, the light-water backfit prebreeder, which produces fuel for an advanced light-water breeder. The advanced light-water breeder would consist of a conventional PWR system, except that its core configuration would be similar to that of the Shippingport movable-fuel LWBR but with less neutron-absorbing structure, a more uniform power distribution, and a slightly higher average power density. The basic parameters describing a typical plant are given in Table 3-10.

3.3.3 STATION LAND USE

Information given in Section 2.3.3 for the LWBR type I module concept applies to this concept as well.

3.3.4 STATION WATER USE (R.G. 4.2/3.3)

Information given in Section 2.3.4 for the LWBR type I module concept applies to this concept as well.

3.3.5 HEAT-DISSIPATION SYSTEMS (R.G. 4.2/3.4)

Information given in Section 2.3.5 for the LWBR type I module concept also applies to this concept. There would be an approximate 5% increase in the heat-dissipation rate of the backfit prebreeder, but this is expected to be relatively inconsequential.

3.3.6 RADWASTE SYSTEMS AND SOURCE TERMS (R.G. 4.2/3.5)

Information given in Section 2.3.6 for the LWBR type I module concept also applies to the advanced-breeder concept except for the addition of boron-recycle equipment to the Chemical and Volume Control System. Boron would be added to the coolant of the breeder for reactivity control during shutdown but would be removed for normal operation. The backfit prebreeder, on the other hand, would utilize boron in the coolant during normal operation. This would allow the formation of more radioactive tritium in the coolant as a result of neutron interactions with boron. Furthermore, the discharge burnup in the prebreeder would be three times higher than that of the breeder, bringing the radioactive release rates of the backfit prebreeder in line with those of a typical PWR. Tables 3-11 and 3-12 show typical releases from the prebreeder cycle. The releases from the breeder would be the same as shown in Section 2.3.6.4.

3.3.7 CHEMICAL AND BIOCIDAL WASTES (R.G. 4.2/5.3)

Information given in Section 2.3.7 for the LWBR type I module concept also applies to this concept.

3.3.8 EFFECTS OF OPERATION OF THE HEAT-DISSIPATION SYSTEM (R.G. 4.2/5.1)

The thermal efficiency of the prebreeder would be very similar to that of the reference LWR; the efficiency of the advanced breeder would be a fraction of a percent higher than that of the LWR. Consequently, the effects of the dissipated heat would be very slightly less than for the LWR.

3.3.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATIONS (R.G. 4.2/5.2)

The dose percentages for the light-water backfit prebreeder from liquid pathways by isotope are presented in Table 3-13; those from noble-gas releases and radioiodine and particulate releases are in Table 3-14 and 3-15. The comparable dose contributions from the breeder type I modules are shown in Table 2-15, 2-16 and 2-17. These values are similar to those for the reference LWR case and should not represent any problem to licensing.

3.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES (R.G. 4.2/5.3)

The chemical and biocidal discharges would be similar to those from the reference LWR; therefore the effects are similar.

3.3.11 OCCUPATIONAL EXPOSURE

Information given in Section 2.3.11 for the LWBR type I module concept also applies to this concept.

Table 3-10. Basic parameters describing the light-water backfit prebreeder concept supplying advanced breeder concept

Parameter	Backfit Prebreeder	Advanced Breeder
Fuel cycle	U-Th recycle	U-Th breeder
Burnup, MWd/MT	34,800	10,100
Base reactor thermal output, MWt	3,800	2,900
Electrical output, MWe	1,295	1,035
Normalized reactor output, MWe	1,000	1,000
Heat rate, Btu/kW-hr	9,990	9,570
Heat dissipation rate at 1,000 MWe, Btu/hr	6.6×10^9	6.2×10^9

Table 3-11. Liquid-radioactive-release source terms for a typical commercial PWR plant described here with a light-water backfit prebreeder

Nuclide	Source term[a] (Ci/yr)	Nuclide	Source term[a] (Ci/yr)
Bromine-82	0.00007	Barium-137m	0.01
Bromine-83	0.0001	Cesium-138	0.00002
Rubidium-86	0.00004	Barium-139	0.00004
Strontium-89	0.0002	Barium-140	0.0002
Strontium-91	0.00006	Lanthanum-140	0.0001
Yttrium-91m	0.00002	Cerium-141	0.00002
Yttrium-91	0.0001	Cerium-143	0.00001
Zirconium-95	0.00002	Praseodymium-143	0.00002
Niobium-95	0.00002	Cerium-144	0.00005
Molybdenum-99	0.0003	Praseodymium-144	0.00002
Technetium-99m	0.0003	Neodymium-147	0.00001
Ruthenium-103	0.00002	Sodium-24	0.0001
Rhenium-103m	0.00002	Phosphorus-32	0.00002
Tellurium-125m	0.00001	Phosphorus-33	0.0001
Tellurium-127m	0.0001	Chromium-51	0.0003
Tellurium-127	0.0002	Manganese-54	0.00006
Tellurium-129m	0.0005	Manganese-56	0.001
Tellurium-129	0.0003	Iron-55	0.0003
Iodine-130	0.0004	Iron-59	0.0002
Tellurium-131m	0.0005	Cobalt-58	0.003
Tellurium-131	0.0001	Cobalt-60	0.0004
Iodine-131	0.14	Nickel-65	0.00002
Tellurium-132	0.01	Niobium-92	0.00006
Iodine-132	0.01	Tin-117m	0.00002
Iodine-133	0.1	Tungsten-185	0.00002
Iodine-134	0.00007	Tungsten-187	0.0005
Cesium-134m	0.00003	Neptunium-239	0.0002
Cesium-134	0.01	All others[b]	0.0001
Iodine-135	0.02		
Cesium-136	0.005	Total[c]	0.3
Cesium-137	0.01		
		Tritium	270

[a]Normalized to 1,000 MWe.

[b]Includes isotopes with discharges of less than 10^{-5} Ci/yr-unit.

[c]Does not include tritium.

Table 3-12. Gaseous-radioactive-release source terms for a typical commercial PWR plant described here with a light-water backfit prebreeder

Nuclide	Source term[a] (Ci/yr)
Krypton-83m	<1
Krypton-85m	11
Krypton-85	380
Krypton-87	2
Krypton-88	14
Krypton-89	<1
Xenon-131m	44
Xenon-133m	80
Xenon-133	7,200
Xenon-135m	<1
Xenon-135	50
Xenon-137	<1
Xenon-139	1
Iodine-131	0.05
Iodine-133	0.06
Tritium	580
Carbon-14	6
Particulates	0.05

[a]Normalized to 1,000 MWe.

Table 3-13. Contributions to doses by liquid effluents for a typical commercial PWR described here with a light-water backfit prebreeder

Isotope	Contribution (%) to organ dose Adult whole body	Critical organ
Tritium	19	3
Cs-134	43	1
Cs-136	3	1
Cs-137	32	1
I-131	1	85
I-133	1	10
Other	4	2
Ratio of dose to that from reference LWR	1.0	1.0

Table 3-14. Contributions to radiation doses by noble gases for a typical commercial PWR plant described here with a light-water backfit prebreeder

Nuclide	Contribution (%) to organ dose	
	Whole body	Skin
Kr-83m	(a)	(a)
Kr-85m	(a)	(a)
Kr-85	(a)	10
Kr-87	(a)	1
Kr-88	8	4
Kr-89	1	(a)
Xe-131m	(a)	(a)
Xe-133m	1	2
Xe-133	85	79
Xe-135m	(a)	(a)
Xe-135	4	3
Xe-137	(a)	(a)
Ratio of dose to that from reference LWR	1.0	1.0

[a]Less than 1%.

Table 3-15. Contribution to radiation doses by radioiodines and particulates for a typical commercial PWR plant described here with a light-water backfit prebreeder

Nuclide	Contribution (%) to organ dose	
	Infant thyroid	Child thyroid
I-131	96	91
I-133	1	1
C-14	2	5
Tritium	1	3
Ratio of dose to that from reference LWR	1.0	1.0

3.4 SAFETY CONSIDERATIONS

3.4.1 UNIQUE SAFETY ASPECTS OF THE CONCEPT

This section describes the major unique features of the prebreeder and breeder concepts, as compared with the reference PWR (Ref. 1), that will need to be considered for licensing.

3.4.1.1 Prebreeder Fuel

The major unique feature of the prebreeder concept is the fuel pellet. Placing the fissile-fuel-bearing uranium dioxide annulus around the thorium dioxide core, which would initially contain no fissile fuel, would cause most of the fission energy to be generated close to the cladding and therefore result in lower average fuel-element temperatures than would be obtained with a solid uranium dioxide pellet. This arrangement would have the advantage of reducing stored heat, which is important during a loss-of-coolant accident. In this concept the annulus would experience peak burnup as high as 119,000 MWd/MT compared with 50,000 MWd/MT in the reference PWR. Currently, there is little information on the structural and thermal characteristics of fuel at high burnup, on fission-gas release, on fuel swelling and cladding strain rate, fuel-cladding response to power transients, and the effect of the high-temperature inner annular surface on defective-rod performance. These matters would have to be investigated before the capabilities of such a system could be established. This fuel system is being evaluated for possible further development under the Department of Energy (DOE) Advanced Water Breeder Applications (AWBA) program, including analytical work to extend the scope of present fuel-element modeling and irradiation testing.

3.4.1.2 Unique Breeder Features

This section discusses the major unique features of the breeder concept that would need to be considered for licensing. Items a through f were evaluated in the Shippingport LWBR safety analysis and were found acceptable in the NRC review. Item h is the feature of the advanced breeder that is uniquely different from the Shippingport LWBR.

a. Tight Lattice

The rod-to-rod spacing in the advanced breeder concept would be about 60 mils rather than the 120 mils used in commercial practice. The tight lattice would require a different fuel-element and grid design and different fuel-assembly procedures compared to those used in commercial practice to avoid rod-to-structure and rod-to-rod contact. The rod-to-rod spacing in the LWBR core at Shippingport is also 60 mils. In setting core operational limits (specifications of set points and allowable power increase) for the Shippingport LWBR, it was assumed that rod-to-structure contact does occur. Extensive in-reactor and out-of-reactor tests with rods in contact have been completed. Design assessments of fuel-rod bowing in the Shippingport LWBR core predict that rod-to-rod contact is unlikely but even if it were to occur, it is calculated to be acceptable. Bowing analysis of the blanket has been completed on a worst case basis by the DOE, and the rod-to-rod spacing has been determined as a function of reactor lifetime. These data demonstrate that adequate margin is incorporated into the Shippingport LWBR design and indicates that acceptable operation of fuel assemblies in a commercial-scale reactor of this design should be feasible.

b. Core Thermal Margins

Thermal analysis programs and correlations for the Shippingport LWBR proved applicable for thermal-hydraulic analyses under widely ranging axial and radial-heat-flux distributions in a close rod array, including a coupled-region interface. The Shippingport LWBR critical-heat-flux correlation conservatively predicts the data for the full range of Shippingport LWBR geometries and heat-flux distributions. Thermal margins for the present concept have been calculated by the DOE based on the approach similar to that used for the Shippingport LWBR. Additional critical-heat-flux testing may be required to confirm the applicability of the modeling of the specific breeder fuel lattice.

c. Provision for Accident Prevention

The probabilities of accident initiation for the Shippingport LWBR core and for the present breeder concept are comparable to that of the reference PWR. The safety and protection of the Shippingport LWBR plant have been designed in accordance with regulatory guidelines and requirements. An emergency core-cooling system appropriate to the LWBR has been incorporated into the Shippingport design. No impediment to providing comparable protection features to a larger plant using the present breeder concept has been identified.

d. Acceptability of Movable Fuel for Reactivity Control

To date, all reactivity-control functions required by an operating nuclear power plant have been satisfactorily performed in the Shippingport LWBR by means of the movable fuel. These include control functions required for shutdown, plant heatup, power operation, and lifetime reactivity changes. Scram reactivity is also provided by the movable fuel. For the present prebreeder concept, the movable fuel would supply only the reactivity control to match part of the power defect plus the buildup of long-lived fission products. The requirement on the movable fuel in the present breeder concept would be much less severe than in the Shippingport LWBR core. However, since the reactivity increase available from one control assembly during power operation is greater in the breeder concept than in current LWRs, consideration will need to be given to control mechanism design to assure that control element ejection is not credible, as was done for the Shippingport LWBR case.

e. Power and Temperature Coefficients

The values of these coefficients and the accuracy with which they have been calculated or estimated are two aspects of this question. The calculation of temperature and power coefficients using the design model has given good agreement with measurements for both critical-experiment configurations and the LWBR core at Shippingport. Measured zero-power temperature coefficients during both the initial and the first periodic testing phases have been calculated satisfactorily for both hot and cold conditions. Performance at power was also calculated satisfactorily. The calculated power coefficient of reactivity was 6% less negative than the measured value, and the calculated temperature coefficient at power was 6% more negative than the measured value. There is a firm basis for expecting that the temperature and power coefficients for the present breeder concept would be similar to those for the Shippingport LWBR core and therefore acceptable with respect to licensing. The temperature coefficient, which depends primarily on the fuel-to-coolant ratio, is expected to remain approximately constant or increase slightly during the life of the core, consistent with the facts that no boron would be used in the core, that the fuel-to-coolant ratio would decrease slightly as movable fuel was lifted, and that the fissile inventory would remain nearly constant during core operation. In the Shippingport LWBR core, the power coefficient is due almost entirely to the Doppler effect and depends primarily on the fuel-to-coolant ratio and the power density (kilowatts per kilogram of heavy metal) at full power. In the present breeder concept, the fuel-to-coolant ratio would be somewhat larger than that of the Shippingport LWBR core and the power density is slightly higher than that of the Shippingport LWBR core. It is therefore expected that the power coefficient for the present breeder concept would be similarly predictable and at least as negative as that for the Shippingport LWBR.

f. Control Stability and Adequacy

The LWBR core at Shippingport has successfully operated as part of an integrated commercial power network. This operation has included long periods at constant power, several weeks of planned swing-load operations, and controlled startup and shutdown periods from zero to full power. In addition, special tests were run to demonstrate the dynamic characteristics and response of the plant to typical load-change rates common in commercial plants. Performance was satisfactory during both swing-load and steady-state operation and in the special tests. The results indicate that control, stability, and adequacy can be maintained in a core with movable-fuel reactivity control.

g. Nuclear Stability

Analytical studies have examined the stability of large uranium-233/thorium reactors with high power densities against spatial xenon oscillations and have compared it with the stability of uranium-fueled PWRs of comparable sizes and ratings (Ref. 2). These studies show that uranium-233/thorium systems are inherently significantly more stable primarily because of lower total xenon yields, a larger fraction of fission yield direct to xenon, and more negative Doppler coefficients of reactivity. Initial physics testing at the Shippingport LWBR has demonstrated that the LWBR module design results in a tightly coupled core and provides further confidence that the present breeder concept would have acceptable stability properties.

h. Advanced Breeder Control

The essential difference between this concept and the presently operating Shippingport LWBR core is that instead of moving bundles of fuel rods, moving fingers of individual thorium dioxide control rods, arranged in a manner similar to the fingers of poison rods now being used in commercial PWRs, would be interspersed throughout the module. Poison-finger rods may also be required either in the same modules as the thorium dioxide finger rods or in alternate modules. A program to determine whether additional development of these alternative control methods should be pursued via design and engineering tests is now being evaluated as part of the DOE AWBA program.

3.5 LICENSING STATUS

The status is essentially the same as described previously in Section 2.5 for the Type I module concept.

As stated in Section 3.4.1.2f, the Shippingport LWBR core has been tested extensively. The results indicate that plant control, stability, and adequacy for utility-type operation are not adversely affected by a core with movable-fuel reactivity control. It is expected that a commercial-size reactor would also have adequate control characteristics.

As noted in Section 3.4.1.2g, LWBRs are expected to have larger margins against spatial xenon instabilities than are current PWRs, partly because of the larger Doppler coefficient of reactivity produced by the thorium. The stability of the prebreeder is expected to be comparable with that of the reference PWR.

The maintenance of a plant containing either of the present prebreeder or breeder concepts would be essentially the same as for a plant containing a conventional PWR core.

3.6 TECHNOLOGY STATUS AND RESEARCH, DEVELOPMENT, AND DEMONSTRATION REQUIREMENTS

3.6.1 PREBREEDER CONCEPT

The main area in which technology would need to be extended is the fuel system of alternating duplex pellets and solid thorium dioxide pellets. This fuel system is being evaluated for possible further development as part of the DOE AWBA program. Duplex pellets have been fabricated, and irradiation testing of fuel rods is in progress. Additional critical-heat-flux testing may be required. Computer models to predict the performance of this fuel system are also being developed.

3.6.2 BREEDER CONCEPT

The areas in which the technology would need to be extended include the fuel-assembly grids, thorium dioxide control fingers, and the plenum. Zircaloy grids are already in use in commercial cores; an extension to larger size and number of rods and to the tighter tolerances associated with the close-packed hexagonal array in the breeder reactor would be required. The thorium dioxide control fingers are essentially fuel rods suspended from one end; their operation in a guide tube with proper coolant flow and acceptable wear and vibration characteristics would have to be demonstrated. The plenum design would require flow testing to demonstrate operation without vibration or unacceptable lateral forces due to crossflow.

The "dual-action" control-rod-drive mechanism, if chosen, would also require an extension of technology. The concepts of roller nuts and magnetic latching for control-rod-drive mechanisms are not new; the novel feature is their application to concentric lead screws and to high loads.

An additional area for research, development and demonstration would be the development of the large pressure vessels required for application of the advanced breeder concept in a large (1,000 MWe) plant. Under the DOE AWBA program, a vendor has completed a study of the feasibility of manufacturing larger reactor vessels than those currently operating or planned; however, additional studies and confirmation of manufacturing capabilities for such large reactor vessels would probably be needed.

In addition to the above efforts on the reactor components, the use of a prebreeder-breeder system would require the recycle of fuel and the research, development, and demonstration necessary to implement reprocessing, refabrication, and waste management.

The requirements for this concept are summarized in Tables 3-16 and 3-17.

3.6.3 CONCERNS NEEDING RESOLUTION

Because of the preliminary nature of the concept, the concerns cited in Section 2.6.2 apply to this concept as well.

Table 3-16. Technological advance requirements for backfit prebreeder concept

Plant component	Operating experience evaluated	Prototype or production components being manufactured	No new knowledge required	Modest improvement in performance or size from present knowledge
Nuclear fuel[a]				√
Reactivity-control systems	√	√	√	
Reactor vessel	√	√	√	
Core-support structure				
Reactor-vessel internals, including shielding, ducting, control-rod guides, baffles, etc.	√	√	√	
Primary-coolant pumps and auxiliary systems	√	√	√	
Primary-coolant chemistry/ radiochemistry control	√	√	√	
Primary-system heat exchangers	√	√	√	
Reactor instrumentation	√	√	√	
Emergency core-cooling/safe-shutdown systems	√	√	√	
Containment, containment-cleanup systems, and effluent-control systems	√	√	√	
Other accident-mitigating systems (i.e., plant-protection systems)	√	√	√	
On-site fuel-handling and storage/ shipping equipment	√	√	√	
Main turbine	√	√	√	
Other critical components, if any	√	√	√	
Balance-of-plant components	√	√	√	

[a]Irradiation test data presently being obtained for prebreeder fuel.

Table 3-17. Technological advance requirements for advanced breeder concept

Plant component	Operating experience evaluated	Prototype or production components being manufactured	No new knowledge required	Contemporary technology with modified configuration/application	Modest improvement in performance or size from available systems
Nuclear fuel	a	√	√		
Reactivity-control systems					√
Reactor vessel					√
Core-support structure				√	
Reactor-vessel internals, including shielding, ducting, control-rod guides, baffles, etc.	√	√			√
Primary-coolant pumps and auxiliary systems	√	√	√		
Primary-coolant chemistry/ radiochemistry control	√	√	√		
Primary-system heat exchangers	√	√	√		
Reactor instrumentation	√	√	√		
Emergency core-cooling/safe-shutdown systems	√	√	√		
Containment, containment-cleanup systems, and effluent-control systems	√	√	√		
Other accident-mitigating systems (i.e., plant-protection systems)	√	√	√		
On-site fuel-handling and storage/ shipping equipment	√	√	√		
Main turbine	√	√	√		
Other critical components, if any	√	√	√		
Balance-of-plant components	√	√	√		

[a]Operating data being obtained from LWBR.

REFERENCES FOR CHAPTER 3

1. Babcock and Wilcox, Reference Safety Analysis Report, BSAR-205, NRC Docket STN 50-561.

2. T. R. England, G. L. Hartfield, and R. K. Deremer, Xenon Spatial Stability in Large Seed-Blanket Reactors, WAPD-TM-606, Bettis Atomic Power Laboratories, April 1967.

Chapter 4

LIGHT-WATER BACKFIT PREBREEDER/SEED-BLANKET BREEDER SYSTEM

4.1 OVERALL DESCRIPTION OF THE CONCEPT

This section describes a conceptual reactor system that would use currently available LWR technology and existing reactor plants to fuel an advanced-technology light-water seed-blanket breeder capable of significantly increasing the utilization of existing supplies of uranium. This reactor system would use a backfit prebreeder (new reactor core in existing PWR vessel and plant) to produce a mixed-uranium-isotope fuel for an advanced light-water seed-blanket breeder.

The backfit prebreeder would make maximum use of current LWR technology to provide a transition from the prebreeder fueled with uranium-235 to the seed-blanket breeder, which eventually would be fueled with equilibrium concentrations of uranium isotopes.

The backfit prebreeder would be identical with current commercial PWRs except for the use of the mixed uranium/thorium dioxide fuel that has been developed for the Shippingport LWBR. Thus, by substitution of fuel, existing PWR plants could be used on a near-term basis to generate fuel for a light-water breeder. This approach requires little technology development effort and a minimum of testing and confirmation effort.

The light-water seed-blanket breeder would be an advanced light-water design in that a seed blanket configuration would be used to maximize neutron economy and thereby improve breeding performance. The breeder would initially be fueled with a uranium/thorium dioxide mixture from backfit prebreeders and additional quantities of mined thorium oxide. Subsequent fuelings of the breeder would use recycled fuel from the breeder. Tables 4-1 through 4-6 list the parameters for the concept.

4.1.1 PREBREEDER

The backfit prebreeder would use an existing commercial PWR (for illustration purposes, the Combustion Engineering (C-E) System 80 plant is assumed) rated for 3,800 MWt (Ref. 1). This prebreeder would be identical with the thorium/uranium-233 recycle design described in Reference 2, except for the method of recycling fuel. Figure 4-1 shows the reactor-core cross section and standard control-element-assembly pattern for the backfit prebreeder.

The conventional C-E System 80 (one of several commercial PWR plants that could be used for backfit) consists of a two-loop PWR with four pumps per loop and supporting auxiliary systems. The reactor core is composed of 241 fuel assemblies and 89 control-element assemblies. The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 143 inches and an active length of 150 inches. Each fuel assembly provides for 256 fuel-rod positions (16 x 16 array), of which 20 positions are occupied by control-assembly guide tubes (five guide tubes each occupying four rod positions). The fuel-rod outside diameter is about 0.380 inch.

The backfit prebreeder would use the standard C-E System 80 plant except for the use of a thorium-based fuel instead of low-enrichment uranium dioxide fuel. Fuel used in both the approach described in Reference 2 and the backfit prebreeder concept would consist of a binary solid solution of uranium dioxide and thorium dioxide. Initially, the uranium dioxide would be highly enriched. As the fuel is irradiated, the fertile thorium-232 would be converted to fissile uranium-233. Although the fission products and thorium could be chemically separated from the fuel during reprocessing, the uranium isotopes would remain mixed. In the backfit-prebreeder concept, in each cycle one-third of the core would be removed from the prebreeder reactor and would be reprocessed and refabricated for use in an LWBR, whereas in the approach of Reference 2 one-third of the core would be recycled back into the original reactor in each cycle.

Reactivity in the prebreeder would be controlled by two independent systems already present in commercial PWRs. The control-element-drive system, consisting of neutron-absorber elements, would be used to provide rapid shutdown. A chemical and volume control system would be used to compensate for long-term reactivity changes and could make the reactor subcritical without the use of the absorber elements. Boric acid dissolved in the coolant would be used as the neutron absorber. In addition, burnable poison rods would be used to compensate for the excess reactivity at the beginning of life.

4.1.2 LIGHT-WATER SEED-BLANKET BREEDER

The primary system of the light-water seed-blanket breeder, except for the reactor and related equipment is made up of components identical with those used in current commercial LWRs. Initial concepts include two reactor-coolant loops, with two steam generators and two reactor-coolant pumps per loop, all housed in a single containment vessel along with the nuclear reactor. Plant rating would be currently targeted at 1,000 MWe at an average operating temperature of 590°F on the basis of the current maximum commercial-reactor-vessel size. Other plant components would be sized consistent with a plant rating of 1,000 MWe unless larger (1,300 MWe) components are required to achieve a 1,000-MWe capability when the breeder core is installed.

The reactor vessel would be similar in size to the vessels used for larger commercial PWR reactors, which have been licensed, and would be identical in construction and materials. The fuel assemblies would be comparable in length to those of current commercial PWRs and similar in cross-sectional area to those used in the Shippingport LWBR. The fuel-module configuration would consist of discrete seed and blanket rod regions (as in the Shippingport LWBR) in an open lattice typical of commercial PWRs.

The seed rods of the breeder would consist of a solid solution of uranium and thorium dioxides in pellet form, stacked in a Zircaloy cladding tube. The cladding outside diameter would be about 0.44 inch, and the active fuel length would be 12 feet, as in commercial PWRs.

Initially, the breeder would be loaded with fuel recycled from the prebreeder. The mixed-oxide fuel would contain thorium dioxide mixed with bred uranium-233, nonfissioned uranium-235, and other uranium isotopes discharged from the prebreeder. Although fission products would be removed, the fuel would be highly radioactive because of the presence of uranium-232 and its daughters. Subsequent loadings would be obtained by the recycle of fuel discharged from the breeder itself or by fuel recycled from other prebreeders.

Blanket rods would consist of thorium dioxide pellets in Zircaloy tubes with dimensions similar to or somewhat larger than those of the seed rods. Fertile thorium-232 in the blanket rods would be converted to fissile uranium-233 during operation at power, resulting in reduced changes in lifetime reactivity in comparison with commercial PWRs and thereby minimizing the need for soluble boron for the control of reactivity.

Shutdown and power-control capability would be provided by groups of individual poison rods in guide tubes located in discrete regions surrounded by regions of seed rods. The cold-to-hot temperature defect would be compensated for by dilution of soluble boron, as in current PWRs.

Compensation for lifetime reactivity changes would be provided by groups of fertile thorium dioxide shim rods in individual guide tubes located in the same discrete regions as the poison rods. All rods--seed, blanket, shim, and shutdown--would be held in place on a triangular pitch of 0.51 inch by Zircaloy grid structures and arranged in an open lattice that would permit free mixing of coolant throughout the lattice.

The drive mechanisms for the shim and shutdown rods would be similar to those used in current commercial PWRs. Mechanisms that drive shutdown rods would have a scram capability, as in commercial PWRs; however, the shim mechanisms have only a shim capability for the control of lifetime reactivity changes and would have no safety function. The seed-blanket arrangement is characterized by a smaller change in reactivity from beginning to end of cycle. The increasing blanket reactivity would compensate for most of the decreasing seed reactivity with depletion.

The resultant small reactivity change associated with depletion can be controlled by the thorium dioxide shim rods, thus minimizing the need for the use of soluble boron during operation at power. At discrete intervals of depletion, one or more groups of shim rods would be moved from a fully inserted to a fully withdrawn position. This approach would minimize peaking of the axial power shape and thereby maximize the operating margin.

Consistent with the objective of relying on current LWR technology, auxiliary systems for soluble-boron addition and removal, long-term shutdown cooling, decay-heat removal, and emergency core cooling would be functionally identical with those used in commercial PWRs or in the Shippingport LWBR. The containment configuration would be equivalent to that of current commercial PWRs.

4.1.3 ACCOMMODATION IN EXISTING PHYSICAL PLANTS

The drive mechanisms, reactor vessel, head, and internals as well as the balance of the nuclear steam supply system, including auxiliary systems for soluble-boron addition and removal, long-term shutdown cooling, decay-heat removal, and emergency core cooling of the backfit plant, should all be reusable with the backfit prebreeder, although some changes in component sizes and capacities may be required to maintain performance characteristics comparable to those of the original PWR design.

4.1.4 FUEL MANAGEMENT AND FUELING ALTERNATIVES

A typical operating system for the backfit prebreeder fueling the seed-blanket breeder would consist of the prebreeder generating fuel each year until enough fuel has been reprocessed to fuel the breeder for a predetermined number of years. This reprocessed fuel would be a mixture of all uranium isotopes generated in the prebreeder and would have to be a quantity sufficient to account for the initial criticality of the breeder, makeup for the breeder, and reprocessing and refabrication losses. The backfit prebreeder would use fuel management, with one-third of the core being replaced each year. The advanced-breeder concept would be batch-loaded and operates for approximately 2.5 years at a capacity factor of 75%. At the end of each batch-loaded cycle, the entire breeder core would be sent to reprocessing, and the reactor would be refueled with a recycled breeder load.

4.1.4.1 Alternative Fueling Options for the System

Several alternative fueling options would be possible for the backfit prebreeder/seed-blanket breeder system. These options could include the following:

1. The prebreeder could be fueled with plutonium-239 as the initial fissile fuel rather than uranium-235.
2. The fuel produced from the prebreeder could fuel other reactor types, such as the LMFBR, using either thorium or uranium-238 as the fertile material.
3. The prebreeder concept could be used as a converter to recycle the mixture of depleted and bred fuel back into the prebreeder rather than to save it for the breeder.
4. Low-enrichment fuel could be used as the prebreeder fuel adding uranium-238 dioxide to the mixture of uranium-235 dioxide and thorium dioxide.

4.1.5 FUEL CYCLES

The prebreeder would consist of a PWR-type-core backfitted into a PWR CE-80 vessel. The fuel would consist of a binary solid solution of uranium and thorium dioxides. Reactivity control would be achieved by poison control rods and dissolved boric acid in the coolant. Spent fuel would be reprocessed to recover the uranium which would be refabricated for the initial fuel loadings in the seed-blanket breeder.

The seed-blanket breeder would use a PWR type vessel somewhat larger than present commercial PWRs. The fuel module would consist of seed blanket regions. Reactivity control would be achieved by movable thorium dioxide rods in the seed region. The seed rods would consist of a solid solution of uranium and thorium dioxides in pellet form. Initially, the breeder would be loaded with fuel recycled from the prebreeder (mixed uranium-fissile fuel: bred uranium-233 and nonfissioned uranium-235). At equilibrium, the breeder would be fueled from recycled uranium (all isotopes) discharged from the breeder plus the mixed-uranium-isotope fuel discharged and stored from the prebreeder. (The prebreeder fuel would be used to fuel the initial cycle of the breeder plus supply all makeup fuel needed.)

4.1.5.1 Backfit Prebreeder HEU(5)-Th

The backfit prebreeder would utilize 93% highly enriched uranium-235 fuel as a binary solid solution of thorium and uranium dioxides in the form of pellets. The spent fuel would be reprocessed to recover all uranium isotopes and would then be fabricated into fuel for the initial fuel requirements for the seed blanket breeder. A typical mass flow diagram for this system is shown in Figure 4-2.

4.1.5.2 Seed Blanket Breeder HEU-Th/Th

The seed blanket breeder would use a large extrapolated-PWR-type vessel. The fuel would consist of seed rods containing a binary solid solution of uranium and thorium dioxides; the blanket would consist of thorium dioxide pellets. The core and blanket would be reprocessed to recover the uranium-233 which is recycled and mixed with makeup also supplied by the initial prebreeder cycles in the form of mixed uranium isotopes. A typical mass flow diagram for this system is shown in Figure 4-3.

4.1.5.3 Quantitative Fuel Inventories

Table 4-7 summarizes the overall fuel-management information, including the separative-work requirements typical of this concept. Table 4-8 shows the calculated isotopic content of the equilibrium cycle for this prebreeder/mixed-breeder concept. Table 4-9 shows the calculated overall isotopic content for the prebreeder one-third core charge and discharge under equilibrium conditions, and Table 4-10 shows the calculated overall isotopic content for the breeder over a 30-year history. Figures 4-2 and 4-3 show the mass flows for the equilibrium prebreeder and breeder cycles, respectively.

4.1.5.4 Reprocessing Considerations

Once the prebreeder fuel load is subject to depletion, significant quantities of uranium-232 and its daughter products would be generated in the thorium matrix of the fuel elements along with fission products. As a consequence, the fuel discharged from any prebreeder or breeder would be highly radioactive, and significant radioactivity would be retained during reprocessing, even after the fission products are separated from the fissile and fertile fuel. To provide for the recycle of fuel, for this or any other breeder system, reprocessing and refabrication facilities would be required. The Shippingport LWBR fuel was fabricated in shielded gloveboxes, but this is not representative of the remotely operated facilities and processes expected to be necessary for recycle and refabrication of fuels of the thorium/uranium-233 fuel cycle on a commercial basis.

The fuel cycle described above is typical of what would be required to achieve the goal of breeding in a light-water concept and has the advantages of minimizing resource requirements and plutonium generation. Other fuel cycles (denatured uranium dioxide, etc.) have been considered for light-water prebreeder/breeder systems. Increased demands on available resources, reduced potential for breeding during the equilibrium cycle, and concerns associated with handling significant quantities of plutonium are disadvantages of such fuel cycles.

Table 4-1. Generalized performance specifications: light-water backfit prebreeder concept supplying seed-blanket breeder concept

Parameter	Prebreeder	Breeder
Power plant performance parameters		
Reactor thermal power output, MWt	3,817	2,993
Net electrical power output, MWe	1,300	1,000
Plant heat rate, Btu/kW-hr	--	10,220
Core performance parameters		
Core heat output, MWt	3,817	2,993
Core volume, liters	40,040	47,200
Core loading, kg		
Heavy metal[a]	93,507	171,504
Fissile fuel	3,088[b]	5,622[c]
Conversion ratio	0.5	1.00[d]
Average seed discharge burnup, MWd/MTHM[e]	33,961	15,300
Peak discharge burnup, MWd/MTHM	54,300	24,500
Fuel type	Binary UO_2/ThO_2	Binary UO_2/ThO_2
Reactor inlet temperature, $^{\circ}$F	565	567
Reactor outlet temperature, $^{\circ}$F	621	617
Fissile inventory ratio		
Initial cycle	0.61	
Equilibrium cycle		1.0[d]

[a]Does not include axial or radial reflectors.

[b]Fissile load for initial cycle of the prebreeder.

[c]Fissile load for initial cycle of the breeder as fueled by the mixed-isotope prebreeder.

[d]For the equilibrium fuel cycle of the breeder.

[e]Heavy metal charged.

Table 4-2. Reactor concept data: light-water backfit prebreeder concept supplying seed-blanket breeder concept

Parameter	Prebreeder[a]	Breeder
Geometric information		
Core height, cm[b]	381	366
Number of core enrichment zones	3	1
Number of assemblies	241	169
Equivalent diameters, cm	373	405
Number of pins per assembly[c]	236	459
Pins pitch-to-diameter ratio	1.33	1.16
Overall assembly length, cm	450	513
Lattice pitch, cm	1.29	1.30
Assembly material	Zircaloy-4	Zircaloy-4
Cladding parameters		
Cladding outside diameter, mils	382	440
Cladding wall thickness, mils	25	27.5
Cladding material	Zircaloy-4	Zircaloy-4
Fissile inventory at beginning of equilibrium cycle, kg	3,088[d]	4,975[e]
External fissile inventory, kg	1,235/2,470[f]	1,990/3,980[f]
Fissile gain or loss, kg/cycle	495 (loss)	0[e]
Specific power, kW/kg fissile	1,231[d]	598[e]
Power density, kW/kg HM	41	17
Power density, kW/liter	95	63

[a]Based on Combustion Engineering Standard System 80 design.
[b]Excluding axial reflectors.
[c]Average number of pins per assembly.
[d]Fissile load for initial cycle of the prebreeder.
[e]Fissile load for equilibrium cycle of the breeder.
[f]Assuming a lag time of 1 to 2 years.

Table 4-3. Fuel-assembly volume fractions: light-water seed-blanket breeder concept

Component	Fuel-assembly volume fraction: Thorium shim rods in, shutdown rods out	All control rods in	All control rods out
Fuel, seed[a]	0.308	0.308	0.308
Fuel, blanket	0.137	0.137	0.137
Coolant[b]	0.371	0.360	0.384
Structure	0.171	0.171	0.171
Control[c]	0.013	0.024	--
Total	1.000	1.000	1.000

[a]Includes pellet volume only.
[b]Includes interstitial coolant.
[c]Reactivity control obtained via movable thorium shim rods and Ag-In-Cd shutdown rods.

Table 4-4. Core-region volume fractions: light-water seed-blanket breeder concept

Component	Volume fraction[a]
Fuel, seed[b]	0.308
Fuel, blanket	0.137
Coolant	0.371
Structure	0.171
Control[c]	0.013
Total	1.000

[a]Includes active core volume only.
[b]Includes pellet volume only.
[c]Thorium shim rods inserted, shutdown rods withdrawn.

Table 4-5. Fuel-assembly volume fractions: light-water backfit prebreeder concept

Component	Volume fraction[a]
Fuel[b]	0.293
Coolant[c]	0.588
Structure	0.119
Total	1.000

[a]All control rods out.
[b]Includes pellet volume only.
[c]Includes interstitial coolant.

Table 4-6. Core-region volume fractions: light-water backfit prebreeder concept

Component	Volume fraction[a]
Fuel[b]	0.293
Coolant	0.588
Structure	0.119
Total	1.000

[a]Includes active core volume only.
[b]Includes pellet volume only.

Table 4-7. Fuel-management information: light-water backfit prebreeder concept supplying seed-blanket breeder system

Average capacity factor, %	75	
Approximate fraction of core replaced	1/3 prebreeder/batch loaded breeder[a]	
Lag time assumed between fuel discharge and recycle reload	1 yr/2 yr	
Fissile-material reprocessing loss fraction, %	1[b]	
Fissile-material fabrication loss fraction, %	1[b,c]	
Fuel requirements, ST/GWe	Prebreeder/breeder system	
	U_3O_8	ThO_2[d]
Initial core	603	76
Equilibrium reload requirement per cycle	24	2
30-year cumulative requirements[e]	3,730/4,587	270/337
50-year cumulative requirements[f]	3,989/4,889	287/356
100-year cumulative requirements[g]	4,639/5,536	331/400
Separative-work requirements, 10^3 SWU/GWe		
Initial core	601	--
Equilibrium reload	24	--
30-year cumulative requirements[e]	3,722/4,577	--
50-year cumulative requirements[f]	3,980/4,878	--
100-year cumulative requirements[g]	4,629/5,524	--
Other data for proliferation-resistance assessment	Prebreeder	Breeder
Fuel-element weight, kg	2.7	3.5
Fresh- and discharge-fuel radiation level, R/hr at 1 m	5,900	5,500
Discharge-fuel energy-gene[illegible] rate after 90-day cooling, W[illegible] element	49	28

[a]Fuel management of breeder is feasible and could result in reduced fissile-fuel requirements.

[b]Losses consist of a total (both reprocessing and fabrication) of 2% for the first 40 years and 1% from 40 to 100 years.

[c]Uranium hexafluoride conversion loss of 0.5% added to 1.0% fabrication loss of all mined uranium.

[d]Thorium dioxide requirements do not include axial and radial reflectors.

[e]Requirement for 11.5/14.1 years of prebreeders, 18.5/15.9 years of breeders.

[f]Requirement for 12.3/15.1 years of prebreeders, 37.7/34.9 years of breeders.

[g]Requirement for 14.3/17.0 years of prebreeders, 85.7/83.0 years of breeders.

Table 4-8. Fuel inventory at the beginning and end of equilibrium cycle: light-water backfit prebreeder concept supplying seed-blanket breeder concept

Isotope	Inventory (kg): Initial cycle,[a] prebreeder		Inventory (kg): initial cycle, mixed breeder		Inventory (kg): Equilibrium cycle, breeder	
	BOEC	EOEC	BOEC	EOEC	BOEC	EOEC
Thorium-232[b]	29,977	29,164	163,864	161,749	163,665	161,545
Protactinium-233	0	41	0	113	0	113
Uranium-232	0	--	--	--	--	--
Uranium-233	0	375	3,055	3,409	4,296	4,195
Uranium-234	64	88	648	762	2,338	2,349
Uranium-235	1,261	350	2,567	1,871	680	670
Uranium-236	0	160	1,179	1,276	632	634
Uranium-238	31	[illegible]6	191	170	--	--
Plutonium-238	0	--	0	--	0	--
Plutonium-239	0	--	0	--	0	--
Plutonium-240	0	--	0	--	0	--
Plutonium-241	0	--	0	-	0	--
Plutonium-242	0	--	0	.	0	--
Fission products	0	--	0	2,050	0	2,050

[a]Fuel inventory for the one-third core fueled and the one-third core discharged each year of the equilibrium prebreeder cycle. Abbreviations: BOEC, beginning-of-equilibrium cycle; EOEC, end-of-equilibrium cycle.

[b]Does not include axial or radial reflectors.

Table 4-9. Backfit-prebreeder charge and discharge data: light-water backfit prebreeder concept supplying seed-blanket breeder concept[a]

Year	Quantity (kg)								
	Th-232[b]	Pa-233	U-233	U-234	U-235	U-236	U-238	Pu, Np isotopes	Total
0[c]	29,977	0	0	64	1,261	0	31	0	31,333
1.0[d]	29,164	41	375	88	350	160	26	--	30,204

[a]Fuel inventory for the one-third core fueled and the one-third core discharged each year of the equilibrium prebreeder cycle.

[b]Does not include axial or radial reflectors.

[c]Beginning of cycle.

[d]End of cycle.

Table 4-10. Light-water seed-blanket-breeder charge and discharge data: light-water backfit prebreeder concept supplying seed-blanket breeder concept[a,b]

Year[c]		Quantity (kg)							
		Th-232[d]	Pa-233	U-233	U-234	U-235	U-236	Total	Fission products
0,	BOC	163,860	0	3,055	648	2,567	1,179	171,309	0
2.5									
	EOC	161,810	113	3,409	762	1,871	1,276	169,241	2,050
	BOC	164,190	0	3,431	742	1,823	1,243	171,429	0
5.0									
	EOC	162,110	114	3,626	862	1,345	1,288	169,345	2,050
	BOC	164,290	0	3,693	852	1,328	1,272	171,435	0
7.5									
	EOC	162,190	115	3,776	973	1,008	1,281	169,343	2,050
	BOC	164,320	0	3,879	970	1,005	1,277	171,451	0
10.0									
	EOC	162,220	116	3,883	1,087	794	1,263	169,363	2,050
	BOC	164,300	0	4,016	1,091	809	1,270	171,486	0
12.5									
	EOC	162,190	116	3,970	1,201	671	1,242	169,390	2,050
	BOC	164,210	0	4,106	1,206	688	1,250	171,460	0
15.0									
	EOC	162,110	116	4,027	1,309	597	1,215	169,374	2,050
	BOC	164,200	0	4,155	1,312	608	1,220	171,495	0
17.5									
	EOC	162,100	116	4,053	1,407	550	1,181	169,407	2,050
	BOC	164,160	0	4,184	1,410	563	1,187	171,504	0
20.0									
	EOC	162,050	116	4,068	1,498	527	1,146	169,405	2,050
	BOC	164,080	0	4,214	1,504	553	1,158	171,509	0
22.5									
	EOC	161,980	116	4,093	1,584	527	1,119	169,419	2,050
	BOC	163,980	0	4,254	1,593	565	1,136	171,528	0

Table 4-10. Light-water seed-blanket-breeder charge and discharge data: light-water backfit prebreeder concept supplying seed-blanket breeder concept[a,b] (continued)

Year[c]		Quantity (kg) Th-232[d]	Pa-233	U-233	U-234	U-235	U-236	Total	Fission products
25.0									
	EOC	161,880	115	4,133	1,666	543	1,099	169,436	2,050
	BOC	163,960	0	4,261	1,669	553	1,104	171,547	0
27.5									
	EOC	161,860	115	4,135	1,735	540	1,068	169,453	2,050
	BOC	163,900	0	4,270	1,739	557	1,076	171,542	0
30.0	EOC	161,810	115	4,144	1,800	547	1,042	169,565	2,050

[a]Charge and discharge data correspond to 30 years of continuous operation of the breeder with no out-of-loads or losses for reprocessing and fabrication.

[b]Because of the small amount of uranium-238 present in the uranium mixture and the stage of the concept development, the uranium-238 chain has not been included in the nuclear calculations.

[c]Abbreviations: BOC, beginning of cycle; EOC, end of cycle.

[d]Does not include axial or radial reflectors.

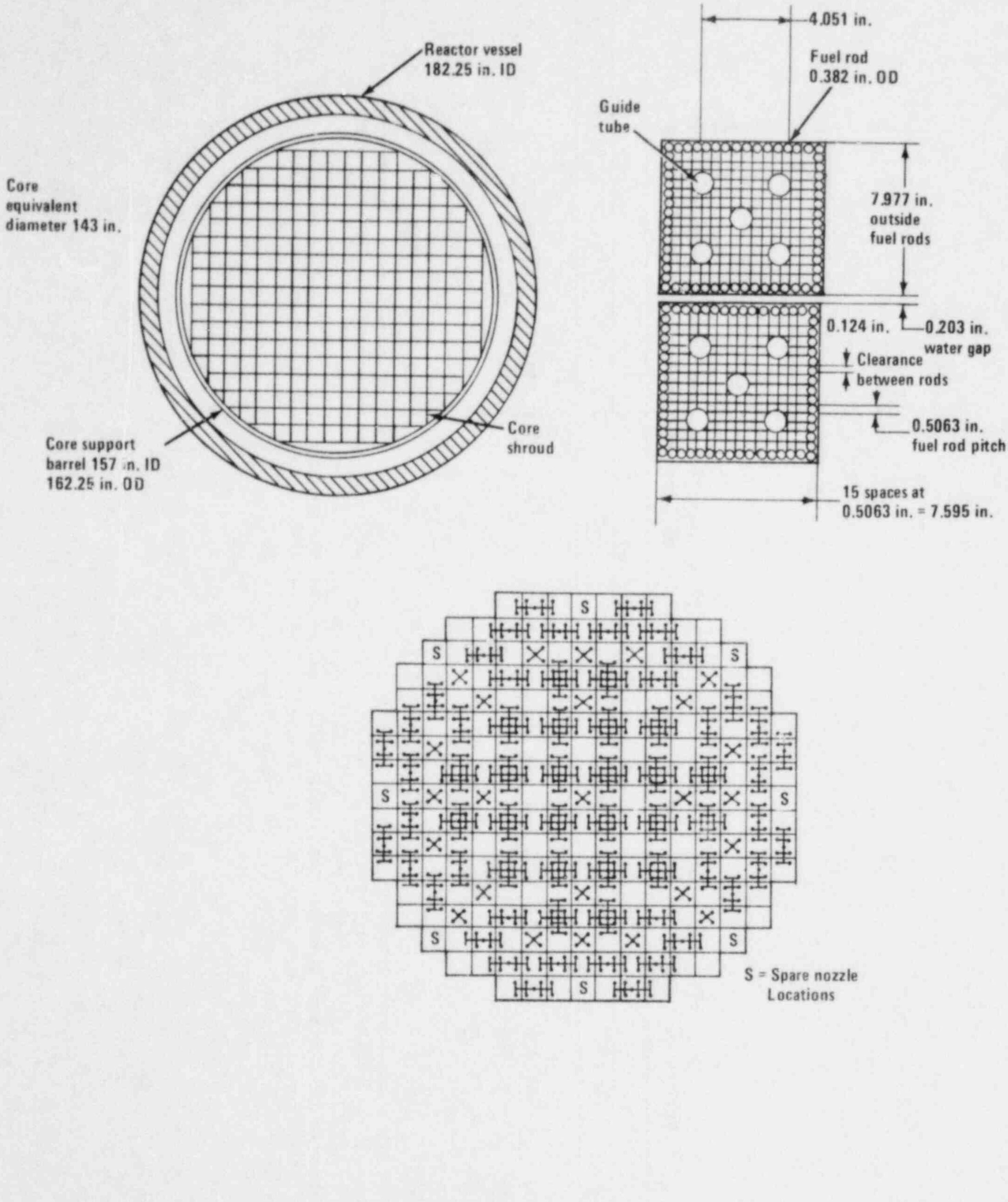
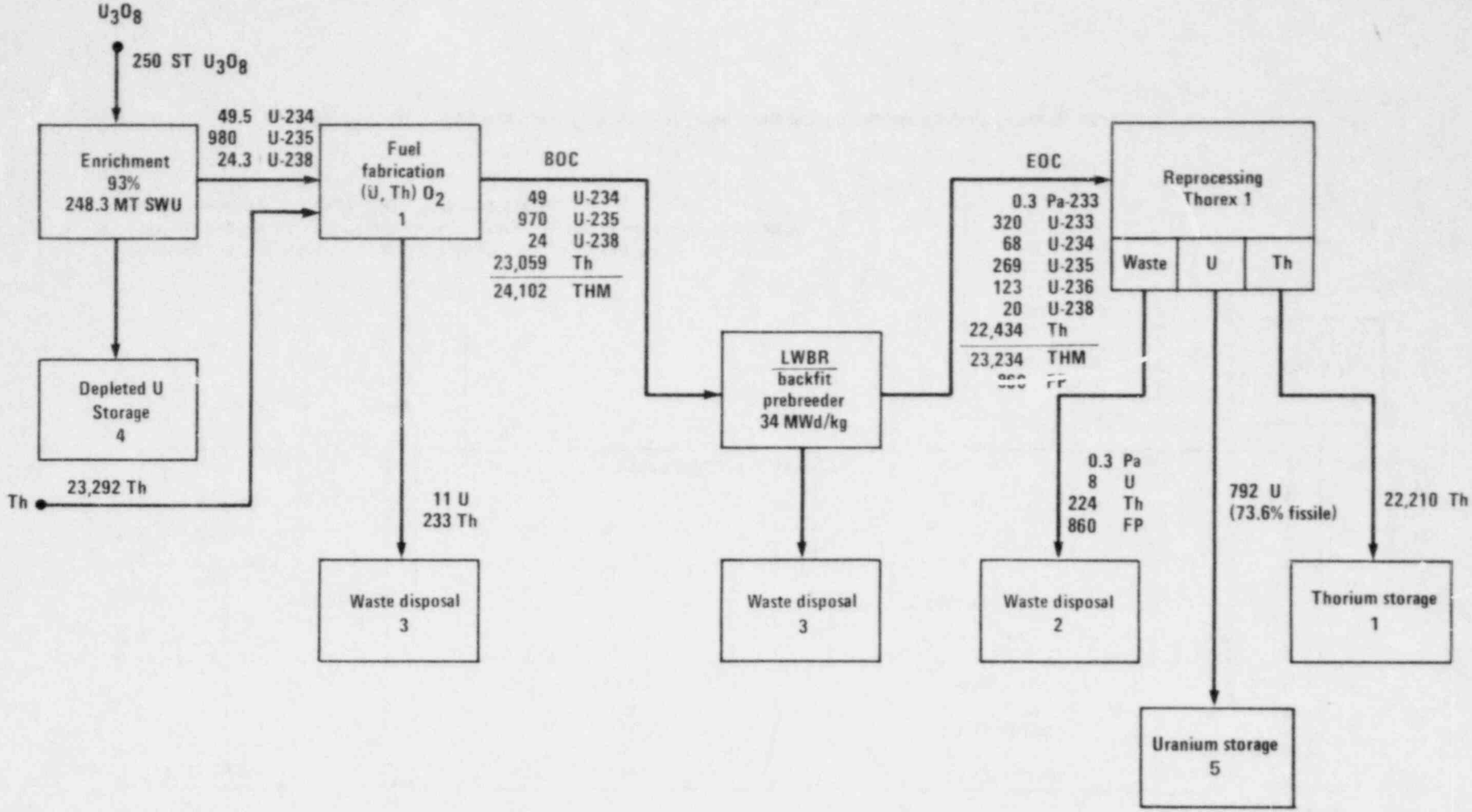


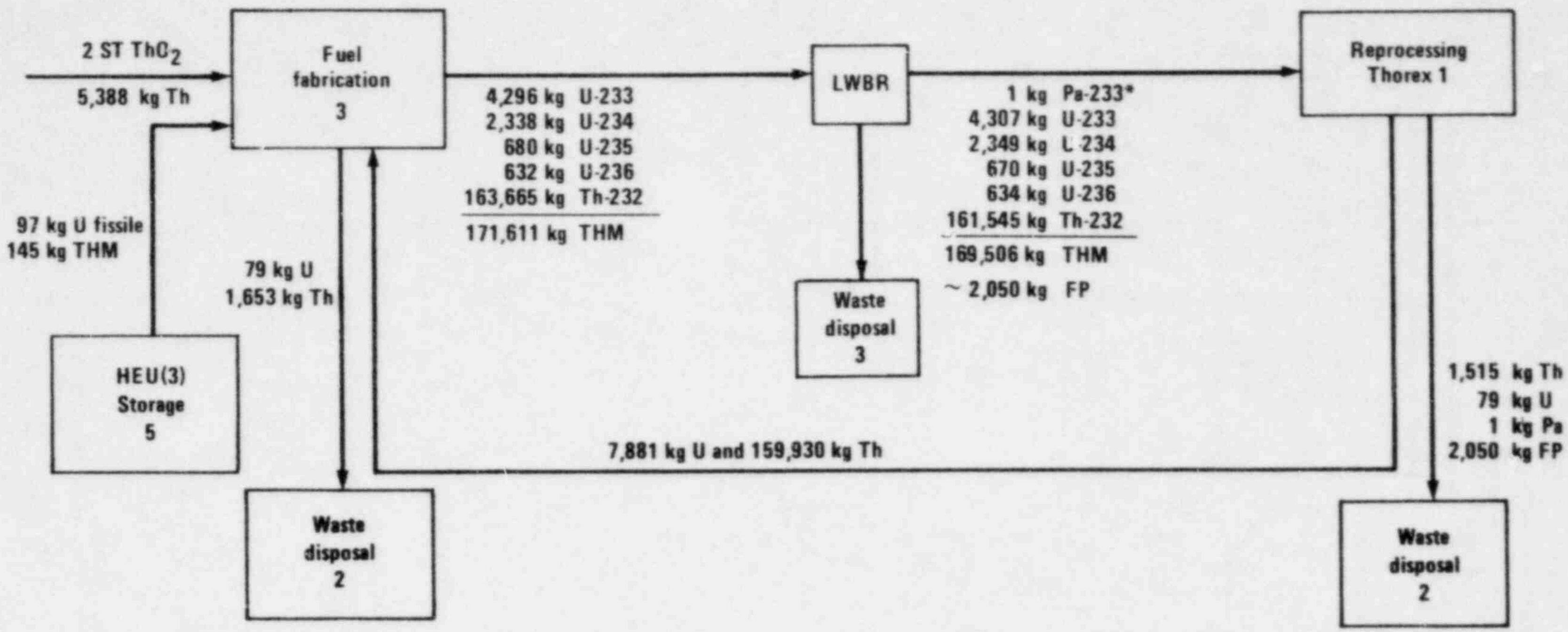
Figure 4-1. Reactor core cross section and standard control element assembly pattern (89 CEAs) for backfit prebreeder concept.



Mass flows in kg per 0.75 GWe-yr.

Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products; THM, total heavy metal.

Figure 4-2. Typical LWBR material flow diagram for a high-enrichment uranium-235/thorium fuel backfit prebreeder concept.



Note: Above equilibrium values are for 0.75 GWe for 2.5-year cycle
*Six-month protactinium-233 decay between reactor shutdown and start of reprocessing.
Abbreviations: FP, Fission Products; THM, total heavy metal.

Figure 4-3. Typical LWBR material flow diagram for seed-blanket breeder concept.

4.2 FUEL MECHANICAL, NUCLEAR, THERMAL-HYDRAULIC, AND MATERIALS CONSIDERATIONS

The light-water backfit prebreeder/seed-blanket breeder system would rely on technology already developed by commercial PWR vendors and the LWBR program. Thus, the feasibility of fulfilling mechanical, thermal-hydraulic, and materials requirements has been established. Both the prebreeder and breeder are equivalent to current commercial PWRs in terms of core fuel geometry (rods), control concept (soluble boron and individual poison rods driven by mechanisms located above the core), and the conventional use of light water as a moderator and coolant. The features that make both the backfit prebreeder or breeder different from current PWRs are discussed below.

4.2.1 URANIUM/THORIUM FUEL

Although not used in current commercial LWRs, the fuel for this conceptual system has been developed and is being proved by test and operation (Shippingport LWBR). The fuel rods for the prebreeder and breeder and the blanket rods for the breeder would be essentially identical with those used in the LWBR in terms of material composition, but the overall rod length and diameter would be larger.

4.2.2 BREEDER SEED-BLANKET CORE ARRANGEMENT

The use of discrete regions of seed and blanket rods would make the core different from current commercial LWRs but similar to the core arrangement of the Shippingport LWBR (individual fuel modules would not be separated hydraulically as in the Shippingport LWBR). Design and testing would be needed to demonstrate the adequacy of a scaled-up lower water/metal ratio to commercial size, but no new technology would be required.

4.2.3 SHIM-ONLY MECHANISMS

The seed-blanket breeder would use separate mechanisms to drive groups of individual thorium dioxide shim rods. These mechanisms would be energized at discrete intervals of depletion to fully insert or withdraw the shim rods. Since these mechanisms would have no scram function, development and production efforts might be less extensive than those associated with shutdown-type mechanisms.

4.2.4 ZIRCALOY GRIDS

It is desirable to use Zircaloy grids for the support and separation of individual fuel rods so as to reduce neutron losses in the breeder. The feasibility of using Zircaloy for grid structures of the backfit prebreeder has been proved by the design and operation of the Combustion Engineering plants. Application to the breeder configuration having a low water/metal ratio would require additional development effort.

4.2.5 NUCLEAR CALCULATIONS

The backfit prebreeder would be based on data presented in Reference 2, which is an assessment of a converter using the thorium fuel cycle. This converter differs from the proposed backfit prebreeder only in that the converter would recycle one-third of the core back into itself each year, whereas the prebreeder would recycle one-third of the core to the breeder each year. The necessary data needed for fueling the breeder and for mining requirement calculations have been extracted from Reference 2.

The nuclear-design calculational methods and cross-section data used in the nuclear analysis of the light-water seed-blanket breeder correspond to the methods and cross-section data used in the nuclear analysis of the Shippingport LWBR. Comparison of experimental and calculational results for initial LWBR physics tests shows that the LWBR calculational model has been very successful in predicting the experimental results.

The program on which the NASAP data for the seed-blanket breeder are based is in the preliminary conceptual stage. At this stage, cell theory is used to calculate the nuclear performance of the core, just as it was used in the early stages of the LWBR program. For preliminary conceptual work, cell theory is considered adequate.

The nuclear analyses of the seed-blanket design concept are based on planar NOVA/SPRITE diffusion-theory models and planar RCP Monte Carlo models of a cluster. The Monte Carlo models are used to normalize the diffusion-theory models and to determine control-rod worth and temperature defect. The diffusion-theory models are used to calculate the nuclear performance of each concept throughout the breeding cycle. The calculated conversion ratio of the diffusion-theory cluster model is modified in a conservative manner to account for neutron leakage, grid material, and structural material that is not explicitly included in the conceptual model.

4.3 ENVIRONMENTAL CONSIDERATIONS

4.3.1 SUMMARY ASSESSMENT

Typical thermal, chemical, and biocidal releases and impacts would be similar to those for the reference LWR. Typical radiological impacts are somewhat smaller than the corresponding values for the reference LWR. This system, therefore, should not experience any special difficulties in licensing for environmental reasons.

4.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM (R.G. 4.2/3.2)

The high enrichment uranium/thorium fueled, light-water backfit prebreeder/seed blanket breeder is a third LWBR scheme that may be capable of meeting current regulations, including Appendix I to 10 CFR 50. This scheme would consist of a conventional PWR vessel and plant with a modified core, the light-water backfit prebreeder, which produces fuel for a light-water seed blanket breeder. The light-water seed-blanket breeder would consist of a conventional PWR system except for a modified reactor and related equipment.

Basic parameters describing typical plants are given in Table 4-11.

4.3.3 STATION LAND USE

Information given in Section 2.3.3 for the LWBR type I module concept would apply to this concept as well.

4.3.4 STATION WATER USE (R.G. 4.2/3.3)

Information given in Section 2.3.4 for the LWBR type I module concept would apply to this concept as well.

4.3.5 HEAT-DISSIPATION SYSTEMS (R.G. 4.2/3.4)

Information given in Section 2.3.5 for the LWBR type I module concept would apply to this concept as well. There would be an approximately 5% increase in the heat-dissipation rate of the backfit prebreeder, but this is expected to be relatively inconsequential.

4.3.6 RADWASTE SYSTEMS AND SOURCE TERMS (R.G. 4.2/3.5)

Information given in Section 2.3.6 for the LWBR type I module concept would also apply to the seed-blanket-breeder concept, except for the addition of boron recycle equipment to the chemical and volume control system and an approximately 50% increase in the discharge burnup. The former would have no effect on radioactivity release rates because the boron in the coolant would be removed during normal operation. The latter would result in an approximately 25% increase in the radioactivity-release rates in the seed-blanket breeder over those of the LWBR type I module (Tables 4-12 and 4-13). The backfit prebreeder, on the other hand, would use boron in the coolant during normal operation. This would allow the formation of more radioactive tritium in the coolant from neutron interactions with boron. Furthermore, the discharge burnup in the prebreeder would be twice as great as that of the breeder, which would bring the radioactivity release rates of the backfit prebreeder in line with those of a typical PWR (Tables 3-11 and 3-12).

4.3.7 CHEMICAL AND BIOCIDAL WASTES (R.G. 4.2/5.3)

Information given in Section 2.3.7 for the LWBR type I module concept would apply to this concept as well.

4.3.8 EFFECTS OF OPERATION OF HEAT-DISSIPATION SYSTEMS (R.G. 4.2/5.1)

The thermal efficiency of the prebreeder would be the same as that of the LWR; the efficiency of the light-water seed blanket breeder would be slightly less. The effects would therefore be very slightly greater than for the reference LWR.

4.3.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATIONS (R.G. 4.2/5.2)

The dose percentages for a typical seed blanket breeder from typical liquid pathways by isotope are presented in Table 4-14; those from noble gas and from iodines and particulates are presented in Tables 4-15 and 4-16. Corresponding typical values for the light-water backfit prebreeder are given in Tables 3-13, 3-14 and 3-15. These dose contributions would be similar to, or smaller than, the corresponding values for the reference LWR and should present no licensing problems.

4.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES (R.G. 4.2/5.3)

The discharges would be similar in kind and in magnitude to those from the reference LWR. The effects would therefore also be similar.

4.3.11 OCCUPATIONAL EXPOSURE

Information given in Section 2.3.11 for the LWBR type I module concept would apply to this concept as well.

Table 4-11. Basic parameters describing the light-water backfit prebreeder concept supplying the seed-blanket breeder concept

Parameter	Backfit prebreeder	Seed-blanket breeder
Type	High-enrichment backfit light-water prebreeder	Light-water seed-blanket breeder
Fuel cycle	U-Th recycle or Pu-Th-U	U-Th breeder or Pu-Th-U
Burnup, MWd/MT	34,000	15,000
Base reactor thermal output, MWt	3,800	3,000
Electrical output, MWe	1,270	1,000
Normalized electrical output, MWe	1,000	1,000
Heat rate, Btu/kW-hr	10,000	10,200
Heat-dissipation rate at 1,000 MWe, Btu/hr	6.7×10^9	6.8×10^9

Table 4-12. Liquid-radioactive-release source terms for a typical commercial PWR plant described here with a seed-blanket breeder installed

Nuclide	Source term[a] (Ci/yr)	Nuclide	Source term[a] (Ci/yr)
Bromine-82	0.00009	Barium-137m	0.006
Bromine-83	0.0001	Cesium-138	0.00001
Rubidium-86	0.00003	Barium-139	0.00003
Strontium-89	0.0003	Barium-140	0.0001
Strontium-91	0.00006	Lanthanum-140	0.00006
Yttrium-91m	0.00001	Cerium-141	0.00001
Yttrium-91	0.0001	Praseodymium-143	0.00001
Zirconium-95	0.00001	Cerium-144	0.00003
Niobium-95	0.00001	Praseodymium-144	0.00001
Molybdenum-99	0.0001	Sodium-24	0.00006
Technetium-99m	0.0001	Phosphorus-32	0.00001
Ruthenium-103	0.00001	Phosphorus-33	0.00006
Rhenium-103m	0.00001	Chromium-51	0.00009
Tellurium-127m	0.0001	Manganese-56	0.0004
Tellurium-127	0.0003	Iron-55	0.00009
Tellurium-129m	0.0003	Iron-59	0.00009
Tellurium-129	0.0001	Cobalt-58	0.001
Iodine-130	0.0003	Cobalt-60	0.0001
Tellurium-131m	0.0003	Nickel-65	0.00001
Tellurium-131	0.00006	Niobium-92	0.00004
Iodine-131	0.09	Tin-117m	0.00001
Tellurium-132	0.006	Tungsten-185	0.00001
Iodine-132	0.006	Tungsten-187	0.0003
Iodine-133	0.05	Neptunium-239	0.0001
Iodine-134	0.00004	Protactinium-233	0.00001
Cesium-134m	0.00003	All others[b]	0.00006
Cesium-134	0.003	Total[c]	0.19
Iodine-135	0.01		
Cesium-136	0.006	Tritium	125
Cesium-137	0.006		

[a]Normalized to 1,000 MWe.

[b]Includes isotopes with discharges of less than 10^{-5} Ci/yr-unit.

[c]Does not include tritium.

Table 4-13. Gaseous-radioactive-release source terms for a typical commercial PWR plant described here with a seed-blanket breeder installed

Nuclide	Source term[a] (Ci/yr)
Krypton-83m	≤1
Krypton-85m	18
Krypton-85	475
Krypton-87	3
Krypton-88	20
Krypton-89	<1
Xenon-131m	29
Xenon-133m	50
Xenon-133	3,500
Xenon-135m	<1
Xenon-135	78
Xenon-137	<1
Xenon-139	1
Iodine-131	<0.04
Iodine-133	0.04
Tritium	220
Carbon-14	4
Particulates	0.03

[a]Normalized to 1,000 MWe.

Table 4-14. Contributions to doses by liquid effluents for a typical commercial PWR plant described here with a seed-blanket breeder installed

Nuclide	Contribution (%) to organ dose: Adult whole body	Contribution (%) to organ dose: Critical organ
Tritium	20	2
Cs-134	28	<1
Cs-136	9	<1
Cs-137	40	<1
I-131	<1	88
I-133	<1	8
Other	3	2
Ratio of dose to that from reference LWR	0.45	0.62

Table 4-15. Contributions to doses by noble gases for a typical commercial PWR plant described here with a seed-blanket breeder installed

Nuclide	Contribution (%) to organ dose	
	Whole body	Skin
Kr-83m	(a)	(a)
Kr-85m	1	1
Kr-85	(a)	19
Kr-87	1	1
Kr-88	19	8
Kr-89	1	1
Xe-131m	(a)	(a)
Xe-133m	1	2
Xe-133	66	59
Xe-135m	(a)	(a)
Xe-135	9	8
Xe-137	(a)	(a)
Ratio of dose to that from reference LWR	0.62	0.65

[a]less than 1%.

Table 4-16. Contributions to doses by radioiodines and particulates for a typical commercial PWR plant described here with a seed-blanket breeder installed

Isotope	Contribution (%) to organ dose	
	Infant thyroid	Child thyroid
I-131	97	93
I-133	1	1
C-14	1	4
Tritium	(a)	2
Ratio of dose to that from reference LWR	0.80	0.77

[a]Less than 1%.

4.4 SAFETY CONSIDERATIONS

The safety-related features typical of the backfit prebreeder/seed-blanket breeder system are shared by the current PWR reactors and the Shippingport LWBR. Natural protection against the adverse effects of reactivity addition and loss-of-flow accidents would be provided by the negative Doppler coefficient of the fuel and the negative temperature coefficient of the coolant. The backfit prebreeder and seed-blanket breeder would have a more negative beginning-of-cycle moderator-reactivity coefficient than does a typical commercial PWR. Related safety features include the reactor-protection system, the decay-heat-removal system, the emergency core-cooling system, and the containment (all systems common to other LWR plants).

4.5 LICENSING STATUS

The only features of this system that have not been proved with respect to licensability are the large-scale use of a close-packed lattice seed-blanket core design and the use of thorium dioxide in control rods. The seed rods of the breeder would be 0.44 inch in diameter, spaced on a triangular pitch of 0.51 inch. The use of this close-packed lattice may have some effect on performance in a loss-of-coolant accident when scaled to commercial size. It is anticipated that some changes in sizing and set points of the auxiliary and emergency core-cooling systems of typical LWR plants may be needed to accommodate this effect. Any uncertainty would have to be resolved by design and prototype testing.

4.6 TECHNOLOGY STATUS AND RESEARCH, DEVELOPMENT, AND DEMONSTRATION REQUIREMENTS

Since the backfit prebreeder/seed-blanket breeder system would rely on technology that has already been developed, no large research and development effort would be required to demonstrate adequacy. However, depending on how the concepts are optimized with respect to breeding and fuel-element power demand, some development and testing would be needed to resolve normal design questions and operational characteristics not directly related to safety or licensing for operation. It is anticipated that the funding and time required to complete these tests and associated developmental procedures and to demonstrate commercial feasibility for the seed-blanket breeder would be comparable to those experienced by the Shippingport LWBR project and commercial PWR manufacturers when new core designs are introduced. Significantly less time and funding would be required to resolve such questions for the backfit prebreeder because only the fuel material would be different from that of current reactors.

The research, development, and demonstration requirements are summarized in Table 4-17 for the prebreeder and in Table 4-18 for the breeder.

Table 4-17. Technological advance requirements: light-water backfit prebreeder concept

Plant component	Operating experience evaluated	Prototype or production components being manufactured	No new knowledge required	Contemporary technology with modified configuration/application
Nuclear fuel	√	√		√
Reactivity-control systems	√	√	√	
Reactor vessel	√	√	√	
Core-support structure	√	√	√	
Reactor-vessel internals, including shielding, ducting, control-rod guides, and baffles	√	√	√	
Primary-coolant pumps and auxiliary systems	√	√	√	
Primary-coolant chemistry/ radiochemistry control	√	√	√	
Primary-system heat exchangers	√	√	√	
Reactor instrumentation	√	√	√	
Emergency core-cooling/safe-shutdown systems	√	√	√	
Containment, containment cleanup systems, and effluent-control systems	√	√	√	
Other accident-mitigating systems i.e., plant-protection systems	√	√	√	
On-site fuel-handling and storage/ shipping equipment	√	√	√	
Main turbine	√	√	√	
Other critical components, if any	√	√	√	
Balance-of-plant components	√	√	√	

Table 4-18. Technological advance requirements: light-water seed-blanket breeder concept

Plant component	Operating experience evaluated	Prototype or production components being manufactured	No new knowledge required	Contemporary technology with modified configuration/application	Modest improvement in perforamnce or size and modified configuration/application
Nuclear fuel	√	√		√	
Reactivity-control systems					√
Reactor vessel			√		
Core-support structure					√
Reactor-vessel internals, including shielding, ducting, control-rod guides, and baffles				√	
Primary-coolant pumps and auxiliary systems	√	√	√		
Primary-coolant chemistry/ radiochemistry control				√	
Primary-system heat exchangers	√	√	√		
Reactor instrumentation	√	√	√		
Emergency core-cooling/safe-shutdown systems				√	
Containment, containment cleanup systems, and effluent-control systems	√	√	√		
Other accident-mitigating systems i.e., plant-protection systems	√		√		
On-site fuel-handling and storage/ shipping equipment				√	
Main turbine	√		√		
Other critical components, if any	√		√		
Balance-of-plant components	√	√	√		

REFERENCES FOR CHAPTER 4

1. Combustion Engineering, Inc., System 80--Preliminary Safety Analysis Report, (CESSAR), Standard PWR-NSSS, NRC Docket STN 50-470.

2. Electric Power Research Institute, Assessment of Thorium Fuel Cycles in Pressurized Water Reactors, EPRI NP-359, February 1977.

Chapter 5

LIGHT-WATER BACKFIT LOW-GAIN CONVERTER USING MEDIUM-ENRICHMENT URANIUM, SUPPLYING A LIGHT-WATER BACKFIT HIGH-GAIN CONVERTER

5.1 GENERAL DESCRIPTION OF THE CONCEPT

The concept described here would use two different types of cores in essentially the same reactor; they are referred to as the low-gain converter and the high-gain converter. The low-gain converter would be fueled with moderately enriched uranium (<20 wt% uranium-235) and thorium; it would produce uranium-233, which would be used, with thorium, to fuel the high-gain converter. The transition from the low-gain converter to the high-gain converter would occur at the beginning of the 14th year of operation.

This concept would use movable-fuel control (lifting and lowering of fuel rods) and is illustrative of a core that could be backfitted into the vessel of a reference pressurized water reactor (PWR) and produce the same power. The reference PWR is taken here to be the Babcock & Wilcox 205 design (Ref. 1). This core would operate in two phases: a low-gain converter phase and a high-gain converter phase. Geometrically, the core would be the same in both phases. The low-gain converter phase would act as a prebreeder.

The low-gain converter phase would be fueled with moderately enriched uranium and thorium, and while operating to generate electricity would also generate significant quantities of uranium-233, which could be used in the high-gain converter or in a light-water breeder reactor (LWBR). During operation at power, the low-gain converter would use movable fuel for reactivity control rather than poison rods or soluble boron; therefore, the rate of uranium-233 production would be higher than that for any of the prebreeder concepts that use poison rods or soluble boron for reactivity control during operation at power.

The high-gain-converter phase would be fueled with uranium-233 and thorium, and would also use movable fuel for reactivity control during operation at power. The high-gain converter would have about twice the power density of the Shippingport LWBR core and would also have a lower fuel-to-coolant ratio; the combination of these features would preclude breeding.

The technology for this concept would be very similar to that for the LWBR core at Shippingport. The deployment of this concept would therefore require only a modest extension of LWBR technology and would involve relatively few new licensing, safety, and environmental considerations.

The conceptual high-gain converter phase would not be self-sustaining and would require makeup uranium-233. Once the high-gain converters were on an equilibrium cycle, one low-gain converter would be required to support about five high-gain converters. Mining of uranium would be required to support the low-gain converter. However, the annual uranium mining requirement for six cores using the present concept (one low-gain converter plus five high-gain converters) would be 20 to 25% of that for six conventional PWR cores.

Tables 5-1 through 5-4 present the parameters for the backfit low-gain and backfit high-gain converters: Table 5-1 gives the generalized reactor performance specifications, Table 5-2 presents reactor design data, and Tables 5-3 and 5-4 give the fuel-assembly and core-region volume fractions, respectively.

The concept described here is similar to the Shippingport LWBR core in that the modules would be hexagonal, the core would be surrounded by a reflecting blanket, and reactivity control during operation at power would be achieved by lifting and lowering the movable fuel. One of the more significant differences between the present concept and the Shippingport LWBR core is the configuration of the movable fuel. In the Shippingport LWBR core, the movable fuel consists of entire cylindrical assemblies (seed assemblies) that can be lifted and lowered; these are surrounded by annular stationary blanket assemblies. In the present concept, the movable fuel would consist of individual rods, containing only thorium oxide, dispersed more or less uniformly throughout the core. These rods, called thorium oxide fingers, would be arranged much like the fingers of poison rods now used in commercial PWRs. Another significant difference from the Shippingport LWBR core is that in the present concept scram reactivity control would be supplied by poison-finger shutdown rods similar to those used in the reference PWR, and soluble boron is assumed to be used for cold shutdown. Thus, in this concept, the movable fuel (thorium oxide fingers) would provide only the reactivity control required during operation at power--namely, the reactivity to follow load changes and to compensate for long-term fission-product buildup. A typical example of the control sequence, starting with the plant cold after a refueling, would be as follows: Initially, the poison-finger rods would be fully inserted and the coolant would contain soluble boron. During plant heatup most of the soluble boron would be gradually removed from the coolant. With the plant hot, criticality would be attained by complete withdrawal of the poison-finger rods and partial withdrawal of the thorium oxide finger rods. From this point on, the movable fuel would be lifted to compensate for the buildup of long-lived fission products.

Features of the module arrangement for the present concept are shown in Figure 5-1. Features of the fuel-rod arrangement inside a module and the movable-fuel control concept are shown in Figure 5-2. There would be 115 hexagonal modules, approximately 12 inches across flats. This array of modules would be surrounded by 45 reflector blanket modules to form a nearly circular array. Each module would contain fixed seed rods, movable thorium oxide finger rods, and poison-finger shutdown rods. The thorium oxide finger rods and poison-finger shutdown rods would move inside Zircaloy-4 guide tubes. The seed rods and the guide tubes would be located on a uniform triangular pitch.

5.1.1 FUEL DESCRIPTION

The fuel would be in the form of cylindrical ceramic pellets inside Zircaloy-4 rods. The pellets in the thorium oxide finger rods would be pure high-density thorium oxide. The diameters of the fuel-rod cladding and the pellets would be between those of the seed rods and those of the blanket rods in the Shippingport LWBR core. For the movable thorium oxide finger rods, the fuel-rod cladding and the pellets would be slightly larger in diameter than the fuel rods; the inside diameter of the guide tubes would be large enough to permit movement of the rods and the flow of coolant inside the guide tubes.

Lateral motion of the fuel rods is assumed to be restrained by a system of grids similar to those used in the Shippingport LWBR core. The grids are assumed to be supported by the guide tubes for the poison-finger control rods. The grids would be Zircaloy-4 rather than the AM-350 steel grids in the Shippingport LWBR core.

Flow orificing, similar to that used in the Shippingport LWBR core, is assumed to be used to minimize pump power by reducing flow to the peripheral blanket modules to the minimum acceptable level. The orifice design has not been developed in detail.

Details of the top and bottom fuel-rod support system, the suspension system, and the core-barrel and thermal-shield arrangement are assumed to be similar to those of the standard PWR.

5.1.1.1 Fuel System for the Conceptual Low-Gain Converter

The fuel for the low-gain converter phase would consist of pellets contained in Zircaloy-4 rods. Except for the reflector blanket modules, the fuel would consist of thorium oxide pellets alternating with duplex pellets. The duplex pellets shown in Figure 5-3 would consist of a ternary mixture of oxides (UO_2-ZrO_2-CaO) in the annulus and cylindrical thorium oxide center. The thorium oxide pellets and the duplex pellets would have essentially the same outer dimensions.

In the reflector blanket modules, all of the fuel pellets would be pure thorium oxide. The enrichment of the initial core and annual make-up uranium would be slightly less than 20%. The uranium-235 content would be about 2.7% of the total heavy metal (uranium plus thorium) for the initial core and 3.2% for the annual makeup.

The design of the fuel rods would lead to thermal characteristics somewhat different from those in the reference PWR core; this point is discussed later (Section 5.2.2.2).

5.1.1.2 Fuel System for the Conceptual High-Gain Converter

The fuel for the high-gain converter phase would consist of pellets contained in Zircaloy-4 rods. Except in the reflector blanket modules, all of the fuel rods would contain pellets consisting of thorium oxide mixed with about 2% uranium oxide as a binary UO_2-ThO_2 solid solution. The uranium would be primarily uranium-233. At the top and bottom of each stack would be a short axial-reflector stack of pure thorium oxide pellets. In the reflector blanket modules, all of the fuel pellets would be pure thorium oxide.

5.1.2 REACTIVITY CONTROL

There would be one control-rod-drive mechanism above each module. In one preliminary concept of the control system, the control-rod-drive mechanism would have independently actuated concentric lead screws. The central lead screw would operate the poison-finger shutdown rods, and the annular lead screw would operate the thorium oxide finger rods. The control-drive train necessary for this would require development. An alternative to this approach would be to use smaller modules with thorium oxide finger rods and poison-finger rods in alternate modules.

For the concentric-lead-screw concept, the guide assembly is assumed to be located above the fuel assembly in the outlet plenum region. This would contain thorium oxide finger guide tubes around a central poison cluster sheath.

Individual guide tubes are assumed to be used in this assembly to protect and guide the thorium oxide finger rods. The central sheath would be an asterisk-shaped hollow column enclosing the poison-finger shutdown rods and the interconnecting "spider." The guide tubes and sheath are assumed to be attached to base plates at both ends to form a sturdy assembly, to protect the finger rods from coolant cross flow in the outlet-plenum region, and to ensure smooth insertion of the finger rods into the fuel assembly.

5.1.3 ACCOMMODATION IN EXISTING PLANTS

If this concept were backfitted into an existing plant, the only major changes required in the plant would be a new closure and new reactor-vessel internals.

5.1.4 FUEL MANAGEMENT

Fuel management for this concept would consist of replacing approximately one-third of the core (excluding peripheral blanket assemblies) annually for the low-gain converter phase and semiannually for the high-gain converter phase. Fresh modules would be installed near the periphery of the core, and the most depleted modules would be removed from near the center of the core.

For the conceptual high-gain converter phase, the two possible fuel-management alternatives would be operation to a higher discharge burnup or operation at a lower power density. Operation at a fixed power density to a higher discharge burnup would reduce the conversion ratio and would increase the mining requirements for producing makeup uranium-233. At a fixed discharge burnup, operation at a lower power density would increase the conversion ratio and would decrease the requirements for mining makeup uranium-233.

5.1.5 FUEL CYCLES

5.1.5.1 Low-Gain Converter

The use of duplex pellets in the fuel assemblies would permit separation during the reprocessing of the uranium isotopes 235, 236, and 238 present in the pellet annulus from the uranium-233 bred in the thorium oxide pellet center. The fuel assemblies would be treated in the head-end step of reprocessing to accomplish this separation, such as by selective dissolution of the pellets. The thorium oxide pellets which alternate with the duplex pellets in the fuel assemblies would also be selectively dissolved.

All the thoria from the fuel assemblies and blanket assemblies would be reprocessed to recover the uranium-233 which would be stored for use in the high-gain converter. The recovered thorium would be sent to interim storage prior to release for the high-gain converter initial core and annual reload fuel.

The uranium recovered from the annulus of the duplex pellets would be recycled to enrichment to provide part of the fuel required for the 20% enriched uranium for fabrication of the low-gain converter fuel. The plutonium recovered from the annulus of the duplex pellet would be sent to interim secure storage. A mass-flow diagram is shown in Figure 5-4.

The fuel-cycle facilities associated with this reactor/fue. .ycle combination are discussed in the following sections of Volume VII of the PSEID:

Enrichment	Chapter 3
Core fabrication 1	Chapter 4
Reprocessing (Purex 1)	Section 5.1
Reprocessing (Thorex 1)	Section 5.4
Thorium storage	Section 6.1
Plutonium storage	Section 6.2
Uranium-233 storage	Section 6.5
Waste disp ·al 2	Section 7.2
Waste disposal 3	Section 7.3

5.1.5.2 High-Gain Converter

The fuel would be fabricated from recycled uranium-233 recovered from reprocessing of high-gain converter fuel and some stored uranium-233 from reprocessing of low-gain converter fuel. The thorium for the fuel and the blankets would be from recycled thorium recovered from reprocessing of high-gain converter fuel and some new thorium. The fuel and blanket rods would be reprocessed to separate the uranium-233, thorium, and fission products. The recovered uranium-233 and thorium would be recycled to fabrication. A mass-flow diagram for a typical cycle is shown in Figure 5-5.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are discussed in the following sections of Volume VII of the PSEID:

Fabrication 1	Chapter 4
Fabrication 3	Chapter 4
Reprocessing (Thorex 1)	Section 5.4
Uranium-233 storage	Section 6.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

5.1.5.3 Quantitative Fuel Inventories

Table 5-5 summarizes the overall fuel-management information, including the separative-work requirements. Table 5-6 shows the detailed isotopic makeup of the prebreeder (low-gain converter) core, Table 5-7 shows the detailed isotopic makeup of the equilibrium cycle for the high-gain converter phase, and Tables 5-8 and 5-9 detail the overall isotopic makeup of the system over its 30-year history. Figures 5-4 and 5-5 show the isotopic mass flows for the low-gain converter and the high-gain converter phases, respectively. The quantities of isotopes in Tables 5-6 and 5-7 have been used to obtain the values shown in Figures 5-4 and 5-5, for a 1,000-MWe system operating at 0.75 annual capacity factor.

5.1.5.4 Fuel Reprocessing and Refabrication

The reprocessing of this fuel would be similar to the reprocessing of LWR fuel in that high-grade fissile material of high toxicity would be generated and handled, although the fission-product concentration in the high-gain converter fuel would be smaller than that in the LWR fuel. The uranium-232 and associated daughter products of the LWBR thorium fuel cycle give off a more penetrating radiation than do the transuranic isotopes of the LWR fuel cycle and would require more highly automated and highly shielded fabrication equipment. Short-term recycling of thorium recovered from reprocessing of thorium-containing fuels would result in highly penetrating radiation unless the thorium were allowed to decay for more than 10 years. Although the penetrating radiation from the uranium-232 accompanying uranium-233 or short-term recycled thorium would present difficulties in fabrication, in transport, and in handling at reactors, it would also deter diversion.

Table 5-1. Generalized performance specifications: light-water backfit low-gain converter using <20 wt% enriched uranium, supplying a high-gain converter

Parameter	Low-gain phase	High-gain phase
Power plant performance parameters		
Reactor thermal power output, MWt	3,900	
Net electrical power output, MWe	1,295	
Plant heat rate, Btu/kW-hr	10,280	
Core design and performance parameters		
Core heat output, MWt	3,900	
Core volume, liters[a]	37,000	
Core loading (first core), kg		
Heavy metal[a]	120,570	141,600
Fissile fuel	4,703	2,602
Conversion ratio		
Cycle average[b]	0.57	--
Average for initial cycle	--	--
Average for equilibrium cycle	--	~0.97
Fissile inventory ratio[b]	0.76	~0.94
Initial cycle	--	--
Equilibrium cycle	--	0.98
Average discharge burnup, MWd/MTHM[c]	26,600	0.96
Peak discharge burnup, MWd/MTHM[c]	~65,400	11,300
Fuel type	Fuel rods--duplex	~28,000
	pellets; control	Binary UO_2/
	fingers--ThO_2	ThO_2
Reactor inlet temperature, °F	560	
Reactor outlet temperature, °F	625	

[a]Excluding axial and radial reflectors.
[b]Includes fissile plutonium production.
[c]Heavy metal charged.

Table 5-2. Reactor design data: light-water backfit low-gain converter using <20 wt% enriched uranium, supplying a high-gain converter

Parameter	Low-gain phase		High-gain phase
Geometric information			
Core height, cm[a]		343	
Number of core enrichment zones		1	
Number of assemblies[a]		115	
Equivalent diameter, cm[a]		371	
Number of rods per assembly[b]		510/108	
Rod pitch-to-diameter ratio[b]		1.241/1.189	
Overall assembly length, cm		449	
Lattice pitch, cm		1.30	
Assembly material		Zircaloy-4 machined grids	
Cladding parameters			
Cladding outside diameter, mils[b]		412/430	
Cladding wall thickness, mils[b]		20.5/21.5	
Cladding material		Zircaloy-4	
Fissile inventory, kg[c]	5,549		2,841
External fissile inventory, kg[d]	1,961/3,922		1,963/3,926
Fissile gain or loss, kg/cycle[d]	486 (loss)		35 (loss)
Specific power, kW/kg fissile[e]	829		1,504
Specific power, kW/kg HM	32.3		27.5
Power density, kW/liter		105	

[a]Excluding axial and radial reflectors.

[b]Values shown are for both seed rods/blanket rods.

[c]At the beginning of the fourteenth cycle for the low-gain phase and at the beginning of an equilibrium cycle for the high-gain phase.

[d]Low-gain-phase value for the fourteenth cycle. High-gain-phase value for the equilibrium cycle.

[e]High-gain-phase value for initial cycle.

Table 5-3. Fuel-assembly volume fractions: light-water backfit low-gain converter using <20 wt% enriched uranium, supplying a high-gain converter

Component	Fuel-assembly volume fraction Safety rods out	Safety rods in
Fuel[a]	0.365 (seed)	0.365 (seed)
	0.085 (blanket)	0.085 (blanket)
Coolant[b]	0.384	0.373
Structure	0.166	0.166
Control[c,d]	--	0.011
Total	1.000	1.000

[a]Includes pellet volume only.
[b]Includes coolant between modules.
[c]Control obtained by means of movable thorium oxide fingers and boron shutdown rods.
[d]Values shown are for the middle of the equilibrium cycle.

Table 5-4. Core-region volume fractions[a]: light-water backfit low-gain converter using <20 wt% enriched uranium, supplying a high-gain converter

Component	Volume fraction
Fuel[b]	0.365 (seed)
	0.057 (blanket)
Coolant[c]	0.419
Structure	0.159
Total	1.000

[a]Excluding axial and radial reflectors.
[b]Includes pellet volume only.
[c]Includes coolant between modules.

Table 5-5. Fuel-management information: light-water backfit low-gain converter using <20 wt% enriched uranium, supplying a high-gain converter

Parameter	Low-gain phase			High-gain phase
Average capacity factor, %	75			75
Approximate fraction of core replaced	1/3 every year			1/3 every 6 months
Lag time assumed between fuel discharge and recycle reload[a]		1 year/2 years		
Fissile-material loss fractions[b]				
Conversion		0.005	0.005	
Fabrication		0.0	0.0	
Reprocessing		0.0	0.0	
Uranium and thorium requirements, ST/GWe		U_3O_8	ThO_2	
Initial core		929	148	
Equilibrium reload		42	0.33	
30-year cumulative requirement[a,c]		2,635/3,320	545/615	
50-year cumulative requirement		3,368/3,985	530/589	
100-year cumulative requirement		5,158/5,719	529/624	
Separative-work requirements, 10^3 SWU/GWe[d]				
Initial core		838		
Equilibrium reload		39		
30-year cumulative requirement[a]		2,460/3,099		
50-year cumulative requirement		3,158/3,737		
100-year cumulative requirement		4,858/5,385		
Other data for proliferation-resistance assessment[e]				
Fuel-element weight, kg[f]	2.2 to 2.4			2.4 to 2.6
Fuel radiation level at 1 meter (R/hr)				
Fresh fuel	0			35
Discharge fuel[g]	9,900			9,100
Discharge-fuel energy-generation rate after 90-day cooling, watts per element	52			39

[a]Fuel requirements are shown for both 1- and 2-year out-of-core time.

[b]Fabrication and reprocessing losses are assumed to be 1% each for the first 40 years of operation and 0.5% thereafter, reflecting improved recycle technology in later generation recycle plants.

[c]Assumes thorium oxide out-of-core time is 10 years during low-gain-phase operation and 1 or 2 years for high-gain-phase operation.

[d]Uranium hexafluoride conversion losses are assumed to occur during the conversion of UO_2 to UF_6. No losses are assumed for reconversion from UF_6 to UO_2.

[e]See also Tables 5-6 through 5-9.

[f]Weight of pellets in a single fuel rod.

[g]Fuel 8 months after shutdown.

Table 5-6. Fuel inventory for the low-gain converter operating with <20 wt% enriched uranium from year 13 to year 14[a]

Isotope	Quantity[b] (kg)							
	Zone 1		Zone 2		Zone 3		Center module	
	BOC	EOC	BOC	EOC	BOC	EOC	BOC	EOC
Thorium-232	32,061.1	31,819.5	31,319.5	31,582.0	31,582.0	31,345.8	843.7	837.4
Protactinium-233	0.0	33.1	10.9	31.8	10.5	31.8	0.0	1.0
Uranium-232	0.0	0.3	0.3	1.0	1.0	1.9	0.0	0.0
Uranium-233	0.0	166.1	188.3	298.5	319.8	387.1	0.0	4.4
Uranium-234	0.0	9.4	9.4	24.6	24.6	42.0	0.0	0.3
Uranium-235	1,929.0	1,613.2	1,609.7	1,244.8	1,244.8	855.6	32.2	24.1
Uranium-236	298.9	349.0	325.9	381.2	381.2	436.2	0.0	1.7
Uranium-238	7,417.4	7,280.4	7,288.9	7,154.3	7,154.3	7,021.6	128.6	126.2
Plutonium-239	0.0	85.1	85.1	117.1	117.1	135.3	0.0	1.5
Plutonium-240	0.0	10.8	10.8	22.2	22.2	29.9	0.0	0.2
Plutonium-241	0.0	4.5	4.5	17.5	17.5	32.6	0.0	0.1
Plutonium-242	0.0	0.2	0.2	1.9	1.9	5.6	0.0	0.0

[a]Excluding axial and radial reflectors.
[b]Abbreviations: BOC, beginning of cycle; EOC, end of cycle.

Table 5-7. Fuel inventory for the high-gain converter equilibrium cycle

Isotope	Quantity[a] (kg)					
	Zone 1[b]		Zone 2		Zone 3	
	BOC	EOC	BOC	EOC	BOC	EOC
Thorium-232	64,896.3	64,712.1	63,052.9	62,887.8	62,887.8	62,732.0
Protactinium-233	0.0	51.9	17.2	50.8	17.3	51.2
Uranium-232	2.2	2.2	2.1	2.1	2.1	2.0
Uranium-233	852.8	798.4	811.2	766.5	800.0	749.4
Uranium-234	352.9	351.8	342.8	338.2	338.2	331.3
Uranium-235	126.4	126.1	122.8	121.2	121.2	118.7
Uranium-236	94.2	93.9	91.5	90.3	90.3	88.5
Fission products	0.0	189.2	164.4	368.7	368.7	553.1

[a]Abbreviations: BOC, beginning of cycle; EOC, end of cycle.
[b]Includes center module.

Table 5-8. Reactor charge data[a,b]

Year	Quantity (kg)							
	Th-232	U-232	U-233	U-234	U-235	U-236	U-238	Total
0.0	151,030.6	0.0	0.0	0.0	4,703.0	0.0	18,840.5	174,574.1
1.0	50,905.8	0.0	0.0	0.0	1,913.4	0.0	7,653.5	60,472.7
2.0	50,905.8	0.0	0.0	0.0	1,913.4	0.0	7,653.5	60,472.7
3.0	50,905.8	0.0	0.0	0.0	1,921.7	52.1	7,634.8	60,514.4
4.0	50,905.8	0.0	0.0	0.0	1,926.7	82.9	7,623.8	60,539.2
5.0	50,905.8	0.0	0.0	0.0	1,939.6	163.5	7,594.7	60,603.6
6.0	50,905.8	0.0	0.0	0.0	1,939.6	163.5	7,594.7	60,603.6
7.0	50,905.8	0.0	0.0	0.0	1,939.6	163.5	7,594.7	60,603.6
8.0	50,905.8	0.0	0.0	0.0	1,945.2	199.0	7,582.0	60,632.0
9.0	50,905.8	0.0	0.0	0.0	1,948.6	219.9	7,574.4	60,648.7
10.0	50,905.8	0.0	0.0	0.0	1,957.4	274.7	7,554.7	60,692.6
11.0	50,905.8	0.0	0.0	0.0	1,957.4	274.7	7,554.7	60,692.6
12.0	50,905.8	0.0	0.0	0.0	1,957.4	274.7	7,554.7	60,692.6
13.0	50,905.8	0.0	0.0	0.0	1,961.2	298.9	7,546.0	60,711.9
14.0	192,709.0	10.5	2,557.3	243.4	34.6	2.6	0.0	195,557.4
14.5	65,353.5	3.6	867.3	82.6	11.7	0.9	0.0	66,319.5
15.0	65,353.5	3.6	867.3	82.6	11.7	0.9	0.0	66,319.5
15.5	65,353.5	3.6	867.3	82.6	11.7	0.9	0.0	66,319.5
16.0	65,353.5	3.6	867.3	82.6	11.7	0.9	0.0	66,319.5
16.5	65,331.4	3.6	865.9	100.2	17.0	1.6	0.0	66,319.7
17.0	65,309.5	3.7	864.5	117.8	22.2	2.3	0.0	66,319.9
17.5	65,287.7	3.7	863.1	135.2	27.4	3.0	0.0	66,320.1
18.0	65,287.7	3.7	863.1	135.2	27.4	3.0	0.0	66,320.1
18.5	65,287.7	3.7	863.1	135.2	27.4	3.0	0.0	66,320.1
19.0	65,287.7	3.7	863.1	135.2	27.4	3.0	0.0	66,320.1
19.5	65,287.7	3.7	863.1	135.2	27.4	3.0	0.0	66,320.1
20.0	65,269.1	3.6	863.5	149.8	30.8	3.5	0.0	66,320.3
20.5	65,250.5	3.5	864.0	164.3	34.2	3.9	0.0	66,320.5
21.0	65,232.1	3.4	864.4	178.8	37.5	4.4	0.0	66,320.7
21.5	65,232.1	3.4	864.4	178.8	37.5	4.4	0.0	66,320.7

Note: See footnotes at end of table.

Table 5-8. Reactor charge data[a,b] (continued)

Year	Quantity (kg)							
	Th-232	U-232	U-233	U-234	U-235	U-236	U-238	Total
22.0	65,232.1	3.4	864.4	178.8	37.5	4.4	0.0	66,320.7
22.5	65,232.1	3.4	864.4	178.8	37.5	4.4	0.0	66,320.7
23.0	65,232.1	3.4	864.4	178.8	37.5	4.4	0.0	66,320.7
23.5	65,213.7	3.3	864.8	193.3	40.9	4.9	0.0	66,320.9
24.0	65,198.0	3.3	864.6	205.2	44.4	5.6	0.0	66,321.0
24.5	65,183.9	3.2	863.9	215.5	48.1	6.5	0.0	66,321.2
25.0	65,183.9	3.2	863.8	215.5	48.1	6.5	0.0	66,321.2
25.5	65,183.9	3.2	863.9	215.5	48.1	6.5	0.0	66,321.2
26.0	65,183.9	3.2	863.9	215.5	48.1	6.5	0.0	66,321.2
26.5	65,183.9	3.2	863.9	215.5	48.1	6.5	0.0	66,321.2
27.0	65,170.1	3.1	863.2	225.6	51.8	7.4	0.0	66,321.3
27.5	65,157.5	3.0	862.6	233.9	55.5	8.9	0.0	66,321.4
28.0	65,145.2	3.0	861.7	242.2	59.2	10.3	0.0	66,321.6
28.5	65,145.2	3.0	861.7	242.2	59.2	10.3	0.0	66,321.6
29.0	65,145.2	3.0	861.7	242.2	59.2	10.3	0.0	66,321.6
29.5	65,145.2	3.0	861.7	242.2	59.2	10.3	0.0	66,321.6

[a]High-gain phase begins at 14 years. Lag time between fuel discharge and recycle reload is assumed to be 2 years.

[b]Data are based on 1,295-MWe plant with an annual capacity factor of 0.75.

Table 5-9. Reactor discharge data[a,b]

Year	Quantity (kg)													
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Fission products
1.0	50,624.8	34.6	0.9	192.0	10.9	938.1	65.1	4,924.0	57.5	7.3	3.1	0.2	56,858.4	380
2.0	50,387.4	34.2	1.7	324.4	26.2	911.6	136.4	6,167.4	103.6	19.6	15.4	1.7	58,129.4	745
3.0	50,150.9	34.2	2.4	413.0	43.5	835.9	204.4	7,241.2	138.6	30.5	33.1	5.7	59,133.3	~1,115
4.0	50,150.9	34.2	2.4	413.0	43.5	835.9	204.4	7,241.2	138.6	30.5	33.1	5.7	59,133.3	~1,115
5.0	50,150.9	34.2	2.4	413.0	43.5	835.9	204.4	7,241.2	138.6	30.5	33.1	5.7	59,133.3	~1,115
6.0	50,150.9	34.2	2.4	413.0	43.5	844.2	248.7	7,223.5	138.3	30.4	33.0	5.7	59,167.7	~1,115
7.0	50,150.9	34.2	2.4	413.0	43.5	849.2	274.9	7,213.1	138.1	30.4	33.0	5.7	59,188.3	~1,115
8.0	50,150.9	34.2	2.4	413.0	43.5	862.1	343.4	7,185.6	137.5	30.3	32.8	5.7	59,241.3	~1,115
9.0	50,150.9	34.2	2.4	413.0	43.5	862.1	343.4	7,185.6	137.5	30.3	32.8	5.7	59,241.3	~1,115
10.0	50,150.9	34.2	2.4	413.0	43.5	862.1	343.4	7,185.6	137.5	30.3	32.8	5.7	59,241.3	~1,115
11.0	50,150.9	34.2	2.4	413.0	43.5	867.7	373.6	7,173.6	137.3	30.2	32.8	5.6	59,264.7	~1,115
12.0	50,150.9	34.2	2.4	413.0	43.5	871.9	391.3	7,166.4	137.1	30.2	32.8	5.6	59,278.4	~1,115
13.0	50,150.9	34.2	2.4	413.0	43.5	879.9	437.9	7,147.8	136.8	30.1	32.7	5.6	59,314.7	~1,115
14.0	149,521.2	102.2	4.1	899.3	78.9	3,737.7	1,168.1	21,582.5	339.0	63.1	54.7	7.7	177,558.8	~1,115
14.5	65,143.6	55.1	3.6	810.4	100.2	17.0	1.6	0.0	0.0	0.0	0.0	0.0	66,131.5	189.0
15.0	64,952.1	55.2	3.6	797.6	115.7	21.8	2.2	0.0	0.0	0.0	0.0	0.0	65,948.2	373.2
15.5	64,769.9	55.6	3.6	777.3	129.5	26.2	2.8	0.0	0.0	0.0	0.0	0.0	65,764.9	557.4
16.0	64,769.9	55.6	3.6	777.3	129.5	26.2	2.8	0.0	0.0	0.0	0.0	0.0	65,764.9	557.4
16.5	64,769.9	55.6	3.6	777.3	129.5	26.2	2.8	0.0	0.0	0.0	0.0	0.0	65,764.9	557.4
17.0	65,769.3	55.6	3.6	777.3	130.0	26.3	2.9	0.0	0.0	0.0	0.0	0.0	65,764.9	557.4
17.5	64,768.8	55.6	3.6	777.2	130.4	26.5	2.9	0.0	0.0	0.0	0.0	0.0	65,764.9	557.4
18.0	64,750.9	55.4	3.5	777.7	144.5	29.7	3.3	0.0	0.0	0.0	0.0	0.0	65,765.1	557.5
18.5	64,733.6	55.3	3.4	778.1	158.2	32.9	3.8	0.0	0.0	0.0	0.0	0.0	65,765.2	557.5
19.0	64,716.3	55.2	3.3	778.5	171.8	36.0	4.2	0.0	0.0	0.0	0.0	0.0	65,765.4	557.5
19.5	64,716.3	55.2	3.3	778.5	171.8	36.0	4.2	0.0	0.0	0.0	0.0	0.0	65,765.4	557.5
20.0	64,716.3	55.2	3.3	778.5	171.8	36.0	4.2	0.0	0.0	0.0	0.0	0.0	65,765.4	557.5
20.5	64,715.8	55.2	3.3	778.6	172.1	36.1	4.2	0.0	0.0	0.0	0.0	0.0	65,765.4	557.5
21.0	64,715.4	55.2	3.3	778.6	172.5	36.2	4.2	0.0	0.0	0.0	0.0	0.0	65,765.4	557.5
21.5	64,697.7	55.0	3.2	779.0	186.4	39.4	4.7	0.0	0.0	0.0	0.0	0.0	65,765.5	557.5

Note: See footnotes at end of table.

Table 5-9. Reactor discharge data[a,b] (continued)

Year	Quantity (kg)												Fission products
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	
22.0	64,682.6	54.9	3.1	779.2	197.7	42.8	5.4	0.0	0.0	0.0	0.0	0.0	65,765.7	557.5
22.5	64,668.7	54.8	3.1	779.2	207.5	46.3	6.2	0.0	0.0	0.0	0.0	0.0	65,765.8	557.5
23.0	64,668.7	54.8	3.1	779.2	207.5	46.3	6.2	0.0	0.0	0.0	0.0	0.0	65,765.8	557.5
23.5	64,668.7	54.8	3.1	779.2	207.5	46.3	6.2	0.0	0.0	0.0	0.0	0.0	65,765.8	557.5
24.0	64,668.3	54.8	3.1	779.2	207.8	46.4	6.2	0.0	0.0	0.0	0.0	0.0	65,765.8	557.5
24.5	64,667.9	54.8	3.1	779.2	208.1	46.5	6.2	0.0	0.0	0.0	0.0	0.0	65,765.8	557.6
25.0	64,653.9	54.7	3.0	779.2	217.9	50.1	7.2	0.0	0.0	0.0	0.0	0.0	65,765.9	557.6
25.5	64,642.0	54.6	2.9	778.6	225.8	53.5	8.5	0.0	0.0	0.0	0.0	0.0	65,766.0	557.6
26.0	64,630.6	54.5	2.9	777.8	233.5	57.0	9.9	0.0	0.0	0.0	0.0	0.0	65,766.1	557.6
26.5	64,630.6	54.5	2.9	777.8	233.5	57.0	9.9	0.0	0.0	0.0	0.0	0.0	65,766.1	557.6
27.0	64,630.6	54.5	2.9	777.8	233.5	57.0	9.9	0.0	0.0	0.0	0.0	0.0	65,766.1	557.6
27.5	64,630.3	54.5	2.9	777.8	233.7	57.1	9.9	0.0	0.0	0.0	0.0	0.0	65,766.1	557.6
28.0	64,630.0	54.5	2.9	777.8	233.9	57.2	10.0	0.0	0.0	0.0	0.0	0.0	65,766.1	557.6
28.5	64,618.4	54.4	2.8	776.9	241.8	60.7	11.3	0.0	0.0	0.0	0.0	0.0	65,766.3	557.6
29.0	64,607.1	54.3	2.8	776.1	249.4	64.1	12.7	0.0	0.0	0.0	0.0	0.0	65,766.4	557.6
29.5	64,596.1	54.2	2.7	775.1	256.9	67.4	14.0	0.0	0.0	0.0	0.0	0.0	65,766.5	557.7
30.0	64,596.1	54.2	2.7	775.1	256.9	67.4	14.0	0.0	0.0	0.0	0.0	0.0	65,766.5	557.7

[a]High-gain phase begins at 14 years. Lag time between fuel discharge and recycle reload is assumed to be 2 years.
[b]Data are based on 1,295-MWe plant with an annual capacity factor of 0.75.

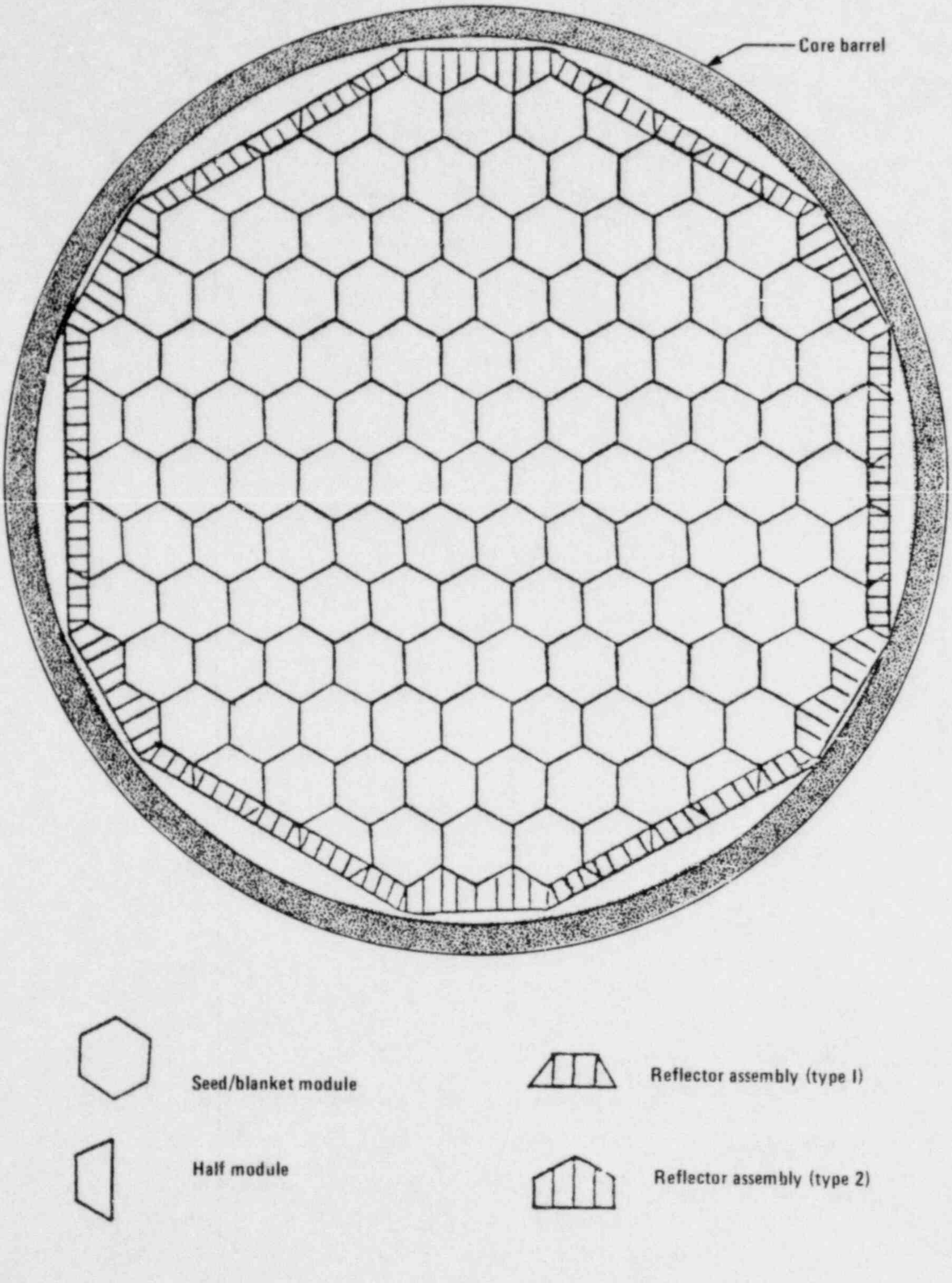


Figure 5-1. Full-core cross section of backfit converter (low-gain phase and high-gain phase).

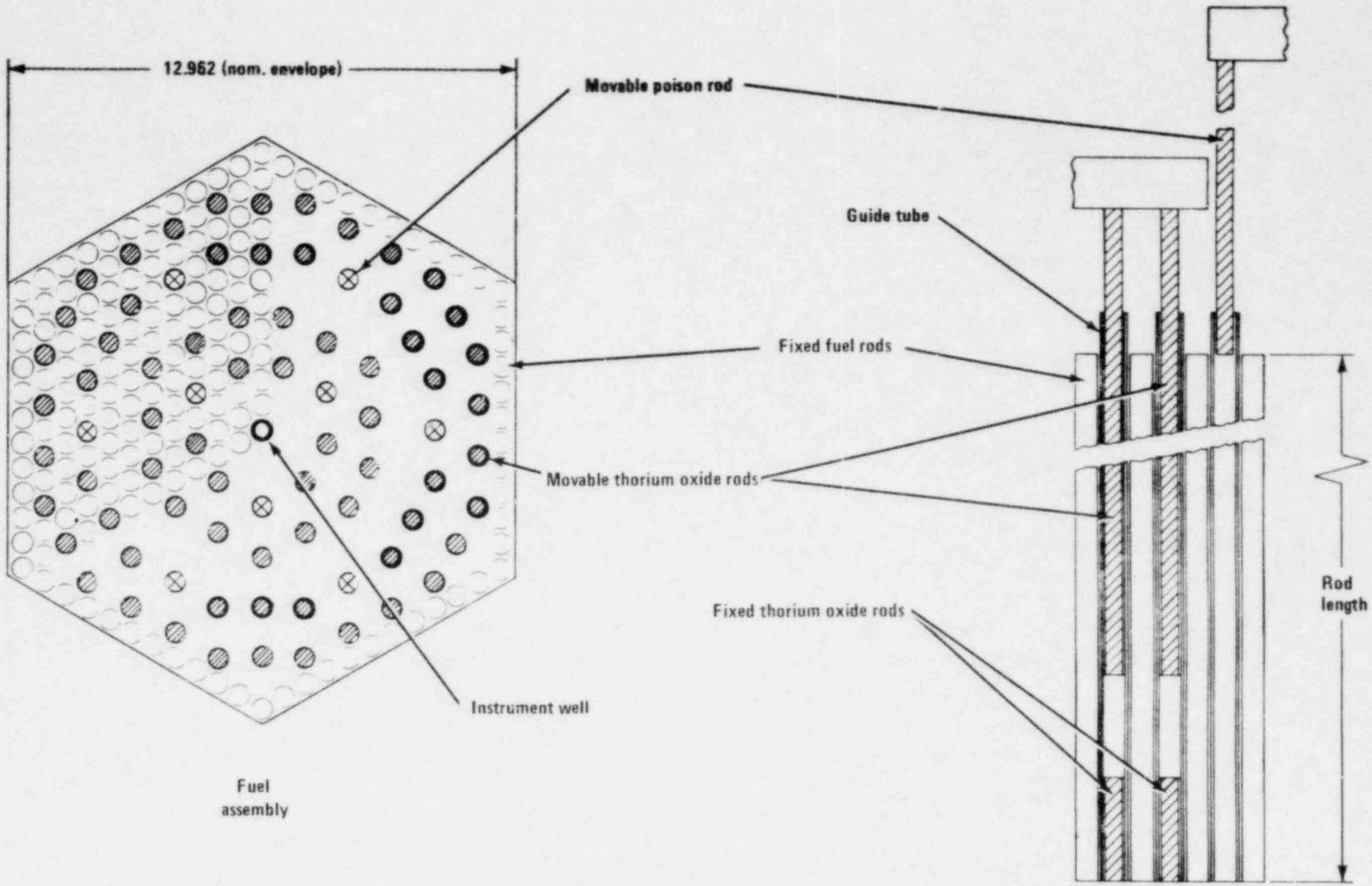


Figure 5-2. Module cross section of backfit converter (low-gain phase and high-gain phase)

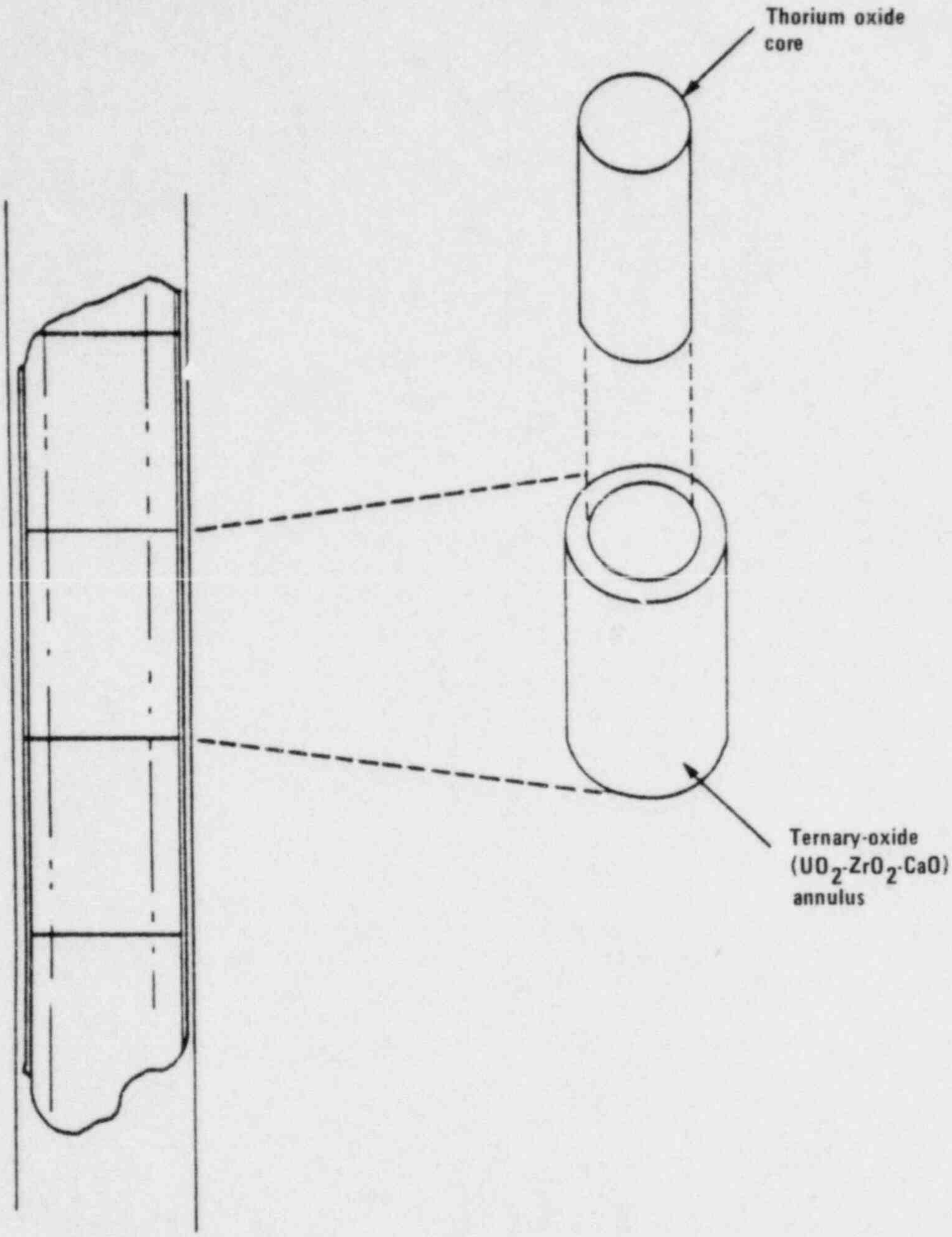
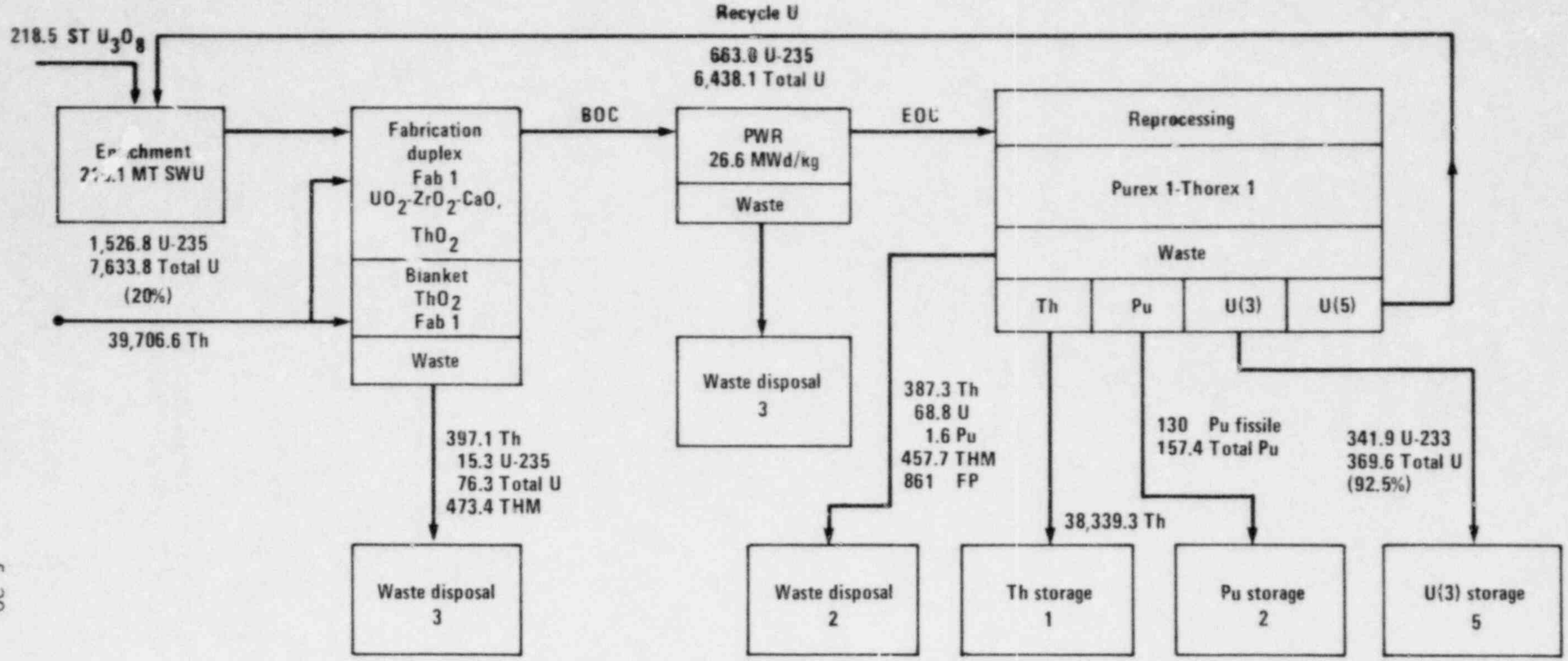


Figure 5-3. Low-gain-phase duplex fuel pellets to be used in place of high-gain-phase UO_2-ThO_2 binary fuel pellets.

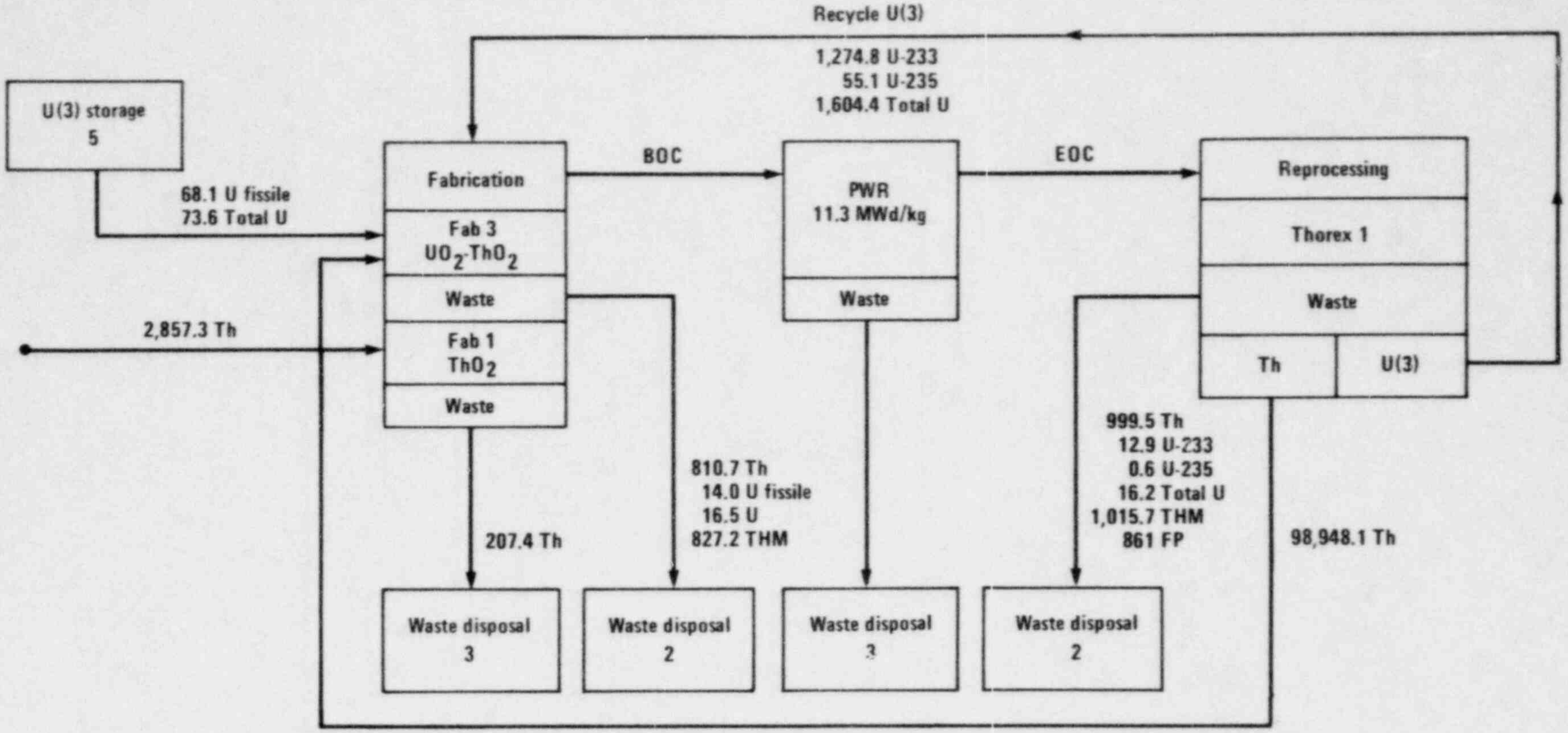


Notes:

1. Mass flows are in kg per 0.75 GWe-yr.
2. Data base: Addendum to NASAP PSEID Vol. III by ANL (February 22, 1979), normalized from a 1,295-MWe reactor; charge data, year 10; discharge data averaged for years 10, 11, and 12.
3. Recovered U-233 assumed to be 92.5% fissile.
4. Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products; THM, total heavy metal.

	BOC	EOC
Thorium	39,309.5	38,726.6
U-233	–	345.3
U-235	1,511.5	669.7
Total U	7,557.4	6,876.5
Pu fissile	–	131.3
Total Pu	–	159.0
THM	46,866.9	45,762.1
FP	–	861

Figure 5-4. Typical LWBR mass-flow diagram for MEU (5)-Th, Th low-gain converter.



Notes:

1. Mass flows are in kg per 0.75 GWe-yr.
2. Fissile pellets 2% UO_2.
3. Makeup U-233 is 92.5% fissile (from prebreeder).
4. Data base: NASAP PSEID Vol. III, LWBR addendum by ANL (February 22, 1979), normalized from a 1,295-MWe reactor, annual requirements for year 20.
5. Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products; THM, total heavy metal.

	BOC	EOC
Thorium	100,787.3	99,947.6
U-233	1,334	1,287.7
U-235	50.2	55.7
Total U	1,037.9	1,620.6
THM	102,425.2	101,568.2
FP		861

Figure 5-5. Typical LWBR mass-flow diagram for HEU(3)-Th, Th high-gain converter.

5.2 FUEL, MECHANICAL, THERMAL-HYDRAULIC, AND MATERIALS CONSIDERATIONS

5.2.1 MECHANICAL CONSIDERATIONS

This section identifies the principal unique mechanical features of the concept. Preliminary reviews indicate that it may be feasible to design and manufacture these reactors using existing design and testing methods, existing manufacturing capabilities, and proven materials. Additional testing would, however, be required to confirm this judgment.

5.2.1.1 Common Considerations

This concept would have drier lattice than the standard PWR. To avoid rod-to-rod contact and to ensure adequate cooling, the fuel rods would have to be placed on a triangular pitch. This pitch arrangement could be accommodated with a minimum amount of structure in a hexagonal module. Because of the drier lattice of the high-gain converter, the weight density (weight per square foot of radial area) would be higher, the core pressure drop would be higher, and the hydraulic lifting pressure would be higher than those in the standard PWR.

There would be two types of movable rods: poison rods and thorium oxide finger rods. As uranium-233 was bred in the thorium oxide finger rods, these rods would begin to produce heat. Thus, some guide tubes (thorium oxide) would be heated while others (poison) would not, and a differential thermal expansion would develop between the guide tubes. The design of the thorium oxide finger rods and guide tubes would have to account for the fissioning of bred uranium-233 in the thorium oxide and the effects of thermal expansion and fuel growth.

The growth of the thorium oxide and the resulting stress on fuel rods are expected to be different from those in the reference PWR.

In addition to the guide tubes for the thorium oxide finger rods and the poison-finger shutdown rods, each fuel assembly would contain one instrument guide tube. The grids, which provide lateral support and alignment for the fuel rods and guide tubes, would be attached only to the poison-rod guide tubes and the instrument guide tube. The poison-rod and instrument guide tubes would provide the primary vertical structure of the assembly and would support the axial loads occurring during core operation.

The number and wall thickness of the poison-rod guide tubes and the wall thickness of the instrument guide tube are variables that could be adjusted on the basis of the calculated loads generated by the fuel assembly during core operation.

The control-rod-drive mechanisms must provide two functions--complete withdrawal and scram of the poison finger shutdown rods and lowering of the thorium oxide finger rods.

One preliminary concept for the control-rod-drive mechanism provides for a single mechanism with two concentric lead screws and two sets of roller nuts. The poison-finger rods would be lifted by the inner lead screw, and the thorium oxide finger rods would be lifted by the outer lead screw.

An alternative control-rod-drive mechanism would be a conventional single-acting drive with scram capability. This would be used in a core with smaller modules having poison-finger rods and thorium oxide finger rods in alternate modules. Such a design would lead to somewhat higher power peaking factors.

Conceptually, the fuel-assembly support structure and the plenum assembly for the core would be similar to those for the standard PWR. The core barrel would be suspended from the reactor-vessel closure flange. The core basket, fuel assemblies, lower support, and flow distributor would be supported by the core barrel. The fuel assemblies would rest on the lower support, which would also provide radial alignment. The flow distributor would contain perforated plates and would be below the lower support. The plenum assembly would contain a fuel-assembly alignment plate, a plenum barrel, and a plenum cover. The primary mechanical consideration for the design of the plenum region would be avoidance of cross flow (which could produce lateral forces on the guide tubes for the poison fingers and thorium oxide fingers and cause rod jamming) and vibration.

5.2.1.2 Fuel for the Conceptual Low-Gain Converter

During power cycling and irradiation, the fuel pellets in the low-gain converter phase would differ in behavior (with respect to thermal expansion, densification, growth, chipping, and cracking) from the fuel pellets in either the conceptual high-gain converter phase or the reference PWR. In addition, the peak fission density in the ternary oxide (fissions per cubic centimeter of ternary oxide) would be higher than that in the reference PWR. Calculations indicate that it may be possible to choose fuel-pellet parameters (density, edge chamfer, end dishing) and certain other parameters (pellet-to-cladding gap, internal rod pressure, and startup-rate limitations) so as to ensure acceptable axial and radial forces on the fuel-rod cladding. Extensive irradiation testing and design would be required to determine the acceptability of using this fuel concept.

5.2.2 THERMAL-HYDRAULIC CONSIDERATIONS

5.2.2.1 Common Considerations

This conceptual reactor would have a higher pressure drop than the reference PWR. Preliminary calculations, however, indicate that it may be possible to maintain adequate coolant flow using the pumps in the reference PWR plant.

Neutron capture in the thorium oxide finger rods would produce uranium-233. After sufficient irradiation, these rods would produce a substantial heat flux, and the cooling requirement would be more severe than that for the poison-finger rods used in the reference PWR for power shaping.

The primary consideration for the design of the plenum region would be avoidance of cross flow, which could produce vibration and lateral forces on the guide tubes for the poison fingers and thorium oxide fingers and cause rod jamming.

5.2.2.2 Fuel for the Conceptual Low-Gain Converter

The alternating pellets used in the fuel rods of the low-gain converter would lead to an alternating high and low heat flux, especially in fresh fuel rods, because most of the heat would be produced in the uranium oxide annulus of the duplex pellet. This effect would have to be included in calculations of the critical heat flux and fuel temperature for normal operation and in calculations of cladding temperature during a loss-of-coolant accident (LOCA).

The ratio of the peak heat flux to the average heat flux would depend on the lengths of the alternating pellets. To establish a basis for predicting performance during normal operation, preliminary tests have been performed with electrically heated single rods to simulate the performance of alternating pellets about 0.4 inch long. Preliminary results indicate that, over the range of expected heat flux and length of alternation, the critical heat flux is determined primarily by the rod average heat flux and is not affected measurably by the peak-to-average heat-flux ratio. Additional tests with full-length rods and rod bundles would have to be performed to confirm this preliminary conclusion.

During normal operation, the peak fuel temperatures in the low-gain converter phase would be lower than those in the reference PWR because most of the heat would be produced in the thin uranium oxide annulus of the duplex pellets. The volumetric heat capacity of thorium oxide is also lower than that of uranium oxide. As a result of these effects, the total heat stored in a fuel rod would be lower for the low-gain converter phase than that for the reference PWR. During a loss-of-coolant accident, the peak temperature of fuel-rod cladding would be affected by the total heat stored in the fuel rods.

5.2.3 MATERIALS

No new or unproved materials would be needed. However, the uranium oxide in the low-gain converter duplex pellets would be irradiated to a higher burnup than that in the reference PWR. Furthermore, the use of Zircaloy in the grids of a close-packed-hexagonal rod array would require confirmatory testing and analyses.

5.2.4 VALIDATION OF CALCULATIONS

Calculations have been performed to estimate reactivity levels, loading requirements, lifetime, and mass flows for the core. Sufficient calculations were performed to estimate the time-dependent effects of fuel recycle on these parameters. Nuclear cross sections for use in depletion calculations were generated by methods that explicitly represent the space and energy effects on neutron resonance capture. Resource requirements were estimated from the time-dependent mass-flow data. In addition, thermal and hydraulic calculations have been performed to estimate core power capability.

5.2.4.1 Calculations of Nuclear Performance

Converter nuclear performance was determined from point-depletion calculations utilizing four neutron-energy groups with breakpoints at 0.8 MeV, 5.53 keV, and 0.625 eV. The neutron cross sections used in these calculations were obtained from detailed Monte Carlo calculations for representative fuel assemblies. The point-depletion results were used to estimate reactivity levels, lifetime, and mass flows for the core.

Effective few-group microscopic cross sections were generated with the RCP01 Monte Carlo program. The Monte Carlo model used 31 energy intervals to describe neutron energies between 0 eV and 10 MeV, with each interval being further divided into as many as 1,000 subintervals to permit accurate representation through all resonances. The primary source of basic cross-section information was the ENDF/B data libraries.

Detailed hexagonal fuel assemblies were represented in the RCP01 calculations, including explicit geometric representations of the fuel pellet, cladding, moderator, and, where appropriate, guide tube for each fuel-bearing and non-fuel-bearing rod in the assembly. The calculated isotopic reaction rates were used to generate highly accurate few-group microscopic cross sections, appropriate for an entire assembly, for use in the point-depletion model; low-gain-phase nuclide inventories were chosen to be representative of midlife depletion conditions. To facilitate the rapid examination of a number of high-gain concepts, several different RCO01 assembly calculations were made to span the range of fuel temperature, moderator temperature, and fuel-to-coolant ratio anticipated for the high-gain-phase concept. In addition, heavy-metal isotopic mixes characteristic both of initial and of equilibrium-cycle loadings were represented for the high-gain phase.

Few-group microscopic data from the RCP01 Monte Carlo results were employed in the four-neutron-energy-group point-depletion model. This model solves equations describing depletion chains for all important heavy-metal isotopes and the dominant fission-product chains for xenon and samarium. Additional fission-product absorption was incorporated via a residual-fission-product nuclide. The point-depletion results were used to estimate core reactivity levels and lifetime and to provide the ratios of heavy-metal isotopes as a function of fuel depletion. The performance of the high-gain converter phase was evaluated from this depletion model with appropriate adjustments to the calculated conversion ratios to account for leakage and noncritical reactivity levels in the computations. The estimates of the high-gain phase performance and the isotopic ratios as a function of fuel depletion were then combined to obtain the desired estimates of core mass flows.

The methods of calculating nuclear performance are derived from those developed and validated as part of the Shippingport PWR and LWBR programs.

The requirements for U_3O_8, ThO_2, and separative work were estimated from the low-gain-phase and high-gain-phase mass flows normalized to 1,000-MWe reactors. The low-gain phase is assumed to operate continuously until sufficient uranium-233 has been generated to supply both in-core and out-of-core inventories for the high-gain phase. After the high-gain phase starts operating, one low-gain converter is assumed to support about five high-gain converters, providing the required uranium-233 for the high-gain phase.

The resource requirements for such a low- and high-gain converter system were estimated on a cycle-by-cycle basis from the previously calculated mass flows. Integral results for 30, 50, and 100 years of operation were obtained as sums of the cycle-by-cycle results normalized to an integrated system capacity of 1,000 MWe.

5.2.4.2 Thermal Calculations

Thermal performance has been analyzed with a simplified calculational model. This model has been qualified by performing detailed module calculations that allowed for the transfer of two-phase fluid properties in three dimensions to predict local fluid conditions and the critical heat flux. These detailed calculations were made with the computer program HOTROD, which was used for the thermal analysis of the Shippingport LWBR core.

The simplified model relates the steady-state overpower thermal performance of the proposed concept to that of a reference commercial design, such as the Babcock & Wilcox standard 205 design. The difference in total reactor flow between the proposed concept and the reference design is determined from changes in parameters that affect the flow, such as the core hydraulic diameter, total core flow area, fuel-rod length, and number and type of grids. Mass velocity and inlet temperature are calculated from the flow for a specified core-average temperature. The hot-channel critical-heat-flux performance is then determined by factoring in changes in the parameters that affect the critical heat flux, such as mass velocity, hydraulic diameter, inlet temperature, power peaking factors, and channel length.

Commercial design procedures, methods, hot-channel factors, and critical-heat-flux correlations provided the basis for the analysis. The peak linear power would be maintained at a level that results in acceptable fuel-element and LOCA performance.

5.2.4.3 Mechanical Calculations

Structural components for this concept have been sized on the basis of existing commercial designs. Detailed analyses have not been performed, but preliminary analyses of key structural support components have been performed using hand calculations to estimate stresses.

5.3 ENVIRONMENTAL CONSIDERATIONS

The nonradiological effects of the light-water low-gain converter supplying a high-gain converter would be similar to those for the LWR, and the comments made in Section 2.3 with respect to land use, water use, heat dissipation, and chemical and biocidal wastes apply to this concept as well.

With respect to radiological effects, e activity release paths and radioactive-waste-processing systems for this concept are similar in most respects to those of a typical LWR. Slight variations, however, may exist between the quantities of radioactive isotopes released from the LWR core and the quantities that would be released from the reactor described here because of differences in fuel composition, reactivity-control systems, and discharge burnup, with burnup having the most significant effect on the quantities of radioactivity released from the core.

The difference in burnup would result in a slight decrease in the amount of radioactivity released in comparison with a typical LWR. Differences in fuel composition would result in small amounts of release from isotopes such as protactinium-233, which would not exist in measurable quantities in a typical LWR. Differences in reactivity-control systems would also result in differences in radioactivity release rates. Since no boron is used in the core for reactivity control, certain releases related to the presence of boron would not occur. The primary effect is that this would eliminate the main source of tritium.

The overall result of these differences would be a reduction in the amount of radioactivity released to the environment in comparison with a typical LWR. The actual amounts of radioactive liquids and gases released to the environment and related dose rates would be very similar to those for the concept of the light-water breeder based on LWBR type I modules (see Tables 2-13 through 2-17).

Radiation exposure to plant workers is related to plant design and is not greatly affected by the installed core. Most of the exposure would be incurred in maintenance, repair, waste-processing, and refueling operations, which would be similar to those in a typical LWR. Occupational exposure from a plant based on this concept would therefore be about the same as that from a typical LWR.

5.4 SAFETY CONSIDERATIONS

This section discusses the major unique features of the concept, compared with the reference PWR, that would need to be considered for licensing.

5.4.1 COMMON CONSIDERATIONS

The design features discussed in Sections 5.4.1.1 through 5.4.1.7 were evaluated in the Shippingport LWBR safety analysis; they were reviewed and found to be acceptable. Item 5.4.1.8 is the feature of the concept that is uniquely different from the Shippingport LWBR.

5.4.1.1 Tight Lattice

The rod-to-rod spacing in the present concept is 100 mils rather than the 120 mils used in commercial PWRs. Such a spacing would impose restrictions on fuel-element and grid design and assembly. These restrictions would be more stringent than those in commercial practice to avoid contact between the fuel rods and the grid structure and between fuel rods. The rod-to-rod spacing in the LWBR core at Shippingport is 60 mils. In setting core operational limits (specifications of set points and allowable power increase) for the Shippingport LWBR, it was assumed that rod-to-structure contact does occur. Extensive in-reactor and out-of-reactor tests with rods in contact have been completed. Design assessments of seed and blanket fuel-rod bowing predict that rod-to-rod contact is highly unlikely. Bowing analysis of the blanket has been completed on a worst-case basis, and the rod-to-rod spacing has been determined as a function of reactor lifetime. These data demonstrate that adequate margin is incorporated into the Shippingport LWBR design and indicate that acceptable operation of fuel assemblies in a commercial-scale reactor of the present concept should be feasible.

5.4.1.2 Core Thermal Margins

Thermal-analysis programs and correlations for the Shippingport LWBR have been proved applicable for thermal-hydraulic analyses under widely ranging axial and radial heat-flux distributions in a close rod array, including a coupled-region interface. The critical-heat-flux correlation for the Shippingport LWBR conservatively predicts the data for the full range of LWBR geometries and heat-flux distributions. Thermal margins for the present concept have been calculated by an approach similar to that used for the Shippingport LWBR. Additional critical-heat-flux testing might be required to confirm the applicability of the modeling to the specific fuel lattice of the current concept.

5.4.1.3 Provision for Accident Prevention

The probabilities of accident initiation for the Shippingport LWBR core and for the present concept are comparable to that for the reference PWR. The safety and protection of the Shippingport LWBR plant have been designed within regulatory guidelines and requirements. An emergency core-cooling system appropriate for the LWBR has been designed into the Shippingport reactor. No impediment to providing comparable protection features in a large plant using the present concept has been identified.

5.4.1.4 Acceptability of Movable Fuel for Reactivity Control

To date, all reactivity-control functions required by an operating nuclear power plant have been satisfactorily performed in the Shippingport LWBR by means of the movable fuel. These include control functions required for shutdown, plant heatup, power operation, and lifetime reactivity changes. Scram reactivity control is also provided by the movable fuel. For the present concept, the movable fuel would supply only the reactivity control to match part of the power defect, plus the buildup of fission products, including equilibrium xenon. The requirements on the movable fuel in the present concept, therefore, would be much less severe than they are in the Shippingport LWBR core.

5.4.1.5 Power and Temperature Coefficients

There are two aspects to the question of the range of power and temperature coefficients that could be expected in the two phases of the present concept. The first aspect concerns the accuracy with which the calculational model can predict these coefficients. The second aspect concerns the values of the coefficients that have been calculated or estimated.

The calculation of temperature and power coefficients using the design model has resulted in good agreement for both the critical-experiment and the operational configurations of the Shippingport LWBR core. Measured zero-power temperature coefficients during both initial and the first periodic testing phases have been calculated satisfactorily for both hot and cold conditions. Performance at power was also calculated satisfactorily. The calculated power coefficient of reactivity was 6% less negative than the measured value, and the calculated temperature coefficient at power was 6% more negative than the measured value.

In regard to the second aspect, there is a firm basis for expecting that the temperature and power coefficients for the present high-gain converter phase would be similar to those for the Shippingport LWBR core and, therefore, would be acceptable with respect to licensing considerations. The temperature coefficient, which depends primarily on the fuel-to-coolant ratio, is expected to remain approximately constant during the life of the core, consistent with the facts that little or no boron would be used in the core during operation at power and that the fissile inventory would remain nearly constant during core operation. In the Shippingport LWBR core, the power coefficient is due almost entirely to the Doppler effect and depends primarily on the fuel-to-coolant ratio and the power density (kilowatts per kilogram of heavy metal) at full power. In the present conceptual high-gain converter phase, the fuel-to-coolant ratio would be between that of the LWBR core and that of the reference PWR core, and the power density would be higher than that of the Shippingport LWBR core. It is therefore expected that the temperature and power coefficients for the present conceptual converter phase would be at least as negative as those for the reference PWR.

Temperature and power coefficients have also been calculated for prebreeder concepts fueled with thorium and slightly enriched uranium-235 and have been found acceptable. Therefore, it is expected that the temperature and power coefficients for the conceptual low-gain converter phase would also be acceptable.

5.4.1.6 Control Stability and Adequacy

The LWBR core at Shippingport has successfully operated as part of an integrated commercial power network. This operation has included long periods at constant power, several weeks of planned swing-load operations, and controlled startup and shutdown periods from zero to full power. In addition, special tests were run to demonstrate the dynamic characteristics and response of the plant to typical load-change rates common in commercial plants. Performance was satisfactory during both swing-load and steady-state operation and in the special tests. The results indicate that control stability and adequacy can be maintained in a core with movable-fuel reactivity control.

5.4.1.7 Nuclear Stability

Analytical studies have examined the stability of large uranium-233/thorium reactors with high power densities against spatial xenon oscillations and have compared it with the stability of uranium-fueled PWRs of comparable sizes and ratings (Ref. 2). These studies show that uranium-233/thorium systems are inherently significantly more stable, primarily because of lower total xenon yields, larger fractions of xenon-chain yield directly to xenon, and larger-in-magnitude (more negative) Doppler coefficients of reactivity. Initial physics testing at the Shippingport LWBR have demonstrated that the LWBR module design results in a tightly coupled core and provides further confidence that the present high-gain converter concept would have acceptable stability properties.

5.4.1.8 Reactivity Control

The essential difference between this concept and the presently operating Shippingport LWBR is that, instead of movable bundles of fuel rods, movable fingers of individual thorium oxide control rods, arranged much like the fingers of poison rods in commercial PWRs, would be interspersed throughout the module. A program to establish the expected performance of such a design and to provide supporting physics and engineering test data is being evaluated as part of the Department of Energy (DOE) Advanced Water Breeder Applications Program (AWBAP).

5.4.2 FUEL FOR THE LOW-GAIN CONVERTER PHASE

Placing the fissile-fuel-bearing ternary oxide annulus around the thorium oxide core, which initially contains no fissile fuel, would cause most of the fission energy to be generated close to the cladding and therefore would result in lower average fuel-element temperatures than those obtained with a solid uranium oxide pellet. This arrangement would also have the advantage of reducing stored heat, which would be important during a loss-of-coolant accident. In this concept, the annulus would experience a peak burnup as high as 119,000 MWd/MT compared with 50,000 MWd/MT in the reference PWR. Currently there is little information on the structural and thermal characterization of fuel at high burnup, fission-gas release, fuel swelling and cladding-strain rate, fuel-cladding response to power transients, and the effect of the high-temperature inner annular surface on defected-rod performance. These matters would have to be investigated before the capabilities of such a system could be established. This fuel system is being evaluated for possible further development under the DOE AWBAP, including analytical work to extend the scope of present fuel-element modeling and irradiation testing.

5.5 LICENSING STATUS

The status is essentially the same as described previously in Section 2.5 for the Type I module concept.

The principal licensing issues for the reactor concepts reported here are those generic to the recycle of fuel from fission reactors and the management of their wastes. The uranium-232 produced by the irradiation of thorium fuel has some advantages and some disadvantages in comparison with uranium fuel recycle.

As mentioned in Section 2.4.1.6, the Shippingport LWBR has successfully operated as part of an integrated commercial power network. This has included long periods at constant power, several weeks of planned swing-load operations, and controlled startup and shutdown periods from zero to full power. In addition, special tests were run to demonstrate the dynamic characteristics and response of the plant to typical load-change rates common in commercial plants. Performance was satisfactory during both swing-load and steady-state operation and in the special tests. The results indicate that plant operability would not be adversely affected by a core with movable-fuel reactivity control.

It was noted that the high-gain converter concept is expected to be more stable against spatial xenon instabilities than are current PWRs, partly because of the larger-in-magnitude (more negative) Doppler coefficient of reactivity produced by the thorium. No calculations on the stability of the low-gain converter concept have been made, but the stability is expected to be comparable to that of a conventional PWR.

The maintenance of a plant containing either phase of the present concepts would be essentially the same as that for a plant containing a conventional PWR core.

5.6 STATUS OF TECHNOLOGY AND OF RESEARCH, DEVELOPMENT, AND DEMONSTRATION

5.6.1 GENERAL CONSIDERATIONS

The areas in which technology would need to be extended include the fuel-assembly grids, thorium oxide control fingers, and the plenum. Zircaloy grids are already in use in commercial cores; an extension to larger size and number of rods and to the tighter tolerances needed for the close-packed-hexagonal array of the high-gain converter reactor would be required. Essentially, the thorium oxide control fingers would be fuel rods suspended from one end; operation in a guide tube with proper coolant flow and wear and vibration characteristics would have to be demonstrated to be acceptable. The plenum design would require flow testing to demonstrate operation without vibration or unacceptable lateral forces from cross flow.

No critical experiments have been performed for systems that use both uranium-238 and thorium in cylindrical fuel elements, either in duplex or homogenized configurations, as with denatured fuel. Experimental confirmation of the nuclear design analysis methods for such configurations would be required for this or any concept using this combination of fertile fuel.

If the "dual-action" control-rod-drive mechanism were chosen, this would also require an extension of technology. The concepts of roller nuts and magnetic latching for control-rod-drive mechanisms are not new; the novel feature would be the application of these to concentric lead screws and to high loads.

In addition, the use of this concept would require the recycle of fuel and the research, development, and demonstration necessary to implement reprocessing, refabrication, and waste management.

The research, development, and demonstration requirements for this concept are summarized in Table 5-10.

5.6.2 FUEL SYSTEM FOR THE CONCEPTUAL LOW-GAIN CONVERTER PHASE

Existing technology would need to be extended for the fuel system of alternating duplex pellets and solid thorium oxide pellets. This fuel system is being evaluated for possible further development as part of the DOE AWBAP. Duplex pellets have been fabricated, and irradiation testing of fuel rods is in progress. Additional critical-heat-flux testing would probably be required. Computer models to predict the performance of this fuel system are also being developed.

Table 5-10. Technological advance requirements

Plant component	Operating experience evaluated	Prototypic or production components being manufactured	No new knowledge required	Contemporary technology with modified configuration/ application	Modest improvement in performance or size from available systems	Modest improvement in performance or size and modified configuration applications
Nuclear fuel	H[a]	H	H			L[b]
Reactivity-control systems					√	
Reactor vessel	√	√	√			
Core-support structure				√		
Reactor-vessel internals, including shielding, ducting, control-rod guides, baffles	√	√			√	
Primary coolant pumps and auxiliary systems	√	√	√			
Primary coolant chemistry/ radiochemistry control	√	√	√			
Primary-system heat exchangers	√	√	√			
Reactor instrumentation	√	√	√			
Emergency core-cooling/ safe-shutdown systems	√	√	√			
Containment, containment-cleanup systems, and effluent-control systems	√	√	√			
Other accident-mitigating systems (i.e., plant-protection systems)	√	√	√			
On-site fuel-handling and storage and shipping equipment	√	√	√			
Main turbine	√	√	√			
Other critical components, if any	√	√	√			
Balance-of-plant components	√	√	√			

[a]Operating data being obtained from LWBR.

[b]Irradiation test data presently being obtained for fuel for low-gain-converter (prebreeder) phase fuel.

Abbreviations: H, high-gain-converter phase; L, low-gain-converter (prebreeder) phase.

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1. Babcock & Wilcox, Reference Safety Analysis Report, BSAR-205, NRC Docket STN 50-561.

2. T. R. England, G. L. Hartfield, and R. K. Deremer, Xenon Spatial Stability in Large Seed-Blanket Reactors, WAPD-TM-606, Bettis Atomic Power Laboratories, April 1967.

APPENDIX A

U.S. Nuclear Regulatory Commission Review of Safeguards Systems for the Nonproliferation Alternative Systems Assessment Program Alternative Fuel-Cycle Materials

BACKGROUND

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for the physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage, and theft. The U.S. Nuclear Regulatory Commission (NRC) has considered whether strengthened physical protection may be required as a matter of prudence (Ref. 1). Proposed upgraded regulatory requirements to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgraded rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM SAFEGUARDS BASIS

The desired basis for the NRC review of safeguards systems for the Nonproliferation Alternative Systems Assessment Program (NASAP) alternative fuel-cycle materials containing significant quantities of strategic special nuclear material (SSNM),[a] greater than 5 formula kilograms,[b] during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I[c] material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, proposals have been made to increase radioactivity of strategic special nuclear materials (SSNMs) that are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh-fuel material to require that, during the period after export from the United States and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover low-radioactivity SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time required in obtaining material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements imposed by the Nuclear Non-Proliferation Act of 1978 include application of International Atomic Energy Authority (IAEA) material accountability

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[a] $\geq$20% U-235 in uranium, $\geq$12% U-233 in uranium, or plutonium.

[b] Formula grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium); Ref. 10 CFR 73.30.

[c] IAEA definitions of highly enriched uranium (>20%).

requirements to nuclear-related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by the IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative that could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC REVIEW

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches (such as upgraded physical protection, improved material control and accountancy, dilution of SSNM, decreased transportation requirements, few sites handling SSNM, and increased material-handling requirements as applied to each fuel material type) should be assessed. The evaluation should consider, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb-making purposes; the relative impacts on domestic and on international safeguards; the impact of radioactive contaminants on detection for material control and accountability, measurement, and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tampering or breaching; the increased public exposure to health and safety risk from acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, the NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation, we request that the NRC assess the differences in the licensing requirements for the domestic facilities, transportation systems to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel-cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, and on the international and national safeguards systems of typical importers for protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, as well as the potential advantages in detection or deterrence should be described in detail. The potential role, if any, that added radioactivity could or should play should be clearly identified, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its acquisition by foreign countri   or weapons purposes. Licensability issues that must be addressed by research, development, and demonstration programs also should be identified.

Table A-1 presents a listing of unadulterated fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import, and export:

Table A-1. Minimum radiation levels for various fuel material types

Fuel Material Type	Minimum radiation level during 2-year period, rem/hr at 1 meter (Ref. 6)	
	Mixed[a]	Mechanically attached[b]
PuO_2, $HEUO_2$ powder or pellets[c]	1,000/kgHM	10,000/kgHM
PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets[c]	100/kgHM	10,000/kgHM
LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[a]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[b]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[c]HEU is defined as containing 20% or more U-235 in uranium, 12% or more of U-233 in uranium, or mixtures of U-235 and U-233 in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are indicated in the following table and references.

Table A-2. Candidate methods and radiation levels for spiking fuel materials

Fuel material type	Minimum 2 year radiation level, rem/hr at 1 meter	Process	Minimum initial radiation level, rem/hr at 1 meter	Reference
a. PuO_2, HE UO_2 powder or pellets	1000/kgHM	Cobalt-60 addition	1300/kgHM	2, 3, 5, 6
b. PuO_2-UO_2 and HE UO_2/ThO_2 powder or pellets	100/kgHM	Cobalt-60 addition	130/kgHM	2, 3, 5, 6
		Fission product addition (Ruthenium-106)	400/kgHM	
c. LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	Co addition	13/assembly	2, 3, 5, 6
		Fission product addition (Ruthenium-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1000 (30 days)/ assembly	4
d. LMFBR or GCFR fuel assembly	10/assembly	Cobalt-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ruthenium-106)	40/assembly	2, 3, 5, 6
		pre-irradiation (40 MWd/MT)	1000 (30 days)/ assembly	4

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3. J. E. Selle, P. Angelini, R. H. Rainey, J. I. Federer, and A. R. Olsen, Practical Considerations of Nuclear Fuel Spiking for Proliferation Deterrence, ORNL TM-6483, Oak Ridge National Laboratory, October 1978.

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APPENDIX B

Responses to Comments by the U.S. Nuclear Regulatory Commission
PSEID, Volume III, Light-Water Breeder Reactors

Preface

This appendix contains comments and responses resulting from the U.S. Nuclear Regulatory Commission (NRC) review of the preliminary safety and environmental submittal of August 1978. It should be noted that the NRC comments are the result of reviews by individual staff members and do not necessarily reflect the position of the Commission as a whole.

RESPONSES TO GENERAL COMMENTS

1. Regarding the NRC request to reduce the number of reactor concepts and fuel-cycle variations, the Nonproliferation Alternative Systems Assessment Program (NASAP) set out to look at a wide variety of reactor concepts and fuel cycles with potential nonproliferation advantages. These various concepts have differing performance characteristics in other important respects, such as economics, resource efficiency, commercial potential, and safety and environmental features. The relative importance of these other characteristics and trade-offs has been determined and the findings are incorporated in the NASAP final report.

2. Regarding the comment on the need to address safeguards concepts and issues, some concepts for providing protection by increasing the level of radioactivity for weapons-usable materials have been described in Appendix A to each preliminary safety and environmental information document (PSEID). Appendix A has been revised to reflect NRC comments.

   An overall assessment of nonproliferation issues and alternatives for increasing proliferation resistance is provided in Volume II of the NASAP final report and reference classified contractor reports.

RESPONSES TO SPECIFIC QUESTIONS

Question 1

Additional questions have been raised regarding the stability of the light-water fast-breeder reactor (LWBR) core with duplex pellets due to the delayed heating of the pellet core (which contributes the major part of the Doppler feedback).

Response

This question was addressed in a memo from Parker, U.S. Department of Energy, Division of Naval Reactors (DOE-NR), to L. Lois, U.S. Nuclear Regulatory Commission (NRC), on May 3, 1979. While a detailed evaluation of possible oscillatory behavior has not been performed, there is no basis for anticipating that such behavior could occur since the reactivity shape and time responses for the solid and duplex pellets are very similar. Such a calculation could be performed to confirm acceptable performance if development of the duplex pellet fuel is continued.

Question 2

The question of recriticality has been answered partly for the homogeneous binary pin where, however, questions relating to configurations reflected by water or hydrogenous concrete remain.

Response

The homogeneous binary fuel rods being considered for the NASAP concepts are the same type as used in the Shippingport LWBR. Based on calculations performed for the LWBR for various reflective conditions which showed acceptable performance, the DOE-NR concludes that recriticality would not occur for homogeneous binary fuel rods as considered in the Nonproliferation Alternative Systems Assessment Program (NASAP) water-cooled breeder concepts.

Question 3

The lower power density in the thorium prebreeder and breeder cores gives rise to lower ΔTs across the core and then higher flows (for similar power levels); this necessitates the use of larger pumps, larger safety equipment, and even a larger pressure vessel. For such a design, it is not clear that the core or the plant will behave in a manner similar to a previously analyzed PWR; neither is it clear that the type and/or design as well as response of the engineered safety features will be the same.

Response

The NRC concerns are not uniformly applicable to all the core concepts. It is more logical to consider two general groups of reactors in the LWBR NASAP concepts. These are (a) the backfit prebreeders and (b) the seed-blanket prebreeders and breeders and are addressed separately below.

a. Backfit prebreeders. Three of the four LWBR reactor pairs employ backfit prebreeders that would be direct mechanical backfits to current PWR cores. These cores would have ΔTs nearly identical with the reference core and except for small differences in kinetics parameters and reactivity coefficients, core and plant performance would be similar to that of current PWRs.

b. Seed-blanket breeders (and LWBR prebreeder). These reactors in general would require increased pump power because they have larger hydraulic resistance and this may result in a larger reactor vessel. It is recognized that there will be some difference in the behavior of the core and the plant compared to conventional PWRs. However, based on the following considerations, the DOE-NR considers there is a basis for concluding that a satisfactory plant and associated engineered safety features could be designed and built for a commercial-scale seed-blanket breeder reactor:

(1) The Shippingport LWBR has successfully operated as part of an integrated commercial power network and no safety-related licensing issues have been identified that are significantly different from conventional PWRs.

(2) No major differences in plant or reactor behavior have been identified in the operation of 1,000- to 1,100-MWe light-water-cooled nuclear plants compared to the earlier commercial plants that had power ratings in the range of 200 to 300 MWe.

(3) An alternative approach would be a seed-blanket core with a power level somewhat lower than a conventional PWR would have for that particular size of reactor vessel and that would maintain components such as pumps at the present size.

Question 4

In addition, we believe that if the concept of the duplex fuel is important to the viability of the breeder concept, additional information must be provided in Volume III or Volume VII concerning the development status of the disassembly operation shown in Figure 2-6.

Response

Because no "disassembly" operations would be required, additional research, development, and demonstration would be required to establish the dissolution conditions. It is noted, however, that none of the breeder concepts and not all of the prebreeder concepts utilize duplex pellets, and that duplex pellets are not essential to the success of the development of the LWBR concepts.

Question 5

Taken as a whole, the LWBR fuel design concept is extremely complex, as compared with the LWR. Differing enrichments, duplex fuel pellets, stationary (blanket) versus movable (seed) components, thorium fingers, tertiary oxides, differing grid materials, taken with the various permutations and combinations afforded by the various breeder-prebreeder options pose potential problems with respect to the development and verification of adequate design bases, design limits, and acceptance criteria.

Response

No single concept would employ all of the features identified by the NRC. The NRC listing portrays the spectrum of possible designs. For instance: duplex fuel pellets and movable fuel occur together only in the prebreeder based on LWBR Type I modules of the three concepts reviewed by the NRC; this is also the only one with ternary oxide fuel; movable seeds and thoria fingers do not occur together in any core; each concept employs only one type of grid material. The final NRC evaluation should reflect this fact.

Question 6:

There is probably an over-reliance on extrapolation of Shippingport technology in regard to LWBR fuel-design licensability.

Response:

There are basically two different fuel systems being considered for the LWBR systems: (1) the binary UO_2-ThO_2 fuel system similar to that being used in the Shippingport LWBR, and (2) the duplex-pelleted fuel system being developed for possible use in the prebreeder reactors. These are addressed separately below.

a. Binary UO_2-ThO_2 fuel. The DOE/NR considers there is no question of licensability for this fuel system; any questions relate solely to sizing of the reactor core for a given power rating and sizing of safety systems. Continued satisfactory operation of the Shippingport LWBR is considered to prove the feasibility of the UO_2-ThO_2 fuel system and to provide a reliable basis for extrapolation. There would be a high degree of similarity between the reactor core environment in the Shippingport LWBR (which was designed to simulate a commercial-scale breeder reactor by using reflector-blankets) and that of a commercial-scale LWBR. Therefore, there is not felt to be an "over-reliance" on the Shippingport fuel technology in extrapolating to a commercial-scale LWBR using the binary UO_2-ThO_2 fuel system.

b. Duplex-pelleted fuel. The DOE/NR is not counting on the Shippingport operation to provide a basis for showing that the duplex-pelleted fuel system is acceptable. As specifically identified in each chapter of the applicable LWBR preliminary safety and environmental information document (PSEID) (Volume III), the behavior of the duplex-pelleted fuel is an uncertainty and a fuel-test program is underway as part of the DOE Advanced Water Breeder Applications program to determine how much future development should be done. For this fuel system, therefore, the DOE/NR understands the uncertainty involved and is not counting on the Shippingport operation to prove out or confirm the performance.

Question 7:

Based on the preliminary review, some additional outstanding issues for the LWBR are as follows.

- Zircaloy core support grid (now designed of stainless steel)
- Potential consequences of molten ThO_2 (e.g., autocatalytic behavior)
- Control element performance with respect to its hydraulic support system
- Potential effect of oversized coolant-handling systems in a backfit prebreeder (e.g., oversized pumps, oversized safety injection)

- Potential effect of the unique radioactive materials contained in the reactor on siting criteria (e.g., uranium-233)
- Analysis of the potential effect of the design-basis accident and the low-probability accident
- Adequacy of the proposed thorium-finger control system
- Fuel reprocessing and remote fabrication
- Validity of assuming the extrapolability of the Shippingport technology and safety implications
- Required level of effort to address the above issues

Response:

The DOE/NR disagrees with some of the issues raised as explained below:

a. On portion of excerpt regarding "consequences of molten ThO_2"

The DOE/NR is not certain of the precise nature of the NRC concern. Molten ThO_2 does not appear capable of autocatalytic behavior since the thorium in ThO_2 is in its highest oxidation state. Since the NR knows of no mechanism by which the temperature of molten ThO_2 could be further raised by an exothermic oxidation reaction, there appears to be no basis for concern about "autocatalytic" behavior of molten ThO_2.

b. On portion of excerpt regarding "hydraulic support system"

The meaning of the NRC comment concerning a "hydraulic support system" is not clear. The Shippingport LWBR core has a bypass inlet flow (BIF) system but it is not a "hydraulic support" system. The BIF system exerts a downward force, not a support force; this opposes the upward flow force and allows the seed module to fall under the force of gravity in the event of a scram. The "support system" for the control element is therefore mechanical, not hydraulic.

c. On portion of excerpt regarding "oversized coolant-handling systems"

This is not correct. The backfit prebreeder concepts would not require "oversized" coolant-handling systems since they would have power density and core T comparable to a commercial PWR.

d. On portion of excerpt regarding "siting criteria"

The DOE/NR acknowledges that a complete safety analysis would need to be performed for specific core and plant designs; however, it is not anticipated that the results would be significantly different from those reported in ERDA-1541, Final Environmental Statement, Light Water Breeder Reactor Program, Volume 3, page IX-207ff. It can be concluded from ERDA-1541 that no significant difference in radiological impact would be expected from the postulated accidents whether they occurred in conventional PWRs, LWBR prebreeders, or LWBR breeders.

e. On portion of excerpt regarding "extrapolability of the Shippingport technology"

The DOE/NR has addressed this issue in its response to Question 7.

f. <u>The DOE/NR comment on portion of excerpt regarding "level of effort"</u>

This does not appear to be a licensing question and hence is not appropriately identified as an outcome of the NRC review.

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