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Nuclear Proliferation and Civilian Nuclear Power

Report of the Nonproliferation Alternative Systems Assessment Program

Volume IX: Reactor and Fuel Cycle Descriptions

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U.S. Department of Energy
Assistant Secretary for Nuclear Energy

December 1979

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Nuclear Proliferation and Civilian Nuclear Power

Report of the Nonproliferation Alternative Systems Assessment Program

Executive Summary

- Volume I: Program Summary**
- Volume II: Proliferation Resistance**
- Volume III: Resources and Fuel Cycle Facilities**
- Volume IV: Commercial Potential**
- Volume V: Economics and Systems Analysis**
- Volume VI: Safety and Environmental Considerations for Licensing**
- Volume VII: International Perspectives**
- Volume VIII: Advanced Concepts**
- Volume IX: Reactor and Fuel Cycle Descriptions**

Nuclear Proliferation and Civilian Nuclear Power

Report of the Nonproliferation Alternative Systems Assessment Program

DOE/NE-0001 (VOL. 9)

Volume IX: Reactor and Fuel Cycle Descriptions

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U.S. Department of Energy
Assistant Secretary for Nuclear Energy
Washington, D.C. 20545

December 1979

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INTRODUCTION

The Nonproliferation Alternative Systems Assessment Program (NASAP) has characterized and assessed various reactor/fuel-cycle systems. Volume IX provides, in summary form, the technical descriptions of the reactor/fuel-cycle systems studied. This includes the status of the system technology, as well as a discussion of the safety, environmental, and licensing needs from a technical perspective. This information was then used in developing the research, development, and demonstration (RD&D) program, including its cost and time frame, to advance the existing technology to the level needed for commercial use. Wherever possible, the cost data are given as ranges to reflect the uncertainties in the estimates.

Volume IX is divided into three sections: Chapter 1, Reactor Systems; Chapter 2, Fuel-Cycle Systems; and the Appendixes.

Chapter 1 contains the characterizations of the 12 reactor types listed below:

1. Light-water reactor (LWR). The current pressurized water reactor (PWR) and boiling water reactor (BWR) are characterized. Extensive information is provided on potential improvements in uranium utilization.
2. Heavy-water reactor (HWR). Data are supplied primarily on a CANDU-type pressurized HWR modified for U.S. conditions.
3. Water-cooled breeder reactor (WCBR). Data are supplied for five light-water breeder reactor (LWBR) concepts, heavy-water breeder concepts, and a variety of supporting prebreeder cycles.
4. High-temperature gas-cooled reactor (HTGR). Data are supplied for both steam-cycle and direct-cycle HTGR concepts.
5. Gas-cooled fast reactor (GCFR). Data are supplied for oxide-fueled GCFRs.
6. Liquid-metal fast breeder reactor (LMFBR). Extensive data are supplied on alternate fuels (oxide, carbide, metal) and alternate fuel cycles for LMFBRs. R&D facilities are described.
7. Spectral-shift-controlled reactor (SSCR). Data are supplied on the SSCR, which is based on modification of the current PWR.

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8. Accelerator-driven reactor (ADR). Data are supplied on a water-cooled fuel regenerator and a sodium-cooled fuel producer.
9. Molten-salt reactor (MSR). Data are supplied on a denatured MSR.
10. Gaseous-core reactor (GKR). Data are supplied on a mixed-flow gas-core reactor with a fertile molten-salt blanket.
11. Tokamak fusion-fission hybrid reactor (THR). Data are supplied on a hybrid fissile-fuel producer.
12. Fast mixed-spectrum reactor (FMSR). Data are supplied on a fast uranium-metal-fueled reactor on a once-through fuel cycle.

As NASAP progressed, evaluation of the technical data and feedback from the assessment studies indicated areas where further design analysis was needed. The results of these analyses are presented together with the reactor characterizations. For example, extensive data on LWR improvements and alternate breeder cycles are included.

Chapter 2 contains similar information developed for fuel-cycle facilities in the following categories:

1. Mining and milling. Data are supplied on uranium resources and on the technology of uranium production from high- and low-grade ores. Thorium supply is also covered.
2. Conversion and enrichment. Data are supplied on the conversion of uranium oxide to UF_6 feed for enrichment, and the major enrichment technologies are characterized.
3. Fuel fabrication. Facilities used for production of fuel elements for the various NASAP cycles are characterized. The impact of radioactive fuel on this process is covered.
4. Spent-fuel reprocessing. Data are supplied on the reprocessing of spent fuel from the various NASAP fuel-cycle concepts.
5. Waste handling and disposal. Data are supplied on the actual and potential methods of disposition of wastes from the various proposed NASAP fuel cycles.
6. Transportation of nuclear materials. Data are supplied on the facilities used to transport nuclear fuels and wastes.

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The Appendixes contain selected detailed data compilations used extensively in both the technical analyses and system-assessment studies:

Introduction. A table identifies the mass flow data used in the NASAP resources/economics analyses.

Appendix A. The reactor/fuel-cycle systems studied by NASAP are listed. Mass flow diagrams are provided for the majority of these cycles.

Appendix B. Detailed mass flow data for the NASAP reactor/fuel cycles are listed.

Appendix C. Key parameters used in NASAP assessments are summarized, including new-technology introduction dates, the projected U.S. demand for nuclear power, and the estimated unit costs of reactor/fuel-cycle operations.

Acronyms.

Glossary.

Bibliography.

The reactor and fuel-cycle characterizations of Volume IX form the basis for assessments presented in the other volumes of the NASAP report. Volumes II, III, IV, V, and VI of the NASAP report focus on the proliferation resistance, the resource requirements, the commercial potential, the economics, and the licensability of the major reactor/fuel-cycle systems. These include the LWR, the HWR, the WCBR, the HTGR, the GCFR, and the LMFBR options. Volume VII covers international aspects of nuclear proliferation and civilian nuclear power. Volume VIII presents an integrated assessment of the advanced concepts -- the ADR, the MSR, the GKR, the THR, and the FMSR. Volume I is the Program Summary.

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1. REACTOR SYSTEMS

1.1 LIGHT-WATER REACTORS

There are two types of light-water reactors (LWRs) in operation -- the pressurized-water reactor (PWR) and the boiling-water reactor (BWR). Each of these currently operates in the once-through (OT) mode utilizing low-enriched uranium (LEU) of ~3 wt % U-235 enrichment. Domestically, two-thirds of the LWRs that are planned, in construction, or in operation are PWRs and one-third are BWRs.

LWRs are the most widely used power reactors in the United States and the world. Operation by electric utilities is well established, the plants are commercially available, their overall performance has been satisfactory, and the LWR nuclear steam supply system (NSSS) and fuel technology are in a continual state of improvement.

1.1.1 Reactor/Fuel-Cycle System Descriptions

The NASAP evaluations described in this chapter are based on PWR and BWR NSSSs and balance-of-plant designs that are typical of reactors currently in operation and under construction.

General PWR performance specifications are summarized in Table 1.

The reference plant design for the NASAP study has a core thermal power of 3800 MWt and a net electrical power output of 1270 MWe. The reactor core is fueled with 241 fuel assemblies, each of which contains 236 fuel rods having an active length of 3.8 m. The current fuel consists of low-enriched UO₂ pellets encapsulated in a Zircaloy-4 cladding. Control rods travel into the fuel assembly. Figure 1 shows a cross section of the core and fuel assemblies; Figure 2 indicates the vertical arrangement of the reactor. The current designs developed by the PWR vendors [Babcock & Wilcox (B&W), Westinghouse, and Combustion Engineering (CE)] are described in detail in each vendor's standard plant safety analysis report.

General BWR performance specifications are summarized in Table 2. The reference BWR plant design (BWR/6) has a core thermal power of 3800 MWt and a

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Table 1. General Reactor Performance
Specifications — PWR

Power plant performance	
Core thermal power, MW	3800
Electrical power, MW	
Gross	1344
Net	1270
Thermal efficiency, %	33.4
Reactor parameters	
Core volume, liters	40,050
Equivalent core diameter, m	3.66
Core height, m	3.81
Core power density, MW/liter	0.095
Coolant flow rate, Mg/sec	20.66
Coolant inlet temperature, °C	296
Coolant outlet temperature, °C	327
Primary system pressure, MPa (psia)	15.5 (2250)
Number of fuel assemblies	241
Fuel parameters	
Average fuel temperature, °C	688
Maximum fuel temperature, °C	1882
Cladding temperature, °C	342
Total heavy metal, kg	99,313 ^a
Fissile material, kg	2,185
Discharge exposure, MWd/t	
Average	30,400
Peak	55,000
Conversion ratio, equilibrium cycle	0.59

^aInitial core with shims. Equilibrium core without shims has a heavy-metal content of 102,775 kg.

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Figure 1. Reactor Core Cross Section of Commercial PWR.

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Figure 2. Reactor Vertical Arrangement of Commercial PWR.

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Table 2. General Reactor Performance
Specifications — BWR

Power plant performance	
Core thermal power, MW	3800
Electrical power, MW	
Gross	1344
Net	1270
Thermal efficiency, %	33.4
Reactor parameters	
Core volume, liters	75,045
Equivalent core diameter, m	5.01
Core height, m	3.81
Core power density, MW/liter	0.051
Coolant flow rate, Mg/sec	14.8
Coolant inlet temperature, °C	278.3
Core pressure, MPa (psia)	7.27 (1055)
Core average void fraction	0.431
Number of fuel assemblies	848
Fuel parameters	
Average fuel temperature, °C	638
Maximum fuel temperature, °C	1840
Cladding temperature, °C	330
Total heavy metal, kg	154,600
Fissile material, kg	2,859
Discharge exposure, MWd/t	
Average	28,400
Peak	40,000
Conversion ratio, equilibrium cycle	0.59

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net electrical power output of 1270 MWe. The reactor core is fueled with 848 fuel assemblies, each of which contains 62 fuel rods having an active length of 3.8 m. The current fuel consists of slightly enriched UO_2 pellets encapsulated in a Zircaloy-2 cladding. The control blades travel outside the fuel assemblies, as shown by the core configuration in Figure 3. A vertical view of the reference BWR reactor system is given in Figure 4.

The current LWR design can be operated in either the once-through or the re-cycle mode, and both types of fuel cycles have been considered. These are described in greater detail in the following sections.

Once-Through Cycles

The current PWR using 3.0 to 3.2% enriched LEU fuel achieves a burnup averaging 30,000 to 34,000 MWd/t. The current BWR using 2.9% enriched LEU fuel achieves a burnup of about 28,000 MWd/t. Spent fuel is presently stored at the reactor site and eventually can be stored at an away-from-reactor storage facility. If reprocessing is not adopted, the spent fuel would ultimately be sent to a geologic spent-fuel repository.

Three classes of improved LWR once-through cycles characterized by lower U_3O_8 requirements are being considered. The first class consists of retrofittable fuel design and fuel management improvements that are expected to reduce U_3O_8 requirements by about 15%, while the second class consists of more extreme modifications generally thought to be retrofittable in the longer term which, when added to the first class, offer a considerably larger potential, although the technical and economic acceptability of certain of these remains to be assessed. Calculations indicate the improvement potential offered by a composite of most options in this category when combined with those in the first class to be in the 25% range (21% for PWRs and 28% for BWRs). While these composite analyses included most options in this second class, a few potential options which have been suggested were not included due to difficulties in representing these options in multicycle analyses with current analytical tools. Finally, it is possible that there are additional potentially retrofittable options which might be deployed by the year 2000 but which have not yet been characterized. Thus the total potential attainable for once-through LWRs from all options in this second class is (when combined with the first class) probably in the 25 to 30% range.

The third class consists of options considered nonretrofittable in currently operating plants or plants under construction because of technical or economic considerations. These options, when combined with retrofittable

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Figure 3. BWR Core Configuration.

Figure 4. BWR/6 Reactor Assembly.

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options into an advanced LWR design, could offer savings in excess of 30%. The practicality and benefits of many of these remain to be assessed.

NASAP has chosen "target" improvements in U_3O_8 utilization of 15% in the near term (by 1990) and 30% in the longer term (2000 and beyond). A 15% near-term target improvement was chosen in recognition of the possibility that not all of the retrofittable options would be fully or universally adopted by the end of the decade but that some improvements would undoubtedly be achieved during the 1980s. The additional uranium improvements expected in the long term are then expected to be accomplished partially by the more complete realization of the retrofittable options and by developing some part of the remaining longer-term nonretrofittable options.

A large number of design changes are being considered, as discussed in Section 1.1.2. The LWR that achieves approximately 15% reduction in U_3O_8 requirements will typically require the use of extended burnup fuels in the range of 45,000 to 50,000 MWd/t, annual refueling, an increased number of fuel batches, and low-leakage fuel management. The parameters of improved fuel cycles in PWRs and BWRs are given in Tables 3 and 4, respectively. The improved PWR case shown in Table 3 incorporates extended exposure and low-leakage fuel management. For the improved BWR case shown in Table 4, other improvement options in addition to extended exposure that are currently being implemented in BWRs are incorporated. These options are axial enrichment zoning (blankets), a three-enrichment-zone initial core, and improved refueling patterns. These are discussed in more detail in Section 1.1.2.

Results of an investigation of use of <20% U-235-Th fuel in the once-through mode indicate that, for burnups of about 33,000 MWd/t, operation in the once-through mode with full MEU-Th fueling requires 40% more U_3O_8 and 214% more separative work than the once-through LEU cycle, making this fuel uneconomical in current LWR once-through cycle operations. This penalty results from the fact that Th-232 has a thermal cross section about 2.7 times that of U-238, resulting in a fissile inventory penalty for the once-through cycle. On the other hand, potential reactor performance improvement and resource benefits can be provided with limited use of ThO_2 rods in UO_2 lattices in BWRs. This mixed-lattice once-through fueling option is discussed in more detail in Section 1.1.2.

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Table 3. Fuel Management Information —
PWR, Once-Through Cycle

	Composite improved PWR	Extended-exposure, low-leakage PWR	Reference PWR
Average capacity factor, %	75	75	75
Fraction of core replaced per refueling	0.2 ^a	0.2	0.33
Refueling interval, years	1	1	1
Fissile fabrication loss fraction	0.015	0.015	0.015
U ₃ O ₈ requirements, st/GWe ^b			
Initial core	376	459 ^c	394
Annual equilibrium	150	167	194
30-year cumulative	4789	5260	6040
Separative work requirements, 10 ³ separative work units (SWU)/GWe ^b			
Initial core	196	252	200
Annual equilibrium reload	106	116	117
30-year cumulative	3267 ^d	3588	3600
Equilibrium cycle enrichment, %	4.70 ^d	4.44	3.00
Plutonium in spent fuel			
Annual equilibrium discharge			
Plutonium fissile, kg/GWe	95	116	173
Plutonium total, kg/GWe	140	171	245
30-year cumulative discharge			
Plutonium fissile, kg/GWe	2575	3317	5169
Plutonium total, kg/GWe	3807	4888	7143
Residual U-235 in spent fuel			
U-235, wt %	0.48	0.72	0.85
Annual equilibrium discharge, kg/GWe	69	101	220
30-year cumulative discharge, kg/GWe	1854	2962	6498
Batch-average discharge burnup, MWd/t	57,130	53,670	30,400
Peak discharge exposure, MWd/t	69,450	61,700	55,000

^aThis is a nominal value. The use of fuel reinsertion from early batches lowers this value by about 2.4% on the average.

^bTails composition 0.2 wt %.

^cLong initial cycle and short second and third cycles required to make transition into equilibrium cycle with acceptable power distribution.

^dActive fuel region; blanket enrichment, 0.711 wt %.

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Table 4. Fuel Management Information — BWR,
Once-Through Cycle

	Composite-of- improvements BWR	Improved BWR (including increased burnup)	Reference BWR
Average capacity factor, %	75	75	75
Fraction of batch replaced per refueling	0.16; 0.07 ^b	0.14	0.24
Number of fresh fuel assemblies per refueling	135	122	200
Refueling interval, years	1.0	1.0	1.0
Conversion loss fraction	0.005	0.005	0.005
Fabrication loss fraction	0.010	0.010	0.010
U ₃ O ₈ requirements, st/GWe			
Initial core	444	496	454
Annual equilibrium reload	146	163	204
30-year cumulative	4631	5175	6443
Separative work requirements, 10 ³ SWU/GWe ^a			
Initial core	198	235	198
Annual equilibrium reload	92	109	124
30-year cumulative	2836	3366	3887
Equilibrium reload enrichment, wt % U-235	3.30	3.81	2.94
Plutonium in spent fuel			
Annual equilibrium discharge			
Plutonium fissile, kg/GWe	130	125	181
Plutonium total, kg/GWe	210	190	256
30-year cumulative			
Plutonium fissile, kg/GWe	3780	3549	5207
Plutonium total, kg/GWe	6100	5370	7313
Residual U-235 in spent fuel			
U-235, wt %	0.51	0.68	0.96
Annual equilibrium discharge, kg/GWe	96	119	281
30-year cumulative, kg/GWe	3005	3725	8988
Discharge fuel burnup, MWd/t	43,000; 49,000 ^b	47,000	28,400
Peak discharge exposure, MWd/t	<66,000	<66,000	<44,000

^aTails composition 0.2 wt %.

^bNormal reload fuel; reinserted fuel made up of high-enrichment discharge rods.

Recycle Systems

If reprocessing were allowed, several other LWR fuel cycles would be feasible, thus improving U_3O_8 utilization beyond that obtainable in the once-through cycles. NASAP has evaluated these cycles in PWRs; equivalent cycles are applicable to BWRs as well. The recycle fuels are encapsulated into fuel assemblies that are dimensionally identical to the present LEU assemblies. The reactor design requirements are very similar to those of the LEU cycle except for minor differences in nuclear control systems, which must be tailored to the nuclear properties of the fuel. The resource and economic characteristics of recycle LWRs can be improved by design modifications that are analogous to the once-through modifications as discussed in Section 1.1.2. However, for the recycle mode, the design which would be employed to achieve an economically optimum system is not the same as the design which would optimize recycle fuel utilization. It is anticipated that only those once-through improvement options which prove economically attractive in recycle conditions would be carried over to the recycle mode.

LWR Uranium/Plutonium Cycle -- In the past, the once-through fuel cycle was generally assumed to be a transient phase ultimately leading to recycle of the plutonium and residual uranium contained in the spent fuel discharged from the LWR. The motivation for the uranium/plutonium fuel cycle was a perception of improved economics resulting from a reduction in U_3O_8 and separative work requirements relative to the once-through cycle. The LWR uranium/plutonium cycle envisions an LWR using LEU dioxide fuel for the fresh makeup fuel and mixed-oxide fuel composed of diluent uranium and plutonium recovered from spent fuel by reprocessing. The recycle mode in which the fissile material recovered from the spent fuel of one reactor is recycled back into that reactor is called self-generated recycle. Either natural uranium or depleted uranium is added as diluent in the mixed-oxide fuel. The residual uranium (with U-236) recovered from reprocessing of the spent LEU fuel becomes feed material for enrichment or can be used as a diluent for mixed-oxide fuel. Fission products and actinides are disposed of permanently in appropriate geologic sites. Relative to the once-through LEU cycle, the self-generated uranium/plutonium recycle option leads to a reduction in cumulative U_3O_8 requirements averaged over the 30-year life of the reactor of about 32% for the PWR and about 40% for the BWR for burnups in the range of 30,000 MWd/t.

The LWR uranium/plutonium cycle can be conducted in the self-generated recycle mode described above or in the plutonium burner mode, where reactors are fueled entirely with mixed-oxide fuel containing plutonium fuel diluted

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with natural, depleted, or recycle uranium. The burner reactor concept requires an external source of plutonium from either other LWRs or from breeder reactors.

The uranium/plutonium cycle can also be carried out with the proliferation countermeasure of coprocessing. With this scheme, the uranium and plutonium are not completely separated during reprocessing. Coprocessing is expected to have minimal impact on reactor performance and minimal impact on the cost of reprocessing and refabrication.

A number of more extreme proliferation countermeasures, such as spiking with Co-60, have also been proposed. These countermeasures would be expected to have only a minor impact on reactor performance but could greatly reduce the economic benefits of LWR recycle by increasing the costs of reprocessing and refabrication.

LWR Low Enriched Uranium-only Recycle -- This cycle is similar to the uranium/plutonium cycle except that plutonium is not recycled into the LWR. The alternatives for disposition of the plutonium are that it could be (1) used in a breeder fuel, (2) safeguarded in storage for future use, or (3) disposed of with wastes in appropriate geologic sites. This results in U_3O_8 savings, for both current designs and designs involving once-through improvements. It is unlikely that this cycle would be economic unless there is ultimately a use for the plutonium.

LWR Medium Enriched U-235-Thorium -- This cycle has been called the denatured uranium/thorium cycle because fissile U-235 or U-233 is always diluted with U-238 in the fresh fuel. Consideration of this cycle is motivated by proliferation questions in that proponents have claimed that the proliferation risk would be less than that of the uranium/plutonium cycle.

The initial fuel is a mixture of 22% uranium oxide (20% U-235) and 78% thorium oxide with a net fissile enrichment of about 4.4% (in PWRs). The spent fuel is reprocessed to separate the uranium, plutonium, and thorium. The uranium, which contains U-233, is blended with highly enriched U-235 makeup to obtain a net fissile enrichment of 4.3% (in PWRs) and refabricated in a remote (and high-cost) facility into UO_2 - ThO_2 recycle fuel. A similar scheme is envisioned for this cycle in BWRs. The remote facility is needed because of gamma radiation emitted in the recycle uranium, which originates from U-232 daughter products. Wastes from reprocessing are sent to a geologic waste repository. The alternatives for disposition of plutonium are

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(1) use in a breeder or burner reactor, (2) safeguarded storage for future use, or (3) disposal with the wastes.

For burnups in the range of 33,000 MWd/t, the U_3O_8 requirement of a PWR operating on this <20% U-235-Th cycle with recycle of uranium is approximately the same as that of a PWR with self-generated recycle of uranium and plutonium, while separative work requirements are ~30% greater.

LWR Plutonium/Thorium Recycle -- This cycle utilizes plutonium extracted from spent fuel of the LEU cycle and produces U-233 for other reactors on the denatured U-233-Th cycle or on the highly enriched U-233-Th cycle.

The fuel is a mixture of plutonium and thorium oxides, with a net fissile enrichment for PWRs that ranges from 4.5 to 6.4%. The spent fuel is reprocessed to separate U-233, which is recycled into another reactor, and plutonium and thorium, which are fabricated for recycle in the plutonium/thorium reactor with plutonium makeup. Wastes from reprocessing are sent to a geologic waste repository.

Denatured U-233-Th Cycle -- The denatured U-233-Th cycle requires an external source of U-233. Potential sources of U-233 are LWRs on the plutonium/thorium cycle or fast breeder reactors containing thorium blankets. Either of these sources utilizes plutonium fuels; the denatured U-233 cycle, therefore, is driven by a plutonium cycle that might be sited under highly safeguarded conditions, while the dispersed denatured reactors could be sited with greater flexibility.

The fuel is a mixture of about 29% uranium oxide (12% U-233 in U-238) and 71% thorium oxide. The spent fuel is reprocessed to separate the plutonium, which is fabricated for recycle into a safeguarded plutonium-fueled reactor, and the uranium and thorium, which are recycled into the denatured LWR with some U-233 makeup. Wastes from reprocessing are sent to a geologic waste repository.

1.1.2 Status of Technology

Once-Through Cycles

The most widely used nuclear power reactors in the United States and the world are LWRs; thus, they are well established as commercial nuclear power

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plants. Moreover, electric utilities have a large and growing experience base for operating LWRs, and their overall performance has been satisfactory, though subject to improvements.

The LWR users and vendors have historically improved their systems to minimize power costs and to improve safety and reliability. Trends have favored larger plants, standard plant designs, higher levels of quality assurance and reliability, and longer fuel exposure. These trends have been applicable to both once-through and recycle systems. With President Carter's policy to defer fuel reprocessing, focusing attention more strongly on improved once-through cycles in LWRs has become desirable. NASAP has sponsored a number of investigations of methods to improve U_3O_8 utilization and economics in once-through fuel cycles. Participating organizations include Combustion Engineering (CE); General Electric (GE); Westinghouse; Babcock and Wilcox (B&W); Exxon Nuclear; Massachusetts Institute of Technology (MIT); Pennsylvania State University; Pickard, Lowe, and Garrick; Argonne National Laboratory (ANL); and Oak Ridge National Laboratory (ORNL).

Fuel design options have been identified that offer potential improvements in fuel utilization and economics, early implementation, and the likelihood that the concept will be suitable for licensing and deployment into currently operating reactors or current commercial offerings without undue difficulty, cost, or impact on plant operational procedures. The degree of overall fuel utilization improvement that results through combinations of selected improvements has been determined to establish how the specific improvements combine. In addition to identifying and characterizing promising once-through fuel-cycle improvement options, a determination has been made of the impact of these once-through improvements on performance in the recycle mode of operation, that is, which improvements also carry over into the recycle mode and whether the incentive for recycle is diminished as a result of the implementation of once-through cycle improvements.

There is significant potential for improving U_3O_8 utilization in the once-through cycle in current LWRs (as indicated by the following discussion). However, because of the anticipated reduction in power-generation cost resulting from perceived improved availability and reduced personnel exposure due to less frequent refueling shutdowns, there are practical considerations that tend to limit the degree of U_3O_8 savings which will actually be attained. For example, the current utility trend toward longer cycles (e.g., 18-month cycles) has a negative effect on U_3O_8 utilization.

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Compared with a 12-month cycle, an 18-month cycle in a PWR requires about 7 to 10% more U_3O_8 at current burnup design limits, with the U_3O_8 penalty associated with longer cycles becoming smaller as burnup design limits increase; BWRs show a similar trend. Therefore, to a certain degree, U_3O_8 utilization improvement options (such as development of technology for extended fuel burnups) are required to counteract the negative effect of the trend toward longer cycles. Deployment of U_3O_8 utilization improvement options to a large enough extent can result in appreciable U_3O_8 savings even if longer cycles are adopted. Furthermore, the potential improvements tend to provide larger percentage savings for 18-month cycles than for 12-month cycles, resulting in increased incentive to develop the various options.

The once-through fuel-utilization improvements are categorized into those suitable for deployment in existing designs and those that are not retrofittable. Most of the NASAP investigations have covered near-term retrofittable improvements which could be implemented before 1990. These can be subdivided into the following classes:

- fuel-assembly changes leading to more reactive geometries,
- extended-exposure assembly designs,
- fuel-management changes,
- plant operational changes.

These classes of modification are discussed below for PWRs and BWRs.

Retrofittable PWR Improvements -- Table 5 provides a summary of U_3O_8 savings from individual retrofittable improvement options for PWRs. Results shown are based primarily on analyses performed by CE for the reference NASAP LWR design described in Section 1.1.1. However, for certain improvement options, a range has been presented to reflect estimates provided by other NASAP participants. The ranges result from differences in PWR designs analyzed, differences in level of calculational detail, and differences in reference values of fuel-management parameters. Fuel-utilization benefits

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Table 5. Reduction in U_3O_8 Requirement for Various Potentially Retrofittable Improvement Options in PWRs

Improvement type	Reduction in 30-year U_3O_8 requirement (%) ^a
Increased burnup	
50 MWd/kg, annual refueling	11
50 MWd/kg, 18-month refueling	5
Lattice changes	
Annular pellets; reduced fuel-rod diameter	2-3
Variable-lattice fuel assembly design	1-4
Improved fuel management and control design	
Low-leakage fuel management	2-3.5
Other	1
Enrichment zoning/fertile blankets	
Pinwise enrichment zoning within assembly	0.1
Axial blanket	2-4
Radial blanket	b
End-of-cycle coastdown	3
Full use of batches in startup core	0.5

^aRelative to 30 MWd/kg, annual refueling.

^bNo feasible retrofittable scheme has been identified.

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attainable from individual options are not necessarily additive when options are combined. In fact, some are mutually exclusive. Table 6 lists a number of retrofittable PWR improvement options which provide a composite saving of about 21% when employed with a 12-month cycle.

Each type of once-through fuel-cycle improvement presents a technique of reducing the fissile material loading required at beginning of cycle per unit energy production. This can be done by reactor design and operating strategies that improve in situ production and utilization of bred fuel, more fully deplete the U-235 component, or reduce nonproductive losses of neutrons in the reactor. The beginning-of-cycle fissile requirement consists of two components:

1. the fissile material depletion component, consisting of fissile material consumed to produce power,
2. the fissile inventory component required for criticality at end of cycle.

While the first component decreases with increasing conversion ratio (which favors both recycle and once-through options), the second decreases with decreasing U-238 captures (thus favoring once-through options but not recycle). Therefore, certain once-through cycle improvements decrease, rather than increase, the conversion ratio. The effect on conversion ratio determines fuel-utilization aspects of operation in the recycle mode, as discussed in Section 1.1.2.

Fuel-Assembly Changes Leading to a More Reactive Assembly -- As previously mentioned, the beginning-of-cycle fissile inventory requirement can be decreased either by decreasing the fissile material depletion component or by decreasing the end-of-cycle fissile inventory component. Increasing the water-to-fuel ratio reduces the end-of-cycle fissile inventory component but this reduction is partially offset by an increase in the fissile material depletion component because the conversion ratio is reduced.

Increases in the water-to-fuel ratio can be obtained by decreasing fuel-rod diameter slightly while maintaining fuel-rod pitch, by employing pellets with an inert diluent, or by utilizing annular fuel pellets (annular fuel pellets have a void central region, thereby reducing the fuel volume). Since these options do not change the overall dimensions of the fuel assembly, they are retrofittable into existing reactors. The small reduction in heat transfer

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Table 6. PWR Improvements for
Composite Case

Extended exposure, annual cycle
Low-leakage fuel management
Lattice optimization
Axial blankets
Preplanned coastdown
Full use of early batches of startup core

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area resulting from reducing rod diameter is not expected to necessitate a reduction in reactor power output; the heat transfer area is not altered when fuel with an inert diluent or annular fuel is used. Fuel assemblies with reduced fuel-rod diameter are currently being developed by some PWR vendors to improve fuel-cycle economics.

For any of the options mentioned, there is a constraint on the degree the water-to-fuel ratio can be increased which limits the U_3O_8 savings to much less than the maximum achievable. The moderator temperature coefficient becomes less negative with increasing water-to-fuel ratio. To avoid operational problems and an adverse impact on certain accidents, such as anticipated transient without scram, the moderator temperature coefficient should remain negative throughout the cycle. The reduction in U_3O_8 requirements provided by these options is 2-3% when this constraint is considered.

Annular pellets, pellets employing an inert diluent, and reduced-diameter fuel rods all reduce the heavy-metal loadings, thus requiring higher burnups for the same energy production. Thus, U_3O_8 savings by these methods are not entirely additive to those obtained through increasing fuel burnup.

Another option that takes advantage of increased reactivity associated with a wetter lattice is the variable-lattice fuel assembly concept. This is a concept in which each fuel batch is irradiated for its first one or two cycles of core residence with a dry (undermoderated) lattice to increase local conversion ratio and to reduce excess reactivity, thereby reducing neutron capture in control poison. After the initial cycle(s) of irradiation, the water-to-fuel ratio of the assembly lattice is increased, resulting in an increase in the assembly k -infinity because of the change in the optimum water-to-fuel ratio with burnup and the large inventory of fissile plutonium built up during the undermoderated period of irradiation. This change in water-to-fuel ratio would be accomplished during refueling outage by removing a small percentage of fuel rods from the lattice. These rods would be reassembled into other assemblies and reinserted. Preliminary analyses have indicated that this option would require employing more rods per assembly than current designs in order to satisfy departure-from-nucleate-boiling-ratio (DNBR) requirements.

Initial scoping analyses have indicated U_3O_8 savings of from 1 to ~4%. Areas of concern that would need further evaluation if the potential for U_3O_8 savings determined by more detailed neutronic assessments appears favorable are (1) operating limits associated with avoidance of

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departure from nucleate boiling, (2) potentially less negative moderator temperature coefficient, and (3) mechanical concerns relative to reassembly of irradiated rods.

Extended-Exposure Assembly Designs -- One of the most straightforward means of improving uranium utilization and power generation economics is to develop technology supporting increased fuel exposure (burnup) limits. A higher discharge burnup improves U_{308} utilization because of the increased production and in situ utilization of bred plutonium and the more complete depletion of U-235. Current discharge exposure limits represent assurance of acceptable fuel performance and licensing acceptance; they do not represent an optimum with regard to U_{308} use or with regard to fuel-cycle cost.

At present, fuel is discharged from a typical PWR after it has achieved a batch-average burnup ranging between 30,000 and 34,000 MWd/t. The design burnup for PWRs planned to go into commercial operation in the early to mid 1980s is typically in the range of 34,000 MWd/t. A substantial reduction in U_{308} requirements would be realized if the discharge exposure were increased to the 50,000-MWd/t range.

Two methods of increasing the discharge burnup are to (1) increase the cycle length and maintain the batch size or (2) maintain the cycle length and increase the number of batches. Many PWRs currently have an annual cycle length and use "three-batch" fuel management.

If the batch size is maintained and the cycle length increased to 18 months, the 50% increase in discharge exposure (from ~30,000 to 45,000 MWd/t) would result in a 4 to 6% reduction in U_{308} requirements. If the annual cycle is maintained and the number of batches increased, thus increasing the burnup of the batch being discharged, a larger saving in U_{308} is achieved. In PWRs, the use of five core batches with annual refueling and a discharge burnup of 50,000 MWd/t instead of the use of three core batches with annual refueling and a discharge burnup of 30,000 MWd/t results in a reduction of U_{308} requirements of ~11%. Table 7 compares resource requirements for a reference three-batch 30,000-MWd/t annual-refueling once-through cycle with those for a five-batch 50,000-MWd/t annual-refueling once-through cycle. It is highly desirable to employ extended exposure in combination with the low-leakage fuel-management option. This combination leads to a combined saving of about 13%. Additional U_{308} savings could be obtained by further extending burnup and number of batches, but for 12-month cycles the additional savings accrue at a greatly diminishing rate.

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Table 7. PWR Fuel Resource Requirements
for Extended-Exposure Cycles^a

	Standard three- batch annual refueling	Extended burnup/five-batch annual refueling
Fuel exposure, MWd/t	30,400	50,600
U ₃ O ₈ requirements, short tons/GWe		
LEU once-through		
Initial core	394	471 ^b
Equilibrium cycle	194	171
30-year cumulative	6,040	5,395
Self-generated recycle (SGR) plutonium and uranium recycle		
Initial core	394	471 ^b
Equilibrium cycle	120	125
30-year cumulative	4,130	4,285
Separative work, 10 ³ SWU/GWe		
LEU once-through		
Initial core	200	264 ^b
Equilibrium cycle	117	118
30-year cumulative	3,600	3,670
SGR plutonium and uranium recycle		
Initial core	200	264 ^b
Equilibrium cycle	81	84
30-year cumulative	2,750	2,775

^a75% plant capacity factor and 0.2 wt % tails assay. Numbers presented include 0.5% conversion loss, 1% fabrication loss, and, for recycle cases, 1.0% reprocessing loss.

^bLong initial cycle and short second and third cycles required for transition into equilibrium cycle with acceptable power distributions.

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Fuel-cycle costs (shown in Table 8) have been estimated for the reference case and for the extended-exposure case. These costs have been developed at U_3O_8 prices of \$40 and \$100/lb using unit costs for fuel-cycle operations from the NASAP data base (Appendix C, Vol IX). As can be seen, the incentive to extend fuel burnup increases as U_3O_8 prices increase.

As has been indicated, several utilities are interested in longer cycle lengths than the NASAP reference of 12 months. Major considerations are that longer cycles offer the potential for improved availability and reduced personnel radiation exposure due to the less frequent refueling operations. Results obtained by Combustion Engineering, Inc. indicate that the 18-month cycle with a burnup of about 39,000 MWd/t, breaks even in U_3O_8 requirement with the 12-month 30,400-MWd/t cycle.

Before extending the batch-average burnup in PWRs much beyond 34,000 MWd/t, the effect on fuel performance must be carefully assessed. Realistic burnup limits for current fuel designs must be carefully estimated and design changes evaluated as to their potential for increasing fuel exposure while maintaining or improving fuel performance.

It has been determined that no significant balance-of-plant modifications are anticipated in going to higher burnups. However, RD&D would be required to develop new fuel designs capable of reliably and consistently achieving burnups of 40,000 to 50,000 MWd/t. The initial phase of the RD&D program should consist of irradiating a few fuel assemblies to the discharge exposures anticipated for high-burnup LWR fuels. These irradiations will provide information required for the design of high-burnup fuel. The effects that must be considered are

- fuel-rod internal pressure,
- pellet-cladding interaction,
- fuel-assembly dimensional changes,
- external cladding corrosion.

Each of these concerns is discussed in more detail.

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Table 8. PWR Fuel-Cycle Cost for Extended-Exposure Cycles

Cycle	Burnup (MWd/t)	30-year levelized fuel-cycle cost (mills/kWhr) ^a	
		\$40/lb U ₃ O ₈	\$100/lb U ₃ O ₈
Once-through	30,400	6.0-6.5	10.4-10.9
Once-through	50,600	5.6-6.0	9.6-10.0
Plutonium/uranium SGR	30,400	5.6-6.6	8.7-9.7
Plutonium/uranium SGR	50,600	5.3-5.9	8.6-9.2

^aJanuary 1978 \$.

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1. Fuel-rod internal pressure. The fission gas released from the pellet as the fuel is exposed increases the internal rod pressure above that caused by internal helium prepressurization. For any specific design, end-of-life internal rod pressure limits the attainable fuel exposure. It is necessary to determine the degree to which burnup limits for current fuel designs can be extended and to evaluate design changes which would allow even higher exposures. Fuel design modification options which potentially would reduce internal gas pressure include (1) increasing the volume of the fission-gas plenum by extending the length of the fuel rod to provide a longer fission-gas plenum, (2) employing alternate fuel pellet geometries, such as annular pellets, and (3) reducing the initial helium prepressurization.

The use of annular pellets reduces fuel-rod internal pressures for two reasons: (1) the central region provides additional space to accommodate fission gas and (2) fuel temperatures are lower for annular pellets because the high-temperature central region is removed. Conservative models of fission-gas release indicate that the reduction in fission-gas release caused by the lower temperature and the additional space to accommodate fission gas proves to outweigh the higher burnup of annular fuel (the burnup of annular fuel is increased because the number of metric tons of heavy-metal fuel loading is reduced), and significant reductions in internal rod pressures occur.

Fission-gas-release data from high-burnup fuel will be needed to determine the extent of fuel-rod design change required for high-burnup fuels.

2. Pellet-cladding interaction (PCI). During irradiation, cladding creep-down and fuel swelling eventually lead to pellet-cladding contact. Certain fission gases such as iodine in combination with high localized stresses are known to cause PCI by stress-corrosion cracking. Stress corrosion in combination with irradiation damage of Zircaloy will potentially increase the probability for PCI, resulting in a need to develop fuel more resistant to PCI failure. PCI may be aggravated by the longer residence time associated with extensions in burnup because of the greater number of operational power ramping cycles to which the fuel would be subjected. Also, the modified fuel management schemes which are anticipated to accompany extended burnup cycles may aggravate PCI somewhat due to the possibly more severe changes in power duty experienced from one cycle to the next. To establish whether fuel having an extended burnup has a greater tendency to fail due to PCI, power ramp tests will have to be performed for high-burnup fuel. Results of these tests could be used to establish the acceptable combinations of local burnup, power level, and rate of power change for high-burnup fuels.

3. Fuel-assembly dimensional changes. Irradiation can cause both stress-free and stress-induced permanent dimensional changes in the components of the fuel assembly. Although engineering evaluation and tests are needed to determine whether current fuel assembly designs have adequate margin at extended burnup or whether design changes are required, some dimensional changes can likely be accommodated by provision in the fuel-assembly design.

4. External cladding corrosion. Under normal steady-state operation, current LWRs have not experienced excessive corrosion on the outside surface of fuel rods. During operation an oxide layer forms as a result of a chemical reaction between Zircaloy and the coolant, and circulating corrosion products (crud) released from steam generators and piping can deposit on the oxide film. If exposures and residence times are increased significantly, the thicker layer of oxide and crud deposits may possibly increase the oxide-cladding interface temperature sufficiently to further accelerate the corrosion of the Zircaloy cladding. If exposures and core residence times are increased significantly, the adequacy of correlations currently used to describe in-core Zircaloy corrosion as a function of time and temperature will have to be reevaluated. In particular, the combined effects of increased fast- and thermal-neutron irradiation damage on cladding corrosion rates must be carefully examined relative to safety and licensing concerns.

Electric Power Research Institute (EPRI), Department of Energy (DOE), and the nuclear industry are very actively pursuing extended-burnup fuel development programs. Currently, DOE has under way several high-burnup demonstration programs involving vendor/utility teams. These programs are intended to demonstrate the extended burnup capability of current fuel designs. In the first of these, in Duke Power Company's Oconee reactor, five B&W fuel assemblies that had completed their planned three-year irradiation period have been examined and reloaded to burn an extra cycle. They will be examined further following irradiation to determine the fuel performance, with four of the assemblies attaining a burnup of about 40,000 MWd/t. DOE is also sponsoring an effort involving Arkansas Power and Light and B&W to design fuel capable of achieving a batch-average discharge burnup of 45,000 MWd/t. Demonstration plans involve irradiation of two sets of lead test assemblies followed by an entire reload batch of approximately 59 assemblies. EPRI programs for obtaining fuel-performance data for extended burnups of existing fuel designs are being pursued in the Zion, Peach Bottom, Calvert Cliffs, Trojan, and Oyster Creek plants.

DOE is also currently sponsoring LWR fuel-cladding interaction remedy programs to develop and demonstrate fuel designs capable of improved fuel-rod cladding integrity under conditions of power ramping and other maneuvers, through increased resistance to fuel-cladding interaction both at current

burnups and at extended burnup levels. Fuel concepts selected for evaluation are annular pellets, graphite, zirconium, or copper coatings on the inside cladding surface and packed particle fuel (sphere-pac and vipac). Information developed in these programs will be beneficial in selecting design modifications necessary for accommodating extended exposures as well as improving fuel performance.

Fuel Management Changes -- Increasing the refueling frequency is one of the most effective methods of reducing U_3O_8 requirements but one which introduces undesirable downtime with current LWR designs. It is doubtful that utilities would adopt options that sacrifice availability for fuel savings unless this were justified by very high U_3O_8 prices. This option is therefore classified as nonretrofitable and is discussed with other non-retrofitable options later in this section.

There are several other potential fuel-management and control changes, each of which could provide small improvements in resource utilization. These involve more uniform batch burnup, better poison-control schemes (which minimize end-of-cycle poison residual), and low-leakage fuel-management schemes.

A low-leakage fuel-management option involves loading fresh fuel in the inner portion of the core in a checkerboard array along with the fuel batch consisting of the most burned assemblies, rather than loading fresh assemblies on the core periphery, thus reducing radial core leakage. This fuel-management option is called the in-out-in scheme. This option has been investigated thoroughly by several PWR reactor vendors and will be put into practice in some of the standardized plants scheduled for operation in the early to middle 1980s.

Low-leakage fuel-management schemes that load fresh assemblies in the core interior with burnable poison are particularly attractive because they complement the extended-exposure option. The higher feed enrichment required for extended exposure results in increased core leakage when fresh assemblies are loaded on the core periphery. The benefit of low-leakage fuel management is greatest for smaller PWR cores that have inherently large radial leakage. For these plants, 3-4% savings in U_3O_8 are attainable by this technique. Estimates of U_3O_8 savings for large PWRs (1100-1300 MWe) range from 2 to 3.5%. The low-leakage fuel-management option requires either fabrication and disposal of lumped burnable poison clusters or fabrication of fuel pellets containing gadolinia. Additional benefits may be obtainable through optimization of burnable poison loadings to minimize the end-of-cycle reactivity penalty associated with the residual poison. Benefits attainable

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by coupling low-leakage fuel management with extended exposure (50,000 MWd/t) for the 12-month cycle are expected to be about 13%.

Fuel batches of the start-up core which are discharged after the first and second irradiation cycles achieve low exposures (in the range of 10,000 to 24,000 MWd/t). Although this fuel does have a lower initial enrichment than batches which achieve full burnup, there is significant residual reactivity remaining in these batches at discharge. Reinsertion of this early-batch fuel has been employed, and this reduces U_3O_8 requirements. Although the lifetime-average U_3O_8 reduction is small (less than 1%), the benefits are realized early in reactor lifetime. Many reactors presently have these startup batches in their spent-fuel pool.

Plant Operational Changes -- Uranium utilization can also be improved if the utility plans to operate the reactor during the last days of the cycle prior to shutdown for refueling at slightly reduced electrical power. Operation at reduced thermal power to gain reactivity needed to achieve cycle length is called preplanned coastdown. A gain in reactivity to allow continued operation at reduced thermal power can be achieved through decreased moderator temperature, reduced Doppler broadening in U-238, and reduced xenon level. With preplanned coastdown, batchwise enrichment loadings would be chosen so that the coastdown period is required in order to meet cycle lifetime requirements. Also, if the feedwater temperature is reduced by bypassing some feedwater heaters, extended operation at full thermal power is attained but at a lower thermal efficiency. This technique of feedwater temperature reduction precedes the thermal power coastdown period.

The optimum coastdown period is a function of U_3O_8 price and replacement power costs. While there is a broad minimum in relative energy cost, for annual cycles with ~50,000 MWd/t fuel a coastdown period of 8 to 14 days each cycle yields near-minimum total energy costs for replacement power costs ranging from 20 to 40 mills/kWhr with \$40/lb U_3O_8 . If coastdown is preceded by feedwater temperature reduction during this 8- to 14-day period, the electrical power will have decreased to the 90 to 75% range and will result in about a 3 to 4% reduction in U_3O_8 requirements per unit of energy generated and a 2% reduction in relative energy cost. Results showing similar trends are obtained for annual cycles employing ~30,000 MWd/t fuel. Preplanned coastdown has been employed to some extent. Many utilities have chosen to reserve coastdown exclusively as an option for controlling the timing of refueling outages to periods of low load demand.

Miscellaneous PWR Fuel Improvement Options -- Other potential options, such as the use of duplex pellets with a natural uranium inner region, optimization of axial shim (burnable poison rod) cutback, use of natural uranium as shims, pellet diameter reduction around control-rod guide tubes, and assembly reinsertion after removal of high-burnup rods, provide individual reductions in U_3O_8 of between 0 and 0.3%.

Zircaloy-4, which is used in all current commercial offerings as fuel-pin cladding and in some offerings as the grid material, consists of several isotopes having differing absorption cross sections. Small U_3O_8 savings of about 2% would result if isotopically improved (by molecular laser isotope separation processes) Zircaloy, enriched in the low-cross-section isotopes, were used for cladding, grids, and guide-tube material. Commercial prospects for isotopically improved cladding in PWRs are quite uncertain at this time, and the option is considered only a long-range possibility.

Enrichment zoning in the horizontal plane ($x-y$) within a PWR assembly will not lead to significant U_3O_8 savings because the assembly radial power (and, therefore, discharge burnup) distribution is fairly flat. Savings in U_3O_8 from this option are estimated to be negligible.

The use of retrofittable axial enrichment zoning (blankets) typically involves use of lower-enriched pellets in the top and bottom few inches of the fuel stack and an increased enrichment in the remainder of the fuel stack. The lower-enriched pellets at the top and bottom of the stack can be natural or depleted uranium or simply of a lower enrichment than the remainder of the fuel stack. Since the overall length of the fuel column is maintained the same as the reference design with this application of enrichment zoning, its use has a relatively small effect on axial neutron leakage. The improvement in U_3O_8 requirement for the once-through cycle results in part from the reduction in axial leakage but primarily from the displacement of the higher-enriched fuel at the ends of the fuel stack, which cannot be fully utilized because of the lower neutron flux at these locations.

A 2 to 4% reduction in U_3O_8 requirements can be obtained by replacing 6 in. of enriched fuel with natural or depleted UO_2 pellets. Very similar savings result if pellets of lower enrichment than the remainder of the fuel stack are used in this 6-in. region. Use of enrichment zoning is achieved at the expense of an increase in peak linear heat rate caused by the reduced length of the higher-enriched region. This would lead to a reduction in thermal operating margin.

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In large PWRs, because the peripheral fuel assemblies constitute about 20% of the core volume, use of radial blanket material in these locations is not practical due to the large increase in power density that would be required in the remainder of the core. The use of radial blanket elements of a size appropriate to fill in corners around the periphery of a core to reduce leakage has been investigated by CE. In this case, the lower-burnup rods in discharged assemblies were assumed to be reassembled into these blanket elements. A reduction in U_{308} of $\sim 1.5\%$ resulted from the reduced core leakage and the increased burnup obtained in these rods. For reactors currently operating or under construction, it is very doubtful that the core design changes required to backfit the core to accommodate the blanket elements could be justified.

A variation of the blanket concept in PWRs is to use fertile rods (perhaps containing ThO_2) or rods containing a mixture of fertile material and burnable poison in certain poison pin locations within the fuel elements. Fertile material is added to the core, while not eliminating locations in the core that have been designated as fuel-rod locations. Only minor U_{308} benefits would be expected because most PWRs use burnable poison only in initial cores. However, the scheme does have potential for reactors employing in-out-in fuel management, which appears desirable for extended-burnup cores.

Retrofittable BWR Improvements -- Several options for improving fuel utilization of BWRs have also been identified. Certain of these options, which involve a slightly modified fuel assembly design and modified fuel management, are currently being backfitted into BWRs.

Typically, the BWR fuel assembly resides in the reactor core for a four-year period. Low-enrichment rods are on the outside of the fuel assembly, and the enrichment increases toward the interior of the assembly, with the highest enrichment in the center. This grading reduces power peaking in the very heterogeneous fuel/control-blade configurations of the BWR. The center of the assembly has two water rods introduced to increase reactivity of the central zone and thus improve power flattening. Gadolinia is introduced into selected fuel rods for reactivity and power distribution control with burnup.

Improvement options for BWRs are presented below:

1. Recently introduced fuel design and fuel management modifications that improve BWR uranium utilization:

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- natural-uranium axial blankets,
- three-assembly-average-enrichment initial core,
- improved refueling patterns.

2. Other potential options for improving BWR uranium utilization:

- increased burnup,
- end-of-cycle coastdown,
- reassembly and reinsertion of high-enrichment-discharge fuel rods,
- increased water-to-fuel ratio,
- coolant density spectral shift,
- selective use of ThO_2 rods to breed U-233 followed by disassembly and reassembly of ThO_2 rods into new assemblies.

Blankets improve fuel utilization by capturing in fertile material neutrons that otherwise would leak from the core. This breeds fissile material, some of which is burned in situ. General Electric is currently utilizing natural-uranium axial blankets in initial and reload BWR cores. In most cases involving the blanketed fuel assembly, a 6-in. blanket region is used at the top and bottom of the fuel column, with the length of the enriched region being decreased by 8 to 12 in. depending on the reactor design. While the U_3O_8 savings offered solely by axial blankets has not been evaluated, the combination of axial blankets with the other improvement options currently being backfitted and discussed below (i.e., three-assembly-average-enrichment initial core and improved refueling patterns) provides a 6% reduction in U_3O_8 requirements and a 10% reduction in separative work requirements.

A recent BWR fuel management change is the use of a three-assembly-average-enrichment initial core. In earlier BWR designs employing a uniform

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assembly-average enrichment in the initial core, the fuel in the batches that were to be discharged after the initial cycle(s) of operation was not fully utilized. An improved first-core loading pattern employs three different assembly-average enrichments. The enrichment zone around the core periphery consists of natural uranium. The use of natural uranium in the peripheral fuel-assembly locations decreases neutron leakage significantly relative to the reference uniform-enrichment case. By use of a three-enrichment-zone initial core, the U_3O_8 requirements of the initial core are reduced ~10%. Small additional U_3O_8 savings are attainable by further increases in the number of zones.

A recent improvement in refueling patterns for reload batches in BWRs involves loading last-cycle fuel in peripheral core locations to minimize leakage. High-reactivity (fresh) fuel is loaded just inside the periphery to flatten the power distribution. An advantage of the improved refueling patterns is that movement of fuel from low-power regions at the core periphery to high-power regions in the center of the core is largely avoided, thus reducing the probability of failure resulting from PCI. Therefore this option not only improves U_3O_8 utilization but also improves reliability and performance.

As is the case for the PWR, increasing fuel burnup while maintaining refueling frequency (i.e., increasing the number of batches) improves fuel utilization. The reduction in U_3O_8 requirements as burnup is increased is shown in Figure 5 for an annual cycle, an 18-month cycle, and a 24-month cycle. As shown, if fuel exposure is increased from ~28,400 to ~47,000 MWd/t for the annual cycle, the equilibrium cycle U_3O_8 requirement is reduced by ~13%. For an 18-month cycle, a burnup of ~37,000 MWd/t is required to break even in U_3O_8 with a 12-month cycle at a fuel burnup of 28,000 MWd/t. In assessing the technical viability of extended exposure, control requirements resulting from the higher fuel enrichment and transient performance need to be evaluated in detail.

Uranium can be saved by varying the electrical power during a planned cycle coastdown, accomplished by thermal power reduction only or by operating the reactor for a period at 100% thermal power but at reduced feedwater temperature, followed by thermal power coastdown. If the thermal power coastdown is preceded by feedwater temperature reduction at full power, a 5% saving in U_3O_8 relative to the reference case results from a six-week coastdown (to ~90% power), while a 10% U_3O_8 saving results from a 12-week coastdown (to ~75% power). This assumes that the cycle coastdown is employed in each cycle.

Figure 5. Relative U_3O_8 Requirement vs Burnup for Various Cycle Lengths in BWRs.

Because of the heterogeneity introduced by the large water gaps employed in the BWR, there is a large spectral variation from the assembly periphery to the assembly interior. To flatten the assembly power distribution, the peripheral rods in the assembly are loaded with a lower enrichment than rods nearer the assembly interior because of the more thermal spectrum near the water gap. A typical rod-by-rod discharge burnup distribution shows that the burnup distribution is fairly uniform; thus the uranium utilization in the interior rods is poor. The uranium utilization could be improved significantly if the interior rods are removed at discharge, reassembled, and reinserted for one additional cycle. This would require that the mechanical integrity of the fuel rods be maintained to higher burnups. During the additional irradiation cycle, the exposures of the reassembled rods would be extended by ~6,000 MWd/t.

This option requires the development of poolside techniques for removal and reassembly of large numbers of fuel rods. BWR fuel assemblies have frequently been taken apart, inspected, reassembled, and reinserted routinely. With current procedures, the dismantle and reassembly process would need to be performed between refueling outages, resulting in the reassembled bundles being ready for reinsertion one year after discharge.

Because of the axial void distribution in BWRs, the bottom half of the fuel rods is more heavily exposed than the top half, resulting in an axial variation in U-235 and plutonium concentration at discharge. If inversion of the fuel rods were performed during the reassembly procedure discussed above, additional reactivity could be obtained. Estimates of U₃O₈ savings attainable from reassembly (and inversion) of high-enrichment-discharge rods range from ~4 to 12%. The effect of reassembly (and possible rod inversion during reassembly) on the core and on assembly power distributions and various licensing concerns needs to be investigated before the technical viability of this option can be determined.

The relative fuel density for a given water-to-nonwater volume ratio can be reduced by using hollow annular pellets. Results based on scoping studies and one-dimensional analyses indicate that the attainable reduction in U₃O₈ requirement is in the 0-to-5% range.

In the BWR spectral-shift scheme by void control, the reactor is operated with low coolant recirculation flow throughout most of the cycle to increase conversion ratio and reduce parasitic capture in control absorbers, followed by operation at maximum coolant recirculation flow at end of cycle. The potential U₃O₈ savings are in the range of 4 to 8%. The degree to which BWR spectral shift by void control could be employed in existing plants with

no major changes in fuel or plant hardware depends on the particular plant design and remains to be determined. Operation at less than rated flow depends on the available excess thermal margins. All modern BWRs have flow control; some have the capability to control core inlet temperature by bypassing feedwater heaters. Many BWRs have excess flow capacity or have the potential for excess flow capacity. To determine the acceptable flow range at 100% power operation for retrofit of this concept and to obtain NRC acceptance would require further experimental examination of BWR transient CHF characteristics, flow-induced vibration measurements, steam separator performance measurements and other testing involving primary-system components.

Preliminary evaluations have indicated that improvements in reactor performance (thermal and transient margins) and small reductions in U_3O_8 use can be achieved through the use of a mixed-lattice UO_2/ThO_2 fuel assembly design employing standard BWR fuel assembly hardware combined with the reassembly-and-reinsertion option. In the mixed-lattice design, 4 pure ThO_2 rods (or 12 in alternate designs) are located at the corners of the BWR fuel assembly. The remaining UO_2 rods are adjusted in enrichment to obtain a desirable local power distribution and to achieve the desired reactivity lifetime.

Due to the use of ThO_2 in the mixed-lattice design, less gadolinia is required for power distribution and reactivity control. The ThO_2 rods (which would contain ~1.2 to 1.5% U-233 upon initial discharge) would be removed and reassembled into fuel elements that would be loaded on the core periphery (or in other reactors) and irradiated until their mechanical lifetime is reached. With this reassembly-and-reinsertion option, U_3O_8 requirements can be reduced by ~4% relative to the reference LEU once-through cycle. Alternatively, use of the four- ThO_2 -corner-rod design would produce about 24 kg of U-233 per 0.75 GWe-year, which could be recovered and reused at a later time if a thorium cycle were adopted.

Fuel-cycle evaluations have been made with a three-dimensional multi-cycle nuclear and thermal-hydraulic computer program for a composite case, with the following options:

- four-enrichment initial core,
- axial blankets,

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- improved fuel management with last-cycle low-reactivity fuel on periphery,
- spectral shift using flow control (80 to 120% of rated flow),
- feedwater temperature reduction (235 to 120°C) and thermal coastdown to 80% power,
- high-enrichment-discharge UO_2 fuel-rod reassembly and reinsert,
- increased burnup (51,000 MWd/t).

This analysis indicated a reduction in U_3O_8 requirements of 28% relative to the reference case as presented in Table 4. If the additional option involving use of ThO_2 corner rods combined with reassembly and reinsertion of these rods were included in a composite case, potential would exist for savings of ~30% relative to the reference case. This composite case is presented as an indication of the potential for U_3O_8 utilization improvement. A thorough analysis of safety, licensing, and operability of certain options has not been performed. Furthermore, economic aspects of some options have not been thoroughly evaluated. Therefore, the degree to which the various options which have been included in this composite case would actually be implemented is somewhat uncertain.

Nonretrofitable LWR Once-Through Improvements -- More extensive improvements than those discussed above would be required to achieve a reduction of over 30% in U_3O_8 requirements in LWRs.* Advanced once through LWR designs have not been evaluated within NASAP to the degree required to examine technical or economic feasibility. Nonretrofitable improvement options are considered to be those options which could provide resource and economic benefits if incorporated into a new design but for which the cost of retrofitting into current plants would be prohibitive and/or for which design

*The practically achievable fuel utilization in LWRs employing various fuel utilization improvement techniques (e.g., increased burnup, reflector blankets, movable fuel control, low-neutron-absorption structural material), including some under development by the water-cooled breeder reactor program, is estimated to be about 17 MW-years (thermal) per short ton of U_3O_8 . Compared with the fuel utilization of the NASAP reference PWR shown in Table 3, this represents a reduction in U_3O_8 requirements of about 31%.

constraints (e.g., pressure vessel or core internals design) or technical constraints would prohibit retrofit. A thorough examination of technical and economic aspects of options currently classed as nonretrofitable might determine that certain of these are retrofitable. Furthermore, some options may be retrofitable only in special instances; for example, an option might be implemented during an extended outage for extensive repairs of other equipment. A brief discussion of options currently classed as nonretrofitable follows:

- Rapid refueling allowing short operating cycles. Increasing the refueling frequency for a given discharge exposure allows increasing the number of fuel batches in the core and thereby improving fuel utilization. For 50,000-MWd/t fuel, shortening the cycle length from 12 to 6 months provides a 9% reduction in U_3O_8 requirements. Compared with the reference annual cycle with 30,000-MWd/t fuel, the combination of six-month cycles and 50,000 MWd/t fuel would reduce U_3O_8 requirements by 18 to 20%. For current economic conditions, commercial adoption of six-month fuel cycles would require new reactor plant designs specifically designed for refueling in a time period approaching one week. If U_3O_8 prices increase significantly, six-month cycles might be commercially acceptable even if refueling required a slightly longer period of time. Normal plant maintenance could be done every 12 to 18 months during longer shutdown periods. Special pressure-vessel closures and equipment for rapid refueling would have to be developed.
- Movable fuel control. Movable seed-blanket control concepts such as used in the LWBR have a large potential for improving fuel utilization by eliminating neutron losses to control poison. The seed-blanket control concept is described in detail in Section 1.3. Neutrons captured in fertile control material could provide additional bred fuel to be consumed in long-burnup cycles. Major design changes would be required.
- Extensive use of blankets and improved reflectors. Potential use of radial and axial blankets has been discussed above. Radial blankets are feasible in present BWRs but not in PWRs; a larger-diameter core and pressure vessel would be required for the PWR.

Current-design PWRs use a stainless steel baffle (or shroud) around the core periphery. Analyses show that improved neutron economy, resulting from reduced lower radial leakage, could be obtained if

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Zircaloy were to replace the stainless steel in the baffle. Retrofitting existing operating plants would entail significant difficulties and excessive costs. In new plants, uranium savings of 1 to 2% might be realized, depending on the core size and on the fuel management employed (low-leakage vs. standard out/in).

- Low-specific-power core. Gains in fuel utilization in the range of 3% can be made through increasing core size and employing lower specific power, thus allowing the use of more in-core batches. These benefits would tend to be offset by the higher fuel inventory cost and the increased plant capital cost.
- Wide range spectral shift through void control in BWRs. Future BWRs could be designed to accommodate a wider range of spectral-shift operation than could be achieved in existing BWRs. Substantial reductions in U_3O_8 requirements are possible through this technique.

Recycle Systems

Technical Status of the Recycle Mode in LWRs -- Extensive safety and licensing reviews of uranium/plutonium mixed-oxide fuels have been carried out for the Big Rock Point, Quad Cities Unit 1, and Dresden Unit 1 reactors, and large numbers of test mixed-oxide fuel elements have been successfully irradiated. No fundamental technical problems have been uncovered which would preclude the use of uranium/plutonium fuels in current LWRs. At least one U.S. vendor offers an NSSS that is designed to accept either conventional UO_2 or a full core loading of mixed uranium/plutonium oxide. All vendors offer the capability for a certain degree of plutonium fueling. However, the commercial use of recycle fuel has not been licensed domestically, as a result of the U.S. policy to defer commercial reprocessing.

Extensive irradiations of thorium-based fuels have been carried out in LWRs (Indian Point 1, Elk River, and a partial loading of $(Th,U)O_2$ fuel in Dresden 1), and thorium-based fuels are being actively developed in the LWBR program. $(U-233/Th)O_2$ fuel is currently undergoing irradiation in the Shippingport LWBR demonstration. While fuels containing ThO_2 have performed well, their development is not as advanced as that of either conventional UO_2 or mixed UO_2 - PuO_2 fuels.

Effect of Retrofittable Once-Through Improvements on the Recycle Mode -- A number of promising options to improve the uranium utilization of the once-

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through cycle have been identified. The effect of these once-through improvements on the performance of the recycle mode of operation -- that is, whether these improvements also carry over to recycle or whether the incentive for recycle is diminished as a result of the implementation of once-through cycle improvements -- is summarized below.

The uranium utilization of the once-through fuel cycle is related directly to the fissile material (U-235) charged per unit of energy production, while in the case of self-generated recycle, uranium demand is related to the conversion ratio. Thus, once-through improvements will have differing effects on the recycle mode depending upon how the conversion ratio is altered. Those once-through improvements which reduce uranium demand by increasing the conversion ratio will result in a similar improvement in the recycle case. Those once-through improvements which do not alter the conversion ratio but obtain their benefit from more completely utilizing the fissile content of the fuel will not alter the uranium requirements in the recycle mode. In this case, the percentage benefits of recycle are reduced since the uranium requirements with recycle do not change, while the uranium requirements of the once-through mode are reduced so that they more closely approach those of recycle. Last, if once-through improvements reduce the conversion ratio, uranium requirements in the recycle mode are increased. Benefits to the once-through mode in this case are derived from the more complete utilization of fissile material, which proves to outweigh the adverse effect of reduced conversion ratio. If such once-through improvements were employed in the recycle mode, resource utilization would be degraded. Improvement options in the recycle mode that improve conversion ratio may or may not prove economically attractive, depending on the competition between carrying charges and fissile material costs.

It is sometimes possible to optimize the recycle mode for fuel utilization in precisely the opposite direction to that which proved beneficial for once-through fueling, so as to increase rather than decrease the conversion ratio. In these instances, resource utilization in the recycle mode can be improved by independent optimization. When considering such independent optimization for the recycle mode, care must be taken to ensure that the resulting designs remain technically and economically sound.

1. Extended exposure. With the extended-exposure annual-refueling option for once-through cycle in the PWR, the equilibrium cycle conversion ratio is reduced slightly (0.57 vs 0.59) relative to the reference case. A comparison between the resource utilization aspects for the standard fuel exposure case and the extended-burnup option for a PWR with self-generated recycle is provided in Table 7. Because of the decrease in conversion ratio, the extended-exposure self-generated plutonium-recycle option requires about 4% more

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U₃O₈ over 30 years than the self-generated plutonium-recycle option with 30,000 MWd/t burnup.

Uranium utilization in the closed cycle could be improved if the discharge burnup were decreased below current values, but economic considerations would limit the practicality of this approach. Reducing the discharge exposure results in increased fuel reprocessing and mixed-oxide fabrication requirements. Since both reprocessing and mixed-oxide fabrication are expensive, economic considerations generally preclude such reduced discharge exposures. An economic comparison of both the once-through cycle and the self-generated recycle options for the 30,000-MWd/t exposure and for the 50,000-MWd/t exposure is shown in Table 8. At a low U₃O₈ price (\$40/lb), the extended-exposure recycle option is economical relative to standard-exposure recycle. Savings from reduced requirements for reprocessing and mixed-oxide fabrication are larger than the cost of the additional U₃O₈ required at higher burnup. As is shown in Table 8, at a higher U₃O₈ price (\$100/lb), the higher U₃O₈ requirement of the extended-exposure option results in fuel-cycle costs essentially equal to those of the standard-exposure case.

The effect of certain other once-through improvement options on the recycle mode for PWRs is briefly reviewed below, based on the impact of the modification on conversion ratio.

2. Lattice changes. With an increase in lattice water-to-fuel ratio for PWRs, an increase in U₃O₈ requirements for recycle results from the increased fissile depletion component of the fissile loading requirement. While a wetter lattice results in a 2 to 3% reduction in U₃O₈ requirements in the once-through mode, the reduced conversion ratio results in an increase of about 3% in uranium requirements under recycle conditions.

3. Enrichment zoning (blankets). As has been indicated, the primary reason that axial blankets provide U₃O₈ savings in the once-through cycle is that the top and bottom few inches of the fuel stack in the reference case are underutilized. Replacing these pellets with lower-enriched, natural, or depleted pellets results in better U₃O₈ utilization. The reduction in axial neutron leakage is slight, since the height of the fuel column is not changed. Thus there is only a very slight effect on core conversion ratio, and the effect on recycle is small. The U₃O₈ saving provided by axial enrichment zoning for PWRs with recycle has been estimated to be about half the saving provided for the once-through cycle.

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4. Full use of initial batches from startup core. The full use of initial batches from the startup core will not lead to a reduction in uranium requirements in the closed cycle since in either event (i.e., whether these batches are reinserted for additional burnup or not), the bred and residual fissile material in these batches will be recovered and subsequently recycled. In fact, reinserting these initial batches in later cycles will delay recycle and increase their discharge exposure and therefore increase the 30-year cumulative uranium requirements of the recycle mode.

5. Improved fuel management and control design. Fuel-management schemes differing from those currently used can be employed to reduce radial leakage from the core. In such schemes, new fuel assemblies are mainly loaded into the interior of the core, while partially irradiated fuel is positioned at the periphery. Evaluations of these low-leakage fuel management schemes indicate that the benefit is derived almost solely from a reduction in core leakage. It is estimated that low-leakage configurations will reduce uranium requirements for PWRs by 1.2% in the recycle mode, as compared with the 2 to 4% savings obtained in the once-through cycle.

6. End-of-cycle coastdown. End-of-cycle coastdown has been used in the past primarily as a means of obtaining added cycle length flexibility when unforeseen factors make it desirable to do so (e.g., because of the forced outage of another unit). For the once-through cycle there is, however, both a fuel resource conservation incentive and an economic incentive to practice planned cycle coastdown. A saving in cycle energy costs can be obtained by employing this operating strategy in its economic optimum fashion.

In the recycle mode, there is a negative fuel resource conservation incentive for PWRs because of the reduced thermal efficiency of the generating plant during the coastdown operation. Less electrical energy is produced for a given consumption of fissile material. Since with recycle the residual fissile material in the spent fuel is recovered and subsequently recycled, operation in a less efficient mode results in a net loss of fissile material and therefore a higher uranium makeup requirement. It is estimated that, if end-of-cycle coastdown were employed in the recycle mode, an increase in uranium requirements of about 0.5% would occur relative to the reference recycle case. Thus, in the recycle mode, planned end-of-cycle coastdown could not be employed to improve uranium utilization but still may be attractive from an economic standpoint; coastdown may also be used effectively in the recycle mode to gain cycle length flexibility when the need arises.

As has been indicated in Table 7, self-generated recycle of uranium and plutonium into current PWRs would provide a reduction of 32% in 30-year U_{308}

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requirements (or a reduction of 38% in equilibrium cycle requirements) relative to the current once-through cycle. Self-generated recycle of uranium and plutonium into a PWR employing those once-through improvement options included in the composite improved PWR (see Table 6) would result in a reduction of 30-year cumulative U_{308} requirements of 10% (or a reduction of 16% for the equilibrium cycle) relative to the composite improved once-through case. This represents an increase of 4% in 30-year cumulative U_{308} requirements (or an increase of 5% in equilibrium cycle requirements) relative to the reference three-batch 30,000-MWd/t self-generated-recycle case shown in Table 7. If only those once-through improvement options which also provide resource benefits under recycle conditions are employed in the recycle mode (namely, axial blankets and improved fuel management), a 33% saving in 30-year cumulative U_{308} requirements (or a 40% equilibrium cycle saving) could be attained relative to the current once-through cycle. Finally, if only those once-through improvements which are expected to benefit recycle economics are employed in the recycle mode (e.g., increased burnup, wetter lattice, axial blankets, and improved fuel management), the recycle mode would suffer about a 4% increase in equilibrium cycle and 30-year U_{308} requirements relative to the reference three-batch 30,000-MWd/t recycle case.

For BWRs, the impact on U_{308} utilization under recycle conditions has not been evaluated separately for each individual once-through cycle improvement option. As for PWRs, incorporating extended burnup in BWRs is expected to lead to a resource penalty but an economic saving at current U_{308} prices. Analyses for equilibrium-cycle conditions indicated that for the current improved BWR (a BWR incorporating axial blankets and improved refueling patterns), self-generated recycle of uranium and plutonium provides a 41% reduction in equilibrium-cycle U_{308} requirements relative to the current improved once-through BWR. Analyses of a composite recycle case including increased burnup to 36,000 MWd/t, axial blankets, a three-assembly-average-enrichment initial core, improved refueling patterns, spectral-shift control, and preplanned end-of-cycle feedwater temperature reduction and coastdown shows that self-generated recycle results in a reduction of 42% in 30-year cumulative U_{308} requirements (or a reduction of 45% in equilibrium-cycle requirements) relative to the current improved once-through BWR.

The effect of nonretrofitable improvements on the recycle mode is not discussed here because a thorough evaluation of the degree to which various nonretrofitable options would be employed in an advanced LWR design and their impact on recycle has not been performed.

Long-Term Improvements for the Recycle Mode -- NASAP has sponsored exploratory investigations by ANL, B&W, and MIT of plutonium- and U-233-cycle

systems. These investigations have focused on the optimization of fuel-to-water ratio and on other long-term design changes. Results have indicated that conversion ratios of 0.8 to 0.9 are achievable in both plutonium and U-233 cycles, as compared with 0.6 characteristic of LEU cycles. The high conversion ratios are achieved in "dry" lattices with fuel-to-water ratios higher than those of the conventional PWR.

An MIT study indicates that the economically optimum fuel-to-water ratio for the uranium/plutonium cycle is essentially the standard configuration. The economically preferred configuration for the U-233-Th cycle is expected to contain about half as much water as the standard one. The economic optimum is determined by a balance of fuel makeup cost [proportional to (1 minus the conversion ratio)] and fuel-inventory cost (which increases in the drier lattices).

A B&W study indicates that conversion ratios in a denatured U-233-Th cycle can be increased to about 0.9 by use of a dry lattice, use of blankets, use of thorium for control, and other plant modifications. The trend of changes indicated is characteristic of some of the features of LWBR concepts.

1.1.3 Licensing Issues

There are currently 72 LWRs with operating licenses in the United States. Construction permits have been issued for an additional 88 units.

In spite of the large number of operating plants, LWR safety criteria are under continuing review by NRC, and plant deficiencies must be corrected as they are identified. Significant changes in licensing standards are likely as a result of the Three Mile Island accident of March 1979. Occupational exposure standards have been under recent review and are likely to be tightened.

The nature of most of the licensing issues confronting LWRs appears to be generic to fission energy and does not appear to be resolvable by adopting some alternate cycle.

Licensing issues associated with techniques to improve fuel utilization in LWRs have been discussed in Volume VI. Many of the licensing issues require R&D to develop fuel designs or fuel-management plans that not only improve fuel performance but also maintain or increase operating margins and improve

or maintain performance during postulated transients and accidents. It is evident that NRC must be cognizant of all phases of the fuel-improvement RD&D. Also, certain improvements would require changes in NRC positions, for example, extensive spectral-shift operation in BWRs. Such changes will require thorough demonstrations of safety for the desired new operating conditions.

Many of the licensing issues mentioned for once-through fuel improvements apply to the recycle systems as well. The licensing of recycle fuels will probably require more fuel-safety-related RD&D to satisfy evolving safety standards than would have been required in the past.

Once-Through Cycles

The reference LWR once-through fuel-cycle and reactor system safety characteristics and features are acceptable to NRC. Current LWR fuel is characterized and limited by known licensed-reactor technical specifications (peak kilowatts per foot, departure from nucleate boiling, minimum critical power ratio, and maximum average planar heat generation rate), which also serve as limits for once-through improvement modifications and recycle systems. The reference LWR core and plant designs in turn determine the reactor safety characteristics, namely, shutdown worth, delayed-neutron fraction, xenon oscillation stability, magnitude of reactivity insertion during transients and accidents, thermal-hydraulic performance, and offsite doses following an accident.

Once-Through Improvement Modifications -- A discussion of licensing aspects of extended-exposure fuel is provided in Volume VI and in the Preliminary Safety and Environmental Information Document for PWRs prepared by NASAP. For the higher burnups of ~45,000 to 50,000 MWd/t, use of existing fuel designs may result in unacceptably high fuel failure rates, and modified fuel designs with improved performance at high burnup will have to be developed. Some of the candidate fuel improvements that may result in improved performance characteristics are as follows:

- annular pellets,
- cladding barriers (zirconium, copper, or graphite),
- sphere-pac fuel.

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The effect of extended burnup on reactor characteristics such as shutdown worth, delayed-neutron fraction, moderator temperature coefficient, decay heat rates, and fission-product inventories are discussed in Volume VI.

Initial analyses of core physics safety parameters indicate that a PWR employing five-batch annual fuel management with fuel designed for about 50,000 MWd/t batch-average exposure has only minor changes in important parameters when compared with a PWR employing three-batch annual fuel management with 30,000-MWd/t fuel, thus indicating a high likelihood that the consequences of postulated accidents will remain acceptable. The following discusses the changes in various core physics safety parameters relative to those in a core employing the standard three-batch fuel management.

- The core may have somewhat higher power peaking. Power peaking in initial extended-burnup fuel-management studies has been approximately 5% higher than peaking for reference cases. Additional fuel-shuffle and poison-distribution studies will be required to reduce the peaks.
- Control-rod shutdown worth is reduced both at the beginning and at the end of the cycle. Reactors built to accommodate plutonium-recycle fuel should have sufficient rod worth for five-batch extended-exposure UO₂ cores.
- Delayed-neutron fraction at equilibrium for a five-batch extended-exposure core is slightly smaller than for a reference three-batch core but larger than that anticipated with plutonium recycle.
- The moderator temperature coefficient is more negative for extended-exposure cores, aggravating cold-water incidents; these more negative coefficients must be considered in establishing the required shutdown worth.
- The fission-product inventory will be larger.
- Higher enrichments may require additional licensing evaluation of fuel-fabrication and fuel-storage facilities and of fuel transportation.

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While detailed analyses of the various hypothesized accidents would be required to firmly establish the licensability of extended exposures in PWRs, the preliminary analyses performed by Combustion Engineering, Inc. establish the likelihood of licensability of the extended-exposure option. Additional assessments are under way to identify the problems and development needs related to licensability of extended fuel exposures to help guide the DOE development program.

For the various once-through cycle improvement options, an assessment of the range of events typically reported in the Safety Analysis Report (SAR) and in reload fuel applications is desirable to determine the RD&D needs prerequisite to licensing and implementation of the improvements.

Recycle Systems

Because there are small differences in the neutronic and physical characteristics of uranium/plutonium recycle fuels from those of conventional UO_2 fuels, the transient and accident behavior of the mixed-oxide cores will be somewhat different. Anticipated transient-without-scrum incidents and loss-of-turbine-load transients are reduced in severity in uranium/plutonium recycle cores because of the core's more negative Doppler, moderator temperature, and void reactivity coefficients. However, concerns arise if plutonium segregation occurs during sustained overpower operation. Cold-water-injection transients are also aggravated.

In general, transients, accidents, and their consequences for uranium/plutonium-fueled cores are similar to UO_2 -fueled cores.

NASAP studies of the safety and licensability of thorium-cycle LWRs have been carried out by B&W and GE. The conclusion is reached that the consequences of postulated accidents would remain acceptable if these fuels were employed. However, although preliminary evaluations of the characteristics of thorium-bearing cores indicate that the response during postulated accidents is satisfactory, the physical properties of thorium-bearing fuels and core properties (such as the coefficients of reactivity and control-rod worth) are somewhat different from those of UO_2 -fueled cores; consequently, a thorough reevaluation of the performance of the thorium-fueled LWR for anticipated operational occurrence and other postulated accidents will be necessary. Therefore, reanalyzing the range of events typically reported in the SAR will be necessary to demonstrate that the safety performance of the thorium-fueled LWR falls within the criteria established for uranium-fueled operation.

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Use of tight lattices, which have been proposed for advanced PWR thorium-cycle cores, will require reanalysis and demonstration of safety performance.

1.1.4 Environmental Issues

In general, LWRs have been found to be environmentally acceptable at a variety of sites within the United States, although the plant features necessary to achieve acceptability vary widely from site to site. LWRs are able to limit thermal and radiological releases to satisfy established standards. The main environmental barrier to increased use of the LWR appears to be uncertainty about disposition of spent fuel.

1.1.5 RD&D Needs

Development associated with improved fueling of LWRs may be subdivided into three broad categories: (1) once-through cycle improvements suitable for deployment into existing LWRs or into current commercial offerings which can offer an overall reduction in U_3O_8 requirements in the near term of about 15% and, with more extreme modifications, a potential reduction in the 25% range in the longer term; (2) nonretrofitable improvements, which, when combined with those in category 1, would lead to reductions in U_3O_8 requirements in excess of 30%; and (3) qualification of plutonium-or thorium-cycle fuels.

Because both PWRs and BWRs are subject to improvement, parallel programs on the two reactor types would be necessary in each of the above categories. Furthermore, each of the LWR vendors would (with government support) need to develop and offer more efficient designs. The cost of a program upgrading all of the vendors' offerings for each category of improvements would be several times as costly as a single-vendor program. The multivendor program is needed however, to achieve the full potential savings of U_3O_8 , within a reasonable time frame.

RD&D for Retrofittable Improvements to the Once-Through Cycle

To achieve widespread industry adoption of the target 15% improvement in U_3O_8 utilization with associated economic benefits by the early 1990s, both considerable RD&D and rapid acceptance of innovations by utilities are required. A certain degree of improvement can most certainly be achieved

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before 1990. Although improvements offering a combined saving of about 15% can be demonstrated by about 1990, widespread utility adoption, which would lead to industry-wide savings of 15%, will require an additional period of time.

Fuels development and demonstration needs for the high-burnup option have been considered in detail. Program costs for demonstration in commercial reactors are shown in Table 9. The initial test-rod irradiation project would consist of the engineering, fabrication, licensing, in-core irradiation, and postirradiation examination of a few test rods contained in a single or in a few bundles.

Fuel-performance results will be used to support design and licensing of the later, more extensive lead test assembly and batch irradiation projects.

As has been discussed in Section 1.1.2, DOE programs are under way to demonstrate the performance of current and slightly modified fuel designs to extended burnups in several commercial reactors in joint utility/vendor efforts. Additionally, EPRI is focusing its fuel test and surveillance program on high-burnup irradiations to obtain information required to support licensing of higher exposures. DOE is also sponsoring pellet-cladding interaction (PCI) remedy programs with the objective of demonstrating advanced fuel designs that are resistant to this failure mechanism both at current burnup levels and at higher burnup levels. These projects include both steady-state irradiations and ramp testing of the candidate PCI remedy fuels.

Data needs for design, safety analysis, licensing, and operation of high-burnup fuels include fission-gas release measurements, failure and damage thresholds associated with pellet-fuel-cladding interaction, assembly dimensional stability such as rod spacing, pellet dimensional stability, and Zircaloy corrosion. Power Burst Facility testing of high-burnup rods may be required to confirm acceptable performance of extended-burnup fuels under anticipated transient conditions.

Concerning fission-product release from failed fuel, characterization of failures and measurement of release of fission products from ramp tests of rodlets in a test reactor may be required to confirm that fission-product release from failed fuel is not increased significantly at higher burnups. DOE has a continuing effort to identify data needs to support licensing of high-burnup fuels, and data development programs are expected to be expanded as required to support extended-exposure development and demonstration.

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Table 9. LWR Extended-Exposure Commercial Reactor Demonstration RD&D Program Description and Cost

RD&D area	Cost (\$ million) ^a
R&D program	
Test-rod project	4-5
Engineering	
Material and fabrication	
Licensing	
In-reactor irradiation	
Postirradiation examinations	
Lead test assembly project	10-15
Engineering	
Materials and fabrication	
Licensing	
In-reactor irradiation	
Postirradiation examination	
Demonstration	
Batch implementation project	6-15
Licensing	
Engineering	
Materials and fabrication	
In-reactor irradiation	
Postirradiation examination	

^a January 1978 \$.

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Figure 6 presents a preliminary schedule for development of uranium utilization improvements, including high-burnup fuel. The extension of fuel exposure technology will be a continuous effort. The development and demonstration will proceed through the extension of present designs to extended exposures, while development and irradiation of advanced fuel designs capable of higher burnup proceeds. Batch reload demonstrations of extended-exposure capability of fuel of essentially current design will proceed through the middle 1980s, while demonstrations of advanced designs will proceed through the late 1980s and early 1990s. Burnups in the range of 36,000 to 45,000 MWd/t will be demonstrated during the middle 1980s, while batch burnups in the range of 45,000 to 55,000 MWd/t will be demonstrated during the early 1990s.

Other improvements amenable to retrofitting which result in reduced U_3O_8 requirements could be demonstrated in commercial reactors by the early to middle 1990s, as shown in Figure 6. Some can be demonstrated much earlier than others. These improvements cover a range from relatively straightforward options, some of which are currently being implemented, to more extreme options which require further assessment of technical and economic feasibility based on a thorough evaluation of potential technical difficulties. For PWRs, this range of options includes lattice optimization, axial blankets, improved fuel management and control techniques, preplanned coastdown, full use of early batches, and the variable-lattice fuel assembly. For BWRs, these include multienrichment-zone initial cores, axial blankets, improved fuel management and control, preplanned coastdown, full use of early batches, reconstitution of high-enrichment fuel, the variable-lattice option, and spectral shift through void control.

Preplanned coastdown is expected to have no major development requirements other than development of improved methods for optimization of the coastdown interval through consideration of utility power system planning. Full use of early batches, axial blankets (and radial blankets for BWRs), and improved fuel management is next in terms of requiring minimal RD&D. Nuclear measurements are needed to confirm predictions of power distributions, reactivity coefficients, and other core physics parameters for these options.

Further development and demonstration of fuels containing burnable poisons (e.g., gadolinia) is required in support of more efficient low-leakage fuel-management and control schemes. This includes measurements of power and burnup distributions on lead test assemblies to confirm that behavior is as predicted.

At least one PWR vendor is pursuing generic NRC approval in advance of commercial use of a slightly modified fuel assembly employing a somewhat wetter

Figure 6. Schedule for Development of Uranium Utilization Improvements.

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lattice, somewhat increased burnup, and more neutron-efficient structural material. A first core loading employing the improved fuel assembly is scheduled for operation by 1982. The compatibility of fuel assemblies employing an optimized lattice would have to be demonstrated for plants that would begin to employ such assemblies after initial operation with standard assemblies.

While a thorough evaluation of technical and economic aspects has not been performed for more extreme retrofittable options, such as variable-lattice fuel assembly designs in PWRs and BWRs, BWR bundle reconstitution (and inversion), and BWR spectral shift through void control, development requirements are potentially more extensive.

For the variable-lattice option, the initial number of rods in the fuel-assembly lattice would need to be increased for departure-from-nucleate-boiling concerns. Also, considerable plant modifications would be required, mainly in the refueling equipment. Fuel-bundle reconstitution, which is necessary for this concept, requires unique quality-control procedures to assure reliable application because of the burnup and isotopic distributions between the individual fuel rods. Furthermore, fuel-assembly designs would need to be of the removable-rod type, since removal of fuel rods from an assembly not specifically designed to allow for this possibility would create severe practical problems. It is estimated that the concept could be implemented in seven to ten years.

Reconstitution (and inversion) of high-enrichment BWR rods also would require unique quality-control procedures to assure reliable application. Although BWR fuel assemblies can be manually disassembled and reassembled with differing fuel-rod configurations, as has been demonstrated many times in the past, large-quantity processing may require more automated procedures. Design with additional water rods in the reassembled bundles required by reactivity and power distribution considerations may require critical heat flux and loss-of-coolant-accident-related measurements on full-scale fuel assemblies. Nuclear and fuels performance measurements will be needed to confirm the predicted characteristics of reassembled fuel.

Application of BWR spectral shift by coolant density control would require modifications in current NRC licensing criteria. Achievement of the full potential would require equipment backfits to increase the full-power flow-control capacity and range relative to current BWRs. Spectral shift could be applied to some extent in most current BWRs with little or no component modification.

RD&D for Nonretrofittable Improvements for Once-Through Operation

The nonretrofittable improvements will probably evolve in the form of options to modify designs of the current standard LWR plants. These options include plant designs for rapid refueling, movable fuel or fertile control, wide-range spectral shift through void control in BWRs, extensive use of blankets, and improved reflectors. Reactor and plant designs employing these options have not been studied in detail; therefore, the RD&D requirements are quite uncertain.

A thorough analysis and development of a conceptual design for a composite case incorporating options not retrofittable into current plants as well as options considered retrofittable are required to develop RD&D requirements. However, it is anticipated that major design changes would be required to achieve uranium savings in excess of 30% relative to the reference current design while maintaining a licensable and commercially viable design. Both NSSSs and balance of plant would be affected.

For rapid-refueling schemes, a preliminary evaluation of RD&D costs and time requirements has been made. Two cases were considered. The first (standard) rapid-refueling scheme would involve a cold-shutdown refueling and would require from 12 to 14 days. This standard scheme is incorporated in a Westinghouse plant currently under construction for Houston Lighting and Power and could be demonstrated in about five years. In order to achieve the more commercially acceptable refueling times for six-month cycles of one week or less, the development of a more advanced scheme would be required. Both concepts require significant plant modifications, including changes in pressure-vessels, pressure-vessel closures, and reactor internals and neither are backfittable into existing plants. Currently, utilities have not shown much enthusiasm for the rapid-refueling concepts, mainly because the core loading is not in the critical path to start-up. The RD&D cost for the advanced scheme is estimated to be in excess of \$100 million. The time required before the commercial offering of a new plant employing this scheme is in the range of 15 to 20 years.

A lower-power-density core using longer fuel assemblies and/or an increased number of fuel assemblies would operate with increased margins to safety limits and would provide U_{308} savings while requiring no new technology developments. The low-power-density core is not retrofittable and would require changes in plant design, such as larger pressure vessel and containment structure, modified internals, changes in new- and spent-fuel storage racks, and perhaps changes in fuel-handling equipment. RD&D costs are first-

time engineering costs, since there are relatively few, if any, feasibility questions to be answered. The RD&D program cost per vendor and time required to develop the design so that it can be offered in the marketplace are approximately \$6 million and four to five years.

Current-design PWRs use a stainless steel baffle (or shroud) around the core periphery. Uranium savings of 1 to 2% are possible if Zircaloy were to replace the stainless steel in the baffle. The RD&D program would need to determine the feasibility and potential problems associated with the behavior of Zircaloy under high, long-term irradiation. The available high-burnup experience for Zircaloy-clad fuel can serve as a useful base for further data acquisition.

RD&D for other options classified as nonretrofitable has not been assessed to the detail presented above. The cost of the development program for advanced PWR and BWR designs employing composites of improvement options would be several times the cost of the program to develop retrofitable options, and successful commercial use would not be likely before 2000. A very preliminary estimate for development of nonretrofitable options for a four-vendor program, assuming R&D coordination by the government, is presented in Table 10.

Reactor-Related RD&D for Recycle Fuels

RD&D for uranium/plutonium fuels has been carried into the lead test assembly demonstration phase. The remaining demonstration costs for a full batch of $\text{UO}_2\text{-PuO}_2$ fuel are expected to be about \$15 to \$70 million, as shown in Table 11. The uranium/plutonium fuel would be used in reactors of existing designs.

Estimated RD&D costs for plutonium/thorium fuels and for denatured uranium/thorium fuels are summarized in Table 10 and presented in more detail in Tables 12 and 13, respectively. The RD&D costs have been estimated under the assumption that these fuels will be employed in LWRs of essentially present design. This assumption is not meant to exclude minor changes to reactor design (e.g., changes in the number of control drives, shim loadings, or fuel management) but rather reflects the current belief that design changes necessitated by these fuels will be sufficiently straightforward to be accommodated within the engineering design typically performed for new plants.

The RD&D schedule for recycle fuels is given in Figure 7.

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Table 10. LWR RD&D Cost Summary
(In millions of January 1978 dollars)

	Per vendor cost		
	PWR	BWR	
<hr/>			
Retrofittable once-through cycle improvements			
Improved fuel-bundle configurations	15-25	15-25	
High-burnup fuel development	20-35	20-35	
Improved fuel management	4-7	4-7	
Changes in plant operations	1-3	16-28	
Total	40-70	55-95	
Estimated cost for five fuel vendors carried as part of a national program		150-300 ^a	
Nonretrofittable additional once-through improvements (four reactor vendors)		600-1000 ^b	
<hr/>			
Reactor costs for recycle fuel qualification (excluding costs of fuel-cycle facilities)			
	Plutonium/ uranium	Plutonium/ thorium	Denatured uranium/ thorium
	15-70	100-300	100-300
Estimated cost for five vendors	45-200	300-750	300-750

^aThe cost will probably be in the upper portion of this range.

^bAssumes no dedicated demonstration plant required.

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Table 11. R&D Required to Deploy Plutonium/Uranium Fuel in LWRs

RD&D area	Cost (\$ million) ^a
Additional lead test assembly irradiation	3-7
Licensing	
Fabrication	
Irradiation and postirradiation examination (PIE)	
Possible additional fuel performance testing	5-30
In-reactor properties experiments	
Power-ramp experiments	
Transient tests	
Licensing for full batch	1-5
Safety analyses of modified core	
Prepare preliminary SAR and license	
Demonstration of full batch in LWR	5-25
Engineering	
Fabrication	
Irradiation and PIE	
Total	15-70

^aJanuary 1978 \$.

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Table 12. RD&D Required to Deploy Plutonium/Thorium in LWRs

Research and development	Cost per vendor (\$ million) ^a
Data-base development	
Physics verification program	7-9
Improve cross sections for thorium, U-233, plutonium, etc.	
Measure resonance integrals for thorium fuels at room temperature and at elevated temperatures	
Perform and analyze critical experiments	
Fuel-performance program	30-150
Perform in-reactor properties experiments	
Perform power-ramp experiments	
Perform fuel-rod irradiation experiments	
Perform transient tests	
Demonstration design and licensing	30-60
Develop core design changes as required for plutonium/thorium fuels (e.g., control materials)	
Perform safety analysis of modified core	
Prepare SAR, carry through licensing	
Demonstration of LWR operating on plutonium/thorium fuel	40-70
Total	100-300

^aJanuary 1978 \$.

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Table 13. RD&D Required to Convert LWRs to Denatured Uranium/Thorium Fuel Cycles (20% U-235/U-238-Th or 12% U-233/U-238-Th)

Research and development	Cost per vendor (\$ million) ^a
Data-base development	
Physics verification program	10
Improve cross sections for thorium, U-233, plutonium, etc.	
Measure resonance integrals for denatured thorium fuels at room temperatures and at elevated temperatures	
Perform and analyze critical experiments	
Fuel-performance program	
Perform in-reactor properties experiments	30-150
Perform power-ramp experiments	
Perform fuel-rod irradiation experiments	
Perform transient tests	
Reactor components development (develop handling equipment/procedures for radioactive U-233 containing fresh fuel elements)	5-25
Demonstration design and licensing	20-50
Develop core design changes as required for uranium/thorium fuels (e.g., control materials)	
Perform safety analysis of modified core	
Prepare SAR, carry through licensing	
Demonstration of LWR operating on denatured fuel (probably U-235/U-238-Th)	30-70
Total	100-300

^a January 1978 \$.

DRAFT

I. (Pu, U)O₂ FUEL DEMONSTRATION

ADDITIONAL LEAD
TCOT
ASSEMBLY
IRRADIATION

ENGINEERING,
LICENSING
AND
FABRICATION

IRRADIATION

EXAM

BATCH IMPL-
EMENTATION
PROJECT

ENGINEERING,
LICENSING
AND
FABRICATION

IRRADIATION

EXAM

II. DENATURED UO₂/ThO₂ OR PuO₂-ThO₂ FUEL DEMONSTRATION

TEST ROD PROJECT

ENGINEERING,
LICENSING
AND
FABRICATION

IRRADIATION, EXAM

LEAD TEST
ASSEMBLY PROJECT

FABRICATION
CRITICAL
EXPERIMENTS

ENGINEERING,
LICENSING
AND
FABRICATION

IRRADIATION

EXAM

BATCH IMPL-
EMENTATION
PROJECT

ENGINEERING,
LICENSING
AND
FABRICATION

IRRADIATION

EXAM

Figure 7. Time Line for Implementing Recycle Fuel Options in a PWR.

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Generically, the first phase of such fuel-type-related research consists of the development of a data base from which safety-related parameters and fuel performance can be predicted in subsequent core physics design and safety analysis programs. First, existing thorium materials and fuel-performance information should be thoroughly reviewed, and a preliminary evaluation of safety and licensing issues should be made to identify missing information and to guide the subsequent development program.

The fuel-performance program under LWR data-base development would consist of the establishment of safety-related fuel-performance information for both Pu/Th and denatured U/Th fuels. The initial phase of this program should consist of in-reactor properties experiments, power-ramp tests, transient fuel damage tests, and fuel-rod irradiations. The in-reactor properties experiments would provide information on such properties as centerline temperature, swelling and densification, and fission-gas release. Power-ramp experiments would consist of preirradiation of the fuel-rod segments in existing LWRs and the subsequent power ramping of these segments in special test reactors to establish anticipated fuel performance during power changes typically encountered in the operation of LWRs. The transient fuel damage experiments would be designed to provide information on the performance of the fuels under the more rapid transients possible during operation and in postulated accidents. Finally, the fuel-rod irradiation experiments would provide information on the irradiation performance of prototypical thorium-based fuel rods and, with subsequent postirradiation isotopic analyses, would also provide information on burnup and isotopic content. In addition to the data-base development, some unidentified reactor components development could be expected. To cover this aspect of the program, an estimated cost of \$5 to \$25 million is included. The remaining R&D for these fuels in LWRs would be devoted to developing safety analysis information in preparation for demonstration in a commercial reactor.

1.1.6 Perceived Advantages and Disadvantages

The advantages of the LWR over other nuclear options are that (1) the LWR is available from a large number of commercial sources, (2) there is a large body of commercial experience with gradual improvements in its performance and reliability, (3) there is a large body of successful experience in safeguarding the once-through cycle against proliferation, and (4) the LWR constitutes an economic power-generating system.

The technical disadvantages are primarily the higher U_3O_8 consumption relative to more advanced reactor concepts and the associated environmental impact of mining requirements.

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The U_3O_8 requirement of LWRs can be reduced in an evolutionary manner through once-through fuel-cycle improvements and, if permitted, through recycle. The economic and commercial advantages of the LWR indicate that it will probably be the dominant nuclear system through 2000 and possibly well into the 21st century.

1.2 HEAVY-WATER REACTORS

1.2.1 Reactor/Fuel-Cycle System Description

The conceptual design of the heavy-water reactor (HWR) developed in the United States is based on a modification of the CANDU reactor developed in Canada. It is a heavy-water-moderated, pressurized heavy-water-cooled horizontal pressure tube reactor fueled with slightly enriched uranium dioxide. Because of the low neutron-absorption cross section of deuterium, reactors utilizing heavy water as the moderator theoretically can attain higher conversion ratios than reactors using other moderators.

The HWR design differs from the LWR design primarily in three areas: (1) the HWR fuel is of lower fissile enrichment, (2) the coolant and moderator are separated by a pressure tube, and (3) the fuel management scheme employs continuous on-line rather than periodic refueling. The on-line refueling scheme is utilized to minimize the core average burnup and the need for control poisons, thus improving neutron economy.

In the conceptual United States HWR design, two primary-coolant circulation loops are provided, each serving one-half the reactor core. This main heat transport system circulates pressurized heavy water through the reactor fuel channels to remove heat produced by the fission process. The heat is carried to the steam generators, where it is transferred to light water to form steam, which drives the turbine generator. Each primary-coolant loop consists of two steam generators and two primary-coolant pumps located at the ends of the reactor, so that flow is in one direction through half the fuel channels and in the opposite direction through the other half. In addition to the main heat transport system, a separate system is provided to maintain the temperature of the heavy-water moderator. Because of the low temperature of the moderator, heat extracted by the moderator cooling system is not used to generate electricity but rather is dissipated directly by the heat rejection system. A simplified flow diagram with only one of the primary coolant loops shown is given in Figure 8.

Some of the major differences between the United States concept and Canadian HWR designs are the following:

- Higher capacity. The U.S. capacity rating is 1260 MWe vs. 750 MWe for the largest current CANDU.

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- Modified primary system. Pressure has been increased to 2200 psia, and larger pumps and steam generators are used. These changes improve thermal efficiency and permit use of U.S. component designs. The main objective is reduced cost.
- Modifications in design to accommodate U.S. licensing criteria.
- Fuel. The reference cycle is based on 1.2% enriched UO_2 fuel instead of natural uranium for the CANDU.

There are several alternative fueling modes that can be applied to the HWR, including the following.

Low-Enriched Uranium Once-Through

This reactor/fuel-cycle combination utilizes 1.2% slightly enriched uranium oxide pellet fuel optimized for a once-through cycle. The lifetime U_3O_8 and enrichment requirements* are respectively 59% and 25% of the requirements for an equivalent-sized PWR.

Natural Uranium Once-Through

This reactor/fuel-cycle combination utilizes natural uranium fuel and is the cycle currently used in Canadian reactors. While the fuel costs of this cycle are a little higher than those of the LEU cycle, a number of countries prefer to avoid dependence on enriched fuel. There are no SWU requirements, but the lifetime U_3O_8 requirements of the U.S. HWR concept are 93% of those for an equivalent-sized PWR.

Medium-Enriched Uranium-Thorium Cycle

This is the denatured U-Th cycle that requires reprocessing but may discard the plutonium with high-level wastes. Makeup fuel is enriched to less than

*The measure of effort required to accomplish isotopic separation (enrichment) is defined in terms of separative work units (SWU).

Figure 8. Simplified Flow Diagram of the Heavy-Water Reactor.

20% U-235 or enriched to less than 12% U-233 from an external source. Of the denatured fuel cycles in thermal reactors, the HWR cycle has the best resource utilization and produces the least plutonium, about 15% of the rate of the standard LWR-LEU cycle.

Plutonium-Thorium, Plutonium Recycle

This cycle is used primarily as a transmuter cycle to produce U-233 for a denatured U-Th cycle.

The introduction of the more complex but more efficient fueling modes would be based on the availability of U₃O₈ and its cost. The denatured U-Th cycle would have U₃O₈ requirements about 25% of the standard LWR-LEU cycle or 40% of the HWR-LEU once-through cycle.

The necessary development and testing programs for these fueling modes may be expected to follow the companion fuel cycle for an LWR.

1.2.2 Status of Technology

Current HWR technology is most generally represented by the CANDU pressurized-heavy-water reactor. Reactors of this type with electrical generating capacities of between 500 and 750 MWe are in operation in Canada and in operation or under construction in a number of other nations, including India, Pakistan, Argentina, and Korea. Figures 9 and 10 are two views of the CANDU reactor.

Extensive operating experience has been obtained for CANDU reactors, particularly for the Pickering station, which has four 500-MWe plants. The first Pickering unit began operation in mid-1971, and a fourth unit was started up in mid-1973. Performance of the Pickering station has generally been excellent; its lifetime average capacity factor is 75%, significantly higher than that reported for the 66 LWRs operating in the U.S. Thus, it can be concluded that the technical feasibility of HWRs operating with natural uranium fuel has been demonstrated. Although the operation of HWRs on the natural uranium cycle has been quite satisfactory, as mentioned above, the design described here employs a limited number of modifications. The use of slightly enriched uranium fuel (~1.2 wt %) will increase discharge burnup to

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Figure 9. Heavy-Water Reactor Assembly.

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Figure 10. General Arrangement of Safety-Related Structures in an HWR.

~20,000 MWd/t, compared with 7500 MWd/t for the current Canadian designs. Although major improvements in current fuel technology are not considered necessary to achieve these higher burnups, development testing for mechanical performance acceptability will be required, particularly to accommodate higher primary system temperatures and pressures.

Table 14 contains general HWR performance specifications.

1.2.3 Licensing Issues

The Canadian experience with construction and operation of CANDU reactors provides a favorable outlook on plant safety. More specific findings would require a licensability review of a proposed design by the NRC staff. Many NRC criteria for licensing defined in 10 CFR 50, Appendix A, are specifically worded for LWRs, and the intent of the majority of these criteria would be met by a CANDU-PHWR. Those criteria that would likely be met but that might require ASME Boiler and Pressure Vessel Code case extensions or modifications are related to the fracture prevention of the reactor-coolant boundary and its inspection. Those criteria that might require a design change for a plant sited in the United States are:

- design bases for protection against natural phenomena (10 CFR 50 Appendix A #2),
- containment design (#16),
- electric power systems (#17),
- control room (#19),
- reactor coolant makeup (#33),
- residual heat removal (#34),
- emergency core cooling (#35),

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Table 14. HWR Performance Specifications

<u>Power plant performance</u>		
Thermal power, MW		
Total nuclear	4,029	
Coolant	3,800	
Electrical power, MWe		
Gross	1,343	
Net	1,260	
Thermal efficiency, %	31.5	
<u>Reactor parameters</u>		
Core volume, liters	3,221,000	
Core dimensions, m		
Equivalent diameters	8.8	
Active length	5.9	
Core power density, MW/liter	0.012	
Coolant flow rate, Mg/sec	14.2	
Coolant inlet temperature, °C	299	
/ Coolant outlet temperature, °C	337	
Primary system pressure, MPa (psia)	15.2 (2,200)	
<u>Fuel parameters</u>		
Average fuel temperature, °C	936	
Maximum fuel temperature, °C	1,900	
Cladding temperature, °C	350	
Core fuel loading, kg		
Total heavy metal	166,056	
Fissile material, initial	1,181	
Discharge exposure, MWd/t		
Average	19,750	
Peak (estimated)	25,000	
Average conversion ratio during equilibrium cycle	0.72	

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- containment heat removal (#38),
- containment atmosphere cleanup (#41),
- cooling water (#44),
- containment design basis (#50),
- fracture prevention of containment pressure boundary (#51),
- systems penetrating containment (#59),
- primary containment isolation (#56),
- closed system isolation valves (#57),
- fuel storage and handling and radioactivity control (#61).

The list above is not intended to imply that these aspects have not been fully and satisfactorily considered in the CANDU design; they are, however, indications of areas where the CANDU design is believed to differ from present LWR practice in the U.S. and, therefore, indications of areas needing early attention by the NRC staff.

Other items that might require substantial licensing review are:

- the behavior of the system, including the on-line refueling system, during accidents or natural disasters,
- the use of a coolant with a positive void coefficient,
- an overall power coefficient of reactivity at power near zero that may either be slightly positive or slightly negative,

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- the presence or instability of xenon oscillations if no external control is applied,
- the use of a nonferrous material (Zr-Nb alloy) exposed to the full neutron flux as part of the pressure boundary of the primary heat transport system,
- the use of rolled joints to connect Zr-Nb alloy pressure tubes to the stainless steel end fittings,
- the material and performance characteristics of the thin-walled (relative to the PWR) pressure tubes at the elevated temperatures and pressures.

There are features of the CANDU design that tend to be positive in the safety sense, including the location of control and safety absorbers in the low-pressure moderator, the provision of separate channels (pressure tubes) for each fueled channel (making the introduction of emergency cooling water potentially more predictable), the large cool moderator surrounding the fuel, which acts as a heat sink in the event of a failure of the emergency core-cooling system (ECCS), and the limited installed reactivity of the system. All factors need to be considered in a determination of safety adequacy.

1.2.4 Safety Issues

During normal operation, the main reactivity control is the use of on-power refueling. (For the initial core, soluble boron moderator poison control is also employed until the excess reactivity is depleted.) With on-line refueling, the reactor operates with essentially no excess reactivity (except that contained in the normally inserted adjuster units). Fresh-fuel bundles are charged into the fuel channel by a remotely operated fueling machine, while spent-fuel bundles are simultaneously discharged into another refueling machine at the opposite end of the reactor. The continual charging and discharging of fuel minimizes reactivity-control system requirements to compensate for fuel depletion. An H₂O zonal control system and vertical absorber rods are provided to control the small reactivity changes that occur between refuelings and to adjust the local power distribution within the reactor core. Vertical adjuster rods, control rods containing a graded vertical loading of cobalt poison, are also provided for xenon power override.

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These adjuster rods are normally fully inserted during operation and are withdrawn after shutdown or operation at low power to increase reactor power by providing the reactivity to override the buildup of xenon.

Two diverse reactor shutdown systems are provided. The first system consists of control rods that are released when tripped and fall into the core by gravity, and the second injects gadolinium poison into the moderator. In-core instrumentation is provided to allow automatic control of reactor power and flux shape and to monitor local core behavior. Central to the instrumentation and control system are large-capacity digital computers that are employed for station control, alarm annunciation, and data display. The computer system (1) coordinates reactor power level with turbine demand, (2) manipulates the various reactivity control devices to maintain a desired flux shape, (3) initiates slow and fast power cutbacks to keep the plant parameters within limits to avoid a reactor trip, and (4) controls other plant parameters, such as steam generator and pressurizer pressures and water levels.

1.2.5 Environmental Issues

There appear to be no fundamentally unique environmental problems or issues associated with the HWR concept. However, a number of the attributes of the HWR system are different enough from those of the LWR to quantitatively change the environmental impact. These aspects are as follows.

1. The standard CANDU operates at a primary system pressure significantly lower than that of the PWR. As a result, the net station efficiency is lower, 29 to 30% compared with 33 to 34% for the PWR, and, consequently, about 20% more heat will be rejected to the environment. However, HWRs for U.S. deployment could be designed to employ higher primary system pressures in order to reduce the capital cost of the plant; a conceptual design of this type has been prepared as part of NASAP. As a result, efficiencies of about 31% might be achieved, resulting in only a 10% difference in heat rejection. For equal primary pressures, the HWR will always operate at a slightly lower efficiency and, hence, at a higher heat rejection rate because of heat lost to the moderator; the energy deposited in the moderator must be rejected to the environment because the moderator temperature is too low for electrical power generation.

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2. The natural-uranium-fueled HWR has a comparatively low burden of trans-uranic elements because of the low burnup of this type of fuel. Consequently, the inventory of long-lived radionuclides available for release during postulated accidents is lower than for the LWR. In a slightly enriched uranium fuel cycle where the fuel burnup would be much higher, trans-uranic inventories should be roughly comparable with those of the LWR. The environmental implications of uranium/plutonium and thorium-containing fuels would be expected to be similar to those described for LWRs.

3. Because of the pressure tube design and on-line refueling feature of the HWR, failed fuel can easily be detected and removed from the reactor; thus, the concentration of radionuclides in the primary coolant can be minimized, and the potential for the release to the environment via this pathway is reduced relative to the LWR.

4. Because the HWR is cooled and moderated with heavy water, production of radioactive tritium is much higher than in the LWR. Careful attention is paid in the design of HWRs to minimizing D₂O/tritium losses, because of the radiological impact and the cost of D₂O. Nevertheless, some tritium will reach the environment during normal operation, and considerable inventories of tritium are present for release during accidents.

The main unresolved environmental issue in the U.S., in addition to those such as waste disposal, which are also common to the LWR, is the establishment of appropriate operational limits for tritium release. (Assuming that HWR operation will not be constrained by criteria for tritium release, no environmental R&D should be required, nor should environmental licensing considerations constrain commercial introduction dates.) However, if it proves necessary to reduce emissions because of environmental considerations, R&D would be required to develop practical methods of separating tritium from heavy water.

5. Because the HWR does not have a thick-walled pressure vessel, its decommissioning is expected to be easier than for the LWR. Removal (and replacement) of the pressure tubes has been shown to be relatively easy. Furthermore, the concrete vault in which the calandria is placed does not become strongly radioactive because of the presence of a large light-water shield.

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1.2.6 RD&D Needs

Although one objective has been to minimize the changes in existing CANDU designs so as to enhance the prospects for early deployment and to minimize R&D requirements, several modifications have been incorporated in the HWR reference design to obtain improved uranium resource utilization in the once-through fuel cycle. Foremost of these modifications is the use of slightly enriched uranium fuel, rather than the natural uranium fuel currently employed in CANDU reactors. The conceptual design also utilizes an increased primary system pressure and secondary system pressure to improve system efficiency and reduce the effective capital cost of the plant.

The use of slightly enriched uranium fuel (~1.2%) will increase burnup to ~20,000 MWd/t of uranium, as compared with ~8,000 MWd/t for the current Canadian designs. Although this burnup is achievable from a reactivity viewpoint, development testing for mechanical performance acceptability will be required. Major improvements in current technology are not considered necessary in this area.

Modifications to system structures and components as required to meet U.S. codes and standards are considered within the state of the art and will not require major RD&D or proof testing.

The following areas have been identified as requiring development effort to support the HWR design:

1. Safety-related physics parameters. Critical experiments designed to measure such parameters as core reactivity, power distribution, and control absorber worth should be performed for slightly enriched uranium fuel.

2. Fuel testing. Demonstration fuel assemblies containing an initial enrichment sufficient to achieve the discharge burnups anticipated for the slightly enriched HWR should be irradiated in test reactors. These experiments would provide a measure of the power peaking anticipated in the slightly enriched uranium fuel cycle and demonstrate the performance of such fuel up to the anticipated burnups and for power changes that occur during refueling.

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3. Advanced pressure tube design. The reference NASAP HWR is expected to operate at pressures comparable with those of the PWR to increase net station efficiency and reduce the effective capital cost of the HWR. An experimental program to demonstrate the integrity of the pressure tubes and rolled joints at these higher operating pressures will have to be performed. This program would be similar in nature to tests already performed on CANDU pressure tubes and would be intended to extend the range of validity for experimental information to the higher pressures anticipated for the reference design.

4. Thermal-hydraulic tests. The empirical correlations used to establish the margin to burnout in CANDU reactors cover the range of pressures currently employed. This information will also have to be extended to the higher pressures anticipated in the reference NASAP design.

5. Advanced D₂O production methods. The economics of the HWR depend, to a good degree, on the cost of D₂O. Consequently, there is a strong incentive to develop less expensive methods for producing D₂O. Such advanced D₂O production technologies should probably be considered as ongoing programs; a reasonable target might be to demonstrate such advanced technology within about five years, so that an advanced production facility might be available in a decade or so.

6. Experimental facilities. It is assumed that a cooperative program between the United States and Canada would be required to achieve a reasonable introduction date of the HWR in the U.S. Utilization of existing facilities of Atomic Energy of Canada, Ltd. (AECL) would greatly reduce or eliminate investment in new facilities or modifications to existing facilities in the U.S.

Demonstration or prototype plants, as described, are not considered necessary for the HWR design. This is based on the current status of technology represented by the CANDU.

Reactors of this type having electrical generating capacities of up to 750 MWe are currently in operation in Canada and other countries. It is anticipated that successful performance of any modifications to the CANDU design could be demonstrated through component testing programs, thus eliminating reliance on demonstration or prototype plant facilities.

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It is assumed that the first HWR constructed in the U.S. would be a lead commercial unit, operated in a utility environment with full-scale components of economic size and capable of replication. If the HWR concept is to have an impact on uranium fuel utilization, a significant commitment by utilities will be required in advance of the initial operation of a lead commercial unit. The design, construction, and licensing span for these units would be comparable with those of U.S. LWRs.

The status of technology for the HWR is such that a well-established technological base does exist for the HWR, and relatively low RD&D requirements can be anticipated.

Table 15 presents cost estimates, and Table 16 shows the expected time schedule from the current status to the lead U.S. commercial plant.

1.2.7 Perceived Advantages and Disadvantages

From the resource point of view, the HWR has very favorable characteristics, in both once-through and recycle modes. It has already been commercialized in Canada and other countries and has a proven technology base. Its usefulness with natural uranium fuel is an important consideration for some countries but not for the U.S., where an LEU cycle would be preferred. Disadvantages result from a capital cost higher than that of LWRs. This cost is related to the high cost of D₂O, lack of a commercial design licensed in the U.S., lower thermal efficiency, and problems related to control of tritium releases to the environment.

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Table 15. HWR Once-Through RD&D Costs

RD&D area	Cost ^a (\$ million)
Reactor plant R&D	200-400
Plant demonstration	
Proof-of-principle plant	N.R. ^b
Demonstration plant	N.R.
Lead plant ^c	1400-1600
NSSS vendor support	100-150
Increment for multiple vendors	100-200
Total reactor RD&D	1800-2350

^a January 1978 \$.

^b N.R. = not required.

^c Does not include cost of heavy water, which is estimated to be \$200-300 million, nor the cost of the initial fuel, which is estimated to be \$40-50 million. (See Volume IV.)

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Table 16. HWR Once-Through RD&D Schedule

HWR-OT RD&D activity area	Time frame (years)
Reactor plant R&D	1979-1985
Plant demonstration	
Proof-of-principle plant	N.R. ^a
Start design	
Construction permit	
Operating license	
Demonstration plant	N.R.
Start design	
Construction permit	
Operating license	
Lead plant	
Start design	1980
Construction permit	1985
Operating license	1993
First commercial plant	
Start design	1995
Construction permit	1999
Operating license	2005

^a N.R. = not required.

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1.3 WATER-COOLED BREEDER REACTORS

1.3.1 Reactor/Fuel-Cycle System Description

The light-water breeder reactor (LWBR) is one of a family of water-cooled reactors, similar to a PWR, that are designed to operate at an equilibrium conversion ratio of 1.0 or higher. The general family of water-cooled breeder reactors (WCBRs) includes heavy-water breeder reactors in addition to the LWBRs. Seven alternative concepts for WCBRs have been considered, as follows:

(1) light-water backfit prebreeder/seed-blanket breeder system, (2) light-water seed-blanket prebreeder/breeder system, (3) light-water seed-blanket prebreeder/heavy-water seed-blanket breeder system, (4) heavy-water seed-blanket breeder system, (5) light-water backfit prebreeder supplying advanced breeder, (6) light-water backfit prebreeder using 20 wt % enriched uranium supplying light-water backfit high-gain converter, and (7) prebreeder and breeder based on LWBR Type I modules.

To achieve the goal of a high conversion ratio, the following design modifications from the reference PWR cycle are necessary.

1. The fuel cycle is based on highly enriched U-233-Th (HEU3-Th) fuel instead of LEU. Since U-233 is not found in nature, the WCBR requires closing of the fuel cycle through reprocessing and recycling of fuel. The thorium serves to capture neutrons and produce U-233.

2. The reactor is controlled by movable fuel or fertile rods and safety poison shutdown rods rather than entirely by poisons.

3. In unirradiated fuel assemblies, the U-233 content of some fuel rods is higher than that of other fuel rods; that is, the WCBR would be a seed-and-blanket design.

4. The metal/water volume ratio in the core is increased to the range of 1.5 to 2.5, vs. about 0.8 in a PWR. The reduction of moderator fraction reduces parasitic neutron capture in water.

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5. The specific power (kilowatts of electricity per kilogram of heavy metal) is about one-third to one-half of that of a conventional PWR. This reduces neutron capture in certain short-half-life isotopes (Pa-233, Xe-135).
6. The fuel element burnup averages 10,000 to 15,000 MWd/t vs. 30,000 MWd/t for a typical PWR. This reduces neutron capture in fission products.
7. The power density (kW/liter) would be about one-half of that of a PWR. This would reduce the fraction of lattice volume required for coolant.

If some or all of the above conditions are not entirely achieved, the system would have more parasitic neutron captures and become a high-gain converter. For example, the high-gain converter (Concept 6, defined above) has a specific power of about 9 kWe per kilogram of heavy metal vs. 4.3 to 5.8 kWe/kg for the various breeders.

Since the WCBR requires the preproduction of U-233, a "prebreeder" cycle is necessary to produce the fissile inventory of the breeder. The prebreeder can be any type of reactor containing thorium and a fissile material. For example, U-233 could be produced in an LWR that is modified to contain a thorium reflector or that would substitute thorium rods in whole or in part for standard poison control rods; U-233 could be produced in Pu-Th cycles in LWRs using plutonium recovered from spent LWR fuels, or it could be produced in LMFBRs using Pu-U or Pu-Th as the core fuel with thorium as the blanket material. The cycles involving the LMFBR are discussed in greater detail in Section 1.6.

The DOE/Division of Naval Reactors (NR) has elected to make the production of U-233 in prebreeders an integral part of the LWBR R&D program and has considered various approaches, including (1) conventional PWRs operated with HEU5-Th fuel, (2) conventional PWRs operated with moderately enriched U-235 fuel (MEU5-Th), and (3) seed-and-blanket prebreeder cores that represent the initial years of WCBR operation with conversion to breeder operation after enough U-233 has been accumulated.

One possible scenario is that approximately ten commercial PWRs might be dedicated to a prebreeder cycle producing U-233 for the start-up of about one light-water-cooled breeder reactor (LWBR) per year.

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Another scenario is that U-233 could be supplied as a market commodity for purchase by the operators of WCBR cycles. Thus, it would be unnecessary to store separated U-233 for long periods of time while awaiting the preproduction of enough fuel to start the WCBR fuel cycle in a specific plant.

One facet of the WCBR program has been the utilization of existing LWR technology as much as is feasible; to this end, some of the proposed prebreeders and breeders were designed to backfit into existing PWR plants without derating plant electrical output. For some concepts having a lower power density than exists in present PWRs, derating will be required in some of the backfit applications. For these latter concepts, installation in new plants with larger reactor and containment vessels may be necessary.

1.3.2 Status of Technology

The LWBR Program began in 1965 under the technical direction of the Division of Naval Reactors (NR). At that time, the work on the large seed-blanket reactor, also under the technical direction of NR, was reoriented by the Atomic Energy Commission (AEC) to an R&D program on a thermal breeder for LWRs. Uranium-233 and thorium were selected as the fissile and fertile materials because of their favorable nuclear properties make possible a thermal-neutron breeder.

An LWBR core was installed in the Shippingport Atomic Power Station in 1977 to confirm the LWBR concept. The core has operated successfully for over 12,000 full power hours (as of October 1979) in utility service. This core is expected to provide between three and five years of power operation. Subsequent to the end of operation, special examinations and tests are planned to verify core performance, including an evaluation of breeding characteristics.

The core in this reactor consists of a fuel region surrounded by a reflector region containing only thorium at the beginning of the cycle. In the fuel region, the fuel is a mixture of U-233 dioxide and ThO₂ in the form of pellets stacked in Zircaloy-4 tubes. The tubes are arranged in a hexagonal array to form a fuel module. The fuel region of the core consists of 12 fuel modules.

The central region of each fuel module is referred to as the seed region, and the surrounding region in each module is called the blanket region. The seed region has a higher concentration of U-233 than the blanket region, and it is movable, while the blanket region is fixed. Reactor control is achieved by lifting or lowering the seed regions rather than by parasitic absorption of borated water, as in a PWR.

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The water-cooled breeder designs are similar to the Shippingport LWBR in that (1) the fuel would be a solid solution of UO_2 (primarily $^{233}\text{UO}_2$) and ThO_2 , (2) they would have seed-blanket designs, (3) they would be controlled by movable fuel rods, and (4) they would have a relatively low power density (kilowatts per liter). For the light-water breeder concepts, an additional similarity is that the metal-to-water ratio is high relative to that in a conventional PWR.

The concepts include a range of reactor designs, fuel enrichments, and breeding ratios and utilize different methods of reactor control. Each prebreeder/breeder system is described briefly in Section 1.3.3 below. Several of these systems are described as being backfitted into specific PWR designs such as the Combustion Engineering System 80 and the B&W 205. Although these systems are sized for the specific PWR designs referred to, the systems could be adapted to other reactor designs. All of the systems use ThO_2 as a fertile material. Tables 17 and 18 contain summaries of some of their characteristics.

1.3.3 Proposed Systems

Light-Water Backfit Prebreeder/Seed-Blanket Breeder System (Concept 1)

Description -- The prebreeder plant for this system utilizes a backfitted CE System 80 PWR design with highly enriched U-235 dioxide in the fuel. Boric acid dissolved in the coolant and burnable poisons would be used for reactor control. The average conversion ratio is expected to be 0.5.

The breeder plant for the system would have a somewhat larger core volume than the prebreeder (47,200 liters for a 1000-MWe plant, compared with 40,040 liters for a 1300-MWe LWR), its specific power would be lower, and it would utilize mixed-oxide U-233--U-235-thorium dioxide fuel recycled from the prebreeder. The average conversion ratio for the equilibrium fuel cycle is expected to be 1.0. Cold-to-hot reactivity changes would be compensated by soluble boron in the coolant, but lifetime reactivity changes would be compensated by groups of thoria shim rods (as opposed to the movable seeds of the Shippingport reactor). The components that differ from the standard PWR are the larger reactor vessel, the increase in the number of control-rod drives, the potential need for increased capacity of the safety injection system, and the larger reactor-containment building.

Table 17. Summary of Prebreeder/Breeders

	Prebreeder/breeder with Type I modules, Concept 7		Seed-blanket breeder, Concept 1 breeder, Concept 2 prebreeder/ breeder, Concept 3 prebreeder		Advanced breeder, Concept 5 breeder	Heavy-water seed-blanket breeder, Concept 3 or Concept 4 breeder
	Prebreeder	Breeder	Prebreeder	Breeder		
Thermal output, MW	2,026	2,026	2,993	2,993	2,900	1,540
Net electrical, MW	721	711	1,000	1,000	1,035	500
Core volume, liters	38,600	38,600	47,200	47,200	52,300	160,150
Initial core loading, kg of heavy metal	148,000	164,000	126,300	171,504	235,000	107,650
Fissile material, kg	4,388	3,528	3,783	4,616, 5,622 ^a	4,498	1,996
Conversion ratio	0.72	1.02	0.83	1.03	1.02	1.01
Metal/moderator volume ratio	2.2	2.15	0.87	1.65	2.1	0.13
Average fuel exposure, MWd/tHM	11,200	10,100	20,000	15,300	10,100	11,300
Fuel type	20%U-235/Th ^b	HEU3/Th	HEU5/Th	HEU3/Th	HEU3/Th	HEU3/Th
Reactor, °C						
T _{in}	305	307	299	297	302	282
T _{out}	323	321	324	325	331	320
Control type	Movable seed	Movable seed	Th shim rods	Th shim rods	Movable Th fingers	Th shim rods
Estimated vessel ID, cm	605	605	503	503	513	762
Specific power, kWe/kg of heavy metal	4.9	4.3	7.9	5.8	4.4	4.6

^aThe higher number is for Concept 1 and the lower number for Concept 2. Both values represent a mixed U-233/U-235 charge.

^bDenotes duplex fuel pellets.

Table 18. Summary of Backfit Prebreeder/Converters

Title	Backfit prebreeder Concept 1 prebreeder	Light-water backfit prebreeder Concept 5 prebreeder	Light-water backfit prebreeder/high-gain converter, Concept 6 prebreeder Concept 6 high-gain converter	
			Prebreeder	Converter
Thermal output, MW	3,817	3,800	3,900	3,900
Net electrical, MW	1,300	1,295	1,295	1,295
Core volume, liters	40,040	35,400	37,000	37,000
Initial core loading, kg of heavy metal	93,507	89,800	120,570	141,600
Fissile material, kg	3,088	3,680	4,703	2,602
Conversion ratio	0.5	0.48	0.57	0.94
Metal/moderator volume ratio	0.7	0.7	1.4	1.6
Average fuel exposure, MWd/tHM	33,961	34,800	26,600	11,300
Fuel type	HEU5/Th	16%U-235/Th ^a	20%U-235/Th ^a	HEU3/Th
Reactor, °C				
T _{in}	296	298	293	293
T _{out}	327	330	329	329
Type control	Boron, poison rods	Boron, poison rods	Movable Th fingers	Movable Th fingers
Pressure vessel type	CE-80	B&W 205	B&W 205	B&W 205
Estimated vessel ID, cm	462	462	462	462
Specific power, kWe/kg of heavy metal	14	14	11	9

^aDenotes duplex fuel pellets.

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Spent fuel from the prebreeder is processed to recover uranium, which is stored until sufficient inventory has accumulated for the breeder cycle. For an ex-core lag time of 1 year and a 30-year operating lifetime, the U_3O_8 mining requirements for the system would be 3730 short tons/GWe, increasing to only 3989 for a 50-year operation. The feasibility of fulfilling mechanical, thermal-hydraulic, and materials requirements has been established in the technology already developed by commercial PWR vendors and by the Shippingport LWBR program. Both the prebreeders and the breeder would have operating characteristics similar to those of current PWRs.

Perceived advantages and disadvantages -- An advantage of this system is that the foundations of the design are based on established technology; therefore, the RD&D and the safety/licensing requirements should be satisfied with minimum efforts compared with some of the other concepts. Another advantage is that the 30-year U_3O_8 utilization is considerably better than that of the PWR. The disadvantages pertain primarily to the higher power costs under present economic conditions compared with those of uranium-cycle LWRs. These higher costs stem from the large quantity of costly U-233 that must be supplied in the early years of the cycle and to the high cost of fabricating U-233 fuel for relatively short exposure (15,300 MWd/tHM) in the reactor. The cycle uses HEU-235 in the prebreeder and primarily HEU-233 in the breeder and, therefore, would require more stringent security than the LEU cycle. The use of highly radioactive U-233 fuel requires that the fuel be handled and fabricated remotely. In some cases, radioactivity in fresh fuel is considered to have proliferation resistance attributes (e.g., spiked fuel), but economic penalties result from increased fabrication and handling costs.

Light-Water Seed-Blanket Prebreeder/Breeder System (Concept 2)

Description -- The same reactor vessel and plant are used for both the prebreeder and breeder, which have similar reactor designs. The mechanical design is the same as that of the breeder discussed in the previous section. The fissionable material for the prebreeder design is highly enriched U-235 dioxide, and the average conversion ratio is 0.83. The cold-to-hot temperature defect would be compensated by soluble boron, whereas lifetime reactivity changes would be compensated by fertile thorium shim rods.

The breeder would utilize a thorium based fuel containing an oxide of U-233 and U-235 in an approximate 4:1 ratio for the initial fuel cycle. The conversion ratio is expected to be 1.00. Shutdown and power control capability would be provided by individual poison shutdown rods, and control of the variation in lifetime reactivity would be provided by thorium shim rods. The

shim and shutdown drive mechanisms would be similar to those currently in use.

Plutonium-239 could replace U-235 as the initial fissile fuel, and the prebreeders could be used to supply fuel for other reactor types. These and other variations are available as fueling alternatives. For an ex-core lag time of 1 year and a 30-year operating time, the U_3O_8 mining requirements are 2495 short tons/GWe, increasing to 2929 for a 50-year operation. Because this system would rely on technology already developed by commercial PWR vendors and the Shippingport LWBR program, the feasibility of fulfilling mechanical, thermal-hydraulic, and materials requirements has been established. Furthermore, the system would have the same operating characteristics as the LWRs. Because of the increased size of the core over a standard PWR, the components of the system that will be different from the standard PWR are the same as those described above for the light-water backfit prebreeder/seed-blanket breeder system: the larger reactor vessel and containment building, the larger number of control-rod drives, and the increase in the size of the safety injection system.

Perceived Advantages and Disadvantages -- The advantages of this system are that the same vessel and plant are used for both the prebreeder and breeder concepts. Furthermore, existing LWR technology is utilized, which will reduce the effort required to satisfy safety/licensing requirements. Additionally, the fuel utilization is better than that of the standard LWRs. The disadvantages would be the same as those cited above for the light-water backfit prebreeder/seed-blanket breeder system.

Light-Water Seed-Blanket Prebreeder/Heavy-Water Seed-Blanket Breeder System (Concept 3)

Description -- The prebreeder in this system is the same as the one described for the preceding system. However, the breeder is significantly different, not only because heavy water is used as the moderator-coolant but also because the size of the core for only half the rated power of the previous breeder systems is about a factor of 3.5 larger in volume (160,150 liters as opposed to 47,200). Two such breeders are required to supply a total power of 1000 MWe. The design of the reactor plant is similar to that of a conventional PWR. The reactor is fueled with a mixture of U-233 and U-235 in oxide form with a ratio of the mass of U-233 to U-235 of about 4:1 for the initial fuel cycle. The equilibrium average conversion ratio is estimated to be 1.01. Lifetime reactivity control would be achieved with thorium shim rods. A few of the options available for fueling alternatives are that the prebreeder could use Pu-239 rather than U-235 and that fuel could be produced

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for other reactors using either U-238 or thorium as the fertile material. The U_{308} ore mining requirements for an ex-core lag time of 1 year and an operating time of 30 years is 2171 short tons/MWe. This increases only slightly to 2227 for 50 years of operation. As for plant operability, the prebreeder is expected to have operating characteristics similar to PWRs.

Perceived Advantages and Disadvantages -- An advantage is that the design of the major features of this system is based on established technology, which may facilitate the licensing process. The utilization of fuel is also rather efficient. Lifetime reactivity control in the breeder can be accomplished by shim rods only because the increased reactivity of the blanket with time partially compensates for the decrease in reactivity of the seed.

The disadvantages of this concept include the points cited for the light-water backfit prebreeder/seed-blanket breeder system. In addition, the large reactor vessels, large D_2O inventories, and costs associated with these items may preclude commercial interest in this cycle, inasmuch as more attractive designs using H_2O moderators are available.

Heavy-Water Seed-Blanket Breeder System (Concept 4)

Description -- Mechanically, this concept resembles the heavy-water breeder of the previous section. In general, the system consists of a breeder that is its own prebreeder, and the plant consists of two 500-MWe reactors, both operating first as prebreeders and then as breeders. The initial fuel would be an oxide of highly enriched U-235 with thoria. The cores would be batch-loaded every 2-1/2 years and reloaded with recycled fuel from a companion HWR with similar exposure. As the process is repeated, the concentrations of uranium isotopes would reach equilibrium. Lifetime reactivity control would be by thoria shim rods.

Fueling options would involve the substitution of Pu-239 for U-235 initially. Any such options would not be deleterious to resource efficiency if the Pu-239 were available from spent LWR fuel. For a 2-year ex-core lag time and 30 years of operation, the U_{308} ore requirements are 1804 tons/GWe; after 50 years of operation, the requirement is only 1852 tons/GWe. Because of the reliance on existing technology, the feasibility of fulfilling mechanical, thermal-hydraulic, and materials requirements is established to a large extent.

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Perceived Advantages and Disadvantages -- The advantages are that the same plant is used for both the prebreeder and breeder phases and that the system is efficient from the fuel utilization point of view.

The disadvantages are those cited for the light-water seed-blanket prebreeder/heavy-water seed-blanket breeder system.

Light-Water Backfit Prebreeder Supplying Advanced Breeder (Concept 5)

Description -- The prebreeder plant for the system utilizes a B&W 205 design and, as such, could backfit into an existing PWR vessel with no changes to the plant. Soluble boron would be used for reactivity adjustments during normal operation. A unique aspect of this concept is the use of duplex fuel pellets alternating with thorium pellets. A duplex pellet is a pellet containing a ThO_2 core surrounded by an annulus of fissionable material -- in this case, UO_2 . The outside dimensions are the same as those for the commercial B&W 205 design. The annulus is enriched to 15% with U-235.

The breeder would have a larger core volume than the prebreeder (52,300 liters for a 1000-MWe LWBR, as opposed to 35,400 liters for a 1300-MWe PWR), its specific power would be lower, and it would use highly enriched U-233 dioxide as the fuel. The average conversion ratio is expected to be 1.02. Cold-to-hot reactivity changes would be compensated by soluble boron in the coolant. Normal reactivity control during operation would be accomplished using movable thorium fingers arranged in a manner similar to the poison rods in commercial PWRs. A few of the components of the breeder that differ from the standard PWR are the larger reactor vessel, the control-rod drive mechanism, the potential for a larger emergency core-cooling system, and a larger reactor-containment building.

The mechanical considerations for the prebreeder are such that its successful manufacture appears feasible using PWR technology. A similar conclusion holds for the breeder, except that design changes are required to accommodate the dryer core and the larger reactor vessel than is currently used. These changes are similar to those described earlier for the light-water seed-blanket breeder. Additional testing would be required to ensure that the thermal and hydraulic problems for the breeder would be satisfactorily resolved.

Perceived Advantages and Disadvantages -- The advantages are that the pre-breeder can be backfitted into existing PWRs and that the use of medium-enriched uranium fuel in the prebreeder will not require unusual security precautions. The advanced breeder has superior breeding characteristics compared with other LWBRs, at the expense of a larger core and a tighter lattice. The disadvantages are similar to those already cited for the light-water backfit prebreeder/seed-blanket breeder system.

Light-Water Backfit Prebreeder Using 20 wt % Enriched Uranium Supplying
Light-Water Backfit High-Gain Converter (Concept 6)

Description -- In this system, both the prebreeder and the converter utilize a seed-and-blanket configuration fitted into a B&W 205 vessel. The core would operate in two phases: a low-gain and a high-gain phase. During the low-gain or prebreeder phase, the fuel would consist of duplex pellets alternating with thorium pellets. The annulus of the duplex pellets would be a 20 wt % enriched UO_2 , ZrO_2 , and CaO_2 with U-235 as the fissile material. The average conversion ratio is expected to be about 0.57. Alternate fueling options for the prebreeder could use enriched U-235 or plutonium. Movable thorium fingers would be used for reactivity control for normal operation. Preliminary fuel-cycle data for these options are given in Table 19.

During the high-gain converter phase, the fuel would be highly enriched U-233--U-235-thorium dioxide, and the conversion ratio is anticipated to be about 0.94. Reactivity control during normal operation is again achieved by movable thorium fingers. The fueling alternative for the high-gain phase would be to operate to a higher discharge fuel burnup.

Consideration of the mechanical, thermal-hydraulic, and materials problems of this system indicate that the unique design features would cause no undue problems, but additional testing is required to confirm this judgment, as discussed previously for WCBRs.

A few of the components that differ from those of a standard PWR are the control-rod drives and the larger emergency core-cooling system that would be required to accommodate the increase in fuel mass.

Perceived Advantages and Disadvantages -- The advantages of this system are the backfitting capabilities of both the prebreeder and the converter into an existing PWR plant design and the relatively low inventory of U-233 compared

Table 19. Preliminary Fuel-Cycle Data, Light-Water Prebreeder/High-Gain Converter^a

	High-gain converter cycles					Prebreeder cycles		
	U-233/Th	U-233/Th	U-233/Th	U-233/Th	DU/Th	MEU/Th	HEU/Th	Pu/Th
MWd/t	11,000	11,300	22,600	30,200	30,000	26,600	26,600	30,000
Data source	NR	ANL	ANL	ANL	ORNL/ANL	NR	ORNL	ORNL/ANL
Initial core, kg fissile/GWe								
U-233	1975	1978	2186	2268	2663			
U-235	27	27	30	31	36	3630	3000	
Pu _f								4419
Annual charge, kg fissile/GWe	(0.75 CR)							
U-233	1330	1339	763	611	739			
U-235	18	18	10	8	10	1500	1300	
Pu _f								1357
Annual discharge, kg fissile/GWe								
U-233	1286	1278	659	494	531	345	470	353
U-235	40	41	33	30	32	655	500	5
Pu _f					95	170		669
Conversion ratio ^b	0.94	0.95	0.91	0.89	0.89	0.57	0.57	0.7

^aHeavy metal loading/GWe = $\frac{\text{Core}}{\text{Th reflector blanket}}$ = 109,300 kg + 41,700 kg for high gain converter.
Heavy metal loading/GWe = 93,100 kg + 41,700 kg for prebreeder.

^bANL conversion ratio (CR) reduced by 0.02 to account for probable leakage and control losses.

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with WCBRs with a conversion ratio of 1 or more. The prebreeder and breeder fit into the same vessel. The NASAP evaluation indicated that the light-water prebreeder/high-gain converter concept is more economical than the other LWBR concepts for intermediate-term application and might be competitive with LWRs at U_3O_8 prices above \$100/lb. For this reason, the high-gain converter was considered as a reference in Volumes III and V. The long-term fuel utilization, however, cannot compete with other LWBR concepts.

The disadvantages relative to LWR recycle stem from the need to adopt a complex fuel cycle requiring both prebreeders and high-gain converters. The highly radioactive U-233 fuel would require fully remote fuel handling and fabrication.

Prebreeder and Breeder Based on Light-Water Breeder Reactor (LWBR) Type I Modules (Concept 7)

Description -- The prebreeder and breeder in this system would use core modules that are geometrically identical to the LWBR Type I modules; thus, this system is a direct extension of the Shippingport LWBR design to commercial-sized cores. In addition, the cores of the prebreeder and breeder would have identical dimensions, so that the breeder core could replace the prebreeder with a minimum of mechanical design changes to the vessel or plant. The core volume in each case is somewhat larger than that of the B&W 205 plant (38,600 liters for 100 MWe as opposed to 35,400 liters for 1300 MWe). Reactivity control in both the prebreeder and breeder is by the lifting or lowering of movable seed assemblies relative to fixed blanket assemblies, as is done in the Shippingport LWBR.

The fuel for the prebreeder would be in the form of duplex pellets whose annular region is a moderately enriched (<20% U-235) solid solution of UO_2 , ZrO_2 , and CaO . The conversion ratio is calculated to be 0.72.

The fuel for the breeder design is in pellet form and consists of a solid solution of U-233 dioxide in ThO_2 . The average conversion ratio is expected to be 1.02.

Fueling alternatives for the prebreeder would be similar to those of other prebreeder concepts and include fully enriched U-235 and plutonium. For the breeder, the alternatives are reduction in fissile inventory, operation at higher power density, and operation at higher discharge fuel burnup. Any one of these alternatives will reduce the conversion ratio. Because the system

is an extension of the Shippingport reactor, the mechanical, thermal-hydraulic, and materials features are essentially based on proven concepts.

Perceived Advantages and Disadvantages -- The advantages of this system are that (1) it is a direct extension of a technology presently under test, namely the Shippingport LWBR; (2) both the prebreeder and breeders occupy the same facility with a minimum of mechanical changes; and (3) the breeder concept offers the prospect of efficient fuel utilization.

The disadvantages are similar to those noted for the light-water backfit prebreeder/seed-blanket breeder system. In addition, the rated power level is below that normally ordered in the U.S.

1.3.4 Licensing and Safety Issues

Light-Water Prebreeder/Breeder Reactors

The safety-related features of these systems 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 negative temperature coefficient of the coolant. The systems would have a more negative beginning-of-cycle moderator reactivity coefficient than a typical commercial PWR. Design-related safety features would include the reactor protection system, the decay-heat-removal system, ECCS, and containment (all systems common to other LWRs).

The LWBR systems contain close-packed seed and blanket rods. This close-packed lattice may have some impact on LOCA performance when scaled to commercial size. Some changes in sizing and set points of auxiliary heat-removal system and ECCSs of typical LWRs may be needed to accommodate this impact. Any uncertainty would be resolved by design and prototypic testing.

The following items (1 through 8) are the major unique features of these systems (compared with reference PWR systems) that would need to be considered for licensing. These features were evaluated in the Shippingport LWBR Safety Analysis Report and were found to be acceptable.

1. Tight lattice. Rod-to-rod spacing of 60 to 100 mils in these concepts (compared with commercial practice of 120 mils) would impose fuel element and grid-design considerations and assembly restrictions more stringent than those in commercial practice to avoid fuel-rod-to-grid-structure and fuel-rod-to-fuel-rod contact. The minimum nominal rod-to-rod spacing in the LWBR core at Shippingport is about 60 mils. Core operational limits, such as specifications of set points and allowable power increase, have been set for the Shippingport LWBR, assuming rod-to-structure contact does occur. Extensive test programs with rods in contact with one another both in-pile and out-of-pile have been completed. Design assessments of seed-and-blanket fuel-rod bowing predict that fuel-rod-to-fuel-rod contact is highly unlikely. Bowing analysis of the blanket has been completed on a worst-case basis, and rod-to-rod spacing has been determined as a function of reactor lifetime. These data demonstrate that adequate margin is predicted for the Shippingport LWBR and indicate that acceptable operation of fuel assemblies in a commercial-scale reactor of the present concepts should be feasible.

2. Core thermal margins. Shippingport LWBR thermal-analysis programs and correlations 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 Shippingport LWBR critical heat flux (CHF) correlation conservatively predicts the data for the full range of LWBR geometries and heat flux distributions. Thermal margins for the present concepts have been calculated based on an approach similar to that used for the Shippingport LWBR. Additional CHF testing might be required to confirm the applicability of the modeling to the specific fuel lattice of the current concept.

3. Provision for accident prevention. The likelihood of an accident occurrence for the Shippingport LWBR core and for the present concepts is comparable with that for the reference PWR. The safety and protection of the Shippingport LWBR plant have been designed within regulatory guidelines and requirements. An ECCS appropriate for the LWBR has been designed into Shippingport. Research has identified no impediment to providing comparable protection features in a large plant using the present concepts.

4. Acceptability of movable fuel as means of reactivity control. All reactivity control functions required by an operating nuclear power plant using movable fuel have been satisfactorily performed to date during the operation of the Shippingport LWBR. These include control required for shutdown, plant heat-up, power operation, and lifetime reactivity changes. Scram reactivity is also provided by the movable fuel. For some of the present concepts, the movable fuel would supply only the reactivity control to match part of the

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power defect, plus the buildup of fission products, including equilibrium xenon. Thus, the requirements of the movable fuel in the present concepts should be much less severe than they are in the LWBR core.

For large core applications using fuel management, the amount of control required should be significantly smaller than that required in the Shippingport LWBR demonstration, where fuel remained in place during exposure. However, because 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 action to accommodate incidents such as rod ejection, beyond that already done for the Shippingport LWBR core.

The essential difference between some of the LWBRs in this subsection and the presently operating Shippingport LWBR core is that, instead of movable fuel-rod bundles, movable fingers of individual thoria control rods arranged in a manner similar to the fingers of poison rods now being used in commercial PWRs are interspersed throughout the module. Poison-finger rods may also be required either in the same modules with thoria-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 Advanced Water Breeder Applications (AWBA) program.

5. Power and temperature coefficients. For the breeder systems discussed in this subsection, there are two aspects to the range of power and temperature coefficients that could be expected in the present breeder concepts. The first aspect concerns the accuracy with which the calculational model can predict these coefficients; the second concerns the values of these coefficients that have been calculated or estimated.

With respect to accuracy, the calculation of temperature and power coefficients for the Shippingport LWBR core using the design model has resulted in agreement for both the operating core and critical experiment configurations. 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 measured.

Concerning the values of the coefficients, there is a firm basis for expecting the temperature and power coefficients for the breeder concepts to be similar to those for the Shippingport LWBR core and, therefore, to 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 or to increase slightly during the life of the core. This is consistent with the facts that (1) no boron would be used in the core, (2) the fuel-to-coolant ratio would decrease slightly as movable fuel was lifted, and (3) 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 concepts, the fuel-to-coolant ratio would be somewhat larger, and the power density is slightly higher than that of the Shippingport LWBR core. Therefore, the power coefficients for the present breeder concepts are expected to be similarly predictable and at least as negative as that for the Shippingport LWBR.

6. Control stability and adequacy. The LWBR core at Shippingport has operated successfully as part of the integrated Duquesne Light Company power network. This operation has included long periods at constant power, several months of planned swing-load operations, and controlled start-up 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 with commercial plants. Performance was satisfactory during both swing-load and steady-state operation and for the special tests. These results indicate that control stability and adequacy can be maintained in a core with movable fuel reactivity control.

7. Nuclear stability. Analytical studies have examined the stability of large U-233/Th reactors with high power densities against spatial xenon oscillations relative to the stability of uranium-fueled PWRs with comparable size and rating. These studies show that U-233/Th systems are significantly more stable, primarily because of lower total xenon yields, larger fraction of xenon chain yield directly to xenon, and large-in-magnitude 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 concepts would have acceptable stability properties.

8. Prebreeder fuel. Some of the prebreeders in this subsection are basically thorium-cycle PWRs with licensing and safety problems similar to those covered in Section 1.1.4. Regarding those reactors utilizing duplex

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pellets, placing the fissile fuel-bearing uranium annulus around the thorium core, which initially contains no fissile fuel, would cause most of the fission energy to be generated close to the cladding. This, therefore, would result in lower average fuel-element temperatures than would be obtained with a solid uranium pellet. This arrangement would also have the advantage of reducing stored heat, which would be important during an LOCA. In some concepts, the annulus would experience peak burnup as high as 119,000 MWd/t, compared with 50,000 MWd/t 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 AWBA program, including analytical work to extend the scope of present fuel-element modeling and irradiation testing.

Heavy-Water Breeder Reactors (HWBRs)

The safety-related features of the HWBRs would be similar to those of current PWR reactors. Natural protection against the adverse effects of reactivity addition and LOCAs would be provided by the negative Doppler coefficient of the fuel and negative temperature coefficients of the coolant. Design-related safety features would include the reactor protection system, the decay-heat-removal system, ECCS, and containment (all of which are developed for and common to LWRs).

The unique licensing issue that must be treated for licensing of the HWBR in the U.S. would be related to the use of heavy water as a reactor coolant. Although not significantly different from light water in physical characteristics, irradiated heavy water does contain higher concentrations of tritium. Potential release of tritium via leakage, discharge, and/or breach of containment is an additional plant-design consideration for any heavy-water system. CANDU experience shows that this can be resolved by proper plant design; for example, in 1974, Pickering releases of tritium were <0.3% of limits. The concept discussed herein would have to be evaluated to determine the feasibility of minimizing tritium release.

Additional features that are unique to the use of heavy water and/or the heavy-water seed-blanket core arrangement must be considered for safety and licensability and are included in the following discussion.

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1. Core arrangement. The HWBR core concept would be unlike current light-water designs in many respects, including seed-blanket arrangement, widely spaced fuel rods, low flow velocity, and hydraulically separated fuel and moderator regions. The large coolant inventory and widely spaced rods would be advantageous with respect to inherent core safety, while low coolant velocity reduces the hydraulic loads on core structure. Nuclear stability would be expected to be the same as that for current PWRs, even though the HWBR core would be larger, because the neutron age and diffusion length in heavy water are much greater than those in light water. Effort would be needed to develop nuclear design and thermal-hydraulic techniques and criteria.

2. Potential for sudden change in moderator capability via dilution of heavy water. The HWBR would be designed so that no paths would exist for rapid introduction of nonborated light water into the primary system. Any system malfunction that would introduce light water would result in reactivity excursions either negative or less positive than other postulated accidents.

3. LOCA/ECCS response characteristics. Low power density, large coolant inventory, and large flow area favor good LOCA performance. The effectiveness of the ECCS, which would use borated light water, would be particularly enhanced by the use of upper plenum spray in combination with inlet region injection. A negative void coefficient during blowdown could be assured through core design in the worst case. The large coolant inventory, however, would increase the difficulty of designing a compact containment.

The HWBR would have to be designed to the same criteria and safety margins embodied in regulations currently applied to conventional LWRs. However, additional design effort would be required to account for extensions of thermal-hydraulic correlations and for control of heavy-water inventory in addition to the other safety and/or licensability concerns associated with the unique features identified above.

1.3.5 Environmental Issues

Light-Water Prebreeder/Breeder Reactors

The final environmental statement on the Shippingport LWBR (ERDA 1541) covered both the Shippingport operation and the large-scale application of light-water breeder and prebreeder cores. The statement was published in June 1976, after review by all concerned government agencies and the public

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and after resolution of questions and comments. The environmental effects of the reactors reported here lie within the range of cases considered by that statement.

In the LWBR environmental statement, the environmental impacts of the introduction of LWBR reactor types and the thorium-based fuel cycle that sustains them are compared with the conditions that would prevail with a continuing use of the current-day LWR reactors and the uranium-plutonium fuel cycle that sustains them. The major types of impacts on the environment for the two cycles are similar for a given amount of energy produced. The reactor plants are similar, and there are very similar supporting facilities, including the mine, mill, fabricating plant, and reprocessing and waste-management facilities. The initial impact of introducing LWBR reactor types commercially would be an increased requirement for mining and enriching of uranium, as well as a need for initiating thorium mining and milling. Eventually, however, the U-233 produced in the prebreeders would provide the fissile fuel for high-gain converters or breeders, and the mining and enriching of uranium to support a given capacity would decrease or be eliminated. The reprocessing of fuel and management of radioactive waste would continue. The usage of other natural resources such as water and land would be less than that for current LWRs, because of the significantly reduced mining and reduced energy requirements for enrichment.

The potential hazards inherent in any of the facilities associated with the two cycles would be approximately the same.

The principal environmental and licensing issues remaining with respect to the reactor concepts reported here are those that are generic to the recycling of fuel from fission reactors and management of their wastes. The U-232 resulting from irradiation of thorium fuel provides some nonproliferation advantages and some disadvantages relative to uranium fuel recycling.

Heavy-Water Breeder Reactors (HWBRs)

Except for concerns associated with tritium release, the operational impact on the environment for these systems would be equivalent to that of current commercial LWRs, because of the similarity of the two reactor systems.

Successful operation of CANDU plants has shown that leakage of heavy water and environmental release of tritium can be held to acceptable levels for that specific plant design. Development of the concept discussed herein

would have to include an assessment of the feasibility of maintaining acceptably low leakage rates. Pursuit of the HWBR as a major energy option would also require development of the industrial capacity to produce large quantities of heavy water. The heavy-water inventory of a 500-MWe HWBR primary system would be $\sim 14,000 \text{ ft}^3$.

Differences in reactor design would have no significant impact on the environment during normal operations, and safety systems used in current LWRs could be readily modified in size and performance to ensure that the environmental impact of unexpected transients or accidents would be no greater than that postulated for existing LWR plants. Latest LWR plant-design technology would be relied on to minimize both environmental impact during operation and the consequences of any accident transient. The impacts of mining and enrichment would be less than for comparable light-water reactors because of lower fissile inventory.

1.3.6 RD&D Needs

Fuel-cycle RD&D data are not included in any of the scheduling and cost estimates given below.

Some of the prebreeders utilize HEU-235 while the breeders utilize primarily U-233; therefore, all RD&D involving these fuels will have the additional complications and expense associated with stringent safeguard precautions.

Backfit Prebreeders (Concepts 1 and 5)

Since the backfit prebreeders are basically commercial PWRs with different fuels, the RD&D program would be very similar to that described above for a denatured-uranium-thorium cycle in LWRs. The prebreeders will require the development and qualification of the fuel and demonstration of suitable control and safety characteristics. These concepts will probably not require a demonstration or lead plant.

The RD&D requirements of the light-water backfit prebreeder supplying the advanced breeder (Concept 5) would be extended additionally because the fuel system consists of alternating duplex pellets and solid thorium pellets. This fuel system is being evaluated for possible further development as part of the DOE water-cooled breeder program.

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Duplex pellets have been fabricated, and irradiation testing of fuel rods is in progress. Additional CHF testing would likely be required. Computer models are also being developed to predict the performance of this fuel system.

It is anticipated that the funding and time-span requirements to complete the development work would be comparable with those experienced by commercial PWR manufacturers when new core designs are introduced.

Seed-Blanket Prebreeders, Breeders, and Advanced Breeder (Concepts 1, 2, 3, and 5)

The seed-blanket prebreeder and breeders in Concepts 1, 2, and 3 are of a novel core design that has only been tested in the Shippingport LWBR.

The initial development program would involve a preliminary evaluation of safety and licensing issues to identify missing information and to guide the subsequent programs.

A physics-verification program would be necessary to provide the information required to predict important safety-related physics parameters and to demonstrate the accuracy of such predictions as part of the safety analysis. The physics-verification program would consist of a series of critical experiments. These experiments would serve as a basis for demonstrating the adequacy of cross-sectional data sets and the ability of analytical models to predict such safety-related parameters as reactivity, power distributions, moderator-temperature reactivity coefficients, boron worth, and control-rod worth.

Another major area of data-base development would consist of the establishment of additional safety-related fuel performance information, including transient fuel-damage limits, thermal performance both for normal operation and with respect to LOCA margins on stored heat, dimensional stability (densification and swelling), gas absorption and release behavior, and fuel-cladding interaction.

Subsequent fuel-related RD&D would be devoted to developing core design changes and safety analysis information in preparation for a reactor/fuel-cycle demonstration. Appropriate safety criteria, such as acceptable fuel-design limits and limits on maximum energy deposition in the fuel, must be

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determined. A safety analysis report for the reactor/fuel-cycle demonstration plant would be prepared as part of this development task, which must be pursued with licensing authorities' approval.

An additional area that might require some development is the ~510 cm inside-diameter (ID) pressure vessel, the vessel head, and reactor internals required for this concept. Manufacturing capability for fabricating 635 cm ID vessels for BWR plants presently exists, and it is not anticipated that fabrication of the larger vessel will require additional manufacturing capabilities.

The advanced breeder (Concept 5) is somewhat similar to the seed-blanket breeders, except that it has a slightly larger core and pressure vessel. The RD&D program for the advanced breeder would be essentially as described above for the seed-blanket breeders.

Heavy-Water Breeder Reactors (HWBRs) (Concepts 3 and 4)

The heavy-water seed-blanket breeder and prebreeder would also rely on technology that has already been developed, but additional RD&D effort would be required to demonstrate design adequacy. Heavy-water critical experiments have never been performed with thorium and U-233; these experiments are essential for the reactor design. Experiments with enriched U-235 would also be needed to support the prebreeder design. Extensive testing programs would also be required to confirm low-flow thermal-hydraulic design criteria for the heavy-water breeder and to confirm the calculation of various nuclear and thermal performance parameters of the core arrangement.

Additionally, a fuel-related development program as described for the light-water seed-blanket breeder would be necessary.

A development effort would also be needed to first establish a production capability for the large reactor vessel, the vessel head, and reactor internals required for this concept. In addition, development of systems to control heavy-water inventory and to limit the release of tritium will be necessary. The remaining components of the plant design (the primary-coolant and related auxiliary and safety systems and the steam supply system) would be expected to be very similar in function and form to those of conventional PWRs.

The remaining components of the plant design (the primary-coolant and related auxiliary and safety systems and the steam supply system) would be expected to be very similar in function and form to those of conventional PWRs.

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Existing light-water and heavy-water reactor technology can be relied on in the development of some of the unique features that will be required for this concept. It is anticipated that the funding and the time to commercialization (including development of a manufacturing capability for the large reactor vessel and heavy-water inventory and control) would be greater than those required by the light-water seed-blanket breeder.

Light-Water Backfit Prebreeder Using 20 wt % Enriched Uranium Supplying Light-Water Backfit High-Gain Converter (Concept 6)

Because this prebreeder/breeder system is proposed as a system that would utilize the same reactor vessel and in which the fuel would switch from 20 wt % U-235 to primarily U-233 after 13 years, it was thought to be different from many of the previous systems that could be grouped generically. Although this system would use a standard B&W 205 pressure vessel, the additional RD&D required is similar to that for some of the other systems.

The areas in which the technology would need to be extended include the fuel-assembly grids, the thoria control fingers, and the plenum. Zircaloy grids are already in use in commercial cores, but an extension to a larger size, to a greater number of rods, and to tighter tolerances associated with the close-packed hexagonal array of the high-gain converter reactor would be required. The thoria control fingers would be 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 cross flow. Because the thoria control fingers constitute about one-third of a fuel assembly, the additional weight on the drive mechanism will be considerably greater than that on the standard LWR drives. The mechanism must therefore be sturdier, and the thickness of the vessel head must be increased to accommodate the additional stress. A thorough analysis must be made of the drive mechanisms, the vessel head, and the reactor support structures to include the effects of the additional weight.

The "dual-action" control-rod drive mechanism, if selected, would also require an extension of existing 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.

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In addition to the above RD&D on the reactor components, the use of this concept would require the recycling of fuel and the RD&D necessary to implement reprocessing, refabrication, and waste management.

Existing technology would need to be extended for the fuel system of alternating duplex pellets and solid thorium pellets. This fuel system is being evaluated for possible further development as part of the DOE Advanced Water Breeder Applications (AWBA) program. Duplex pellets have been fabricated, and irradiation testing of fuel rods is in progress. Additional CHF testing would likely be required. Computer models to predict the performance of this fuel system are also being developed.

Prebreeder and Breeder Based on Light-Water Breeder Reactor (LWBR) Type I Modules (Concept 7)

This system is different enough from the previous systems that the RD&D will be considered separately.

The Nuclear Regulatory Staff and the NRC Advisory Committee for Reactor Safeguards (ACRS) performed a safety review of the Shippingport reactor using an LWBR Type I module core and found the design acceptable from a safety viewpoint. However, NRC stated that a number of unknowns, uncertainties, and/or issues will have to be addressed for LWBRs of larger size and/or different fuel cycles from those of the reviewed Shippingport LWBR of 236 MWt.

More information and detailed evaluations will be needed in areas such as:

- adequacy of control systems,
- the potential for recriticality events and the consequence of such events during low-probability meltdown accidents,
- compliance with Appendix K of 10CFR 50 (ECCS Evaluation Models),
- thermal performance characteristics of LWBR fuels, core thermal-hydraulics, calculational models, data base with thorium-based fuels, etc.,

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- differences between current-vintage LWRs and LWBRs and the effect of these differences in relation to normal operation, minor accidents, major accidents, and performance required from safety-related systems,
- stability of the LWBR core with duplex pellets due to the delayed heating of the pellet core,
- potential for separation of ThO_2 and UO_2 in a duplex pellet under transient conditions due to the difference between their melting points (about 400°C).

This information, as discussed previously, would have to be developed individually for each alternate LWBR fuel cycle.

In addition, the characteristically low average power density in this LWBR core (about 50 kW/liter) would require a larger pressure vessel than is currently being fabricated. This extension of technology would require a modest development program. It is anticipated that additional manufacturing facilities would not be necessary for this size vessel.

The main item of reactor RD&D for the prebreeder would be the determination of the performance of the prebreeder fuel. This fuel system is being evaluated for possible further development under the DOE AWBA program that includes analytical work to extend the scope of present fuel-element modeling and irradiation testing.

RD&D Costs and Schedules

For purposes of characterization of WCBR RD&D costs, representative concepts were selected for detailed evaluation. They are Concepts 1, 6, and 7, and their estimated costs and time schedules are shown in Tables 20-22.

1.3.7 Perceived Advantages and Disadvantages

A variety of concepts have been proposed in the generic class of WCBRs. As a class, they require fuel recycle and development of the thorium fuel cycle and offer evolutionary changes to the PWR in the direction of much better

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Table 20. RD&D Summary: Light-Water Backfit
Prebreeder/Seed-Blanket Breeder
(Concept 1)

(In 1978 dollars)

Step	Cost (\$ million)
Generic R&D	375
Support R&D	235
NSSS vendor support	200
Lead plant (1000-MWe seed-blanket breeder)	1000
Breeder core	465
Three prebreeder reloads	<u>165</u>
Total reactor RD&D cost	2440
Event	Year
Decision to build lead plant	1990
Lead plant starts up	2002

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Table 21. RD&D Summary: Light-Water Backfit Prebreeder/
Light-Water Backfit High-Gain Converter
(Concept 6)

(In 1978 dollars)

Step	Cost (\$ million)
Generic R&D	375
Support R&D	125-325
NSSS vendor support	200-300
Lead plant (3800-MWt/1270-MWe high-gain converter)	1100-1200
Prebreeder core	270
High-gain core	320
Total reactor RD&D cost	2390-2790
Event	Year
Decision to build lead plant	1990
Lead plant starts up	2002

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Table 22. RD&D Summary: Prebreeder and Breeder
Based on LWBR Type I Modules
(Concept 7)

(In 1978 dollars)

Step	Cost (\$ million)
Generic R&D	375
Support R&D	60
NSSS vendor support	200
Lead plant (700-MWe scale)	1000
Prebreeder core	240
Breeder core	<u>420</u>
Total reactor RD&D cost	2295
Event	Year
Decision to build lead plant	1985
Lead plant starts up	1997

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fuel utilization and, ultimately, breeding. The continued reliance on a fully commercialized technology, the PWR, is perhaps the most significant advantage of the WCBR.

The disadvantages of the WCBR approach lie in the complexity of the pre-breeder/breeder cycle used for some concepts, the need to provide incentives to stimulate development and interest in thorium fuels, and the need to develop an economic/commercially oriented design.

1.4 HIGH-TEMPERATURE GAS-COOLED REACTORS

1.4.1 Reactor/Fuel-Cycle System Descriptions: Steam-Cycle (SC) Designs and Direct Cycle (DC) Concepts

The high-temperature gas-cooled reactor (HTGR) offers two alternative power cycles for base-load power generation: the steam cycle (SC) and the direct cycle (DC). Most of the past experience has been with SC plants, and the SC data have been the primary data source for NASAP resource and economic assessments. This section presents a discussion of both SC and DC technology, as well as a general description of the fuel cycles common to both.

Steam Cycle (SC)

The reactor core is cooled with helium, moderated and reflected with graphite, and fueled with uranium and fertile thorium. It is constructed of hexagonal graphite blocks, with vertical holes for helium coolant flow, fuel-rod placement, and control-rod movement.

The reactor core and helium coolant are contained in a multicavity prestressed concrete reactor vessel (PCRV). In a 1330 MWe-sized commercial HTGR, helium coolant flows from six steam-driven circulators through the core, through the six main steam generators (each located in a cavity in the PCRV wall), and back to the circulators. The reference HTGR-SC for NASAP is an HTGR design of 1330 MWe output, a net plant efficiency of 39.6%, and a core power density of 7.1 W/cm^3 . General HTGR-SC performance specifications are given in Table 23.

The superheated steam produced in the steam generators is passed through the high-pressure section of the main turbine and then to the helium-circulator steam-drive turbines. On exit from the circulator steam turbines, it passes through the radial reheaters in the steam generators before it enters the intermediate-pressure section of the main turbine. Waste heat is removed from the steam by a water-cooled condenser and rejected through cooling towers to the atmosphere. The components and systems that constitute the NSSS are shown as a perspective cutaway in Figure 11.

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Table 23. HTGR-SC Performance Specifications

<u>Power plant performance</u>	
Thermal power, MW	3,360
Electrical power, MW	
Gross	1,360
Net	1,332
Thermal efficiency, %	39.6
<u>Reactor parameters</u>	
Core volume, liters	4.706×10^5
Core power density, MW/liter	0.0071
Core height, m	6.34
Number of enrichment zones	1
Coolant flow rate, Mg/sec	1.66
Coolant inlet temperature, °C	320
Coolant outlet temperature, °C	750
Primary system pressure, MPa	5.38
Total primary system pressure drop, MPa	0.12
Number of fuel elements	5,288
<u>Fuel parameters</u>	
Average fuel temperature, °C	880
Maximum fuel temperature, °C	1,300
Cladding temperature, °C	
Core fuel loading, kg	
Total heavy metal	30,600
Fissile material	1,350
Discharge exposure, MWd/t	
Average	130,000
Peak	165,000
Conversion ratio	0.59

Figure 11. Nuclear Steam Supply System of the Lead Plant HTGR-SC.

Direct Cycle (DC)

The ability to drive a gas turbine directly with primary helium in an HTGR-DC cycle has long been recognized as an improvement to the HTGR-SC. Development emphasis has shifted recently from the HTGR-SC to the HTGR-DC in the expectation that the direct cycle will prove more economical than the HTGR-SC and because the formation of an international HTGR-DC and process heat program with the Federal Republic of Germany (FRG) is anticipated.

Based on recent conceptual HTGR-DC concepts, the design is presently viewed as offering the promise of improvements through plant simplification, reduced capital costs, increased net plant efficiency, use of economical dry cooling systems, and enhanced cogeneration capabilities. The waste heat produced by the HTGR-DC can be economically rejected by dry cooling towers, resulting in a 40% net plant efficiency; optionally, if wet cooling is used, the waste heat can power a secondary Rankine cycle (HTGR-binary cycle) resulting in an overall net plant efficiency of 46%.

The HTGR-DC plant can combine the existing HTGR core and fuel designs with a closed-cycle helium gas-turbine power conversion loop (PCL) that operates directly with reactor coolant helium. The PCLs are integrated into the PCRV on the basis of both safety and economic considerations; thus, the necessity of providing for failure of large external metallic pressure vessels and ducts is avoided. The PCRV is located within a secondary containment building, which, together with the PCRV, incorporates safety features to limit loss of primary coolant and to limit missile damage in the event of failures in the turbomachinery, shaft seals, generator, heat exchangers, and other components.

Because the helium gas turbines and compressors are small compared with steam turbines (resulting from small volumetric changes and pressure drops across gas turbines and compressors) and because the gas turbines and compressors can be integrated into the PCRV, the large turbine/generator building is eliminated. The end result is a potentially low capital cost system. If dry cooling is required, the capital cost is further reduced because of the greatly reduced cooling tower surface area requirements that result from the high reject temperatures of the Brayton cycle (120 to 150°C). Also, the steam-cycle helium circulators are replaced by the helium compressors mounted in tandem with the gas turbine.

Based on the utilization of an existing 3000-MWt core design, the HTGR-DC embodies three PCLs arranged in a delta configuration, each rated at 1000

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MWt. The simplified isometric diagram of the reactor and primary system (Figure 12) shows the core, turbomachinery, heat exchangers, and entire helium inventory enclosed in the PCR. The isometric view indicates the integrated approach for the direct-cycle plant.

The salient cycle parameters for the plant are given on the simplified loop diagram (Figure 13). As shown in this diagram, each loop includes a single-shaft gas turbine, a recuperative gas-to-gas heat exchanger, and a precooler (gas-to-water exchanger) for cycle heat rejection. A summary of the main features of the direct-cycle HTGR power plant is given in Table 24.

Fuel Design

The fuel element is a hexagonal graphite block that (1) contains the fuel, (2) channels the coolant flow, and (3) acts as the moderator. The hexagonal graphite block contains drilled coolant passages and fuel channels into which the fuel rods are inserted. The individual fuel rods contain the fissile TRISO-coated (four-layered) and fertile BISO-coated (two-layered) particles distributed in a graphite matrix. The initial core elements and the reload elements, whether containing fresh or recycle fuel, are of identical geometry. However, the particle sizes can be adjusted to maximize resonance self-shielding effects, minimize fissile inventory requirements, or maximize conversion ratio.

The fissile particle is a uranium carbide kernel with a TRISO coating. The TRISO coating has four layers: an inner buffer layer of low-density pyrolytic carbon, a thinner layer of high-density pyrolytic carbon, a layer of silicon carbide that provides containment of gaseous and solid fission products at high burnups, and an outer layer of high-density pyrolytic carbon that adds strength to the coating.

The fertile particle is a thorium oxide kernel with BISO coating. The BISO coating has two layers: an inner buffer of low-density pyrolytic carbon and an outer coating of high-density pyrolytic carbon. The latter provides the containment. These fuel elements reside in the core for four years and are removed and replaced by a fuel-handling machine while the reactor is in a shutdown, depressurized mode of operation.

The generic HTGR fuel design exhibits a flexibility in meeting nonproliferation and resource-conservation objectives. The particle nature of the

Figure 12. Integrated HTGR-DC with 3000-MW(t) Reactor Core and Three Power Conversion Loops.

Figure 13. Cycle Diagram of HTGR-DC with Dry Cooling.

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Table 24. Main Features of Nuclear Closed-Cycle HTGR-DC

-
- Power plant life, 40 years
 - Plant availability, 80%
 - 3000-MWt core thermal power rating for three loops
 - 40% efficiency with dry cooling
 - 48% efficiency with ammonia bottoming cycle
 - Reference design based on
 - optimized parameters for minimum overall cost of power
 - integrated configuration
 - 850°C (1562°F) reactor outlet temperature
 - multiple gas turbine loops
 - nonintercooled cycle
 - high degree of heat recuperation
 - state-of-the-art technology, materials, and fabrication methods
 - major components with rail transportation limits
-

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fi ssile and fertile kernels allows the reactor designer to retrofit once-through low-conversion-ratio and recycle high-conversion-ratio fuel cycles by varying the respective fi ssile and fertile particle loading without modifying the heat transfer characteristics of the fuel. High-efficiency HTGR-DC and binary-cycle concepts allow for additional U_3O_8 utilization improvement. The high burnups achievable in spiked HEU fuel (100,000 to 150,000 MWd/t) greatly reduce the sensitivity of the HTGR fuel-cycle costs to high fabrication costs.

The HTGR-SC and the HTGR-DC utilize basically the same fuel design and core configuration, except for the following potential differences:

1. HTGR-DC fertile particles may ultimately require silicon carbide coatings to minimize fission-product releases (convert from BISO to TRISO) to reduce occupational exposure during gas turbine maintenance.
2. HTGR-DC fi ssile particles may also require an additional ceramic coating (like silicon carbide) to further reduce fission-product release rates.
3. HTGR-DC hexagonal graphite fuel-block configurations may require a redesign to increase coolant surface area and/or a decrease in fuel pellet size may be needed so that the ratio of maximum to average fuel temperature would be reduced. However, the fabrication costs would increase. Also, fuel management techniques will have to be modified to further reduce age peaking factors.
4. The HTGR-DC can more effectively utilize the pebble-bed core configuration with the once-through-then-out (OTTO) fuel cycle and on-load refueling to further minimize peak fuel temperatures and improve availability.

Fuel-Cycle Options

20% U-235-Thorium Once-Through Fuel -- Fuel would consist of 20%-enriched uranium carbide fi ssile particles and thorium oxide fertile particles. The HTGR allows much more complete burnup of the charged fuel than the LWR, and, consequently, the once-through fuel can be designed to minimize plutonium discharge quantities and isotopics in the spent fuel. The spent fuel may be

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stored at the reactor site or at an offsite storage facility. Ultimately, the fuel may be sent to a geologic repository if it is decided to forgo reprocessing.

Uranium-Thorium Recycle -- If recycle were permitted, the fuel would be designed to contain more thorium than in the once-through cycle to encourage U-233 production. The spent fuel is disassembled to separate fissile and fertile particles. The seed fissile particles are discarded to high-level waste storage, while the fertile particles are reprocessed to recover U-233, which is denatured to 12% with U-238 and recycled. The thorium recovered during reprocessing is stored for ten years and then recycled. The discarded fissile particles may ultimately be reprocessed to recover uranium and plutonium, or they can be sent to a geologic repository with other high-level wastes.

Over 20% U-235-Thorium with Recycle -- This is the cycle historically planned for the HTGR and the one that provides the best fuel utilization and ultimately the lowest fuel cost (when reprocessing is carried out on a sufficiently large scale). Uranium and thorium are segregated as in the previously described cycles. Both types of particles are reprocessed to recover uranium for recycle in HEU fuel. Fission-product wastes are processed for storage in a geologic repository.

Initially, it is anticipated that the HEU fuel would be enriched U-235. Ultimately, another potential source of HEU would be U-233 from breeder thorium blankets.

NASAP has considered the option of spiking HEU fuels with radioactive species to improve their diversion resistance. Spiking would not greatly affect fuel utilization but would substantially increase HEU (U-235) fabrication costs (though mitigated by high burnup), because recycle U-233 fuel contains a high level of gamma activity from the daughters of U-232 decay. This level of radiation may not constitute an adequate proliferation barrier, however.

Plutonium-Thorium, Plutonium Recycle -- The HTGR can be used to efficiently burn plutonium recovered by reprocessing spent LWR fuel. PuO_2 and ThO_2 are contained in separate particles and reprocessed separately. The plutonium is recycled, while the U-233 is recovered from the fertile particles for use in a denatured or HEU-thorium cycle. Fission products are discarded with high-level waste.

The PuO_2 particles in this cycle can be spiked, if necessary, to improve diversion resistance, though this would increase fuel-cycle costs.

1.4.2 Status of Technology

The HTGR has been under development since the early 1950s. Five reactors have been or are being built -- the U.S. Peach Bottom and Fort St. Vrain, the FRG pebble-bed reactors (AVR and THTR), and the UK Dragon. Within the United States, the 330-MWe Fort St. Vrain HTGR, operating with highly enriched uranium-thorium fuel, is in its initial cycle and has operated to 70% power for 200 effective full-power days. A significant feature of the HTGR is that it achieves a high overall thermal efficiency of 40% (relative to 34% for PWRs) by operating with helium coolant at high temperature to produce steam at conditions that permit use of conventional power conversion equipment.

The HTGR in the United States is considered to be at the prototype development stage. Therefore, requirements for additional RD&D comprise principally the development and demonstration of a commercial-scale demonstration plant. Although the performance of the Fort St. Vrain HTGR cannot be assessed fully until after several years of operation, most of the HTGR component technology for large plants has been shown to be practicable.

The most recent HTGR-SC design modifications have stressed plant improvements to increase plant availability and maintainability while attempting to minimize plant capital cost increases. (These modifications have not been factored into the NASAP assessment but would not affect the estimated cost or fuel utilization.) The following design modifications were for a standardized four-loop 2240-MWt design and a six-loop 3360-MWt design with a net plant efficiency of 39.2%:

- eliminated steam reheat in the primary circuit,
- used electric-motor-driven helium circulators instead of steam-driven circulators,
- used containerized fuel storage,
- specified nonsymmetric PCRV,

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- developed integrated-ring PCRV support,
- incorporated core auxiliary heat exchange bayonet tube design instead of helical tube,
- operated at lower core power density.

The technology needed for the HTGR-DC would also be applicable to high-temperature nuclear process heat.

The status of technology required for the development of the HTGR-DC is essentially the same as that of the HTGR-SC, except in the areas noted below.

1. High-temperature materials. The materials necessary for operation in a helium environment at a reactor outlet temperature ranging from 850°C (1562°F) for the reference HTGR-DC to 1000°C for an advanced HTGR-DC require an extensive materials development program. This program is aimed at the development of materials associated with the fuel, the reactor internals, and the turbomachinery hot duct. While the turbomachinery material requirements can be demanding (depending on helium inlet temperatures), note that existing industrial, marine, and aircraft gas-turbine engines are operating in a peaking mode at higher temperatures and in more severe corrosive and erosive environments than the HTGR-DC turbomachinery will. However, they are more accessible for maintenance. Development of high-temperature materials for the hot duct to the turbomachine is demanding and may require novel solutions to find materials that can last for the desired 280,000 hr of operation. Otherwise, designs can be specified that allow for periodic (six-year) removal of turbine blades.

2. Thermal barrier. The HTGR-DC thermal barrier will utilize the same basic technology as the HTGR-SC, but it must be designed (1) to withstand the rapid depressurization rates associated with the operation of the turbomachinery bypass valves and (2) for a postulated turbomachinery deblading. It must also be designed to operate under the sustained acoustic loading caused by the turbomachinery. Extensive development is required in this area.

3. Turbomachinery. The design of the turbomachinery is considered to be within the present state of the art. Existing industrial, marine, and aircraft engines operate under more demanding conditions, such as higher temperature, more contaminated environment, and more foreign object damage. The

areas unique to the turbomachinery development include (1) blade/rotor containment rings, (2) bearing and seal system, (3) primary-system shaft seal, (4) contaminated turbomachinery maintenance techniques, and (5) sound pressure level attenuation.

Sufficient interest has been generated in the status of HTGR technology to warrant the formation of a small utility HTGR commercialization group, Gas-Cooled Reactor Associates (GCRA). The DOE's recent redirection of GCRA's program to develop the HTGR-DC instead of commercializing the HTGR-SC will delay the introduction of HTGR technology into the commercial nuclear power sector by five to ten years, requiring additional material development and the RD&D of a helium gas turbine. However, a strong international HTGR-DC commercialization program (U.S. and the Federal Republic of Germany) could reduce the delay.

1.4.3 Licensing Issues

Although small HTGRs (Peach Bottom I and Fort St. Vrain) have been licensed in the past, several significant issues are not yet resolved in licensing the large HTGRs. Included in these issues is the flow instability problem with the Fort St. Vrain reactor. A relatively simple fuel modification will soon be tested in this reactor in the hope of alleviating the problem. Based on the NRC staff responses to licensing reviews of the steam-cycle HTGR commercial projects for Delmarva Power and Light Company and the Philadelphia Electric Company and the ongoing GASSAR-6 review, the following issues are unsettled: seismic design and analysis, structural design practices for graphite, in-service inspection, emergency cooling, allowable temperatures to limit fuel failure during operation, accident probability and consequences evaluation, fission-product transport for reactor siting evaluation, plant response to anticipated transient without scram events, design verification programs for prototype components, oxidation of core support structure, preoperational vibration assessment for reactor intervals, and design of the radial flow reheater in the steam generators. Additional safety/licensing issues that may have to be treated result from design/cost improvements: fuel-particle coating and fuel element, primary-coolant system, unlined containment, and PCRIV penetration and closures.

In addition to HTGR generic issues, the HTGR-DC has a number of features that lead to some new licensing questions, as mentioned in Section 1.4.4. The most significant of these are discussed below.

Other licensing concerns of the HTGR-DC result from the higher pressures and temperatures required relative to the HTGR-SC. Fission-gas releases and primary-circuit contamination may be higher if the fuel is not improved. Bypass control valves and pressure relief valves require faster actuation times but under more extreme conditions. The major licensing concern will be the proximity and integration of a turbomachine within the PCR. V.

However, a unique aspect of the HTGR-DC is the potential for classification of major heat transfer components (recuperator and pre-cooler) as nonnuclear components.

1.4.4 Safety Issues

The HTGR has inherent characteristics that make the thermal response of the core to transients very slow and also limit the potential for release during low-probability maximum credible accidents. These characteristics include a large core heat capacity, low power density, exclusive use of refractory materials for the core, a single-phase inert gas with a low stored energy content as the primary coolant, and a structurally redundant concrete reactor vessel that encloses the entire primary coolant system. Accidents involving core meltdown or total loss of coolant are inherently unlikely. Although accidents involving overheating of the core or leakage of coolant can be postulated, the releases associated with such accidents are limited to a small fraction of the core inventory and occur on the time scale of days to weeks -- allowing time for mitigating actions and evacuation of the nearby population.

The factors that make potential accidents low in consequence suggest enhanced siting flexibility. This is an important consideration for nuclear process heat, the economics of which are sensitive to assumptions regarding reactor siting.

One safety issue for the HTGR steam cycle is the possibility of a massive leak of water into the reactor followed by reaction of the steam with core graphite, thus producing potentially explosive hydrogen and oxygen.

The HTGR-DC has a number of safety features in common with the HTGR-SC. The principal ones are the use of the PCR, the prismatic graphite core with encapsulated fuel particles, and the use of three independent auxiliary cooling loops. Safety features such as the control-rod reserve shutdown system and liner cooling system are essentially identical.

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The HTGR-DC control and protection systems are significantly different from those of the HTGR-SC, but the increased simplicity of the gas-turbine concept may lead to enhanced overall plant safety.

The systems safety philosophy on which the HTGR has been based is formulated in a way that makes it applicable to both the SC and DC designs. Of course, safety-related design criteria for individual components such as recuperator, precooler, turbomachines, PCRV shaft seals, and other equipment unique to the DC may be different from those of any nuclear plant design proposed to date.

In addition to HTGR generic issues, the HTGR-DC has a number of features that lead to some new safety questions. The most significant of these are discussed below.

1. Shaft seal failure. The turbomachine/generator shaft penetrates the primary coolant system boundary. Failure of the seal due to machine or shaft malfunction can potentially cause a rapid depressurization to the containment. Design features must be incorporated which ensure that such accidents have an acceptably low probability of occurrence.

2. Internal pressure equilibration accidents. Failure of internal components such as the turbomachines or recuperators can cause rapid pressure equilibration within the PCRV. These pressure-pulse transients are much more severe than those associated with the most rapid postulated PCRV depressurization for the HTGR-SC. The DC pressure equilibration accidents place stringent design requirements on PCRV internals and dictate component designs that may be different from those of the HTGR-SC.

3. Turbomachine failures. In addition to causing rapid pressure transients, turbomachine failures can create missiles from which protection must be provided. The steam cycle also has the potential for internal missiles generated by circulator failures, but the magnitude of the missile problem for the DC is larger.

The water ingress problem, graphite oxidation, and core support oxidation concerns of the HTGR-SC are effectively eliminated by the HTGR-DC design.

1.4.5 Environmental Issues

The radiological impacts of the HTGR may be compared with LWR impacts as follows:

- Smaller amounts of gaseous and liquid wastes are generated. However, there are large quantities of spent fuel and reflector blocks.
- Lesser environmental dilutions are required for the gaseous (100 times) and liquid (10 times) wastes.
- C-14 production is higher, and disposal of C-14 must be provided for in waste treatment.

In general, the greater efficiency of the HTGR (about 40 to 47% for binary cycle vs. about 33%) results in less nuclear fuel consumption, less generation of nuclear wastes, and less rejection of waste heat per kilowatt-hour of electricity produced relative to current-technology LWRs.

Comparative studies have shown the HTGR-DC concept to have the following characteristics relative to LWRs:

- a 25 to 30% lesser effect on water consumption and biota,
- a lesser effect on water quality and atmosphere because only 78% as much waste heat is rejected,
- a smaller loss of plant efficiency when using a dry cooling system.

1.4.6 RD&D Needs

High-Temperature Gas-Cooled Reactor -- Steam Cycle

Experience in HTGR technology has been gained through both national and international programs. The HTGR has evolved through the Peach Bottom and

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Fort St. Vrain demonstration plants and the designs prepared for large commercial plants. In addition, direct fuel-design and operational experience has been gained from the United Kingdom program (primarily Dragon) and the Federal Republic of Germany AVR/THTR program. Also, a formal R&D program has been ongoing with Commissariat à l'Energie Atomique (CEA) in Saclay, France, for developing fuel and component scale test data for inclusion in the United States HTGR program.

The Peach Bottom plant operated satisfactorily for seven years before shutdown, with an ongoing end-of-life surveillance program currently providing feedback to the HTGR R&D programs. The data being generated through this base program include materials data, primarily from steam generator tubes and other component samples and from fuel element and fuel test element postirradiation examination. Graphite performance in terms of irradiation stability and oxidation resistance are main areas of investigation.

The Fort St. Vrain plant, which is currently in its approach to full power, has provided significant technology directly applicable to the HTGR plant design in construction, fabrication, and operation. For example, considerable difficulty was encountered on the Fort St. Vrain steam generators in fitting the outer shrouds to the heat transfer bundles without interference between the shroud and tubes. (This is a problem common to most large helically coiled steam generators.) For the lead plant, increased outer shroud clearances were obtained by modifications to tube design.

Experience in areas such as the refueling penetrations and other major penetrations through the thermal barrier has provided information on the complex heat transfer mechanisms in these areas. In particular, modifications to the control-rod drive assembly were made to reduce a bypass flow which resulted in hot helium reaching the refueling penetration liner. Detailed analyses have been made to check various hot spots adjacent to penetrations and to confirm the local concrete stress levels.

Experience with the circulator service system involving accidental injection of bearing water into the primary coolant and inadvertent circulator trips under transient and low-power conditions has provided valuable system performance data on which to improve the lead plant design. Improvements have been made by providing a core auxiliary cooling system independent of the main helium circulators, thus reducing the complexity of the service system.

In the performance area, the preoperational tests and power operation to date have provided data that have been incorporated into the lead plant design.

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In particular, in the overall plant performance area, much has been learned regarding operation at part load and under transient conditions, the relationship between controls of the reactor system and the plant secondary systems, and the responses to trip conditions. Other data have been gathered which have affected the design and/or development programs for the PCRV, control-rod drives, neutron detection instruments, region flow control systems, fuel-handling system, primary-coolant system, main circulator service system, helium purification system, plant protection system, the overall plant control system, and data acquisition and processing system.

As identified above, the Peach Bottom end-of-life program and the Fort St. Vrain surveillance program, along with data from cooperative programs with other countries, will provide a large amount of data to be incorporated in the HTGR plant development program. Nevertheless, a significant amount of development is still required to realize a commercially feasible system from the present technological status of the HTGR. Additionally, all of the design developments remain to be verified during program demonstrations.

Items requiring further technological development in the overall plant include steam generator and core auxiliary heat exchanger, instrumentation and control, PCRV (including liners, closures, and penetrations), reactor internals and thermal barrier, safety and reliability, systems engineering, components, and materials. In the reactor core area, the program requirements include fuel (particularly MEU-thorium and graphite) development, fission-product/coolant chemistry, and fresh-fuel process development.

Table 25 provides a summary of the HTGR steam-cycle reactor plant components/systems technology requirements. This table additionally indicates the areas in which design and developmental work must be performed.

Table 26 lists the various projects that must be accomplished prior to operation of the first commercial plant. The costs associated with these projects include both design and construction and RD&D where applicable.

Table 27 lists the time frame for the various projects that must be accomplished before operation of the first commercial plant.

High-Temperature Gas-Cooled Reactor -- Direct Cycle

The RD&D needs unique to the HTGR-DC are noted below.

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Table 25. HTGR Advanced Technological Requirements

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

^aFor medium-enriched fuel cycle.

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Table 26. HTGR-SC RD&D Costs^a

RD&D area	Cost (\$ million) ^b
Reactor plant R&D	400-500
Plant demonstration	
Proof-of-principle plant	N.R. ^c
Demonstration plant	N.R.
Lead plant	1300-1500
NSSS vendor support	100-150
Increment for multiple vendors	350-500
Total reactor RD&D	2150-2650

^a Does not include initial fuel costs, which are estimated to be \$130 to 150 million (see Volume IV).

^b January 1978 \$.

^c N.R. = not required.

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Table 27. HTGR-SC RD&D Schedule

RD&D activity area	Time frame (years)
Reactor plant R&D	1979-1989
Plant demonstration	
Proof-of-principle plant	N.R. ^a
Start design	
Construction permit	
Operating license	
Demonstration plant	N.R.
Start design	
Construction permit	
Operating license	
Lead plant	
Start design	1981
Construction permit	1986
Operating license	1993
First commercial plant	
Start design	1991
Construction permit	1995
Operating license	2001

^aN.R. = not required.

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1. Reactor vessel. The reactor vessel for the HTGR-DC plant bears a close resemblance to the steam-cycle HTGR PCRV. The reactor core cavity is centralized in the PCRV, and the vertically positioned heat exchangers are installed in side-wall cavities (pods). While the technology is contemporary, the vessel geometry configuration in the PCRV bottom head is different for the HTGR-DC because horizontal cavities are necessary for the turbomachines. In the vicinity of the turbomachine cavities, horizontal cross tendons are necessary in addition to the vertical tendons. Liner and closure features are almost identical with the steam-cycle HTGR, but modifications to the thermal barrier are necessary, based on rapid depressurization rates. The operating pressure is higher than for the steam cycle (although less than for the GCFR), and this, in conjunction with the previously mentioned geometrical differences, necessitates model testing.

2. Reactor vessel internals. The reactor internals (which include shielding, ducting, control-rod drives, baffles, etc.) are classed in a category requiring modest improvement in performance or size from present (steam-cycle HTGR) knowledge. The reactor internals bear a very close similarity to the steam-cycle plant, but the DC control rods and drives are affected by the thicker PCRV top head (resulting from the higher operating pressure). The higher reactor inlet temperature of the HTGR-DC will affect the design of the reactor internals and their material selection.

3. Primary-system heat exchangers. The technology for the HTGR-DC heat exchangers is regarded as contemporary. The operating environments (temperature and internal pressure differential) are less severe than for the steam cycle, and existing code-approved alloys are utilized. The precooler operates with a maximum metal temperature below 205°C; thus, it is free from creep effects. The large surface area requirements necessitate compact surface geometries, but the tubular surface geometries and fabrication methods are regarded as state-of-the-art technology. Large tubular units of the types selected have been built in Europe and operated successfully in fossil-fueled closed-cycle gas-turbine plants.

4. Other plant accident mitigating systems. In the area of plant protection systems, there are noticeable differences compared with steam-cycle HTGRs. To prevent turbine overspeed, for example, a compressor bypass valve is necessary, and this and other valves in the primary system are used for plant control and protection. The plant protection system is regarded as requiring a modest improvement over that of the steam cycle. A pressure relief valve from the high-pressure helium system is not believed to be necessary in the HTGR-DC plant because the relief function can be done internally within the primary circuit.

5. Helium gas turbine. The helium turbine is a unique component for the DC plant, but its development is considered within the state of the art. All of the aerodynamic, thermodynamic, and structural technology from open-cycle industrial gas-turbine practice is applicable. The turbine inlet temperature of 850°C is modest compared with current industrial gas turbines, and this permits the utilization of existing nickel-base alloys (uncooled blades). Areas in the turbomachine requiring extensive development are the bearings and seals. Helium turbomachines have been built and operated in Europe with rotor sizes of 300 MWe equivalent rating. (Because of the lower operating pressure of the fossil-fueled Oberhausen 2 helium turbine power plant, the actual output is 50 MWe. The high helium volumetric flow in this plant results in component sizes representative of a nuclear gas turbine with a rating of about 300 MWe.)

6. Control valves. As mentioned above, the plant control and protection system does embody helium bypass valves. These valves are integrated into the primary systems and are installed in cavities in the top head of the PCR. Three (of the four) valves operate at elevated temperature (reactor inlet gas) in a dry helium environment. While large, these valves are amenable to state-of-the-art design and fabrication technologies. Because of their important role in the plant control system, an extensive development is planned to ensure a high degree of reliability.

Table 28 lists the various projects that must be accomplished prior to operation of the first commercial plant, assuming a program that is not incremental to the HTGR-SC. Rather, it is a "stand-alone" program.

Table 29 gives the time frame for the various projects that must be accomplished before operation of the first commercial plant, once again assuming a stand-alone program.

1.4.7 Perceived Advantages and Disadvantages

The HTGR is a technologically attractive system that does suffer from uncertainties in cost, licensability, and reliability relative to LWRs. Commercial introduction of the HTGR would require a lengthy and expensive RD&D program and would expose industrial participants to greater risk than a comparable program of evolutionary improvements to LWRs. The fuel-cycle technology is unique and would require an RD&D program relatively independent of water and breeder fuel development.

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Table 28. HTGR-DC RD&D Costs^a

RD&D area	Cost (\$ million) ^b
Reactor plant R&D	600-800
Plant demonstration	
Proof-of-principle plant	N.R. ^c
Demonstration plant	1500-1900
Lead plant	1600-2000
NSSS vendor support	100-150
Increment for multiple vendors	350-500
Total reactor RD&D	4150-5350

^aDoes not include initial fuel costs, which are estimated to be \$260 to \$300 million (see Volume IV).

^bJanuary 1978 \$.

^cN.R. = not required.

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Table 29. HTGR-DC RD&D Schedule

RD&D activity area	Time frame (years)
Reactor plant R&D	1979-2005
Plant demonstration	
Proof-of-principle plant	N.R. ^a
Start design	
Construction permit	
Operating license	
Demonstration plant	
Start design	1980
Construction permit	1985
Operating license	1992
Lead plant	
Start design	1990
Construction permit	1995
Operating license	2002
First commercial plant	
Start design	2004
Construction permit	2008
Operating license	2014

^aN.R. = not required.

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The HTGR technology appears to provide the possibility of efficient U_3O_8 utilization and competitive costs for U_3O_8 prices above \$100/lb. Specific advantages and disadvantages of the HTGR are outlined below.

The HTGR has excellent inherent safety characteristics in the high-melting-point graphite moderator and ceramic fuel, contained within a PCRV. The bulky fuel elements generate only small amounts of plutonium in the reference once-through fuel cycle. The high thermal efficiency of the cycle minimizes thermal impacts on the environment. The direct-cycle alternative has even higher thermal efficiency than the steam cycle, lends itself to coupling with dry cooling towers (attractive for water-deficient sites), and potentially has low capital costs.

In spite of the technological attractiveness of the HTGR, there is no commercial base for supply of these systems or for the supply of fuel-cycle services on a large scale. The uncertainties in economics create a high-risk situation. The direct-cycle alternative is a novel design at an early stage of development and, consequently, involves high risks in early plants.

1.5 GAS-COOLED FAST BREEDER REACTOR

1.5.1 Reactor/Fuel-Cycle System Description

The gas-cooled fast breeder reactor (GCFR) concept being developed in the United States embodies extensions of many of the principles of the current designs of HTGRs, with the general exception of the fuel and mechanical systems related to the fuel. The present GCFR fuel concept is an adaptation of the mixed-oxide fuel being developed for the LMFBR. The significant features of this reactor are the high-pressure helium cooling, the venting of the fuel to a cleanup system, and the direct heat exchange of the helium to generate steam without an intermediate heat exchanger, as in the LMFBR. The components of the reactor coolant system are arranged within a multicavity PCRV. The helium flows through the steam generator, producing "uphill" boiling. From the steam generator, the helium flows to the core cavity, down through the core grid plate, down over the fuel, and back to the steam generator cavity. In this NASAP reference "downflow" core design, the refueling occurs through penetrations in the bottom of the PCRV, and the control rods are inserted in the top of the core. Figure 14 shows this downflow core configuration for a 300-MWe demonstration plant.

Coolant flow in a commercial-sized GCFR is assured by six electric-driven circulators with tandem-mounted pony motors. The use of electric-driven circulators eases control of the reactor during operation at steady state, start-up, shutdown, and transients and accidents. In addition, there are three auxiliary helium circulators and heat exchangers to remove core decay heat.

A recent program decision has been made to adopt an upflow core configuration for the GCFR reference design. The major incentives for considering the upflow coolant configuration included

- the potential for core cooling by natural convection, thus increasing residual heat removal reliability and diversity;
- opportunity for significant plant design improvements, especially in the areas of refueling and main helium circulator location.

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Figure 14. Downflow 300-MWe GCFR Nuclear Steam Supply System.

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In this reference upflow design, helium is pumped through the reactor core by the electric-motor-driven main circulators at the bottom of the steam generator cavities. The helium coolant flows upward through the core and then across to the steam generator cavity. It flows downward through the helical-tube-bundle steam generators and continues downward to the circulator inlets. Figure 15 shows this configuration.

This design provides for pressurized natural convection cooling by properly sizing the elevations of the core auxiliary heat exchanger (CAHE) and the air/water ultimate heat dump in relation to each other and to the reactor core. Figure 16 presents an elevation view of the general location of the above components.

The steam cycle and the approximate operating conditions for the GCFR are shown in Figure 17.

The reference fuel consists of uranium-plutonium oxide pellets within stainless steel rods, with the fuel rods vented to a fission-product collection system through the pressure equalization system. The cladding is roughened over most of its length to augment heat transfer. Figure 18 shows the upflow fuel-rod assembly. Possible alternative fuel forms (carbide and metal) and alternative fuel/blanket configurations, as discussed for LMFBRs, are also applicable to the GCFR. However, other cladding materials can be used in GCFRs, because there is no chemical corrosion caused by sodium; that is, ferritic alloys can be used in the GCFR for additional improvements.

The reactor operates on an approximately annual refueling scheme in which one-third of the core is replaced. The core charge is uranium dioxide/plutonium dioxide, with an average fissile enrichment of about 14% (total fissile plutonium per unit total heavy-metal content) and either natural or depleted uranium dioxide. Core discharge consists of uranium dioxide, plutonium dioxide, and fission products. Fuel is shipped in the form of spent assemblies and processed by a PUREX process similar to that used for LMFBR fuel. The uranium dioxide/plutonium products are conveyed to a fabrication plant and reconstituted into fresh assemblies with outside makeup. Depending on the fuel-cycle scenario, either the fresh fuel is shipped directly to the reactor, or it is first spiked for diversion resistance by adding radioactive material or by preirradiation and then shipped to the reactor.

The blanket charge in the reference design (radial and axial) is thorium dioxide. Discharge blanket assemblies contain thorium dioxide, U-233

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Figure 15. Upflow 300-MWe GCFR Nuclear Steam Supply System.

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Figure 16. GCFR General Arrangement.

Figure 17. Main Loop Cooling System.

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Figure 18. GCFR Fuel Rod Assembly.

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dioxide, and fission products. Processing is done by THOREX, which yields separated and decontaminated thorium dioxide and U-233 dioxide. The latter is denatured with U-238 dioxide and sold for thermal-reactor usage.

The radial blanket consists of three rows: Row 1 contains 60 assemblies, Row 2 contains 66 assemblies, and Row 3 contains 72 assemblies. Fuel assemblies are shuffled inward during refueling operations. The axial blanket is an integral part of the fuel element and is replaced with the element.

Table 30 shows the reactor performance characteristics.

In addition to the reference $\text{PuO}_2\text{-UO}_2$ GCFR core and ThO_2 axial and radial blanket fuel cycle, there are other fuel-cycle options: (1) PuO_2/UO_2 core and UO_2 axial and radial blanket and (2) $\text{PuO}_2/\text{ThO}_2$ core and ThO_2 axial and radial blanket. The bred U-233 is recycled to thermal converter systems with the makeup plutonium obtained from reprocessed thermal converter reactor spent fuel.

Typical fissile performance characteristics of three GCFR fuel cycles are given in Table 31, normalized to 1000 MWe and 70% capacity factor. The breeding ratios are somewhat higher than those of the LMFBR.

1.5.2 Status of Technology

A substantial amount of work has already been completed on materials and methods development that is applicable to both the HTGR and GCFR programs. Some additional development work is required for the GCFR NSSS because of increased primary-coolant (helium) operating pressures (9.0 vs. 5.1 MPa for the HTGR-SC) and differences in reactor internals and core arrangement, though the operating helium temperatures for the GCFR are lower than those for the HTGR (540 vs. 745°C). The lower operating temperatures permit a wider latitude in the selection of materials.

The GCFR was originally developed because it offers the potential for superior breeding performance, resulting from the hard neutron spectrum, and lower capital costs because there is no liquid-metal handling and the intermediate loop is eliminated. As a fast breeder, it was designed to utilize as much as possible the technologies developed for the LMFBR program, such as fuel and plant components. Because it is a gas-cooled reactor, it draws from the developed technology from related gas-cooled programs, especially that

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Table 30. GCFR Performance Specifications:
Pu-U Core, Th Axial and Radial
Blankets

Reactor thermal power, MW	3,287
Gross electrical power, MW	1,320 ^a
Net electrical power, MW	1,200
Thermal efficiency, %	36.5
Capacity factor, %	75
Core volume, liters	14,606
Core power density, MWt/liter	0.211
Core height, m	1.27
Number of enrichment zones	4
Coolant flow rate, Mg/sec (lb/sec)	2.468 (5441)
Reactor inlet temperature, °C	316
Reactor outlet temperature, °C	575
Primary system pressure, MPa (psi)	8.98 (1302)
Total primary system ΔP, MPa (psi)	0.386 (56)
Pin diameter, cm	0.74
Number of fuel elements	253 ^b
Average fuel temperature, °C	1,281
Maximum fuel temperature, °C	2,328
Maximum cladding temperature, °C	750
Initial inventory, kg of heavy metal	153,824
Fissile plutonium loading (239 + 241), kg	4,440
Fissile uranium loading (235), kg	68.4
Fertile uranium loading (238), kg	27,756
Fertile thorium loading (232), kg	121,560
Average discharge exposure, MWd/tHM	81,000
Peak discharge exposure, MWd/tHM	92,000
Breeding ratio, beginning of life	1.51
Breeding ratio, average over equilibrium cycle	1.41

^aElectric-driven circulators assumed.

^bIncludes 31 partially fueled assemblies that have central control rods.

Table 31. GCFR Fissile Performance Characteristics^a

	Breeding ratio	Initial core, plutonium fissile (kg/GWe)	Annual charge equilibrium, plutonium fissile (kg/GWe-year)	Annual discharge at equilibrium (kg/GWe-year)	
				Plutonium fissile	U-233
PuO ₂ and UO ₂ core, ThO ₂ blankets	1.51	4440	1512	1420	440
PuO ₂ and UO ₂ core, UO ₂ blankets	1.54	4288	1452	1910	
PuO ₂ and ThO ₂ core, ThO ₂ blankets	1.35	7088	2329	1654	1041

^aNormalized to 1000 MWe.

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being proved at the Fort St. Vrain reactor. This technology covers the basic technical aspects of the PCRV, helium circulator, steam generator, helium-purification system, core auxiliary cooling systems, containment systems, and a substantial portion of the nuclear plant systems. The primary technological benefit of the HTGR program is expected to be the proof-of-principle experiment presently being conducted at Fort St. Vrain. In addition, relevant information will be exchanged with a number of European organizations in the Federal Republic of Germany and Switzerland which are involved in technology development programs for the PCRV, reactor internals, fuel, core design, and safety-related studies.

Although an attempt is made to carry over as much technology from the LMFBR and the steam-cycle HTGR as possible, there are some features and components that are unique to the GCFR. These may be summarized as follows:

1. core support grid plate and internal shielding,
2. bottom fuel handling (for downflow core concept only),
3. control and instrumentation at a high fast flux,
4. main circulator safety features,
5. central cavity closure,
6. thermal-hydraulic behavior of roughened cladding,
7. performance of vented fuel, with pressure-equalization system,
8. fuel-pellet--cladding interactions,
9. inspection and repair of primary system.

These features are addressed by development programs designed to be complementary to the existing development programs, as shown below.

Fuel Development Status

While the oxide fuels for the GCFR draw heavily on the existing LMFBR fuel development program, the unique (to GCFR) aspects are the subject of several fuels and materials R&D efforts. These have included creep rupture and environmental testing of ribbed (roughened) cladding, irradiation tests of GCFR-type fuel rods at burnups of 121,000 MWd/t in the EBR-II reactor, irradiation of a prototypical GCFR fuel rod to investigate the effects of a step change in linear rod power (currently under way), thermal flux irradiations of two vented fuel rods to demonstrate the feasibility of pressure-equalized and vented fuel, and irradiation of a 12-rod pressure-equalized vented helium-cooled mixed-oxide-fueled bundle with ribbed cladding to a burnup of 78,000 MWd/t in a prototypical helium environment in the BR-2 reactor in Mol, Belgium.

Components Development Status

Components and systems R&D development programs to date have been directed toward (1) establishment of comprehensive development plans, (2) establishment of a design for a circulator test facility for full-scale circulator testing and qualification, (3) testing of scale models of PCRV closures, (4) investigation of materials properties for steam generator tubing and out-of-core structural components, and (5) development of computer codes which predict steady-state and transient behavior of the GCFR for both normal and postulated accident conditions.

Facilities which are planned for support of the GCFR development include the following:

The Core Flow Test Loop (CFTL), to test electrically heated models of GCFR core assemblies in both normal and accident conditions. The CFTL facility, which will be located at ORNL, is currently in Title II design. The design of the first test bundle has been completed. Computer code methods development in support of fuel assembly development has included adaptation of LMFBR codes to the GCFR and generation of new codes. These codes are used for thermal-hydraulic analysis and for statistical analysis of hot-channel performance.

The GRIST-II (Gas-Reactor In-Pile Safety Test) is aimed at demonstrating fuel melting, vaporization, and relocation phenomena which could possibly be encountered in core accident transients. The experiment is located at the

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ANL-TREAT reactor facility at Idaho Falls, Idaho, and is currently in Title I design with ANL and EG&G participants.

The Low Power Safety Experiment (LPSE) has as its objective determination of the limiting characteristics of the GCFR core assemblies by investigating the physical phenomena of low-power/low-flow accident conditions up to and including conditions resulting in fuel cladding failure. The experiment is under way at the Los Alamos Scientific Laboratory.

An assessment of required development programs beyond those planned for the demonstration plant is as follows:

1. Prestressed-concrete reactor vessel (PCRV). Additional loops are required for the commercial-plant PCRV. Verifying the PCRV analytical calculations with additional scale-model tests is necessary, although the basic technology for the demonstration plant is applicable to the commercial plant.
2. Core support structure. The technology for the demonstration plant can probably be extended to the commercial plant. No additional development is expected.
3. Core locking and reactivity control mechanism. The same mechanical principles are used for the demonstration and commercial plants, and no additional development is expected.
4. Core shielding. The neutron fluxes at the interface between blanket and shield are expected to be similar for the demonstration and for the commercial plants. Additional shielding experiments may be needed to explicitly model the configuration for the larger core and shielding layout.
5. Main helium circulator. The pressures and temperatures for the demonstration and commercial plants are almost the same. However, the flow rates and associated horsepower ratings are significantly different, and this fact may make extensive full-power testing and demonstration necessary, depending on the degree of extrapolation required.
6. Steam generator. The development tasks for the demonstration plant are assessed to be adequate for commercial-plant application, except for low-flow

boiling stability and gas flow distribution, which will require additional development.

7. Core auxiliary cooling system (CACS). The component technology for the demonstration plant is adequate for the commercial plant. The number of CACS loops in the commercial plant (three or six) has not been chosen yet, and the nature of additional development work depends on the choice of the number of loops. If a three-loop concept is chosen, larger components need to be developed; a revised PCRV design is necessary for a six-loop concept.

8. Refueling system. The basic technologies for the demonstration and commercial plants are similar. However, a scaled-up configuration and geometry may require additional development efforts in the design of fuel-handling equipment such as in-vessel machines, control systems, and casks.

9. Other methodologies. Physics calculations and safety evaluations are assessed to be the same for the demonstration and commercial plants. Radiological impacts of scaled-up configurations should be evaluated for the commercial plant.

Table 32 summarizes the requirements for technological advances in GCFR plant components.

To further accelerate GCFR RD&D and define the utility requirements for a commercial plant, a significant number of utilities have formed a GCFR RD&D group called Helium Breeders Associates (HBA).

1.5.3 Licensing Issues

Since 1974, General Atomic Company (GAC) has been developing information relative to licensing concerns expressed in the "pre-application" review. Progress on this topic was briefly presented to ACRS and the NRC staff in July 1977, but NRC has no present plans to review this newly available information.

Based on the licensing concerns of the Nuclear Regulatory Commission and ACRS, General Atomic and other program participants have pursued a program to resolve the outstanding safety/licensing issues as discussed in the following paragraphs.

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Table 32. GCFR Technological Advance Requirements

Plant component	Technological advance requirement					
	None	Modified configuration/ application	Modest improvements in performance or size	Modest improvement in performance or size and modified configuration/ application	Major improvement in performance or size	Major improvement in performance or size and modified configuration/ application
Nuclear fuel				X		
Reactivity control systems	X					
Reactor vessel			X			
Core support structure				X		
Reactor vessel internals, including shielding, ducting, control-rod guides, and baffles		X				
Primary-coolant pumps and auxiliary systems				X		
Primary-coolant chemistry/radiochemistry control		X				
Primary-system heat exchangers			X			
Reactor instrumentation				X		
Emergency core-cooling/safety shutdown systems		X				
Containment, containment cleanup systems, and effluent control systems		X				
Other accident-mitigating systems (i.e., plant protection systems)		X				X ^a
On-site fuel-handling, storage, and shipping equipment				X		
Main turbine		X				
Other critical components, if any						
Balance-of-plant components	X					

^aPostaccident fuel containment system only - if required.

Other fast and thermal reactor systems can provide built-in thermal capacity and/or natural-circulation cooling characteristics, which eliminate the need for rapid restoration of forced circulation of primary coolant to prevent fuel overheating. Studies are continuing on other GCFR design arrangements that provide sufficient natural-circulation cooling if coolant pressure is maintained. As previously mentioned, an "upflow" core design has been developed.

The licensability of the GCFR is not well established, according to NRC. Some of the concerns regarding licensing feasibility are core-cooling reliability, containment design bases, core-disruptive accidents including PCRV response, reliability of shutdown systems, in-service inspection, reactor vessel failure mechanism, and neutron streaming within the reactor vessel. All major issues will be faced in the licensing review of the GCFR demonstration plant.

1.5.4 Safety Issues

The conditions identified in the NRC staff safety evaluation of the GCFR were specific in many areas, and their satisfactory resolution could cause substantial design changes. The major safety considerations that were considered unresolved were:

1. acceptable power density levels and thermal margins,
2. definition of depressurization accidents,
3. definition of core-disruptive accidents,
4. diversity in reactor shutdown system,
5. adequacy of core-cooling systems,
6. containment design bases for combustible gas, plutonium aerosols, core-disruptive accidents, and postaccident heat removal.

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An experimental and analytical program is under way in the United States and in Europe to address these considerations.

Note that the GCFR concept has safety advantages of a strong vessel (PCRV) for containment of accidents which is the equivalent of a core catcher. It does not have large positive void coefficients characteristic of present LMFBR designs. Studies of the commercial-sized GCFRs show that their overall safety characteristics are expected to improve as plant size increases. Reactivity and kinetics parameters of GCFR cores ranging from 300 to 1500 MWe have been evaluated; results indicate that, as plant size increases, the Doppler coefficient becomes more strongly negative and fuel, cladding, and coolant specific worths decrease significantly, so that whole-core cladding and coolant worths used for the 1500-MWe cores are only slightly higher than those for the 300 MWe core.

The development of the "upflow" core design will enhance the safety characteristics of the GCFR and also improve the operability of the reactor. Also, because penetrations in the bottom of the PCRV are eliminated, consequences resulting from a hypothetical core-disruptive accident (HCDA) are mitigated. Instead, the fuel handling will be through the top of the PCRV.

1.5.5 Environmental Issues

The GCFR environmental effects are similar to those described for the LMFBR. However, the vented fuel design greatly reduces releases of noble gases and metallic fission products to the containment and eventually the environment. In combination, the vented fuel cleanup system and the primary helium purification system virtually eliminate release of radioactivity to the environment including hard-to-handle tritium. Because of the inert nature of the helium, separation of radioactive material from the primary coolant is greatly facilitated. Because of the reduced inventory of fission products in the fuel, release of fission products from the fuel during an HCDA is greatly reduced.

1.5.6 RD&D Needs

Although reliance is placed on HTGR and LMFBR technology, the technology and experience (including safety experiments) are not as well developed for the GCFR as for the LMFBR. If the GCFR concept is to be exploited, it is likely that the first significant step would be the construction and operation of a proof-of-principle demonstration plant of about 300 MWe with components that

are half commercial size. Some thought is being given to demonstrating full-sized components in a two-loop 400-MWe plant. The experience obtained from this plant would permit more confident projections to be made for a commercial-size plant. If this route were followed, it is reasonable to expect that construction of a commercial-size demonstration plant could not begin before the 1990s. The present developer of the GCFR concept, GAC, is continuing with GCFR design and is working in association with a utility group (HBA) to develop a demonstration plant for possible construction in the 1980s. HBA is charged with managing and further accelerating the national GCFR program.

The current GCFR concept, being investigated by GAC, is based on the uranium-plutonium fuel cycle, but with thorium blankets to breed U-233. An alternative would contain uranium blankets, for which there is more experience derived from the LMFBR program.

Table 33, GCFR RD&D costs, lists the various projects that must be accomplished before operation of the first commercial plant, and Table 34 gives the time frame for the various projects that must be accomplished before operation of the first commercial plant. These costs and schedules assume that a three-plant scenario is necessary for the commercialization of the GCFR. A two-plant scenario, such as mentioned earlier, would considerably reduce RD&D costs by eliminating the capital costs of the second plant and associated RD&D. This scenario would also permit earlier deployment.

1.5.7 Perceived Advantages and Disadvantages

Advantages of the GCFR, as compared with the LMFBR, include a somewhat higher attainable breeding ratio, a potential for lower plant capital costs, and the possibility of higher plant availabilities resulting from the use of inert helium coolant and the elimination of intermediate loop components. The utilization of HTGR component development and LMFBR fuels development experience could shorten the development time. From a safety viewpoint, there is very little helium reactivity effect, and the helium is chemically inert. Also, there is no need of an intermediate heat exchanger loop. While there is a potential for steam leaks into the primary system, experiments in critical-experiment facilities have indicated thus far that this may not be a serious problem. The major disadvantage of the GCFR relative to the LMFBR is that little experience has been gained on the concept and no reactor of this type has been operated. Consequently, there is uncertainty as to whether the advantages would be realized in practice. Other disadvantages are the needs to contain high-pressure helium and to provide ensured cooling for the core at all times.

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Table 33. GCFR RD&D Costs^{a, b}

RD&D area	Cost (\$ million) ^c
Reactor plant R&D	500-1,000
Plant demonstration	
Proof-of-principle demonstration plant #1	1,100-1,400
Plant #2 (900 MWe)	1,500-2,100
Lead plant	1,600-2,200
NSSS vendor support	100-150
Increment for multiple vendors	400-800
Total reactor RD&D	5,200-7,650

^aDoes not include initial fuel costs, which are estimated to be \$550-600 million.

^bAssumes continuation of major portions of the LMFBR program, costs of which are not included here.

^cJanuary 1978 \$.

Table 34. GCFR RD&D Schedule

RD&D activity area	Time frame (years)
Reactor plant R&D	1980-1990
Plant demonstration	
Demonstration plant #1	
Start design	1980
Construction permit	1985
Operating license	1992
Plant #2	
Start design	1990
Construction permit	1995
Operating license	2002
Lead plant	
Start design	2000
Construction permit	2005
Operating license	2012
First commercial plant	
Start design	2014
Construction permit	2019
Operating license	2024

1.6 LIQUID-METAL FAST BREEDER REACTOR

1.6.1 Reactor/Fuel-Cycle System Description

The liquid-metal fast breeder reactor (LMFBR) is a concept that uses a liquid metal (sodium) to remove heat from the core, has little neutron moderation, and, therefore, operates with high-energy (fast) neutrons. Compared with the thermal spectrum, the fast-neutron spectrum results in higher neutron yields per fission (thus making additional neutrons available for absorption in fertile material) and provides some additional direct fission of fertile material. The LMFBR creates about 10 to 40% more fissile material than it consumes, employing either plutonium or U-233 as the fissile fuel and either U-238 or Th-232 as fertile materials.

Because the operating temperature (480 to 650°C) of the reactor coolant is well below its boiling temperature (about 940°C at atmospheric pressure), the primary system in present plant designs is not pressurized. The reactor operates at temperatures well above those characteristic of LWRs (290°C) and can produce steam that is comparable in temperature and pressure with that produced by fossil-fueled boilers. As a result, the thermal efficiency is higher (about 36 to 40%) than that of LWRs (about 33 to 34%).

The primary sodium circuit of the LMFBR is placed within a reactor-containment building for isolation. The major varieties of the reactor system include those in which the plant is configured into a set of major components connected by piping, known as the loop-type reactor (Figure 19), and those in which these components are placed within a single vessel, known as the pool-type reactor (Figure 20). Both operate near atmospheric pressure, so that the pipe-break problem that dominates LWR safety is somewhat alleviated.

Description of Proliferation-Resistant LMFBR Core Design Study

Recent concerns raised over the potential for nuclear weapons proliferation have resulted in an assessment of proliferation-resistant fueling options for fast breeder reactors. The Proliferation-Resistant LMFBR Core Design Study (PRLCDS) was initiated by the Division of Reactor Research and Technology, U.S. Department of Energy, in October 1977. The objective of the study was to establish the proliferation resistance and performance potential of a wide range of sodium-cooled fast-reactor core designs.

Figure 19. Loop Type LMFBFR.

Figure 20. Pool Type LMFBFR.

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The various LMFBR fueling alternatives examined during the PRLCDS were designated as listed below. The contractors participating in the study are identified in Table 35.

<u>Nomenclature</u>	<u>Core fuel</u>	<u>Blankets</u>
Reference breeder	Pu-U	U
Blanket transmuter	Pu-U	Th
Full transmuter	Pu-Th	Th
Denatured breeder	U-233-U-238	Th
U-233-Th breeder	U-233-Th	Th

In addition, the fuels considered were in the form of oxides, carbides, and metals, and the core configurations studied were characterized as homogeneous and heterogeneous. A homogeneous core is one in which the central region is fuel and the outer periphery is a blanket of fertile material; a heterogeneous core is one composed of annular regions alternating between fuel and fertile material. Only the homogeneous configuration was studied for the metal fuels. Figure 21 is a diagram of the homogeneous and heterogeneous core configurations.

The performance characteristics and principal fuel parameters for each core design are identified in Tables 36 and 37. The design characteristics of the fuels are considered state of the art (i.e., successful achievement of current R&D goals is assumed).

General Electric (GE) was responsible for developing homogeneous core designs using oxide fuel. The GE reference design and the blanket transmuter have essentially the same design configuration. The GE full transmuter has a larger pin diameter and a lower pitch-to-diameter ratio. These changes were made to increase the breeding ratio, which was significantly reduced by the addition of thorium to the core. The fuel pin diameter of the denatured design was increased further to keep the enrichment of U-233 in U-238 below 12%. The residence time of this design was also increased to offset the higher fabrication costs associated with fuel containing U-232.

The Westinghouse Advanced Reactors Division (WARD) was responsible for developing heterogeneous core designs using oxide fuel. Small and large fuel pin diameters were used. The large pin was employed to increase the fuel volume fraction, thus obtaining a high breeding ratio. The small pin was employed to decrease the fissile inventory. Both the reference design

Table 35. PRLCDS Contractor Participation^a

Reactor type	Oxide		Carbide		Metal, homogeneous
	Homogeneous	Heterogeneous	Homogeneous	Heterogeneous	
Reference	GE ^b	WARD ^c	CE ^d	AI ^e	ANL ^f
Blanket transmuter	GE	WARD	CE	AI	
Full transmuter	GE	WARD	CE	AI	
Denatured	GE	WARD	CE		
U-233-Th	ANL		ANL		ANL

^aHanford Engineering Development Laboratory (HEDL) was the technical manager for the study and developed the ground rules required to ensure consistency among the designs. ORNL evaluated the performance of particulate fuel in a heterogeneous design. In addition, WARD provided support in specifying fuel pin design limits and provided fuel pin verification analysis at the end of the study.

^bGeneral Electric.

^cWestinghouse Advanced Reactors Division (WARD) supplied two designs for each type, one employing a large pin and one employing a small pin.

^dCombustion Engineering (CE) also provided two designs with fuel intended for coprocessing.

^eAtomics International.

^fArgonne National Laboratory.

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Figure 21. LMFBR Configurations.

Table 36. LMFBR Performance Characteristics

Parameter	Oxide fuels									
	Homogeneous				Heterogeneous					
					Large pin		Small pin			
	Pu-U/U	Pu-U/Th	Pu-Th/Th	Denatured	Pu-U/U	Pu-U/Th	Pu-U/U	Pu-U/Th	Pu-Th/Th	Denatured
Breeding ratio	1.32	1.31	1.22	1.23	1.44	1.40	1.27	1.25	1.30	1.28
Compound system doubling time, years	14.8	15.8	29.6	21.2	18		22			
Fissile inventory, kg BOC	3635	3526	4212	3893	5203	5509	3674	3854	6317	5296
Fissile gain, kg/year										
Pu	246	-67	-588	467	282	-171	161	-259	-621	463
U-233	0	301	750	-288	0	407	0	388	778	-247
Peak exposure, MWd/t	95,000	95,000	91,000	90,000	89,000	94,000	152,000	156,000	91,000	90,000
Fuel residence time, full-power years	1.73	1.73	1.73	2.11	2.1	2.1	2.1	2.1	2.1	2.1

Table 36 (continued)

Parameter	Carbide fuels							Metal fuel, homogeneous, Pu-U/U
	Homogeneous				Heterogeneous			
	Pu-U/U	Pu-U/Th ^a	Pu-Th/Th	Denatured	Pu-U/U	Pu-U/Th	Pu-Th/Th	
Breeding ratio	1.42	1.39	1.19	-1.21	1.55	1.42	1.28	1.54
Compound system doubling time, years	9.7	12	33.2	18.7	12.9	18.3	35.0	9.2
Fissile inventory, kg BOC	3155	3035	3636	2990	4671	4780	5469	3160
Fissile gain, kg/year								
Pu	321	-20	-576	398	353	-232	-600	378
U-233	0	304	693	-248	0	494	769	0
Peak exposure, MWd/t	104,000		123,000	109,000	88,000	94,000	100,000	62,000
Fuel residence time, full- power years	1.7	1.7	1.9	2.0	2.1	2.1	2.1	1.4

^aThis design is preliminary rather than final. Consequently, it is not consistent with the other homogeneous carbide designs.

Table 37. LMFBR Fuel-Design Characteristics

Parameter	Oxide fuels									
	Homogeneous				Heterogeneous					
					Large pin		Small pin			
	Pu-U/U	Pu-U/Th ^a	Pu-Th/Th	Denatured	Pu-U/U	Pu-U/Th	Pu-U/U	Pu-U/Th	Pu-Th/Th	Denatured
Pin OD, cm	0.737	0.737	0.787	0.864	0.787	0.787	0.584	0.584	0.787	0.787
Number of pins	271	271	271	271	271	271	271	271	271	271
Cladding, cm	0.030	0.030	0.033	0.038	0.033	0.033	0.030	0.030	0.033	0.033
Ground rule minimum	0.030	0.030	0.030	0.030	0.030	0.030	0.030	0.030	0.030	0.030
Cladding thickness to OD ratio	0.041	0.041	0.042	0.044	0.042	0.042	0.052	0.052	0.042	0.042
Ground rule minimum	0.039	0.039	0.039	0.039	0.039	0.039	0.039	0.039	0.039	0.039
Smear density, %	90	90	90	90	91	91	91	91	91	91
Ground rule maximum	90	90	90	90	90	90	90	90	90	90
Peak linear power, kW/m	50.5	50.5	55.8	54.1	46.9	48.6	43.6	46.6	42.6	49.2
Ground rule maximum	49.2	49.2	45.9	54.1	49.2	49.2	49.2	49.2	45.9	54.1

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Table 37 (continued)

Parameter	Carbide fuels							Metal fuel, homogeneous, Pu-U/U
	Homogeneous				Heterogeneous			
	Pu-U/U	Pu-U/Th ^a	Pu-Th/Th	Denatured	Pu-U/U	Pu-U/Th	Pu-Th/Th	
Pin OD, cm	0.940	0.940	0.940	0.889	0.940	0.940	0.940	0.711
Number of pins	169		169	169	169	169	169	271
Cladding, cm	0.038	0.038	0.038	0.036	0.038	0.038	0.038	0.048
Ground rule minimum	0.030	0.030	0.030	0.030	0.030	0.030	0.030	0.030
Cladding thickness to OD ratio	0.041	0.041	0.041	0.040	0.041	0.041	0.041	0.068
Ground rule minimum	0.039	0.039	0.039	0.039	0.039	0.039	0.039	
Smear density, %	77	77	78	78	81	81	81	75
Ground rule maximum	82	82	82	82	82	82	82	
Peak linear power, kW/m	120.4		122.7	125.6	78.4	85.6	69.9	49.2
Ground rule maximum								

^aThis design is preliminary rather than final. Consequently, it is not consistent with the other homogeneous carbide designs.

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and the blanket transmuted were developed using the large- and small-diameter fuel pins. The full transmuted used only the large pin to increase the fuel volume fraction and thereby offset the effect of thorium in the core. The large pin was also used for the denatured design in an attempt to keep the enrichment of U-233 in U-238 below 12%.

Combustion Engineering (CE) was responsible for developing homogeneous core designs using carbide fuel. The designs were based on large-diameter fuel pins that yielded values for fuel-cycle costs and doubling time in the optimum range. The fuel pin diameter for the denatured design was somewhat smaller than those for the reference and transmuted designs. This allowed lowering the fissile inventory as much as possible while maintaining a fissile enrichment lower than 12%.

Atomics International (AI) was responsible for developing heterogeneous core designs using carbide fuel. All designs use the same fuel pin, the same subassembly geometry, and very similar core configurations. The fluence in the AI designs was kept low by employing a relatively low linear heat rate. This alleviated problems with bundle-duct interaction and allowed them to obtain low pitch-to-diameter ratios.

Argonne National Laboratory (ANL) was responsible both for developing a homogeneous reference core design using metal fuel and for developing homogeneous core designs containing U-233 in thorium. The design was based on a fuel pin with a low linear heat rate, thick cladding, and a relatively short residence time. However, even with this conservative approach, ANL researchers were able to obtain a high breeding ratio (1.54) and low compound-system doubling time (9.2 years) for their design with metal fuel. Three designs containing U-233 in thorium were also developed. These designs were developed using either oxide, carbide, or metal fuel, and they all have relatively low breeding ratios and long doubling times.

Comparison of Fueling Alternatives

Reference Designs -- A summary of the reactor performance specifications for the homogeneous reference design is provided in Table 38.

For reference homogeneous designs containing plutonium-uranium in the core and uranium in the blankets and using oxide fuel, the breeding ratio is estimated to range from 1.30 to 1.36. The breeding ratio for reference designs using carbide fuel is estimated to range from 1.38 to 1.46, and the breeding

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Table 38. LMFBR General Performance Specifications,
GE Reference Design

Power plant performance	
Core thermal power, MW	2,740
Net electrical power, MW	1,000
Thermal efficiency, %	36.5
Reactor parameters	
Core and blanket volume, liters	20,853
Equivalent core diameter, m	
Without blanket	2.82
With blanket	3.71
Core height, m	1.22
Coolant inlet temperature, °C	343
Coolant outlet temperature, °C	499
Fuel parameters	
Pin OD, cm	0.74
Cladding temperature, °C	558
Core fuel loading, initial core, kg	
Total heavy metal	85,200
Fissile material	3,635
Discharged exposure, MWd/t	
Average	61,000
Peak	95,000
Conversion ratio, equilibrium cycle	1.32

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ratio for reference designs using metal fuel is estimated to range from 1.50 to 1.60. Thus, it may be concluded that the breeding ratio of the carbide fuel is approximately ten points higher (one point = breeding ratio of 0.01) than that of the oxide fuel and, likewise, that the breeding ratio of metal fuel is approximately ten points higher than that of carbide fuel. Based on this information, doubling times of the various fuels are expected to range as follows: oxide fuel, 13 to 16 years; carbide fuel, 9 to 11 years; and metal fuel, 8 to 10 years. It was found that the doubling times decrease as one proceeds from oxide to carbide to metal fuel because the breeding ratio increases while the inventory generally decreases.

The estimated breeding ratio ranges for the designs are: reference heterogeneous oxide designs, 1.40 to 1.46; and reference heterogeneous carbide designs, 1.50 to 1.58. In general, the heterogeneous design appears to have a breeding ratio approximately 10 to 12 points higher than the corresponding homogeneous design.

The doubling time of the heterogeneous design is approximately three to four years longer than that of the homogeneous design, primarily because the increase in the heterogeneous design fuel inventory has a greater effect than the increase in the breeding ratio. However, this difference may be narrowed with further design optimization.

The reference designs in this study can be ranked in terms of increasing breeding ratio as follows: oxide, carbide, and metal. However, the designs can be ranked in terms of our increasing knowledge about their fuel and safety performance in the following order: metal, carbide, and oxide. Thus, the fuel systems that show the maximum breeding also have the smallest available data base.

Blanket Transmuter Designs -- It was found that the homogeneous blanket transmuter produces a significant amount of U-233, with little reduction in breeding ratio. The breeding ratio of the homogeneous blanket transmuter using oxide fuel was only one point lower than that of the corresponding reference design. Similarly, the breeding ratio of the homogeneous blanket transmuter is approximately equal to that of the homogeneous reference design, regardless of fuel type.

The principal feature of the homogeneous blanket transmuter was that the net fissile production was switched from plutonium to U-233. In addition, the fissile plutonium loss in the core with either oxide or carbide fuel was small.

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The net fissile production rate of the heterogeneous blanket transmuter was reduced by the addition of thorium to the core, and it is significantly less than that of the heterogeneous reference design. The effect of adding thorium to the core is also reflected in the breeding ratio, which is lower for the heterogeneous blanket transmuter than for the corresponding reference design. For example, the breeding ratio of the heterogeneous blanket transmuter is 4 points lower using oxide fuel and 10 to 13 points lower using carbide fuel.

Full Transmuter Designs -- The advantage of a full transmuter is that a large amount of U-233 is produced, while the disadvantage is that a large amount of plutonium is simultaneously destroyed. The U-233 production rates in the full transmuter do not differ significantly from homogeneous to heterogeneous design or from oxide to carbide fuel. This is due to the small difference in theoretical density between thorium oxide and thorium carbide and also due to the negligible fast-fission cross section of thorium.

The net fissile production rate of the full transmuter design was generally lower than that of other designs because of a significant amount of thorium in the core. Furthermore, the breeding ratio of the full transmuter designs in this study was 10 to 27 points lower than the corresponding reference designs. In addition, the inventory of the full transmuter designs was 15 to 20% higher than the inventory of the associated reference designs. The combination of a lower breeding ratio and a higher inventory produced a considerably longer compound system doubling time for these designs.

Denatured Breeder Designs -- The principal advantage of these denatured breeder designs, from a proliferation standpoint, is that the fissile material in the fresh fuel cannot be chemically separated from the fertile material. The designs, were, however, difficult to develop because the concentration of U-233 in U-238 was required to be below 12% to maintain proliferation resistance. This, coupled with the relatively poor fast-fission properties of some of the uranium isotopes (U-234, U-235, U-236), makes it very difficult to develop a concept with a breeding ratio approaching that of the plutonium-fueled designs.

The breeding ratio of the homogeneous denatured design using oxide fuel was approximately nine points lower than that of the corresponding reference design. Also, the breeding ratio of the homogeneous carbide design was approximately 21 points lower than that of the corresponding reference design. In general, the breeding ratio of the denatured designs in this study was 9 to 21 points lower than that of the corresponding reference designs.

U-233-Th Breeder Designs -- The breeding ratios of these designs ranged from 12 to 31 points below those of the corresponding reference designs, even when a very aggressive design philosophy was used. These designs have the advantage of a very low sodium void worth; however, this advantage is probably not enough to offset the severe resource disadvantages associated with them.

Symbiotic Systems

Three missions were considered for the FBR in support of a nuclear nonproliferation policy: (1) achieving maximum growth, (2) supplying moderate amounts of denatured fuel internationally, or (3) supplying large amounts of denatured fuel from energy centers. In this study, these missions were called the reference regime, the international fuel supply regime, and the energy-center regime.

In the reference regime, the emphasis was on growth (short doubling time); hence, the reference designs exhibited the best breeding performance. The growth rate was highest for designs using metal fuel and lowest for those using oxide fuel.

In the energy-center regime, the emphasis was on a high external energy support ratio. A nuclear system based on the full transmuter exhibited the best performance in this regime. Such a system was capable of supporting ~4 GWe of denatured LWRs externally for each 1 GWe internally. In this regime, an outside supply of fissile fuel is required, unless the symbiotic dispersed reactors have a conversion ratio near 1.0 or above (denatured LMFBRs or LWBRs).

In the international fuel supply regime, the emphasis was on supplying a moderate amount of denatured fuel. This can be obtained by either a full transmuter supplying a denatured FBR or a blanket transmuter supplying a denatured LWR. The choice between the two systems will depend on whether a denatured breeder can be deployed internationally and on the relative economics. If the breeder can be deployed internationally, the nuclear system could grow at ~3%/year. If a denatured breeder cannot be deployed internationally, the nuclear system would probably be based on the blanket transmuter.

If a blanket transmuter is used to supply a denatured LWR, the nuclear system may or may not grow, depending on the amount of denatured fuel required. If it is necessary to support less than 1 GWe externally for each 1 GWe internally, the nuclear system will generally grow. Conversely, if it is necessary to support more than 1 GWe externally for each 1 GWe internally, the

nuclear system will generally decay. In this regime, it is easier to support growth with carbide or metal fuels than with oxides because of the higher breeding ratio.

1.6.2 Status of Technology

The development of the LMFBR in the United States has been proceeding with a high priority since 1967 as a means of ensuring long-range supplies of nuclear fuel. The feasibility of developing commercially attractive LMFBRs is founded on an extensive body of both LWR experience and sodium reactor technology. The LMFBR has a prior record of accomplishment and accumulated experience. The initial prototype, Experimental Breeder Reactor-I (EBR-I), was, in fact, the first nuclear reactor to produce electricity and began operating in 1951. Further, Experimental Breeder Reactor-II (EBR-II) has been operating successfully since 1964.

Over the span of nearly 28 years since EBR-I first went critical (August 24, 1951), 16 civilian reactors ranging from 1 MW to 1000 MWt and cooled with liquid sodium have been started up around the world. Together, they represent some 3300 MW of thermal capacity and 124 reactor-years of liquid-metal-cooled reactor operations and maintenance experience. As shown in Figure 22, nine of these reactors are currently in use, with three in the Soviet Union, two in France, and one each in the United Kingdom, Japan, Federal Republic of Germany, and the United States. Twelve of the 16 sodium reactors were equipped to produce steam, and 11 were equipped to produce electricity.

In addition, there are six more sodium-cooled breeder reactors currently under construction, including the BN-600 in the Soviet Union; FBTR in Madras, India; FFTF in the United States; PEC in Italy; SNR-300 in the Federal Republic of Germany; and the Creys-Malville Super Phenix in France, representing an additional increment of over 5700 MWt. All of these except PEC and FFTF are designed to produce steam and electricity.

Beyond those sodium-cooled breeder reactors currently under construction, there are another 10 or 12 LMFBR power reactors in design or planned among the leading breeder nations, averaging over 1000 MWe (2600 MWt) each. Furthermore, there are over 100 sodium test loops and rigs that have been built and operated at three dozen laboratories and test facilities around the world. A summary listing of the principal LMFBR facilities is presented in Table 39.

Figure 22. World Liquid-Metal-Cooled Reactor Facilities.

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Table 39. LMFBR Projects

Name	Country	Type	Power		Operation
			MWt	MWe	
Operable					
BR-10 ^b	Soviet Union	Loop	10		1973 ^a
DFR ^b	United Kingdom	Loop	60	14	1959
EBR-II	United States	Pool	62.5	20	1963
Rapsodie	France	Loop	40		1967
BOR-60	Soviet Union	Loop	60	12	1969
BN-350	Soviet Union	Loop	1000	350(equiv)	1972
Phenix	France	Pool	567	250	1973
PFR	United Kingdom	Pool	600	250	1974
Joyo	Japan	Loop	100		1977
In commissioning					
KNK-2	Federal Republic of Germany	Loop	58	20	1977 ^c
Under construction					
BN-600	Soviet Union	Pool	1470	600	1979
FFTF	United States	Loop	400		1979
PEC	Italy	Loop	135		1980
Madras FRTR	India	Loop	40	15	1979
SNR-300	Federal Republic of Germany	Loop	762	327	1981
Super Phenix	France	Pool	2900	1200	1982
In design					
CRBR	United States	Loop	975	350	N.A. ^d
Monju	Japan	Loop	714	300	N.A. ^d
CFR-1	United Kingdom	Pool	3100	1300	
	Soviet Union	N.A.		1600	N.A. ^d
Saone	France	Pool		1200-1800	N.A. ^d

^aFollowing conversion from BR-5, which operated from 1959 to 1971.

^bDecommissioned March 1977.

^cFollowing conversion from the thermal-neutron KNK-1, which operated from 1971 to 1974.

^dNot available.

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The U.S. and foreign approaches to the development of the LMFBR differ significantly in basic philosophy. The United States has taken a comprehensive technology approach to the LMFBR. The emphasis is on understanding the full range of technology problems; thus, a broad array of facilities and programs has been developed in addition to the LMFBR experimental reactor projects. In contrast, the United Kingdom, the Soviet Union, and France have taken a reactor-centered approach in their fast-reactor programs. Emphasis in these foreign efforts has been on the earliest possible reactor operation, which has meant moving directly from the experimental reactors to 250-MWe demonstration plants. Thus, the general emphasis on early demonstration plants has resulted in a number of foreign demonstration reactors operating at this time.

Because the FBRs have a more difficult problem than LWRs with regard to the damaging effects of fast-neutron irradiation on structural and fuel-related materials, the first stage in the U.S. program has emphasized developing the capability to acquire a comprehensive understanding of materials behavior in the fast breeder environment. The first step in achieving this understanding required converting an existing facility, the EBR-II, to an irradiation test reactor.

These EBR-II experiments lead logically to the second stage of the breeder program, the design and construction of the Fast Flux Test Facility (FFTF). The FFTF will have the instrumented capabilities in-core to provide information about neutron flux effects and reactor fuel, material, and component behavior under reactor conditions. The information gained about fuels and materials will provide the basis to develop the fuels and materials necessary to realize the ultimate potential of commercial LMFBRs. The fuel/materials information will be especially important for development of the advanced mixed-oxide stainless steel cores.

Fuel Development Status

While the performance characteristics of the oxide, carbide, and metal fuels differ considerably, the amount of development work that has been performed on each of the three fuels also differs considerably. For example, the amount of irradiation testing performed on oxide fuel (steady-state, transient, and off-normal) is considerably greater than that performed on either carbide or metal fuel. Carbide fuel is currently under irradiation, but the number of tests is limited. Metal fuel has been irradiated in EBR-II for many years; however, the metal fuel used in this study is of the uranium-zirconium type, for which there has been very little irradiation experience.

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Even though carbide fuel development has been more actively pursued in recent years, the irradiation data base is less than for oxides, and additional work is required to bring the carbide fuel to a stage of development equivalent to that of oxide fuel. Nevertheless, the incentive for development is there because, if carbide fuel can be satisfactorily developed, improved breeding performance is expected.

Component Development Status

Component test facilities are an integral part of the U.S. development program. The High-Temperature Sodium Facility (HTSF) at the Hanford site will proof-test every major reactor component before it is installed in FFTF. Within this facility, the Composite-Reactor Component Test Assembly (CRCTA) has tested FFTF prototypical components.

However, while the FFTF will be the central test facility, the entire LMFBR program is supported by a number of other major facilities. The Energy Technology Engineering Center (ETEC), formerly the Liquid-Metal Engineering Center (LMEC), at Santa Susana, California, will proof-test prototypic liquid-metal components such as pumps, valves, heat exchangers, and steam generators under normal and potentially abnormal conditions. For example, the Sodium Pump Test Facility (SPTF), one of the ETEC facilities, will permit full-scale testing of large sodium pumps for both FFTF and the Clinch River Breeder Reactor Project (CRBRP). In physics development, zero-power reactors have been utilized. At these facilities the neutronic characteristics of proposed reactors are simulated, allowing the studies needed for development and validation of nuclear design approaches to be carried out. The Zero-Power Plutonium Reactor (ZPPR) in Idaho is the largest facility of its type in the world. There the entire reactor cores and blankets of the FFTF and the CRBRP have been simulated neutronically. The facility has been expanded to allow similar work to be conducted on full-sized commercial plant designs.

1.6.3 Licensing Issues

The Enrico Fermi reactor was a licensed reactor, and the FFTF has been evaluated by the NRC staff and the Advisory Committee on Reactor Safeguards (ACRS) and found acceptable for construction. An additional NRC review before operation of the FFTF is currently under way. The licensing evaluation of the CRBR conducted so far indicates that the state of technology and experience should eventually result in a safe design, but the particular design proposed has not yet been licensed for construction. The major concerns that remain unresolved include instrumentation to detect core

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abnormalities which could lead to accidents, inspection of the reactor system to provide continued confidence in the system integrity, decay heat removal system reliability, scram system, residual heat removal system, fuel rod and subassembly failure detection, and containment design to withstand low-probability accident consequences. All of these topics are being addressed in the United States and in other countries, and the outlook appears favorable for their resolution.

The alternative fuel cycles have an effect on plant design, safety, and licensability. Of significant importance for licensing is the status of demonstrated technology and analytical techniques; these are presently concentrated on the mixed oxides of U-238 and plutonium, including most of the safety experiments and analysis models, but should be applicable to the fissile U-233 and fertile thorium arrangements. The use of U-233 and thorium cores reduces the positive void coefficient compared with the plutonium-uranium cores; this reduction can substantially alter accident sequences (especially for whole-core accidents), with a potential for reduction of the probability of hydrodynamic disassembly. This result can also be achieved with the plutonium-uranium cores by using a heterogeneous (parfait) core arrangement.

Licensing of large LMFBRs will have to be preceded by a determination that deployment of the technology is in the national interest, considering public concerns about reprocessing, the spread of plutonium fuels, and reactor safety. No serious technical obstacles are currently foreseen, in view of favorable foreign experience and assuming a systematic RD&D program in the United States. The fewest technical barriers are seen for licensing uranium-plutonium oxide fuels, where a large body of experience has been accumulated. A similar depth of experience would be required either for alternate fuels such as carbides or metals or for alternate fuel cycles.

1.6.4 Safety Issues

The major safety issues of the LMFBR are the following:

- chemical activities of sodium and the potential for reaction with water and oxygen,
- the potential for leakage of coolant from the system, thereby removing cooling from the core,
- the possibility of voiding sodium,
- the potential for reconfiguration of the core into a critical mass following core melting, leading to an accident,

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- use of plutonium and the potential hazards to man from this toxic substance,
- fuel-coolant interaction.

Safety Effects of Reference Designs

The above issues have been considered in the LMFBR design, and discussion of these issues is summarized in the following paragraphs.

Sodium has been safely used as coolant in reactors and test loops for about 25 years. The LMFBR makes use of a secondary sodium loop, which separates the radioactive primary loop from the steam generators. The presence of water is generally prohibited in areas containing radioactive sodium, except in specially designated areas for cleaning large components that may have been in contact with sodium. In those cases, after drainage of sodium, inert gas flushing, and steam washing, the component may finally be washed safely with water.

The presence of oxygen is prohibited in areas around components that contain radioactive sodium. In particular, the primary heat transport system components are located in cells rendered inert with nitrogen, and the sodium heat transport system within the primary circuit has an argon cover gas. In reactors containing these features, inadvertent spills of radioactive sodium would be directed into an inert area, and no sodium burning could be supported.

An LMFBR system is designed so that leakage of primary-system sodium is highly improbable. Nevertheless, features are included in the design to ensure that the reactor core can be cooled even if a leak should develop. The sodium inventory is sufficient to ensure that no leak could result in loss of cooling capability. The major heat transport system components are provided with surrounding guard vessels to limit the leakage of sodium, so that the reactor core and the coolant nozzles on the reactor vessel remain submerged at all times. The piping between the major components is elevated, so that a pipe leak might result in a loss of siphon in that loop but would permit continued core cooling and decay-heat removal through the remaining heat transport system loops.

The possibility of sodium boiling gives rise to potential reactivity changes that could be positive in certain portions of the core, thus enabling both a

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power increase and a simultaneous possible reduction of cooling capability. In fact, there is a large margin between the operating temperatures of the sodium coolant (maximum 650°C) and the boiling temperature of the sodium (~940°C), and no initiating mechanism of any likelihood has been identified that could cause sodium to boil.

One method of voiding sodium that could be postulated is by gas entrainment and gas collection, or sweep, through the core. The design incorporates a vortex suppressor plate beneath the sodium surface in the reactor vessel, vent lines at high points in the system to vent any small amounts of gas that may accumulate, and multiple barriers to prevent the entry of materials, such as pump lubricant, that could result in the formation of gas.

For recriticality to be inadvertently obtained within the LMFBR core, the core would have to be reconfigured in some way following melting or disruption of the fuel. No initiating mechanism for such gross core damage has been identified, except for postulated hypothetical events (such as those involving the arbitrary assumption of complete failure of the shutdown systems) used to judge the adequacy of containment barriers.

If the initial core disassembly is mild, leading to gradual meltdown, an energetic recriticality event is possible, in principle. For oxide fuel (uranium- as well as thorium-based), recriticality appears to be precluded by natural dispersive mechanisms (fission gas, cladding vaporization, and fuel vaporization). Fission gas acts early, while cladding vaporization tends to ensure long-term dispersals at decay-heat power levels. (The relatively high melting point of $\text{ThO}_2\text{-UO}_2$, compared with the boiling point of steel, may even play a favorable role for early fuel dispersal.)

A concern exists for the potential release of plutonium to the environment. In calculating the plutonium doses that could result from accidents involving the release of plutonium, conservatism is introduced by assuming that plutonium aerosol acts as a gas and could, therefore, leak out of the containment at a very slow rate and be inhaled. The containment is designed and tested to a leak tightness of 0.1 volume percent per day under the maximum design-basis accident conditions.

Safety Impact of Alternate Designs and Fuels

Safety characteristics vary among the design choices (homogeneous or heterogeneous), among the fuels (oxide, carbide, or metal), and among the fuel cycles (Pu-U/U, Pu-Th/Th, denatured, etc.).

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In relation to accident prevention (the ability to safely shut down the reactor when called on and to subsequently remove decay heat), no significant safety implications of other fuel types compared with the reference oxide fuel (uranium-plutonium) appear to be present. Based on inherent prompt negative feedback characteristics, the metal- and carbide-fueled reactors would place demands on required response of the plant protection system similar to those on that of the reference oxide fuel. The Doppler coefficient for the metal fuel (~ 0.003) is lower than that for oxide fuel (~ 0.008). The degree to which the lower Doppler coefficient of reactivity for the metal fuel affects reactor transients needs to be examined for low and highly irradiated fuel conditions.

In the area of postaccident heat removal, the potential safety feature of sodium boiling for removing decay heat (in the case of insufficient time for establishing single-phase natural circulation) may not be available for metal fuel to the same degree as for oxide and carbide fuels. This is because the eutectic temperature between the metal fuel and cladding is below the boiling temperature of the sodium.

However, it can probably be shown that the duration of sodium boiling before reestablishing single-phase natural circulation of relatively low-temperature sodium is short compared with the required time for any significant cladding penetration.

The sodium void worth is a parameter that governs the severity of certain unlikely LOCAs, which potentially lead to core-disruptive accidents. The lower positive void coefficient is the major incentive for using a heterogeneous core, which therefore may be considered "safer." Cycles that call for U-233 or thorium in the core generally have less sodium void worth than uranium-plutonium fuels; the sodium worth generally parallels the breeding ratio.

Much attention has been given to detailed mechanistic calculations of core-disruptive accidents using accident analysis computer codes (SAS, MELT, and SIMMER). These codes do not readily lend themselves to an assessment of safety characteristics of the alternative fuel candidates because the available codes are usually developed specifically to handle the reference oxide fuel or because they are still largely in a developmental stage. Thus, relying on physical principles is necessary in assessing safety implications of the alternative fuels.

Following this approach, significant differences in safety characteristics are much more dependent on whether the fuel is in the oxide, carbide, or

metal form than on whether the fuel is U-233 or Pu-239 or the fertile material is U-238 or thorium, with the exception of the sodium void effect alluded to earlier.

For oxide fuels, the cladding boiling point is roughly equal to or somewhat lower than the melting point of the fuel, suggesting that steel vapor from cladding boiling can provide an effective dispersal mechanism. Explosive sodium-vapor formation is not likely to be involved for oxide fuel/sodium systems. Uranium-plutonium metal fuels that are bonded with and have become logged with sodium have a potentially similar dispersive mechanism because sodium bond would begin boiling very nearly coincidentally with initiation of fuel disruption (i.e., at the metal fuel melting temperature, which is below the cladding melting temperature). The dispersal potential appears less favorable for thorium-based metal fuels because the cladding melts at a lower temperature than the fuel does. This melting temperature would allow the sodium bond to escape before fuel disruption. No significant differences appear between uranium-plutonium carbide and thorium-based carbide.

In the case of a loss-of-heat-sink accident, oxide and carbide fuels would experience disruption only after the coolant level dropped below the core because sodium boiling would appear capable of removing decay-heat power levels. This favorable safety characteristic may not be available for the metal fuels because the low-melting eutectic temperature between the fuel and cladding is such that fuel disruption may occur in the presence of coolant. However, in this case, continued sodium vaporization may provide fuel dispersal, thereby preventing energetic recriticalities.

Radiological consequence problems related to aerosol behavior in the secondary containment are likely to be affected by the fuel type. However, no qualitative difference in allowable release is expected because of a change to alternate fuel cycles,

1.6.5 Environmental Issues

The environmental impact of the LMFBR employing a uranium-plutonium fuel cycle will be less than for the LWR on a slightly enriched uranium cycle with uranium-plutonium recycle. The LMFBR is designed to operate at a higher thermal efficiency as compared with the LWR (37% as compared with 33%), resulting in about 19% lower environmental heat load per unit of electrical production. Uranium enrichment is eliminated, and the need for uranium or thorium mining and milling is essentially eliminated, except for small makeup quantities of fertile material. Routine releases of radionuclides in LMFBR

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facilities are not expected to be significantly different from LWR facilities.

With regard to the environmental impact of postulated plant accidents, plant design features to mitigate these accidents should ensure that all off-site exposure limits are met. Breeder fuels, which contain high concentrations of toxic plutonium, present greater risks of occupational exposure than LEU fuels. However, proper facility design and operation should alleviate these risks.

It is not anticipated that the environmental impacts associated with the use of carbide and metal fuels will differ from those associated with oxide fuels. Although the offsite consequences of plant accidents involving fuel release to the containment building will have to be analyzed, no qualitative differences are expected.

1.6.6 RD&D Needs

An RD&D program leading to the commercialization of LMFBRs was initially set forth in the LMFBR program plan issued by the Atomic Energy Commission in 1968. This program has been revised and updated to reflect advances in technology and the continued refinement and definition of objectives. In the development of the program, there was an underlying emphasis on facilities: facilities to permit testing of physics, fuel components, and instrumentation and facilities to design and proof-test needed components. In this discussion of RD&D needs, it is recognized that a comprehensive development program on LMFBRs has been in progress in the United States for many years and that much of the technology needed for commercial LMFBRs either is already in hand or will probably be generated within the next decade. In addition, every technical uncertainty need not be resolved within the U.S. program alone. The sizable body of LMFBR experience that has been developed in various foreign programs has already demonstrated the technical feasibility of the LMFBR concept, and the new 1200- to 1500-MWe LMFBR plants being designed and constructed in Europe should reduce the technical uncertainty in commercial-size plants over the next 10 to 12 years.

Research Facilities

Several of the existing research facilities, such as FFTF and EBR-II, were discussed in Section 1.6.2. Additional facilities that will be used as part of the base program include the following:

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Transient Reactor Test Facility (TREAT) -- The TREAT is a pulsed test reactor. Current experiments are designed to examine fuel failure under transient overpower and loss-of-flow conditions and to provide data for reconstruction of the mechanisms that produced fuel failure and subsequent movements of sodium, fuel, and cladding.

Sodium Loop Safety Facility (SLSF) -- The SLSF is designed to produce information needed to demonstrate the behavior of FBR fuel elements or subassemblies for postulated abnormal or hypothetical accident situations.

Thermal-Hydraulic Out-of-Reactor Safety Facility (THORS) -- THORS provides an engineering-scale high-temperature sodium loop for thermal-hydraulic testing of simulated reactor subassemblies at normal operating conditions.

Sodium Boiling Test Facility (SBTF) -- The SBTF is a single-channel sodium loop for the study of free-convection and low-flow forced-convection boiling dynamics and heat transfer.

Components and Materials Evaluations Loop (CAMEL) -- CAMEL is designed to perform out-of-reactor tests undertaken to examine the hydraulic aspects of fuel sweep-out or plug formation under simulated transient overpower conditions.

Containment Systems Test Facility (CSTF) -- The CSTF is designed to obtain baseline information on aerosol behavior in large vessels and on the performance of emergency air-cleaning systems.

Out-of-Pile Expulsion and Reentry Apparatus (OPERA) -- OPERA was constructed at ANL primarily for prototype studies of coolant behavior after postulated pump-coastdown and flow-blockage transients.

Large Sodium Fire Facility (LSFF) -- The LSFF at Hanford Engineering Development Laboratory (HEDL) has been modified to perform a variety of investigations of phenomena associated with sodium spills. These include the evaluation of cavity-liner designs, sodium-concrete interactions, hydrogen recombination, large-scale "feature" tests, and filter-loading studies on emergency air-cleaning systems.

Direct Electric Heating (DEH) -- The DEH apparatus at ANL relies on the electrical resistance of the fuel pellets to generate time-varying thermal transients in a restrained fuel column.

Fuel Cladding Transient Tester (FCTT) -- The FCTT at HEDL provides data on the response of unirradiated and irradiated cladding materials to internal pressure and thermal loadings.

Fission-Gas Release Mechanism (FGRM) -- The FGRM facility at HEDL is designed to provide data on transient fission-product release from samples of irradiated fuel.

Alternate-Fuel-Cycle Research

Fuels of $\text{ThO}_2\text{-UO}_2$, $\text{ThO}_2\text{-PuO}_2$, and UO_2 , compared with $\text{UO}_2\text{-PuO}_2$ fuel, are expected to be stronger and to have lower creep rates and lower thermal expansion. Little is definitely known about their fuel swelling rates or fission-gas behavior under fast-flux, high-power-density conditions. The higher strength, however, may give more fuel-cladding mechanical interaction than $\text{UO}_2\text{-PuO}_2$ at comparable temperatures and heat ratings. High-strength cladding materials, therefore, might be necessary for these fuels. Thermal performance limits need to be determined.

Carbide fuel development has been under way in the United States since 1957 to provide an improved breeding option by capitalizing on the inherent nuclear advantages that these fuels have over oxide fuels. Efforts have progressed from an exploratory phase in the period before 1974 to the current concept-screening phase, which began in 1974 and will be completed in the period between 1979 and 1981. Other than the fact that a detailed design data base remains to be generated on the specific reference carbide design that is ultimately selected (as is currently being developed for oxide fuel), no significant performance-related barriers have been identified that would preclude the use of carbide fuel in future breeder reactors.

Three basic design concepts for carbide fuel elements are recognized: (1) sodium-bonded high-density pellets, (2) helium-bonded low-density pellets, and (3) helium-bonded high-density particulate fuel. Each of the three designs is under development in LMFBR fuel programs in several different countries. France has selected sodium-bonded pellet fuel for development, the Federal Republic of Germany helium-bonded pellet fuel, and the United Kingdom

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and Switzerland helium-bonded particulate fuel. The United States program presently includes approximately equal emphasis on sodium-bonded and helium-bonded pellet fuels.

The worldwide irradiation data base on UC-PuC fuels is rapidly expanding on the various fuel element design concepts. It includes steady-state endurance tests over a variety of operating conditions, transient overpower and undercooling experiments, and tests of breached fuel elements. In the U.S. program, EBR-II is being used to irradiate both sodium-bonded and helium-bonded fuels to identify the performance limitations of each concept as a basis for selecting one fuel element design for detailed testing. To date, under the conditions tested, neither design has demonstrated any significant performance limitation.

Because few irradiation data are available on alternative carbide fuels, a considerable program of steady-state and transient testing would be required to design and construct reactors using such fuel.

With regard to metal fuels development, thorium-uranium and thorium-plutonium alloys have been tested as prototype sodium-bonded fuel elements to high burnup in only a few cases. Uranium-molybdenum alloys have been evaluated extensively, primarily to determine the optimum heat treatment for dimensional stability and to minimize high-temperature swelling. It is believed, however, that information more pertinent for LMFBR application is provided by the U-Fs* alloy used in the Mark II driver fuel element in EBR-II. There has been favorable experience with this driver fuel.

Irradiation tests of metal fuels to target burnups are needed to provide required information on the possible influence of fission products, constituent migration, and other irradiation effects on fuel-cladding compatibility. If the Th-Pu alloy is selected for development, large-scale irradiations should be preceded by an alloy development program to raise the solidus temperature of the thorium-plutonium-uranium fuel alloy that will exist at the end of life of the fuel element. Zirconium additives are a logical approach, based on experience in related uranium-plutonium alloys.

*Fissium (Fs) is an equilibrium concentration of fission-product elements left by the pyrometallurgical reprocessing cycle used for EBR-II. Simulated U-Fs alloy contains 2.4% molybdenum, 1.9% ruthenium, 0.3% rhodium, 2.0% palladium, 0.1% zirconium, and 0.01% niobium.

The predicted capability of the candidate alloy fuel elements to accommodate design transients should be verified by transient tests on both unirradiated and highly irradiated fuel elements.

The performance characteristics of a design are also dependent on the behavior of the structural material in the core. The stainless steel used as structural material in the PRLCDS was superior to the steel used in the first core of the FFTF. This difference in steel could affect the results of this study, because designs with high fluence are desirable. For example, the magnitude of the fast flux is generally lowest in an oxide fuel design, higher in a carbide fuel design, and highest in a metal fuel design. This effect is caused by the removal of the moderating fuel diluent as one proceeds from oxide to carbide to metal.

Thus, for the same residence time, the fluence will tend to be higher in the carbide and metal designs, and, thus, their performance characteristics may be more sensitive to the properties of the structural material. However, the fluence can always be decreased in these designs by reducing either the residence time or the linear heat rate, though, such reductions will effect an increase in the fuel-cycle cost.

Test Facilities and Demonstration Plants

Most of the principal test facilities that will be required in support of an LMFBR demonstration plant and commercial plant are currently in existence. Table 40 identifies the principal facilities.

Five new facilities that have been under consideration are the Safety Research Experiment Facility (SAREF), the High-Performance Fuels Laboratory (HPFL), the Hot Experimental Facility (HEF), which is intended for the LMFBR fuel reprocessing demonstration at ORNL, and the Fuel Storage Facility (FSF) at HEDL.

Some modifications of existing test facilities will be required to handle prototype components. The principal ones are as follows:

1. The SPTF will have to be expanded from a 24,000-gpm pump rating to a capacity four times that size.

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Table 40. List of Principal LMFBR Test Facilities

Facility	Name	Location
FFTF	Fast Flux Test Facility	HEDL
CRBRP	Clinch River Breeder Reactor Project	Oak Ridge
EBR-II	Experimental Breeder Reactor II	ANL-W
ZPPR	Zero Power Plutonium Reactor	ANL-W
SAREF ^a	Safety Research Experimental Facility	ANL-W
SPTF	Sodium Pump Test Facility	ETEC
ETR-SLF	Engineering Test Reactor — Sodium Loop Safety Facility	INEL (NRTS)
TREAT	Transient Reactor Test Facility	ANL-W
LLTR	Large-Leak Test Rig	ETEC
SOWAT	Small-Leak Sodium-Water Reaction Test Rig	GE
HTSF	High-Temperature Sodium Facility	HEDL
SCTL	Small Components Test Loop	ETEC
SCTI	Sodium Components Test Installation	ETEC
HTF	Hydraulic Test Facility	Rocketdyne
PAL	Prototype Applications Loop	HEDL
LTV	Large Test Vessel	AI-San Susana
	Building 006, 009, 028	AI-San Susana
HFEF/N	Hot Fuel Examination Facility/North	ANL-W
HTL	Hydraulic Test Loop	AI-San Susana
CMR-HCF	CMR-Bldg. (Wing 9) Hot Cell Facility	LASL
RM	Radiometallurgical Laboratory	HEDL
FMEF ^a	Fuels and Materials Examination Facility	HEDL
SDCTF	Shut Down Component Testing Facility	GE-San Jose
CHTR	Chain Hoist Test Rig	AI-San Susana
	125,000-gpm Hydraulic Test Facility	Edwards AFB
STCL	Source Term Control Loop	HEDL
CHCF	Component Handling and Cleaning Facility	ETEC
TTF	Thermal Transient Facility	ETEC
STL	Sodium Test Lab (Bldg. 32)	ETEC
HPFL ^b	High Performance Fuels Laboratory	HEDL
PBF	Power Burst Facility	INEL
	Chemistry Research	Rocky Flats
FFM	Fuel Failure Mockup	ORNL
SMF	Shielded Material Facility	HEDL
HEF ^a	Hot Experimental Facility (reprocessing)	ORNL
FSF	FFTF Fuel Storage Facility	HEDL

^aIn planning.

^bHas been combined with FMEF.

2. The Composite-Reactor Component Test Activity within the HTSF at HEDL will need to be modified to accommodate large fuel-handling equipment, primarily the ex-vessel transfer machine of the A-frame concept.

The major reactor facilities that would be required for commercialization of LMFBRs include the FFTF, the demonstration plant, and a lead plant. Since 1977, the LMFBR program has been redirected to emphasize alternative fuels and fuel cycles that might offer nonproliferation advantages. The Administration has proposed that CRBRP be discontinued and that a conceptual design study be undertaken to develop the design for the LMFBR demonstration plant.

Costs and Schedule

The estimated costs and a potential schedule for the LMFBR RD&D program are identified in Tables 41 and 42, respectively. For the purposes of this report, it has been assumed that the costs identified in the LMFBR Program Overall Plan for R&D are generically applicable, that the demonstration plant has a capital cost of about 1.5 times that of a 900-MWe LMFBR, and that the lead plant cost is at the high end of the range of the Prototype Large Breeder Reactor (PLBR) studies.

The RD&D costs for alternate fuel cycles and fuel materials are included within the range identified for the Reactor R&D Program.

1.6.7 Perceived Advantages and Disadvantages

The major advantage of LMFBRs is their ability to use U-238 or thorium as the resource base instead of the much more limited supply of U-235, which is the resource base for LWRs. The U.S. or world resource of fertile fuels is many times that of fossil fuel, making breeder reactors an "inexhaustible" source of energy.

Compared with other advanced reactor concepts, the LMFBR has benefited from extensive development and reactor operating experience in the U.S. and abroad. Many facilities are contributing technology on sodium handling, materials, fuels, core heat transfer, heat exchangers, fuel handling, steam generators, sodium impurity control, pumps, control systems, shielding, physics, fuel fabrication, and reactor safety. LMFBRs have the advantage of low-pressure primary and secondary systems, a highly conductive coolant that normally operates ~330°C below its boiling point, and a refractory fuel that

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Table 41. LMBFR RD&D Costs^a

RD&D area	Cost (\$ million) ^b
Reactor plant R&D	5,300-6,000
Plant demonstration	
Proof-of-principle plant	N.R. ^c
Demonstration plant, 900 MWe	1,700-2,100
Lead plant, 1200 MWe	1,800-2,200
NSS vendor support	100-150
Increment for multiple vendors	0
Total reactor RD&D	8,900-10,450

^aCosts include the initial fuel costs, which are estimated to be \$300 to \$350 million (see Volume IV).

^bJanuary 1978 \$.

^cN.R. = not required.

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Table 42. LMFBR RD&D Schedule

RD&D activity area	Time frame (years)
Reactor plant R&D	1979-2004
Plant demonstration	
Proof-of-principle plant	N.R. ^a
Start design	
Construction permit	
Operating license	
Demonstration plant	
Start design	1979
Construction permit	1984
Operating license	1990
Lead plant	
Start design	1988
Construction permit	1994
Operating license	2001
First commercial plant	
Start design	1999
Construction permit	2003
Operating license	2009

^aN.R. = not required.

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is well characterized and relatively inert to sodium. Because of the apparent advantages of the breeder, most technologically advanced nations have, or are taking part in, breeder development programs.

The disadvantages of the LMFBR originate from safety concerns and the cost of alleviating these concerns. They include the flammability of sodium, a positive sodium void coefficient (although advanced heterogeneous cores reduce this to a low level), the necessity to design for high temperatures and to accommodate thermal shock characteristics of sodium, and the need to mitigate the possible consequences of sodium-water reactions originating from the steam generators. These problems lead to a need for technical sophistication and expertise greater than that required for current reactor types (LWRs or HWRs).

With regard to core configuration, heterogeneous cores have higher breeding ratios than homogeneous cores and somewhat longer compound system doubling times as a result of higher fissile inventories. The sodium void worth of the heterogeneous designs is significantly lower than that of the corresponding homogeneous design.

The carbide and metal fuels have advantages over the oxide designs with regard to breeding ratio and doubling time. However, the RD&D needs for carbide and metal fuels are greater than for the oxide fuel. The licensability of oxide fuels is more plausible than that of the alternatives; however, more RD&D may make the advanced fuels equally acceptable.

Finally, the introduction of alternative fuels (U-233) or alternative fertile materials (thorium) into the LMFBR reduces the breeding ratio and increases the doubling time. The alternative fuel cycles appear to be technically feasible, and they may have some attractive safety features. However, because the bulk of RD&D thus far has been expended on the uranium-plutonium cycle, somewhat more time and money would be required to commercialize an alternate cycle.

1.7 SPECTRAL-SHIFT-CONTROLLED REACTORS

1.7.1 Reactor/Fuel-Cycle Systems Description

The spectral-shift-controlled reactor (SSCR) is an advanced converter reactor that is based on PWR technology and offers improved resource utilization, particularly when recycling is employed. The SSCR differs from the conventional PWR in that it is designed to minimize the number of captures in poison rods and to utilize to the greatest possible extent captures of excess neutrons in fertile material as a method of reactivity control. The resulting increase in the production of fissile material serves to reduce fuel makeup requirements.

In the conventional PWR, long-term reactivity control is achieved by varying the concentration of soluble boron in the coolant to capture the excess neutrons generated throughout plant life. This soluble boron concentration is relatively high at beginning of cycle (about 700 to 1500 ppm) and is gradually reduced during the operating cycle (by feed and bleed chemistry control) to compensate for the depletion of fissile inventory and the buildup of fission products.

The SSCR consists basically of the standard PWR with the conventional soluble-boron reactivity-control system replaced with spectral-shift control. Spectral-shift control is achieved by the addition of heavy water to the reactor coolant, in a manner analogous to the use of soluble boron in the conventional PWR. Because heavy water is a poorer neutron moderator than light water, the introduction of heavy water shifts the neutron spectrum in the reactor to higher energies and results in the preferential absorption of neutrons in fertile materials. In contrast to the conventional PWR, where absorption in control absorbers is unproductive, the absorption of excess neutrons in fertile material breeds additional fissile material, increasing the conversion ratio of the system and decreasing the annual makeup requirements. At beginning of cycle, a high (~60 wt %) D₂O concentration is employed to increase the absorption of neutrons in fertile material sufficiently to control excess reactivity. Over the cycle, the spectrum is thermalized by decreasing the D₂O/H₂O ratio in the coolant to compensate for fissile-material depletion and fission-product buildup, until at end of cycle, essentially pure light water (~2 wt % D₂O) is present in the coolant.

The basic changes required to implement spectral-shift control in a conventional PWR are not too difficult. In the conventional PWR, pure water is

added and borated water is removed during the cycle to compensate for the depletion of fissile material and buildup of fission-product poisons. The borated water removed from the reactor is processed by the boron concentrator, which separates the discharged coolant into two streams, one containing pure unborated water and the second containing boron at high concentrations. The latter stream is stored until refueling or any cold shutdown, where it is used to provide the boron necessary to hold down the excess reactivity introduced by the loading of fresh fuel. In the SSCR, the major components of the nuclear steam supply system remain essentially unchanged from those employed in a conventional poison-controlled PWR, except that the function of the boron concentrator is replaced by that of the D_2O upgrader. The function of this upgrader is to separate heavy and light water so that concentrated heavy water is available for the next refueling. The upgrader consists of a series of vacuum distillation columns that utilize the differences in volatility between light and heavy water to effect the separation. Although the boron concentrator and the upgrader perform analogous functions and operate using similar processes, the D_2O upgrader is much larger and more sophisticated, consisting of three towers, each about 2.13 m in diameter and 68.6 m tall. Boron, however, is still needed for refueling shutdowns; therefore, some boron processing equipment is still required. Several additional changes will be required to realize spectral-shift control in practice. These include modifications to minimize and recover D_2O leakage, to facilitate refueling, and to remove boron from the coolant after refueling.

The fuel-cycle options that apply to an SSCR are the same as those for an LWR (Section 1.1.1). In the once-through cycle, the standard 30,000-MWd/t burn-up would require an enrichment below 3% in the SSCR. The PWR fuel improvements would be applicable here. U_{308} requirements are about 90% of those of the PWRs in once-through cycles. General SSCR performance specifications are given in Table 43.

The recycle options are both plutonium-uranium cycle and thorium cycle, as described in Section 1.1.1 for the LWR. The SSCR has been historically associated with thorium cycles, and the denatured uranium-thorium cycles in the SSCR require only about 83% of the fuel makeup of corresponding PWR cycles.

1.7.2 Status of Technology

Initial analyses of SSCRs were carried out in the U.S. in the early 1960s by M. C. Edlund, and an experimental verification program was performed for both LEU and HEU-Th fuels by B&W. Edlund's studies, which were performed for

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Table 43. SSCR Performance Specification

<u>Power plant performance</u>	
Core thermal power, MW	3,800
Electrical power, MWe ^a	
Gross	1,336
Net	1,258
Thermal efficiency, %	33
<u>Reactor parameters</u>	
Core volume, liters	40,050
Core dimensions, m	
Equivalent diameter	3.66
Active length	3.81
Core power density, MW/liter	0.095
Coolant flow rate, Mg/sec	20.66
Coolant inlet temperature, °C	296
Coolant outlet temperature, °C	327
Primary system pressure, MPa (psia)	15.5 (2,250)
<u>Fuel parameters</u>	
Average fuel temperature, °C	688
Maximum fuel temperature, ^b °C	1,882
Cladding temperature, °C	342
Core fuel loading, kg	
Total heavy metal	
UO ₂ once-through	102,775
Denatured ThO ₂	93,555
Fissile material	
UO ₂ once-through	2,100
Denatured U-233-ThO ₂ core	2,200
Denatured U-235-ThO ₂ core	3,200
Discharge exposure, MWd/t	
UO ₂ once-through (nominal)	30,400
UO ₂ once-through (high burnup)	50,650
Conversion ratio	
UO ₂ once-through equilibrium cycle	0.67
Denatured U-235-ThO ₂ equilibrium cycle	0.72
Denatured U-233-ThO ₂ equilibrium cycle	0.84
<u>Other relevant data</u>	
Fuel stack density, g/cm ³	
UO ₂ once-through	10.054
Denatured ThO ₂	9.173

^aThese values are for typical mechanical-draft cooling towers.

^bPeak fuel temperature, centerline at 100% power.

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reactors designed specifically for spectral-shift control, indicated that the inventory and consumption of fissile material could be reduced by 25 and 50%, respectively, compared with poison control in reactors fueled with highly enriched U-235 and thorium oxide. They also indicated that a 25% reduction in uranium ore requirements could be realized with spectral-shift control using the LEU cycle.

The spectral-shift-control concept has been demonstrated by the Vulcain reactor experiment in the BR-3 nuclear plant at Mol, Belgium. After two years of operation as a conventional PWR, the 11-MWe BR-3 plant was modified for spectral-shift-control operation and successfully operated with this mode of control between 1966 and 1968. The Vulcain core operated to a core average burnup of 23,000 MWd/t (a peak burnup of around 50,000 MWd/t) and achieved an average load factor of 91.2 and a primary plant availability factor of 98.6. The leakage rate of primary water from the high-pressure reactor system to the atmosphere was found to be negligible, about 30 kg of D₂O/H₂O mixture per year. After the Vulcain experiment was completed, the BR-3 was subsequently returned to conventional PWR operation. In addition to demonstrating the technical feasibility of spectral-shift control, the Vulcain experiment identified the potential engineering problems inherent in converting existing plants to the spectral-shift mode of control.

One technology that would require substantial expansion if significant deployment of the SSCR were to occur is heavy-water separation capacity. The existing plants in Canada and at Savannah River, South Carolina, could be used to obtain the D₂O inventory for a limited number of reactors; however, additional capacity would be required for significant commercial deployment. Because the technology of heavy-water production is well developed both in the U.S. and abroad, no new advances in this area are required for the commercial system. There is, nevertheless, a strong incentive for developing a low-cost heavy-water separation process, because the heavy-water inventory represents a substantial cost (\$50 to 60 million) to the SSCR.

The only major component not already fully developed for the SSCR is the D₂O upgrader. While D₂O upgraders are being used in conjunction with CANDU-HWRs, the size of these upgraders is different from those that would be required for the SSCR.

Another area of the technology that would require extension is that of valves and seals to minimize the leakage of the heavy-water mixture. It is important to reduce the leakage of the D₂O mixture from an economic standpoint because of the radiological hazard from tritium that is produced in the mixture. Although extensive experience exists from the efforts on the HWR,

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additional development will be required because of the larger-size components and the higher pressures under which the SSCR would operate.

In spite of the fact that the thermal hydraulics of the D₂O mixture are expected to be similar to those of H₂O, a testing program like the one carried out for the PWRs would have to be undertaken to establish the thermal margins.

Additionally, the appropriate concentrations of D₂O in the mixture during refueling would need to be determined. The appropriate concentrations would be based on considerations of excess reactivity, radiological hazard from tritium, and excess D₂O inventories.

1.7.3 Licensing Issues

For the evaluation of the safety aspects of an SSCR, it is expected that the majority of the safety criteria applied to the evaluation of a conventional PWR will be applicable.

It is not expected that safety/licensing considerations will impose any significant constraints on the commercial introduction date. Because of the general similarity of the SSCR to a conventional PWR, the resolution of the specific SSCR safety/licensing items should be accomplished within a reasonable time after submittal of a detailed design to NRC for review.

1.7.4 Safety Issues

Because of the closeness of the SSCR design concept to that of a conventional PWR, no new fundamental problems related to safety, licensing, and environmental impact are expected. However, there are a number of safety/licensing-related design aspects that will require evaluation.

Because soluble boron is not utilized in the SSCR, the moderator-temperature coefficient of reactivity is more negative early in the cycle, and, hence, warm-water accidents (e.g., loss of load, loss of flow, and accidents that increase core reactivity and power) in the SSCR are expected to be less severe than in the conventional PWR. Since the coolant in the SSCR at the end of cycle is essentially pure water, similar consequences of cold-water accidents are anticipated at end of cycle in the SSCR; however, because

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of the absence of boron and the hard neutron spectrum at beginning of cycle, cold-water accidents early in the cycle are major items that must be explored.

The effectiveness of the emergency core-cooling system, which injects highly borated light water during accidents and anticipated operational occurrences, must also be evaluated; this system may be less effective in the SSCR because of the presence of light water in the injected borated water. The system also might require either an increase in system boron concentration or an increase in injection capacity to obtain an equivalent degree of effectiveness.

Because of the harder neutron spectrum resulting from the presence of D_2O in the coolant, a reduction of the control-rod effectiveness might be anticipated. However, the thermal diffusion length is greater in D_2O , and the control rods are "seen" by neutrons from a greater distance. As a consequence of these competing effects, the control-rod worth is comparable with that of a PWR. The Doppler coefficient at the beginning of a cycle is more negative than for a PWR because of the larger fractional neutron absorption in the fertile material. As a result of the greater mean path length of the neutrons in the D_2O -containing coolant, the neutron fluence at the wall of the reactor vessel is about 30% higher than for a PWR. The resulting increase of the nil-ductility transition temperature by about $20^\circ F$ at the end of the vessel life appears to be tolerable.

Inadvertent injection of unborated light water into the coolant system will cause an increase in reactivity when the D_2O concentration is high. This accident is similar to unborated water addition in the PWR. It appears that, with similar controls, as in the PWR, the D_2O dilution accident in the SSCR can be prevented or its consequences mitigated.

To minimize the tritium radiological hazard during refueling, a reference procedure has been developed that is a variation of the standard refueling procedure. It is based on the concept of exchanging the residual H_2O/D_2O coolant with H_2O before opening the reactor vessel before refueling. To salvage as much D_2O as possible, the coolant will be drained and the water level lowered below the reactor vessel flange to the centerline of the primary-coolant pipes. The remaining coolant volume will be flushed out to one of the holdup tanks with H_2O from the refueling water tank using existing pumps and pipes of the shutdown cooling system, as well as some additional hardware. Subsequently, the reactor vessel will be filled with H_2O to the vessel flange, and the refueling procedure will be continued as in a conventional PWR. The susceptibility of this refueling scheme to procedural violations requires evaluation.

1.7.5 Environmental Issues

No new fundamental problems related to the environmental impact of SSCRs are expected; however, a number of design aspects related to environmental impacts will require evaluation. The increased production of tritium in the SSCR requires evaluation of its potential environmental impact. It is anticipated that tritium control in the reactor plant will allow environmental impact criteria to be met.

Tritium control, as it affects operating personnel exposure and consequent constraints on space and equipment access, would require a detailed evaluation. The tritium production and tritium concentration in SSCR coolant are considerably higher (about 20 times) than in a conventional PWR. This can lead to tritium concentrations in the coolant at beginning of cycle that are several hundred times those experienced in PWRs.

1.7.6 RD&D Needs

This section describes the RD&D program for the SSCR concept, which is basically a modified PWR in which long-term reactivity is controlled by withdrawal of heavy water rather than soluble boron from the primary coolant. In the SSCR concept, the use of a system of control rods for reactivity control in start-up, shutdown, and load-changing, as well as the use of highly borated water for the emergency core-cooling systems, remains unchanged.

Two types of fuel cycles are under consideration for the SSCR and are treated separately herein. The first is the UO_2 once-through cycle using LWR fuel assemblies of a standard design but qualified for extended burnup in the SSCR. The second applies to a denatured-thorium fuel cycle with recycle of U-233. In the latter case, the RD&D considerations covered here pertain only to the reactor design and not to the fuel-cycle facilities.

Proof of the spectral-shift-control principle has been demonstrated with the Belgian BR-3, and a plant design was developed and considered for commercial marketing in the 1960s by a U.S. reactor vendor. This fact, coupled with the fact that the technology for heavy-water handling has been well developed for Canadian WRs, leads to the opinion that a standard PWR design could be modified and licensed as an SSCR without the need for a prototype. Such a plant would carry a small capital cost risk, since it could be operated as a standard PWR in the unlikely event that the spectral-shift-control feature were unsuccessful.

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In the RD&D program under consideration for the SSCR, the standard design of a large PWR (1200-MWe class) would be revised in accordance with the SSCR concept, followed by licensing and construction of a new plant based on the revised PWR design.

Although the SSCR concept is based on revision of the standard PWR design, a number of generic programs will be required to establish the technology for this adaptation. These programs are described below.

1. Experimental physics data base development. Substitution of D₂O in the reactor primary-coolant system will significantly alter the following safety-related physics parameters, thus requiring the development of a new data base. The parameters are:

- reactivity coefficients and predictions with core life,
- core neutron flux and power distribution,
- D₂O worths,
- control-rod worths,
- effects of the presence of light water in the injected borated water from the safety injection system,
- neutron spectral dependence on D₂O,
- investigation of neutron flux at the pressure vessel wall by non-destructive test (NDT) consideration.

A critical facility will be required for the above investigations.

2. Thermal-hydraulic investigations. Although departure-from-nucleate-boiling correlations for D₂O will be close to those for H₂O, there probably will be enough difference to require the development of experimental

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correlations for D₂O/H₂O mixtures. Other thermal-hydraulic investigations may be necessary.

3. Component development. Component development for the SSCR will be concerned mainly with the following items:

- upgrading valves and seals to reduce D₂O leakage, which is necessary both from an economic standpoint and to minimize the potential radiological hazard from tritium;
- making the D₂O upgrader for the SSCR identical in concept to the upgraders used in HWRs and the last stage of D₂O production facilities (because of size considerations and D₂O/H₂O mixture-control requirements, special development would be required);
- providing a D₂O-concentration measuring device with appropriate range;
- providing special components for refueling and for coolant exchange during refueling.

4. Plant systems development. Most of the plant systems for the existing PWR design would be maintained, with possibly some minor modifications for D₂O. New system design and development would be required for the following systems:

- D₂O upgrading system,
- refueling system,
- coolant exchange system.

5. Fuel development.

- a. Once-through cycle. Standard-design LWR fuel assemblies would be used in the SSCR with a once-through cycle; extended burnup fuel developed for the LWR could also be employed in the SSCR.
- b. Recycle. In addition to consideration of fabrication, reprocessing, and high-level waste disposal (areas that are not the subject of this report), the thorium-loaded fuel for the SSCR would require investigations to demonstrate its suitability with respect to the following:
 - in-reactor densification,
 - swelling,
 - fission-gas release,
 - thermal conductivity,
 - pellet-cladding interaction,
 - reactivity coefficients.
- c. Refueling methods. Special refueling methods will have to be developed for the SSCR because of the radiological hazard of tritium. D_2O will have to be exchanged for borated H_2O during refueling.

6. Generic safety and licensing. No new fundamental safety and licensing problems are anticipated because of the basic similarity to the conventional PWR design; however, investigations will be required in the following areas to ensure licensing:

- warm- and cold-water accidents at beginning and end of cycle,
- effectiveness of the safety injection system,

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- change in control-rod worth,
- LOCA with D₂O/H₂O coolant mixture and high-burnup fuel,
- nil-ductility transition temperature change for reactor pressure vessel,
- tritium release and radiological hazards.

After completion of the above-described technology generic programs, a large demonstration plant would be designed, licensed, and constructed. This project would be carried out by revising a standard PWR plant design in accordance with the SSCR concept. Over and above the normal engineering, design, licensing, and construction required for the standard PWR plant, this project would require the following special additional activities:

- safety and licensing work to prepare the specific SARs and environmental report for the SSCR plant,
- special engineering for changes in plant systems, equipment, and structures,
- procurement and installation of special equipment.

After successful operation of the demonstration plant, special activities relative to the lead commercial plant would generally be required. However, other than the scale-up in size, these requirements should be as described for the demonstration.

A schedule for the overall SSCR once-through RD&D program is shown in Table 44, and the costs for such a program are shown in Table 45. The cost estimates are based on the following assumptions:

- Estimates are based on 1978 costs.

Table 44. SSCR RD&D Schedule

SSCR once-through RD&D activity area	Time frame (years)
Reactor plant R&D	1979-1983
Plant demonstration	
Proof-of-principle plant	N.R. ^a
Start design	
Construction permit	
Operating license	
Demonstration plant	N.R. ^a
Start design	
Construction permit	
Operating license	
Lead plant	
Start design	1981
Construction permit	1984
Operating license	1991
First commercial plant	
Start design	1990
Construction permit	1993
Operating license	2000

^aNot required.

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Table 45. SSCR Once-Through
RD&D Costs^a

RD&D area	Cost (\$ million) ^b
Reactor plant R&D	50-130
Plant demonstration	
Proof-of-principle plant	N.R. ^c
Demonstration plant	N.R. ^c
Lead plant (1200 MWe)	1150-1270
Total reactor RD&D	1200-1400

^aDoes not include initial costs of fuel, which are estimated to be \$90 to 100 million. (See Volume IV.)

^bJanuary 1978 \$.

^cNot required.

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- Operating costs are not included for the demonstration plant and the lead commercial plant.
- No allowance is made for payments to the utility operating the demonstration plant because of the possible necessity to operate at a lower capacity factor with an experimental plant.

1.7.7 Perceived Advantages and Disadvantages

Major advantages of the SSCR are that its design is based on PWR technology and that a standard SSCR can be designed using a licensed standard PWR plant as a starting point. The RD&D cost of developing an SSCR is therefore modest. Also, the SSCR has significantly better fuel utilization than the LWR.

The disadvantages of the SSCR are: (1) its economics are unfavorable (i.e., the improved fueling characteristics do not pay for the D₂O), (2) the need to produce D₂O is an added complication, and (3) other routes to improving the PWR are available, including those discussed in Section 1.1.2 and under WCBR technology. Some of these PWR improvements, however, would also benefit the SSCR.

1.8 ACCELERATOR-DRIVEN REACTORS

The accelerator-driven reactor systems are characterized by the use of high-energy particle accelerators to drive fission reactor assemblies. The accelerated particle beam impinges on a target, which becomes an intense neutron source. The source neutrons are then absorbed in the blanket system to create fission power and possibly to breed fissile material. Several combinations of accelerator, targets, and fission blanket assemblies have been proposed. Two of these concepts have been reviewed for NASAP: the Linear Accelerator Fuel Regenerator (LAFR) proposed by Brookhaven National Laboratory and the Ternary Metal-Fueled Electronuclear Fuel Producer (TMF-ENFP) proposed by the Oak Ridge National Laboratory. Section 1.8.1 discusses the LAFR concept, while Section 1.8.2 is devoted to the TMF-ENFP concept.

1.8.1 Linear Accelerator Fuel Regenerator (LAFR)

Reactor/Fuel-Cycle System Description

The LAFR would use a high-energy proton beam from a linear accelerator with a liquid lead-bismuth target to provide an intense neutron source (Figure 23). The concept is designed to produce fuel by enriching the fissile content of LWR fuel elements in the fission blanket. Thermal power production would be incidental to the enrichment process. The proposed fuel cycle would use standard LWR fuel assemblies made with slightly enriched uranium. After fabrication, these assemblies would be irradiated in individual pressure tubes surrounding the LAFR neutron source for approximately one year. (Shorter exposures in each of several tubes might be required to obtain the desired quantity and distribution of fissile material in each element.) At the end of this irradiation, the assemblies would be removed from the LAFR and used as LWR reload fuel. After depletion in LWRs for the nominal time of three to four years, the fuel would be returned to the LAFR for further irradiation, and the fuel would again be enriched in plutonium. After a second depletion in LWRs, the fuel elements are assumed to be mechanically unsuitable for continued use and would be discarded. One LAFR would be expected to enrich fuel for three LWRs.

The major parts of the LAFR facility would include the accelerator, the liquid lead-bismuth target system (which would require heat removal), the fission blanket system (which would be cooled by a steam and water combination), and the conventional steam plant (which would use the heat from the target and the blanket to produce steam and drive a conventional turbine generator). The power output from the generator is expected to be about 450

Figure 23. Linear Accelerator Fuel Regenerator (LAFR).

MWe. The power requirements of the accelerator are expected to be 900 MWe; thus, the LAFR would require about 450 MWe of net power from outside sources.

Status of Technology

The basic principles of using an accelerator to produce neutrons and of using neutrons to breed fissile material are well known. Thus, on the surface, only further engineering development would seem to be required to produce a commercial-sized facility. However, several large extrapolations of existing technology have been assumed that are currently beyond the state of the art and would require substantial RD&D, as discussed in Section 1.8.3. The most formidable example is the use of standard LWR fuel at greatly increased burn-up. In principle, fabrication and use of the LWR are a well-established technology. However, for the LAFR application, the standard LWR fuel design will be subjected to more than twice the average exposure normally experienced in LWR applications. A significant part of this irradiation will include high-energy particles with highly anisotropic fluxes. These altered operating conditions will probably require substantial amounts of fuel development effort and may not be technically feasible.

Another example is the accelerator beam performance, as measured by a combination of voltage, current, and duty factor. The proposed beam would deliver continuously 300 mA of protons at 1.5 GeV to the liquid-metal target with a plant operating factor of 0.75. Again, in principle, accelerator designs to meet these requirements can be extrapolated from present designs. However, not only must a size factor of 200 to 300 be overcome, but the accelerator must also be highly reliable and operate almost continuously. In addition, the efficiency (assumed 50%) with which input electricity is converted to the 450 MW of beam power will be an important economic consideration. All performance goals are significantly beyond the current state of technology and, therefore, may require substantial amounts of accelerator development effort.

Vacuum system, liquid-metal target circulating system, reactor pressure vessel, calandria pressure tubes, fuel handling, and material compatibility aspects probably will also require significant engineering development. Even the proposed steam cycle is somewhat unconventional in that an expansion valve is used to obtain the appropriate steam and water combination for fuel cooling. This is not a standard commercial item (at least in the size required) and, therefore, would probably require development effort. The proposed two-phase cooling cycle is not considered essential to the concept, however; gas and heavy water are alternative coolants.

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In general, the principles to be used by the LAFR design are known, but the required extrapolation will involve substantial RD&D.

Licensing Issues

The most distinguishing feature of the LAFR system from the standpoint of licensing is that the fissile blanket is a subcritical assembly. This fact may ease many licensing issues normally associated with criticality. However, blanket meltdown, transient behavior, and potential criticality will require investigation and must be demonstrated to be manageable. The addition of the accelerator and target system will directly affect licensing by increasing shielding requirements and by producing spallation products.

A major licensing concern will be the use of the LAFR-irradiated fuel in LWRs. The fuel will receive high-energy irradiation, which may cause substantial variation in isotopic distributions among different assemblies and within the same assembly. The effects of these distributions on power peaking, operating margins, and reactor control will have to be investigated before the LWRs can be licensed to accept such fuel. In addition, it will be necessary to establish inspection and quality assurance methods to prove that refreshed fuel meets whatever acceptance criteria are established for reinsertion of such elements into an LWR. The large extrapolation of fuel performance will also be a possible licensing issue, as will be the effects of the high-energy irradiation.

Safety Issues

The design-basis accidents have not yet been defined for the LAFR concept. The possibility of blanket meltdown will be one candidate, with coolant pipe ruptures as one possible mechanism for causing this accident. Several mechanisms may exist for overpower transients that could possibly cause meltdown. Changes in the operating characteristics or failure of the lead-bismuth nozzles could cause unacceptable power peaks or power transients. Sudden changes in the proton beam may also have the same effect.

Several engineered safety systems may have to be incorporated to prevent blanket meltdown. A shutdown heat removal system will probably be required, as well as emergency spray and flooding systems. The possibility of criticality under flooded conditions in the pressure tubes will have to be

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addressed, because flooding might occur as a result of the action of safety systems or as the result of sudden cooling of the water-steam mixture used to remove heat from the fuel being irradiated.

Containment isolation in the event of primary coolant boundary breach in the blanket system must be considered. The target region would apparently be operated at a very high vacuum in direct communication with the accelerator beam tube. This places one relatively large containment penetration in direct communication with the primary-coolant system boundary. Thus, at least one isolation valve would probably be required where the accelerator beam penetrates the containment boundary. The speed with which this valve could be closed and the potential damage to the valve caused by the proton beam may be important issues in containment design. Other related issues are (1) the effect on the accelerator of sudden large releases of steam and water into the vacuum region and (2) the confinement of volatile and other radioactive spallation products that would be formed in the lead-bismuth target.

Shielding of the LAFR will probably be significantly more difficult than for an LWR. The shielding requirements for the coolant system should be similar to those of a BWR, but special consideration would be required in the region above the blanket to shield the fuel shuffling and refueling operations while protecting the vacuum regions from excessive exposure to water. The shielding of the target area and accelerator beam tube will pose additional problems because of the high energy of the proton beam and resulting neutrons. The spallation products that remain in the lead-bismuth are likely to make the lead-bismuth systems highly radioactive and again greatly increase shielding requirements over those of similar LWRs.

Environmental Issues

The LAFR would not affect the environment in any way that is fundamentally different from reactors now operating. The net power requirements of the LAFR imply that the overall thermal efficiency of a system including the LAFR will be reduced, thus increasing the net thermal discharge per unit of electrical output. Gaseous and liquid radioactive wastes should be similar to those from LWRs of similar fuel mass, with the possible exception of the radioactive spallation products generated in the lead-bismuth target. The uncontrolled release of the spallation products to the environment should be prevented, possibly through the provision of additional levels of confinement, for at least part of the lead-bismuth system. Also to be considered is the toxicity of lead (and possibly, to a lesser degree, bismuth) vapors that could be released from any uncontained portions of the molten lead-bismuth circuits.

Because it is unlikely that the LAFR would be colocated with the LWRs that use its product, the shipment of nuclear fuel between the various facilities may also represent an environmental issue. The "first-round" fuel leaving the LAFR would contain relatively little fission-product activity, but it would contain substantial plutonium. The material that would be returned after one exposure in an LWR would be very similar to material that is currently discharged from LWRs. However, to limit the size of the "in-process" fuel inventory, the material would have to be shipped to the LAFR after only a short cooling period in the reactor's spent-fuel storage facility. The re-freshed fuel, on leaving the LAFR, would contain substantially more plutonium (per element) than is currently encountered in LWR fuel and about the same fission product inventory as spent fuel from current LWRs. Finally, the second-round spent fuel from the LWR would contain more long-lived fission products, more plutonium, and more transplutonium actinides than spent fuel from current LWRs.

1.8.2 Ternary Metal-Fueled Electronuclear Fuel Producer (TMF-ENFP)

Reactor/Fuel-Cycle System Description

The TMF-ENFP would use a high-energy proton beam from a linear accelerator with a liquid sodium target to produce a high-intensity neutron source that would drive a blanket of fission fuel assemblies (Figure 24). The sodium would serve as both a target and a coolant to transport heat produced in the target and blanket to an intermediate sodium coolant loop. The intermediate loop would produce steam in a steam generator, with the steam being used to drive a conventional turbine generator plant.

The fuel for the TMF-ENFP would be composed of plutonium, thorium, and uranium. During the exposure of the fuel in the ENFP, the plutonium would function as a neutron-multiplying medium and be consumed, while U-233 would be bred from the thorium. The uranium would provide fertile material for some plutonium regeneration, diluent U-238 for denaturing the U-233, and some fast-neutron multiplication. The proposed fuel cycle would use pyrometallurgical reprocessing and fabrication of the TMF-ENFP fuel to extract the bred U-233 and produce fresh fuel for the reactor. The bred U-233 would be fabricated into denatured LWR fuel and would be sufficient to fuel approximately five LWR plants. The fuel discharged from the LWRs would be reprocessed to recover the U-233 and plutonium. The recovered U-233 would be recycled back to the LWRs as denatured fuel, while the plutonium would be used as part

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Figure 24. Ternary Metal Fueled Electronuclear Fuel Producer (TMF-ENFP).

of the plutonium requirements for the TMF-ENFP. The rest of the plutonium requirements would have to be provided from a separate source. To keep the fuel production rate high, the TMF-ENFP would be refueled about every 90 days.

Status of Technology

As with the LAFR, the basic principles of using an accelerator to produce neutrons and of using neutrons to breed fissile material are well known. The use of sodium as a coolant is being fully developed in the LMFBR program. The technology for the metallic fuel and pyrometallurgical reprocessing and refabrication has been demonstrated on a small scale at EBR-II. However, the vanadium-titanium cladding that is proposed for the metallic fuel has not been established as a practical technology. In addition, large extrapolations of size will have to be demonstrated to yield commercial-sized equipment and plants.

Although the primary particle energy proposed for this concept is only 1 GeV (vs. 1.5 GeV for the LAFR), the extrapolation from current technology that would be required is very nearly the same as that for the LAFR accelerator discussed in Section 1.8.1. The requirements are well defined, but a substantial development effort will be needed to meet them.

This concept requires that the accelerator beam be deflected through 90° to permit it to enter the target blanket region from above, and the deflection device could require considerable development. Also, the target window that must separate the vacuum of the beam tube from the sodium target will be difficult to develop, particularly because it will probably be a part of the primary-coolant pressure boundary and must be capable of frequent replacement due to radiation damage. These requirements will lead to difficult design criteria in areas such as sodium compatibility and leak-tightness.

In general, the principles to be used by the TMF-ENFP have been demonstrated, but substantial RD&D will be required to extrapolate the accelerator to the necessary size and to design an adequate target window. Additional engineering development will be required to provide safe and reliable fuel and plant design.

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Licensing Issues

Because the TMF-ENFP will be cooled with sodium, the licensing issues should be similar to those of the LMFBR. Licensing will have to address all the safety issues discussed below. Adequate levels of containment should be provided by following LMFBR current design, with the possible exception of the isolation valves for the accelerator beam tube. In addition to these safety issues, licensing will have to address the packaging and disposal of high-level wastes from the reprocessing and fabrication facilities. Aside from these issues, no unique licensing issues are anticipated.

Safety Issues

The design-basis accidents have not yet been defined for the TMF-ENFP concept. The possibility of fuel meltdown in regions of high power will be one concern. Interruption of coolant flow, such as by rupture of coolant pipes, coolant flow blockage in the fueled regions, or loss of coolant pump power, would be a possible mechanism for this accident. Several mechanisms may exist for overpower transients, which could possibly cause meltdown. Changes in the proton beam or in beam focusing could cause unacceptable power peaks or power transients, and accidental criticality could also result in overpower conditions.

Several engineered safety systems may have to be instituted to prevent blanket meltdown. A shutdown heat removal system will be required and probably emergency cooling and flooding systems.

Criticality accidents in the blanket regions as well as in the reprocessing and fabrication facilities will have to be addressed. The blanket regions are to be designed to remain subcritical, but this capability will have to be demonstrated under all possible conditions. Containment isolation in the event of primary-coolant boundary breach must be considered, with the rupture of the target window being one of the more likely failures to be evaluated. Such a rupture would require prompt closure of the valve(s) in the opening where the accelerator beam penetrates the containment boundary. The speed with which this valve can be closed and the potential damage caused to the valve by the proton beam may be important issues in the containment design. Definition of design-basis accidents will establish criteria for containment design.

Shielding for the TMF-ENFP should be only slightly more difficult than that for a normal LMFBR design. The shielding requirements for the accelerator

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beam tube will have to be investigated, and shielding around the beam deflection mechanisms may introduce some additional difficulties. The spallation products from neutron production in the sodium may cause some additional safety and handling problems in the radioactive waste handling systems but should not significantly affect shielding requirements in this design.

Environmental Issues

The TMF-ENFP would have essentially the same environmental effects associated with any large LMFBR, except for effects that might be uniquely associated with criticality in the reactor core. In addition, the plant would impose all the effects of a full-scale fuel reprocessing and fabrication facility and a physically large accelerator. The large power requirements for the accelerator are expected to reduce the overall thermal efficiency and increase the net thermal discharge per unit electrical output. No gaseous or liquid radioactive effluent would be discharged from the plant during normal operation.

Onsite reprocessing and fabrication operations would take place in a closed cycle with no routine effluent discharges. High-level radioactive wastes would be extracted by reprocessing for eventual transport to a waste repository. The concentrated fission-product waste should be similar to the high-level waste that would be produced in a conventional fuel reprocessing plant. Ultimate disposal of fission-product wastes would be identical to procedures used in the closed LWR fuel cycle.

1.8.3 RD&D Needs

Technical requirements for the drive accelerator of either the LAFR or the TMF-ENFP are quite similar up to the point of interfacing with the driven reactors. RD&D requirements for the accelerator portion, the reactor portion of the LAFR concept, and the reactor portion of the TMF-ENFP concept are covered separately.

Accelerator RD&D Requirements

In principle, the technology for providing a beam of sufficient intensity exists today; however, linear accelerators with the combination of very high energy (1.5 GeV) and a relatively high continuous beam current (300 mA)

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suitable for this application have not been built and operated. Therefore, many RD&D problems relative to scaling up the accelerator size and properly interfacing it to the target may be anticipated.

From a review of the linear accelerator requirements and design aspects, the following special RD&D items may be identified:

1. Scaling of existing linear accelerator technology to meet the requirements for high energy and high continuous proton beam current.
2. The ability of a high-energy, high-current linear accelerator to operate reliably at a high duty factor, which remains to be demonstrated. Most present-day accelerator applications are experimental in nature and require intermittent operation for relatively short durations.
3. Development of an efficient and reliable ion source for proton injection in accordance with the high-beam-current requirement.
4. Development of the special radio-frequency power supplies. These supplies will represent a very high (900-MW) electrical load on the facility turbine generator or the utility grid and thus will pose a very special coordination problem in terms of start-up, operation at power, and shutdown (especially sudden shutdown).
5. Coordinating the control of the linear accelerator from a safety standpoint with the engineered safety features and safety circuits of the driven reactor, which will require special development.

Linear Accelerator Fuel Regenerator RD&D Requirements

The overall RD&D program, summarized in Table 46, is parallel to that of the more developed concepts and includes research and development, a proof-of-principle engineering-scale plant, a demonstration plant, and the lead commercial plant. The LAFR cannot be developed and commercialized in less than about 40 years.

The LAFR, exclusive of the linear accelerator, is divided into the accelerator interface, the target blanket assembly, the fuel assembly, the target liquid-metal cycle, and the steam-power conversion cycle for purposes of assessing the RD&D requirements.

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Table 46. LAFR Estimated RD&D Cost Summary

RD&D area	Estimated cost ^a (\$ million)
<u>Research and development</u> ^b	
Basic research	700—1200
Development facilities	800—1200
Regulatory studies	500—800
Subtotal	2000—3200
<u>Demonstration facilities</u>	
Proof-of-principle plant	900—1400
Demonstration plant	1300—2000
Lead commercial plant ^c	1300—2000
Total estimated cost	5500—8600

^a Estimated in 1978 dollars. Does not include interest during construction.

^b Assumes existing LWR testing facilities are used for fuel and component testing.

^c The LAFR facility requires a net input of approximately 450 MW of electrical power for normal operation.

Accelerator interface --

1. A special transition section will be required between the end of the accelerator and the target face to spread the collimated beam of the accelerator to match the target array. This is considered a special development item, particularly because its failure could have serious safety implications.
2. Special development of the accelerator isolation valves probably will be necessary. These, in effect, will serve to isolate the primary containment from the accelerator beam tube under accident conditions. Thus, they will permit the accelerator seismic and safety classifications to be lower than those of the target blanket.

Target blanket assembly --

1. Neutron flux and thermal power gradients in the blankets and individual fuel assemblies will have to be investigated very carefully.
2. Design of the target to achieve uniform proton stoppage over its length will require special investigation.
3. Special work will be required to optimize the lead-bismuth jet spray arrangement.
4. The need for a barrier to separate the target from the blanket must be investigated. This may be necessary to protect the pressure tubes from lead-bismuth and would have the beneficial effect of significantly reducing the volume to be operated in a vacuum.
5. The neutron reflector and shield will require special design work.
6. Use of a two-phase water coolant will require extensive investigation and testing, including methods of forming and controlling this mixture. Both dryout and flooding are a safety concern in this respect. Further studies may indicate that either gas or heavy water would be a more appropriate coolant.
7. It must be demonstrated that the blanket will remain subcritical under all burnups, material changes, geometric configurations, and tube flooding conditions that are physically possible, or, alternatively, a reactor shutdown system must be developed and incorporated.
8. The use of Zircaloy as a structural material for the pressure tubes must be qualified, especially because pressure tube failure and possible failure propagation will be a safety item.

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9. The compatibility of carbon steel, stainless steel, or other materials exposed to lead-bismuth in the LAFR environment must be investigated.
10. Emergency core-cooling systems and the shutdown heat removal system will require special design and development because of the large amount of decay heat, unusual LOCA potential, and the need for frequent shutdowns for fuel replacements or shuffling.
11. The requirements for refueling and fuel shuffling in the LAFR will involve the development of a very complex fuel handling system which will probably be of the remote type, to keep radiation exposure of personnel within allowable limits.

Fuel assemblies -- The feasibility of the fuel cladding materials to sustain extended burnups in the power reactor and the irradiation cycle of the LAFR will have to be demonstrated.

Target liquid-metal cycle --

1. Induced radioactivity and spallation products in the lead-bismuth caused by high-energy neutrons and protons will require investigation.
2. Compatibility of lead-bismuth with the materials used in the piping system will have to be demonstrated.
3. Special pumps and heat exchangers probably will have to be developed.

Steam-power conversion cycle --

1. The feasibility of using expansion valves to flash feedwater to the desired two-phase coolant mixture for the blanket will have to be demonstrated. The severe conditions imposed on valves for this type of flashing service, along with the requirements for two-phase mixture control, may require development of special valves. An alternate arrangement that may be considered is the use of special steam generators to produce the two-phase mixture. These steam generators could utilize forced, variable-flow water recirculation to control the outlet quality.
2. Special-design moisture separators would be required.

Ternary Metal-Fueled Electronuclear Fuel Producer RD&D Requirements

The TMF-ENFP, exclusive of the linear accelerator, is divided into the accelerator interface, the target blanket assembly, the fuel assemblies, heat transport systems, and the steam-power conversion cycle, for the purposes of assessing the RD&D requirements.

Accelerator interface -- The TMF-ENFP reactor concept includes a target region that is a central vertical column of liquid sodium with the proton beam entering through a special window at the top. This type of interface results in the following RD&D problems.

1. The transition section between the accelerator output and the target window must provide for a 90° deflection of the proton beam. This is considered to be a major RD&D item, considering the anticipated difficulty in providing such a deflection of the high-energy, high-current proton beam within a reasonable radius.
2. The alternative arrangement of bringing the proton beam straight in horizontally, as in the case of the LAFR, does not appear feasible at this time. However, this may have to be investigated further if the present configuration is not found to be feasible.
3. The special window at the entrance of the proton beam to the target region will serve as the barrier between the hard vacuum of the accelerator beam tube and the interior of the target blanket to the accelerator beam tube. This window will have to be developed to (1) withstand the pressure differential under both normal and accident conditions, (2) withstand the metallurgically degrading effects of proton stoppage in the window and possible exposure to liquid sodium or sodium vapor, and (3) minimize proton beam loss.
4. Because the target window may not be relied upon to maintain containment integrity under accident conditions, a set of isolation valves, which would close automatically under accident conditions, would be required at the containment penetration by the accelerator beam tube. As in the case of the LAFR, these valves are expected to be a special development item.

Target-blanket assemblies --

1. The neutron source strength required to produce the TMF-ENFP performance parameters and the proton beam energy and current requirements to

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produce the required neutron source strength have not been established. This will require further analysis and/or experimental work.

2. The metal fuel would operate at a very high power density and with a severe radial power gradient. Considerable effort will be required to try to improve the power distribution and ensure that coolant flow restriction or blockage will not cause fuel melting and reconfiguration.
3. The present core design in the conceptual model is based on the fuel assembly design used in the EBR-II reactor, which has long operating experience with metallic fuel and sodium cooling. Although this may serve as the basis for the TMF-ENFP fuel assembly design, additional investigation will be needed to verify and/or optimize the following parameters:
 - percentage of the plutonium, uranium, and thorium components of the ternary metal fuel;
 - fuel pin diameter, pitch, length, and cladding thickness;
 - relative volumes of fuel, coolant, and structure;
 - improved cross sections required for thorium and for isotopes in the thorium depletion chains, such as U-233 and protactinium.
4. The eutectic NaK alloy may be preferable to sodium because of its lower melting temperature. However, this would have to be investigated from the standpoint of neutronics for any possible degradation of performance.
5. A basic problem with the TMF-ENFP concept is its change in subcritical multiplication and, therefore, its increase in power level with time. Extensive analysis will be required to arrive at an optimization of the multiplication factor while maintaining a safe margin of subcriticality. Possible methods of countering the effect of variation in multiplication constant are
 - design of the balance of the plant to handle the relatively high end-of-life thermal power generation,
 - adjustment of the neutron source level by control of the accelerator proton beam current,
 - introduction of time-variable neutron poison in the blanket to achieve a constant multiplication factor.

Fuel assemblies -- As previously mentioned, the TMF-ENFP fuel assemblies are based on the EBR-II design; however, because of major differences in

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operating conditions, such as proton and neutron fluxes, between the two reactors, special investigations will be required in the following areas:

1. Considerable investigation and qualification of the ternary metal fuel under the TMF-ENFP design conditions will be necessary.
2. Use of vanadium/titanium alloy as fuel cladding material will have to be qualified. There is limited experience with this in the environment of high proton, neutron, and gamma fluxes, high temperature, and sodium wetting exposure. Therefore, considerable investigative work could be expected in this area.
3. Fuel fabrication and reprocessing are integral to the TMF-ENFP concept. Pyrometallurgical techniques, as used with the EBR-II--type fuel, will be the basis for these processes, but additional RD&D work may be expected because of the special requirements imposed by the TMF-ENFP.

Heat transport systems -- It is reasonable to expect that an intermediate heat transport system would be required for the TMF-ENFP. In accordance with the LMFBR design, this likely would be a sodium-cooled heat transport system to move heat from the primary-system sodium to the steam generators. Technology for the LMFBR primary and secondary sodium heat transport systems is reasonably well developed, and no special RD&D problems are anticipated.

Steam-power conversion cycle -- A conventional steam and power conversion system could be postulated with no special RD&D requirements in this area anticipated, with the exception of steam generators. High-pressure, high-temperature superheated steam conditions (480°C, 13.8 MPa) typical of fossil-fueled power plants could be expected.

It may be anticipated that steam generator design for the TMF-ENFP would probably follow the LMFBR steam generator technology; therefore, some RD&D in this area may be necessary insofar as it is still required for the LMFBR program.

The TMF-ENFP RD&D program is assumed to benefit to a significant extent from developed and developing LMFBR technology, especially with respect to the heat transport systems and the steam-power conversion cycle. However, it is apparent that the TMF-ENFP RD&D program would be more complex, time-consuming, and costly than the LMFBR program because of the need for the accelerator, the accelerator interface, an advanced metal fuel, development of fuel reprocessing/recycle for the dispersed LWRs, and development of pyroprocessing for the ternary metal fuel.

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A summary of the RD&D costs of the TMF-ENFP is given in Table 47. The TMF-ENFP probably cannot be developed and commercially deployed in less than 45 years.

1.8.4 Perceived Advantages and Disadvantages

Linear Accelerator Fuel Regenerator

The principal advantage of the LAFR is its ability to produce reactor fuel from fertile material in situ, thereby significantly extending uranium resources. (The net advantage of the LAFR with respect to uranium resource utilization should be evaluated by comparison with once-through LWRs that achieve the same fuel exposure levels as those proposed for the LAFR, namely, 70 MWd/kg rather than the 30 MWd/kg currently achieved in LWRs.) In addition, the LAFR would do this without the necessity of reprocessing or re-fabrication of radioactive fuel.

Among the technical disadvantages of the concept will be the complexity of the LAFR plant. The addition of the accelerator and the circulating lead-bismuth target to the nominal equipment for an LWR adds significant complexity to the plant design, construction, and operation. In itself, this additional complexity over normal LWRs would tend to decrease the plant capacity factor. In addition, the LAFR must be refueled annually and may require intermediate fuel shuffling to compensate for the steep flux gradients from the LAFR target area. Depending on final designs for fuel handling, these operations might also have an unfavorable effect on plant capacity factor.

Another potentially serious disadvantage is that LAFR-produced fuel will not be very uniform fuel. Isotopic distribution and irradiation may vary among fuel assemblies and even within the same assembly. These variations may be perceived by reactor operators as undesirable, particularly because they could greatly complicate the fuel management tasks in the LWRs that use the fuel and reduce safety margins. A great deal of positive experience would be required to overcome any such perceived disadvantage, even if the perception should be incorrect.

The high power requirements of the LAFR facility will make start-up and shut-down operations more difficult than for LWRs.

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Table 47. TMF-ENFP Estimated RD&D Cost Summary
(Assumes LMFBR research and development continues)

RD&D area	Estimated cost ^a (\$ million)
<u>Research and development</u>	
Basic research	500-1,200
Development facilities	850-1,300
Regulatory studies	750-1,000
Subtotal without major plant construction	2,100-3,500
<u>Demonstration facilities</u>	
Proof-of-principle plant	1,400-2,000
Demonstration plant	2,400-3,500
Lead commercial plant (560 MWe)	2,400-3,500
Total estimated cost	8,300-12,500

^aEstimated in 1978 dollars. Does not include interest during construction. Estimated costs are incremental to LMFBR program costs.

Ternary Metal-Fueled Electronuclear Fuel Producer

The primary advantage of the TMF-ENFP is its ability to produce reactor fuel from fertile material with a minimal consumption of uranium; this could significantly extend uranium resources. In addition, the plant would function as an "incinerator" for plutonium produced in other power reactors.

One of the biggest disadvantages of the TMF-ENFP concept is that it requires full-scale reprocessing and refabrication of fissile material, including plutonium. Another disadvantage will be the complexity of the TMF-ENFP plant. The addition of the accelerator, beam deflector, target window, and fuel reprocessing and fabrication facilities to the equipment nominally required for a sodium-cooled reactor significantly increases the complexity of the plant design, construction, and operation. This additional complexity over normal LWRs would tend to decrease the plant capacity factor. In addition, the 90-day refueling schedule will also have an unfavorable effect on plant operating costs and capacity.

Another potential disadvantage is the sodium-handling aspects of the concept. Even though the LMFBR program is addressing the important safety problems of handling sodium, this activity may still be perceived as a disadvantage by prospective plant operators.

1.9 MOLTEN-SALT REACTORS

1.9.1 Reactor/Fuel-Cycle System Description

The molten-salt reactor (MSR) concept is distinguished by the use of a molten mixture of inorganic fluoride salts as a circulating fluid (Figure 25). Fuel salt, consisting principally of the molten fluorides of lithium, beryllium, thorium, and uranium, flows through a reactor core vessel, where it is heated by the fission process. It is then pumped through a number of parallel loops to shell-and-tube heat exchangers, where the heat is transferred to an intermediate fluid (coolant salt), nominally a eutectic mixture of sodium fluoroborate (NaBF_4) and sodium fluoride (NaF). The fuel salt then returns to the reactor, while the coolant salt is circulated to the steam system to produce high-temperature, high-pressure steam for the generation of electricity.

The fluoride-salt mixtures have inherently low vapor pressures, allowing the systems to be operated at high temperatures ($\sim 700^\circ\text{C}$) with essentially no overpressure in the primary circuit. The resultant generation of high-temperature, high-pressure steam permits efficient production of electricity. Since the fission process occurs directly within the fuel salt in an MSR, the salt retains many of the fission products and is, accordingly, highly radioactive. This radioactivity requires that all items that are normally contacted by fuel salt -- the entire primary circuit and most appendages (for example, the drain tanks and the fuel cleanup system) -- be enclosed within a primary containment and be amenable to remote inspection and maintenance.

The fuel salt proposed for MSRs is capable of dissolving sufficient quantities of uranium and/or thorium to allow it to serve as a fissile medium, a fertile medium for production of plutonium or U-233, or both, as required by the fuel cycle. Because the salt constituents have significant neutron-moderating effects, the reactors are generally regarded as thermal-neutron systems. Hence, fast breeding is usually not considered, and moderators such as graphite are often added to the core to enhance neutron thermalization.

The fuel cycle selected as the NASAP reference is based on denatured uranium and is entitled the denatured molten salt reactor (DMSR). Concept characteristics are given in Table 48, and an illustration appears in Fig. 25. In its simplest form, the DMSR would be fueled initially with denatured U-235 (20% enrichment), and all subsequent additions would be limited to denatured U-235 or depleted U-238. The thorium concentration would be allowed to decline slowly with time (by Th-232 conversion to U-233) to help compensate for

Figure 25. Simplified Flowsheet for Molten-Salt Reactor.

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Table 48. Denatured MSR Characteristics

Power plant performance parameters	
Reactor thermal power output, MW	2,250
Electrical power output (MW)	
Gross	1,035
Net	1,000
Plant heat rate, Btu/kWhr	7,680
Core design and performance parameters	
Core heat output, MW	2,250
Core volume, liters (excluding reflector)	4.8×10^5
Core loading at end of life	
Heavy metal charged, kg	151,000
Fissile fuel, kg	0.8
Fuel type	Fluoride salt
Reactor inlet temperature, °C	565
Reactor outlet temperature, °C	704
End-of-cycle excess reactivity	Zero (0)
Initial core requirements, st/GWe	
ThO ₂	165
U ₃ O ₈	883
30-year cumulative requirements, st/GWe	
ThO ₂	165
U ₃ O ₈	1,888
Separative work requirements, 10 ³ SWU/GWe	
Initial core	805
30-year cumulative	1,719

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the buildup of U-238 from subsequent additions of denatured fuel. During the first five years, some depleted U-238 would have to be added to maintain denaturing as the dominant fissile uranium nuclide gradually shifts from U-235 to U-233. The subsequent additions of denatured U-235 would bring in more than enough U-238 for denaturing, and the fissile uranium isotopic enrichment would gradually decline. At the end of reactor life (30 years at 75% capacity factor), the fuel charge would be removed for storage and eventual disposal.* This fuel might conceivably be treated for recovery of valuable materials (uranium for reenrichment, Li-7, beryllium, thorium, etc.) and disposal of the fission products. Even with no credit for recoverable values in the spent fuel, the lifetime U_3O_8 requirement for an MSR on this fuel cycle would be only ~1700 metric tons.

Reactors based on other fuel cycles have also been considered. These include simple uranium burners, positive-gain breeders in the Th/U-233 cycle (which require highly enriched fissile fuel), and plutonium burners that might produce U-233 as a by-product.

The addition of an on-line fuel processing facility to the once-through DMSR for removal of fission products would permit the achievement of break-even breeding in a denatured reactor. This modification of the basic fuel cycle would require a larger capital investment in the plant, along with added operating complexity and expense, but it would also eliminate the need for fissile fuel additions during the life of the plant. In the limit, the fuel salt from one plant could be removed at the end of plant life and used for startup of a similar new plant. Implementation of the fuel processing option, which could be added to existing once-through DMSRs, would significantly reduce the plant's fuel cycle cost, as well as the demand for mined U_3O_8 .

The technology of the once-through fuel cycle is simpler than that of the breeder cycle, in view of the absence of the on-line chemical processing facility for fission-product cleanup. The development of the processing plant has long been recognized as being one of the most difficult technical tasks in MSR development and as probably being on the critical time path for the commercialization of breeding MSRs. In addition, the elimination of a processing plant might make the concept more attractive to utilities.

Although this fuel cycle would eliminate chemical removal of fission products, it would not eliminate all chemical treatment of the salt. Because

*Recovery of only the denatured uranium could be carried out at the reactor site just prior to decommissioning.

the primary circuit would occasionally be opened to the atmosphere for maintenance and because the fuel salt has limited tolerance for oxide contaminants (as from atmospheric oxygen and moisture), a hydrofluorination facility for oxide removal would probably be required. This facility would have to provide for occasional batchwise treatment of a fraction of the fuel salt. Although radioactive, it could be relatively simple to operate and maintain. The reactor operator would also be required to maintain close surveillance and regulation of the reduction/oxidation (redox) potential of the salt and its general chemical composition. All these requirements would be less demanding than fission-product processing. The continued addition of denatured fissile material and the simple chemical treatments required to maintain the fuel in operating condition might require replacement of a portion of the fuel charge during the 30-year operating life of the reactor.

1.9.2 Status of Technology

The molten-salt concept has been developed through the reactor experiment stage [the Molten-Salt Reactor Experiment (MSRE), an ~8-MWt reactor, operated successfully in the period 1965 to 1969]. Thus, basic technology for a commercial reactor is known but not fully developed. The engineering development needs for a commercial reactor include additional developing and proof-testing of the basic materials, developing a commercial source for the materials of construction (modified Hastelloy-N and graphite), scale-up of fuel and coolant pumps and heat exchangers, design of steam generators, and other aspects of reactor engineering technology.

The fuel cycle for the once-through DMSR is based on processing operations that were demonstrated in the MSRE. At most, this cycle might require infrequent, batchwise hydrofluorination (for oxide removal) and small-batch fluorination for uranium recovery, if limited salt discard is required for chemical reasons. Thus, the fuel technology for the once-through DMSR is expected to be feasible but needs further engineering development.

Development of the capability for processing DMSR fuel to remove soluble fission products could be largely independent of the reactor development effort; thus, the process might be commercialized on a time scale different from that for the reactor plant. A reductive-extraction/metal-transfer process was under development in connection with the Molten-Salt Breeder Reactor (MSBR) Program, and all basic steps in that process would be applicable to the treatment of DMSR fuel. (Some additional, but similar, steps might be required to retain protactinium and plutonium and to remove fission-product zirconium from the DMSR.) Aspects of the MSBR process have been demonstrated

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in the laboratory, and some steps have been tested on a pilot-plant scale. Nevertheless, substantial engineering development would be required to commercialize the process for any MSR.

A comprehensive report on the status of MSR technology was developed and published in 1972 by Oak Ridge National Laboratory. Since that time, a limited amount of development work (with some significant results, however) was performed in the period 1974 to 1976 in a few technology areas. An updated status summary with emphasis on changes from the 1972 status is currently in preparation.

1.9.3 Safety Issues

Breach of the reactor-coolant safety boundary, resulting in a spill of highly radioactive salt into the primary containment, will probably be the design-basis accident for DMSRs. Possible initiators of this accident include pipe failure, missiles, and pressure or temperature transients in the primary-coolant system. Research to date indicates that these initiators are highly unlikely, but this would be a prime area of concern for future safety R&D. In the event of a salt spill, a system of drains would act to channel the salt to the cooled drain tank. The primary-system containment, defined as the set of hermetically sealed concrete-shielded equipment cells, would probably be designed to withstand such a spill, but cleanup operations would be difficult.

A unique safety feature of the MSR is that, under accident shutdown conditions, the fuel material would be drained to the drain tank, which contains a constantly operating cooling system, instead of requiring activation of an idle ECCS, as in the LWR. The reactor and containment would be designed so that the fuel salt would reach the drain tank under any credible accident conditions. It should be noted that the decay heat is associated with a very large mass of fuel salt, and therefore the possibility of melt-through of the primary system seems remote.

The neutronics of the MSR are favorable from a safety point of view. Because of on-line refueling,* the amount of excess reactivity that must be controlled (and could be available under accident conditions) is small ($<1\% \Delta k/k$). Because the temperature reactivity coefficient of the fuel salt is negative and the delayed-neutron fraction is unusually low, the power reacts

*Makeup fuel can be added at will in arbitrary increments.

rapidly to damp temperature fluctuations. This response is large enough to limit any credibly rapid reactivity change. The small positive void coefficient and the extremely high boiling temperature of the fuel salt (~800°C above normal operating conditions) make core voiding an unlikely accident initiator. Also, the main heat source is lost along with the cooling capability in a voided region. The MSR core is not difficult to cool after short transients, because there is no large amount of heat stored in solid fuel that must be removed rapidly after shutdown to avoid meltdown. However, the decay heat of the fuel salt must eventually be removed, probably by the drain-tank cooling system. Continued circulation and cooling in the primary circuit may also provide satisfactory heat removal for some situations.

1.9.4 Environmental Issues

MSRs would not affect the environment in any way that is fundamentally different from reactors now operating. The expected higher thermal efficiency will reduce the net thermal discharge per unit of electrical output. No gaseous or liquid radioactive effluent would be discharged from the plant during normal operation. Minor amounts of such effluents would result from maintenance operations involving opening of the primary system.

Onsite reprocessing, if used, would operate in a closed cycle with no effluent discharges. Most high-level radioactive wastes would be retained in the fuel salt and in gas containers for eventual transport to a waste repository.

A large amount of tritium (~2400 Ci/day) is generated in MSRs as a result of neutron reactions with the lithium in the fuel salt. This tritium will diffuse through metal walls such as heat exchanger tubes, providing a potential route for transport of gaseous tritium through the secondary-coolant loop to the steam generators. Recent experiments have shown that tritium is oxidized in the secondary coolant (sodium fluoroborate), which can block further transport of tritium. As a result, it is expected that the release of tritium to the environment from MSRs may be no greater than that from LWRs and within current guidelines.

MSRs with processing would produce a concentrated fission-product waste similar to the high-level waste that would be produced in a conventional fuel reprocessing plant and that could be prepared for ultimate waste disposal in an analogous manner. To date, no development work has been done on methods of preparing fission-product wastes from MSRs for ultimate disposal.

1.9.5 Licensing Issues

The most distinguishing characteristic of the MSR system from the standpoint of reactor safety and licensing is that the primary system coolant serves a dual role as both the medium in which heat is generated within the core and the medium that transfers heat from the core to the primary heat exchangers. Thus, the entire primary system will be subject to both high temperatures (700°C at core exit) and high levels of radiation by a fluid containing most of the daughter products of the fission process. However, the primary-system design pressure will be relatively low.

The major licensing issue is the absence of barriers normally present in power reactors: the solid fuel, the fuel cladding, and the reactor vessel. The entire reactor-coolant system theoretically provides one level of confinement analogous to the fuel cladding in a solid-fuel reactor.

Although much larger than the analogous LWR system, it will not be subject to rapid thermal transients with the possibility of melting, as in various LWR and LMFBR accident scenarios. Two levels of confinement would be provided in addition to the primary loop in the present DMSR concept, the primary containment cell and the secondary containment building. Developing a reactor-coolant system that will be reliable, remotely maintainable, inspectable, and structurally sound over the plant's lifetime will probably be a key factor in demonstrating ultimate safety and licensability.

Other issues that would likely arise during licensing considerations for an MSR include

- the ability to provide for reliable draining of spilled fuel salt (from a postulated breach of the primary-coolant system boundary to the drain tank),
- the ability to reliably drain salt from the intact reactor-coolant circuit,
- the ability to ensure residual heat removal in the fuel-salt drain tank,
- the ability to ensure cooling of the liners of the equipment cells (primary containment walls and floors) in the event that spilled fuel salt fails to flow to the drain tank,

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- the ability to prevent the escape of stored fission-product gases,
- the development of licensing criteria for a liquid-fueled reactor.

1.9.6 RD&D Needs

The following programs are viewed as necessary steps leading to the orderly development of a lead commercial MSR plant.

1. Base technology

- Fuels conception and development
- Reference and advanced structural alloys determination/selection
- Shutdown mechanisms
- Physics
- Analytical methods

2. Laboratory process and experimental facilities

- Fuels testing and demonstration
- Component adequacy/improvement and compatibility testing
- Shielding and safety development
- Reliability and availability testing
- Transient behavior
- Materials development, testing, and demonstration

3. Proof of principle

- Experimental operations
- Component characteristics in operation
- Structure and alloy behavior under radiation, temperature, and cycling environment

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- Startup and shutdown procedure development
- Operations requirement and on-line testing

4. Demonstration plant

- Scale-up and development
- Power generation
- Transient condition determination
- Full-scale operations requirements
- Maintenance procedure and equipment development
- Licensability requirements
- Environmental considerations

5. Lead commercial plant

- Full-power generation
- Cost/savings analysis
- Component and system reliability/availability enhancement
- Constructability and scheduling improvement
- Advancement of the art

To extend present technology to that required for the construction and operation of a 1000-MWe commercial MSR would require developing commercial sources of construction materials for the reactor vessel, primary heat transport system, and secondary heat transport system. In particular, supply sources for the modified Hastelloy-N alloy, Li-7, and graphite are required. Additionally, successful demonstration of capability to scale up the fuel and coolant pumps and heat exchangers is required, as is the design of the steam generators, preheaters, reheaters, and other aspects of reactor engineering technology.

Although an experimental reactor has been built and operated, a need still remains for the development of instrumentation and controls for an MSR. The

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technology required to provide instruments capable of indicating the status of the reactor core has not been developed. Similarly, valves capable of providing a positive shutoff of flow at one moment and permitting flow at the operators' direction have not been developed. Previous experimental work on such valves has disclosed that Hastelloy-N valves in fuel-salt service tend to weld themselves closed when in the shutoff position. These valves then remain closed and are ineffective in performing their intended service.

As described above, the feasibility of MSR operation has been demonstrated. To establish this technology, it is necessary to develop and demonstrate, first at reduced scale and later at full scale, the concepts, components, and systems of an MSR plant.

Such a program must provide all the essential components, equipment, and appurtenances of the plant and demonstrate ability to meet the lifetime requirements of the plant, utilizing both scheduled and unscheduled maintenance to ensure continued operations.

In addition to the three major construction projects (the demonstration plant, lead commercial plant, and standard commercial plant), a number of large-size test facilities will be required to demonstrate various aspects of the design/development process. These facilities include (1) a steam-generator-tube test stand, (2) a pump test stand, (3) a components test facility, (4) a model steam generator test installation, (5) a full-scale mockup molten-salt test reactor, (6) a process/instrument test facility, and (7) an engineering laboratory/decontamination facility.

As in all cases involving predictions, there is no assurance that a means of resolving outstanding engineering and developmental problems can be obtained in the time allotted for such work. Therefore, the time needed for resolving these problems, including the development of components and the successful demonstration of component reliability, availability, capability of in-service inspection, remote maintenance, and licensability, can only be estimated.

Commercial feasibility of an MSR is also dependent on the production of the required quantities of lithium fluoride from 99.99% Li-7. At present, there is no facility within the U.S. producing, or even capable of producing, this required salt in the quantity/purity required for the operation of an MSR. It is believed that U.S. industry could provide the remainder of the components to constitute the initial fuel charge without undue resource development.

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It appears that some additional small-scale experiments may be required to define more fully the physical properties of specific fuel compositions, behavior of actinide oxides in fluorides, solubility limits of trifluorides in the MSR fuel, and optimum procedures for removal of oxides from contaminated fuels. However, no experiments are believed to be required to prove the principle of the MSR or of the denatured fuel composition.

The status of MSR technology is sufficiently advanced that a decision to build a small DMSR demonstration plant could be made by 1990, provided that a successful technology development program is undertaken in the interim. The lead commercial plant would be two technological generations beyond, so that little commercial impact would be possible within the NASAP planning horizon (2025).

A summary of the RD&D costs of the DMSR concept is given in Table 49. The DMSR probably cannot be developed and commercially deployed in less than 40 years.

1.9.7 Perceived Advantages and Disadvantages

Advantages offered by the MSR, in general, are associated principally with its high level of neutronic performance resulting from the absence in the core of structural material that would parasitically absorb neutrons. In an MSBR (with on-line fuel processing to remove fission products), a breeding ratio of up to about 1.07 could be realized with a low fissile inventory. For the DMSR without fuel reprocessing, a very long (30-year) once-through fuel cycle can be attained with a U_{308} resource requirement that is approximately one-third of that of a once-through LWR. In addition, the DMSR could eliminate shipments of plutonium-containing materials from the reactor site except at the end of plant life.*

The principal disadvantages of the DMSR are associated with its new and generally unfamiliar technology and the long time scale for deployment. This will act as a barrier to utility acceptance of the concept. The fuel is a liquid that is "unclad" and circulates throughout the entire reactor-coolant circuit. Thus, the reactor-coolant pressure boundary becomes the first barrier for the confinement of fission products, and the entire circuit is made highly radioactive, imposing a requirement for inspection and maintenance by totally

*The net amount of plutonium produced in a 1-GWe DMSR during 30 years of operation at 75% plant factor would be much less than that for a comparable once-through LWR, because most of the plutonium would be consumed in situ.

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Table 49. DMSR Estimated RD&D Cost Summary

RD&D area	Estimated cost ^a (\$ million)
<u>Research and development</u>	
Basic research	300-600
Development facilities	1500-2500
Regulatory studies	1000-2000
Subtotal	<u>2800-5100</u>
<u>Demonstration facilities</u>	
Proof-of-principle plant	800-1200
Demonstration plant	1200-1600
Lead commercial plant (1000 MWe)	1200-1600
Total estimated cost	<u>6000-9500</u>

^aEstimated in 1978 dollars. Does not include interest during construction.

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remote procedures. The liquid salt also releases nearly all the noble-gas fission products and some tritium,* which would have to be collected and stored for decay and ultimate disposal. In addition, the accumulation of fission products for 30 years would lead to a higher inventory of nongaseous fission products at a DMSR site than at an LWR site. Ultimate disposal of the spent fuel from a once-through MSR fuel cycle is a problem which has not been addressed and which may be more difficult than the disposition of solid fuels.

*Most of the tritium would transfer into the cover gas over the secondary salt, from which it would have to be collected and stored.

1.10 GASEOUS-CORE REACTORS

1.10.1 Reactor/Fuel-Cycle System Description

The gaseous-core reactor (GKR) concepts (Figure 26) are characterized by the use of a fissioning gas mixture to sustain the fission reaction. This circulating fuel gas also serves as the primary heat transport medium. Concept characteristics are given in Table 50. The fuel-gas systems can be operated at high temperatures (725°C), allowing more efficient production of electricity. Because the fission process occurs directly within the fuel gas, the gas retains many of the fission products and will be highly radioactive. This radioactivity requires that all items normally contacted by fuel gas (i.e., the entire primary circuit) be enclosed within a primary containment and be amenable to remote inspection and maintenance.

Two fuel-gas mixtures with sufficient quantities of uranium and plutonium to serve as a fissile medium, $\text{UF}_6\text{-He}$ and $\text{UF}_6\text{-CF}_4$, have been proposed for GKR. These mixtures would also have to include a substantial concentration of free fluorine gas to stabilize the plutonium (formed by neutron capture in U-238) as PuF_6 . Because a thermal neutron spectrum is needed for efficient nuclear performance, moderators such as graphite or beryllium are added to the structure and reflector regions to enhance neutron thermalization.

Several different fuel cycles have been proposed, including the graphite-moderated heterogeneous GKR and the mixed-flow gaseous-core reactor (MFGCR). The MFGCR has been characterized for NASAP and will be discussed as representative of the GKR technology.

In its proposed form, the MFGCR uses seven core cavities to permit fissioning of the fuel gas, $\text{UF}_6\text{-CF}_4$. These core cavities would be surrounded by beryllium moderators and imbedded in a molten-salt blanket. Heat would be removed from the fuel gas by a gas-to-gas heat exchanger. The intermediate helium gas would then be cooled by a steam generator, which would produce steam for a conventional electrical generating plant.

The use of the molten-salt blanket allows the system to be operated without the introduction of fresh fissile fuel after the initial reactor loading. The molten-salt blanket would require reprocessing to remove uranium for transfer to the fuel gas (and possibly fission products), while the fuel gas would require reprocessing to remove fission products. The two processing

Figure 26. Mixed-Flow Gaseous-Core Reactor Conceptual Configuration.

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Table 50. MFGCR Characteristics

Reactor thermal power output, MW	2,610
Electrical power output, MW	
Gross	1,100
Net	1,000
Plant heat rate, Btu/kWhr	8,900
Core heat output, MW	2,544
Reactor inlet/outlet temperatures, K (°F)	600 (1,000°)
Core specific power, kW/kg U-233	48,000
Core power density, kW/kg uranium	4,800
Core fuel inventory (10% enrichment)	
U-233, kg	53
U-238, kg	477
External inventory	
Out-of-core primary system (10% enrichment)	
U-233, kg	111
U-238, kg	1,000
Blanket (13% enrichment)	
U-233, kg	38
U-238, kg	256
Blanket process system (13% enrichment)	
U-233, kg	9.5
U-238, kg	64
Fissile U-233 loading, kg	212
Fissile plutonium inventory (maximum): Pu-239 + Pu-241, kg	66
Other fissile inventory: U-233 (in startup increments or from Pa-233 decay), kg	123
Initial core requirements, st/GWe	
U ₃ O ₈	65
ThO ₂	147
Annual reload requirements, st/GWe	
U ₃ O ₈	0.11
ThO ₂	1.5
30-year cumulative requirements, st/GWe	
U ₃ O ₈	68
ThO ₂	194
Separative work requirement (initial core only), SWU	64,000

facilities also provide for continuous interchange of uranium between the fuel gas and the blanket.

In the simplest form, the MFGCR would be fueled initially with medium-enriched denatured U-235, and all subsequent fuel additions would be limited to U-238 and thorium. Other chemical additions likely would be required to maintain the desired chemical compositions. At the end of a 30-year reactor life at 75% capacity factor, the fuel charge and the blanket salt would be removed for storage and eventual disposal or reuse. These materials might conceivably be reprocessed for recovery of valuable materials (uranium for reenrichment, Li-7, beryllium, and thorium) and disposal of the fission products. Even with no credit for recoverable values in the spent fuel, the lifetime U_3O_8 requirement for an MFGCR on the proposed fuel cycle would be only 68 st/GWe.

If it is presumed that the core and blanket concepts can be engineered, the successful development and implementation of the processing plants will be one of the most challenging difficulties of the concept and could be on the critical path for MFGCR commercialization. In addition, the processing of fuel gas and molten-salt blanket will add substantial complexity to plant operation and require a highly competent, specially trained operating staff.

1.10.2 Status of Technology

The MFGCR concept is essentially unproved. Although simple critical experiments have been performed with UF_6 , the feasibility of sustaining a power-producing fissioning gas in a flowing system has not been demonstrated, even at moderate temperatures, and extrapolation to the proposed high temperatures and pressures could be substantially more difficult. The ability of the gas to retain the bred plutonium in a gaseous form at practical F_2 pressures has not been demonstrated and is seriously questioned. The required technologies for reprocessing the fuel gas and molten-salt blanket have not been developed beyond vague conceptual designs, nor has material compatibility with the high-temperature fuel gas been demonstrated. Even after these basic questions have been addressed, an enormous engineering development program would be required to produce a working system of commercially acceptable size. Even conventional items such as heat exchangers, pumps, valves, and instrumentation would require considerable development beyond their current status to accommodate the hot fuel gas.

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In summary, the technology for the MFGCR is considered to be in the earliest stages of development, and the technical feasibility of the concept is questionable.

1.10.3 Safety Issues

Breaches of the fuel-gas primary-coolant loop and of the molten-salt blanket system will probably be the design-basis accidents. Either of these accidents would release substantial amounts of fission products to the primary containment. In addition, failure of the fuel loop would spread condensed uranium and plutonium compounds throughout the containment. A blanket failure and spill would also cause substantial contamination, primarily uranium, which would tend to remain with the nonvolatile salt. The high pressure and temperature of these systems combined with the corrosive nature of the fluids will make these accidents difficult to prevent and contain.

Because the processing systems will be necessarily connected to the fuel-gas and molten-salt systems, positive isolation valves will probably be necessary to prevent system blowdown through the processing systems. Positive shutoff valves for this application will be difficult to design and demonstrate because of the operating conditions and corrosive fluids.

Verification of fuel-gas and molten-salt system boundaries throughout the plant life will be prompted by these major safety concerns and could affect plant operation through in-service inspection requirements and preventive equipment-maintenance procedures. A related problem will be associated with the interface between the fuel-gas and molten-salt systems. The safety significance of various sizes of leaks between the two systems has not yet been established, but it could be substantial.

The potential for criticality accidents is low, because the loss of coolant also implies loss of fuel and probably reactor shutdown. Possible criticality accidents in the processing systems will have to be addressed in the plant design.

Transient behavior of the reactor has not been analyzed; thus, some potential may exist for unstable transient behavior, which would cause power excursions. However, if the fuel-gas temperature coefficient of reactivity is strongly negative, as expected, it should have a major stabilizing effect on the dynamic characteristics of the system. Pressure-relief methods will probably have to be incorporated into the design to protect the primary systems.

1.10.4 Environmental Issues

The GKR's would not affect the environment in any way that is significantly different from reactors now operating. The expected higher thermal efficiency will reduce the net thermal discharge per unit electrical output. No gaseous or liquid radioactive effluent would be discharged from the plant during normal operation. Minor amounts of such effluents would result from maintenance operations involving opening of the primary system.

Onsite reprocessing would operate in a closed cycle, with no effluent discharges. High-level radioactive wastes would be extracted by reprocessing for eventual transport to a waste repository.

A significant amount of tritium may be generated in MFGCRs as a result of neutron reactions with the lithium in the blanket salt. Tritium will diffuse through metal walls such as heat exchanger tubes, providing a potential route for transport of gaseous tritium. Without special tritium-handling features, the release of tritium to the environment from MFGCRs may be significantly greater than that from LWRs. If helium is used as the secondary coolant for the blanket salt, tritium confinement will be relatively easy, because tritium can be readily removed from helium.

GKR's with processing would produce a concentrated fission-product waste similar to the high-level waste that would be produced in a conventional fuel-reprocessing plant. Concentrated fission-product waste could be prepared for ultimate waste disposal in an analogous manner to high-level waste from LWR recycle operations. To date, no development of methods of preparing fission-product wastes from GKR's for ultimate disposal has been performed.

1.10.5 Licensing Issues

The most distinguishing characteristic of the MFGCR system, from the standpoint of reactor safety and licensing, is that the primary-system coolant serves as both the medium in which heat is generated within the core and the medium that transfers heat from the core to the primary heat exchangers. Thus, the entire primary system will be subject to both high temperatures and high levels of radiation by a fluid containing many of the daughter products of the fission process. The entire reactor-coolant system serves the same level of confinement as the cladding in a solid-fuel reactor and may be subject to relatively rapid thermal transients because of the potential for wide temperature swings in the circulating gas. The present MFGCR concept calls for two additional levels of confinement to help prevent radioactivity

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releases to the environment. However, developing a reactor-coolant system that will be reliable, remotely maintainable, inspectable, and structurally sound over the lifetime of the plant will probably be the key factor in demonstrating ultimate safety and licensability.

Other issues that would likely arise during licensing considerations for the MFGCR include the ability to

- provide for reliable containment of the fuel gas,
- ensure residual heat removal from the fuel gas and from the empty primary system,
- control power transients and possible reactor instabilities,
- prevent the escape of stored fission-product wastes.

1.10.6 RD&D Needs

The GKR is in an early stage as far as the research, design, and development program is concerned. Before the GKR can be developed into a practical power plant, basic research must be performed to demonstrate the feasibility of the concept. The principal technological barrier is to demonstrate that the proposed materials of construction can withstand the fuel gas at elevated temperature and pressure.

Little is known about the behavior of the fuel and its chemistry at elevated temperatures. The stability of the UF_6 , PuF_6 , and fission-product components in the reactor environment must be established. Areas requiring study include fuel-gas radiolytic dissociation, fuel-gas stability over the temperature range of interest, and fuel-gas corrosivity when exposed to containment materials. In addition, the thermal and chemical properties of fission-product fluorides and the effect of fission products and radiation damage on core structural materials require detailed study. Fuel systems and expected long-term component service life should be demonstrated early in the program. The same is true, to a lesser degree, for the molten-salt portion of the fuel cycle. In addition, the reactor-physics aspects of the proposed core materials and fuel-gas mixtures must be proved sound by experimental programs.

The sustainer-blanket design is based on MSR technology to take advantage of the development that has gone into that concept. Utilization of MSR

technology might reduce the lead time required to develop the sustainer MFGCR. Early demonstration reactors may be constructed without the blanket and may be fueled with some nominal enrichment of U-235.

Additional development problem areas are fuel-gas cleanup, blanket processing, system dynamics and control, safety systems and their analysis, and design of specific components and systems.

Early RD&D activities should be concentrated on questions of technical feasibility, including compatibility of construction materials with $\text{UF}_6\text{-F}_2\text{-CF}_4$ at high temperatures, thermal and radiation stability of UF_6 and PuF_6 , reactor physics, and behavior of fission products created in the gaseous fuel.

The MFGCR plant would use a mix of technologies. The technology required for the steam system is reasonably well established. All other areas require at least some development to establish commercial acceptability. Applicable technology is being developed in the following areas:

- Four UF_6 nuclear experiments have operated over the past 20 years but at very mild conditions compared with those proposed for the MFGCR.
- Substantial experience in handling UF_6 gas is available from the development of enrichment facilities.
- Molten-salt-blanket technology would use the same data base as the MSBR.

Table 51 outlines the programs required to achieve a commercial GCR plant. They include basic research, laboratory and experimental facilities, test loops, a proof-of-principle plant, a demonstration plant, a lead commercial plant, and, finally, the standardized commercial plant.

The following additional activities are assumed to be included in the program:

- development of unique instrumentation and control systems,
- development of unique personnel-monitoring and protection systems,

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Table 51. MFGCR Integrated Projects Required to
Achieve Commercialization

Basic research data

- Nuclear design and systems evaluation
- High-temperature UF₆ chemistry
- Fluid dynamics of convection energy transport
- Thermal and chemical properties of fission-product fluorides
- Fuel-gas and containment-materials compatibilities

Laboratory process and experimental facilities

- Reactor-loop critical assembly
- Fission-product extraction loop
- Fission-product plate-out loop
- U-233 extraction and fluorination loop
- Plutonium extraction and fluorination loop
- Blanket uranium and plutonium extraction fluorination and reduction loop
- Materials development and testing
 - Long-term corrosion and erosion resistance to fuel gas
 - Fission products and radiation damage
 - Physical properties at expected temperatures and pressures

Component development and testing

- Intermediate heat exchanger
- Instrumentation
- Control and power lead seals
- Fuel-gas cleanup and reprocessing
- Waste handling equipment
- Circulators
- Emergency cooling equipment
- Reactor-vessel internals
- Remote handling equipment

System development and testing

- Remote handling systems
- Remote cleanup systems
- Remote maintenance systems
- Criticality control systems
- Containment systems
- Fuel/coolant injection systems

High-temperature loop (1 MWt)

Proof-of-principle plant (50-100 MWt)

Demonstration plant (1000 MWt)

Lead commercial plant (2600 MWt)

Standardized commercial plant (2600 MWt)

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- development of unique cleanup and high-vacuum air cleaning (HVAC) systems for primary containment systems and for accessible areas in the reactor building,
- development of unique fire-protection and explosion-proof systems,
- development of a reactor-shield cooling system and a molten-salt-blanket heat-removal system,
- development of post-accident heat-removal and cleanup systems,
- development of fission-product and radioactive-waste storage systems,
- reactor kinetics program for determining responses to normal control transients such as load changes, abnormal operation, etc. (a model development program for transients is needed),
- safety/accident analysis to determine the hazards for the plant concept (fire, explosion, and other accident modes must be investigated, as well as nuclear-related events).

The status of GKR technology would require extensive RD&D, leading to a possible decision by about 1990 to 1995 to build a proof-of-principle plant. A summary of the RD&D cost estimate for the MFGCR is given in Table 52. The MFGCR, if feasible, probably cannot be developed and commercially deployed in less than 50 years.

1.10.7 Perceived Advantages and Disadvantages

The advantages of the MFGCR stem from its low fuel inventory and very good uranium resource utilization.

The greatest disadvantages of the MFGCR will be its great complexity and unfamiliar technology. Flowing fuel gas, intermediate-coolant loops, molten-salt-blanket system, fuel-gas and blanket reprocessing, and conventional steam plant must all operate together. The complexity of the plant and the requirement for much remote maintenance will have an adverse effect on plant reliability and capacity factor. The unfamiliar technology is also likely to meet resistance from potential utility users. Handling and containing the radioactive fuel gas is likely to be an ongoing safety problem that will be viewed unfavorably by utilities.

Table 52. MFGCR Estimated RD&D Cost Summary

RD&D area	Estimated cost ^a (\$ million)
<u>Research and development</u>	
Basic research	1,200–2,500
Development facilities	2,800–4,000
Regulatory studies	1,500–2,500
Subtotal without major plant construction	5,500–9,000
<u>Demonstration facilities</u>	
Proof-of-principle plant	900–1,500
Demonstration plant	1,400–2,000
Lead commercial plant (1000 MWe)	1,400–2,000
Total estimated cost	9,200–14,500

^a Estimated in 1978 dollars. Does not include interest during construction.

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If the GKR were technologically feasible, the overall time scale of commercialization would be longer than that of the DMSR, and cost would be greater.

1.11 TOKAMAK FUSION-FISSION HYBRID REACTOR

The tokamak fusion-fission hybrid reactor (THR) concept that was characterized and assessed as part of NASAP (Figure 27) represents one of many possible fusion hybrid systems. The characterization was limited in scope and was of uncertain quality, thus making equally uncertain an extrapolation of the data presented in the study to a commercial adaptation of the hybrid. The assessment of the specific concept presented here should therefore not be construed to apply generally to fission-fusion hybrids. The hybrid concepts and sequences of development leading to commercialization are subject to debate, and the concept as depicted probably does not represent an optimum design.

1.11.1 Reactor/Fuel-Cycle System Description

The Tokamak Fusion-Fission Hybrid Reactor (THR) concept utilizes a tokamak deuterium-tritium (D-T) fusion reactor as a neutron source surrounded by a blanket containing fertile material. The neutrons emitted from the plasma are slowed down in the blanket and captured by the fertile nuclei to produce fissile material. Analytical studies have shown that blankets containing Th-232 or U-238 have potential for producing sufficient fissile fuel to operate several fission reactors and simultaneously convert to electrical power the energy deposited in the blanket.

A preconceptual design of a commercial tokamak hybrid reactor has been carried out to provide a basis for studying this alternative. The tokamak was proposed to operate with a D-T plasma that produces 1140 MW in the ignited mode. The plasma region has a 5.4-m major radius and a 1.0-m minor radius. The confining toroidal magnetic field is maintained by D-shaped Nb₃Sn superconducting magnets with a maximum field of 12 teslas at the coil.

The satisfactory performance of the THR depends on the satisfactory operation of the fusion device and the selection of the fertile fuel form. Tritium for the operation of the fusion device will also be bred in the blanket. Trade-offs may be required to minimize the competition for neutrons by appropriate selection and composition of the fissile and fusile breeding materials. Preliminary investigations have been made on a number of fertile fuel alternatives for the tokamak hybrid. These include (1) a once-through cycle using a natural-uranium-fueled hybrid for power production only, (2) recycle to thermal reactors with the dual role of power and fuel production, (3) refresh

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Figure 27. Tokamak Hybrid Reactor Cross Section.

fuel cycle where the hybrid is used to reenrich PWR fuel, and (4) plutonium-thorium fuel cycle where the hybrid breeds U-233. In the fourth alternative, the U-233 is sold, and the plutonium is recycled. These fuel-cycle options have also been proposed for use with the linear-accelerator-driven fuel regenerator and reactor options discussed in Sections 1.8.1 and 1.8.2. Option 2 was treated more extensively in the THR characterization; this fuel cycle will require reprocessing and fabrication facilities for both the blanket fuel and the product LWR fuel.

The performance characteristics of the THR operating with these fuel cycles are based on one-dimensional radiation transport calculations and idealized geometries. The blanket materials are homogenized, thus removing the effects on performance of neutron losses caused by radiation streaming through voids in the blanket. However, the results of these calculations are encouraging and indicate fissile fuel production rates on the order of 1.5 fissile atoms per fusion neutron, depending on the fertile material and the competition for the plasma neutrons when tritium breeding is simultaneously required.

1.11.2 Status of Technology

The THR concept is in the very earliest stages of development; the hybrid concept is being investigated at a number of laboratories in conceptual design studies. The THR concept is not a proven alternative, and its adoption as a viable energy or fuel producer depends on the development of a fusion driver with favorable ignition characteristics, burn-time duration, and duty factor. The development of a tokamak with these plasma characteristics cannot be anticipated until near the turn of the century and depends heavily on the success of the ongoing fusion research and development program.

The feasibility of the THR blanket has been investigated using one-dimensional neutronics methods. The nuclear performance using idealized geometries is compared as a function of fertile fuel composition, tritium breeding material, and structural components with calculations using a homogenized mixture in cylindrical shells surrounding a D-T plasma zone. The nuclear data for some of the materials used in these analyses have not been fully evaluated, so results must be treated as tentative.

The long-term operation of fusion devices depends on the lifetime of the first structural wall and other potentially sensitive components (e.g., fuel cladding, cooling tubes, insulators) in the intense neutron environment. The required material properties are beyond the state of the art. These selections of materials for use in the first structural wall and in diverters (if

they are used) will depend on the results of continuing materials research. The compatibility of these materials with the fertile, cooling, and moderating materials used in the blanket must also be established. Conditions of high-energy neutron flux, high temperature, and thermal-mechanical cycling will impose severe operating limits on the system, and the integrity of the blanket and other vital components must be ensured. Both the cooling system for the first wall and the hybrid blanket will require development.

Other areas requiring comprehensive technological advances include magnetics, cryogenics, diagnostics, and fuel handling.

1.11.3 Safety Issues

The design-basis accident for the fusion-fission hybrid has not yet been defined. The fueling, radiation, and thermal-mechanical properties for the fusion driver will be considerably different from those found in conventional fission reactors. The inclusion of lithium in the hybrid blanket introduces additional hazards from the build-up of fusion products in the fissile-fuel-breeding modules.

The fusion component of these systems will introduce significant safety problems, which will require the development of engineered safety systems and passive safety design features. The pulsing of the tokamak will impose cyclic stresses on the blanket containment. Effects of an imbalance in the magnetic field caused by failure of some of the poloidal or toroidal magnets may result in a breach of the reactor vessel leading to release of the blanket fuel and fusile material to containment. Large amounts of tritium will be present in the fuel, and the technology and criteria for containment design have not yet been developed. The confinement of tritium has been cited as a potential hazard in hybrid reactors. Current fusion RD&D programs are addressing tritium monitoring, confinement, and recovery.

Containment of liquid lithium in the blanket must be maintained because the accidental release of lithium could result in an explosion, if it contacts water or concrete. Other hybrid designs may not require the use of liquid lithium.

An intensive research and development program to understand materials properties under high and low temperature, intense radiation, and cyclic loading conditions must be pursued to qualify materials for hybrid reactors. The consequences of a LOCA must be studied. The power density in hybrids is

typically less than that in LWRs, and the THR is designed to provide decay-heat cooling by natural convection after an LOCA. Most hybrid systems that have been proposed eliminate the possibility of criticality regardless of reconfiguration geometry.

Loss of onsite power could cause loss of coolant flow and require immediate reactor shutdown. However, loss of onsite power will result in the immediate loss of the fusion driver neutral beam systems, which will prevent the start of any new plasma ignition. The superconducting magnets will then need an active trip to interrupt plasma confinement and stop the fusion reaction. Inclusion of a passive impurity injection system to provide plasma cooldown may also be advisable. The vacuum pumping system and tritium processing stream will also have to be investigated for problems caused by loss of power.

Mitigation of potential fuel-handling and processing accidents must consider the fissile-fuel-laden blanket modules as well as tritium. Some modules may require extensive disassembly for transportation, processing, or storage. Because the geometry and composition of these vary with design, the maintenance, handling, shipping, and processing requirements must be established before effective accident analysis can begin.

1.11.4 Environmental Issues

Radioactive materials in a hybrid power plant may be found in four forms: tritium, actinides, activation products, and fission products. Tritium may be released by permeation at high temperatures and by leakage through valves and seals. Accidental failure of tritium-containing components could potentially result in the release of large amounts of tritium.

The fission modules will contain significant levels of actinides and fission products. Events leading to fuel failure could potentially cause release of these radioactive materials. Characterization of the blanket fuel behavior under accident conditions will require extensive analysis, and the inventory of specific isotopes generated by the fusion neutron spectrum must also be established to evaluate the consequences of a blanket disruption.

The 14-MeV fusion neutron source will result in highly activated structural materials, the presence of which will require design of shielding to protect plant personnel during maintenance operations. Control and disposal of radioactive waste materials and decommissioning procedures will also require additional evaluations.

The THR is expected to operate at high temperatures and high thermal efficiency. Therefore, heat rejection and water use should be favorable compared with other reactor systems. The THR will require reprocessing and fabrication facilities for both the THR fuel and the burner reactor fuel. The environmental impact of these facilities would be the same as for similar facilities for other fuel cycles. Uranium mining and milling environmental impacts should be greatly reduced from those of the LWR with low-enriched uranium feed, and enrichment services would not be required for the THR. Use of uranium enrichment tails or reprocessed uranium from other fuel cycles could further reduce the mining and milling impacts.

1.11.5 Licensing Issues

Although a specific design-basis accident for the THR has not yet been defined, a number of potential design-basis accidents can be identified. These include the release of tritium, a plasma energy dump, inadvertent criticality, fission-product release, electromagnetic system excursions, vacuum and cryogenic system breakdowns, fuel-pin failures, loss of coolant, accidents caused by hazardous materials such as hydrogen and lithium, and missile-generated accidents.

The containment design criteria must be sufficient for enclosing a reactor that will contain fissile fuels, have a high thermal energy storage, have decay-heat removal problems, and contain hazardous materials such as hydrogen and lithium. If liquid lithium is used, the entire liquid-metal safety problem must be addressed. Due to these factors, the licensing effort for the THR containment is expected to differ significantly from that for the LWRs.

In a final THR design, it will be necessary to investigate compliance with all relevant regulatory guidelines. Although a number of existing guidelines, which apply mainly to LWR technology, can be applied to the THR concept, extensive development of new guidelines to fit the fusion/fission hybrid technology is expected. New or significantly revised criteria may be anticipated to account for the unique combination of a hybrid blanket with a fusion driver, the use of tritium, the use of hazardous materials such as hydrogen and liquid lithium, and the use of high-energy and cryogenic electromagnetic systems.

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1.11.6 RD&D Needs

The THR will require an extensive RD&D effort to bring it to the demonstration stage. The THR uses an ignited tokamak as the fusion driver. Before such an ignited tokamak is obtained, at least the following components must be developed:

1. Neutral beams in the 100-MW class with 150-kV operating energies or equivalent microwave heating systems.
2. Large superconducting magnets. THR utilizes Nb₃Sn as the superconductor, which allows higher fields than niobium-titanium but is considered by many to be more difficult to work with.
3. Durable first-wall material.
4. Large tokamak start-up systems.
5. Impurity control systems.
6. Pellet or gas-puffing fueling systems.

In addition, more must be learned about plasma confinement scaling with size, temperature, and density and about alpha-particle confinement. All these areas are being addressed in the controlled-fusion RD&D program, but the present schedule will not lead to an experimental device incorporating all these features before the early to middle 1990s.

Additional areas that must be addressed concerning the fission as well as the fusion aspects of the THR are discussed below.

1. Nuclear fuel. Improvements are needed to make the fissile and fusile fuel compatible with the hard anisotropic neutron spectrum. Significant material and geometry testing and development will be required.
2. Blanket modules. The configuration of the blanket must allow proper cooling and efficient breeding while maintaining assurance against possible criticality and providing for a reasonable operating life. This will require extensive neutronic, thermal-hydraulic, and stress modeling.
3. Remote handling. The remote handling capabilities required for a tokamak reactor to be developed during the reactor design process will be considerably complicated by use of a hybrid blanket.

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4. Primary cooling system. The primary cooling system design will be dependent on blanket module configuration and energy density. Most of the development required is more appropriately assigned to the module rather than to the coolant system.

5. Tritium breeding and extraction. Extraction of bred tritium from either primary coolant or stationary breeding material is being addressed in several small programs. The breeding material and the primary coolant may have to be tailored to optimize fissile fuel production.

6. Reactor instrumentation and control. The THR will require instrumentation and control similar to that for a pure fusion tokamak combined with a system similar to that employed in a fission reactor. Development of such a combination should not require significantly more effort than a pure fusion system.

7. Emergency cooling. Developing an emergency cooling system compatible with relatively high local energy densities, the possible presence of liquid lithium, and the primary cooling system will present difficult challenges. The need for such a system may place constraints on fuel module design and performance, tritium breeding material, and primary coolant choice.

8. Safety and licensing. An extensive effort will be required to address the safety and licensing aspects of a fusion reactor coupled with the fission products and actinides in the hybrid blanket. Key issues include criticality, afterheat, first-wall fatigue or burn-through, tritium containment, lithium chemical reactions, magnet failures, magnetic field effects, cryogenic hazards, and high-voltage hazards.

In addition to the required fusion research, the THR will require development of blanket fuel and plant equipment. The RD&D program will require construction and operation of a prototype THR. The demonstration hybrid reactor would produce substantial quantities of fissile fuel and electric power with reliability acceptable to a utility. It would be a licensed facility operating on a utility grid. All technical problems would be successfully addressed in such a device, and the only additional steps necessary for commercialization would be moderate scale-up in size and improvement in production techniques to construct the lead commercial plant.

The summary of the RD&D costs of the THR is given in Table 53. The THR probably cannot be developed and commercially deployed in less than 50 years.

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Table 53. THR Estimated RD&D Cost Summary

(Assumes fusion research and development continues)

RD&D area	Estimated cost ^a (\$ million)
<u>Research and development</u>	
Basic research	1,500-3,000
Development facilities	1,700-3,500
Regulatory studies	1,500-2,500
Subtotal without major plant construction	4,700-9,000
<u>Demonstration facilities</u>	
Proof-of-principle plant	1,600-2,500
Demonstration plant	2,800-4,000
Lead commercial plant (1000 MWe)	2,800-4,000
Total estimated cost	11,900-19,500

^aEstimated in 1978 dollars. Does not include interest during construction. Estimated costs are incremental to fusion program costs.

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1.11.7 Perceived Advantages and Disadvantages

The greatest potential advantage of the THR is that it produces substantial quantities of fissile fuel which can be used to fuel reactors. The THR would also produce significant electrical power. As many as 5 to 20 reactors may be fueled by a single fusion-fission hybrid depending on the design of the hybrid, and the design and fuel cycle chosen for the supported reactors. The THR concept assessed in NASAP will support five LWRs on the uranium-plutonium fuel cycle.

The requirements for the fusion driver for the hybrid are less stringent than those for pure fusion devices, and thus the hybrids may become feasible earlier than pure fusion plants.

One disadvantage of the THR concept is that the appropriate fusion driver has not been adequately demonstrated. Moreover, the concept combines two technologies -- fusion and fission -- each with its attendant problems and difficulties.

1.12 FAST MIXED-SPECTRUM REACTORS

1.12.1 Reactor/Fuel-Cycle System Descriptions

The fast mixed-spectrum reactor (FMSR) is a new concept in fast-reactor design for the production of electric power. The design and its limitations, possibilities, and effects of uncertainties are presently being explored in an initial feasibility study by Brookhaven National Laboratory. Both gas-cooled and sodium-cooled versions of the reactor are being studied.

Although the FMSR is a substantial departure from conventional fast breeder designs with regard to important operating characteristics, it is closely related to breeders in fundamental technology. As a result, only extensions of current fast breeder R&D programs would be needed to establish feasibility and prepare for demonstration of the concept. An important exception is that a new high-risk metal fuel development program would be required. The FMSR has several features that differentiate it significantly from conventional fast breeders. It would operate on a once-through-and-store fuel cycle, with no fuel reprocessing being required. When equilibrium operation is achieved, after the first core and transition cycles, the new fuel charged at reload time would consist only of natural or depleted uranium, with no fissile make-up required. The plutonium burned in the reactor would be produced in situ by neutron capture, as a result of the high breeding ratio or conversion ratio of the FMSR. Fuel would remain in the reactor a very long time (about 17 years according to current design studies), and the burnup of heavy metal that would be achieved in this period is substantial (about 13 to 15 atom %). The combination of refueling with natural or depleted uranium and the high burnup would make the FMSR more than 15 times as efficient in uranium utilization as a once-through LWR (measured in terms of energy per unit mass of natural uranium) for both the gas-cooled and liquid-metal-cooled FMSR reactor operating on their respective equilibrium cycles. However, the initial fuel inventory of the FMSR would require about 1800 short tons of U_3O_8 , about four times that required for the initial loading of a standard LWR.

The reactor could be started up with fuel at an average U-235 enrichment of about 7% and a maximum enrichment of about 11%. However, the discharged fuel would contain substantial amounts of plutonium -- about 600 kg/year for a 1000-MWe plant. This plutonium would be produced over and above any needed for continuation of the FMSR fuel cycle. Therefore, it would be available for use in burner reactors in a tandem fuel cycle if this were desired.

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Unlike the conventional fast breeder reactor, whose core makeup feed consists of plutonium as well as natural uranium, the FMSR is designed to be self-sustaining on an equilibrium feed of natural or depleted uranium alone. No enrichment or reprocessing plants are needed once the reactor has achieved an equilibrium fuel cycle. A unique feature of the FMSR is a moderator-bearing region. The objective is to provide additional reactivity at low enrichment and to improve power flattening and design flexibility.

To achieve the design objectives (principally the once-through fuel cycle based on natural-uranium makeup), the concept requires a high breeding gain and good neutron economy. These requirements have led to a design based on the use of metal fuel. This one feature almost doubles the number of fertile atoms per unit volume of the fuel, significantly increases the fast-fission contribution of U-238 and Pu-240, reduces the capture-to-fission ratio of fissile isotopes, and, because of the relatively harder neutron spectrum, leads to a smaller reactivity loss due to fission-product absorption.

Core Layout

Figure 28 is a cross-sectional diagram of the FMSR core. The schematic core layout should be considered as one suggested approach rather than a fixed representation of FMSR design. In the several loading and shuffling strategies that have been explored, the same general features have been used. The hexagonal subassemblies contain fuel or moderator and steel. The nonshaded hexagons could contain steel that might be used as a thermal shield. The hexagons with dots in the center contain moderator. Hexagons marked "F" are locations of fuel-bearing control rods, while those marked "S" are representative locations of shutdown rods. All other hexagons represent fuel-bearing subassemblies. Those marked with a "1" are in the moderated zone; those marked with a "2" are in the hard-spectrum region, and those marked "3" are in a transition region, where spectrum softening occurs because of the presence of the surrounding moderator.

Figure 29 shows a cross-sectional view of the proposed subassemblies. Fueled subassemblies would nominally contain 271 fuel-pin locations in a hexagonal array within the hexagonal duct walls that confine the coolant flow. The distance between duct wall flats is approximately 20 cm. The external dimension of moderator-filled subassemblies would be essentially identical to those of the fueled subassemblies. The dashed circle in the center designates space to be allocated for cooling.

FAST-MIXED SPECTRUM REACTOR CONCEPT

FAST FUEL

MOD. FUEL

MODERATOR

F - FUEL-BEARING CONTROL RODS

S - SHUTDOWN RODS

1 - MODERATED FUEL ZONE

2 - HARD SPECTRUM FUEL ZONE

3 - TRANSITION FUEL ZONE

Figure 28. Cross-Sectional View of the FMSR.

FUELED
SUBASSEMBLY MODERATOR-FILLED
SUBASSEMBLY

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Figure 29. Cross-Sectional View of the FMSR Subassembly.

Fuel-Shuffling Strategies

Several fuel-shuffling strategies have been investigated. In one of these, fresh fuel as natural or depleted uranium is inserted first into locations near the center of the fast-spectrum zone. Here it displaces fuel which, in turn, is moved to a new position farther out radially. Fuel from that position is moved to a next position farther out. Thus, fuel moves in successive shuffling steps from the center of the fast-spectrum region to the transition zone and then into the moderated region, from which it is discharged when its irradiation limit has been reached. Calculations indicate that this strategy would satisfy neutronic requirements of breeding and criticality in the steady-state mode of operation, but the disadvantage would be that the variation in plutonium content and, therefore, power generation would be high for fuel in its initial location at the center of the fast-spectrum zone.

In a second strategy, the fresh fuel would be placed first in the transition zone at the outside of the fast-spectrum region. Its low initial plutonium content would offset the tendency toward power peaking in this location. After some period of residence, the fuel would then be moved to the central zone of the fast region. From this point on, it would be moved in succession through several locations in the fast-spectrum region in a sequence designed to flatten the power generation rate. The fuel would then skip over the transition zone and would be shuffled outward through the two radial zones of the moderated region before discharge. This version of the reactor would also satisfy the neutron physics requirements.

A third strategy is based on loading fresh fuel first into the outermost ring of the moderated region. The fresh fuel (depleted or natural uranium) acts as a strong absorber of radially leaking neutrons. After a period of residence, it is moved to the inner ring of the moderator region. The fuel remains in this position until the bred plutonium enrichment reaches ~3%, at which time the fuel is shuffled to the fast region. This strategy results in lower power swing for the fuel during its first cycle in the zone. In addition, this strategy also yields an acceptably flat radial power distribution.

Studies to date have focused on the third strategy. The primary basis for emphasizing this strategy was to reduce potential local hot-spot effects in the moderated region, which would result if more enriched fuel were placed there. The disadvantage of the strategy is that the power density in the outer moderated region is low, requiring the fast core to carry a higher power load and leading to a higher net fluence damage to the cladding. Other shuffling strategies may very well turn out to be more advantageous, and these will be studied later.

1.12.2 Status of Technology

The FMSR would primarily depend on technology developed for the fast breeder program. In most respects, this is mainline LMFBR or GCFR technology, though in some important respects the technology chosen requires advancements beyond the current state-of-the-art FBR technology. Current FBR fuel and development is directed toward oxide and carbide fuels. Thus, the development of metallic fuel will require a substantial new program. The nonnuclear parts of the plant could be identical to those of the corresponding fast breeder, though the gas-cooled FMSR would operate at a higher pressure.

The FMSR concept developed by Brookhaven National Laboratory is an extension of the heterogeneous fast breeder reactor and is also a variation of the coupled fast-thermal reactor studied earlier by R. Avery of Argonne National Laboratory. The objective of Avery's study was to increase the prompt neutron lifetime and thus greatly reduce the energy yield of hypothetical major accidents.

The fuel, cladding, and duct designs are being developed in cooperation with Argonne National Laboratory. The fuel design follows as closely as possible the design features of the successful Mark II metal driver fuel, which has been operated in EBR-II to burnups in excess of 13 atom %. The concept requires very long fuel residence times (about 17 years) and very high fuel burnup (142,000 MWd/t average and 180,000 MWd/t peak) and subjects fuel components to very high neutron fluence (8 to 10×10^{23} nvt). The effects of this mode of operation require materials to behave satisfactorily under conditions considerably beyond those explored experimentally thus far. Materials with the desired properties are currently not available, and a substantial period of development and testing is needed. The exposures expected for LMFBR oxide fuel are in the range of 50,000 to 100,000 MWd/t, achieved in about two to three years. Since the exposure and residence time for the FMSR are several times as large, the testing period to achieve representative results would take about 6 to 15 years for each trial fuel configuration. High exposures can be achieved in shorter periods of time in accelerated tests, and these might be definitive in some cases. However, time periods on the order of the lifetime of the fuel in the reactor are needed to properly evaluate effects such as fuel-cladding interactions. The FMSR RD&D period might be lengthy because of this constraint alone. In addition, there is risk that the long exposures will not be technically feasible.

The dense packing of fuel in the reactor core which is required to achieve high breeding ratio has an effect on cooling that has not been fully explored. The pumping power of the gas-cooled reactor tends to be high, and studies are under way to improve estimation of pumping power and to reduce it.

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The physics analysis to date indicates that designs can utilize either sodium or helium as the coolant. Additional physics analyses and a program of critical experiments would be required to confirm the design basis, that is, to confirm that the reactor could be fueled at equilibrium with natural or depleted uranium.

The reactor can be made to operate with any of several fuel-shuffling strategies. In each case, however, it appears highly desirable to surround the central hard-spectrum region of the core by a moderator zone consisting of graphite or beryllium. The role of the moderator differs somewhat depending on fuel-shuffling strategies; its principal purpose is for power flattening and reactivity management. Its presence outside the active core region provides some boost in reactivity, thus reducing the reactivity swing during the intervals between fuel shuffling. Note that core reactivity increases with burnup due to the large internal breeding ratio.

The FMSR is compatible in most respects with balance of plant designs corresponding to conventional fast breeders. The one significant departure affecting the nonnuclear part of the helium-cooled version is the current choice of high pressure (about 140 bars) for the helium-cooled version. The reason for this tentative choice is the desire to (1) increase the effective heat capacity of the gas flowing through the core, (2) take maximum advantage of the better heat transfer characteristics of metal fuel, and (3) reduce total pumping power requirements.

The reliance on FBR technology should lead to plant capital costs essentially the same as for FBRs, and the FMSR should have acceptable fuel-cycle costs. The use of natural or depleted uranium feed would permit fuel to be made using hands-on methods, with no criticality control restraints. Reprocessing would not be required.

1.12.3 Licensing Issues

Licensing issues have not yet been defined in any detail. Because the FMSR resembles conventional fast breeder counterparts in many respects, many licensing issues would be common. The FMSR does have some additional issues, including

- the lower melting point of the metal fuel than that of the ceramic fuel ordinarily used, and
- the high fluence to which cladding and structure are exposed.

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On the other hand, the FMSR has a longer neutron lifetime than the conventional FBR, mitigating the effects of certain transients. On balance, it is plausible that the FMSR would be licensable provided the fuel, cladding, and duct materials are demonstrated.

1.12.4 Safety Issues

The safety-related features of the FMSR, in general, will resemble those of other fast reactor concepts, with safety features of the gas-cooled version being similar to its counterpart GCFR and safety features of the sodium-cooled version being similar to its counterpart LMFBR. In each case, the principal differences will stem from differences in their respective fuels and the inclusion of a moderated region in the FMSR fuel. Although careful safety analyses have not yet been done, the important differences can still be delineated and discussed.

Neutron Lifetime

Because the FMSR is in large part a fast reactor, its neutron lifetime will be short, and a relevant safety analysis must focus on its effects on reactor transients. The incorporation of a moderator zone in the FMSR will tend to lengthen the neutron lifetime, thus diminishing the violence of a prompt power excursion that might ensue from large inadvertent reactivity additions. The value of the neutron lifetime will depend on the specific design of the reactor core and, in particular, on the degree of power sharing between the hard-spectrum region and the moderated region. Core designs currently under study cover a wide range of power sharing. The basis of selection of a particular core design must take into account its corresponding neutron lifetime and its effect on the severity of reactor transients; the delayed-neutron fraction β -effective of the FMSR design used in the present study is 0.0044, which is significantly higher than that of corresponding breeders. This effect will tend to mitigate the effect of inadvertent reactivity insertions.

Spatial Transients

The use of two differing material regions in the FMSR raises the possibility of occurrence of decoupled transients involving only part of the core. Such a possibility always exists in a large reactor. However, the question is somewhat more significant for FMSR because of the possibility that a transient could involve neutronic characteristics of only one of the two regions.

The possibility for such a transient and its implications must be explored.

Fuel Slumping vs. Fuel Expansion

Vibratory-compacted fuel is proposed for use in the gas-cooled FMSR, and Argonne National Laboratory's Mark II fuel is proposed for the sodium-cooled version. The vibratory-compacted fuel will have a density well below the theoretical value. The Mark II fuel will also have a low density as soon as release of gaseous fission products begins to induce porosity. In either case, the issue must be raised and answered as to whether fuel melting could lead to slumping, which might densify the core and result in a large reactivity increase. Conversely, the dissolved fission-product gas in both kinds of fuel should cause the fuel to foam or froth on melting, inducing expansion of the reactor core rather than compaction; this could result in a substantial negative reactivity that would shut down a transient at the onset of fuel melting. Fuel frothing did occur to a substantial degree in the EBR-I meltdown.

For the sodium-bonded fuel such as used in the Mark II design, appreciable liquid sodium penetration into the interconnected porosity as burnup proceeds has been observed. In any transient that caused the fuel temperature to exceed the boiling point of sodium (880°C), fuel dispersion could occur before melting of the uranium (1040°C).

In transients leading to fuel melting, it is currently expected that the fuel would disperse rather than slump, yielding beneficial negative reactivity feedback. Nevertheless, the phenomena require careful exploration, starting with experiments in TREAT and in Sandia Laboratory's Advanced Core Performance Reactor (ACPR).

Sodium Void

The sodium void coefficients require further analysis. In the one calculation made thus far for one particular design, the reactivity increased markedly on total sodium voiding. The spatial distribution of the sodium void coefficient and the overall void coefficient will depend strongly on the degree of power sharing between the two reactor core regions and the fuel-shuffling strategy. A core design must properly take into account the resulting sodium void coefficients and their effects on transients.

Doppler Coefficient

Although the contribution from the hard-spectrum region to the negative Doppler coefficient will be small, the contribution by the moderated zone can be significant, depending on the degree of power sharing. Core design must also take into account the importance of the Doppler effect to overall negative reactivity feedback coefficient. For the one core design utilized in the present study, the Doppler coefficient is sizable and negative: $[-4 \times 10^{-5} \text{ (k/}^\circ\text{C)/k}]$.

Recriticality

The average fissile enrichment of the fuel is quite low (5% plutonium) in both versions of the reactor. It is unlikely that the fuel can be made critical at this average enrichment without some neutron moderation. Even with the addition of moderator blocks from the moderated zone, recriticality would still be questionable.

Studies are planned to determine under what circumstances, if any, recriticality might be possible. If there are such circumstances, designs will be explored to prevent reassembly in the required geometry.

In any event, present plans include provisions for the conceptual design of an in-vessel molten core retention system with active cooling. If necessary, this system would incorporate neutron poisons to prevent recriticality.

Appropriate design measures can be taken to ensure that no accident could credibly release energy with mechanical work after a first pulse. With a long neutron lifetime, the total mechanical work should be small and the effect mild.

Auxiliary Heat Removal System

Backup cooling for a sodium-cooled version would be supplied by auxiliary pumps in a bypass to the main coolant circuit. Such a system would be designed so that no credible rupture of the primary system could uncover the core. Backup cooling for the gas-cooled version would meet the same problems encountered in the design of the GCFR. The reliability of circulator operation and its ability to force sufficient gas through the core if a massive

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depressurization were to occur are specific problem areas. The implications of these questions have yet to be explored. Resolution of these questions will depend on solution of similar questions for the GCFR.

Fuel Melting Temperature

The melting point of metal fuel is well below that of ceramic fuel. However, the thermal conductivity is high, and the maximum fuel temperature will be low. The margin to fuel melting is comparable with that in an oxide-fueled reactor.

Fuel-Cladding Interaction

Uranium and steels undergo mutual interdiffusion at higher temperatures, and eutectic formation can occur. For this reason, it has been suggested that a vanadium barrier be inserted between the uranium and the steel. Even though this barrier will be only a few thousandths of an inch thick, a metallurgical bond will be required between the steel and vanadium to eliminate interface thermal resistance. An experimental program will be required to establish the adequacy of the barrier for avoiding fuel-cladding interaction over the long fuel lifetime.

Fuel Bowing

The effects of fuel bowing in altering reactivity are well known. The use of fuel assembly duct walls and spacer grids, combined with the poor structural rigidity of the fuel, should limit the effects of fuel bowing to an acceptable level.

Fuel Coolant Interaction

Cladding failure during a reactivity transient could release molten fuel into the sodium of the sodium-cooled version of the reactor. The possibility of coolant boiling and, consequently, enhanced mechanical work in interactions between metal fuel and sodium has not yet been explored and, therefore, must be the subject of experimental research. The presence of fission gas in the intergranular positions coupled to the diffusion of sodium into the pores of the fuel is expected to mitigate considerably the effect of fuel-coolant interactions.

Low Fuel Enrichment

The low fissile enrichments of the fuel would tend to reduce the chance of recriticality should meltdown occur even though suitable designs should prevent recriticality. The tendency of metal fuel to form inseparable eutectics with steel will further reduce the effective fissile density. Added to these effects is the tendency of fuel from the higher-power-density regions to be diluted by fuel from the other areas, further decreasing the fissile enrichment and decreasing the chances of recriticality. When estimates of large mechanical energy release have been made in the past for LMFBF accident sequences, they have been the result of large energy developed from recriticality. FMSR designs therefore offer the possibility of assured low mechanical energy release from reactor accidents.

1.12.5 Environmental Issues

The FMSR concept has not yet been developed to the point where specific environmental issues can be defined and addressed. However, it is not anticipated that the FMSR will present any additional environmental issues over and above those associated with conventional fast breeders. Since the FMSR does not require reprocessing and fabrication or reprocessed material, reduction of environmental impacts can be expected in these areas (relative to the LMFBF). However, provision must be made for long-term spent-fuel storage.

1.12.6 RD&D Needs

Preliminary studies on the FMSR focused primarily on the gas-cooled version. A detailed study should be made to evaluate both the gas-cooled and the sodium-cooled versions, with one of the two being selected. In some manner, a choice should be made among the proposed moderator materials -- beryllium, graphite, and zirconium hydride.

Alternate design options and fuel cycles should be investigated for optimization purposes. As examples, the inclusion of an internal moderator zone, the use of U-233-Th, and the use of partially enriched uranium as a feed (for a fallback option) should be evaluated.

A reactor physics program must be initiated to study the safety-related reactivity coefficients, the shielding requirements, power distributions,

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reactivity swings over fuel cycles, reactivity control and control-rod performance, critical experiments, computer codes applicable to the FMSR, and computer code validation.

The thermal hydraulics of the system should be analyzed for hot channels and hot spots, orificing options, duct design options, and the effects of different control systems.

An extensive fuel and materials program must be undertaken. This should include

- Vipac fuel development and fabrication,
- investigation of alternative fuels, such as sintered fuels,
- fuel properties, such as swelling and slumping,
- feasibility of vanadium barriers and sodium bonding,
- investigation of cladding and duct materials that would include high-temperature properties, compatibility with fuels, decarburization, thermal aging, and corrosion mass transport,
- a comprehensive irradiation testing program that would include fuels, fuel bonds, cladding, and ducts.

Studies of the mechanical design of the core and the PCRV must be carried out. These would include analyses of ducts such as vented ducts, grid-supported ducts, and constrained and free-standing hexagonal and circular ducts. Also included would be shielding and PCRV studies.

Although the FMSR is predicated on the use of existing LMFBR and GCFR technology, it is sufficiently different in design and exposure requirements that a very extensive, comprehensive, and time-consuming RD&D program would be required. A summary of the RD&D cost estimates for the FMSR is given in Table 54. The principle of FMSR operation would probably be demonstrated in existing LMFBR facilities. The FMSR probably cannot be developed and commercially deployed in less than 45 years.

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Table 54. FMSR Estimated RD&D Cost Summary
(Assumes LMFBF research and development continues)

RD&D area	Estimated cost ^a (\$ million)
<u>Research and development</u>	
Basic research	500-1000
Development facilities	700-1200
Regulatory studies	750-1000
Subtotal without major plant construction	1950-3200
<u>Demonstration facilities</u>	
Proof-of-principle plant ^b	200-400
Demonstration plant	1200-1800
Lead commercial plant (1000 MWe)	1200-1800
Total estimated cost	4550-7200

^aEstimated in 1978 dollars. Does not include interest during construction. Estimated costs are incremental to LMFBF program costs.

^bProof-of-principle plant costs cover only operation of existing LMFBF facilities.

1.12.7 Perceived Advantages and Disadvantages

The principal advantages of the FMSR are summarized as follows:

No fuel reprocessing facility would be needed for the FMSR fuel cycle. Of course, fuel could be reprocessed, if desired, but FMSR operation does not require the use of plutonium in feed fuel.

No enrichment capacity would be needed for the fuel cycle, apart from that to supply the 7 to 11% U-235 for the start-up core and perhaps fissile material for control elements.

The FMSR would be much more efficient in use of uranium resources than LWRs are. If fuel recycle were ultimately allowed, the FMSR would be an excellent breeder.

The principal disadvantages of the FMSR are as follows:

Because the technology is less developed than that of the conventional LMFBR/GCFR, even more time and cost would be required for RD&D, and the beneficial effects of this technology would not be available until after 2025. There is also considerable risk of not meeting the technological objectives.

FMSR has a high front-end loading of U_3O_8 and enrichment when compared with most other reactor/fuel cycles.

2. FUEL-CYCLE SYSTEMS

Power reactor fuel is prepared in numerous facilities that employ chemical or physical techniques to convert the initial feed materials into finished fuel element assemblies for reactor operation. In addition, the fuel cycle systems include transportation and waste handling facilities. Reactors operating on the once-through fuel cycle system require uranium, uranium enrichment, fuel fabrication, transportation, and spent fuel storage facilities. A typical once-through fuel cycle system is shown in Figure 30.

Reactors utilizing recycled U-233 or plutonium require the addition of reprocessing, recycle fuel fabrication, and high-level waste disposal facilities in order to "close the fuel cycle." A typical closed fuel cycle system for plutonium fuel is shown in Figure 31.

The individual facilities that may be combined to form a specific fuel cycle system are discussed in this Chapter. Each section describes the several variations in design, construction, or operation that may be necessary for the range of fuel cycle systems studies by NASAP. Guidance in the selection of specific facility options may be obtained by referring to Appendix A of this volume.

2.1 MINING AND MILLING

2.1.1 Uranium Resources and Supply

The amount of uranium available during any time period, the price of the uranium, and the reactor fuel requirements influence the choice of reactor systems deployed. For example, restricted or expensive uranium favors deployment of more efficient reactor/fuel cycles; abundant cheap uranium favors reliance on current cycles. The amount of uranium that can be supplied is influenced by the availability and costs of production, facilities for mining and milling, and manpower to staff these facilities.

Description of Resources and Status of Technology

Uranium Resources -- On the basis of geologic knowledge, the availability of uranium is classified into two basic groups: (1) reserves of known mineral

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Figure 30. Illustrative Once-Through Fuel Cycle.

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Figure 31. Illustrative Closed Fuel Cycle.

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deposits of such size, grade (fraction uranium), and composition that the uranium could be recovered at an indicated cost with current technology and (2) potential resources that are estimated to exist but have not been sufficiently delineated to be designated as reserves. Potential resources are put into one of these subdivisions according to their geologic setting.

- "Probable" potential resources are those estimated to be in known uranium districts, in extensions of known deposits, and in new deposits within trends or areas of mineralization that have been identified by exploration.
- "Possible" potential resources are those estimated to occur in new deposits in formations or geologic settings productive elsewhere within the same geologic province or subprovince under similar geologic conditions, and within the same geologic province or subprovince under different geologic conditions.
- "Speculative" potential resources are those estimated to occur in new deposits in formations or geologic settings not previously productive within a productive geologic province or subprovince and within a geologic province or subprovince not previously productive.

The term "productive," used in the definitions of possible and speculative potential resources, is arbitrarily defined as >10 short tons of U_3O_8 in production and/or ore reserves.

These resource classifications include a measure of producibility at the listed production cost. The uranium availability is therefore described as an "economic resource" rather than in terms of the total uranium content of a given area. Currently, two economic levels have been selected for uranium resources on a worldwide basis. These cost categories are \$30 and \$50/lb U_3O_8 (\$80 and \$130/kg uranium). The costs, often described as "forward costs," include the direct costs of mining, transportation, processing, and capital for providing and maintaining the production facility. Past costs are not included. The market price of U_3O_8 is expected to be about 1 1/2 to 2 times the forward costs. Domestic resources estimated as of January 1979 for the selected forward production costs per pound of U_3O_8 of \$30 and \$50 or less are shown in Table 55.

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Table 55. Domestic Uranium Resources (January 1, 1979)

Category	U ₃ O ₈ at various forward costs ^a (short tons)	
	≤\$30/lb	≤\$50/lb ^b
Reserves	690,000	920,000
Probable potential	1,005,000	1,505,000
Subtotal	1,695,000	2,425,000
Potential		
Possible	675,000	1,170,000
Speculative	300,000	550,000
Total	2,670,000	4,145,000

^aJanuary 1978 \$.

^bIncludes resources at forward cost of \$30/lb U₃O₈.

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In addition, 120,000 short tons of U_3O_8 is considered to be producible as a by-product of phosphate and copper production, for a grand total of about 4.3 million short tons of U_3O_8 at a forward cost per pound of \$50 or less. The 1.8 million short tons of U_3O_8 represented by reserves, probable potential, and by-product in the \$30 cost category is considered by DOE to be sufficiently reliable to use in projecting production capability and supply and demand forecasts.

World uranium resources at forward costs of up to \$50/lb U_3O_8 are shown in Table 56. In addition to the nearly 6 million short tons of U_3O_8 in the resource and probable potential categories, a recent study called the International Uranium Resource Evaluation Project (IUREP) indicates that an additional 8 to 19 million short tons of U_3O_8 may exist as speculative potential resources.

In addition to domestic resources that have been estimated at forward costs of up to \$50/lb U_3O_8 , there are higher-cost, generally lower-grade, resources. Some of these may be in conventional sandstone or vein-type formations that might have forward costs of $> \$50/lb U_3O_8$ because of their small sizes or greater depths. Deposits other than sandstone or vein-types that are generally of lower grade or having higher forward cost have also been identified. These types of deposits are being evaluated through the National Uranium Resource Evaluation (NURE) program.

Certain very large but very low-grade deposits that have been known for many years are worthy of additional comment; however, these deposits currently cannot be recovered at an acceptable cost and have not been included in NURE's economic resources. Of particular interest are the shales and phosphate deposits. The Chattanooga shale of Tennessee is the chief potential source for uranium from shale in the U.S. Over 5 million short tons of U_3O_8 at an average assay of 60-ppm uranium and 10 gal of oil/ton of shale (Fischer assay) are known to exist. Similarly, a large resource is known to exist in phosphate deposits, particularly in Florida, where uranium concentrations are approximately 100 to 300 ppm.

Estimates of the uranium contained in these types of domestic deposits are shown in Table 57. Relatively long lead times coupled with favorable selling prices will be required to convert these low-grade resources into a potential supply. There has been some research into extracting U_3O_8 from seawater, which is estimated to contain 5 billion short tons of U_3O_8 (Table 57), recoverable at hundreds of dollars/lb U_3O_8 .

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Table 56. World Uranium Resources by Continent
(Excludes Peoples' Republic of China,
USSR, and Associated Countries)

	U ₃ O ₈ (thousands of short tons)			
	Reserves		Probable potential	
	≤\$30/lb	≤\$50/lb ^{a, b}	≤\$30/lb	≤\$50/lb ^{a, b}
North America ^c	920	1180	1540	2340
Africa	680	740	200	260
Australia	380	380	60	60
Europe	0	500	60	120
Asia	20	60	0	30
South America	60	50	20	20
Total (rounded)	2060	2910	1880	2830

^a January 1978 \$.

^b Includes resources at forward cost of \$30/lb U₃O₈.

^c Includes revisions since original OECD report.

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Table 57. Domestic Low-Grade Uranium Resources

Type deposit	Estimated U ₃ O ₈ (short tons)
Low-grade conventional deposits	400,000
Phosphates	4 million
Shales	5 million
Lignite and coal	30,000
Seawater (worldwide)	5 billion

Conversion of Resources into Supply -- The uranium industry is well established with many years of experience. During 1978, the domestic mining industry produced about 18,500 short tons of U_3O_8 , which was almost evenly divided between open-pit and underground mines. The selection of a mining method is usually based on the economics of recovery, which in turn are determined by the thickness, grade, lateral extent, configuration, and depth of the deposit.

Exploration is the basis for expanding uranium resources. There has been a marked increase in worldwide exploration during the last three years, particularly in the United States and Canada. During 1978, ~47 million feet of exploratory and development drilling was completed in the U.S., resulting in a 30,000-short-ton net increase in \$50/lb U_3O_8 reserves. Figure 32 illustrates the growth in domestic exploration efforts.

The U.S. government also spends \$70 to \$80 million annually for uranium resource activities. A portion of this expenditure is for NURE, a comprehensive evaluation program based on hydrogeochemical and airborne surveys whose goals are to provide comprehensive information on uranium resources and to increase the amount of known uranium resources.

The uranium mill extracts uranium from the ore by mechanical and chemical processing of the mineral into a semirefined uranium concentrate sometimes called yellow cake (U_3O_8). Two alternative methods using either sulfuric acid or sodium carbonate leach processes are employed for uranium extraction. The selection of the process depends on the chemical characteristics of the ore. The uranium leach solution is then purified by either ion exchange or liquid-liquid extraction techniques and converted into a solid uranium product. There were 20 conventional uranium mills operating in the U.S. during 1978, with a combined processing capacity of 21,000 short tons of U_3O_8 per year. These active mills are located in Colorado, New Mexico, Texas, Utah, Washington, and Wyoming. During 1978, two new conventional mills were started up, two existing mills were in the process of expansion, and six mills were in the planning stages for probable startup during the next several years. Uranium production requirements to meet different nuclear reactor building schedules are discussed in Volume III.

Other methods of uranium recovery, including in situ leaching, recovery from mine water, and by-product production, accounted for about 3% of the total U_3O_8 production. Methods of recovering uranium from wet-process phosphoric acid and the acid leaching liquors from porphyry copper ores have been developed. In the wet process, crushed and ground phosphate rock is mixed

Figure 32. Annual Uranium Drilling Costs and Total Footage.

with sulfuric acid and allowed to digest before insoluble calcium sulphate is filtered off, and the acid solution is concentrated. The uranium goes into the phosphoric acid product, which is used for the manufacture of triple superphosphate fertilizer. Similarly, uranium in acid-leached copper ore passes into the product stream and becomes available for by-product recovery to the copper extraction process. Solvent-extraction processes are used to remove the uranium from the impure acid solutions. The extractants most often used are based on organic phosphate compounds in an aliphatic diluent. Several by-product plants are in operation, and others are under construction. Uranium recovery from wet-process phosphoric acid has progressed more slowly than anticipated because of initial plant startup problems. Some of the phosphoric acid producers are also waiting until current plants are proven successful before installing their own recovery circuits. At present, the potentially available quantity of by-product uranium is limited by the production of phosphoric acid. DOE has estimated that some 100,000 short tons of U_3O_8 will be available from phosphoric acid production by the year 2000. Rates of uranium production from phosphates have been estimated as high as 15,000 short tons of U_3O_8 per year by 2025. Production of up to 20,000 short tons of U_3O_8 through the rest of the century has been projected from copper leaching operations, and these operations may yield over 1000 short tons of U_3O_8 per year after 2000.

Other Potential Sources of Uranium Supply -- Shales, low-grade conventional deposits, coals, and seawater represent other potential sources of uranium supply for the future. At present, there is no significant domestic effort for developing these sources of supply, because their respective selling prices would need to be significantly higher than for conventional resources. The principal attraction of the Chattanooga shale is the enormous amount of uranium contained in this deposit. The problem is the very low concentration (60 ppm) of the uranium. If lower grades to 45 ppm uranium are included, the Chattanooga shale may contain over 10 million short tons of U_3O_8 . A recent study was made for DOE on the economics and feasibility of producing uranium from Chattanooga shale. Table 58 shows some key characteristics of the shale processing plant and shale mines that were derived from this study. Extensions of the study indicate that, with oil priced at about \$25 per barrel, uranium might be produced economically as a by-product of shale oil recovery, but not as the sole product. Two major problems are (1) the large mining and milling operations that are required and (2) the need to dispose of a large amount of waste. About 9 tons of Chattanooga shale would have to be mined to produce 1 lb of U_3O_8 , compared to 1/2 ton of high-grade ore needed today. A capital investment of >\$2 billion would be needed to obtain about 50,000 bbl per day of oil and 1200 short tons per year of U_3O_8 . A large infrastructure, work force, and supplier organization would be required for such operations. The environmental impact of mining, processing, and waste disposal for such large quantities of material would

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need to be carefully investigated and planned for. Many years would be required to implement such a major project, and it would not be undertaken until adequate prices are clearly perceived, both for the shale oil and the uranium. The current outlook for technology, economics, and lead time makes it unlikely that any significant commercial production could be expected before 2000. Thereafter, shales could be a significant source of oil and by-product uranium. Production of over 25,000 short tons of U_3O_8 per year has been projected as a possibility by DOE-URE from this resource by 2025. A significant RD&D program would be required to exploit the shales (Table 58).

Coals now used as fuel in steam generating plants generally contain only about 1 ppm uranium, and, therefore, are not feasible uranium sources. On the other hand, certain deposits of coal, particularly some of the thinner lignite beds of the western U.S., contain higher concentrations of uranium, which could possibly be recovered either directly from the lignite or from ash after burning. Small volumes of these high-grade lignites have been mined in parts of North and South Dakota and Montana. However, these uraniumiferous coals have higher ash contents and much lower heating values than nonuranium-bearing coals, thus detracting from their suitability as fuels. Some are so poor in quality that they do not support combustion by themselves. The amount of uranium in the ash from high-grade U.S. lignites is estimated to be about 4000 short tons of U_3O_8 available at \$30/lb at a grade of 0.10% uranium. Because of the poor fuel quality of these lignites, their utilization is uncertain.

There is uranium equivalent to 5.2 billion short tons of U_3O_8 in the oceans at a concentration of 0.003 ppm. Recovery from seawater has been studied in the United Kingdom since 1964, and more recently in Japan, Germany, and Italy. Three schemes have been considered: use of pumping, use of tides to avoid pumping, and use of natural ocean currents. The civil engineering requirements of all three are formidable and enormously expensive.

A recent DOE-sponsored study* estimated that 0.5×10^9 m³ per day of seawater would have to be treated to produce 550 short tons of U_3O_8 per year, assuming a uranium absorption efficiency of 70%.

*From a study for DOE-Grand Junction by the Exxon Nuclear Company, Inc.

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Table 58. Key Characteristics of a Chattanooga Shale Processing Plant, Mines, and Waste Disposal

Shale processing plant				
	Production			
	Daily		Yearly	
Shale feed (55 ppm U), tons	100,000		35×10^6	
Oil product, bbl	49,000		17.5×10^6	
U ₃ O ₈ product, short tons	3.30		1,155	
Ammonia product, tons	490		0.17×10^6	
Sulfur product, tons	2050		0.72×10^6	
Waste tailings, tons	80,000		28×10^6	
Mines and waste disposal				
<p>Three underground mines, 100 to 300 ft deep, each covering an area of about 4 miles long by 3 miles wide, producing 36,000 tons per day for 20 years using room and pillar mining system; total employment for each mine would be 950 people; 70% of waste tailings would be back filled into mines as 50% shale/water slurry (56,000 tons per day); 30% of waste tailings (24,000 tons per day equivalent to 10 acre-ft per day) are disposed of and covered with topsoil and grasses.</p>				
Financial aspects				
<p>Capital investment for shale processing plant and three mines is \$2.1 billion. The following product sales will result in either a 10% or 15% after-tax return on investment (ROI):</p>				
Product	10% ROI		15% ROI	
	Unit sales price (\$)	Annual income (\$ million)	Unit sales price (\$)	Annual income (\$ million)
Oil	15-25/bbl	262.5-437.5	15-25/bbl	262.5-437.5
U ₃ O ₈	114-38/lb	262.5-87.5	178-103/lb	412.5-237.5
Ammonia	120/ton	20.6	120/ton	20.6
Sulfur	40/ton	28.7	40/ton	28.7
Electricity	0.2/kWhr	27.2	0.2/kWhr	27.2
Total		601.5		751.5

The plant capital cost was estimated to be \$6.2 billion and the cost of the U_3O_8 product to be \$600 to \$3600/lb. Since the research is at an early stage, the engineering difficulties are very formidable, the costs are likely to be extremely high, and no production from seawater is expected through 2025.

Licensing, Safety, and Environmental Status

Uranium Mining -- The federal government has no licensing authority over uranium mines, although the NRC reviews the safety and environmental impact of mines when they are part of a mine-mill complex. However, various federal regulations apply to the occupational health and safety of miners and to environmental impacts on air and water quality.

Uranium mining operations have similar health and safety effects as other heavy metal mines. In addition there are occupational hazards from radiation exposure.

The principal environmental effects of uranium mining result from the disturbance of land by mining operations and the release of naturally occurring radioactive elements from the ore body. Open-pit mining has greater land impacts than underground mining because of the large open excavations, the large piles of earth and rock overburden that are created, and the extensive network of roads required for the operation. These disturbances are largely temporary. The principal radioactive effluent is radon (Rn-222).

Uranium production from low-grade resources would be expected to have increased environmental effects. A major reduction in ore grade would necessitate processing larger amounts of ore. In addition, the locale of the mine and mill operation may change from the arid western states to a humid southeastern area where environmental impacts would be less.

Uranium Milling -- The NRC licenses uranium mills. During the licensing process the NRC reviews the safety and environmental impacts of these facilities. Various other federal regulations apply to the occupational health and safety of the mill workers and to the environmental impacts on air and water quality. Efforts by the NRC are under way to provide uniform and effective regulations for mill tailings.

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The most likely types of accidents associated with uranium mill operations are inadvertent discharges of tailings or solutions to nearby surface waters or a fire in a solvent-extraction circuit. Potential failures that could result in off-site release of tailings include failure of a tailings dam, rupture of a tailings pipeline, and an operator permitting the overflow of a tailings pond. A mill solvent-extraction circuit that contains several thousand gallons of mostly kerosene solvent represents a potential for a serious fire.

The principal radiological concern of the milling operation is the release of Rn-222 from the ore and mill tailing and the potential resultant exposures to the mill workers. Population doses from these sources are quite low because the mills are generally located in remote and sparsely populated areas.

After milling operations are completed, radon continues to be released from the tailings. Current NRC regulatory objectives are to reduce radon releases to about twice the background level by isolating or stabilizing the tailings piles.

RD&D Needs

Conventional mining and milling of uranium from high-grade ore is an established industry. Improvements are expected as a part of normal operations within the industry. However, increasing interest in the release of small amounts of radioactive or toxic material will probably result in the need for some R&D to meet the as-low-as-reasonably-achievable (ALARA) discharge requirements.

A substantial RD&D program will be needed to develop low-grade resources into significant uranium supply. The Chattanooga shale has the greatest potential for providing a significant long-term source of supply. The following developments will be necessary for the utilization of this supply from the Chattanooga shale.

- Shale-oil-processing technology will have to be expanded from the initial laboratory testing of eastern Devonian shale into full-sized hydroretorting or shale-oil processing plants. This will require (1) process development studies in facilities of the type that are available at the Institute of Gas Technology, (2) construction and operation of a very large-sized pilot plant, and (3) construction of the lead plant(s).

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- Uranium and other metal extraction technology will have to be developed and operated in a pilot plant. High levels of uranium recovery, as well as good economics, would be stressed in this development effort. Swedish technology, developed at the Ranstad pilot plant, may be applicable to the United States shales.
- Environmentally acceptable disposal technologies will have to be developed for the process waste. The following areas of technology development are required: (1) dewatering of the tailings, (2) backfilling below-ground mines with tailings, (3) storing the remaining tailings above ground, and (4) controlling seepage of liquid from these wastes.
- New or improved below-ground mining technology may have to be developed that would remove large volumes of shale, allow the backfilling of the mine with tailings, and limit subsidence of the surface.
- Socioeconomic and environmental impact studies will need to be made for operations of the magnitude needed to develop this source of uranium supply.
- Additional by-product recovery development studies may be desirable for reducing the amount of toxic materials that might otherwise be present in the mill tailings. This could be a means of reducing the bulk of the tailings and may also result in a possible source of revenue.

A schedule and cost summary for developing the Chattanooga shale through RD&D is shown in Table 59.

Economics

The cost of uranium mines and mills varies significantly with the size and composition of the ore body, the accessibility, the availability of labor, and other factors. The cost of typical U.S. facilities for producing U_3O_8 from ore is presented in Table 60. A mine/mill complex is relatively small, but it requires considerable preproduction expenditure, land acquisition, exploration, and development.

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Table 59. Chattanooga Shale RD&D Estimates

Schedule	Years
Process development	
Development operations	1980-1990
Support operations	1986-1995
Pilot plant (~100 short tons U ₃ O ₈ per year)	
Start design	1981
Start operation	1986
Lead plant (1200 short tons U ₃ O ₈ per year)	
Start design	1988
Start operation	1997
<u>Cost</u>	<u>\$ million^a</u>
Process development	45-55
Pilot plant	400-500
Lead plant	2900-3500

^aJanuary 1978 \$.

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Table 60. Estimated Costs,^{a,b} Ranges, and Averages by Function

	Low	High	Average
Acquisition and exploration costs			
\$/lb U ₃ O ₈ recovered	0.30	7.30	2.40
Ore haulage costs			
\$/ton of ore	0.05	5.10 ^c	1.50
Mining costs			
Open pit, \$/ton of ore			
Capital ^d	6	26	14
Operating	5	15	9
Total	14	34	23
Underground, \$/ton of ore			
Capital ^d	1	33	10
Operating	22	45	31
Total	23	64	41
In situ, \$/lb U ₃ O ₈			
Capital ^d	1	4	4
Operating	4	13	7
Total	5	16	11
Milling costs			
Conventional, \$/ton of ore			
Capital ^d	1	7	1
Operating	4	16	7
Total	5	22	8
In situ, \$/lb U ₃ O ₈ recovered			
Capital ^d	1	2	2
Operating	3	7	5
Total	4	8	7
Royalty costs			
\$/lb U ₃ O ₈ recovered			
Conventional mining	0	5.70	1.30
Nonconventional	1.00	2.50	1.90

^aAs used in 1977 30-year estimate of "could" production capability — January 1977 \$.

^bWhere both uranium and vanadium were assumed to be recovered, only costs allocated to uranium are shown.

^cSecond highest — to show highest might reveal company involved.

^dForward cost as of January 1, 1977.

2.1.2 Thorium Mining and Milling

Description

Most of the world's thorium production has been from monazite sands, which are mainly phosphates of the rare-earth elements in placer deposits. Placer deposits may be formed in a river or in coastal locations. Monazite-bearing stream placers in the Piedmont region of North and South Carolina were the world's first source of thorium, with production starting in 1887. Production from this source ceased in 1917 and was supplanted by lower-cost monazite beach placers in India, Malaysia, Brazil, and Australia. Recent U.S. production of thorium has been a by-product of monazite recovered in mining titanium from beach placers in Georgia and Florida.

The most promising thorium deposits for large-scale exploitation are thorite-bearing veins such as those in Colorado, Idaho, and Montana.

Total U.S. thorium reserves have been estimated at 188,000 tons as thorium dioxide. Probable potential resources are estimated at almost three times the estimated reserves. Approximately 75% of the U.S. thorium reserves producible at a forward cost of <\$30/lb of thorium dioxide occur in vein deposits. More than half of the \$30 reserves in veins is estimated to be available in the Lemhi Pass district of Idaho and Montana, which lies astride the Continental Divide.

Most of the remaining thorium reserves and resources in veins occur in the Wet Mountain district in Colorado. Flat or mildly dipping veins near the surface would be mined first by open-pit methods. Underground mining may have to be used later as surface deposits are depleted.

Status of Technology

At present, there is no industry in the U.S. for producing thorium as a primary product and no U.S. industrial experience in producing thorium from thorite vein ores. However, conventional open-pit and underground mining methods can be used to mine the ore. A plant for milling and refining thorium ore can use technology developed for uranium ore. A modest amount of R&D will be required to develop extraction technology for vein ores and to analyze and measure the transport and dispersions of potentially harmful radioactive or toxic elements in the effluents from thorium mines and mills.

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Licensing, Safety, and Environmental Status

Since there is no domestic thorium mining and milling industry, there has been no licensing history for such activities. However, licensing considerations for uranium mines and mills would be directly applicable to thorium mining and milling, and no insurmountable problems unique to thorium are foreseen.

The health and safety effects postulated for a typical uranium mine and mill are also appropriate for a conceptual thorium mill. The principal environmental effect from thorium mining and milling is due to the release of thorium daughter products. Radon-220, with a half-life of 55.6 sec, is the thorium decay product of most concern, whereas with uranium, one is concerned primarily with Rn-222 (half-life = 3.62 days). Because of the shorter half-life of Rn-220, only ore very near the surface contributes to radioactive emissions to the atmosphere from in-place ore. However, mined ore has a higher volumetric source strength, and the overall release rate of Rn-220 from thorium mining and milling operations is greater than that of Rn-222 from uranium mining and milling. With the use of about 1 ft of earth over the dry tailings pile after mill operations have ceased, radiation due to the tailings pile and airborne particulates would be reduced to negligible concentrations compared with the concentrations during operations. After a decay period of about ten half-lives for Ra-228 (57.5 years), the tailings pile will be equivalent to a thorium ore body containing 0.05% thorium oxide equivalent.

2.2 CONVERSION AND ENRICHMENT

2.2.1 Conversion of Uranium Feed to UF₆ for Enrichment

Before uranium is enriched in the U-235 isotope by either the gaseous diffusion or gas centrifuge process, it must be converted to the volatile compound uranium hexafluoride (UF₆).

Description and Status of Technology

There are presently two processes that have been used commercially for the conversion of uranium ore concentrates to UF₆ in the United States. One method, fluorination-fractionation, consists of reduction of the U₃O₈ ore concentrate to uranium dioxide (UO₂), hydrofluorination to uranium tetrafluoride (UF₄), and fluorination to uranium hexafluoride (UF₆), followed by fractional distillation of the UF₆, which provides the final product purification step. The fluorination-fractionation process, with a nominal capacity of 13,000 tons of uranium per year, is used by the Allied Chemical facility at Metropolis, Illinois. An alternate method, wet processing, employs a liquid-liquid extraction step at the head end of the process to prepare a high-purity uranium feed from uranium ore prior to the reduction, hydrofluorination, and fluorination steps.

The wet process, with a nominal capacity of 9,900 tons of uranium per year, is used by the Kerr McGee facility at Gore, Oklahoma. Government-owned feed conversion plants have employed the wet process to convert both uranium ore concentrates and Government-reprocessed recycle uranium into UF₆ for enrichment. Power reactor recycle uranium would also be converted into UF₆ for re-enrichment utilizing either of these processes. The two domestic facilities noted above are providing commercial conversion services and have operated successfully for some time. Conversion facilities for recycle uranium have been built at the Allied General Nuclear Services (AGNS) reprocessing plant, but the facilities have not been licensed.

Licensing, Safety, and Environmental Status

Both the Kerr McGee and Allied Chemical uranium ore conversion processes are licensed. These plants have also expanded their production capacities and have undergone licensing revisions. Licensing of a facility to convert

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reprocessed uranium into UF_6 for re-enrichment has not been completed. Licensing action for the Allied General Nuclear Services (AGNS) reprocessing plant was suspended when the decision was made to delay commercial spent reactor fuel reprocessing.

The most likely types of accidents associated with conversion processes involve the release of strong acids and are similar to those for hydrometallurgical processing of other heavy metals. Spillage of acidic solutions of uranium and kerosene-type solvent fires are also sources of potential accidents.

Differences are noted in the nature of the effluents from the two processes. The bulk of the impurities in the feed to the fluorination-fractionation process are rejected as solid fluorides, while the bulk of the impurities from the wet process are discarded in the acidic raffinate stream. Waste treatment processes are employed to neutralize the excess acid or fluoride that may be present with the discarded impurities and to limit the solubility of these materials and, thus, their release to the biosphere.

Conversion of reprocessed reactor recycle uranium into UF_6 should not be the source of additional significant safety or environmental problems if spent-fuel reprocessing is accepted.

RD&D Needs

The basic conversion processes have been operated on a commercial scale for many years. As a result, there is little need for basic RD&D efforts in support of the conversion processes. However, some development effort may be desirable to support the first conversion facility for reprocessed reactor recycle uranium.

2.2.2 Uranium Enrichment

Many of the reactor fuel-cycle systems studied in the NASAP investigation require fuel enriched in U-235 that is produced using the gaseous diffusion process, the only domestic process employed for commercial uranium enrichment. Recently, the gas centrifuge has become competitive with the gaseous diffusion process, and a large share of additional worldwide enrichment capacity will undoubtedly be provided by this process. Several advanced isotope separation (AIS) processes that appear to offer significant economic

potential are in the early stages of development. A brief description of these processes is included in the following discussion.

Gaseous Diffusion Process Description

Enrichment of uranium by the gaseous diffusion process depends on the small difference in molecular mobility between gas molecules of different isotopic species. Molecules of the light isotopes will have velocities slightly greater than molecules of the heavy isotopes and, therefore, come in contact with the barrier walls more frequently. A fraction of the UF_6 gas introduced into porous barrier tubes diffuses through the barrier walls, is slightly enriched in U-235; the remaining undiffused and slightly depleted portion is recycled. Figure 33 illustrates the basic concept of the gaseous diffusion process. It shows schematically what takes place in a single stage. The degree of enrichment in a single stage is small, but the effect can be multiplied by building a cascade consisting of many stages (Figure 34).

The enriched stream from the bottom stage (see Figure 34) enters the middle stage compressor and, after being partly compressed, is mixed with the depleted stream from the top stage. This mixture is then compressed still further and is fed to the converter in the middle stage. The enriched stream from the middle stage moves to the next compressor upstream, and the depleted stream is sent to the stage below. Groups of stages are coupled in this fashion to make up cells, which, in turn, make up the cascade.

Currently, nuclear power industry customers desiring enrichment services enter into a contract for these services with DOE. The customer provides the feed, which usually is virgin material with a normal assay of 0.711 wt % U-235 and a negligible U-234 content (0.004 wt %); however, the feed may consist of reactor returns (reactor recycle material), in which case the isotopic composition of the uranium would vary. There has been no recent commercial fuel recycle to the existing diffusion plants, and essentially none is forecast through the 1980s. In the case that there would be reactor fuel recycle, the recycled uranium from LWRs and other reactors would be very slightly enriched in U-235 and may contain a significant quantity of U-236 plus small quantities of U-232 and U-233 as well. The U-236 comes from non-fission neutron capture in U-235; the U-232 and U-233 arise from neutron interactions with some of the uranium and thorium isotopes. The U-232 decays, producing very radioactive daughter products, some of which are solids that accumulate in the process equipment. The introduction of U-232 is thus a source of concern to an isotope separation plant, and reprocessed uranium must therefore conform to established specifications.

Figure 33. Gaseous Diffusion Stage.

Figure 34. Gaseous Diffusion Stage Arrangement in a Cascade.

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The specifications for UF_6 feed to the gaseous diffusion plants allow a maximum of 0.11 ppm (U-232/U-235) and 500 ppm (U-233/U-235). A recent study indicated that significant quantities of U-232 would be present in spent fuel from reactors employing extended burnup. These reactors currently employ the once-through fuel cycle. The spent fuel is stored rather than recycled. Depending on the length of storage prior to recycle, the U-232 may build up to the maximum permissible concentration for reenrichment. Any U-232 present in the cascade feed stream tends to move up the cascade with the U-235, while about one-half of the U-236 will be removed in the tails stream.

Product is withdrawn and transported to commercial plants for conversion and fabrication into reactor fuel elements. At the customer's option, the depleted uranium stream or tails are either returned or retained permanently by the U.S. Government.

Technology Status -- The gaseous diffusion process has been commercially deployed for over 30 years by the United States and other countries and is not subject to extensive development efforts in the future. Current development efforts are considered primarily as supporting and maintaining a mature technology. However, in the United States, two major expansion programs are in progress to increase the capacity of the diffusion complex, namely the Cascade Improvement Program (CIP) and the Cascade Upgrading Program (CUP). The technological base for both of these programs has been tested as full-scale prototypes in the production plants. The CIP will provide a significant increase in separative work* capacity by improving process equipment efficiency. New, more efficient barrier and compressor modifications will improve the efficiency of plant power usage by over 30%. This will increase the annual system separative capacity by about 5.5 million SWU with no increase in power usage. The CUP will increase throughput of the three gaseous diffusion plants by increasing the power capability from 6065 to 7380 MW. The additional separative work achieved will be about 4.6 million SWU/year. The CUP involves the upgrading of plant utility systems to permit supplying additional power and arranging for disposition of the associated process heat.

It is clear from the above discussion that improved technology is being implemented in the three U.S. gaseous diffusion plants and will be completed in 1981. The completion of these programs will allow the United States to continue providing a large fraction of worldwide separative work requirements at competitive prices.

*The measure of effort required to accomplish isotopic separation (enrichment) is defined in terms of "separative work units" or "SWU."

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The enrichment capacity at DOE's three plants before the CIP and CUP was 17.2 million SWU/year. Completion of the CIP/CUP improvements will provide an additional 10.1 million SWU/year, which, in the long term, is fully committed to toll enrichment customers already under contract and to U.S. Government requirements. The current contract commitments require that not only the CIP and CUP be completed, but that additional enrichment capacity be built as discussed in the section below (Gas Centrifuge Process Description).

Licensing, Safety, and Environmental Status -- All existing gaseous diffusion facilities are owned by the Federal Government and are currently not subject to regulatory review and licensing by the NRC. Private ownership of the facilities is not presently envisioned. Based on the excellent safety record of the gaseous diffusion plants to date, the risk to the public due to an expansion to 27.3 million SWU/year and continued operation is small. Furthermore, the selection of an alternative fuel cycle will have no effect on plant operation as long as feed material specifications remain unaffected by this choice.

The major environmental impacts of gaseous diffusion plant operation are the heat rejected to the environment from the diffusion plants and the impacts of generating the 7380 MWe power consumed. Although there are no immediate plans to expand the diffusion plants further, the expansion of enrichment capacity via gaseous diffusion would appear to be environmentally acceptable.

RD&D Needs -- Some RD&D effort will be expended in studies of the reliability of important mechanical components through the year 2000 and beyond as part of normal efforts to maintain a mature process. Also, an ongoing RD&D effort aimed at supplying part of the space heating requirements at the cascade facility by recovering energy from the cascade coolant will be continued. A program aimed at utilizing this residual heat for additional applications is being initiated.

Gas Centrifuge Process Description

The degree of enrichment in a single centrifuge machine depends on the mass difference of the isotopes being separated, the length of the rotor, and, most importantly, the speed of rotation. In large, high-speed machines, it is possible to obtain a degree of enrichment many times that of a diffusion stage. Gaseous UF_6 at a very low pressure is fed into a rotor that spins at high speed inside an evacuated casing. The gas accelerates to approximately the speed of the rotor. Centrifugal force causes the heavier U-238

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molecules to move closer to the wall of the rotor, producing partial separation of the U-235 and U-238 isotopes. This separative effect is increased by an axial countercurrent flow of gas within the centrifuge. Feed is introduced near the middle of the rotor, and enriched and depleted streams are removed near the ends. Figure 35 shows the basic principle of a gas centrifuge.

Because the throughput of a single centrifuge is small, a large number of machines must be connected in parallel to achieve the total flow necessary for a single stage of enrichment in a large-capacity plant. Also, the desired enrichment for power reactor fuel is not attainable in a single stage, and, thus, several stages must be connected in series to form a cascade (Figure 36).

The most significant difference between the gaseous diffusion and gas centrifuge processes is power use. A centrifuge plant requires only about 4% of the power needed for a diffusion plant of equal separative work output. Thus, for a plant of 8.8 million SWU/year capacity, about 2400 MW of electrical power would be required for a diffusion plant and only about 100 MW for a centrifuge plant. In addition to the energy saved by this lower power demand, corresponding environmental benefits result from lower demands on water resources for cooling. It is anticipated that UF₆ feed and withdrawal systems will be generally similar to those used in the diffusion plant so that diffusion plant experience will be directly applicable to a large centrifuge plant.

Technology Status -- Results from the U.S. centrifuge program have established the potential attractiveness of the gas centrifuge process. The scope of the program has been expanded to ensure capability in all areas essential to the successful construction and operation of a production-scale facility. Major areas of the current program include:

- Centrifuge Physics -- experimental and theoretical studies to acquire a better understanding of centrifuge theory and operation;
- Materials Development -- investigation, development, and testing of materials;
- Technology Development -- development and testing of alternate concepts to provide fallback alternatives to selected critical plant elements;

Figure 35. Principle of a Gas Centrifuge.

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Figure 36. Arrangement of Centrifuges in a Cascade.

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- Machine Manufacturing Development -- evaluation, development, implementation, and testing of manufacturing methods for centrifuges;
- System Manufacturing Development -- evaluation, development, implementation, and testing of manufacturing methods for plant process elements other than centrifuges;
- Development Facility Operations -- operation of experimental facilities required for screening tests, reliability tests, interface tests, and cascade testing of centrifuge systems;
- Recycle/Assembly Development -- evaluation, development, and testing of methods and tooling for diagnostic systems, repair, and assembly/disassembly of centrifuges;
- Support and Special Studies -- miscellaneous activities such as economic and analytical studies, facility engineering studies, and security support.

Several development facilities have been or are being constructed to improve centrifuges, associated equipment, and operating procedures. These include the following.

1. The Equipment Test Facility, which began operation in 1971, was developed for examining the reliability of four types of high-capacity centrifuges. Reliability testing is continuing on some of these earlier machines.
2. The Component Preparation Laboratories was built to aid the development and demonstration of techniques for manufacturing centrifuges and to reduce the time and cost of the manufacturing process. These manufacturing techniques are being provided to private industry.
3. The Component Test Facility, which began initial operation in 1975, is used to test the reliability and operability of substantial numbers of centrifuges. This facility serves as a pilot-plant cascade and now operates centrifuges obtained from three different sources. Although a test facility, it has a production capacity of about 50,000 SWU/year.

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4. The Advanced Machine Development Laboratories are being used for improving and testing centrifuges.
5. The Advanced Equipment Test Facility became operational in the spring of 1978. This facility is used primarily to test the reliability of the production centrifuges that will be used in the Centrifuge Plant Demonstration Facility and the full-sized Gas Centrifuge Enrichment Plant. The facility is also operated to test plant subsystems.

Licensing, Safety, and Environmental Status -- The Gas Centrifuge Enrichment Plant to be built at DOE's Portsmouth, Ohio, facilities will not be subject to regulatory review and licensing by NRC. Specific safety and environmental criteria for large gas centrifuge facilities have not yet been delineated. The accident potential and resulting consequences for a centrifuge plant are reduced, however, primarily because of lower in-process inventories and quantities of required oil, chemicals, and chromated recirculating cooling water. The lower inventory also reduces the criticality potential. Losses of recirculating cooling water due to line breaks would have reduced consequences because of the greatly reduced heat load compared with the gaseous diffusion plants.

There is little difference between the environmental impacts of constructing a gaseous diffusion and a gas centrifuge plant, except that the former requires concurrent construction of a much larger electrical power source and its redundant backup. The major differences during operation are those related to the heat removal system and the power consumed. Because both the heat removal and power requirements are higher for a gaseous diffusion plant than for a gas centrifuge plant of similar capacity, the potential environmental impact of operating a gas centrifuge enrichment plant is expected to be significantly less than for a gaseous diffusion plant.

RD&D Needs -- The development program has as its purpose the technological support of the Gas Centrifuge Enrichment Plant being designed and on which major construction will begin shortly, and the support of additional plants that may be built subsequently. The first 1.1 million SWU/year increment of the plant is scheduled to begin operation in 1986. The full plant is scheduled for completion in about 1994 and will have an annual capacity of 8.8 million SWUs. The cost is expected to be about \$5.1 billion in FY 1978 \$. The major capital facilities involved, their purpose, and their funding levels are shown in Table 61.

Table 61. Major Capital Facilities in the Centrifuge RD&D Program^a

Title	Purpose	Funding (\$ million) ^b
Equipment Test Facility	Centrifuge reliability testing	7
Component Preparation Laboratories (Torrance, California, and Oak Ridge, Tennessee)	Fabrication and assembly development	64
Advanced Machine Development Laboratory	Centrifuge improvement and testing	1
Component Test Facility and Advanced Equipment Test Facility	Pilot plant production and reliability testing	46
Centrifuge Plant Demonstration Facility	A production plant unit cascade	60

^aIncludes only those projects authorized through FY 1978.

^bJanuary 1978 \$.

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The Centrifuge Plant Demonstration Facility (CPDF) will be in operation by mid-1982 and is estimated to cost \$60 million. This facility will be used to test not only centrifuges but also other cascade equipment and systems. Much equipment, including all centrifuges for this project, will be procured from industry. The facility will be used to obtain information to show that producible machines and the necessary industrial vendors are available and that the design, construction, and operating criteria are satisfactory for use in the full-sized plant.

As testing of machine and cascade auxiliary equipment in the CPDF proceeds, further extensive evaluation and qualification of industry-produced machines and plant components will be conducted to achieve cost and performance goals. These activities, identified in the ten-year development plan, will demonstrate machine producibility and operability by the manufacture and testing of machines and plant components in Government facilities and by early operation of the first production train, which consists of six cascades. Machines and plant components beyond the CPDF will incorporate improvements from the development program.

Other developmental activities related to gas centrifuge technology include the following:

1. meeting water effluent quality standards,
2. potential for reduction in Freon coolant releases,
3. improved handling of buried solids,
4. improved protection against accidental releases of UF₆,
5. contaminant control, especially if recycle uranium is to be returned in the future for re-enrichment,
6. control of solvents used in the manufacturing of centrifuge components,

7. tests to ensure the protection of personnel in the event of an essentially instantaneous disintegration of a rotor, and
8. reduction in noise levels.

Advanced Isotope Separation (AIS) Program

Several advanced isotope separation (AIS) processes are in early stages of investigation, and preliminary results are encouraging. An extensive development effort on three of the more attractive processes is being funded by the U.S. Government. A decision date has been set (the end of 1981) for eliminating two of the processes and concentrating the effort on the most promising candidate. Preliminary estimates of the price of separative work are less than one-half the current price, and, if this potential is realized, the AIS could provide a means of economically lowering the tails assay of the nation's enrichment complex. Projected RD&D costs for the selected AIS process are shown in Table 62. The schedule and capital costs for the major RD&D facilities are shown in Table 63.

Atomic Vapor Laser Isotope Separation Process

The atomic vapor laser isotope separation (AVLIS) process is based on the isotopically selective excitation and ionization of uranium metal vapor by laser photons. The laser or lasers preferentially excite U-235 atoms, causing their subsequent ionization. The U-235 ions may be separated from U-238 neutrals by use of an electric field that diverts the ions to a product collector, while the non-ionized atoms go to a tails collector. Although little or no U-238 is photoionized, a significant amount is collected in the product. Some U-238 ions may be created by the vapor source or by charge exchange, and these will be diverted to the product collector by the electric fields that divert the U-235 ions. Additionally, vapor will be scattered directly onto the product collector. Both of these effects act to reduce the product assay obtainable with a given feed.

At this time, the AVLIS process is still in the intensive R&D phase with emphasis on fundamental process physics and materials issues. No separations have been demonstrated in systems comprised of components that can be expanded to production systems. Current problem areas include system materials compatibility caused by the corrosive nature of high-temperature uranium, the accurate measurement of the population of electronic states accessible to the

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Table 62. AIS RD&D Effort

Fiscal year ^a	Funding (\$ million) ^b
1979	53
1980	50
1981	77
1982	128
1983	139
1984	85
1985	107

^aFunding requirements beyond FY 1985 are as yet undefined.

^bJanuary 1978 \$.

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Table 63. Major Capital Facilities for the AIS R&D Program

Title	Expected operation	Funding (\$ million) ^a
Prototype test beds	FY 1980	30
Development module	FY 1984	25-40
Demonstration facility	FY 1988	100

^aJanuary 1978 \$.

proposed laser systems, and the design of a practical collector that will operate at a high enough vapor density to make the process economical.

Molecular Laser Isotope Separation Process

The molecular laser isotope separation (MLIS) process is based on the isotopically selective excitation and decomposition of UF_6 gas. A mixture of UF_6 and a suitable inert carrier gas is expanded through a supersonic nozzle to obtain low temperatures. Under these conditions, the UF_6 is supersaturated, and a large fraction is in the ground vibrational state. The cold gas mixture is irradiated with an appropriate infrared laser, selectively exciting $^{235}\text{UF}_6$ to a higher vibrational state by the absorption of one or more photons per molecule. Subsequently, the mixture is irradiated with a second laser in the ultraviolet frequency band. This results in decomposition of the excited $^{235}\text{UF}_6$ to form a $^{235}\text{UF}_5$ enriched product, which is collected as a solid. Staging is likely to be required to achieve the desired degree of isotopic separation.

At this time, the MLIS process is still in the intensive R&D phase, where the major emphases are on laser development and fundamental process physics. Separations have not been demonstrated in systems comprised of components readily scaled up to production systems. Lasers of the required type, power, and pulse repetition rate are not presently available. Lasers are the pacing item in the development of this technology. The optical problems associated with the transport of laser light into the interaction region may be critical because of the high pulse energy and high average power required for the process. These problems have not been solved.

Plasma Separation Process

The plasma separation process (PSP) is based on the principle of ion cyclotron resonance in which one isotopic constituent of a stable plasma is excited to a higher ion temperature (and a larger cyclotron orbit) by means of a resonant radiofrequency (RF) electric field superimposed on a perpendicular constant magnetic field. One method used for collection of the desired isotope is by careful geometric design of static collectors, which favor deposition of the more energetic ions on their surfaces.

Use of a spatially neutral and magnetically stabilized plasma is expected to prevent space charge and electrode-related design limitations of conventional

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mass spectrometer devices, such as the calutron and quadrupole, thus allowing orders-of-magnitude enhancement of uranium throughput per unit volume of apparatus.

Processing occurs in a large horizontal cylindrical vacuum chamber located inside the field of a large cryogenically supercooled electromagnet so that the magnetic lines of force are parallel to the axis of the tank and a constant magnetic field is maintained (10 to 20 thousand gauss). The baseline process can be described in terms of five basic subsystems (source, excitation, collection, diagnostics, and control) and module hardware.

Schematic diagrams of the three AIS processes are presented in Figures 37, 38, and 39.

Licensing, Safety, and Environmental Status

As mentioned earlier in the discussion of the gas centrifuge program, enrichment plants will be owned by the U.S. Government and will not be subject to regulatory review and licensing by the NRC. The AIS technology is not developed sufficiently to define accident and safety considerations which should be considered in the design and operation of such facilities. Also, environmental issues related to AIS processes are not well defined. A large percentage of the issues to be resolved relate to occupational safety. For example, if economic tails stripping is achieved as a result of the program, national uranium resources would be extended and the significant environmental impact of mining and milling would be reduced. Whether such a reduction would be offset by the impact of AIS facility operation is not known.

Economics

Economic parameters for a mature enrichment technology are displayed in Table 64. An inspection of this data leads to the following observations.

- Mature plants are large and capital intensive.
- The major advantage of the gas centrifuge technology over gaseous diffusion is reduced power consumption.
- The AIS processes will potentially have lower capital costs than the mature processes and may have low power consumption.

Figure 37. Conceptual Schematic of a Molecular Laser Isotope Separation Process.

Figure 38. Conceptual Schematic of an Atomic Vapor Laser Isotope Separation Process.

Figure 39. Conceptual Schematic of a Plasma Isotope Separation Process.

Table 64. Economic Summary for Enrichment Technology Alternatives

Technology	Plant size (10 ⁶ kg SWU/year)	Capital cost ^a (\$ million)	Power consumption (MWe)	O&M cost ^b (\$10 ⁶ /year)
Gaseous diffusion	9	5300	2450	30
Gas centrifuge	8.8	6800	105	150
Advanced systems	8.8	<1000	~100	N.A. ^c

^aIncluding interest during construction, FY 1978 \$.

^bExclusive of power.

^cN.A. = not available.

2.3 FUEL FABRICATION

2.3.1 Introduction

Fuel fabrication technologies that may be applied to the principal NASAP reactor/fuel-cycle systems are described in this section. Fuel fabrication processes, their technology status, and the facilities needed to implement the processes are discussed. Finally, the RD&D requirements to support development of a commercially viable fuel fabrication technology are described.

2.3.2 General Considerations on Fuel Fabrication Technology

General Process Descriptions

Most of the commercial power reactors in use or under development utilize a fuel element fabricated by loading fertile and/or fissile material into fuel rods and assembling these rods into a fuel element. Although reactor designers have considered using oxide, carbide, nitride, and metal fuel, the oxide form currently is preferred. The oxide can be loaded into the fuel rod in the form of pellets or as particulate material, either as crushed and sized material in the vipac process or as sized microspheres in the sphere-pac process.

Because most reactors today use pellet fuel, this section discusses the pelletization process with only a brief discussion of the particulate fuel used in HTGRs and the sphere-pac process.

A brief description follows of the generic fuel-fabrication process used to make pelletized fuels keyed to the functional flow diagram shown in Figure 40.

Step 1.0: Reprocessing Plant Product -- Materials processed in the plant will include products of a reprocessing plant.

Step 2.0: Feed Material Preparation -- Uranium either as UO_2 or UF_6 , plutonium in the oxide form, mixed oxides (MOX), and ThO_2 will be received, and each lot will be analyzed for accountability and process control. UF_6

Figure 40.. Generic Functional Flow Diagram for Fuel Production.

will be converted to UO_2 before use. These materials will be blended to the assay required for each particular reactor fuel, and the blend will be analyzed to verify composition and homogeneity.

Step 3.0: Fuel Production -- This process function is to produce the oxide pellets. Blended oxides of the appropriate fissile/fertile materials are pressed into pellet form, sintered, ground to size, then inspected, analyzed, and weighed. PWR pellets are about 1/3 in. in diameter, BWR pellets are about 1/2 in. in diameter, and FBR pellets are about 1/4 in. in diameter. The pellets are generally about 90 to 95% of theoretical UO_2 density.

Step 4.0: Pin Fabrication -- Fuel pellets are loaded into Zircaloy or stainless steel tubes with the bottom end cap prewelded and the rod preloaded with buffer pellets. The open rod end is decontaminated, and the upper end cap is welded in place. If pressurization is required, the upper end cap is laser drilled, the pin is pressurized, and the hole is sealed by defocusing the laser. The fuel pins are then inspected and assayed to check fissile content, pellet placement, and surface flaws.

Step 5.0: Bundle Assembly -- Fuel pins will be assembled into a fuel element or bundle.

Step 6.0: Scrap Handling -- Wherever possible, scrap material, such as powder or pellets, will be recycled through the same unit operations. All other nonspecification material would be dissolved in nitric acid and converted by coprecipitation to powder for recycle after a solvent-extraction step.

Step 7.0: Waste Processing -- All liquid effluent from the fuel-fabrication plant is monitored for radioactivity. Nonradioactive liquid waste is released to the process sewer system. All radioactive liquid wastes are collected, and the radioactive constituents are removed and disposed of by mixing with cement or other solidifying agents. The radioactive solid is suitably packaged for shipment to a waste-disposal facility.

Solid nonradioactive waste is consolidated and disposed of in burial grounds. Solid radioactive waste (alpha waste) is collected, processed, and packaged for future disposal.

HTGRs utilize significantly different fuel designs and fabrication processes. The HTGR fuel element is a graphite block with a hexagonal cross-section drilled lengthwise with two different hole sizes: one to allow passage of the helium coolant and the other to contain the fuel rods. Fuel rods are formed by molding selected blends of fuel particles with graphite pitch, loading them into the graphite block, and baking the assembly at high temperature to graphitize the pitch. Because of its limited use, the HTGR fuel-rod fabrication process will not be discussed in detail.

The gel-sphere-pac process (for fabricating oxide fuels) utilizes high-density spheres of the required fuel, which are loaded into the fuel cladding by low energy vibration, to give a high-smear-density (85 to 88% theoretical density) fuel rod. Preparation of dry gel spheres is based on three major steps: (1) preparation of a special solution ("broth"), (2) gelation of broth droplets to give semirigid spheres, and (3) washing and drying to give a dry gel-sphere product suitable for calcination to product spheres. Gelation is accomplished chemically by the use of ammonia, which is formed within the broth droplet utilizing hexamethylenetetramine decomposition.

The sphere-pac process requires three size fractions to produce high-density fuel rods. For a mixed-oxide fuel, the fissile (U,Pu) O_2 spheres are prepared in large coarse (nominal 1200 μm diameter) and small coarse (nominal 300 μm diameter) sizes. The fertile fine UO_2 spheres are prepared with a nominal diameter of 40 μm and provide about 20% of the total heavy metal. It is anticipated that sphere-pac fuels will achieve 85 to 88% of theoretical density.

Process Facility Design

While the basic process functions remain unchanged, the design of the process facilities depends primarily on the characteristics of the fuel material being handled (i.e., its radiation, fissile content, and toxicity). Evaluation of the principal NASAP reactor/fuel-cycle systems indicates that the fuel elements for these systems can be fabricated in three types of facilities:

- Type A -- for fuels containing U-235, U-238, and/or thorium. These are fuels of relatively low toxicity and radioactivity.
- Type B -- principally for uranium-plutonium fuels. These are fuels with high toxicity and low-to-moderate radioactivity.

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- Type C -- for fuels with high gamma emission levels, such as U-233 fuels or spiked plutonium or HEU fuels.

Some of the options utilize a spikant, potentially Co-60, to enhance the radiation level of the fuel, potentially increasing its proliferation resistance. While neither the need for spiking nor the radiation levels required have been established, the assumption is made in this section that the spikant, possibly Co-60, will be added to the fuel before fabrication, requiring all spiked fuel to be fabricated in a Type C facility.

Type A Facility -- This facility and process line is designed to handle low-energy alpha-emitting fuels such as the low-enriched-uranium fuels used in the LWR and HWR. It is characterized as a facility using contact operation and contact maintenance. Fuel containment is provided by use of open-face hoods and dust-collecting systems.

Typically, a two-story steel structure would house the following facilities:

- UF_6 conversion -- process area containing UF_6 to UO_2 conversion equipment;
- pelletizing -- process area where UO_2 powder is pelletized;
- wet scrap recovery -- process area for recovery of uranium by dissolving solid scrap material, separating uranium, and recycling to conversion;
- support areas -- offices, laboratory, change rooms, safety, and laundry facilities;
- fuel rod assembly area -- process area for insertion of LEU fuel pellets into fuel rods;
- fuel bundle assembly -- process area for assembly of fuel rods into bundle assemblies and storage for shipping.

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The building is a carefully designed, airtight structure with controlled ventilation. A typical production facility would have a capacity of 520 metric tons of heavy metal per year and contain five production lines.

The facility is designed and operated in a manner to ensure that personnel are not exposed to radiation in excess of as-low-as-practical levels that will meet or be lower than those specified by regulatory agencies (e.g., 10 CFR 20). Full-scale licensed commercial facilities of this type are currently producing LWR fuels (see Section 2.3.3), and no licensing problems are expected for new plants.

Type B Facility -- This facility and process line is designed to handle material emitting alpha, beta, and low gamma radiations and includes fuel for the LWR (uranium-plutonium), GCFR (uranium-plutonium), and LMFBR (uranium-plutonium using recycled PuO_2). The composition of reactor grade plutonium is typically 58% Pu-239, 23% Pu-240, 14% Pu-241, and 5% Pu-242. The radiation dose at 1 m from 1 kg of unshielded material would be below 0.1 R/hr with the primary source being the buildup of Am-241. Fuel containment during processing is provided through use of glovebox-type enclosures or specially designed equipment that provides its own primary containment. Operation of the process equipment will be remote through a control room or panel. Equipment maintenance will be performed by individuals after the station area has been cleared of fissile material and decontaminated to the extent required. A typical production line would have a capacity of 500 metric tons heavy metal per year for LWR fuel or 250 metric tons heavy metal per year for LMFBR fuel. Criticality control limits the individual production line capacity. A typical plant would contain four or five production lines.

The facility would consist of a primary process building, a support building plus auxiliary buildings located on a relatively spacious site, thus providing space to accommodate the building and any necessary exclusion area.

The primary process building is a reinforced concrete shear-wall structure. It is classified as a critical structure and must withstand design-basis accidents (such as fires and explosions) and natural phenomena (earthquakes and tornadoes) while continuing to provide containment. The first floor contains the special nuclear material (SNM) receiving and assay areas and the remotely operated and shielded process operations. The basement contains material storage areas and utilities. The fabrication line control room, analytical chemistry control room, wet scrap control room, and an observation corridor overlooking the fuel fabrication area are located in a mezzanine above the process floor. Normal access to the fuel building is through personnel and

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equipment air locks that are controlled from the guard station in the support building.

The facility is designed and operated in a manner to ensure that personnel exposure to alpha, beta, gamma, and neutron radiation is below that required by Federal regulations (principally 10 CFR 20 and applicable regulatory guides) and meets those levels deemed by the licensing agency (NRC) to be as low as practically achievable. Both continuous and partial air monitors, liquid effluent monitors, and appropriate sampling will be used to monitor radiation levels and to isolate streams if their activity level exceeds a prescribed level. The design and operating procedure should ensure that the effect of the plant on the environment is minimized. Criticality control will be obtained by designing the system to a safe geometry for the fissile level expected (<20%) and by use of a strict accountability system. Safeguard systems that meet the requirements in the Code of Federal Regulations (10 CFR 70 and appropriate regulatory guide) will be used.

Type C Facility -- This facility would satisfy fabrication requirements for fuel cycles utilizing fuels having high gamma activity used in LWRs, LMFBRs, HTGRs, and LWBR converters and breeders. Typically the radiation level at 1 m from 1 kg of unshielded fuel would be from 10 to >100 R/hr. The facility and process lines are designed to handle high-energy gamma-emitting material such as U-233 (with associated U-232 chain) or any spiked fuel material. Fuel containment during processing is achieved by using remotely operated and maintained canyon or cell operations containing equipment designed to the extent possible to provide its own primary containment. Because of radiation levels, all operations of process equipment are remote.

The main building design is based on the concept of heavily shielded process canyons with surrounding areas for control rooms and laboratories -- all surrounded by an external building shell capable of withstanding natural and man-made design-base accidents. The outer shell of the process building (including the emergency power supply) also provides the final radioactive material containment barrier with the canyon construction providing secondary containment. Primary containment is within the process equipment within the main building. The fuel processing lines and the support lines are located and contained in independent cells. The cells are separated by concrete shielding to reduce in-cell radiation levels. Access to the independent cells is provided by removable covers. The building is provided with a central crane that can remove equipment from the cells through the roof covers into the repair facility. There is no manual personnel access to equipment cells up to and including fuel-rod final-end-cap welding; hence, all operations are fully remote. Contact maintenance in steps subsequent to the end

cap welding (i.e., fuel element assembly, storage, and shipping) is possible if desired.

The facility is designed and operated to ensure that personnel exposures will be both below the level required by the regulations (10 CFR 70 and appropriate regulatory guides) and acceptable to the as low as reasonably achievable (ALARA) requirement of the licensing agency (NRC). Both continuous and portable air monitors, liquid effluent monitors, and appropriate sampling will be used to monitor radiation levels in both process and working areas and to isolate streams, if their radioactivity levels exceed a prescribed level. The design of the plant is expected to minimize any effect on the environment. Criticality control will be based both on using safe geometries for process equipment and on a strict accountability system. Appropriate safeguards systems will be provided as required by the regulations and the licensing agency (10 CFR 70 and appropriate regulatory guides).

2.3.3 Status of Technology

Type A Facilities

The processes used in the Type A facility for LEU fuels are well developed and fully capable of producing an economically competitive product. Currently, there are five major licensed LEU fuel fabrication facilities in operation in the U.S. (Table 65).

Generally, further R&D efforts would be utilized to meet ever-changing environmental and regulatory requirements while maintaining or improving product quality and cost. Thus, the impetus to perform the indicated R&D will come from the regulatory agencies [NRC, EPA, and the Occupational Safety and Health Administration (OSHA)] and competitive pressures (cost and quality) rather than an incompletely developed process. Any development work desired for the LEU Type A fabrication plant design does not involve the development of new technology or even hardware development. What is needed is the application of existing materials handling and diagnostic capabilities to the LEU plant operation with the implementation of the associated software and sequencing programs.

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Table 65. Major U.S. Fabrication Facilities
for Low-Enriched Uranium Fuel

Company	Location	Estimated capacity (metric tons per day)
Westinghouse	Columbia, S.C.	5-7
General Electric	Wilmington, N.C.	5
Babcock & Wilcox	Lynchburg, Va.	3-5
Combustion Engineering	Windsor, Conn.	2-3
Exxon	Richland, Wash.	2-3

Type B Facility

The Type B facility is designed to fabricate standard mixed-oxide (uranium-plutonium) fuels. Development of the technology for plutonium recycle began with the Plutonium Utilization Program at Hanford in 1956, and continued with the Saxton MOX fuel development program. While government-supported R&D was essentially complete by 1970, major industrial programs were initiated in 1967 and culminated in demonstration programs using mixed-oxide fuel in commercial reactors. As a result of the experience gained in both the U.S. and in similar foreign plutonium recycle programs, it was demonstrated that plutonium recycle was technically feasible.

The nature of plutonium, particularly its toxicity, is such that many of the fabrication operations cannot be performed in equipment typical of that used in commercial LEU Type A fuel fabrication facilities. Further, the present development effort has largely been carried out in laboratory or small engineering-scale facilities. Although these small facilities have been constructed, only the B&W-operated NUMEC plant at Apollo, Pennsylvania, is operating (Table 66).

In view of the above experience and proposed changes in allowable dose rates and plant emissions, a considerable R&D program focusing on engineering development is needed to advance the technology to a commercial status.

Type C Facility

This facility is designed to fabricate fuels having high gamma activity. While various areas of technology have been investigated by industry and government laboratories, the development has progressed to a hot engineering-scale development stage in only a few cases. Since the fabrication operations cannot be performed using equipment or technologies developed for the LEU Type A facilities, an extensive R&D program focusing primarily on engineering development studies will be needed to advance the current state of technology to that required for a commercial facility.

2.3.4 Licensing, Safety, and Environmental Status

The present commercial fuel fabrication industry is based on LWR fuel. Several of these licensed fabrication facilities are considering expansion of

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Table 66. Mixed-Oxide Fuel Fabrication Facilities in the U.S.

Organization ^a	Location	Capacity (metric tons per day)	Status
Babcock & Wilcox-NUMEC	Apollo, Penn.	0.1	Operating
Exxon	Richland, Wash.	0.1	Shutdown
Westinghouse	Cheswith, Penn.	0.1	Shutdown

^aIn addition to the above, mixed-oxide facilities have included United Nuclear (Pauling, N.Y.), Kerr-McGee (Cimmaron, Okla.), General Electric (Vallecitos, Calif.), and government laboratories.

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capacity and process modifications. No problems are expected in amending the licenses to cover these changes.

To date, there have been no off-site environmental effects from accidents in low-enrichment uranium dioxide fuel fabrication plants. Accidents have been rare in these plants because the plant design and operation are safety oriented. Occupational dose commitments and chemical exposures resulting from accidental releases are also estimated to be quite low because the materials are not very hazardous and the plants are licensed and regulated by the government.

The very small quantities of radionuclides now released in the LWR fuel fabrication effluent gases and liquids result in concentrations of radioactive material in these media that are about 0.04 and 10%, respectively, of 10 CFR 20 limits at the site boundary.

Kerr-McGee and B&W (Lynchburg facility) have licensed development facilities that have been working on plutonium fuel rod production for the Fast-Flux Test Facility (FFTF). The Kerr-McGee plant is now on standby, and Kerr-McGee is considering decommissioning it. Westinghouse has filed an application to construct and operate a commercial plant in South Carolina. The suspension of the Generic Environmental Statement for LWR Mixed Oxide Fuel (GESMO) hearings resulted in NRC stopping all case review of the Westinghouse application.

Licensing of plutonium-containing mixed-oxide fuel [10 CFR 70.22(f)] requires plants to be designed, fabricated, constructed, tested, and operated under quality-assurance programs. Quality assurance comprises systematic actions necessary to provide confidence that structures, systems, components, and operations will perform satisfactorily in service (10 CFR 50, Appendix B).

All operations at mixed-oxide fuel fabrication plants that involve handling plutonium (except that contained in shipping containers or fuel elements) are carried out inside process enclosures such as cells or gloveboxes. These enclosures provide confinement of plutonium in the event of equipment failure. The process building, essential equipment, and supporting systems are designed to withstand natural phenomena, including tornadoes, earthquakes, and floods. During the life of a mixed-oxide fuel fabrication plant, equipment or accessory failures may occur. Monitors are installed to detect any failure or abnormal condition that could cause safety-related damage. Corrective action is automatically provided. The ventilation system is designed

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to function during and after accidents and to pass all plant-ventilation air through high-efficiency particulate air (HEPA) filters before releasing it to the atmosphere. In summary, a mixed-oxide fuel fabrication plant is:

1. designed, fabricated, constructed, tested, and operated according to a quality-assurance plan;
2. designed to cope with accidents;
3. designed to minimize the off-site consequences of potential accidents.

Some incidents, such as glove tears or other glovebox malfunctions, are expected to occur as part of the normal operation. Other more serious accidents, such as glovebox-window breakage, will occur far less often. Although their off-site consequences are judged to be insignificant, such accidents are included in the estimate of airborne effluents resulting from normal operations.

Analysis of the consequences of such an accident and minimization of the impact receive significant treatment in design and licensing actions. An extensive set of regulatory requirements exists to protect the environment from releases from recycle facilities. These are expected to be reviewed and redefined before fuel recycle is permitted.

In order to be licensed, fabrication plants for plutonium recycle fuel must meet the same regulatory requirements to protect the environment as LWR reprocessing plants. The design of the fuel fabrication plant must adhere to the concept of no releases of radioactive liquids to the environment. Aqueous solutions containing plutonium will be treated for plutonium recovery, concentrated, and solidified. Noncondensables will be filtered and released as gas. Plutonium-containing solids are treated for plutonium recovery, residues are converted to a nonleachable form, transferred to stainless steel containers, and stored in a stable geologic formation. Solids that contain small amounts of beta and gamma activity are buried at licensed sites. Gases containing radioactive and noxious material are filtered and processed before release.

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No sphere-pac fabrication facility has been licensed in the U.S., nor has an application for a license, a safety analysis report, or an environmental impact report been filed with NRC. Such filing is not probable until the current restrictions on use of plutonium are lifted and reprocessing plants are authorized for operation. The operations and equipment required for a sphere-pac plant are sufficiently similar to those used in uranium/plutonium dioxide fuel fabrication plants and reprocessing plants, so that licensing problems should be equivalent in magnitude.

A licensed plant for HTGR U-235 fuel manufacture currently exists at General Atomic in San Diego, California. Future plants must be designed to meet future regulatory requirements; these may increase costs and engender delays if regulations change during plant construction. The various U-235 enrichment fuel-cycle alternatives for the HTGR present no basic differences, from a safety and licensing standpoint, from the previous reference high-enrichment (~90%) uranium cycle.

Recycled thorium (together with recycled U-233) has associated significant gamma activity and radon emissions, causing most operations to be remote and highly automated. The current limited experience does not provide assurance that licensing difficulties will be readily overcome. No facilities utilizing remote operation and remote maintenance have been licensed to date.

2.3.5 RD&D Needs

General RD&D Program

The facilities and studies required to carry out an R&D program include the following steps:

Cold Laboratory Development -- The feasibility of the concept is developed and demonstrated using laboratory-scale equipment and cold or simulated fuels. These facilities continue at a reduced scale to support the engineering-scale programs.

Hot Laboratory -- Small-scale unit operation equipment is demonstrated using full-level radioactive materials. These studies continue at a reduced scale to support the engineering-scale programs.

Cold and Hot Engineering Development -- Engineering-scale equipment is designed and demonstrated using cold (nonradioactive) or synthetic process material. Successful equipment models are then used in hot (radioactive) engineering tests to provide preliminary operational efficiency, control, and maintenance information needed to design the full-size equipment to be used in the pilot plant test.

Cold Prototype and Hot Pilot Plant -- A fully integrated, remotely operated (and maintained for high gamma fuels) fabrication facility is constructed and demonstrated. This facility will demonstrate not only normal operations but also the operations that would be required to recover from malfunction of process equipment and operational error. Current concepts incorporate approximately full-scale equipment items in the pilot plant. Successful operation of the facility will constitute demonstration of all equipment concepts. The facility would operate for 10 years. During this period, the fuel is utilized in development and prototype reactors. A characteristic feature of the hot pilot plant is its flexibility. Equipment can be quickly and readily replaced and different equipment and processes demonstrated. Before the facility processes radioactive materials, all systems are evaluated in the "cold prototype" mode. This 1- to 2-year period is used to confirm equipment design, transfer system interface, and maintenance requirements.

Pilot plant throughput, to support about 1 GWe of installed electrical capacity, will be about 50 metric tons per year for LWK, LWBR, or HWR fuel, or about 25 metric tons per year for LMFBR fuels. These throughput differences result primarily from differences in core size and the fissile content of the fuel.

Lead Plant -- A lead commercial-scale fuel fabrication plant would be built and operated to process fuel for commercial reactors. This plant would demonstrate that a commercial plant can operate in a safe, environmentally acceptable manner with full accountability and safeguards for nuclear fuel. As it is the lead or first-of-a-kind full-scale plant, it will be more conservatively designed than will the follow-on or mature plants. Continuing operation of the plant will demonstrate the economic feasibility of commercial fabrication of nuclear fuels. Throughput would be about 500 metric tons of heavy metal per year for LWR, HWR, and LWBR fuels, and 250 metric tons per year for HTGR and LMFBR fuels (Table 67). These throughput differences result from differences in core size and reload requirement. In addition, the process-line size would vary to compensate for the fissile content of the fuels. A plant would normally have several parallel fabrication lines.

Mature Plants -- The commercial plants will be redesigned and constructed on about the same scale as the lead plant and will differ only to exploit the

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Table 67. Estimated Cost Range for Development and Commercialization of a New Fuel Fabrication Technology (\$ million)^a

	Type A (uranium fuel)	Type B (plutonium fuel)	Type C (high gamma fuel)
Base technology R&D	5-15	110-140	190-270
Hot pilot plant	None	120-150	240-300
Licensing support for lead plant	None	75-100	150-250
Total cost of RD&D	5-15	305-390	580-820
Cost of lead plant	^b	425-600 ^a	750-1200 ^c
Lead plant capacity (metric tons of heavy metal per year)	500	250-500 ^d	250-500 ^{d, e}
GWe supported	15-20	15-20	15-20

^aJanuary 1978 \$.

^bFor purposes of comparison, a 500-metric-ton-heavy-metal-per-year plant would cost about \$60 million.

^cCost of lead plant would be supported by sale of fabrication services and is not considered an RD&D cost. Lead-plant cost tabulated does not include licensing support, which is part of RD&D.

^dFast-breeder fuel would be at the low end of the capacity range.

^eA facility for supporting 15 to 20 GWe of HTGR capacity using U-233 fuel would have a capacity of about 150 to 200 metric tons of heavy metal per year. The cost of the program would be expected to fall within the range reported above.

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experience and data acquired by operation of the lead plant. Throughput would be the same.

Key items that must be considered throughout the RD&D program include:

- shielding of equipment from gamma radiation,
- completely remote maintenance,
- modular equipment with high reliability,
- facility layout and plant systems interaction problems,
- extensive nondestructive assay equipment development,
- process cell maintenance problems.

RD&D Cost and Schedule

Estimated cost ranges and times for the development and commercialization of a new fuel refabrication technology (Table 67) show that the total cost to develop a new technology for recycle fuel would range between \$305 to \$820 million. The costs of the program are modest during initial base technology R&D, and they increase during later years to cover continued R&D, the hot pilot plant, and licensing support for the lead plant. Also, the costs of facilities for Type C plants (U-233 or spiked fuel) are at the high end of the range of costs.

The scheduling of nuclear fuel fabrication plants is directly related to the construction of nuclear reactors that will use the fuel. It takes less time to construct fuel fabrication plants than reactors, but a full fuel core representing a significant production effort in the fabrication plant must be available before reactors start up. For a particular fuel, then, the research and development phase must be completed, a fabrication facility must be constructed, operation testing and startup must be completed, and at least one full core must be produced during or before reactor construction.

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The laboratory and engineering phase R&D can be completed in about 12 years from the start of a program for a Type B or Type C fuel. Construction of the hot pilot plant can be initiated about 8 years after program start, somewhat overlapping the R&D. Operation of the hot pilot plant begins about 15 to 18 years after the start of the program, with the longer time required for the Type C fuel. The development of technology for LMFBR $\text{PuO}_2\text{-UO}_2$ fuel has progressed to the point where a hot pilot plant could be authorized for construction now.

A typical construction schedule for a remote-operated contact-maintenance lead or Type B pilot plant is shown in Figure 41, assuming the R&D phase has been completed. Thus, a pilot plant would require approximately 3 years of production to produce the first full-sized reactor core and slightly less than 2 years to produce the next and subsequent cores. Construction would have to start at least 9 years before the reactor started up to ensure production of one full-sized core. The schedule could be relaxed slightly to produce a smaller demonstration-sized reactor core.

Construction time for HTGR fabrication facilities is about the same as for the oxide facilities, and requires 6 years to construct a pilot plant, a lead plant, or a mature plant. The time to full production is estimated to be 3 years.

2.3.6 Plant Costs in a Mature Industry

Direct investment cost ranges for mature industry fabrication plants each able to support 15 to 20 GWe of nuclear capacity are estimated at about \$50 to \$180 million for Type A plants, \$400 to \$600 million for Type B plants, and \$700 to \$1200 million for type C plants. These cost estimates do not include interest during construction, which would add 35 to 40% to the investment cost, or owner's cost during construction, which would add 5 to 10% to the Type B and C plant costs and 15 to 40% to the less-capital-intensive Type A plant costs.

The upper end of the investment cost range for contact (Type A) plants is for the fabrication of HTGR fuels. The direct plant investment cost for low enriched UO_2 fuels for LWRs is about \$60 to \$70 million. For type B and type C facilities, the costs at the lower end of the cost range are for water-reactor fuels and those at the higher end are for fast-breeder fuels. Furthermore, the high end of the range assumes that the reductions in plant costs may be offset by additional costs incurred during licensing, while the

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MILESTONE EXPLANATION

1. LICENSING BEGINS
2. CONSTRUCTION BEGINS
3. CONSTRUCTION COMPLETED
4. HOT OPERATION BEGINS
5. PLANT OPERATION AT FULL CAPACITY

NOTE: THIS SAME CONSTRUCTION SCHEDULE IS APPLICABLE TO HTGR FABRICATION FACILITIES.

Figure 41. Pilot/Lead Plant Construction Schedule for a Type B Facility.

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low end reflects a net cost reduction achieved by learning and technology advancements.

As the fuels industry grows, plant sizes may become larger. Some savings may be made because of economies of scale. This savings may be reduced, however, because of the need for parallel lines in the fabrication operation. It is estimated that doubling the plant size can be done with an additional capital investment of 60 to 70% of the base figures.

The ranges of estimated typical industry unit costs in \$/kg of heavy metal fabricated into fuel are given in Table 68. The unit costs at the lower end of the cost ranges are for high-capacity plants with favorably resolved cost uncertainties. The costs at the higher end of the uncertainty range assume that cost uncertainties are unfavorably resolved. The wider cost ranges for HTGRs are due to the different methods for fuel loading and the different fuel fabrication processes now under consideration. The range will be narrowed as the industry matures.

Type A facilities are the least capital intensive, while Type C facilities are the most capital intensive. The unit costs given in Table 68 are composed of a capital and operating cost component. For LWR Type A fuels, about 40% of the unit cost is related to the initial capital investment, while 60% is derived from operating costs including materials. For LWR Type B fuels, this capital cost component rises to about 70%; it is about 80% for Type C fuels. These percentages are somewhat higher for the HWR fuel but are lower for LMFBR and HTGR fuel because of the more complicated LMFBR assembly and greater relative material costs for these fuels. Once built, unit costs of Type B and Type C facilities cannot be reduced by technological improvements. As capital intensive facilities, they are relatively insensitive to operating cost increases.

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Table 68. Estimated Unit Cost Range for Fuel
Fabrication, \$/kg•Heavy Metal^{a, b}

Reactor type	Type A (uranium fuel)	Type B (plutonium fuel)	Type C (high gamma fuel)
LWR	110-140	300-550	450-850
HWR	60-120	250-450	350-700
HTGR	300-900	<i>c</i>	600-2100
FBR ^d	<i>c</i>	400-800	700-1400

^a\$/kg of heavy metal fabricated.

^bJanuary 1978 \$.

^cNot applicable.

^dBased on core plus axial blanket material throughput.

2.4 FUEL REPROCESSING

2.4.1 Introduction

Fuel reprocessing technologies are discussed in this section; the critical issues involved in fuel reprocessing are identified. Also, the research and development requirements to advance fuel reprocessing technology to commercial use are described.

2.4.2 General Considerations on Fuel Reprocessing

Currently there are no commercial fuel reprocessing plants in operation in the United States. However, about 250 metric tons of uranium of LWR fuels and 375 metric tons of uranium of metallic fuel were processed commercially in the United States at the NFS Plant at West Valley, New York. That facility has been shut down permanently because the cost of backfitting modifications to meet constraints on emissions and of improving resistance to natural phenomena such as earthquakes and tornados could not be economically justified by the owners.

Aqueous Reprocessing

The principal fuel reprocessing technologies that may be applied to the diverse reactor fuel designs utilized by the NASAP reactor/fuel-cycle systems (see Appendix A) are currently based on fuel breakdown by mechanical steps, such as shearing, followed by fuel dissolution and aqueous reprocessing. Aqueous reprocessing basically involves a series of operations in which (1) spent fuel and blanket materials are dissolved in nitric acid; (2) the resulting nitric solution, after adjustment and clarification, is treated in a solvent extraction process where the plutonium, uranium, and thorium are separated from the fission products and from each other; (3) the resulting product streams are converted to appropriate solid form and composition for transfer to a refabrication plant; and (4) the waste streams are processed to produce solid waste products suitable for transfer to and storage in a federal waste repository. The basic steps for all aqueous reprocessing schemes are presented in Figure 42. The more important process steps, along with features unique to certain fuel types, are described below. Steps 1.0 and 2.0 are related only to material handling and are not described further.

Figure 42. Generic Flow Diagram for Aqueous Fuel Reprocessing.

Step 3.0: Mechanical Head End -- Most of the commercial power reactor fuel assemblies consist of a number of parallel tubes of either zirconium alloy (typical for LWRs, HWRs, or LWBRs) or stainless steel (typical for LMFBRs) containing pellets of the oxides of fissionable material enclosed within a metal shroud. The fuel pellets are exposed in the mechanical head end using physical equipment to disassemble and shear the tubes into 1- to 2-in. segments.

HTGR fuel, however, is unique, consisting of fuel spheroids or microspheres in a graphite matrix. The HTGR fuel reprocessing sequence starts with the head-end operation, where the fuel particles are separated from the graphite block by crushing and fluidized-bed burning. Subsequent head-end operations separate the fissile particles containing the residual U-235 from the thorium fertile particles containing bred U-233 and convert the fissile particle to an oxide suitable for dissolution. The burned fertile particles can be dissolved after the initial crushed fuel element burning operation. The burned fissile particles contain an inert silicon carbide coating that must be crushed and inner carbon coatings that must be burned prior to dissolution. Gaseous fission products, C-14 and tritium, are recovered from the combustion off-gas.

Step 4.0: Tritium Removal -- This operation removes and recovers tritium and some of the other gases from the sheared fuel by heating the fuel in an oxidizing atmosphere prior to dissolving the fuel. The interstitially bound gases are released as the oxide fuels pass through a phase change as they are oxidized to U_3O_8 . In the case of the HTGR fuel, as stated under step 3.0, tritium is removed with the combustion off-gas. Subsequent HTGR processing is similar to that of oxide fuel.

Step 5.0: Dissolution -- The sheared and oxidized fuel is dissolved in nitric acid. After separation from the undissolved cladding and insoluble constituents, the solution is transferred to the solvent extraction area. The cladding is washed and discharged as a waste stream. All thorium based fuels are difficult to dissolve and probably will require a nitric-HF dissolvent.

Step 6.0: Feed Preparation -- The dissolver batches are accumulated and their pH adjusted for solvent extraction.

Step 7.0: Solvent Extraction -- Solvent extraction separation makes use of the relative differences in the equilibrium distribution between aqueous and organic phases for the various elements. Tributyl phosphate (TBP), usually

in concentrations of 15 to 30% in dodecane, a kerosene-like diluent, is an extremely effective organic extractant because of the addition compounds it forms with uranium, plutonium, and thorium, making them soluble in the organic phase, whereas the fission products are not. Reprocessing schemes involving only uranium and plutonium are keyed to a basic solvent extraction process known as PUREX, and those involving thorium, to a process known as THOREX. In Table 69, the reference PUREX and THOREX processes are listed and characterized along with four other process schemes -- two keyed to PUREX and two keyed to THOREX -- modified for possible enhanced proliferation resistance.

The basic PUREX process, PUREX 1, is illustrated in Figure 43. The process results in a stream of fission product waste, a stream of purified uranium, and a stream of purified plutonium. By varying the conditions in the solvent extraction cycle, plutonium can be stripped with a uranium diluent, so that plutonium is never by itself in solution -- PUREX 2 (coprocessing). It is also feasible to leave the coprocessed plutonium in a partially decontaminated state so that it is always highly radioactive -- PUREX 3.

PUREX 4 is a variation of the coprocessing option in which a spikant (probably Co-60) is added to the product streams to provide additional proliferation resistance.

The basic THOREX process, THOREX 1, is similar to the PUREX process except that it involves fertile thorium and fissile U-233. It results in purified thorium and purified uranium streams. A THOREX process modified to accommodate plutonium, as well as uranium and thorium, can be visualized by inserting a second partitioning column after the one shown in Figure 43. This THOREX 3 process would conceivably result in purified streams of plutonium, thorium, and uranium. Note that, by varying the conditions in the partitioning columns, the thorium conceivably may be carried along with either the uranium or the plutonium streams. The THOREX process for HTGR fuel with its special head end has been designated as THOREX 5.

Step 8.0: Product Conversion -- The product streams from the PUREX and THOREX solvent extraction systems are converted to solid before shipment to the fuel-fabrication facility. Plutonium streams are precipitated as the oxalate, which is then calcined to plutonium oxide, packaged, and shipped. Uranium is converted to uranium trioxide by thermal decomposition, followed by reduction with H_2 to UO_2 . In the coprocessed case, plutonium and uranium are coprecipitated and reduced. Thorium is precipitated as the oxalate and converted to the oxide by thermal decomposition.

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Table 69. Aqueous Reprocessing Schemes

Aqueous scheme	Reactor fuels	Product		
		Stream 1	Stream 2	Stream 3
PUREX 1 - Reference	LWR U/Pu recycle	U	Pu	
PUREX 2 - Coprocessing	LWR U/Pu recycle	U-Pu	U	
	FBR U/Pu core and blanket	U-Pu	U	
PUREX 3 - Coprocessing with partial decontamination	LWR U/Pu recycle	U-Pu ^a	U	
	FBR U/Pu core and blanket	U-Pu ^a	U	
PUREX 4 - Coprocessing with added spikant	LWR U/Pu recycle	U-Pu ^a	U	
	FBR U/Pu core and blanket	U-Pu ^a	U	
THOREX 1 - Reference HEU/Th	LWR HEU/Th	U	Th	
	LWBR HEU/Th	U	Th	
	FBR Th blanket	U	Th	
THOREX 3 - Mixed U-Pu-Th	LWR Denatured U/Th	U	Pu	Th
	LWBR Denatured U/Th	U	Pu	Th
	FBR Denatured U/Th	U	Pu	Th
	FBR Pu/Th core	U	Pu	Th
	LWBR Pu/Th	U	Pu	Th
THOREX 5 - HTGR	HTGR Denatured U/Th	U	Pu	Th
	HTGR HEU/Th	U	Th	

^aRadioactive.

Figure 43. Schematic Representation of Countercurrent Extraction Equipment in Basic PUREX Process.

Step 9.0: Waste Processing -- All of the steps in the fuel processing facility produce waste. The off-gas, particularly from the tritium removal and dissolution steps, is treated to recover tritium and Kr-85. These are packaged and sent to waste disposal. All other off-gases are filtered to keep the release of particulate radioactive material to ALARA levels. Cladding hulls are packaged and sent to waste disposal. High-level fission product waste is solidified, usually in the form of a glass, before being sent to waste disposal. In the case of HTGR processing, the CO₂ off-gas must be collected in order to isolate the C-14 produced in the graphite moderator.

Pyrometallurgical Process

One NASAP reactor/fuel-cycle system utilizing a pyrometallurgical reprocessing system was evaluated (see Appendix A, System 6.6). This is a low decontamination process applicable principally to metal-fueled FBRs. In principle, it might be useful also with oxide fuels if appropriate head end and conversion steps are added. The proposed nonaqueous process is in the conceptual stage, supported by small hot cell data. The process for metal fuels consists of removing the fuel clad mechanically, dissolving the fuel in molten metal (Cu-5% Mg), and extracting the alkaline earth and some of the rare earth fission products into a molten salt. The plutonium, a portion of the uranium, and remaining rare earths are then extracted into a second molten salt (MgCl₂ and MgF₂), which is back extracted into molten zinc contained in another vessel. Separation of the zinc from the actinides occurs by evaporation of the acceptor alloy, leaving a mixture of uranium and plutonium behind that is contaminated with fission products, predominantly of the rare earth group.

Materials of construction for process equipment are a challenging problem because of corrosive and high-temperature operating conditions (tungsten for the molten metal equipment, beryllia for the zinc vaporization retorts), and the loss of product to waste may be high.

2.4.3 Critical Issues

The nuclear industry encountered major difficulties in implementing a nuclear fuel reprocessing capability in the United States even before the President indefinitely deferred the fuel-cycle option for nonproliferation reasons. Mechanical head end equipment and liquid effluent problems plagued operation of the first small-scale commercial reprocessing plant at West Valley, New York, and start-up problems led to the decision not to begin hot operations at the GE plant at Morris, Illinois. Factors contributing to the problems of the industry were the adverse public reactions to releases of radioactivity,

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leading to stricter requirements for control of wastes and effluents that came into existence in the 1970s. As a result, there have been major uncertainties in the fuel recycle program in the United States which continue to the present.

A basic problem is projecting the future growth and direction of the commercial nuclear power industry. Specifically, it is necessary to predict the growth scenario of specific reactors and the associated fuel-cycle system so that industrial fuel reprocessing can be established at the appropriate time and on the scale needed. This is critical from the standpoint of establishing the scale of the technology to be developed and the principal steps to be included in the development.

In addition, there are difficulties inherent in resolving the technical problems associated with particular reactor/fuel-cycle systems, including:

1. development of equipment and processes to separate and recover the fertile and fissile material in spent-fuel assemblies in a form (solid) suitable for both storage and shipment,
2. development of the process and equipment to convert high-level wastes into a solid form suitable for terminal storage in a Federal waste repository and to remove and collect gaseous radioactive effluents before release to the atmosphere, and
3. implementation of appropriate safety (seismic, shielding, effluent controls, etc.) and safeguard (security-accountability) criteria.

2.4.4 Licensing, Safety, and Environmental Status

Three commercial fuel reprocessing facilities have been constructed, but only one of these, the Nuclear Fuel Services (NFS) plant at West Valley, New York, was actually operated. The NFS plant, as constructed and operated in the late 1960s and early 1970s, would not meet current regulations. The General Electric plant at Morris, Illinois, was not commissioned because of technical problems and the volatile status of licensing and environmental issues, and licensing proceedings on the Barnwell Nuclear Fuel Plant (Allied General Nuclear Services) in South Carolina have been suspended. Licensing proceedings for the construction of a fourth plant (Exxon Nuclear Corporation) were begun but were later suspended. Safety analyses and environmental

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reports for the latter two facilities were prepared by the applicants, but final actions have not been taken by the NRC. Generic environmental and safety evaluations of fuel reprocessing have been prepared by the NRC.

Generic problems continue to impede the licensing of any reprocessing facility. These include:

- establishment of regulations for allowable releases limits for gaseous fission product effluents. Kr-85 release limits have recently been specified by EPA, but the technology to ensure adherence to these limits remains to be demonstrated. In the case of C-14 and tritium, permissible release limits have not been established. Although satisfactory processes have been demonstrated on a laboratory scale, they have not been scaled up;
- establishment of acceptable high-level waste forms. Waste management technology development has been impeded because the choice of an acceptable final waste form has not been made and, in turn, because it may depend on the selection of final repository;
- establishment of the appropriate level of safeguards, the required accuracy for accountability for strategic nuclear materials, and the possible need for proliferation countermeasures, such as coprocessing and spiking of plutonium product and inclusion of proliferation-resistant engineering features.

Fast-breeder reprocessing plants are expected to handle spent fuel containing relatively more plutonium and short-lived fission products than plants designed for LWR fuel. Licensing breeder fuel reprocessing plants may present additional problems related to spent-fuel characteristics and to unique LMFBR head-end procedures (e.g., sodium handling). The technical data required to assess the safety and environmental aspects of FBR fuel processing will be provided by the planned RD&D program.

Thorium fuel reprocessing plants are sufficiently different from uranium-plutonium recycle plants that some additional obstacles to licensing could be anticipated. Technology for controlling the emissions of Rn-220 and compacting the large quantities of C-14 containing waste in the case of HTGR fuel would have to be demonstrated in RD&D programs. Demonstrations of the reliability of remotely operated equipment to isolate radioactive-partitioned products -- U-233 and Th -- could also cause licensing delays.

2.4.5 RD&D Program Needs

Status of Technology

Facilities to reprocess fuel from power reactors exist in Belgium, France, Germany, Great Britain, India, Italy, and Russia. A number of countries are operating small reprocessing facilities in connection with RD&D programs. Small experimental LMFBR reprocessing facilities have been operating in France and the United Kingdom. Hot cell studies, directed principally toward FBR fuel reprocessing development, have been carried out in the United States over the last decade. The basic technology for reprocessing of uranium and uranium/plutonium oxide pellet fuels with low burnup exists in the PUREX process. However, a commercial reprocessing plant for mixed-oxide power reactor fuels that conforms to current U.S. Federal and state requirements has not yet been operated. Specific areas that still require some development work include:

- operation and maintenance of the mechanical head-end equipment,
- methods for handling highly radioactive residues that remain after the dissolution of high-burnup fuel,
- technology for reducing radioactive off-gas releases (e.g., Kr-85, iodine, and tritium) to conform to anticipated regulations,
- remotely operated and directly maintained conversion processes for plutonium from power reactor fuels,
- high-level waste solidification and vitrification to prepare for terminal storage.

Coprocessing solvent extraction has the technical problems of the reference PUREX process and, in addition, would require careful optimization of flow rates and mass balances and very careful process control. Conversion of the coprocessed product to an acceptable form for fabrication also would require some development. Several conversion processes are currently under development in the United States.

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Coprocessing with low decontamination introduces additional developmental needs. In reprocessing, expanded remote operation and maintenance of the plant would be required. If decontamination factors of at least 100 are maintained, only modest changes to present technology would probably be required. At a minimum, the chemical processes would be more uncertain. Retention factors for the held-over fission products would have to be established in hot cell tests, with subsequent demonstration at the production level. In product conversion, existing approaches would have to be modified to accommodate remote operation and maintenance.

The technology for recycle of fast reactor fuels based on the uranium-plutonium cycle is similar to that for LWR fuels on the same cycle; there are, however, important differences that make fast reactor fuel recycle technology more difficult than recycle technology for LWRs. Significant differences exist between FBRs and LWRs because of the higher irradiation levels, the higher fission product concentrations, the higher fissile contents of the fuel, and the different fuel element cladding and designs.

Areas requiring more RD&D emphasis include:

- development and maintenance of shearing equipment for highly irradiated fuels;
- dissolution of fuels with higher plutonium content, feed clarification, and collection of larger quantities of insoluble fission product residues;
- possible increased solvent degradation because of higher radiation levels.

Fuel recycle needs for LMFBRs and GCFRs are quite similar, so the preceding discussion is generally applicable to both fuels. However, for the LMFBR, fuel cleaning for sodium removal is needed.

The technology for reprocessing thorium-based oxide pellet fuels is less advanced than that for uranium-based fuels. The THOREX process has been used to process irradiated thorium oxide fuels of low burnup in Government plants and, in limited quantities, in a small-scale industrial plant. The Federal Republic of Germany is undertaking laboratory-scale development of thorium fuel reprocessing for its gas-cooled pebble-bed reactors. Thorium oxide

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fuels have not been processed in a large-scale plant specifically designed for thorium processing, nor has highly irradiated thorium fuel been processed by the THOREX process in engineering-scale equipment.

The principal differences between the reprocessing development required to reprocess metal-clad thorium-based oxide fuels and graphite-based HTGR fuel occur in the head-end treatment. Partitioning of fuel materials from both classes of reactor fuel can then be accomplished by a THOREX-type solvent extraction process.

In the case of metal-clad oxide fuels, additional head-end process RD&D is required to determine how to dissolve ThO_2 fuel in the presence of zirconium. Significant waste handling problems are encountered, because fluoride is required to dissolve ThO_2 .

Thorium-based fuels containing much greater quantities of uranium and plutonium, than handled in prior applications of the THOREX process technology, would be encountered from denatured fuel and LWBR prebreeder fuel cycles. Thus, additional R&D will be required to address the unique problems associated with these fuel compositions and to validate the conceptualized flow sheets used for the evaluation of alternatives.

In the case of the head-end process development for graphite-based HTGR fuels, development work is needed with irradiated materials in the crushing, burning, and particle separation operations and in the treatment of C-14-containing off-gases associated with the head end of the reprocessing plant.

Specific areas of solvent extraction process development work required to reprocess all thorium-containing reactor fuel include:

- development and maintenance of shearing equipment for highly irradiated fuels;
- technology development for containing Rn-220 and other radioactive gases to conform to regulations;
- recovery of fully irradiated thorium in large-scale facilities;

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- partitioning of fuel solutions containing uranium, plutonium, and thorium;
- recovery and handling of highly radioactive product streams;
- process and equipment design integration;
- high-level waste concentration and vitrification.

General Development Program and Schedule

The sequential steps in a reprocessing development program would be process flow sheet development, pilot plant hardware development, construction and operation of a cold integrated pilot facility focusing on the mechanical head end, and then operation of a "hot" pilot plant.

The hot pilot plant would operate initially as a development tool but could accommodate the reprocessing requirements of early reactors. An important judgment is that a hot pilot plant of approximately one-tenth full commercial scale is required to complete the development for the complex fuel cycles. Included are breeder cycles and all thorium cycles for LWR and LWBR reactors. Because of the uniqueness of the HTGR fuel, special consideration must be given to the selection of an appropriate pilot plant size. For coprocessing and/or spiked uranium-plutonium LWR fuels, the commercial industry could possibly evolve without this pilot plant phase. However, such a pilot plant could improve technology for a modified LWR uranium-plutonium reprocessing plant.

The generic steps required to bring a nuclear reprocessing concept to a commercial scale are described below.

Cold Laboratory Development -- The chemistry of the concept is developed and demonstrated using laboratory-scale equipment and cold or tracer-level synthetic process materials. These facilities continue at a reduced scale to support the engineering-scale programs.

Hot Cell Studies -- The chemistry of the various operating steps is demonstrated in remotely operated laboratory-scale equipment using highly radioactive materials and preferably samples of actual spent fuel.

These facilities continue at a reduced scale to support the engineering-scale programs.

Component Development -- Engineering-scale equipment is designed and demonstrated using cold or tracer-level synthetic process materials. Successful equipment models from this step may be modified for use in the other engineering-scale operations described below.

Cold Prototype -- Pilot or commercial-sized equipment that can be remotely operated and maintained is designed and demonstrated using cold or tracer level processing solutions. For the pilot scale, this becomes an integrated operation (cold) of the entire flow sheet. For the commercial-sized equipment, only partial mockup of the process line is required.

Hot Pilot Plant (150 Metric Tons of Heavy Metal per Year Throughput) -- A fully integrated, remotely operated (and maintained) reprocessing facility is constructed and operated initially with irradiated fuel from demonstration reactors and then using commercial nuclear fuels. This facility will demonstrate not only normal operations but also the operations that would be required to recover from malfunction of process equipment and from operational error. The scale of this equipment is less than that which would be economic for commercial operations but large enough to completely demonstrate any anticipated engineering problems. Extended operation of several years is normal, and, during this period, the spent fuel from a few reactors can be accommodated. A characteristic feature of the hot pilot plant is its designed flexibility. Equipment can be quickly and readily replaced and different equipment and processes demonstrated. The hot pilot plant can support about 5 GWe of nuclear capacity.

Lead Plant (1500 Metric Tons of Heavy Metal per Year Throughput) -- A commercial fuel reprocessing plant would be built and operated to process fully irradiated commercial fuel. This plant would be industrially financed except for licensing support, considered a demonstration cost which might be Government supported. This plant would demonstrate that a commercial-scale plant can operate in a safe, environmentally acceptable manner and with full accountability and safeguards of nuclear fuel. This plant may be more conservatively designed than will be required for following plants in anticipation of unforeseen difficulties. Continuing operation of the plant will

demonstrate the economic feasibility of commercial reprocessing of nuclear fuels. The lead plant can support about 40 to 60 GWe of nuclear capacity.

Mature Plant (1500 Metric Tons of Heavy Metal per Year Throughput) -- The commercial plants will be redesigned and constructed on about the same scale as the lead plant and differ only to exploit the experience and data acquired by the operation of the lead plant. The capacity of the commercial-sized plant is partially based on the size of the typical reactor spent-fuel discharge, minimization of spent fuel through product inventory, and nuclear material safeguards requirements.

RD&D Cost and Schedules

While fuel recycle RD&D needs can be identified for a variety of alternate fuel cycles and systems, the launching of a major developmental effort to integrate these activities into a specific integrated fuel cycle must await a U.S. decision on the fuel cycle and reactor development strategy that would best support our nonproliferation objectives and our energy needs. To what extent it would be expeditious to develop individual cycles for an integrated recycle development program will depend on the cycles that are developed and relative timing. If a number of related cycles were developed in the same facilities, the total costs would be only moderately higher than the costs associated with any one cycle.

The existence of major uncertainties in the fuel recycle development and demonstration programs make cost projections highly uncertain. There are, first, difficulties inherent in projecting the costs of process and equipment development programs that address the resolution of technical problems associated with particular reactors and fuel cycles. In addition, there are uncertainties common to projecting costs and schedules for all fuel recycle development programs; specifically, uncertainties in the future size of the commercial nuclear industry cause problems in program definition. Similarly, growth rates for particular reactor types may be much smaller than others, or the fuel loads may be smaller because of higher fuel burnup. Thus, smaller fuel-cycle plants would be required.

Estimated cost ranges for the development and commercialization of a new reprocessing technology are presented in Table 70. From this table, it can be seen that the total cost to develop a new reprocessing technology would range from about \$250 million for the LWR uranium-plutonium cycle to about \$2 billion for the more complex cycles. The major costs associated with commercialization of reprocessing lie at the far end of the RD&D progression,

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Table 70. Estimated Cost Range (\$ Million) for Development and Commercialization of a New Reprocessing Technology^{a, b}

	LWR (U/Pu cycle)	Water reactor ^c (Th cycle)	HTGR (Th cycle)	FBR (U/Pu cycle)	FBR (U/Pu core, Th blanket)
Base technology RD&D	100-150	300-450	300-500	350-500	450-600
Hot Pilot Plant	None	600-900	700-1000	600-900	700-1000
Licensing support for lead plant	150-250	200-300	200-350	250-300	300-400
Total cost of RD&D	250-400	1100-1650	1200-1850	1200-1700	1450-2000
Cost of lead plant ^d	900-1600	1500-2100	1600-2300	1500-2100	2200-3000
Lead plant capacity					
Metric tons of	1500	1500	400-600	1500	1500
heavy metal per					
year					
GWe supported	55	25-60	40-60	40-50	40-50

^aEstimated lapsed time from initial development through demonstration of lead plant ranges from 12 years for established technology to 30 years for new technology.

^bJanuary 1978 \$.

^cIncludes LWR, HWR, and LWBR.

^dLead plant is a commercial facility. It is not included in the RD&D program, and costs are assumed to be recoverable in fees. The tabulated cost of the plant is direct investment and does not include licensing support.

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namely, in the steps involving hot pilot plants, large-scale prototype equipment development, and the lead plant. Base technology RD&D to identify process and equipment concepts may require 2 to 6 years. The engineering phase of the development program, including hot testing, may require 10 to 15 years.

Lead plant design and construction might require 8 to 12 years. There can be considerable overlapping of phases so that, for a given fuel cycle, the total lapsed time from initial development to commercialization of fuel recycle ranges from about 15 to 30 years. The total time would depend on the initial technology status, the degree to which the RD&D program steps are telescoped to save time, and the stage to which the development program must be carried. The thorium cycles would be at the far end of the development time range.

The cost of reprocessing RD&D for fuel cycles that incorporate countermeasures such as coprocessing or spiking is essentially the same as for the generic fuel cycle. These fuel-cycle modifications have little or no impact on the reprocessing plant operations.

Development Plan for Recycle of LMFBR Fuel

An important NASAP scenario (Figure 44) is based on the commercial introduction of the uranium/plutonium-fueled LMFBR when constraints develop to the supply of U_3O_8 for LWRs on the once-through cycle. Under conditions that combine high power growth with U_3O_8 supply on the low side of the expected supply capability, one might anticipate a lead LMFBR plant to start operating in 2001 and the first commercial plant to start operating in 2009. Fuel-cycle support would be required also for a demonstration breeder to start up in 1990.

The preferred source of plutonium fuel for the breeders would be by reprocessing spent fuel from LWRs. Start-up of the Barnwell plant in about 1987 and operation for a year could supply plutonium for the demonstration plant. Alternatively, plutonium could be obtained by reprocessing LWR fuel in a foreign PUREX plant. There are no small-scale facilities in existence in the United States that could recover plutonium from power reactor fuels, and extensive additions would be required at Savannah River to permit such reprocessing there.

A plutonium fabrication hot pilot plant committed about 1981 would support fuel fabrication for the demonstration LMFBR and for the subsequent lead

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Figure 44. LMFBR Deployment Scenario.

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plant. A lead commercial fabrication plant would probably be committed about the time the first commercial LMFBR was committed.

The reprocessing hot pilot plant to support the demonstration plant would be committed in 1982 and start up in 1991. The same facility, which could support up to about 5 GWe of breeders, would suffice for all three developmental LMFBRs. Fuel loadings for the lead plant (2001) and the first commercial plant (2009) would be supplied from the same sources represented as options for the demonstration plant. The first commercial-scale reprocessing plant operation in the United States might credibly be postponed until 1997, at which time the commercial breeder would be ordered. At that time, a decision must be made on whether to reprocess the LWR fuel for plutonium supply and the breeder fuel for recycle in the same plant or provide different facilities for each. Such decisions would depend on certain technical judgments and such matters as rates of introduction of the breeders, decisions on whether to allow recycle of plutonium in LWRs and regulatory requirements that are not yet established. A commercial breeder would not be ordered unless there was a clear national commitment to allow reprocessing.

The above scenario, illustrated in Figure 44 represents the earliest commitment to reprocessing leading to an LMFBR deployment. If the need for the breeder were deferred to a later date, the timing of fuel-cycle support could also be deferred. Keeping the option for a commercial breeder in 2009 open requires an active plutonium fabrication and spent-fuel reprocessing R&D program in the 1980s.

Development Plan for Recycle of LWBR Fuel

A second important NASAP option (Figure 45) is based on the commercial introduction of the U-233-Th fueled LWBR when constraints develop to the supply of U_3O_8 for LWRs on the once-through cycle. A first commercial plant might start up as early as 2007. Fuel-cycle support would be required for a demonstration/lead plant to start up in about the year 2000.

The LWBR (which could be of either the HEU or denatured U-233 type and which probably would be introduced as a high-gain converter) requires preproduction of U-233 in prebreeder. The most practical scenarios for the prebreeder are the use of LWRs on the U-235/thorium and/or the plutonium/thorium cycle as U-233 producers (see Volumes III and V of the NASAP report) or the start up of the LWBR with a high/gain U-235/thorium or plutonium/thorium cycle (as suggested in Volume IV). The plutonium for plutonium/thorium loadings for

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Figure 45. LWBR Deployment Scenario.

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LWRs would be extracted from spent LEU fuel, as in the LMFBR scenario. The Barnwell plant could supply fuel for plutonium/thorium prebreeders starting about 1988.

The pilot plant would be required to process spent thorium fuel from the prebreeders as well as spent LWBR fuel. The hot pilot plant could be started on spent thorium fuel from LWRs in 1995 to extract U-233 for the LWBR demonstration. Comparatively little work has been done to develop processes for water-reactor/thorium fuels. Flow sheets have been devised, but many problems are recognized, and an intensive development program is required. After the demonstration/lead plant starts operation, the pilot plant would start processing of LWBR fuel about 2003. The hot pilot plant would similarly support the first commercial LWBR plant.

A prebreeder U-235/thorium and/or plutonium/thorium fuel-fabrication plant would be required to start up about 1990 to supply fuel for several prebreeders. An intensive development program, including irradiation testing, is required to meet the schedule. This plant might be a large pilot plant or a smaller production plant similar to an LWR uranium-plutonium mixed-oxide plant. There would be a need to start up a hot U-233 fabrication pilot plant about 1997 to fabricate fuel for the LWBR demonstration/lead plant.

Commercial LWBR plants would probably not be committed until after commercial thorium cycle reprocessing services were planned and were seen to be coming available from a licensed fuel-cycle plant under construction. Such a commitment might be expected about the year 2000 for a reprocessing lead plant to start up about 2005 to 2010.

The above appears to be a rather ambitious program especially in the thorium fuel cycle. Supplementary U-233 production could be gained from internal ThO_2 blanket rods associated with LEU fuel bundles, for example, in corner rods of BWR bundles (see Section 1.1.2, Once-Through Cycles). Some U-235 could be substituted for U-233, if necessary, in the early LWBR loadings. However, both of these approaches introduce new licensing problems for which solutions must be found through additional R&D.

2.4.6 Plant Costs in a Mature Industry

The estimated investment cost ranges for mature fuel reprocessing plants with capacities of 1500 metric tons of heavy metal per year are shown in Table 71. Both direct and capitalized investment costs are presented. The capitalized

Table 71. Estimated Cost Ranges for Mature Industry Fuel Reprocessing^{a, b}

	LWR (U/Pu cycle)	Water reactor (Th cycle)	HTGR (Th cycle)	FBR (U/Pu cycle)	FBR (U/Pu core Th blanket)
Direct investment, \$ million	800-1400	1200-1700	1300-1900	1200-1700	1800-2400
Capitalized investment, \$ million	1200-2100	1700-2400	1800-2700	1700-2400	1700-3400
Operation and materials cost, millions of dollars per year	40-60	50-70	50-70	50-70	60-90
Unit cost, dollars per kilogram of heavy metal	230-400	280-420	330-450	280-420	430-560

^a1500 metric tons of heavy metal per year plant size, except for the HTGR plant which has a capacity of 400 to 600 metric tons of heavy metal per year.

^bJanuary 1978 \$.

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costs include both interest during construction (IDC) and owners cost during construction. These latter costs amount to 40 to 45% of the direct costs. The cost ranges include the effect of plant design uncertainties, as well as the effect of fuel differences.

The lower number in the cost range assumes a plant design philosophy similar to the AGNS plant at Barnwell, South Carolina, with semi-remote maintenance. The costs at the upper end of the range assume a fully remote "Canyon" type facility, which might be required in anticipation of more stringent safeguards and operating criteria than required by current regulations.

As the fuels industry grows, plant sizes may become larger than the 1500-metric tons heavy metal per year reference size for LWR fuels. Also, in some cases, such as the HTGR, the reprocessing plant size may never reach 1500-metric tons heavy metal per year. It is estimated that doubling the plant size may be done with a 50 to 60% increase in capital investment. The capital investment requirement for plant size of 750 metric tons of heavy metal per year (one-half the reference size) would be about 70 to 80% of the investment for the reference-sized plant.

The ranges of estimated typical industry unit costs in dollars per kilogram of heavy metal charged to the reactor are also shown in Table 71. These unit cost ranges are based on the investment and operating costs also given in the table and an annual fixed charge rate of 22.6% on capitalized investment. Reprocessing costs are very capital-intensive. The operating and materials costs make up only 9 to 16% of the total unit cost.

2.5 WASTE HANDLING AND DISPOSAL

2.5.1 Introduction

Many types of waste are generated in the nuclear fuel cycle. Factors that determine the character of the wastes are the types and operating characteristics of the reactors, the processes used in manufacturing and reprocessing the fuels, and the waste management or treatment practices adopted at the plants or mandated by the government. Wastes are also generated in the decommissioning of nuclear facilities. Since many nuclear waste management standards have not yet been firmly established, any discussion of waste management is tentative.

The predominant form of waste from the once-through fuel cycle is the spent fuel discharged from the reactor. The spent fuel is generally stored underwater at the reactor site for some time. Following a suitable period of cooling, the spent fuel may be moved to an away-from-reactor (AFR) storage site. Ultimately, if fuel reprocessing were not authorized, the spent fuel would be suitably packaged and placed in a geologic repository. Storage of spent fuel in a repository can be conducted in a retrievable mode in anticipation of a later decision to allow reprocessing and to confirm the safe performance of the geologic medium.

While much R&D work has been done on the closed fuel cycle, there remains a shortage of commercial plant operating experience upon which estimates of future performance can be reliably based. This is especially true of those operations that will generate the most radioactive wastes: mixed-oxide fuel fabrication and large-scale commercial reprocessing. A principal source of radioactive waste in the closed nuclear fuel cycle is the reprocessing plant where the integrity of the fuel cladding is breached. Fission gas is released, collected, and packaged; fission products are concentrated into high-level wastes, and irradiated fuel cladding and hardware are disassembled and collected. Reclamation of mixed-oxide fuel fabrication scrap also contributes to the production of transuranium (TRU) wastes.

This section provides descriptions of "typical" wastes by drawing on available operating experience at both DOE and commercial facilities and from the results of the ongoing Alternate Fuel Cycle Evaluation Program. On the basis of their packaging, shipping, and probable disposal requirements, eight types of radioactive wastes generated by the closed fuel-cycle operations are defined: high-level wastes, fuel cladding and hardware, noble gases, iodine, C-14, tritium, alpha or TRU wastes, and low-level beta-gamma wastes (material

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containing $<10 \text{ nCi } \alpha/\text{g}$). Table 72 shows where these wastes occur in a generically closed nuclear fuel cycle. The wastes are collected at the site where they occur, treated to convert them to a stable form, packaged, and transported (as discussed in Section 2.6) to disposal sites.

Table 73 is a summary of the types of disposal sites envisioned for each category of waste.

Table 74 provides some perspective on the distribution of volume, radioactivity, and thermal output among the various categories of waste in the closed fuel cycle. It is evident that the major part of the activity and thermal heating is in the high-level waste, while the low-level wastes are extremely voluminous. The volume of spent fuel generated in the once-through fuel cycle is approximately equal to the sum of the high-level waste and cladding hulls in the closed fuel cycle, but there is no equivalent to the bulky TRU wastes generated in the fabrication and reprocessing plants.

All waste management systems rely on a multibarrier approach to achieve the desired level of isolation. The first engineered barrier is the chemical form of the waste itself, which should be stable and resist attack from the storage environment. The waste is generally placed into a container, which forms a secondary engineered barrier. Finally, hazardous wastes are to be stored in geologic repositories that are selected for their highly stable character, thus forming a natural barrier that isolates the waste from a nuclide transport medium path to the biosphere. Section 2.5 discusses the handling of the various wastes, their disposal, and the RD&D requirements of the nuclear waste program.

2.5.2 Waste Handling

Spent Fuel

At-reactor Storage -- Spent fuel discharged from a reactor is normally stored at an onsite pool in racks that space the fuel or are poisoned in a manner precluding accidental criticality. The pool provides shielding and cooling. Reactor sites were originally designed to have space for about four to five annual fuel discharges; this storage capacity has been increased at most reactors so that spent fuel can now be stored onsite for about 5 to 10 years. Prior to the deferral of spent-fuel reprocessing, spent fuel was assumed to be stored at the reactor for about one year.

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Table 72. Estimate of Where Radioactive Wastes Occur
in a Generically Closed Nuclear Fuel Cycle

	Fabrication (%)	Reactor (%)	AFR Facility (%)	Reprocessing (%)
High-level waste	0	0	0	100
Fuel cladding and hardware	0	0	0	100
Noble gases	0	1	1	98
Iodine	0	1	1	98
C-14	0	10	0	90
Tritium				
LWR	0	20	2	78
LMFBR	0	95	<2	3
CANDU	0	97	1	2
TRU wastes	75	1	<1	24
Low-level beta-gamma wastes	10	80	<1	10

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Table 73. Proposed Waste Disposal Options

Waste category	Waste disposal options
Spent-fuel elements in once-through cycle	Temporary storage at reactor site, optional transport to AFR storage, and final disposition in a geologic repository (salt, basalt, etc.)
High-level wastes	Geologic repository
Fuel cladding and hardware	Geologic repository
Noble gases	Compressed in cylinders or sorbed on zeolite and stored under administrative control -- final disposition not yet determined
Iodine	Final disposition not yet determined
C-14	Final disposition not yet determined but possibly as low-level waste
Tritium	Storage at designated sites in solid or liquid form
Alpha (TRU) wastes	Geologic repository
Low-level beta-gamma wastes	Storage at designated sites

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Table 74. Fuel-Cycle Wastes from the Production of 1 GW-Year of Electricity^a

Category of waste	Volume (m ³)	Activity ^b (MCi)	Thermal power ^b (kW)	Plutonium residue (kg)	Number of shipments
Transuranium wastes					
High-level solidified	3.1	170	960	1.6	2
Cladding hulls	2.7 ^c	2.0	10.1	1.3	2
Intermediate-level solid	140 ^d	0.012	0.057	0.5	66
Low-level solid	480 ^d	0.049	0.04		17
Non-TRU wastes					
Noble gases	0.1	0.25	0.37		0.4
Iodine	0.05	1×10^{-6}			0.2
C-14		2×10^{-5}			
LWR tritium (water)	140	6×10^{-4}			9
Fission-product tritium (solidified)	0.35	0.018	7×10^{-4}		1
Low-level solid	2400 ^d	0.002	0.007		180
Ore tailings	42,000	5×10^{-4}	0.011		

^aMixed (uranium-plutonium) oxide-fueled LWRs.

^bActivity and thermal power at time of waste generation -- current processing basis.

^cCompacted to 70% of theoretical density.

^dUncompacted volume.

Some consideration is being given to the expansion of at-reactor storage pool capability by some utilities as an alternative to the use of an AFR storage facility.

The storage of spent fuel in water-filled basins has been practiced for about 20 years, when the first power reactor began discharging fuel. Spent fuel has also been stored in the water-filled basin at the Nuclear Fuel Services Reprocessing Plant at West Valley, New York, and at the Midwest Fuel Reprocessing Plant at Morris, Illinois. Consequently, this method of storing spent fuel is based on fully developed technology and does not require additional RD&D.

Away-from Reactor (AFR) Storage Facility -- Typically this would be a large pool containing storage racks capable of handling 5,000 to 15,000 metric tons of heavy metal in the form of spent-fuel assemblies. The AFR facility would accept fuel cooled for 5 years or more following discharge from a reactor and would include facilities for shipping and receiving fuel by both truck and rail, for encapsulating failed fuel elements, for water treatment, for cooling towers, and for waste monitoring and treatment.

Spent Fuel Encapsulation -- Spent-fuel elements will be encapsulated before they are placed in the geologic repository. The encapsulation facility would probably be located at the site of the geologic repository. Spent fuel cooled 10 or more years after discharge from the reactor, would be received, and assemblies prepared and placed in canisters, welded closed, tested for leaks, and placed in the repository either in the temporary (retrievable) or permanent mode. This facility would also have the ability to overpack failed fuel canisters with a second canister, and to weld, test, and transport these overpack canisters in a manner similar to standard canisters. The estimated annual capacity to encapsulate fuel would be commensurate with the geologic repository receiving rate.

The fuel characteristics are assumed to allow two BWR assemblies per canister, but only one PWR assembly per canister. Current experience indicates that leaking fuel is not a problem to extended water-pool storage. A program is under way at the Nevada Test Site to encapsulate several spent LWR fuel elements and to conduct in situ tests for thermal and radioactivity interactions with several geologic formations. In situ tests were previously conducted in a salt formation using encapsulated test reactor fuel elements.

High-Level Wastes

High-level wastes are defined in federal regulations as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels." These wastes contain virtually all of the nonvolatile fission products, several tenths of one percent of the uranium and plutonium originally in the spent fuels, and all the other actinides formed by transmutation of the uranium and plutonium in the reactors. They can be generally characterized by their very intense, penetrating radiation and their high heat generation rates. The wastes are temporarily stored as acid solutions in stainless steel tanks (or they are neutralized and stored in carbon steel tanks). They may have volumetric heat generation rates as high as 10 W/liter (120 Btu/hr/ gal).

Storage of high-level liquid wastes from the weapons production program has been practiced for 35 years, and almost 340,000 m³ (90 million gal) of acid and alkaline solutions and sludges have been accumulated in about 200 tanks at DOE reprocessing sites during this period. Although more than 20 carbon steel tanks storing neutralized defense wastes are known to have developed leaks, the released radionuclides have been retained in the soil within the immediate vicinity of the vessels, and no radiation exposure to personnel or to uncontrolled areas has resulted. The experience gained with such storage has led to the design of new liquid storage systems that justify greater confidence in containment integrity. The substantial advantages to be gained from reduced radiation levels and heat evolution rates during subsequent waste processing will probably make some interim liquid storage desirable. The complexity and cost of acceptable safe systems will tend to limit their capacity to no more than a few years of plant generation.

Regulations call for high-level liquid wastes from commercial operations to be solidified within 5 years after they are generated and for the resultant stable solids to be shipped to a federal repository within 10 years after the liquids are generated. Methods for solidification of high-level wastes from reprocessing plants range from calcination to incorporation in glass. The actual volumes of waste will depend on many factors that cannot be determined without the benefit of actual operating experience. Although waste management economics will undoubtedly tend to force the final volumes toward smaller values, additional wastes not considered here, arising from mal-operation, equipment decontamination, etc., will tend to increase them. Similarly, possible future requirements for higher-quality waste products (less leachable in water, more radiation resistant) may also result in higher volumes of waste per metric ton of fuel processed.

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In a typical vitrification process, high-level, liquid waste is taken from interim storage in tanks to an atomizing nozzle at the top of a spray dryer. The spray-dried product is calcined, mixed with glass frit, melted, and poured into canisters. The canisters of vitrified waste are welded closed and stored under water. Water storage of the waste is required because of the high heat generation (15 to 20 W/liter). Shipment entails removing the canisters from water storage and placing them in overpacks on an as-needed basis, and loading into rail transport casks for transport to a federal repository. The development of processes for vitrification of high-level wastes is under way in several European countries and the United States. France is operating a hot pilot plant with a capacity of 20 kg/hr at Marcoule. Pacific Northwest Laboratory (PNL) operates engineering-scale cold and hot pilot plants. The scaleup of such facilities to hot operation at high capacity and with demonstrated remote maintenance technology constitutes a major RD&D need.

Fuel Cladding and Hardware

Fuel cladding wastes are the residual Zircaloy, Inconel, and stainless steel hulls and structural components of the fuel assemblies that remain after the fuel core has been dissolved. Although the radioactivity arises mainly from neutron-induced isotopes, the hulls are similar in some respects to high-level wastes in that they may contain up to several hundredths of a percent of the plutonium originally in the spent fuel and have heat generation rates of about 2 to 4 W/liter (50 to 100 W/ft³).

One method for packaging fuel hulls and hardware is fixing in Portland cement inside 55-gal steel drums to provide a package suitable for shipment to permanent storage. The process involves the direct addition of the existing metallic forms to Portland cement grout inside standard 55-gal steel drums, a technique used commercially to immobilize radioactive waste from power reactors. Hulls intentionally are not compacted to ensure that the centerline temperature of the cement aggregate does not exceed 100°C, the temperature at which water or hydration tends to leave the cement.

Fuel hulls and hardware fixation in cement should be a relatively straightforward operation. However, design details must be developed and certain critical equipment demonstrated. Separation of end fittings from hulls may be required after dissolving to aid in the encapsulation process. Further study is required for the Zircaloy fines fixation step.

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Another method for packaging fuel hardware and hulls is to compact the material in a press, place the compacted material in a canister (similar in size to high-level waste canisters), weld the canister shut, and ship it to a waste repository for disposal.

End fittings with only low levels of induced radiation may be disposed of by shallow land burial.

Noble Gases

The noble-gas fission products consist principally of stable and short-lived isotopes of krypton and xenon. Typically, from 5 to 6 kg is present in each metric ton of spent fuel from LWRs and LMFBRs, and, while xenon comprises about 95% of the weight of the mixture, the only radioisotope remaining after 150 days decay is 10.8-y half-life Kr-85. These gases are currently discharged to the atmosphere following interim holdup for decay. Over 99% of them would be released at the reprocessing plant, and the remainder are discharged at power stations as a consequence of their leakage through defective fuel cladding. Although off-site radiation exposures have been small compared with current guidelines for population exposure, the noble gases will probably be recovered in the future as a result of stated policies of maintaining radioactive releases to the environment at the lowest practicable levels.

Several processes are available or are under development for the recovery of the noble gases from plant off-gas streams. Initially, it is likely that the mixed gases will be collected in gas cylinders, shipped to a remote site, and stored under conditions that will promote long-term integrity of the cylinders. A typical off-gas treatment flowsheet is shown in Figure 46. Proven unit operations are available for all the off-gas steps; however, operation of large-scale systems at full radiation levels must still be demonstrated.

The quantity of noble gases to be collected is roughly the same for all fuel cycles and depends primarily on the fuel burnup.

Iodine

Iodine is a semivolatile fission product which, because of its complex physical and chemical properties and its high biological significance, has always required special attention to ensure adequate safety management. About 0.01 g of iodine isotopes is formed per MWd(t), and the isotope of greatest

Figure 46. Off-Gas Treatment Flowsheet.

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concern during reactor operation and fuel reprocessing is 8.05-d half-life I-131. After one year's decay, the isotope of concern for waste management is 1.6×10^7 y half-life I-129. The quantity of iodine formed is roughly independent of the fuel cycle.

Research and development work aimed at reducing iodine releases from fuel reprocessing plants to "near zero" levels shows promise of at least 99.9% removal from the other fuel constituents by volatilization at the head end of the process. Once separated, it would subsequently be trapped and refined separately from the other waste streams. Several dry and wet processes for the removal of iodine from the various off-gas streams of a nuclear fuel reprocessing plant have been used or investigated. Dry processes use beds containing solids such as charcoal (at ambient or cryogenic temperatures) and silver-zeolite. Wet processes use scrubbing columns filled with solutions such as 10 M NaOH, $\text{HNO}_3 + \text{Hg}(\text{NO}_3)_2$, or 20 M HNO_3 (the Iodex process). The final form into which the iodine may be processed for packaging, shipment, and disposal has not been defined and requires additional RD&D. Such concentrates of I-129 must be stored permanently because of their very long half-life.

Carbon 14

Carbon 14 (5730-y half-life) is produced in oxide-fueled reactors principally by an (n,p) reaction with N-14 impurity in the fuels but also as the product of an (n, α) reaction with oxygen. In the HTGR, the principal source is from (n, γ) reaction with the C-12, C-13 of the graphite fuel, and reflector blocks. It is estimated that 60 to 70 Ci of C-14 is generated per GWe in LWRs, from 10 to 20 Ci/GWe-year in LMFBRs, and about 200 Ci/GWe-year in HTGRs. The assumption most often made is that >90% of the C-14 will be released at reprocessing plants as C-14-contaminated CO_2 in oxide fuel-dissolver off-gas or in the HTGR graphite-fuel block burner off-gas. In this event, the C-14 can be removed from the off-gas by sorption on molecular sieves or by scrubbing with $\text{Ca}(\text{OH})_2$ or $\text{Ba}(\text{OH})_2$ slurries.

However, removal of C-14 from HTGR fuel reprocessing off-gas poses a somewhat greater problem than it does for LWR or FBR fuel reprocessing off-gas, because, in the former case, it is diluted with a large quantity of nonradioactive CO_2 derived from burning the graphite fuel blocks. As a consequence, about 1400 55-gal drums of C-14-contaminated CaCO_3 in concrete per metric ton of heavy metal reprocessed (14,000 55-gal drums per GWe-year) must be shipped to a suitable isolation site. This compares with <1 55-gal drum of final C-14-contaminated waste per metric ton of LWR or LMFBR fuel and represents a substantial waste operation.

Tritium

About 20% of tritium (12.3-y half-life) wastes are generated at LWR nuclear power stations and the remainder at fuel reprocessing plants. Tritium in LWR wastes arises principally from neutron reactions with light elements such as lithium and boron that may be present in the primary coolants. New power stations may have the capability of recycling the coolant until the tritium concentration reaches ~ 5 $\mu\text{Ci/ml}$ and then maintaining that concentration by adding fresh water to the coolant as required. Operating experience with this technique is lacking; however, a reasonable estimate is that, on the average, about 20,000 gal of tritiated water per year will be stored in tanks at each power reactor until it must be shipped to disposal sites. Additional work on the disposal of this waste may be required.

About 700 to 1200 Ci per metric ton of fuel is produced in fission and appears in wastes from fuel reprocessing. A few percent may be associated with the fuel cladding, but most is present with the fuel materials and is currently released as water vapor to the atmosphere. Future plants having head-end operations like "voloxidation" should be able to separate and recover the tritium in a relatively small volume. This concentrated solution could then be converted into an appropriately stable, solid form [perhaps as $\text{Ca}(\text{OH})_2$ or sorbed on zeolite], packaged, and shipped to a designated storage or disposal site.

Tritium behaves differently in LMFBRs than in LWRs. As much as 95% may diffuse through the stainless steel cladding during reactor operation and appear as a tritiated sodium sludge in the primary-coolant cold traps. Processes for recovery and packaging of the tritium from LMFBR fuels may not be needed at the reprocessing plant, and special precautions to dispose of the sodium sludge from the reactor may be required.

Heavy-water-moderated reactors generate tritium by neutron activation of the deuterium to an extent far in excess of that produced in the fuel. For example, it is estimated that for a 1000-MWe plant, the annual production of tritium in the fuel could be about 15,000 Ci, while that in the heavy-water moderator could be 600,000 Ci. The necessity to minimize losses of heavy water because of its expense, as well as because of the hazard from release of the tritium, requires carefully engineered containment systems and elaborate provisions to ensure occupational safety. No provisions have been made for disposal of "waste" D_2O containing large amounts of tritium. This may require a RD&D program to develop safe handling and shipping procedures.

Transuranic Wastes

The so-called transuranic (TRU) or alpha wastes are defined as those solid materials that contain plutonium or other long-lived alpha emitters in concentrations greater than 10 nCi/g. The criterion of 10 nCi/g is adopted as a lower activity density for TRU wastes, because this corresponds to the upper range of alpha-emitting isotopes in naturally occurring deposits. They may have low enough external radiation levels that they can be handled directly without supplementary shielding (surface dose rates of <10 mrem per hour) or they may require biological shielding and reactor handling techniques for surface dose rates >>10 mrem per hour. The 75% of the alpha or TRU wastes generated in the closed nuclear fuel cycle originating at fabrication plants belongs to the first category, while the 25% generated at the reprocessing plant is in the second category and require shielding. It is reasonable that wastes containing less alpha activity than approximately 10 nCi/g be regarded as disposable in carefully selected surface burial grounds, while alpha wastes must be stored in geologic repositories that offer assurance of permanent containment. The intent is to minimize the total volume of waste to be stored in the geologic repository.

TRU wastes consist of a wide assortment of solid materials, including items made of paper, cloth, wood, plastic, rubber, glass, ceramic, and metal, as well as salts and sludges that arise in the treatment of liquid waste streams and filters from cleanup of off-gases. The TRU-element contents of these wastes range from trace amounts to a hundred or so milligrams per liter and average about 9 mg per liter. The densities of the uncompacted wastes vary from about 2 lb/ft³ to as much as 200 lb/ft³. About one-half to two-thirds of these wastes (by volume) are combustible and can, via incineration, be reduced by factors of about 50 and about 20 in volume and weight, respectively, or they can be reduced in volume by factors of 2 to 10 through compaction. The compacted waste may be encased in cement or other fixative agents in 55-gal drums. The TRU wastes generated at reprocessing plants generally contain about 1 mg of plutonium per liter.

Low-Level Beta-Gamma Wastes

Diverse solid wastes contaminated with beta-gamma activity (containing <10 nCi/g) are common to all facilities handling radioactive materials. They may range from concentrates of the radionuclides generated in the decontamination of plant effluent streams to almost every conceivable type of contaminated solid refuse from plant operations. They normally require minimal shielding and, since they are designated to contain less than 10 nCi of long-lived alpha activity per gram, are disposable in surface burial grounds.

Beta-gamma wastes are compacted in a manner similar to the handling of alpha wastes.

Decommissioning and Decontamination

Experience in decommissioning and decontamination of nuclear facilities has been limited to uranium milling, several developmental reactors, and fuel-cycle facilities associated with development or defense programs.

Decommissioning requirements were not directly addressed as part of a federal statute until 1978. Congress passed the Uranium Mill Tailings Radiation Control Act of 1978 in November of that year. This act amended the Atomic Energy Act of 1954 to require a decommissioning plan and proof of financial ability to execute it as part of the license application for a uranium mill. As a result, abandoned mill tailings piles are now being stabilized and covered to protect the environment and eventually to restore the land area for other uses.

Firm requirements for decommissioning and decontamination (D&D) of nuclear facilities do not currently exist. The NRC staff is conducting some studies of its regulatory position with respect to D&D of facilities. NRC studies are expected to cover the following: pressurized-water reactors, boiling-water reactors, multiple reactor facilities, uranium mills, uranium fabrication, by-product utilization, fuel reprocessing, mixed oxide fabrication, and low-level waste burial grounds.

Three alternatives for decommissioning nuclear reactors were previously identified in Regulatory Guide 1.86. These alternatives are protective storage, entombment, and dismantling.

Protective storage consists of preparing the facility to be left in place safely for an extended period. Radiation monitoring, environmental surveillance, and security procedures are designed to ensure the health and safety of the public. This mode may be adopted as a permanent solution for facilities not contaminated with long-lived nuclides.

Entombment involves sealing highly radioactive or contaminated components within a high integrity shielding structure. All fuel assemblies, process materials, and other radioactive wastes are removed prior to this entombment. The materials removed would be prepared for ultimate disposal as either

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high-or low-level waste. Unrestricted access to all uncontrolled parts of the facility is possible after completion of the entombment.

Dismantling involves the removal of all radioactive components and materials that exceed the criteria for unrestricted access. This process may be undertaken shortly after cessation of power production operations.

The volume of radioactive wastes from decommissioning and decontamination is large enough so that there is an economic incentive for reduction of waste volume. This may be accomplished in a variety of ways: evaporation, calcination, incineration, mechanical compaction, or cutting.

The estimated cost for immediate dismantling of a commercial size PWR has been reported to be approximately \$40 million. This figure represents roughly 5% of the estimated capital cost of \$800/kWe for a 1000-MWe LWR. The creation of a sinking fund which accumulates 5% of the initial plant capital investment at the end of plant life has been suggested as a means of funding decommissioning. The decommissioning sinking fund allowance can then be included in economic evaluations as an additional component of the annual fixed charge rate on capital investment.

The alternatives for decommissioning other fuel-cycle facilities are the same as those outlined above for nuclear reactors. Decommissioning costs for highly radioactive fuel-cycle facilities, such as reprocessing plants or plutonium fuel fabrication plants, are expected to be comparable to those of nuclear reactors.

2.5.3 Waste Disposal

Site-Selection Criteria and Options

Six candidate technologies have been examined for the ultimate disposal of high-level and TRU wastes by the President's Interagency Review Group (IRG):

- placement in mined repositories;
- placement in deep ocean sediments;

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- placement in very deep drill holes;
- placement in a mined cavity that leads to rock melting;
- partitioning of reprocessing waste, transmutation of heavy radio-nuclides, and geologic disposal of resulting fission products;
- ejection into space.

Current evaluations indicate that of the options, mined repository (i.e., in geologic formations) disposal will be available soonest. It is conceivable that deep-ocean sediment, deep drill hole disposal, transmutation, and space disposal may be available at a more distant time. The major RD&D effort is directed toward placement in mined repositories.

The geologic formation housing the repository may have to satisfy certain physical requirements. Some of the possible requirements are listed below.

- The depth of the formation is great enough to provide a barrier to physical transport of the wastes to the biosphere.
- The physical and chemical properties of the rock provide heat dissipation and do not degrade the waste.
- There is essentially no likelihood of earthquake damage to the repository.
- The disposal medium minimizes groundwater interactions that could potentially transport waste constituents.
- The formation does not contain mineral resources that may attract future human mining activity.
- Surrounding media are stable and do not interact with the repository medium.

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The geologic media considered most promising for disposal sites are salt, basalt, granite, and shale. A preliminary evaluation of the relative properties of these media has been reported by DOE. These properties are not of equal importance, and they have not been weighted for relative importance. Typical media properties are rated on a relative scale of 3 to 1, with a rating of 3 indicating greater potential for isolation than a rating of 1.

Property	Salt	Granite	Shale	Basalt	Desired characteristic
Natural moisture content	3	3	2	3	Low moisture
Strength	1	3	2	3	High strength
Coefficient of linear thermal expansion	2	3	3	3	Low expansion
Heat capacity	3	2	2	2	High heat capacity
Thermal conductivity	3	2	2	1	High conductivity
Permeability	3	3	3	3	Low permeability
Porosity	3	3	1	3	Low porosity
Solubility	1	3	3	3	Low solubility

The underground area required for the disposal of wastes depends on the disposal medium and the fuel cycle, as illustrated in Table 75.

Table 75 indicates that, in granite or basalt, LWR wastes from the once-through cycle can be stored somewhat more compactly than high-level wastes from recycle. However, in salt the high-level waste can be stored more compactly. Since each repository is expected to occupy about 2000 acres, from two to four sites would be needed for the first 10,000 GWe-year of power production. Variations in repository area requirements result from differences in thermal loading and mechanical characteristics of the medium and quantities and compositions of the emplaced wastes.

Table 75. Acreage and Thermal Loadings Estimated for Conceptual Repositories^a (Preliminary LWR Data)

	Salt		Granite		Shale		Basalt	
	kW/acre ^a	GWe-yr/acre ^b	kW/acre	GWe-yr/acre	kW/acre	GWe-yr/acre	kW/acre	GWe-yr/acre
Once-through cycle ^c	40	1.28	100	3.20	58	1.48	100	3.20
Self-generated recycle ^d	76	2.04	95	2.55	50	1.61	95	2.55

^aFar-field average thermal load limits.

^bAssumes waste is ten years out of reactor at time of emplacement and that only spent-fuel assemblies and HLW are buried.

^cValues based on 2:1 ratio of PWR to BWR fuels.

^dOnly values for solidified HLW listed.

Site-Selection Process

Site selection is partly a technical process and partly a political process. The technical process consists of the following three stages.

1. Broad considerations of tectonic stability and hydrology are used to select regions on a national scale.
2. Candidate areas on the order of 10,000 km² are identified on the basis of faulting, seismicity, hydrology, stratigraphy, and geologic structure. Local geologic data are developed at this time.
3. Candidate areas about 20 km² are selected and ranked in order of desirability. Detailed information about sites will be obtained by exploration methods such as drilling and other geophysical techniques. This exploration must be carefully planned and executed to prevent compromising the repository's integrity. Rock properties, stress states, and hydrologic gradients should be measured in situ. Sociopolitical factors, including proximity to potentially valuable resources, should be evaluated early in Stage III prior to beginning intensive site exploration. Some sites will be eliminated because of such considerations.

The practical aspects of gaining access to land for reconnaissance and exploration, at least over the near term, may impose severe restrictions on the area considered. Although satisfaction of appropriate technical criteria and siting considerations is essential at each stage, other factors, such as ease and cost of access, distance from other societal activities, and societal acceptance of the location as a candidate repository site, etc., are relevant to the site selection process and could dominate. Large areas of the country may be ruled out by these latter nongeologic factors, apart from their possible attractiveness from a geologic perspective, while other areas may be highly favored by these nongeologic factors. Geologically, suitable sites may have to be selected for detailed investigation based on availability or simply on ownership by appropriate government agencies.

Typical Geologic Waste Repository Description

In the simplest form, a geologic repository is not unlike a mine, and it consists of access corridors and disposal rooms excavated deep within the

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formation. Various supportive facilities are provided on the surface over the repository, and shafts are constructed to provide access from these facilities to the disposal level. A repository encompassing ~2000 acres of subterranean area has been selected as a convenient size for excavation and waste emplacement activities. The surface facilities that support the repository during its operating life represent the only visible evidence of the repository and would be confined within a fairly small area of perhaps 200 acres. Figure 47 depicts a repository concept in a salt formation.

The following sequence of events describes the operation of a typical repository. This example is based on the disposal of spent fuel. Spent fuel will be received in shipping casks, normally on railcars, but with repository capability provided for receiving occasional truck shipments. After inspection of the cask, the spent fuel will be unloaded in a hot cell in the waste handling building. Spent fuel that was not previously placed in canisters will be canistered and the canisters inspected remotely by means of viewing windows and remote manipulators. The canisters will be transferred by automatic machinery to a shielded transfer cask. Each canister will be fully shielded and confined at all times, first by the hot cell and then by the transfer cask.

The transfer cask will be loaded into the waste handling cage for transfer down the waste handling shaft to the mine, 2000 ft below the surface. At the mine level, the transfer cask will be loaded onto a manned, rubber-tired transporter for hauling to the storage room. At the storage location, the transporter will position the transfer cask over a previously drilled storage hole in the salt floor of the mined storage room and lower the fuel assembly into the hole.

The storage hole will have previously been provided with a steel sleeve to ensure that loose salt will not lodge against the canister and impede its retrieval.

After canister emplacement, the transporter will place a shielding plug in each hole above the canister to reduce radiation in the storage room to acceptable levels. Sleeves and shield plugs, however, will be provided during the retrievability period characteristic of initial operation of early repositories. After this period, canisters will be stored in bare holes and backfilled to room level with crushed salt. Rooms will also be backfilled as they become full.

Figure 47. Artist's Concept of a Geologic Repository and Its Support Facilities.

High-level waste, intermediate-level mixtures of actinides and fission products, and cladding waste from recycle operations would be handled similarly to spent fuel in the above example of repository operation.

Low-level TRU (or alpha) wastes in drums will be removed from the shipping container by forklift truck, inspected, loaded in a transfer fixture, and then transferred to the same waste handling cage used for fuel assemblies. After being lowered to the mine level, the drums will be removed from the transfer fixture, palletized, hauled by truck to the storage area, and stacked in tiers.

2.5.4 Alternate Cycles Comparison

Wastes in the once-through cycle are characterized by a high degree of containment within the fuel element cladding of the highly hazardous alpha material and fission products. Tritium and beta-gamma low-level wastes generated at the reactor site may be retained at the reactor until disposal at a low-level waste burial site.

The AFR storage facility appears to be primarily applicable to the once-through fuel cycle. AFR storage would be unnecessary in a mature recycle economy, because spent fuel would be reprocessed after leaving the reactor site. Up to several AFR facilities would be utilized per geologic repository in a mature once-through fuel cycle. Reactor operators may choose to allow spent fuel to decay at the reactor site, thus reducing demand for AFR storage.

Table 75 (see Section 2.5.2, Noble Gases) indicates nominal repository requirements for wastes generated in a uranium-plutonium self-generated recycle mode with wastes of the once-through cycle. As the result of reprocessing and mixed-oxide fabrication, the recycle wastes are much larger in total volume. High-level waste, cladding waste, and TRU wastes are the dominant forms that must be stored in a geologic repository. They have a total volume of about 230 m³/GWe-yr, in contrast to the spent-fuel volume of the once-through cycle of about 15 to 20 m³/GWe-yr. It is possible that some reduction in this volume of TRU wastes may be effected through further RD&D.

A fully developed geologic repository in salt is expected to contain at least 50,000 metric tons of spent fuel (equivalent to about 2500 GWe-yr of wastes) or the wastes recovered from recycle of fuel from about 3000 GWe-yr of power

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production. The high-level wastes from reprocessing are expected to require somewhat less mine storage space than spent-fuel elements.

The spent fuel from the extended-burnup LWR once-through cycle is similar to that from the standard cycle except for being about one-third less per GWe-yr by volume, having higher activity and greater plutonium content.

Table 76 compares the radioactivity of wastes from alternate cycles in a repository as a function of time. High-level waste in the LWR-LEU cycle would correspond to waste generated when LWR fuel is reprocessed and the plutonium stored for later use. LWR-U/Pu cycle waste would be generated after plutonium recycle in LWRs. Data are plotted for these wastes as well as LWR-once-through, HTGR-HEU/Th, and LMFBR-U/Pu systems in Figure 48. For the first 100 years, wastes from all cycles are roughly comparable in activity, which is dominated by fission products. The activity decays a factor of 10 between year 1 and year 10, and a second factor of 10 between year 10 and year 100.

At 1000 years, the activity is dominated by heavy elements, principally Np-239, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, and Cm-245. The spent fuel from LWR once-through cycles and the high-level wastes from LWR-U/Pu recycle have the highest activity at 1000 years; the principal actinides in the former are the residual plutonium isotopes, whereas the latter has a greater content of the higher actinides formed by irradiation of plutonium in a thermal neutron spectrum. LMFBR wastes are somewhat less active, because the fast neutron spectrum promotes fission of the TRU elements. The lowest activity is noted in high-level waste from the LWR-LEU cycle (relatively little plutonium, americium, or curium) or the HEU-Th cycle (U-233-Th wastes relatively benign after 1000 years).

The radioactivity of the HEU-Th cycle waste or the high-level LEU waste falls to a level about equal to that of the original uranium ore within 1000 years, as shown in Figure 48. The activity of the ore (principally Rn-222) is about 20 Ci/short ton U₃O₈. The activity of LMFBR waste has decayed to the level of ore before 10,000 years. The activity of spent LWR fuel or of LWR-U/Pu waste requires about 100,000 or more years to reach that level.

It should be noted that reprocessing parameters may be modified, at modest cost, to reduce plutonium losses to waste and to recover other TRU elements. Thus, plutonium-uranium fuel cycles could be modified to recycle a greater fraction of the actinides, thereby reducing the 1000-year activity level by about an order of magnitude. The recycled actinides may be fissioned in a

Table 76. Radioactivity of Wastes from Alternate Cycles
Stored in a Geologic Repository
(Ci/GWe-yr)

Time (year)	Once-through cycles			
	Standard PWR		High-burnup PWR	
	Heavy elements	Fission products	Heavy elements	Fission products
10	2.8×10^6	1.0×10^7	3.1×10^6	1×10^7
100	2.3×10^5	1.1×10^6	2.5×10^5	1×10^6
1,000	5.8×10^4	7.3×10^2	5.2×10^4	7×10^2
10,000	1.5×10^4	7.1×10^1	1.2×10^4	7×10^1
30,000	5.8×10^3	6.7×10^1		
100,000			1.0×10^3	

Time (year)	Reprocessing plant wastes							
	Standard PWR-LEU		PWR-U/Pu		LMFBR-Pu/U		HTGR-HEU-Th	
	Heavy elements	Fission products	Heavy elements	Fission products	Heavy elements	Fission products	Heavy elements	Fission products
1	5.6×10^5	1.4×10^8	7.8×10^6	1.5×10^8	1.5×10^6	2.2×10^8	1.2×10^6	6.1×10^7
10	7.1×10^4	1.0×10^7	1.5×10^6	9.0×10^6	1.4×10^5	7.1×10^6	2.5×10^5	1.0×10^7
100	1.2×10^4	1.2×10^6	1.7×10^5	1.0×10^6	6.2×10^4	8.0×10^5	9.2×10^4	1.1×10^6
1,000	3.2×10^3	7.7×10^2	3.9×10^4	7.8×10^2	1.5×10^4	6.8×10^2	1.7×10^3	6.1×10^2
10,000	7.5×10^2	7.3×10^2	1.2×10^4	7.3×10^2	1.5×10^3	5.9×10^2	6.1×10^2	6.0×10^2
100,000	8.9×10^1	5.8×10^2	4.4×10^3	5.9×10^2				

^aWaste recovered from reprocessing standard LEU fuel.

^bRecycle of fissile values.

Figure 48. Radioactivity of Wastes from Alternative Cycles.

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fast-reactor spectrum. If such a strategy were adopted, TRU wastes at fuel fabrication plants would also require tighter standards of plutonium control.

Long-term waste management considerations therefore favor the fission of plutonium and other TRU elements in a fast spectrum and the use of uranium fuels in a thermal spectrum.

Reprocessing may be used to mitigate long-term waste hazards, but this would occur at the expense of increased short-term effluents and a more bulky set of wastes.

Some other fuel-cycle-related observations on wastes follow.

- The HTGR cycle creates a very large quantity of C-14 wastes, about 3000 m³ per GWe-yr. This burden is partly offset by the low activity level of these wastes.
- Large quantities of tritium are generated in the heavy water or heavy-water-moderated reactors. Since the heavy water is not discarded (at least not yet), this tritium represents an environmental risk rather than a waste material. It will probably be necessary to provide for disposal of this material at some future time.

2.5.5 Waste Licensing Issues

Although a great deal of effort has been expended to develop methods of nuclear waste treatment and disposal, the lack of agreement for waste disposal forms and of standards for licensability constitute a major deficiency of the nuclear fuel cycle. It has been assumed that wastes from any of the major fuel cycle options can be disposed of in an acceptable way. However, major decisions must be made to resolve the outstanding issues. Major economic impacts may result from mandated standards for waste management.

The issues may be classified under waste treatment standards, geologic repository standards, treatment of low-level wastes, and the need for AFR facilities.

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The critical issues in waste treatment include the lack of agreement on the ultimate configuration of the fuel cycle, the lack of agreement on the required degree of immobilization of hazardous materials within the waste medium, the lack of expert agreement on low-level radiation-exposure risks, and the ambiguity of the ALARA emission standard for radioactive effluents. Depending on the interpretation of these parameters, the technology for waste treatment existing today may be fully adequate to meet societal needs, or it may fall considerably short. Developing accepted standards may be a difficult and time-consuming step, and one which slows the progress of the overall waste program.

The standards and regulations for a geologic waste repository will have to deal with many difficult issues, including:

- the extent to which government-owned waste disposal facilities will be licensed and the process for licensing action;
- how to define "permanent" isolation (the degree of assurance required from each of the multiple isolation barriers);
- the extent of potential hazard to future generations from the TRU wastes, principally plutonium after the first 100 to 500 years (conceivably, future standards might limit TRU content of the repository, forcing as high as practicable recycle of TRU into reactors);
- the extent to which stored wastes must be retrievable;
- the extent to which state and local entities will be compensated for the possible undesirable impacts and risks of a repository and the extent to which state and local bodies will be authorized to establish criteria.

The issues for low-level beta-gamma waste disposal facilities parallel the issues for high-level repositories.

AFR facilities are controversial in that they require spent-fuel transportation in contrast to the alternative of enlarging at-reactor storage pools. Because AFR facilities are temporary, their deployment "postpones" terminal

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waste disposal; their use may also introduce states' and local rights' issues that may delay licensing.

2.5.6 RD&D Needs

The RD&D program is separated into the waste treatment and waste disposal requirements.

Waste Treatment

The support R&D for waste treatment covers development of a number of forms for containment of high-level waste, iodine, and other radioactive materials to be stored for long periods of time. The other major R&D requirements for waste treatment center on two types of facilities: cold testing facilities and hot pilot plants.

The cold facility is required to develop the mechanical hardware that is needed for remote operation and maintenance of the waste treatment systems. Clearly, the requirements for this facility vary from cycle to cycle, but it is needed in all cases. The estimated cost for a typical facility is about \$50 to \$100 million to be spent over a period of 20 years. This facility might be constructed and operated for a shorter time period at increased expense, but it is unlikely that sufficient design information could be obtained in less than five years.

It is assumed that parallel R&D programs would cover spent-fuel encapsulation and high-level waste treatment for permanent storage. The former operation would provide wastes to be placed into early repository tests (on a retrievable basis) and provide a firm design basis for the commercial-scale encapsulation facility associated with operation of a lead geologic repository. A spent-fuel encapsulation pilot plant program has been estimated by NASAP to cost \$100 to \$200 million.

The high-level waste pilot plant would be associated with the reprocessing hot pilot plant (capacity of 150 metric tons of heavy metal per year) and would develop the generic procedures for vitrifying high-level waste. These techniques are required, even in the event that commercial reprocessing were not authorized in the U.S., to treat wastes that have been generated in the past, such as those at the NFS plant at West Valley, New York. The general character of high-level waste is not expected to be fuel-cycle dependent;

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however, the difficulty of pilot plant operations will probably be strongly influenced by the activity level. This is in turn influenced by the fuel burnup and ex-reactor cooling time.

The high-level waste pilot plant is required to demonstrate that the waste treatment systems can be satisfactorily integrated with each other and with the fuel recycle operations that occur on the site. The hot waste project includes several years of cold operation after construction is completed. Hot operation would begin before designs are finalized on the commercial waste facility. The pilot plant estimated R&D costs are \$200 to \$400 million, which may be minimized by proper coordination of the effort with fuel recycle studies at an early stage. On the other hand, the costs may be at the high end of the indicated range if extensive retrofitting of hot facilities is required to integrate the waste treatment systems and test the concepts, or if wastes from several fuel-cycle options must be processed. The estimated time requirement is 13 years. This facility might be constructed and operated for a shorter time period at increased expense, but it is unlikely that sufficient design information could be obtained in less than 8 years.

Costs and schedule for this R&D effort are shown in Table 77.

R&D on waste treatment may include work on actinide partitioning and transmutation in fast reactors as well as incineration and other compaction and fixation techniques. This process, if successfully developed, could substantially reduce long-term hazards (greater than 1000 years) from terminal waste storage. The program is necessarily long-term in that it is coupled to reprocessing and fast-reactor development and deployment.

Waste Disposal

The major RD&D efforts that must be accomplished to ensure safe disposal of solidified waste are (1) increased knowledge of rock behavior as it applies to thermomechanical and thermohydrologic interactions; (2) development of a data base to identify nuclide leaching rates, groundwater chemistry, nuclide solubility in groundwater, and nuclide transport through rocks; and (3) increased knowledge of the overall impact of radiation on disposal media.

Several different geologies are candidates for the siting of geologic repositories. Salt has been selected as a leading candidate geology, but other geologies are also being evaluated. The R&D requirements for in-depth examination of different geologies differ in kind but not in general content.

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Table 77. RD&D Cost and Schedule in Support
of a Lead 2000-Acre Repository

<u>Waste treatment</u>	<u>\$ million^a</u>	<u>Timing</u>
Supporting R&D	150-300	1979-1995
Cold testing facility	50-100	1990 operation
High-Level Waste Pilot Plant ^b	200-400	1992 operation
Spent Fuel Encapsulation Pilot Plant ^c	100-200	1990 operation
Subtotal	500-1000	
<u>Waste repository</u>		
Supporting R&D ^d	1000-2000	1979-1995
Total RD&D	1500-3000	

^aJanuary 1978 \$.

^bMatches the scale of the Reprocessing Hot Pilot Plant.

^c200-metric-ton-per-year scale.

^dSupporting geologic R&D covers several alternate media/
sites.

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An important step in the R&D program that will provide experimental verification of the design and the models is the construction of an experimental test facility. Test facilities may be housed in existing underground openings in the respective formations. Potential sites for these facilities are under review in several regions that are also being investigated for suitable sites for repositories. Some of the tests to be carried out at candidate sites are:

- Geologic response tests -- heat and deformation in pillar geometry for design configuration, corrosion, waste form/host rock interaction, migration rates for mixed waste nuclides through geologic media, gas permeability and reconsolidation of backfill material, and borehole and shaft sealing against inflow of water;
- Mechanical or equipment tests -- retrievability of emplaced canister, placement and removal of backfill materials, corrosion of waste packages, and functional tests of underground handling and monitoring equipment;
- Evaluation of accident conditions -- ability of personnel and equipment to cope with selected situations and evaluation of design-basis accidents.

The cost and schedule of geologic R&D efforts leading to a repository are most uncertain. Several media/sites will be investigated, but the depth of each investigation has not been determined. Even after a repository is in full-scale operation, it appears likely that geologic research into alternatives would continue at a significant level. These activities are required to support all fuel-cycle alternatives. Because of the uncertainties, the anticipated cost of supporting RD&D facilities for a lead repository, as displayed in Table 78, is estimated to be \$1 to \$2 billion through 1995. This assumes that several alternative sites would be evaluated.

2.5.7 Waste-Management Costs

Waste management of spent fuel in the once-through cycle includes onsite storage at the reactor site, optional storage at an AFR facility, and disposal at a geologic repository. As indicated in Table 78, the cost of a 5,000- to 15,000-metric-ton AFR facility would be about \$200 to \$300 million. A geologic repository in salt is expected to cost \$500 million to \$1 billion.

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Table 78. Estimate of Lead Plant Waste Disposal Costs

Required facilities	Estimated cost (\$ million) ^a	Estimated completion date
Lead AFR storage, 5,000–15,000 metric tons	200–300	1983–1985
Lead encapsulation plant, 6,000 metric tons per year	300–400	1994
Lead geologic repository, 45,000 metric tons	500–1,000	1994
Total facilities	1,000–1,700 ^b	

^a January 1978 \$.

^b Cost of these facilities to be recovered through fees.

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The associated fuel encapsulation plant is expected to cost \$300 to \$400 million.

DOE has estimated a spent-fuel disposal charge of \$117/kg heavy metal and an AFR storage plus disposal charge of \$232/kg heavy metal. This charge includes cost recovery for the waste management facilities as well as the associated R&D. In view of the major uncertainties in waste facility standards, such cost estimates must be regarded as highly uncertain.

Costs for disposal of suitably packaged high-level waste from a reprocessing plant are expected to be about half the cost of spent-fuel disposal. This is based on the assumption that the high-level waste will be packaged in fewer canisters and require fewer cask shipments and fewer handling operations at the disposal facility.

However, when the costs of high-level waste treatment and the waste treatment and disposal of other wastes generated from the closed fuel cycle are considered, overall waste costs in the closed fuel cycle are probably greater than for the once-through cycle. Those costs are usually allocated to reprocessing and fuel fabrication; direct cost comparisons are not currently available between closed and once-through fuel cycles.

2.6 TRANSPORTATION SYSTEM

2.6.1 Description and Status of Technology

Radioactive materials in a wide variety of physical and chemical forms must be transported between facilities that are situated in diverse geographic locations. Materials such as uranium or thorium ore concentrates, uranium hexafluoride, fresh reactor fuel, spent reactor fuel, plutonium, and many forms of radioactive waste have been shipped routinely in the United States for many years. Other materials that may be expected in the future to require transportation on a commercial scale are solidified high-level waste, cladding hulls, radioactive gases, and fresh and spent fuel from alternative reactor fuel cycles. Although packaging for these materials has not yet been developed, the development of appropriate transportation systems is not expected to pose major technical barriers. However, transportation of highly radioactive materials through populated areas is a very sensitive subject, and institutional barriers must be recognized.

The radioactive materials in the nuclear fuel cycle are transported primarily by truck or train; the choice is essentially an economic one. Barge and sea transport have not been considered for domestic shipments in this report, although in many cases they are viable economic alternatives and are a practical necessity for many international shipments. Air transport is not considered for commercial shipments in the fuel cycle.

Figure 49 shows the transportation system for the PWR once-through fuel cycle; Figure 50 shows the system for the PWR with recycle fuel. In this report, any necessary recycle uranium or plutonium conversion facilities are assumed to be colocated with the reprocessing facilities. The transportation systems discussed below are arranged in descending order of the associated radioactivity.

Spent-Fuel Transportation

Spent fuel contains uranium, plutonium, other transuranic radionuclides, and large quantities of highly radioactive fission products. The radioactive decay processes generate a large amount of decay heat. The spent-fuel assemblies are stored at the reactor a minimum of 120 days to allow radioactivity and heat generation levels to decrease before transport to AFR storage or reprocessing facilities. At 120 days, the LWR spent-fuel assembly contains about 2×10^6 Ci of activity and generates about 9.4 kW of heat. Casks are

Figure 49. Transportation System for the LWR Once-Through Fuel System.

Figure 50. Transportation System for the LWR Recycle Fuel Cycle.

licensed for spent LWR fuel cooled only 120 days; however, the current practice is to let the spent fuel cool much longer before shipment.

The higher heat generation of fuel from the fast breeder has a marked impact on cask design or the amount of fuel that can be transported in a cask. For example, an existing cask design will accommodate six assemblies after a 365-day cooling period. If 90-day-cooled fuel is to be shipped, only one assembly could be shipped due to the additional heat and radiation loads.

Spent-fuel shipping casks are designed with a thick (typically 15-cm) layer of steel, lead, or depleted uranium for gamma shielding and with a water or solid hydrocarbon (e.g., epoxy resin) layer for neutron shielding. Steel liners provide structural strength and additional gamma shielding. Heat removal is usually accomplished by natural convection and conduction inside the cask and by natural convection and radiation from numerous fins on the exterior of the cask. Although not required for safety, mechanical heat removal systems are sometimes provided to reduce cask cooldown times, thus reducing turnaround times and shipping costs. All these factors make spent-fuel shipping casks large, heavy, and costly. Spent-fuel shipping casks using existing designs are expected to cost around \$1 million for truck transport and \$4 million for rail transport.

Spent fuel has been shipped by both rail and truck in the United States for many years, and several types of shipping casks have been designed. Table 79 lists information about casks currently available or proposed for domestic use. Figure 51 shows a detailed view of the rail-transported NLI 10/24 cask, and Figure 52 shows the cask loaded on a rail car. About 50% of the current reactors that are operating or under construction do not have rail sidings, thus precluding the use of rail casks at these sites. The other factors influencing the choice of rail or truck transport are largely economic ones: the cask payload; loading, unloading, and travel time; and cask availability. Rail shipments have payload and loading/unloading time advantages, but current shorter truck travel times at least partially compensate for these advantages. Utilities have purchased both types of casks to meet the needs of their particular situation. Reprocessing facilities have been designed to receive both types of casks.

After ten years, the radioactivity level in the spent fuel would be lower by a factor of 10, and the heat generation rate would be lower by a factor of more than 10. Subsequent transportation of ten-year-old spent fuel to a geologic repository or other facility could be made in the existing spent-fuel casks. However, because of the lower radioactivity and heat generation levels, it could be more economical to design and license separate casks for

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Figure 51. NLI 10/24 Irradiated-Fuel Shipping Cask.

Drawing Courtesy of
NL Industries, Inc.

Figure 52. NLI 10/24 Irradiated-Fuel Cask and Carrier.

Table 79. Licensed and Available Shipping Casks for
Current-Generation LWR Spent Fuel

Cask designation	Number of assemblies		Loaded cask weight (metric tons)	Usual transport mode	Shielding		Cavity coolant	Maximum heat removal (kW)	Status
	PWR	BWR			Gamma	Neutron			
NFS-4 (NAC-1)	1	2	23	Truck	Lead and steel	Borated water and antifreeze	Water	11.5	Licensed
NFS-5	2	3	25	Truck	Uranium and steel	Borated water and antifreeze	Water	24.7	SAR submitted
NLI 1/2	1	2	22	Truck	Lead, uranium, and steel	Water	Helium	10.6	Licensed
TN-8	3		36	Truck ^a	Lead and steel	Borated solid resin	Air	35.5	Licensed
TN-9		7	36	Truck ^a	Lead and steel	Borated solid resin	Air	24.5	Licensed
TN-12	12	32	97	Rail	Steel	Borated solid resin	Air	135	Licensed in Europe only
IF-300	7	18	63	Rail ^b	Uranium and steel	Water and antifreeze	Water	76 ^c	Licensed
NLI 10/24	10	24	88	Rail	Lead and steel	Water	Helium	97 ^d	Licensed

^aOverweight permit required.

^bTruck shipment for short distances with overweight permit.

^cLicensed decay heat load is 62 kW.

^dLicensed decay heat load is 70 kW.

ten-year-old or older fuel assemblies. It is expected that such casks could carry twice as many assemblies as current casks.

In LMFBR fuel cycles, some new considerations are introduced because of the higher volumetric heat generation rates and the presence of a fuel assembly duct that reduces heat transfer. It may be necessary to ship the spent fuel in sodium to promote cooling, but this would occur at the expense of more complex handling procedures. Several conceptual spent-fuel cask designs have been considered, one of which is shown in Figure 53. As previously indicated, the carrying capacity of the cask is related to the cooling time prior to shipment. For example, 90 (or 180) day cooling would reduce the payload from six to one (or four) assemblies per cask.

High-Level Waste Transportation

High-level waste is the primary waste generated when spent fuel is reprocessed. The waste will be solidified in borosilicate glass or a ceramic matrix and placed in stainless steel containers. High-level waste includes all except the gaseous fission products from the fuel, all of the transplutonic actinides, and about 0.5% of the uranium and plutonium originally in the spent fuel. There is no high-level waste in commerce today. Casks designed specifically for shipment of solidified high-level waste have not been built; however, they are expected to resemble the casks currently available for rail shipment of spent fuel. It is expected that, because of the higher payload per shipment, rail will be used exclusively for shipments to geologic disposal. There are two options available: (1) utilize an existing rail cask with a modified internal basket and possibly with several basket designs depending on the relative heat generation rates or (2) design a new cask optimized for the specific use.

The radiation level and heat generation rate of the high-level waste may vary significantly among the various fuel cycles. For plutonium recycle, the neutron source from trace actinides is ~30 times as large as for initial LWR cycles or fresh UO_2 waste, and the heat generation rate may be 20 to 100% higher than for initial cycles or fresh UO_2 waste.

Cladding Waste Transportation

Cladding waste is the general term denoting the solid metal residues from fuel reprocessing operations. The residues include short lengths of fuel cladding or hulls with some residual fuel, massive end fittings, fuel support

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Figure 53. Conceptual Design for Irradiated LMFBR Fuel Shipping Cask.

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grids, and assorted springs, spacer elements, and fuel bundle support rods. The radioactivity of cladding wastes arises both from neutron-induced isotopes and from the fission products and actinides present in the small amount of fuel that adheres to the cladding. Cladding wastes generated to date in commercial fuel reprocessing operations have been disposed of by onsite burial. Casks have been designed specifically for rail or truck shipment of fuel bundle residues to geologic disposal but have not been built. These casks are expected to resemble casks currently available for shipment of spent fuel but should be somewhat simpler in design because heat removal requirements would be reduced.

The casks would not require neutron shielding and would require only a few inches of gamma shielding. Heat loads would be low enough that cooling fins would not be required.

The Savannah River Laboratory has proposed that uncompacted cladding material be cemented into 55-gal drums. The radiation level would be in the range of 300 R/hr from this type of package. A conceptual cask for transporting six 55-gal drums of cladding would weigh about 25 metric tons and cost about \$240,000.

Plutonium Oxide Transportation

Plutonium is separated from the spent fuel at the reprocessing facility and converted to PuO_2 powder before being shipped. Plutonium has been transported for many years in a variety of forms and containers for research and weapons programs. The most common package is the 6M type, consisting of a 5-liter steel drum with a 12.7-cm inner vessel of schedule 80 steel pipe. The inner and outer vessels are separated by an insulating material. The PuO_2 is packaged in two metal containers holding about 2.2 kg of PuO_2 . The contents are limited to this amount because of a package heat dissipation limit of 10 W. If not limited by radiation dose or criticality considerations, 120 packages are shipped in a standard-sized truck van.

Figure 54 shows a PuO_2 overpack (the PPP-1M) that was designed for commercial-scale operations to reduce handling requirements and thereby reduce occupational radiation doses, as well as costs. The PuO_2 -filled containers, holding 32 kg of PuO_2 , are put into the overpack, which is about 17 in. in diameter and 63 in. long. Neutron and gamma shielding is provided by lead and a proprietary solid organic material. The design includes both a primary and a secondary pressure vessel, which is now required by NRC regulations. The overpack has a gross weight of 1800 lb and, for safeguard

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Figure 54. Modified Overpack (PPM-IM) for PuO_2 Shipment.

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purposes, has been designed so that no lifting lugs are required. Special handling equipment is required to load and unload the overpack.

Proliferation countermeasures would have impacts on PuO_2 shipping. If co-processing were mandated, then a PuO_2 - UO_2 mixture would be shipped; essentially the same types of shipping containers used for PuO_2 could accommodate the mixed oxides. If spiking were required, then heavier gamma shielding would be necessary, as discussed below. Figure 55 shows one concept for transporting overpacks to meet safeguard requirements. Eight overpacks are transported in a penetration- and accident-resistant material developed by Sandia Laboratories, essentially in a vault.

U-233 and Spiked Oxide Powder Transportation

Uranium-233 oxide powder has the same safeguards requirements as plutonium oxide powder. In addition, the U-233 oxide may contain several thousand parts per million of U-232. Some of the U-232 daughter radionuclides emit very energetic gamma rays, particularly 2.62 MeV from Tl-208. Therefore, the unshielded gamma radiation level may be about 100 rem/hr per kilogram of U-233 at 1 ft (assuming 1000 ppm U-232). The shielding provided by the overpacks and the accident-resistant material to meet the criteria of 10 CFR 71 for PuO_2 may not suffice for shipment of U-233 and may have to be supplemented with additional shielding.

In some NASAP fuel cycles, a spikant is added to the powder to produce radiation levels up to about 1000 rem/hr at 1 m. Shielding in addition to that provided by the overpack is likely to be required.

Fresh-Fuel Transportation

Fresh-fuel assemblies are usually shipped enclosed in a plastic bag and placed in a container that supports the fuel assembly along its entire length (sometimes called a strongback). Typically, one or two fuel assemblies are packaged within one container.

Containers for PWR fuel which meet the requirements of 10 CFR 71 are lightweight metal, about 5 m long and 1.2 m in diameter, and weigh up to 4800 kg when loaded. Six PWR containers (12 fuel assemblies) are shipped on a truck trailer using straightforward tie-down techniques.

Figure 55. Vault Concept with Top-Plug Loading for Enhanced Security During Shipment.

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The primary consideration for fresh-fuel shipment in other fuel cycles may be the neutron or gamma shielding requirements. If the radiation levels are only slightly greater than for low-enriched UO_2 PWR fuel, shielding materials could be placed inside the existing containers. Space for the shielding and reduction of the radiation source would result if only one fuel assembly per container were shipped. Relicensing would be required, but it should not be difficult.

If radiation levels are relatively severe, for example, as would be the case for spiked fuel or for recycle fuel containing U-232, spent-fuel casks could be used for transporting fresh fuel. Existing casks are overdesigned for such purposes and are thus uneconomical; for a mature cycle, it would be economical to design and license fresh-fuel containers for the specific fuel assemblies concerned. Other variations may also be practical. For example, spiked fuel could be shipped in spent-fuel casks that would otherwise be returning empty to the reactor. In this case, rerouting from the reprocessing facility via the fabrication facility would be necessary (if they were not colocated), as well as additional cask licensing.

Low- and Intermediate-Level Waste Transportation

Low- and intermediate-level wastes are packaged according to their physical and chemical composition and radiation level. As a result, the outer container may range from wood boxes or 55-gal drums to shielded casks. In general, the waste is packaged in a stable solid form that has low leachability and is noncombustible. Shipment of radioactive noble gases in pressurized gas cylinders may be an exception to this general approach. Technology for the above operations is available.

U-235 and Thorium Oxide Powder Transportation

Fresh ThO_2 and enriched UO_2 , recycle ThO_2 after storage for ten years, and fully purified reprocessed enriched UO_2 can be handled without shielding. The enriched UO_2 is packaged within an overpack to meet the shipping requirements of 10 CFR 71, the amount of uranium within the inner container being determined by the enrichment and criticality limit. The ThO_2 and UO_2 containers are steel drums or cans with gasketed lids. Almost all shipments of oxide powder to the fabrication facility are made by truck.

Uranium Hexafluoride Transportation

Uranium hexafluoride (UF_6) is a chemically active material that reacts with water, many organic compounds, and some metals. At room temperature, UF_6 , a white solid that vaporizes below the boiling point of water, is shipped as a solid in steel, Monel, or other metal cylinders that are designed for the handling of gases. These cylinders may contain as little as 55 lb of highly enriched UF_6 or up to 14 tons for low-enriched, natural, or depleted UF_6 . The typical cylinder for shipping enriched uranium for LWR fuel contains up to 2.5 tons of UF_6 (Figure 56). When these cylinders are used to ship enriched uranium, they are placed within protective overpacks that meet the requirements of 10 CFR 71 (Figure 57). These overpacks are boxes or cylinders with about 6 in. of fire-resistant foam between steel shells. The maximum enrichments, amounts, and numbers of cylinders that can be shipped on a single truck or rail car are shown in Table 80.

Uranium and Thorium Ore Concentrate Transportation

Uranium and thorium ore concentrates are typically packaged and shipped in strong 55-gal drums with gasketed lids. These materials are shipped in truck or rail car as radioactive materials but do not require shielding or special overpacks.

2.6.2 Licensing, Safety, and Environmental Status

The shipment of fissionable materials is covered by NRC regulation 10 CFR 71. This regulation requires that shipments be able to withstand both normal shipping damage and accident conditions that could be encountered. Tests are prescribed to simulate impact, fire, and submersion in water. The degree to which technical work and demonstration may be required to license a new shipping container depends in part on the similarity to other tested designs and the type of fuel-cycle material it will contain. In addition to physical testing, there are sophisticated analytical procedures that are used to evaluate the designs. The licensing effort is determined by the hazard of the material being transported, the design innovations, the existence of applicable operating data, and the relevance of the design standards and analytical model being used. State and local concerns over radioactive shipments may play a role in the selection of transportation mode and routing.

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GENERAL DATA

OTHER DESCRIPTIVE TERMINOLOGY USED — 2-1/2-TON[^]

ENGINEERING DRAWING
REFERENCE

UNION CARBIDE CORPORATION,
ORGDP: D-P 35721-F, REV. 3

NOMINAL DIAMETER	30 IN.
NOMINAL LENGTH	81 IN.
WALL THICKNESS	1/2 IN.
NOMINAL TARE WEIGHT	1,400 LB
MAXIMUM NET WEIGHT	5,020 LB
NOMINAL GROSS WEIGHT	6,420 LB
MINIMUM VOLUME	26 CU FT
BASIC MATERIAL CONSTRUCTION	STEEL
SERVICE PRESSURE	200 PSIG
HYDROSTATIC TEST PRESSURE	400 PSIG
ISOTOPIC CONTENT LIMIT	5.0% ²³⁵ U MAX WITH MODERATION CONTROL

Figure 56. UF₆ Cylinder Model 30B.

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AVERAGE WEIGHTS

SHIPPING CONTAINER WEIGHT	1750 LBS.
FILLED 30" CYLINDER	6400 LBS.

TOTAL WEIGHT 8150 LBS.

Figure 57. Horizontally Loaded Protective Outer Package for UF_6 Cylinder.

Table 80. UF₆ Shipping Containers

Cylinder	UF ₆ weight	Maximum percent of U-235	No. per truck	No. per rail car
5A	55 lb	100	50	
12A	460 lb	5.0		
30B	2.5 tons	5.0	5	16
48X	10 tons	4.5	2	4
48Y	14 tons	4.5	1	4

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In addition to the regulations that have been developed to protect both the population and the environment from radiation or pollutants, there are regulations intended to protect against the diversion of sensitive nuclear materials. These safeguard requirements may involve escort vehicles, the necessity of using special communications systems with established check-in reports, and guards to escort the shipment.

An annual radiation dose of about 7 total-body man-rem is associated with transportation of radioactive materials (fissile and non-fissile) in an LWR once-through fuel cycle. Transport workers receive ~60% of this total dose. Of the various transport stages, shipments of irradiated fuel account for about 60% of the total dose from transportation. The GESMO study estimated a total population dose (workers and general public) of 1790 man-rem/GWe-yr for the entire LWR once-through fuel cycle. The radiological impact of normal transportation requirements would amount to 0.5% of the radiological impact for the entire fuel cycle.

The rate of release of heat to the air from a spent-fuel cask will be about 10 to 70 kW, depending on the type and amount of irradiated fuel contained. This might be compared with the rate at which waste heat is released from a 100-hp truck engine operating at full power, which is about 50 kW. The temperature of the air which contacts the loaded cask will be increased a few degrees, but the temperature of the air a few feet from the cask would remain unaffected.

Irradiated mixed-oxide fuel assemblies for LWRs will average heat generation rates about 10 to 20% over uranium fuel assemblies. Moreover, neutron radiation levels will be up to two orders of magnitude higher than for UO₂ fuel, so that some increase in neutron shielding may be necessary. The safety and environmental impacts of transporting fuel-cycle materials for alternative reactors would be similar to those for the LWR, given appropriately designed shipping containers.

The occupational risks of transportation accidents are small. Analyses for various fuel cycles indicate that the possibility of transportation accidents is on the order of 10^{-6} per vehicle mile. Shipping casks and other containers of radioactive material are designed to NRC standards and will maintain their integrity during quite severe accidents. The GESMO study estimated that one fatal injury per ten reactor years and \$475 property damage

per reactor year would be associated with transportation accidents for the LWR fuel cycle.

2.6.3 RD&D Status

The primary RD&D needs for the transportation system are

- design and licensing support for fresh fuel with high gamma-radiation levels,
- design and licensing support for spent casks for all cycles other than the standard once-through LWR, especially for fuel cooled for short periods of time,
- design and licensing support for high-level waste casks and cladding-hull casks,
- design and licensing support for recycle powder containers and transport vehicles,
- optimization of the low-level waste TRU transportation system,
- determination of interface requirements between the transportation system and the fuel-cycle facilities, especially the geologic repository.

Of these, probably the most critical are those that concern the most radioactive situations: high-level waste and spent-fuel casks and, to a lesser extent, cladding waste casks and casks for fresh fuel producing high gamma-radiation levels. In particular, the retrofitting or, probably more desirable, the designing of a new cask for spent fuel and high-level waste containing a high level of neutron emission (such as LMFBRs, plutonium recycle LWRs, and modified once-through LWRs) involves some problems. While the technology exists today to design safe transportation systems, technology must be developed in parallel with a developing fuel cycle to ensure that all licensing, safety, and environmental constraints are satisfied.

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The design, test, and demonstration or licensing schedule and costs for several generic types of shipping containers are shown in Table 81.

Transportation RD&D is not usually a critical item in the overall schedule for developing a fuel cycle.

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Table 81. Transportation RD&D

Area	Schedule for complicated packaging system (months)
Engineering design, analysis, and licensing	22-49
Prototype fabrication	
Truck cask	21-26
Rail cask	26-36
TRU waste package	8-14
Production fabrication	
Truck cask	16-18
Rail cask	23-26
TRU waste package	4-7
Functional and compliance checks	2-4
Overall schedule from start to completion of first production unit	
Truck cask	60-90
Rail cask	80-115
TRU waste package	35-55
Area	Costs for complicated packaging systems (thousands of \$) ^a
Engineering, design analysis, and licensing	150-1500
Prototype fabrication	
Truck cask	1600-1700
Rail cask	6800-8300
TRU waste package	250
Production fabrication (per unit) ^b	
Truck cask	1200
Rail cask	4800
TRU waste package	100
Functional and compliance checks (per unit)	3.2-33.8

^a January 1978 \$.

^b Costs estimated on the basis of the ratio of prototype to production fabrication times and includes a 25% surcharge for first of a kind.

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APPENDIXES

INTRODUCTION

The following appendixes contain the detailed fuel cycle and economic data which formed the basis for the NASAP economic and resource assessments. Throughout the NASAP report, consistent, simplified reactor and fuel cycle nomenclature is used. This is summarized in the table below, together with a more detailed nomenclature used only in Appendix B to describe all fuel cycles whether or not they were used further in the NASAP assessments. The table also gives cross-references to which fuel cycle mass flows were used in the analyses of Volumes III and V.

Appendix A contains descriptions of the various NASAP reactor/fuel-cycle systems and options. Also included in Appendix A are mass flow diagrams for various NASAP reactor/fuel-cycle systems. Appendix B presents a summary of the NASAP mass flow data base. The methodology and abbreviations used in naming the reactors and fuel cycles studied in the NASAP program are also included here. Appendix C contains costs and other economic-related information used in the analysis throughout the NASAP study.

Table I.1. Reactor/Fuel Cycles Included in NASAP Reactor Mass Flow Data Base

Reactor	Simplified fuel cycle designation used throughout NASAP report	Detailed fuel cycle designation used in Appendix B	Fuel cycle references			Source of data
			(a)	(b)	(c)	
PWR	LEU Current, OT	LEU5(U)	X	X	X	Combustion Engineering, Inc.
	LEU X, OT	LEU5(U)-X			X	"
	LEU, SGR	LEU5, Pu(U)-RUPu,	X	X	X	"
	LEU X, SGR	LEU5, Pu(U)-RUPu, X	X	X	X	"
		Pu(U)		X		"
		Pu(U)-X				"
	Pu(Th), Recycle	Pu(Th)-RPu	X	X	X	"
		Pu(Th)				"
		Pu(Th)-RPu, X				"
		Pu(Th)-X				"
	MEU3(Th), Recycle	MEU3 (U, Th)-RUF	X	X	X	"
		MEU3 (U, Th)-RUF, X				"
	MEU5(Th), Recycle	MEU5 (U, Th)-RUF		X	X	"
	LEU 15% imp., OT	LEU5(U)-15%	X	X	X	"
	LEU 30% imp., OT	LEU5(U)-30%	X	X	X	"
BWR		LEU5, Pu(U)-RUPu, Sc, Cp				"
		LEU5(U)-F18				"
		LEU5(U)-F18, X				"
		LEU5(U)				"
LWPR		LEU5(U)-STI				General Electric Company
		LEU5(U)-X				"
	HEU5(Th)-NR1	HEU5(Th)-RU5-B	X		X	Division of Naval Reactors, DOE
		HEU5, U3 (Th)-RUF-SB				"
	MEU5(Th)-NR5	MEU5(U, Th)-RU5-B		X	X	"
		MEU5(U, Th)-RU5				"
	MEU5(Th)-NR7	MEU5(U, Th)-RU5-T1M			X	"
		HEU3, U(Th)-RUF-SB				"
	HEU5(Th)-HGC	HEU5(Th)-RU5-HGC-B		X		Oak Ridge National Laboratory
		Pu(Th)-RPu-HGC		X		Argonne National Laboratory

(a) Mass flows used for resource/economics analysis, Volumes III and V of NASAP report.

(b) Mass flows used for analysis presented in Section A.1.1 through A.1.6 of the Appendix to Volume III. These are summarized in Tables B.3 through B.6 of Appendix B.

(c) Mass flow charts are presented in Appendix A of this volume.

Table I.1. (Continued)

Reactor	Simplified fuel cycle designation used throughout NASAP report	Detailed fuel cycle designation used in Appendix B	Fuel cycle references			Source of data
			(a)	(b)	(c)	
LWBR	HEU3(Th), Recycle-NR1	HEU3, U5(Th)-RUF-SB	X		X	Division of Naval Reactors, DOE
		HEU3, U5 (Th)-RUF				"
	HEU3(Th), Recycle-NR5	HEU3(Th)-RUF-AD			X	"
		HEU3(Th)-RUF-HGC, B				"
	HEU3(Th), Recycle-NR7	HEU3(Th)-RUF-TIM		X	X	"
		Pu(U)-RPu-HGC				Argonne National Laboratory
		HEU3(Th)-RUF-HGC				"
		HEU3(Th)-RUF, X(22)-HGC				"
		HEU3(Th)-RUF, X(30)-HGC		X		"
	MEU3(Th), Recycle-HGC	MEU3(U, Th)-RUF-HGC	X	X	X	"
		MEU3(U)-RUF-HGC				"
HWBR	HEU3(Th), Recycle	HEU3(Th)-RUF			X	Division of Naval Reactors, DOE
HWR	Nat U; OT	NU5(U)			X	Combustion Engineering, Inc.
	LEU, OT	LEU5(U)	X	X	X	"
	Pu(Th), Recycle	Pu(Th)			X	"
	MEU3(Th), Recycle	MEU3(U, Th)-RUF			X	"
	MEU5(Th), Recycle	MEU5(U, Th)-RUF			X	"
		Pu(U)-RUPu				"
		MEU5(U, Th)-RUF, X		X		"
SSCR	LEU, OT	LEU5(U)			X	Combustion Engineering, Inc.
		LEU(U)-X				"
	MEU5(Th), Recycle	MEU5(U, Th)-RUF			X	"
	MEU3(Th), Recycle	MEU3(U, Th)-RUF			X	"
		Pu(Th)-RPu				"

(a) Mass flows used for resource/economics analysis, Volumes III and V of NASAP report.

(b) Mass flows used for analysis presented in Section A.1.1 through A.1.6 of the Appendix to Volume III. These are summarized in Tables B.3 through B.6 of Appendix B.

(c) Mass flow charts are presented in Appendix A of this volume.

Table I.1. (Continued)

Reactor	Simplified fuel cycle designation used throughout NASAP report	Detailed fuel cycle designation used in Appendix B	Fuel cycle references			Source of data
			(a)	(b)	(c)	
HTGR		LEU5(U)				General Atomic Company
	MEU5(Th), OT	MEU5(U, Th)			X	"
	MEU5(Th), OT-Opt.	MEU5(U, Th)-FS	X	X		"
		MEU5(U, Th)-RU5				"
	MEU5(Th), Recycle	MEU5(U, Th)-RUF		X	X	"
		MEU5(U, Th)-RUF, FS				"
		MEU5(Th)-RUF				"
	HEU5(Th), Recycle	HEU5(Th)-RUF			X	"
	HEU3(Th), Recycle	HEU3(Th)-RUF	X		X	"
LMFBR	Pu(Th), Recycle	Pu(Th)			X	"
	MEU3(Th), Recycle	MEU3(U, Th)-RUF	X	X	X	"
	Pu(U)/U	Pu(U)/U/U-RPu, HM		X	X	General Electric Company
	Pu(U)/Th	Pu(U)/Th/Th-RPu, HM		X	X	"
	Pu(Th)/Th	Pu(Th)/Th/Th-RPu, HM		X	X	"
	MEU3(Th)/Th	MEU3(U)/Th/Th-RUF, HM		X	X	"
	Pu(U)/U	Pu(U)/U/U-RPu, HT	X		X	Westinghouse, Inc.
GCFR	Pu(U)/Th	Pu(U)/Th/Th-RPu, HT	X		X	"
		Pu(U)/U/U-RPu, HM				General Atomic Company
		Pu(U)/Th/Th-RPu, HM			X	"
		Pu(Th)/Th/Th-RPu, HM				"
FMSR		NU5(U)				Brookhaven National Laboratory

(a) Mass flows used for resource/economics analysis, Volumes III and V of NASAP report.

(b) Mass flows used for analysis presented in Section A.1.1 through A.1.6 of the Appendix to Volume III. These are summarized in Tables B.3 through B.6 of Appendix B.

(c) Mass flow charts are presented in Appendix A of this volume.

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Appendix A

NASAP REACTOR/FUEL-CYCLE SYSTEMS

Presented in Table A.1 is a list of the reactor/fuel-cycle systems which were studied by NASAP. The numbering system in Table A.1 follows the reactor descriptions and accompanying section numbers found in Chapter 1 of this volume. For example, all LWR cycles are numbered 1.1, 1.2, 1.3, HWRS are identified as 2.1, 2.2, 2.3, The fuel-cycle facilities required for each cycle are identified in Table A.1. Descriptions of the fuel-cycle facilities (fuel fabrication, reprocessing, waste management, and storage) are listed at the end of Table A.1. Also included in Appendix A are the mass flow diagrams for most of the NASAP reactor/fuel-cycle systems (Figures A.1-A.42).

Table A.1. NASAP Reactor/Fuel Cycle Systems

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
1.0 <u>Light Water Reactors (LWR):</u>					
1.1 <u>PWR LEU Current, OT:</u>					
This reactor-fuel cycle combination is a standard PWR using 4.3% low-enriched uranium oxide pellet fuel achieving 30 MWd/kg average burnup operating on a once-through cycle. Spent fuel will be stored at the reactor site or away-from-reactor storage facility. Ultimately, the spent fuel will be sent to a geologic spent fuel repository.	Fab 1	None	WM 2	ST 3	Reference NASAP reactor/fuel cycle base case. Present type that is widely deployed in the U.S.
	(U pellets)		WM 3	ST 4	
	Fab 4 option		WM 5		
			WM 6		
1.2 <u>PWR LEU X, OT:</u>					
This reactor-fuel cycle combination is a PWR using 4.3% low-enriched uranium oxide pellet fuel, modified to achieve 50 MWd/kg average burnup to decrease uranium requirements, operating on a once through cycle. Spent fuel will be stored at the reactor site or away-from-reactor storage facility. Ultimately, the spent fuel will be sent to a geologic spent fuel repository.	Fab 1	None	WM 2	ST 3	13% reduced uranium requirements from reference case.
	(U pellets)		WM 3	ST 4	
	Fab 4 option		WM 5		
			WM 6		
1.2a <u>PWR LEU 15% imp, OT:</u>					
Similar to Case 1.2, designed to achieve a 15% reduction of U ₃ O ₈ consumption over Case 1.1.	Fab 1	None	WM 2	ST 3	
	(U pellets)		WM 3	ST 4	
	Fab 4 option		WM 5		
			WM 6		
1.2b <u>PWR LEU 30% imp, OT:</u>					
Similar to Case 1.2, designed to achieve a 30% reduction of U ₃ O ₈ consumption over Case 1.1.	Fab 1	None	WM 2	ST 3	
	(U pellets)		WM 3	ST 4	
	Fab 4 option		WM 5		
			WM 6		
1.3 <u>PWR LEU, SGR (Spiked):</u>					
This reactor-fuel cycle combination is a PWR using 3% low-enriched uranium oxide pellet fuel and self-generated recycle fuel of partitioned uranium and plutonium. The Pu is spiked with cobalt-60 and recycled to fabrication. Fresh make-up fuel is LEU(5). Excess U from reprocessing is recycled to enrichment. It is the same as 1.5 with a spikant added.	Fab 1	Purex 4	WM 1-5	ST 4	Self-generating spiked fuel recycle.
	(U pellets)			ST 2 Pu	
	make-up				
	Fab 3				
	(U-Pu pellets)				
	Fab 6 option				

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
1.4 <u>PWR MEU 3(Th) Recycle:</u> This reactor-fuel cycle combination is a PWR using U-233 denatured with U-238 and mixed with thorium oxide to fabricate pellet recycle fuel. The spent fuel is reprocessed to recover the U-233/U-238 which is blended with additional U-233 from a secure storage center. Recovered plutonium is spiked and sold to a secure storage center.	Fab 3 (U-Th pellets) Fab 6 option	Thorex 3	WM 1-5	ST 1 ST 2 Pu ST 2 U(3)	
1.5 <u>PWR LEU, SGR:</u> PWR using 3% low-enriched uranium oxide fuel and self-generated recycle fuel of partitioned uranium and plutonium oxide; GESMO fuel cycle. Same as Case 1.3 without the spikant.	Fab 1 (U pellets) Fab 2 (U-Pu pellets)	Purex 1	WM 1-5	ST 2 Pu ST 4	
1.6 <u>LWR LEU, SGR, Coprocessed and Partially Decontaminated:</u> This reactor-fuel cycle combination is a PWR using fuel assemblies made up of 3% LEU oxide rods and MOX rods of coprocessed partially decontaminated U-Pu for self-generated recycle. The LEU oxide rods are fabricated in a contact facility and transferred to the final assembly area of the MOX fabrication facility. Spent fuel assemblies are reprocessed using the "CIVEX" process. The partially decontaminated, recovered U-Pu mixture is spiked and recycled as feed to MOX fuel remote fabrication. Excess recovered uranium is decontaminated and sent to enrichment. Wastes from LEU oxide fuel rod manufacture are sent to a low-level shallow land waste disposal site. Wastes from reprocessing and MOX fuel rod manufacture are sent to a geologic waste repository.	Fab 3 U-Pu & Fab 1 (U pellets)	Purex 3	WM 1-5	ST 4	Low-decontamination coprocessed fuel (Civex)

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
1.7 <u>LWR Pu(Th), Recycle:</u>					
This reactor-fuel cycle combination is PWR using Pu-Th sphere-pac fuel. The spent fuel is reprocessed to recover and separate the U-233, plutonium, and thorium. The recovered U-233 is denatured to 12% fissile with depleted uranium and sent to a secure storage facility. The recovered plutonium is recycled to fuel fabrication where it is mixed with make-up plutonium from a secure storage facility and with new thorium oxide. The thorium recovered during reprocessing is sent to an interim thorium storage facility where it will be allowed to decay for at least 10 years. Wastes from fuel fabrication and reprocessing will be sent to a geologic waste repository.	Fab 5 (Pu-Th)	Thorex 3	WM 1-5	ST 1 ST 2 U(3) ST 4 ST 2 Pu	Pu transmuter to produce U(3).
1.8 <u>LWR HEU 5(Th), Recycle:</u>					
This reactor-fuel cycle combination is a PWR using 4.4% fissile in total heavy metal fuel of make-up U-235, recycle U-233 and U-235, and thorium oxides. The spent fuel is reprocessed to separate the uranium, plutonium, and thorium. The uranium is recycled to fabrication where it is mixed with HEU from a secure center and thorium oxide. The thorium is placed in 10-yr interim storage. Wastes from reprocessing and fabrication are sent to a geologic waste repository.	Fab 3 U(3)6(5), Th (pellets) Fab 1 U(5), Th (pellets) mak-eup	Thorex 3	WM 1-5	ST 1 ST 2 Pu ST 2 U(3) ST 4	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
2.0 <u>Heavy Water Reactors (HWR):</u>					
2.1 <u>HWR LEU, OT:</u>					
This reactor-fuel cycle combination is a CANDU-type HWR using 1.2% slightly enriched uranium oxide pellet fuel optimized for a once-through cycle. Spent fuel will be stored at the reactor site or away-from-reactor storage facility. Ultimately, the spent fuel will be sent to a geologic spent fuel repository.	Fab 1 U pellets	None	WM 2 WM 3 WM 5 WM 6	ST 3 ST 4	HWR optimized for once-through 40% reduced uranium requirements from LWR reference case.
2.2 <u>HWR Nat U, OT:</u>					
This reactor-fuel cycle combination is a CANDU HWR once-through system using natural uranium fuel. The spent fuel is stored at the reactor site or at an away-from-reactor storage facility. For ultimate disposal, the fuel will be sent to a geologic spent fuel repository.	Fab 1 U-pellets	None	WM 2 WM 3 WM 5 WM 6	ST 3	
2.3 <u>HWR MEU 3(Th), Recycle:</u>					
This reactor-fuel cycle combination is a CANDU HWR using denatured U-233-thorium fuel. The spent fuel is reprocessed to separate the uranium, plutonium, and thorium. The recovered uranium is recycled to fabrication where it is mixed with EEU(3) from a safe storage facility and virgin thorium. The plutonium is sent to a safe storage facility. The thorium is sent to 10-yr interim storage. Wastes from fabrication and reprocessing are sent to a geologic waste disposal site.	Fab 3 U-Th pellets	Thorex 3	WM 1-5	ST 1 ST 2 Pu ST 2 U(3)	

A-5

D R A F T

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
2.4 <u>HWR MEU 5(Th), Recycle:</u> This reactor-fuel cycle combination is a CANDU reactor using recycled denatured U-233, U-235 mixed with make-up U(5). The spent fuel is reprocessed to separate the uranium, plutonium, and thorium. The U(3) (5) recovered during reprocessing is recycled to fabrication. The recovered plutonium is sent to a secure storage facility. The recovered thorium is placed in 10-yr interim storage. Wastes from fabrication and reprocessing are sent to a geologic waste storage repository.	Fab 3 U(3)&(5), Th pellets	Thorex 3	WM 1-5	ST 1 ST 2 Pu	Mass flow data not available.
2.5 <u>HWR Pu-Th, Recycle:</u> Plutonium-thorium fuel; spent fuel reprocessed to recover 233-U and Pu. U(3) is sent to secure storage. Pu recycled. Th to 10-yr interim storage. Wastes to geologic repository.	Fab 3 Pu-Th pellets Fab 6 option	Thorex 3	WM 1-5	ST 1 ST 2 U(3) ST 2 Pu	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
3.0 Water-Cooled Breeder Reactors (WCBR):					
3.1 Backfit Prebreeder/Advanced Breeder:					
<p>The prebreeder would consist of a PWR-type core backfit into a B&W 205 vessel. The fuel would consist of thorium pellets alternating with duplex pellets. The duplex pellets would consist of a uranium annulus around a cylindrical thorium center. Reactivity control would be the same as that in a standard PWR.</p> <p>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. Reactivity control would be achieved by movable thorium control rods.</p>					
3.1.1 LWPR MEU 5(Th) - NR 5:					
The prebreeder would utilize 15.4% enriched uranium-235 [U(5)], thorium duplex fuel. The spent duplex fuel would first be reprocessed to recover the uranium-235 from the uranium annulus and would then be reprocessed to recover the bred uranium-233 from the thorium. The recovered U-235 would be recycled to enrichment, the Pu and bred U-233 would be sent to a secure storage center, and the Th would be sent to 10-yr interim storage.	Fab 10 (U-Th duplex pellets) Fab 1 (Th)	Purex 1 Thorex 1	WM 1-5	ST 2 Pu & U(5) ST 1 ST 4	Cross-over of bred U from Th may be a problem during Purex, requiring a dedicated enrichment facility due to U-232 and U-233.
3.1.2 LWBR HEU 3(Th), Recycle - NR 5:					
The advanced breeder would use large extrapolated PWR type vessel modified for a tight lattice hexagonal fuel bundle, and thorium control rods. The fuel would consist of binary solid solution of highly enriched uranium and thorium in the form of pellets. The spent fuel would be reprocessed to recover the U-233 and thorium which would be recycled to remote fuel fabrication.	Fab 1-blanket (Th) Fab 3-Core (U-Th)	Thorex 1	WM 1-5		LWBR designed to operate at high power density and fuel burnup to improve near-term economics. Highest conversion ratio/lowest specific power. Requires larger core and reactor vessel than standard PWR. Self-sustaining breeder. Approximately 1.3 prebreeder discharges required to sustain 1 advanced breeder.

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D R A F T

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
3.2 Prebreeder/Breeder, LWBR Type I Modules:					
<p>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.</p>					
<p>High-gain converter phase would not be self-sustaining and would require makeup U-233. At equilibrium, one low-gain converter would be required to support about five high-gain converters. Annual uranium requirement 20-25% less than conventional PWRs.</p>					
3.2.1 LWBR MEU 5(Th) - NR 7:					
<p>The prebreeder would use less than 20% enriched uranium-235 fuel in the form of UO_2-ZrO_2-CaO_2 (ternary) duplex pellets alternating with ThO_2 pellets in the fuel rods. The duplex pellet would consist of a ternary annulus around a cylindrical thorium center. The spent fuel would be reprocessed in two stages to recover the U, Pu, and Th. The first stage would recover the U-235 remaining in the UO_2-ZrO_2-CaO annulus. The second stage would recover the bred U-233 from the thorium. The U-235 would be recycled to enrichment. The Pu and the bred U-233 would be sent to secure storage, and the Th would be sent to 10-yr interim storage.</p>					
	Fab 1-blanket Fab 10-core (U-Th duplex pellets)	Thorex 1 Purex 1	WM 1-5	SI 2 Pu & U(3) SI 1 SI 4	Will require two-stage reprocessing of the duplex fuel pellets. Cross-over of bred U from Th may be a problem during Purex requiring a dedicated enrichment facility due to U-232 and U-233.
3.2.2 LWBR MEU 3(Th), Recycle - NR 7:					
<p>The breeder would be fueled with a binary solid solution of highly enriched uranium and thorium in the form of pellets. The spent fuel would be reprocessed to recover the U-233 which would be recycled to remote fuel fabrication. The Th would be separated and also recycled to remote fuel fabrication.</p>					
	Fab 3-Core (U-Th)	Thorex 1	WM 1-5		2.2 discharges from the low-gain converter required to sustain one high-gain converter. Semi-annual refueling.

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
3.2 Backfit Prebreeder/Seed Blanket Breeder:					
<p>The prebreeder would consist of a PWR type core backfit into a PWR CE-80 vessel. The fuel would consist of a binary solid state solution of uranium and thorium. 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.</p> <p>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 solid solutions of UO_2 and ThO_2 in pellet form. Initially, the breeder would be loaded with fuel recycled from the prebreeder (mixed uranium-fissile fuel: bred U-233 and nonfissioned U-235). At equilibrium, the breeder would be fueled from recycled U (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.)</p>					
3.3.1 LWPR HEU 5(Th) - NR 1:					
<p>The backfit prebreeder would utilize 93% highly enriched U-235 fuel in the form of binary solid solution UO_2-ThO_2 pellets. The spent fuel would be reprocessed to recover all uranium isotopes and would be stored for fabrication into fuel for the initial fuel requirements for the seed blanket breeder. The recovered thorium is sent to 10-yr interim storage.</p>					
	Fab 1 (U-Th) pellets	Thorex 1	WM 1-5	ST 2 U(3) ST 1 ST 4	
3.3.2 LWBR HEU 3(Th), Recycle - NR 1:					
<p>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; the blanket would consist of thorium pellets. The core and blanket would be reprocessed to recover the U-233 and thorium which are recycled and mixed with make-up highly enriched U-233 from storage.</p>					
	Fab 1-blanket (Th) Fab 3-Core (U-Th)	Thorex 1	WM 1-5	ST 2 U(3)	Self-sustaining breeder. Two and one-half year batch refueling cycle.

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
3.4 <u>LWBR High Gain Converter:</u>					
3.4.1 <u>LWBR MEU 3(Th), Recycle - HCG:</u>					
The fuel for this reactor, the LWACR, would consist of ThO ₂ pellets with 12% UO ₂ in Zircaloy rods. The spent fuel would be reprocessed, and the uranium would be recycled to fabrication where it would be mixed with uranium containing 42% U-233. The thorium would be sent to 10-yr interim storage and the plutonium sent to secure storage.	Fab 3 (U-Th) pellets Fab 1-blanket (Th pellets)	Thorex 3	WM 1-5	ST 1 ST 2 Pu ST 2 U(3)	
3.5 <u>HWBR HEU 3(Th), Recycle:</u>					
This reactor-fuel cycle combination is a heavy water seed-blanket reactor system using U(3)-thorium fuel and thorium blanket rods. The spent fuel is reprocessed to separate the U-233 and the thorium. The recovered U-233 is recycled to fabrication where it is mixed with make-up U(3) from storage and new thorium. The thorium recovered during reprocessing is sent to 10-yr interim storage. Wastes from fabrication and reprocessing are sent to a geologic waste repository. While this reactor will be completely refueled each 2-1/2 years, the mass-flow diagram shows the annual requirements.	Fab 3 U(3)-Th pellets (core) Fab 6 option Th pellets (blanket)	Thorex 1	WM 1-5	ST 1 ST 2 U(3)	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
4.0 High-Temperature Gas-Cooled Reactors (HTGR):					
4.1 HTGR MEU 5(Th), OT:					
This reactor-fuel cycle combination is a high temperature gas cooled reactor (HTGR) using 20% U(5) fissile particle and ThO ₂ fertile particle fuel operating on a once-through fuel cycle. Spent fuel will be stored at the reactor site or away-from-reactor storage facility. Ultimately, the spent fuel will be sent to a geologic spent fuel repository.	Fab 7 U/Th	None	WM 2 WM 3 WM 5 WM 6	ST 3 ST 4	25% reduced uranium requirements from LWR reference case.
4.2 HTGR MEU 3(Th), Recycle:					
This reactor-fuel cycle combination is a high temperature gas cooled reactor using recycled, denatured U(3) as 12% fissile particles and thorium oxide fertile particles. The spent fuel is reprocessed to separate the fissile and fertile particles, which are reprocessed separately. The U and Pu from the fissile particles are recovered, separated, and sent to secure storage. The fertile particles are reprocessed to recover bred U(3) which is diluted with D(U) to 12% fissile and recycled to fabrication. The Th is sent to a 10-yr interim storage.	Fab 7 U/Th 2 particles	Purex 1 Thorex 1	WM 1-5	ST 1 ST 2 Pu ST 2 U(3) ST 4	
4.3 HTGR HEU 5(Th), Recycle:					
This reactor-fuel cycle combination is an HTGR with 3 different kinds of fissile particles, each mixed with the same type of ThO ₂ fertile particle to form 3 fuel element species. The initial and make-up fuel element consists of HEU(5) fissile and separate ThO ₂ particles; the recycle U-233 fuel element consists of the bred HEU(3) fissile and ThO ₂ fertile particles; the recycle U-235 fuel element consists of residual U-235 fissile and ThO ₂ fertile particles. The 3 elements are processed in the head end of a reprocessing plant; the fissile particles from the makeup elements are processed in a Purex plant. The recovered U-235 is recycled one time. Fertile particles from all of the fuel elements and U-233 fissile particles from the recycle U-233 elements are processed in a Thorex plant to recover the U-233 for recycle. The thorium is sent to 10-yr storage and a small amount of Pu recovered and sent to storage.	Fab 7 U/Th 3 particles	Purex 1 Thorex 1	WM 1-5	ST 1 ST 4 ST 2 U(5)	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
4.4 HTGR HEU 5(Th), Recycle (Spiked):					
This reactor-fuel cycle combination is identical to 4.3 with addition of a radioactive spikant added to the recycle streams in reprocessing prior to shipment to refabrication.	Fab 7 U/Th 4 particles	Purex 1 Thorex 1	WM 1-5	ST 1 ST 2 U(5) ST 4	
4.5 HTGR MEU 5(Th), Recycle (Denature in Situ):					
This reactor-fuel cycle combination is an HTGR using fuel consisting of recycle DU(3) and make-up DU(5) fissile particles and Th-U(5) fertile particles that will be denatured in the reactor. The spent fuel is reprocessed to separate the uranium, plutonium, and thorium. All recovered uranium is less than 12% fissile U(3). DU(3) from fertile particles is recycled to refabrication. The DU(5) and DU(3) from fissile particles are sent to storage. The plutonium is sent to a secure storage facility. The thorium is placed in 10-yr interim storage. Wastes from reprocessing and fabrication are sent to a geologic waste repository.	Fab 7 U/Th 3 particles	Purex 1 Thorex 3	WM 1-5	ST 1 ST 2 Pu ST 4 ST 2 U(5)	No mass flow data available.
4.6 HTGR HEU 3(Th), Recycle (Spiked):					
This reactor-fuel cycle combination is an HTGR using spiked, highly enriched U-233 fissile and thorium fertile particles. The spent fuel is reprocessed to separate the thorium and the U-233. All recovered U-233 from fertile and fissile particles is homogenized and recycled to refabrication where it is mixed with make-up U-233 from a secure storage facility for the manufacture of fissile particles. A radioactive spikant is added to the U(3) during the final product homogenization prior to shipment to refabrication. The fertile particles are fabricated from new thorium. The thorium recovered during reprocessing is sent to 10-yr interim storage. Waste from fabrication and reprocessing is sent to a geologic waste repository.	Fab 7 U/Th 2 particles	Thorex 1	WM 1-5	ST 1 ST 2 U(3)	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
4.7 <u>HTGR Pu(Th), Recycle:</u>					
This reactor-fuel cycle combination is an HTGR using fuel composed of highly enriched plutonium oxide fissile particles and thorium oxide fertile particles. The spent fuel is reprocessed to separate the plutonium, U-233, and thorium. The recovered plutonium is recycled to refabrication where it is mixed with Pu of high content and fabricated into fissile particles. Fertile particles are fabricated from new thorium. The U(3) recovered during reprocessing is mixed with depleted uranium to a final assay of 12% fissile and sent to secure storage. The recovered thorium is sent to 10-yr interim storage. Wastes from fabrication and reprocessing are sent to a geologic waste repository.	Fab 7 Pu/Th 2 particles	Purex 1 Thorex 1	WM 1-5	ST 1 ST 2 L(3) ST 2 Fu	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
5.0 Gas-Cooled Fast Breeder Reactors (GCFR):					
5.1 GCFR Pu(U)/Th:					
<p>This reactor-fuel cycle combination is a gas cooled fast reactor using a uranium-plutonium mixed oxide homogeneous core and thorium oxide blankets. The core is coprocessed to recover plutonium mixed with uranium which is blended with make-up Pu-U, 20% fissile, from a safe secure storage facility and with depleted uranium to attain the desired 14.0% fissile assay and quantity for feed to the fuel fabrication operations. The core assemblies are preirradiated for spiking prior to shipment to the reactor. Depleted uranium is mixed with the recovered U-233 from blanket reprocessing to produce 12% fissile assay denatured U-233 for storage or sale. The thorium recovered from blanket reprocessing is sent to storage for a decay period of at least 10 years. New or decayed thorium is fabricated into blanket elements.</p>	Fab 2-Core (U-Pu)	Purex 2- Core	WM 1-5	ST 1	
	Fab 1-Blanket (Th)	Thorex 1- Blanket		ST 2-U(3) ST 2 Pu ST 4	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
6.0 <u>Liquid Metal Fast Breeder Reactors (LMFBR):</u>					
6.1 <u>LMFBR Pu(U)/U:</u>					
This reactor-fuel cycle combination is an LMFBR using recycled coprocessed U-Pu mixed oxide in a homogeneous core and recycled uranium mixed with make-up depleted uranium in the axial and radial blanket assemblies. The core fuel is reprocessed separately from the blanket assemblies. All of the coprocessed, recovered Pu-U from the core is mixed with make-up uranium and some of the U-Pu recovered from blanket reprocessing for feed material to core fabrication. The excess coprocessed U-Pu from blanket reprocessing is sent to secure storage for later use in LWRs or LMFBRs. All other recovered U from blanket reprocessing is recycled to blanket fabrication after being mixed with make-up depleted uranium.	Fab 2-Core (U-Pu) Fab 1-Blanket (U)	Purex 2- Core Purex 2- Blanket	WM 1-5	ST 2 U-Pu ST 4	High Pu production to provide for long-term high-growth rate with minimal need for mined uranium.
6.2 <u>LMFBR Pu(U)/U-(Spiked):</u>					
	Fab 2-Core (U-Pu) Fab 1-Blanket (U)	Purex 2-Core Purex 4- Blanket	WM 1-5	ST 2 U-Pu ST 4	
6.2.1 <u>LMFBR Pu(U)/U (Spiked)-Heterogeneous:</u>					
This reactor-fuel cycle combination is an LMFBR using 14.3% fissile assay mixed U-Pu recycle fuel in the core and depleted U in the blanket assemblies. The core and blanket assemblies are reprocessed separately. The core is coprocessed, and all of the recovered U-Pu is mixed with a portion of the U-Pu recovered during blanket reprocessing to provide feed material to core fabrication. The excess U-Pu recovered during blanket reprocessing is sent to secure storage. The balance of the uranium recovered during blanket reprocessing is mixed with make-up uranium to provide feed material for blanket fabrication. Elements are pre-irradiated before shipment.					

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	* Comments
<p>5.2.2 <u>LMFBR Pu(U)/U (Spiked) Homogeneous:</u></p> <p>This reactor-fuel cycle combination and mass flows are identical to that discussed in Section 6.1 above, except that the excess mixed oxide is sent to storage and the core fuel assemblies are preirradiated before shipment.</p>	<p>Fab 2-Core (U-Pu) Fab 1-Blanket (U)</p>	<p>Purex 2- Core Purex 4- Blanket</p>		<p>ST 2 U-Pu ST 4</p>	
<p>6.3 <u>LMFBR Pu(U)/Th (Spiked):</u></p> <p>Uranium-plutonium oxide core and thorium oxide blanket; same as 6.2, except U-233 is recovered from blanket fuel and sold as denatured fuel. Pu makeup from LWRs.</p>					<p>Fast breeder to produce U(3) for denatured U(3) supply to converter reactor while still providing Pu for recycle. Westinghouse large pin transmuter; General Electric transmuter.</p>
<p>6.3.1 <u>LMFBR Pu(U)/Th (Spiked) Heterogeneous:</u></p> <p>This reactor-fuel cycle combination is an LMFBR using a 16% fissile U-Pu core and an axial blanket of depleted uranium. In addition this reactor has internal and radial blanket assemblies of thorium oxide. Core fuel assemblies are preirradiated for spiking prior to shipment. Core and axial blanket assemblies are coprocessed, and the recovered U-Pu is mixed with make-up U-Pu as feed to core fabrication. Excess depleted uranium from core reprocessing is used as diluent to the U(3) recovered from the internal and radial blanket reprocessing and as feed to axial blanket fabrication. Make-up depleted uranium is required to complete axial blanket fabrication feed material requirements. The denatured (in process) U(3) is stored in an interim storage facility. New thorium is used for internal and radial blanket fabrication.</p>	<p>Fab 2-Core (U-Pu) Fab 1-Blanket (Th) Fab 1-Axial Blanket (U)</p>	<p>Purex 4- Core and Axial Blanket Thorex 1- Inner & Radial Blankets</p>	<p>WM 1-5</p>	<p>ST 1 ST 2 Pu & U(3) ST 4</p>	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
<p>6.3.2 <u>LMFBR Pu(U)/Th (Spiked) Homogeneous:</u></p> <p>This reactor-fuel cycle combination is an LMFBR using a homogeneous core of U-Pu and axial and radial blankets of thorium oxide. Core assemblies are preirradiated for spiking before shipment. Core and blanket assemblies are processed separately. Core assemblies are coprocessed, and all of the recovered U and Pu are recycled to fabrication. Make-up Pu from secure storage and depleted uranium are mixed with the recycled U-Pu as feed to core fabrication. Blanket assemblies are fabricated from new thorium. The U(3) recovered during blanket reprocessing is denatured with the addition of depleted uranium in process. The DU(3) is sent to safe storage. The recovered thorium is stored for 10 years.</p>	<p>Fab 2-Core (U-Pu) Fab 1-Blanket (Th)</p>	<p>Purex 2- Core Thorex 1- Blankets</p>	<p>WM 1-5</p>	<p>ST 1 ST 2 Pu & U(3) ST 4</p>	
<p>6.4 <u>LMFBR Pu(Th)/Th (Spiked):</u></p> <p>This reactor-fuel cycle combination is an LMFBR using 14.2% plutonium-thorium mixed oxide homogeneous core and thorium oxide blankets. The core and blanket are reprocessed separately. All of the plutonium and part of the thorium recovered during reprocessing are recycled to core fabrication after being mixed with Pu-Th make-up material from secure storage. The thorium recovered during core reprocessing is mixed with the thorium recovered during blanket reprocessing and sent to interim thorium storage for 10-yr decay. The U(3) recovered during core reprocessing is mixed with depleted uranium and the U(3) recovered during blanket reprocessing to produce a 12% fissile denatured product which is sent to secure storage. Blanket assemblies are fabricated from new thorium.</p>	<p>Fab 3-Core (Th-Pu) Fab 1-Blanket (Th) Fab 6-Core Option Fab 4-Blanket</p>	<p>Thorex 3- Core Thorex 1- Blanket</p>	<p>WM 1-5</p>	<p>ST 1 ST 2 U(3) ST 2 Pu ST 4</p>	<p>General Electric transmuted. Full production of U(3) for maximizing number of denatured fueled reactors.</p>

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
6.5 <u>LMFBR MEU 3(Th)/Th:</u>					
This reactor-fuel cycle combination is an LMFBR using a 10.1% fissile recycle U-233 oxide homogeneous core and a thorium oxide blanket. The core and blanket are reprocessed separately. The denatured U-233 recovered during core reprocessing is mixed with the HEU(3) from the blanket reprocessing and make-up UC(3) to provide the feed for fabrication. The plutonium recovered during core reprocessing is placed in secure storage. Thorium recovered during blanket reprocessing is placed in interim storage for 10 years. Blanket assemblies are fabricated from new or decayed thorium.	Fab 3-Core (U) Fab 1- Blanket (Th)	Purex 1- Core Thorex 1- Blanket	WM 1-5	ST 1 ST 2 Pu ST 2 U(3) ST 4	General Electric denatured system. High performance denatured fuel reactor U(3) to improve fuel utilization of system.
6.6 <u>LMFBR Pu(U)/U (Metallic Fuel):</u>					
This reactor-fuel cycle combination is an LMFBR using metallic U-Pu alloy core and U metal blanket fuel elements. The core and blanket fuel are coprocessed separately in the reprocessing facility. The U-Pu recovered during core reprocessing is mixed with some of the U-Pu from blanket reprocessing for recycle to core fabrication. Excess U-Pu from blanket reprocessing is diluted to 20% fissile and sent to secure storage. Excess uranium from blanket reprocessing is recycled to blanket fabrication where it is mixed with make-up depleted uranium. Wastes from core fabrication are sent to a geologic waste repository. Wastes from blanket fabrication are sent to low-level shallow land burial facility.	Fab 9	Pyrometal- core Pyrometal- Blanket	WM 1-5	ST 2 Pu ST 4	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
7.0 Spectral Shift Control Reactors (SSCR):					
7.1 SSCR LEU, OT:					
This reactor-fuel cycle combination is a standard LWR reactor vessel with the BOP converted to SSCR. The reactor will use 2.8% U-235 fuel optimized for a once-through fuel cycle. The spent fuel will be stored at the reactor or at an away-from-reactor storage facility with ultimate disposal in a geologic spent fuel repository. Wastes from fabrication will be sent to a low-level shallow land waste disposal site.	Fab 1 U-Pellets	None	WM 2,3, 5,6	ST 3 ST 4	
7.2 SSCR MEU 3(Th), Recycle:					
This reactor-fuel cycle combination is a standard LWR vessel with the BOP converted to SSCR. The reactor is fueled with 12% fissile denatured U-233-thorium oxide fuel. The spent fuel is reprocessed to recover the uranium which is recycled to fabrication where it is mixed with make-up HEU(3) from a secure storage facility and new thorium. The plutonium recovered during reprocessing is sent to a secure storage facility. The thorium recovered during reprocessing is sent to a 10-yr interim storage. Wastes from fuel fabrication and from reprocessing are sent to a geologic waste repository.	Fab 3 U-Th pellets	Thorex 3	WM 1-5	ST 1 ST 2 Pu ST 2 U(3)	
7.3 SSCR MEU 5(Th), Recycle:					
This reactor-fuel cycle combination is a standard LWR reactor vessel with the BOP converted to SSCR. The reactor is fueled with HEU(5) make-up fissile material and recycled U(3) U(5) mixed with new thorium oxide. The spent fuel is reprocessed to recover the uranium, plutonium, and the thorium. The uranium consisting of denatured in situ U(3) and U(5) is recycled to fuel fabrication where it is mixed with virgin thorium oxide and HEU(5). The plutonium is sent to a secure storage facility, and the thorium is placed in 10-yr interim storage. Wastes from fabrication and reprocessing are sent to a geologic waste repository.	Fab 3 U(3)&(5)-Th	Thorex 3	WM 1-5	ST 1 ST 2 Pu	

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
8.0 Accelerator-Driven Reactors (ADR)					
8.1 ADR/PWR-LAFR:					
Load 2% U-235 fuel into linear accelerator fuel regenerator; enrich via irradiation to increase Pu content; transfer enriched fuel to PWR; return to LAFR to re-enrich, to PWR, etc.; ship depleted fuel to spent fuel storage and then geological disposal.	Fab 1 or Fab 2	None	WM 2 WM 6	ST 3	BNL design.
8.2 ADR/TMF-ENFP Th-Pu:					
Pu(Th) fuel irradiated in target-blanket to produce U-233; reprocess fuel to recover U-233; U-233 denatured and sold to secure storage center, plutonium recycled; initial Pu from reprocessed fuel from other reactors.		Tborex 3	WM 2 WM 3 WM 5		ORNL design method to burn excess reprocessed plutonium.

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
9.0 <u>Molten Salt Reactor (MSR):</u>					
9.1 <u>DMSR MEU 5(Th)-OT:</u>					
This reactor-fuel cycle combination is a molten salt reactor (MSR) using MEU 5(Th) initial fuel. Fission products and plutonium are allowed to build up in the circulating fuel until the end of reactor life. Make-up 20% fissile uranium salt is added to maintain a constant fissile content in the reactor charge. Gaseous fission products are removed continuously from the circulating fuel and packaged for permanent disposal. Uranium is recovered from the fuel charge at the end of reactor life, and other materials are packaged to be sent to a geologic waste repository.			WM 1 WM 4		ORNL design.

Table A.1. Continued

	Fuel fabrication option	Reprocessing option	Waste management option	Storage option	Comments
10.0 <u>Gaseous Core Reactor (GKR):</u>					
10.1 <u>GKR MEU 5/Th Blanket:</u>					
Load MEU, U-235 (20%) to core and thorium to molten salt blanket; break-even breeding of U-233, Pu stays in core; bred U-233 moved from blanket to core as make-up fuel; reprocessing is conducted continuously at the reactor site.		Onsite	WM 1-5		SSAI design, no external facilities.
11.0 <u>Tokamak Fusion-Fission Hybrid (THR):</u>					
11.1 <u>THR Fuel Producer:</u>					
Load natural U or Th into hybrid blanket; reprocess to recover Pu or U-233 to be used in other reactors; recycle fertile material to FFR; reprocessing wastes to geologic disposal.	Fab 1 U or Th pellets	Purex 1 or Thorex 1	WM 1-5		PNL design.
12.0 <u>Fast Mixed Spectrum Reactor (FMSR):</u>					
12.1 <u>FMSR LEU - OT:</u>					
Reactor is loaded initially with LEU averaging 7% U-235; Pu is bred in situ; equilibrium feed is natural U; spent fuel is stored and will be sent to a geologic repository.	Fab 9 (not remote)	None	WM 1 WM 3 WM 5 WM 6	ST 3 ST 4	BNL design.

D R A F T

FUEL CYCLE OPTIONS

Fuel Fabrication

Fab 1

Uranium and thorium materials that have very low activity are fabricated into pellets which, in turn, make up fuel rods and fuel assemblies. All of the operations are conducted in a conventional industrial manner.

Fab 2

Uranium and plutonium materials that have some significant level of activity associated with them are fabricated into pellets which, in turn, make up fuel rods and fuel assemblies. Because of its high degree of automation, the operational activity is remote, while the maintenance will still be a contact (or hands-on) activity.

Fab 3

Uranium and thorium or uranium and plutonium material with a very high level of radiation is fabricated into pellets which, in turn, make up fuel rods and fuel assemblies. Because the radiation levels are extremely high, all aspects of both operation and maintenance are performed remotely.

Fab 4

Uranium or thorium materials that contain a very low level of radiation are fabricated into spheres of three size categories. These spheres, when loaded into fuel pins and subjected to low-frequency vibrations, produce a moderately high material density in the pin. These pins are then arranged together to produce fuel assemblies. Because there are no significant radiation hazards, both operational and maintenance activities are performed in a contact (or hands-on) manner.

Fab 5

Uranium and plutonium materials that contain a moderate level of radiation are fabricated into spheres of three different sizes. These spheres are then loaded into pins, and the spheres form high-density material under conditions of low-frequency vibration. The pins are then arranged together to produce fuel assemblies. Because of the radiation levels, all operational activities are conducted remotely, while maintenance activities, because they involve less time and minimum material, are conducted in a contact (or hands-on) manner.

Fab 6

Uranium and thorium or uranium and plutonium materials that contain a very high level of activity are fabricated into spheres of three different sizes. These spheres are used to fill fuel pins under conditions of low-frequency vibration. The finished pins are then arranged together to form fuel assemblies. Because the radiation levels are so high, both operational and maintenance activities are performed remotely.

Fab 7

Uranium is processed to yield uniform UCO spheres which are coated with three layers: inner pyrolytic carbon, silicon carbide, and outer pyrolytic carbon. Thorium is processed to yield uniform ThO₂ spheres which are coated with two layers: an inner low-density pyrolytic carbon layer and an outer high-density pyrolytic carbon layer. These two types of particles are combined to form fuel rods which are then loaded into graphite blocks, the final element form. Remote operation and maintenance are required for U-233 fuels, while direct contact operation is acceptable for U-235 fuels.

Fab 8

Spent fuel is removed from its cladding and is processed through dry operations such as grinding, oxidizing, and reducing designed to restore original fuel mechanical and thermal properties. Fissile material may be added to increase the fissile content of the fuel material, which is then fabricated by one of the standard techniques into pins and assemblies. All operational and maintenance activities are performed remotely.

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Feb 9

The metallic fuel is cast into cylinders of precisely controlled diameter. These metallic cylinders are placed in pins, which are then sealed. The pins are collected into fuel assemblies. All operational and maintenance activities are performed remotely.

Feb 10

Uranium and thorium that have very low activities (and utilize contact operations) or that have very high levels of radiation (and utilize remote operations and maintenance) are fabricated into duplex fuel pellets alternating with thoria pellets. The annulus of the duplex pellet contains the fissile uranium, while the center contains thoria.

Reprocessing

PUREX 1

Spent fuel containing plutonium, uranium, and fission products is dissolved in nitric acid for subsequent processing by TBP solvent extraction. The products of this process are a stream of fission product waste which is solidified, a stream of purified uranium which may be converted to UF_6 , and a stream of purified plutonium which is converted to PuO_2 .

PUREX 2

Spent fuel containing plutonium, uranium, and fission products is dissolved in nitric acid for subsequent processing by TBP solvent extraction. The products of this type of processing are a stream of fission product waste, which is solidified, and a stream of plutonium-uranium without separation from each other. This product stream may be converted to a mixed oxide $(Pu-U)O_2$ with the same Pu/U ratio as the entering spent fuel, or the ratio may be modified by adding plutonium from an external source.

D R A F T

PUREX 3

Spent fuel containing uranium, plutonium, and fission products is dissolved in nitric acid for subsequent processing by TBP solvent extraction. The results of this form of processing are three streams: a fission product waste stream which is solidified, a plutonium-uranium product stream (<20% Pu) containing 1 to 10% of the original fission products which is converted to highly radioactive mixed oxide (Pu-U)O₂, and a purified uranium stream which may be converted to UF₆. An additional part of this process is the use of designs that reduce plant flexibility for the production of high-purity plutonium.

PUREX 4

Spent fuel containing uranium, plutonium, and fission products is dissolved in nitric acid for subsequent processing by TBP solvent extraction. The results of this form of processing are three streams: a fission product waste stream which is solidified, a plutonium-uranium product stream (<20% Pu) to which a Co-60 spike is added prior to conversion to (Pu-U)O₂, and a purified uranium stream which may be converted to UF₆.

THOREX 1

Spent fuel containing thorium, U-233, and fission products is dissolved in nitric-hydrofluoric acid for subsequent processing by TBP solvent extraction. This method of processing produces three streams: a fission product waste stream which is converted to an immobile solid form, a purified thorium stream which is converted to ThO₂, and a purified U-233 stream which is converted to UO₂.

THOREX 3

Spent fuel containing thorium, uranium, plutonium, and fission products is dissolved in nitric-hydrofluoric acid for subsequent processing by TBP solvent extraction. This processing alternative produces four streams: a fission product waste stream which is converted to a solid immobile form, a purified plutonium stream which is converted to PuO₂, a purified thorium stream which is converted to ThO₂, and a purified uranium stream which is converted to UO₂.

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THOREX 5

Spent-fuel microspheres contain two types of material, thorium-uranium-233 fission products and uranium-plutonium fission products. The two types of microspheres are separated physically before each is dissolved in acid and processed by TBP solvent extraction. The fertile particle is processed into three output streams: a fission product waste stream for solidification, a purified thorium stream for conversion to ThO_2 , and a purified U-233 stream for conversion to UO_2 . The fissile particle is processed to three similar output streams: a fission product waste stream for solidification, a purified plutonium stream for conversion to PuO_2 , and a purified uranium stream for conversion to UO_2 .

THOREX 6

This process is the same as the previous process (THOREX 5) with the exception that the second group of material, the fissile material, has two waste streams -- the fission product stream and the uranium stream. (The plutonium in the fissile particle is sent to waste with the fission products.) Both uranium and fission products are solidified for disposal.

THOREX 7

Spent-fuel microspheres that contain two material types, uranium-plutonium fission products and denatured uranium-thorium fission products, are physically separated in the reprocessing plant and individually dissolved in nitric acid and nitric-hydrofluoric acid respectively. The solutions of the two types of microspheres are individually processed through TBP solvent extraction systems. The fissile particle produces three streams: fission product waste, low enriched uranium, and plutonium. The fertile particle also produces three streams: fission product waste, low enriched uranium (U-233), and thorium. All the actinide streams are converted to dioxides, and the wastes are converted to solid immobile forms.

Pyrometal 1

Metallic spent reactor fuel containing uranium, plutonium, and fission products is dissolved in a copper alloy for molten salt-molten metal processing. There are three streams resulting from such processing: a salt waste stream containing fission products, a Pu-U metallic phase (<20% Pu), and a

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uranium metallic phase. The first metallic product is fabricated into recycle core elements, and the second metallic product is fabricated into recycle blanket elements.

Waste Management

WM 1

Waste from a reprocessing plant includes a high-level fission product stream, intermediate-level concentrates, process discards such as rags and papers, cladding hulls, and selected volatile radionuclides. These are reduced to a compacted stable form with low leach rates and packaged for shipment to a deep geologic federal repository.

WM 2

Waste from a fabrication plant includes intermediate-level concentrates, process discards such as rags and papers, and reject cladding. These are reduced to compacted stable form with a low leach rate and packaged for shipment to a deep geologic federal repository.

WM 3

Waste from a power reactor includes muds, resins, slurries, rags, and papers. These are reduced to a compacted stable form with low leach rates and packaged for shipment to a licensed shallow land burial site.

WM 4

Waste from reprocessing and fabrication operations that contain high levels of radiation and/or long-lived actinide wastes are solidified, placed in canisters, and buried in stable deep geologic formations (salt, granite, basalt) in such a manner to dissipate heat and control any radionuclide release.

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WM 5

Low-level wastes from reactors in relatively untreated forms are disposed of in shallow land burial sites which are selected and developed for the favorable geologic and hydrologic characteristics.

WM 6

Spent fuel, as with waste from reprocessing and fabrication, can be packaged and buried in stable geologic formations. Some of the engineering specifics are different between spent fuel and high-level waste disposal, but concepts and approaches are similar.

Storage

ST 1

Thorium recovered from reprocessing operations is stored as ThO_2 to permit high-activity decay daughter isotopes to decay. Minimal cooling is required for this storage, and when the troublesome isotopes have decayed, the material can be used like natural ThO_2 .

ST 2

Recovered plutonium and/or U-233 can be stored until it is needed for recycle fuel fabrication. Moderate levels of shielding and some cooling are required for both of these materials, and criticality potential must be considered in design and operation of storage vaults.

ST 3

Spent fuel is stored at large secured storage pools as an interim measure until disposal or reprocessing facilities are available. The technology for such operations is the same as currently practiced for at-reactor spent-fuel storage but will be modified to provide more fuel storage in a smaller area.

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ST 4

Depleted uranium, which can come from reprocessing or enrichment, is stored in bulk quantity until its use is required. No heating, shielding, or criticality problems are associated with its storage.

Notes:

1. Mass flows in kgs per 0.75 GWE-year.
2. Abbreviations: THM, total heavy metal; tSWU, metric tons of separative work; FP, fission products; BOC, beginning of cycle; EOC, end of cycle; WM, waste management; ST, storage.
3. NASAP data base, average for years 20, 21, 22, normalized from a 1,270-MWe reactor.

U-235
Total U
Pu fissile
Total Pu
THM
FP

	BOC	EOC
U-235	813	220
Total U	26,977	25,858
Pu fissile	—	178
Total Pu	—	246
THM	26,977	26,104
FP		839

Figure A.1. System 1.1: PWR LEU Current, OT.

Notes:

1. Mass flows in kg per 0.75 GWe-year.
2. Abbreviation: see note 2 in Figure A.1.
3. NASAP data base, average for years 20, 21, 22, normalized from a 1,270-MWe reactor.

U-235
Total U
Pu fissile
Total Pu
THM
FP

BOC	EOC
693	117
16,130	15,074
	117
	180
16,130	15,269
	848

Figure A.2. System 1.2: PWR LEU X, CT.

NOTES:

1. MASS FLOWS IN kg/0.75 GWe-yr
2. DATA SUPPLIED BY CE

U-235

TOTAL U

Pu FISSILE

TOTAL Pu

THM

FP

	BOC	EOC
U-235	671	108
TOTAL U	16,184	15,032
Pu FISSILE		124
TOTAL Pu		190
THM	16,184	15,222
FP		

Figure A.3. System 1.2a: PWR LEU 15% imp, OT.

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NOTES:

1. MASS FLOW DATA IN kg/0.75 GWe-yr
2. NORMALIZED FROM A 1270 MWe REACTOR

U-235

TOTAL U

Pu FISSILE

TOTAL Pu

THM

FP

BOC	EOC
540	54.3
13,497	12,480
	97.6
	162.1
13,497	12,643

Figure A.4. System 1.2b: PWR LEU 30% imp, OT.

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- U-235
U-236
U-238
U TOTAL
Pu FISSILE
Pu TOTAL
THM
FP

BOC		EOC	
MAKEUP	RECYCLE	MAKEUP	RECYCLE
562.2	63.8	163.8	33.9
		66.1	6.3
16900	8902	16509	8706
17462	8965.8	16739	8746
	264.6	115.7	177.1
	436.2	158.2	331.5
17462	9402	16897	9078
		587	248

Figure A.5. System 1.3: PWR LEU, SGR (Spiked).

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Abbreviations: See note 2, Figure A.1.
3. NASAP data base, year 25, normalized from a 1,270-MWe reactor.

U-233
U-235
Total U
Pu fissile
Total Pu
Th
THM
FP

	BOC	EOC
U-233	772	468
U-235	55	59
Total U	7,196	6,700
Pu fissile		17
Total Pu		94
Th	17,358	16,914
THM	24,554	23,708
FP		833

Figure A.6. System 1.4: PWR MEU 3(Th), Recycle.

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NOTES:

1. MASS FLOW DATA IN kg/0.75 GWe-yr
2. NORMALIZED FROM A 1270 MWe REACTOR

U-235
U-236
U-238
U TOTAL
Pu FISSILE
Pu TOTAL
THM
FP

BOC		EOC	
MAKEUP	RECYCLE	MAKEUP	RECYCLE
562.2	63.8	163.8	33.9
		66.1	6.3
16900	8902	16509	8706
17462	8965.8	16739	8746
	264.6	115.7	177.1
	436.2	158.2	331.5
17462	9402	16897	9078
		587	248

Figure A.7. System 1.5: PWR LEU, SGR.

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NOTES

1. 56.6 kg EXCESS Pu GENERATED
2. MASS FLOWS IN kg/GWe-yr
3. NORMALIZED FROM A 1270 MWe REACTOR

U-235
U-236
TOTAL U
Pu FISSILE
TOTAL Pu
TOTAL
FP

BOC		EOC	
MAKEUP	RECYCLE	MAKEUP	RECYCLE
6C0.2	49.1	139.2	48.5
	25.4	72.6	25.3
20170	6507.8	18615.2	6489
	265.7	219.8	76.5
	437.4	373.4	130.1
2C170	6944	18989	6619

Figure A.8. System 1.6: LWR LEU, SGR, Coprocessed and Partially Decontaminated.

NOTES:

1. MASS FLOWS IN kgs per 0.75 GWe-yr
2. NASAP DATA BASE, YEARS 21/25 NORMALIZED FROM A 1270 MWe REACTOR.

U-233
U-235
TOTAL U
Pu FISSILE
TOTAL Pu
Th
THM
FP

BOC	EOC
	270.1
	2.6
	292.2
1563.8	860
2947.3	2044.6
21600.8	21217.6
24548	23554.2
	823.6

Figure A.9. System 1.7: LWR Pu(Th), Recycle.

NOTES:

1. MATERIAL BALANCE CLOSED ON TOTAL U; U-233 and U-235 BALANCES NOT SATISFIED.
2. MASS FLOWS IN kg/0.75 GWe-yr
3. NORMALIZED FROM A 1270 MWe REACTOR

Th
U-233
U-235
TOTAL U
Pu FISSILE
TOTAL Pu
THM
FP

	BOC		EOC		EOC TOTAL
	MAKEUP	RECYCLE	MAKEUP	RECYCLE	
Th	1512	16783	1475	1636.4	17859
U-233		352.3	21.2	351.5	372.8
U-235	85.5	649.3	29.2	275.4	304.6
TOTAL U	427.5	5828	390.6	532.4	5715
Pu FISSILE			4.6	59.8	64.4
TOTAL Pu			6.2	86.3	92.5
THM	1939.5	22611	1872	2180.3	23675
FP			66.7	777.5	844.3

Figure A.10. System 1.8: LWR MEU(5) (Th), Recycle.

NOTES:

1. MASS FLOWS IN kg/0.75 GWe-yr
2. NORMALIZED FROM A 1260 MWe REACTOR

U-235
U-236
TOTAL U
Pu FISSILE
TOTAL Pu
THM
FP

BOC	EOC
531.7	55.9
	71.0
44321	43186
	149.4
	257.8
44321	54742
	875

Figure A.11. System 2.1: HWR LEU, OT.

NOTES:

1. MASS FLOWS IN <gs PER 0.75 GWe-yr
2. NASAP DATA BASE, YEAR 25 NORMALIZED FROM A 1260 MW_e REACTOR

U-235
TOTAL U
Pu FISSILE
TOTAL Pu
THM
FP

BOC	EOC
1021	348.5
143600	137470
	361.1
	489.6
143600	137961
	991.3

Figure A.12. System 2.2: HWR Nat U, OT.

NOTES:

1. MASS FLOWS IN kgs PER 0.75 GWe-yr
2. NASAP DATA BASE, YEAR 25 NORMALIZED FROM A 1260 MWe REACTOR

U-233
U-235
TOTAL U
Pu FISSILE
TOTAL Pu
Th
THM
FP

	BOC	EOC
U-233	850.8	746
U-235	71.4	68.3
TOTAL U	7496	7315.5
Pu FISSILE		23.5
TOTAL Pu		33.0
Th	50863	50067
THM	58359	57416
FP		857.1

Figure A.13.: System 2.3: MEU3(Th), Recycle.

NOTES:

1. MASS FLOW IN kg PER 0.75 GWe-yr

U-233
U-235
TOTAL U
Pu FISSILE
TOTAL Pu
Th
THM
FP

	BCC	EOC
U-233	836	829
U-235	333	145
TOTAL U	8234	8093
Pu FISSILE		22.5
TOTAL Pu		32.1
Th	55610	54812
THM	63844	62937
FP		906

Figure A.14. System 2.4: HWR MEU5(Th), Recycle.

NOTE:

1. MASS FLOWS IN kg/0.75 GWe-yr

	BOC	EOC
U-232		0.6
U-233		311
U-234		16.8
U-235		1.3
U-236		0.08
Pu FISSILE	1030	483
TOTAL Pu	3137	2135
Th	34263	34008
THM	37400	36473
FP		927

Figure A.15. System 2.5: HWR Pu-Th, Recycle.

Figure A.16. System 3.1.1: LWPR MEU5(Th)-NR5.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Protactinium assumed to decay to U-233.
3. Reprocessing and fabrication losses assumed to be 0.5% each, reflecting improved recycle technology in later-generation recycle plants.
4. Abbreviations: See note 2 in Figure A.1.

U-233
 U-235
 Total U
 Th
 THM
 FP

	BOC	EOC
U-233	1,523	1,549
U-235	91	113
Total U	1965	2,085
Th	103,874	103,018
THM	105,900	105,103
FP		~ 802

Figure A.17. System 3.1.2: LWBR HEU3(Th), Recycle-NR5.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Protactinium assumed to decay to U-233.
3. Abbreviations: See note 2 in Figure A.1.

	BOC	EOC	
		Stage 1	Stage 2
U-233	—	—	437
U-235	2,059	1,327	8
Total U	10,293	9,512	478.7
Pu fissile	—	81.3	—
Total Pu	—	96.5	—
Th	77,716	—	77,048
THM	88,009	9,608	77,527
FP	—	~707	~125

Figure A.18. System 3.2.1: LWPR MEU5(Th)-NR7.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Protactinium assumed to decay to U-233.
3. Abbreviations: See note 2 in Figure A.1.

	BOC	EOC
U-233	1,680	1,722
U-235	64	81
Total U	2,001	2,138
Th	113,715	112,769
THM	115,717	114,906
FP	-	~815

Figure A.19. System 3.2.2: LWBR HEU3(Th), Recycle-NR7.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Protactinium assumed to decay to U-233.
3. Abbreviations: See note 2 in Figure A.1.

*Fission products estimated

	BOC	EOC
U-233	—	320
U-235	970	269
Total U	1,043	800
Th	23,059	22,434
THM	24,102	23,234
FP		~860*

Figure A.20. System 3.3.1: LWPR HEU5(Th)-NR1.

NOTES:

1. MASS FLOWS¹ IN kg PER 0.75 GWe-yr.
2. PROTACTINIUM ASSUMED TO DECAY TO U-233.

U-233
U-235
TOTAL U
Th
THM
FP

	BOC	EOC
U-233	1,708	1,704
U-235	223	219
TOTAL U	3,057	3,102
Th	65,560	64,724
THM	68,617	67,826
FP		~820

Figure A.21. System 3.3.2: LWBR HEU(3)-Th, Recycle-NR1.

NOTES:

1. MASS FLOWS IN kg/0.75 GWe-yr
2. PROTACTINIUM ASSUMED TO DECAY TO U-233

	BOC	EOC
U-233	739	543.6
U-235	20.6	35.2
TOTAL U	6158	5784
Pu FISSILE		57.6
TOTAL Pu		112.2
Th	21683	21128
THM	27841	27024

Figure A.22. System 3.4.1: LWBR MEU 3(Th), Recycle-HGC.

NOTES:

1. MASS FLOWS IN kgs PER 0.75 GWe-yr
2. NASA? DATA BASE, YEAR 25.0 AND 27.5 NORMALIZED FROM 2.5 YEAR CYCLE TO ONE YEAR CYCLE

U-233
U-235
TOTAL U
Th
THM
FP

	BOC		EOC
	SEED	BLANKET	TOTAL
U-233	1355.2		1368.8
U-235	146.4		150.8
TOTAL U	2175.6		2202
Th		84076	83156
THM	2175.6	84076	85358
FP			840

Figure A.23. System 3.5: HWBR HEU3(Th), Recycle.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. NASAP data base, average for years 20, 21, and 22, normalized from a 1,332-MW_e reactor, 1.07-year cycle.
3. Abbreviation: See note 2 in Figure A-1.

U-233
U-235
Total U
Pu fissile
Total Pu
Th
THM
FP

	BOC	EOC
U-233	—	64
U-235	566	46
Total U	2,858	2,252
Pu fissile	—	29.2
Total Pu	—	58
Th	2,469	2,287
THM	5,327	4,597
FP	—	~727

Figure A.24. System 4.1: HTGR MEU5(Th), OT.

Note:

1. Mass flows in kg per 0.75 GWe-yr.
2. NASAP data base, year 16, normalized from a 1,332-MWe reactor, 1.07-yr cycle.
3. Abbreviations: See note 2 in Figure A.1.

	BOC		EOC	
	Fissile	Fissile	Fissile	Fissile
U-233	581.9		125.9	201
U-235	3.2		13.8	3.2
Total U	4,935		4,291.7	226.9
Pu fissile			50.2	
Total Pu			75.1	
Th		8,828		8,450
THM	4,935	8,828	4,366.8	8,676.9
FP			550	140

Figure A.25. System 4.2: HTGR MEU3(Th), Recycle.

NOTES:

1. MASS FLOW DATA IN kg/0.75 GWe-yr
2. DATA NORMALIZED FROM A 1332 MWe REACTOR, 1.07 yr CYCLE
3. A TRUE EQUIL. CYCLE IS NEVER ATTAINED; THE BRED U-235 IS CONSTANTLY BUILDING UP

U-233
U-235
TOTAL U
Pu FISSILE
TOTAL Pu
Th
THM

	BOC			EOC		
	23R	25R	M	23R	25R	CREDIT
U-233	268.4			277.2		
U-235	30.8	32.7	224.5	34.4	36	5.5
TOTAL U	410.1	74.4	241.4	436.2	80.4	48.0
Pu FISSILE				1.2		
TOTAL Pu				5.3		
Th		10407		9843		
THM		11132		10411		

Figure A.26. System 4.3: HTGR HEU5(Th), Recycle.

NOTES:

1. MASS FLOW DATA IN kg/0.75 GWe-yr
2. DATA NORMALIZED FROM A 1332 MWe REACTOR, 1.07 yr CYCLE
3. A TRUE EQUIL. CYCLE IS NEVER ATTAINED; THE BRED U-235 IS CONSTANTLY BUILDING UP

U-233
U-235
TOTAL U
Pu FISSILE
TOTAL Pu
Th
TBM

BOC			EOC		
23R	25R	M	23R	25R	CREDIT
268.4			277.2		
30.8	32.7	224.5	34.4	36	5.5
410.1	74.4	241.4	436.2	80.4	48.0
				1.2	
				5.3	
	10407			9843	
	11132			10411	

Figure A.27. System 4.4: HTGR HEU5(Th), Recycle (Spiked).

NOTES:

1. MASS FLOW DATA IN kg/0.75 GW-yr
2. DATA NORMALIZED FROM A 1332 MWe REACTOR, 1.07 YEAR CYCLE

U-233
U-235
U
Pu FISSILE
Pu
Th
THM

	BOC		EOC	
	RECY	MAKEUP	RECY	NON RECY
U-233	99.3		99.1	8.4
U-235	22.9	393.5	841.5	31.7
U	841.5	1987.3	841.5	1566
Pu FISSILE			27.2	
Pu			51.8	
Th	3905		3649	
THM	6733.6		6109	

Figure A.28. System 4.5: HTGR MEU5(Th), Recycle (Denature in situ).

Figure A.29. System 4.6: HTGR HEU3(Th), Recycle (Spiked).

Figure A.30. System 4.7: HTGR Pu(Th), Recycle.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. NASAP data base, average for years 20, 21, and 22, normalized from 1,120-MWe reactor.
3. Protactinium assumed to decay to U-233.
4. Abbreviations: See note 2 in Figure A.1.

Th
U-233
U-235
Total U
Pu fissile
Total Pu
THM
FP

BOC			EOC			Total
Core	Axial	Radial	Core	Axial	Radial	Blanket
	8,332	11,567		8,075	11,350	19,425
18.9			10.3	227.7	198	425.7
7,658.9			7,019	0.06	0.03	0.09
1,288.6			1,142.9	230	200	430
1,666.2			1,566.2			
9,328			8,585	8,305	11,550	19,855
			872	22.8	16.2	39

Figure A.31. System 5.1: GCFR Pu(U)/Th.

NOTES

1. MASS FLOWING IN kgs PER 0.75 GWe-yr

U-235
TOTAL U
Pu FISSILE
TOTAL Pu
THM
FP-

BOC		EOC	
CORE	BLANKET	CORE	BLANKET
19.3	25.1	10.4	27.2
9630.3	16679.3	3870.4	16265.6
1351.5		1279.7	339.1
1728.6		1729.7	350.3
11359	16679.3	10,600.1	16615.9
		758.7	64.3

Figure A.32. System 6.1: LMFBR Pu(U)/U.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. NASAP data base, average for years 22, 23, and 24, normalized from 0.70 capacity factor.
3. Axial, radial, and internal blankets of UO₂.
4. Abbreviations: See note 2 in Figure A-1.

U-235
Total U
Pu fissile
Total Pu
THM
FP

BOC		EOC	
Core	Blankets	Core	Blankets
20.1	51.3	12.9	41.2
9,269	23,587	8,724	22,970.5
1,693.6		1,492.4	488.6
2,190.3		2,043.2	511.5
11,459.3	23,587	10,767	23,482
		634.2	108.8

Figure A.33. System 6.2.1: LMFBR Pu(U)/U (Spiked) Heterogeneous.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Abbreviations: See note 2 in Figure A.1.

U-235
Total U
Pu fissile
Total Pu
THM
FP

BOC		EOC	
Core	Blankets	Core	Blankets
19.3	25.1	10.4	27.2
9,630.3	16,679.3	8,870.4	16,265.6
1,351.5		1,279.7	339.1
1,728.6		1,729.7	350.3
11,359	16,679.3	10,600.1	16,615.9
		~758.7	~64.3

Figure A.34. System 6.2.2: LMFBR Pu(U)/U (Spiked) Homogeneous.

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Figure A.35. System 6.3.1: LMFBR Pu(U)/Th (Spiked) Heterogeneous.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Protactinium assumed to decay to U-233.
3. Abbreviations: See note 2 in Figure A.1.

BOC		EOC	
Core	AB&RB	Core	AB&RB
Th			14,792
U-233			327.2
U-235		10.2	333.3
Total U	19.2	8,847.9	
Pu fissile	9,620.3	1,286	
Total Pu	1,359.4	1,738.1	
THM	1,738.8	10,586.7	15,125.3
FP	11,359.1	771.7	51

Figure A.36. System 6.3.2: LMFBR Pu(U)/Th (Spiked) Homogeneous.

Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Protactinium assumed to decay to U-233.
3. Abbreviations: See note 2 in Figure A.1.

Th
 U-233
 U-235
 Total U
 Pu fissile
 Total Pu
 THM
 FP

BOC		EOC	
Core	AB&RB	Core	AB&RB
9,784.4	17,054.9	9,080	16,689.1
—	—	495.5	319.0
—	—	0.95	—
—	—	516.3	323.9
1,696.3	—	1,057.7	—
2,169.7	—	1,571.8	—
11,954.1	17,054.9	11,168.1	17,013
—	—	785.6	43.3

Figure A.37. System 6.4: LMFBR Pu(Th)/Th (Spiked).

NOTES:

1. MASS FLOWS IN kgs PER 0.75 GWe-yr
2. PROTACTINIUM ASSUMED TO DECAY TO U-233

Th
U-233
U-235
TOTAL U
Pu FISSILE
TOTAL Pu
THM
FP

BOC		EOC	
CORE	AB + RB	CORE	AB + RB
	16533.6		16241
1204.4		645.9	259.7
70.0		72.2	0.1
12556.2		11243.7	363.3
		498.4	
		532.9	
12556.2		11776.6	16504.3
		754.3	30.6

Figure A.38. System 6.5: LMFBR MEU3(Th)/Th.

NOTES:

1. MASS FLOW DATA NOT AVAILABLE

U-235
TOTAL U
Pu FISSILE
TOTAL Pu

FP

BOC		EOC	
CORE	BLANKET	CORE	BLANKET

Figure A.39. System 6.6: LMFBR Pu(U)/U (Metallic Fuel).

NOTES:

1. MASS FLOWS IN kg/0.75 GWe-yr
2. NORMALIZED FROM A 1.27 GWe REACTOR

U-235
TOTAL U
Pu FISSILE
TOTAL Pu
THM
FP

BOC	EOC
725.4	184.5
26866	25722
	200.7
	282.1
26866	26017
	830.7

Figure A.40. System 7.1: SSCR LEU, OT.

NOTES:

1. MASS FLOWS IN kg/0.75 GWe-yr
2. NORMALIZED FROM A 1.26 GWe REACTOR

	BOC	EOC
Th	18299	17847
U-233	715.6	454.0
U-235	76.1	73.2
TOTAL U	6344	5819.8
Pu FISSILE		75.1
TOTAL Pu		100.1
THM	24643	23795
FP		840.7

Figure A-41. System 7.2: SSCR MEU3(Th), Recycle.

NOTES:

1. MASS FLOWS IN kgs PER 0.75 GW₃-yr
2. PROTACTINIUM ASSUMED TO DECAY TO U-233

	BOC	EOC
Th	19139.1	18706
U-233	346.5	333.1
U-235	545.9	247.5
TOTAL U	5301.4	4831.2
Pu FISSILE		64.7
TOTAL Pu		89.4
Np-237		0.4
THM	24440.5	23652.2
FP		765.4

Figure A.42. System 7.3: SSCR MEU5(Th), Recycle.

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Appendix B

MASS FLOW DATA BASE

This appendix presents a summary of the NASAP alternative fuel cycle mass flows. For completeness, all fuel cycles considered by NASAP are included, whether or not further detailed assessments were performed. Table B.1 presents fuel utilization aspects of thermal reactor fuel cycles, and Table B.2 presents similar information for fast reactors.

These mass flow data are presented on a 75% capacity factor basis. In predicting resource requirements for scenarios calculated on a 70% capacity factor basis, the data base annual equilibrium requirements are multiplied by 70/75, and the initial core charge and final core discharge values are unaltered. The rationale for this procedure is that utilities load fuel into reactors under the assumption that they will achieve high capacity factors, and many power reactors do, in fact, operate at 75% capacity factor or above. The industry average is brought down to 70% because some plants operate at low capacity factors or are shut down for extended periods because of unexpected technical difficulties.

The tables presented in this appendix contain some categories that have been defined in different ways by different authors. Hence, a short description of some of the assumptions used in deriving these tables will be given.

The external source referred to in the Table B.1 entry "Fissile material requirements from an external source" can be either a stockpile, the discharge from another reactor, or U-235 that has been mined. Fissile material whose origin is a previously discharged batch from the source reactor is not included. Tables B.1 and B.2 show resource requirements assuming a 1% fabrication loss, a 0.5% conversion loss, and a 1% reprocessing loss. Hence, Tables B.1 and B.2 show the actual mass flows required in a nuclear economy.

In Tables B.1 and B.2, equilibrium cycle has been assumed to be the charge, discharge, and make-up material associated with the batch loaded in the 22nd year of reactor operation.

Special attention should be given to the columns in Table B.1 labeled "30-year cumulative discharge of fissile material not recycled into source reactor." Since it has been assumed that a 2-year reprocessing-refabrication lag time is required for recycled fuels, the material discharged in years 28,

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29, and 30 cannot be recycled into the source reactor. Hence, the discharge from these years is included in this column along with the discharged material from the previous 27 years that was not recycled.

Since material from years 28, 29, and 30 cannot be recycled into the source reactor, the 30-year cumulative U_3O_8 and SWU requirements do not include a discharge credit for these years. The value of the uranium discharged in years 28 through 30 is shown in a separate table entry labeled "final core + ex-core credit."

No neutronic penalty due to the presence of U-236 has been assumed in calculating the credit for the discharged uranium. In all thorium cycles, the thorium is assumed to be stored for 20 years and then recycled. Twenty years corresponds to 10 half-lives of Th-228. The equilibrium annual ThO_2 requirement is the quantity of ThO_2 charged in the 22nd year of reactor operation.

This appendix also presents a summary of the mass flows for all reactor/fuel cycles assessed in Volume III, presented on a constant 70% capacity factor basis. These data appear in Tables B.3 through B.6.

Table B.3 shows the U_3O_8 requirements for the reactors and fuel cycles. The table includes 30-year lifetime requirements as well as annual equilibrium requirements. Initial core charge requirements and final core discharge credits are also reported, because these values can vary significantly from the annual equilibrium charge and discharge values. Also contained in Table B.3 are the initial and equilibrium enrichments.

Enrichment tails in Table B.3 are assumed to be 0.2% U-235. A 1.0% material loss due to the fabrication process was included in the charge requirements. Also, a 1.0% material loss due to reprocessing was deducted from the discharge credits. A 0.5% material loss due to the U_3O_8 to UF_6 conversion process was included in the charge requirements for those fuel cycles that require enriched uranium.

Several of the fuel cycles that are listed in Table B.3 do not employ annual refueling. However, to aid in the comparison of various fuel cycles, an annual equilibrium charge requirement and discharge credit were defined as the equilibrium charge (or discharge) divided by the equilibrium cycle length. The values of the equilibrium charge and discharge are defined to be the average over all years of reactor operation except for the first and

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final years. Those fuel cycles that do not refuel annually have been footnoted. Since the initial core and final core usually contain more than one fuel batch, the initial core and final core are not annualized.

The 30-year U_{308} discharge includes the final core discharge credit. Hence, the 30-year net U_{308} use includes a credit for the final core.

Table B.4 shows the separative work requirements, and Tables B.5 and B.6 show the fissile mass flows for the reactors of Table B.3 using the same assumptions for reactor capacity factor and fuel-cycle processing losses. Table B.5 reports the initial core charge and final core discharge values. Table B.6 reports the annual average equilibrium mass flows. Both Tables B.5 and B.6 report U-233, U-235, fissile plutonium, and total fissile material use. Discharged Pa-233 is included with discharged U-233. Since Pa-233 has a relatively short half-life (27 days), after 1 year nearly all (99.99%) of the protactinium will have decayed to U-233 prior to reprocessing.

DATA BASE METHODOLOGY

The following paragraphs describe the methodology and abbreviations used in naming the reactors and fuel cycles studied in the NASAP program. These abbreviations are used in the tables appearing in Appendix B and in a simplified form, given in the introduction to the appendixes, elsewhere in the NASAP report.

All NASAP fuel cycles are described with a four-part descriptor. The descriptor has the form: reactor type -- reactor-type fuel isotopics -- fuel-cycle management description -- physical fuel design characteristics.

The list of reactor types covered here includes:

PWR	pressurized-water reactor
BWR	boiling-water reactor
HWR	heavy-water reactor
HWBR	heavy-water breeder reactor
SSCR	spectral-shift-controlled reactor
HTGR	high-temperature gas-cooled reactor
LWPR	light-water prebreeder reactor
LWBR	light-water breeder reactor

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LMFBR liquid-metal fast breeder reactor
GCFR gas-cooled fast reactor

The fuel isotopics segment describes which fissile and fertile components are present in the fuel. If fissile uranium isotopes are present in the fuel, then the level of fissile concentration is specified according to the categories described below.

The fuel isotopics segment has the following format: concentration of fissile uranium, fissile isotope(s) [fertile isotope(s)]/axial blanket material (if present)/radial blanket material (if present).

If an internal blanket is present, its description is given in a footnote.

Concentration of fissile uranium is defined as the ratio of the sum of U-233 and U-235 to total uranium. Four categories exist and are listed below:

<u>Fuel type</u>	<u>Abbreviation</u>	<u>Fissile uranium concentration</u>
Natural uranium	N	0.711%
Low-enriched uranium	LE	>0.711-10%
Medium-enriched uranium	ME	>10-20%
Highly enriched uranium	HE	>20%

Even though fuels containing U-233 are not likely to be produced by an enrichment plant, these designations are also applied to U-233 fuels in the interest of simplicity.

Three possible fissile components have been chosen and are listed below:

<u>Material</u>	<u>Abbreviation</u>
Uranium-235	U5
Uranium-233	U3
Fissile Pu (Pu-239 + Pu-241)	Pu

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Two possible fertile components have been chosen and are listed below:

<u>Material</u>	<u>Abbreviation</u>
Uranium-238	U
Thorium	Th

As an example, suppose the fuel loaded into a heterogenous LMFBR was as follows: 17% U-233 and U-235 in uranium and thorium loaded into the core, thorium in the axial blanket, uranium in the radial blanket, and thorium in the internal blanket. The fuel isotopics segment of the fuel-cycle descriptor would be that shown below:

-- ME U5, U3 (U,Th)/Th/U^a --

where a is a footnote which would state that the internal blanket is composed of thorium. As a second example, consider a PWR operated with a plutonium-uranium fuel cycle (commonly referred to as a Pu-U burner). The fuel isotopics segment of this fuel-cycle descriptor would be that shown below:

-- Pu (U) --

The fuel-cycle management segment of the fuel-cycle descriptor contains a list of abbreviations that describes fuel management options and fuel processing options. No specification is listed if the fuel cycle is a once-through fuel cycle and if the discharge exposure is unique to the fuel cycle. Several PWR fuel cycles have been examined for two levels of discharge exposure (30,000 MWD/t and 50,000 MWD/t).

The format of the fuel-cycle management description is abbreviation 1, abbreviation 2, abbreviation 3, etc.

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<u>Options</u>	<u>Abbreviation</u>
Recycle	
U-235 only	RU5
All fissile uranium	RUF
Plutonium only	RPu
All fissile uranium and plutonium	RUPu
Short-term improvement	STI
18-month refueling	F18
Semiannual refueling	FS
Extended exposure	X
Spikant	
Co-60 added in reprocessing	SC
Preirradiated	SP
Mechanical source attached	SM
Heterogeneous core	HT
Homogeneous core	HM
Coprocessed plutonium, uranium	CP
15% improvement	15%
30% improvement	30%

As an example, consider an LMFBR that used recycled fissile uranium and plutonium in a homogeneous core and was preirradiated. The fuel-cycle management segment would appear as shown below:

-- RUPu, SP, HM --

Any noteworthy peculiarities of a particular system can be included with a footnote.

Since some fuel cycles are distinguished solely by differences in the physical characteristics of the fuel or fuel element design, a physical fuel design descriptor is required. The format of the descriptor is

-- abbreviation 1, abbreviation 2, etc. --

Options available for this section are listed below.

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<u>Option</u>	<u>Abbreviation</u>
Backfit design	B
Seed-blanket design	SB
Type 1 module	T1M
High-gain converter	HGC
Advanced design	AD
Carbide fuel	C
Metal fuel	M

As an example, a seed blanket design operated with carbide fuel would be specified as shown below:

-- SB, C. --

To show how a fuel-cycle descriptor would be created, consider the following example. A writer wishes to prepare an entry in a table for a homogeneous gas-cooled fast reactor that has 12% U-233 in uranium core fuel and thorium in the radial and axial blanket. The fissile uranium is recycled, but the plutonium is stored. The fuel is a carbide design. The fuel-cycle descriptor would be

GCFR - ME U3(U)/Th/Th - RUF, HM - C.

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Table B.1. Resource Requirements of Various
Reactor/Fuel Cycle Options.^a

	PWR LEU5(U)	PWR LEU5(U)-X	PWR LEU5, Pu (U)-RUPu
Reactor power (GWe)	1.270	1.270	1.270
Refueling interval (yr)	1.0	1.0	1.0
Fraction of core replaced per refueling interval	1/3	1/5	1/3
Equilibrium cycle discharge burnup (MWU/tHM)	30360	50650	30360
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	78202/26997	78338/16130	78199/26865
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	3.01	4.30	3.22(0.711) ^h
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	1759	2010	1759
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	---
U-235	822	704	473
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	---
U-235	25694	22315	16342
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^h	---	---	---
U-235	219	117	33 ^t
Pu-fissile ^g	177	118	---
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	7354	4688	1938
Pu-fissile ^g	5486	3781	1455
Pu-total	7543	5699	2408
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	407/338	470/387	407/338
Equilibrium annual ^f	195/159	170/137	115/91
30-yr cumulative	6086/4962	5390/4330	3980/3179
Final core + ex-core credit ^j	0/0	0/0	243/213
Separative work required (tSWU/GWe) ^e			
0.2wt% tails/0.05wt% tails			
Initial core	210/393	260/482	280/391
Equilibrium annual ^f	118/214	118/208	79/140
30-yr cumulative	3637/6622	3650/6466	2673/4798
Final core + ex-core credit ^j	0/0	0/0	85/167
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	---	---	---
Equilibrium annual ^{f,m}	---	---	---
30-yr cumulative	---	---	---

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Table B.1. Continued

	PWR LEU5, Pu(U)-RUPu,X	PWR Pu(Th)-RPu	PWR Pu(Th)
Reactor power (GWe)	1.270	1.270	1.270
Refueling interval (yr)	1.0	1.0	1.0
Fraction of core replaced per refueling interval	1/5	1/3	1/3
Equilibrium cycle discharge burnup (MWD/tHM)	50650	33400	33400
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	78338/16130	73637/24552	73637/24446
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	4.30	0	0
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	2010	---	---
Pu-fissile	---	2489	2489
Equilibrium cycle ^f			
U-233	---	---	---
U-235	503	---	---
Pu-fissile ^l	---	830	1179
30-Yr cumulative			
U-233	---	---	---
U-235	17234	---	---
Pu-fissile ^l	---	28090	36628
Pu-total ^l	---	40010	51819
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	267	288
U-235	9 ⁱ	3	3
Pu-fissile ^g	---	---	427
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	8394	8810
U-235	464	83	102
Pu-fissile ^g	690	5121	14980
Pu-total	1270	11784	30709
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	470/387	---	---
Equilibrium annual ^f	126/98	---	---
30-yr cumulative	4285/3368	---	---
Final core + ex-core credit ^j	k	---	---
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	260/482	---	---
Equilibrium annual ^f	97/170	---	---
30-yr cumulative	3154/5577	---	---
Final core + ex-core credit ^j	k	---	---
ThO ₂ requirement (st/GWe) ^{d, o}			
Initial core	---	89	89
Equilibrium annual ^{f, m}	---	27	29
30-yr cumulative	---	648	926

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Table B.1. Continued

	PWR Pu(Th)-RPu,X	PWR Pu(Th)-X	PWR Pu(U)-RPu
Reactor power (GWe)	1.27	1.27	1.27
Refueling interval (yr)	1.0	1.0	1.0
Fraction of core replaced per refueling interval	1/5	1/5	1/3
Equilibrium cycle discharge burnup (MWD/tHM)	55515	55515	30400
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	73638/14674	73638/14671	80937/26981
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	0	0	0.711
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	---	---	566
Pu-fissile	2733	2733	1622
Equilibrium cycle ^f			
U-233	---	---	---
U-235	---	---	179
Pu-fissile ^l	812	1222	395
30-Yr cumulative			
U-233	---	---	---
U-235	---	---	5801
Pu-fissile ^l	28254	37912	15055
Pu-total ^l	42218	63813	21310
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	197	204	---
U-235	3	4	110
Pu-fissile ^g	---	495	---
30-Yr cumulative discharge of ^d fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	6180	6275	---
U-235	95	103	3742
Pu-fissile ^g	5915	16952	4782
Pu-total	15659	41813	8945
U ₃ O ₈ requirement (st/GWe) ^d 0.2wt% tails/0.05wt% tails			
Initial core	---	---	103/103
Equilibrium annual ^f	---	---	33/33
30-yr cumulative	---	---	1060/1060
Final core + ex-core credit ^j	---	---	0/0
Separative work required (tSWU/GWe) 0.2wt% tails/0.05wt% tails			
Initial core	---	---	0/0
Equilibrium annual ^f	---	---	0/0
30-yr cumulative	---	---	0/0
Final core + ex-core credit ^j	---	---	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	88	88	---
Equilibrium annual ^{f,m}	15	16	---
30-yr cumulative	399	551	---

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Table B.1. Continued

	PWR Pu(U)-RPu,X	PWR MEU3(U,Th)-RUF	PWR MEU3(U,Th)-RUF,X
Reactor power (GWe)	1.27	1.27	1.27
Refueling interval (yr)	1.0	1.0	1.0
Fraction of core replaced per refueling interval	1/5	1/3	1/5
Equilibrium cycle discharge burnup (MWD/tHM)	50650	33400	55515
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	80940/16121	73671/24554	73651/14681
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	0.711	0.200	0.200
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	1904	1905
U-235	562	28	27
Pu-fissile	1783	---	---
Equilibrium cycle ^f			
U-233	---	316	346
U-235	101	---	---
Pu-fissile ^l	491	---	---
30-Yr cumulative			
U-233	---	12258	13097
U-235	3510	52	44
Pu-fissile ^l	18868	---	---
Pu-total ^l	28034	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	56	---	---
Pu-fissile ^g	---	66	54
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	2509	2284
U-235	1890	285	430
Pu-fissile ^g	6522	2011	1644
Pu-total	13475	2673	2330
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	103/103	0/0	0/0
Equilibrium annual ^f	18/18	0/0	0/0
30-yr cumulative	641/641	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	---	73	73
Equilibrium annual ^{f,m}	---	22	12
30-yr cumulative	---	527	315

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Table B.1. Continued

	PWR MEU5(U,Th)-RUF	PWR LEU5(U)-15%	PWR LEU5(U)-30%
Reactor power (GWe)	1.27	1.27	1.27
Refueling interval (yr)	1.0	1.0	1.0
Fraction of core replaced per refueling interval	1/3	1/5	1/6
Equilibrium cycle discharge burnup (MWD/tHM)	33400	33400	33400
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	73656/23746	78199/16185	78199/13497
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	20.0	4.15	4.00
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	2637	1759	1929
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	---
U-235	422	682	548
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	---
U-235	16790	21533	17831
Pu-fissile ^l	---	---	---
Pu-total ^c	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	---	108	54
Pu-fissile ^g	63	124	98
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	1805	---	---
U-235	1807	4544	2742
Pu-fissile ^g	1893	4087	3334
Pu-total	2618	6184	5423
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	664/517	407/338	450/371
Equilibrium annual ^f	107/86	165/132	132/106
30-yr cumulative	4235/3288	5190/4176	4289/3457
Final core + ex-core credit ^j	480/350	0/0 ^j	0/0 ^j
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	600/990	208/391	243/452
Equilibrium annual ^f	104/170	113/199	89/158
30-yr cumulative	3916/6432	3473/6167	2832/5045
Final core + ex-core credit ^j	334/617	0/0 ^j	0/0 ^j
ThO ₂ requirement (st/GWe) ^{d, a}			
Initial core	77	---	---
Equilibrium annual ^{f, m}	23	---	---
30-yr cumulative	545	---	---

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Table B.1. Continued

	PWR LEU5, Pu(U)-RUPu, SC, CP	PWR LEU5(U)-F18	PWR LEU5(U)-F18, X
Reactor power (GWe)	1.27	1.27	1.27
Refueling interval (yr)	1.0	1.5	1.5
Fraction of core replaced per refueling interval	1/3	1/2	1/2
Equilibrium cycle discharge burnup (MWD/tHM)	30400	30868	46824, 62432 ^g
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	78199/27114	78199/39666	78198/23534
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	2.98	3.20, 3.40 ^p	4.79
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	1759	1584	2041
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	---
U-235	521	920	763
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	---
U-235	17383	27181	23760
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	48	236	158
Pu-fissile ^g	---	174	123
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	1543 ⁿ	7533	4884
Pu-fissile ^g	880 ⁿ	5160 ⁿ	3797 ⁿ
Pu-total	1495 ⁿ	5614 ⁿ	5545 ⁿ
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	407/338	479/393	478/393
Equilibrium annual ^f	128/102	204/166	186/148
30-yr cumulative	4253/3390	6477/5257	5768/4616
Final core + ex-core credit ^j	k	0/0 ^j	0/0 ^j
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	208/391	269/495	268/495
Equilibrium annual ^f	86/156	126/228	133/232
30-yr cumulative	2806/5099	4020/7261	4040/7099
Final core + ex-core credit ^j	k	0/0 ^j	0/0 ^j
ThO ₂ requirement (st/GWe) ^{d, e}			
Initial core	---	---	---
Equilibrium annual ^{f, m}	---	---	---
30-yr cumulative	---	---	---

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Table B.1. Continued

	BWR LEU5(U)	BWR LEU5(U)-STI	BWR LEU5(U)-X
Reactor power (GWe)	1.0	1.0	1.0
Refueling interval (yr)	1.0	1.0	1.0
Fraction of core replaced per refueling interval	1/4	1/4	1/7
Equilibrium cycle discharge burnup (MWD/tHM)	28400	28400	47000
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	122200/30520	122290/30540	122290/17570
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	3.01	2.75	3.81
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	2233	2325	2325
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	---
U-235	934	853	680
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	---
U-235	29318	27054	22050
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	280	220	120
Pu-fissile ^g	180	170	130
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	8600 ⁿ	6660 ⁿ	3570 ⁿ
Pu-fissile ^g	5410 ⁿ	3070 ⁿ	3820 ⁿ
Pu-total	7770 ⁿ	7430 ⁿ	5860 ⁿ
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	505/427	528/445	528/445
Equilibrium annual ^f	222/180	201/164	164/132
30-yr cumulative	6831/5660	6356/5215	5278/4269
Final core + ex-core credit ^j	0/0	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	222/429	240/461	240/461
Equilibrium annual ^f	134/243	116/213	109/193
30-yr cumulative	4100/7478	3607/6638	3389/6069
Final core + ex-core credit ^j	0/0	0/0	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	---	---	---
Equilibrium annual ^{f,m}	---	---	---
30-yr cumulative	---	---	---

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Table B.1. Continued

	LWPR HEU5(Th)-RU5-B	LWPR HEU5,U3(Th)-RUF-SB	LWPR MEU5(U,Th)-RU5-B
Reactor power (GWe)	1.30	1.0	1.295
Refueling interval (yr)	1.0	2.50	1.0
Fraction of core replaced per refueling interval	1/3	1	1/3
Equilibrium cycle discharge burnup (MWD/tHM)	33961	20000	34800
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	k/24102	126304/126278	69343/23368
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	93	93	15.4
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	2411	3840	2885
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	88	---
U-235	914	183	705
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	---
U-235	28917	14616	23860
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	296	---	249
U-235	246	---	---
Pu-fissile ^g	---	---	68
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	9218 ⁿ	3802	7367
U-235	8714 ⁿ	978	3235
Pu-fissile ^g	---	---	2083
Pu-total	---	---	2707
U3O8 requirement (st/GWe) ^d 0.2wt% tails/0.05wt% tails			
Initial core	612	975/754	722/565
Equilibrium annual ^f	232/	26/20	179/138
30-yr cumulative	7341/	3972/3075	6030/4680
Final core + ex-core credit ^j	k	248/192	804/631
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	610	992/1576	626/1044
Equilibrium annual ^f	232/	26/40	169/138
30-yr cumulative	7339/	4047/6429	5604/9189
Final core + ex-core credit ^j	k	252/402	689/1160
ThO2 requirement (st/GWe) ^{d,e}			
Initial core	k	155	61
Equilibrium annual ^{f,m}	29	62	20
30-yr cumulative	625	1415	477

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Table B.1. Continued

	LWPR MEU5(U,Th)-RU5	LWPR MEU5(U,Th)-RU5-TIM	LWBR HEU3,U5(Th)-RUF-SB ⁸
Reactor power (GWe)	1.295	0.721	1.0
Refueling interval (yr)	1.0	3.0	2.50
Fraction of core replaced per refueling interval	1/3	1	1
Equilibrium cycle discharge burnup (MWD/tHM)	26600	11200	15300
Total heavy metal loaded (kg/GWe), 134806/46867 initial core/equilibrium core		263850/264028	171309/171509
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	20	20	---
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	3086
U-235	3687	6179	2593
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	50 ^g
U-235	862	768	---
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	8209
U-235	30017	34961	3388
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	342	433	---
U-235	---	---	---
Pu-fissile ^g	127	80	---
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	10444	12589	8424
U-235	4203	8698	1076
Pu-fissile ^g	3949	2319	---
Pu-total	4765	2754	---
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	928/723	1555/1211	0/0
Equilibrium annual ^f	218/169	195/151	0/0
30-yr cumulative	7589/5888	8495/6599	0/0
Final core + ex-core credit ^j	1049/820	2182/1549	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	839/1384	1406/2319	0/0
Equilibrium annual ^f	208/339	186/303	0/0
30-yr cumulative	7152/11671	7924/12965	0/0
Final core + ex-core credit ^j	933/1556	1936/2922	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	148	295	207
Equilibrium annual ^{f,m}	50	98	83
30-yr cumulative	1158	2378	1890

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Table B.1. Continued

	LWBR HEU3,U5(Th)-RUF-SB ^{2a}	HWBR HEU3,U5(Th)-RUF	LWBR HEU3(Th)-RUF-AD
Reactor power (GWe)	1.0	1.0	1.035
Refueling interval (yr)	2.50	2.50	1.0
Fraction of core replaced per refueling interval	1	1	1/3
Equilibrium cycle discharge burnup (MWD/tHM)	15300	11300	10100
Total heavy metal loaded (kg/GWe), 171508/171562 initial core/equilibrium core		215500/215628	313693/105900
Equilibrium [†] cycle makeup feed enrichment (wt% U-235 in U)	---	---	---
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	3751	2978	4331
U-235	912	724	59
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	56 ^g	5 ^g	---
U-235	10 ^g	5 ^g	---
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	9165	6912	7519
U-235	1609	1159	49
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	---	---	---
Pu-fissile ^g	---	---	---
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	8352	6810	7837
U-235	1017	756	742
Pu-fissile ^g	---	---	---
Pu-total	---	---	---
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	209	266	391
Equilibrium annual ^{f,m}	83	106	132
30-yr cumulative	1895	2420	3055

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Table B.1. Continued

	LWBR HEU3,(Th)-RUF-HGC,B	LWBR HEU3,(Th)-RUF-TIM	LWBR Pu(U)-RPu-HGC
Reactor power (GWe)	1.295	0.711	1.0
Refueling interval (yr)	.50	1.0	1.0
Fraction of core replaced per refueling interval	1/3	1/3	1/4
Equilibrium cycle discharge burnup (MWD/LHM)	11300	10100	29240
Total heavy metal loaded (kg/GWe), 150821/51214 initial core/equilibrium core		340891/115717	108156/27039
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	---	---	0.2
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	1995	4946	---
U-235	27	67	203
Pu-fissile	---	---	4687
Equilibrium cycle ^f			
U-233	75	9	---
U-235	---	---	51
Pu-fissile ^l	---	---	232
30-Yr cumulative			
U-233	6704	8572	---
U-235	---	---	1670
Pu-fissile ^l	---	---	13827
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^o	---	---	---
U-235	---	---	31
Pu-fissile ^g	---	---	---
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^o	5104	8426	---
U-235	592	797	921 ⁿ
Pu-fissile ^g	---	---	3706 ⁿ
Pu-total	---	---	5503 ⁿ
U308 requirement (st/GWe) ^u			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	188	425	---
Equilibrium annual ^{f,m}	127	144	---
30-yr cumulative	2770	3337	---

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Table B.1. Continued

	LWBR HEU3,(Th)-RUF, FS-HGC	LWBR HEU3,(Th)-RUF, X(22)-HGC	LWBR HEU3(Th)-RUF, X(30)-HGC
Reactor power (GWe)	1.0	1.0	1.0
Refueling interval (yr)	0.5	1.0	1.0
Fraction of core replaced per refueling interval	1/3	1/3	1/4
Equilibrium cycle discharge burnup (MWD/tHM)	11310	22620	30160
Total heavy metal loaded (kg/GWe), 109344/36448 initial core/equilibrium core		109344/36448	109344/27336
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	---	---	---
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	1998	2208	2291
U-235	27	30	31
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	87	119	128
U-235	---	---	---
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	7105	6950	6981
U-235	---	---	---
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	22 ^t	22 ^t	22 ^t
Pu-fissile ^g	---	---	---
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	3164	1958	1467
U-235	643	646	626
Pu-fissile ^g	---	---	---
Pu-total	---	---	---
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	136	135	135
Equilibrium annual ^{f,m}	90	45	34
30-yr cumulative	1970	1054	825

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Table B.1. Continued

	LWBR MEU3(Th)-RUF-HGC	LWBR Pu(Th)-RPu-HGC	LWBR LEU3(U, Th)-RUF-HGC
Reactor power (GWe)	1.0	1.0	1.0
Refueling interval (yr)	1.0	1.0	1.0
Fraction of core replaced per refueling interval	1/4	1/4	1/4
Equilibrium cycle discharge burnup (MWD/tHM)	29610	29930	28090
Total heavy metal loaded (kg/GWe), 111365/27841 initial core/equilibrium core		110165/27541	117398/29349
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	0.2	---	0.2
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	2690	---	2590
U-235	75	---	202
Pu-fissile	---	4463	---
Equilibrium cycle ^f			
U-233	208	---	356
U-235	---	---	9
Pu-fissile ^g	---	719	---
30-Yr cumulative			
U-233	9807	---	13580
U-235	---	---	564
Pu-fissile ^g	---	26619	---
Pu-total ^h	---	31940	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	360	---
U-235	14 ^t	5	---
Pu-fissile ^g	86	---	260
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	1625 ⁿ	10793 ⁿ	1007 ⁿ
U-235	367 ⁿ	157 ⁿ	135 ⁿ
Pu-fissile ^g	2602 ⁿ	2030 ⁿ	7900 ⁿ
Pu-total	3332 ⁿ	3576 ⁿ	9780 ⁿ
U3O8 requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	0/0	0/0
30-yr cumulative	0/0	0/0	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
ThO2 requirement (st/GWe) ^{d,e}			
Initial core	113	131	33
Equilibrium annual ^{f,m}	27	32	8
30-yr cumulative	673	791	202

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Table B.1. Continued

	HWR NU5(U)	HWR LEU5(U)	HWR Pu(Th)-RPu
Reactor power (GWe)	1.260	1.260	1.26
Refueling interval (yr)	u	u	u
Fraction of core replaced per refueling interval	---	---	---
Equilibrium cycle discharge burnup (MWD/tHM)	6300	19750	23400
Total heavy metal loaded (kg/GWe), 131790/143600 ^u initial core/equilibrium core		131790/44321 ^u	119946/37400 ^u
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	0.711	1.20	---
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	947	947	---
Pu-fissile	---	---	2827
Equilibrium cycle ^f			
U-233	---	---	---
U-235	1031	540	---
Pu-fissile ^l	---	---	611
30-Yr cumulative			
U-233	---	---	---
U-235	31414	16891	---
Pu-fissile ^l	---	---	21832
Pu-total ^l	---	---	40456
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	316
U-235	348	56	1
Pu-fissile ^g	361	149	---
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	9974
U-235	10855	2254	47
Pu-fissile ^g	10895	4762	3699
Pu-total	14720	8065	15343
U ₃₀₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	173/173	173/173	0/0
Equilibrium annual ^f	188/188	114/101	0/0
30-yr cumulative	5738/5738	3549/3159	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	0/0
Equilibrium annual ^f	0/0	31/65	0/0
30-yr cumulative	0/0	922/1917	0/0
Final core + ex-core credit ^j	0/0	0/0	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	---	---	145
Equilibrium annual ^{f,m}	---	---	44
30-yr cumulative	---	---	1032

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Table B.1. Continued

	HWR MEU3(U,Th)-RUF	HWR MEU5(U,Th)-RUF	HWR LEU5,Pu(U)-RPu
Reactor power (GWe)	1.26	1.26	1.26
Refueling interval (yr)	u	u	u
Fraction of core replaced per refueling interval	---	---	---
Equilibrium cycle discharge burnup (MWD/tHM)	14000	13700	14740
Total heavy metal loaded (kg/GWe), 119938/62472 ^u initial core/equilibrium core		119945/63844 ^u	131790/59387 ^u
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	0.2	20.0	0.711
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	1878	---	---
U-235	28	2444	947
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	120	---	---
U-235	---	166	423
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	7982	---	---
U-235	68	9660	14675
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	---	---	80
Pu-fissile ^g	24	22	---
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	4389	3988	3289
U-235	239	815	---
Pu-fissile ^g	735	652	1197
Pu-total	1013	915	2673
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	615/479	173/173
Equilibrium annual ^f	0/0	42/32	77/77
30-yr cumulative	0/0	2430/1893	2681/2681
Final core + ex-core credit ^j	0/0	190/157	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	556/917	0/0
Equilibrium annual ^f	0/0	35/60	0/0
30-yr cumulative	0/0	2126/3553	0/0
Final core + ex-core credit ^j	0/0	110/203	0/0
ThO ₂ requirement (st/GWe) ^{d, u}			
Initial core	132	137	---
Equilibrium annual ^{f, m}	68	70	---
30-yr cumulative	1519	1564	---

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Table B.1. Continued

	HWR MEU5(U,Th)-RUF,X	HTGR LEU5(U)	HTGR MEU5(U,Th)
Reactor power (GWe)	1.26	1.332	1.332
Refueling interval (yr)	u	1.07	1.0
Fraction of core replaced per refueling interval	---	1/3	1/4
Equilibrium cycle discharge burnup (MWD/tHM)	37000	111000	129700
Total heavy metal loaded (kg/GWe), 119945/23676 ^v initial core/equilibrium core		21794/6454	30637/5332
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	20.0	10.1	19.8
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	3044	1098	1359
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	---
U-235	210	619	576
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	---
U-235	9627	19133	18037
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^o	---	---	64
U-235	---	62	46
Pu-fissile ^g	12	42	29
30-Yr cumulative discharge of ^e fissile material not recycled into source reactor (kg/GWe)			
U-233 ^o	2424	---	2238
U-235	677	2416	2040
Pu-fissile ^g	370	1388	909
Pu-total	676	2678	1763
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	766/597	268/213	342/266
Equilibrium annual ^f	53/41	154/121	145/113
30-yr cumulative	2425/1886	4761/3740	4538/3535
Final core + ex-core credit ^j	157/128	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	693/1443	193/337	309/510
Equilibrium annual ^f	47/78	129/217	131/215
30-yr cumulative	2151/3585	3946/6660	4101/6765
Final core + ex-core credit ^j	95/173	0/0	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	133	---	30
Equilibrium annual ^{f,m}	25	---	3
30-yr cumulative	644	---	119

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Table B.1. Continued

	HTGR MEU5(U,Th)-SA	HTGR MEU5(U,Th)-RU5	HTGR MEU5(U,Th)-RUF ^W
Reactor power (GWe)	1.332	1.332	1.332
Refueling interval (yr)	0.5	1.07	1.07
Fraction of core replaced per refueling interval	1/8	1/3	1/4
Equilibrium cycle discharge burnup (MWD/tHM)	118900	62750	96000
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	28843/2915	37215/11748	30377/7226
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	19.8	19.8	19.8
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	---
U-235	998	1795	1307
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	---
U-235	521	573	370
Pu-fissile	---	---	---
30-Yr cumulative			
U-233	---	---	---
U-235	16395	18929	12500
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	78	164	---
U-235	27	---	---
Pu-fissile ^g	20	57	27
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	2660	5084	687
U-235	1388	1651	671
Pu-fissile ^h	621	1751	807
Pu-total	1329	2564	1516
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	251/196	457/397	329/256
Equilibrium annual ^f	131/102	145/112	94/73
30-yr cumulative	4125/3213	4802/3724	3175/2463
Final core + ex-core credit ^j	0/0	411/326	163/130
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	227/374	408/673	297/490
Equilibrium annual ^f	119/196	138/225	88/145
30-yr cumulative	3728/6150	4516/7378	2940/4830
Final core + ex-core credit ^j	0/0	340/575	118/204
ThO ₂ requirement (st/GWe) ^{d, e}			
Initial core	30	36	30
Equilibrium annual ^{f, m}	4	9	5
30-yr cumulative	150	214	125

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Table B.1. Continued

	HTGR HEU5(U,Th)-RUF, SA ^w	HTGR HEU5(Th)-RUF ^x	HTGR HEU3(Th)-RUF
Reactor power (GWe)	1.332	1.332	1.332
Refueling interval (yr)	0.50	1.07	1.07
Fraction of core replaced per refueling interval	1/8	1/4	1/4
Equilibrium cycle discharge burnup (MWD/tHM)	100000	130200	36800
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	38144/3459	46746/11912	79107/20071
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	19.8	93.1	---
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	---	189
U-235	1582	2018	1905
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	77
U-235	363	223	8
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	5215
U-235	12089	9064	557
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	---	---	---
Pu-fissile ^g	24	1	0.2
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	537	1771	3159
U-235	534	703	500
Pu-fissile ^g	759	41	6
Pu-total	1506	158	68
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	398/310	512/396	0/0
Equilibrium annual ^f	92/72	55/43	0/0
30-yr cumulative	3058/2373	2309/1788	0/0
Final core + ex-core credit ^j	141/112	178/138	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	359/593	508/815	0/0
Equilibrium annual ^f	86/141	55/88	0/0
30-yr cumulative	2825/4646	2301/3686	0/0
Final core + ex-core credit ^j	110/188	176/278	0/0
ThO ₂ requirement (st/GWe) ^{d, e}			
Initial core	38	56	97
Equilibrium annual ^{f, m}	5	13	23
30-yr cumulative	140	333	566

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Table B.1. Continued

	HTGR Pu(Th)-RPu	HTGR MEU3(U,Th)-RUF ^g	SSCR LEU5(U)
Reactor power (GWe),	1.332	1.332	1.260
Refueling interval (yr)	1.07	1.07	1.0
Fraction of core replaced per refueling interval	1/3	1/3	1/3
Equilibrium cycle discharge burnup (MWd/tHM)	100000	50000	30390
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	24593/7175	42474/14724	81577/27079
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	---	---	2.7
Fissile material requirements from an external source ^d (kg/GWe)			
Initial core			
U-233	---	1013	---
U-235	---	---	1681
Pu-fissile	1465	---	---
Equilibrium cycle ^f			
U-233	---	265	---
U-235	---	---	742
Pu-fissile ^l	549	---	---
30-Yr cumulative			
U-233	---	9830	---
U-235	---	---	23205
Pu-fissile ^l	17912	---	---
Pu-total ^l	23589	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	177	---	---
U-235	3	---	186
Pu-fissile ^g	---	50	202
30-Yr cumulative discharge of ^c fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	5362	1917	---
U-235	94	179	6304
Pu-fissile ^g	1324	1542	6266
Pu-total	2472	2288	8744
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	389/127
Equilibrium annual ^f	0/0	0/0	175/143
30-yr cumulative	0/0	0/0	5450/4473
Final core + ex-core credit ^j	0/0	0/0	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	0/0	0/0	186/353
Equilibrium annual ^f	0/0	0/0	100/184
30-yr cumulative	0/0	0/0	3085/5682
Final core + ex-core credit ^j	0/0	0/0	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	28	39	---
Equilibrium annual ^{f,m}	7	11	---
30-yr cumulative	176	269	---

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Table B.1. Continued

	SSCR LEU5(U)-X	SSCR MEU5(U,Th)-RUF	SSCR MEU3(U,Th)-RUF
Reactor power (GWe)	1.260	1.260	1.260
Refueling interval (yr)	1.0	1.0	1.0
Fraction of core replaced per refueling interval	1/5	1/3	1/3
Equilibrium cycle discharge burnup (MWD/tHM)	50650	33400	33400
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	81576/16248	74256/24643	74237/24643
Equilibrium cycle makeup feed enrichment (wt% U-235 in II)	3.80	20.0 (93.0) ^h	0.20
Fissile material requirements from an external source ^a (kg/GWe)			
Initial core			
U-233	---	---	1721
U-235	1696	2580	47
Pu-fissile	---	---	---
Equilibrium cycle ^f			
U-233	---	---	258
U-235	627	379	5
Pu-fissile ^l	---	---	---
30-Yr cumulative			
U-233	---	---	10105
U-235	19872	14644	208
Pu-fissile ^l	---	---	---
Pu-total ^l	---	---	---
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	---	---
U-235	98	---	---
Pu-fissile ^g	137	65	78
30-Yr cumulative discharge of ^a fissile material not recycled into source reactor (kg/GWe)			
U-233 ^b	---	1926	2549
U-235	3946	1484	388
Pu-fissile ^g	4379	1943	2229
Pu-total	6558	2715	2958
U ₃ O ₈ requirement (st/GWe) ^d			
0.2wt% tails/0.05wt% tails			
Initial core	388/325	649/506	0/0
Equilibrium annual ^f	151/122	96/74	0/0
30-yr cumulative	4765/3849	3700/2873	0/0
Final core + ex-core credit ^j	0/0	188/150	0/0
Separative work required (tSWU/GWe)			
0.2wt% tails/0.05wt% tails			
Initial core	189/358	587/968	0/0
Equilibrium annual ^f	100/178	94/151	0/0
30-yr cumulative	3087/5521	3518/5714	0/0
Final core + ex-core credit ^j	0/0	141/245	0/0
ThO ₂ requirement (st/GWe) ^{d,e}			
Initial core	---	78	76
Equilibrium annual ^{f,m}	---	24	23
30-yr cumulative	---	560	546

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Table B.1. Continued

	SSCR Pu(Th)-RPu
Reactor power (GWe)	1.260
Refueling interval (yr)	1.0
Fraction of core replaced per refueling interval	1/3
Equilibrium cycle discharge burnup (MWD/tHM)	33400
Total heavy metal loaded (kg/GWe), initial core/equilibrium core	74212/24643
Equilibrium cycle makeup feed enrichment (wt% U-235 in U)	---
Fissile material requirements from an external source ^d (kg/GWe)	
Initial core	
U-233	---
U-235	---
Pu-fissile	2434
Equilibrium cycle ^f	
U-233	---
U-235	---
Pu-fissile ^l	825
30-Yr cumulative	
U-233	---
U-235	---
Pu-fissile ^l	26970
Pu-total ^l	38071
Annual equilibrium discharge ^f of fissile material not recycled into source reactor (kg/GWe)	
U-233 ^b	287
U-235	3
Pu-fissile ^g	---
30-Yr cumulative discharge of ^e fissile material not recycled into source reactor (kg/GWe)	
U-233 ^b	8876
U-235	104
Pu-fissile ^g	4867
Pu-total	11373
U ₃ O ₈ requirement (wt/GWe) ^d 0.2wt% tails/0.05wt% tails	
Initial core	0/0
Equilibrium annual ^f	0/0
30-yr cumulative	0/0
Final core + ex-core credit ^j	0/0
Separative work required (tSWU/GWe) ^e 0.2wt% tails/0.05wt% tails	
Initial core	0/0
Equilibrium annual ^f	0/0
30-yr cumulative	0/0
Final core + ex-core credit ^j	0/0
ThO ₂ requirement (wt/GWe) ^{d,e}	
Initial core	90
Equilibrium annual ^{f,m}	28
30-yr cumulative	656

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Table B.1. Footnotes

^a75% capacity factor.

^bIncludes Pa-233.

^cAssumes entire core is discharged at the end of the 30th year; also includes discharge from years 28 and 29, which are not recycled due to two-year ex-reactor reprocessing-refabrication time.

^dAssumes 1% loss in fabrication, 1% loss in reprocessing, and 0.5% loss in conversion.

^eAssumes Th stored for 20 years and then recycled.

^fTwenty-second year of reactor operation is assumed to be the equilibrium cycle.

^gNo reprocessing losses assumed in once-through systems.

^hParenthesized value is for the recycle assemblies.

ⁱThis quantity is the discharge from the recycle.

^jNo credit given unless fuel cycle being considered is one which recycles uranium.

^kData not supplied.

^l10% Pu-241 decay loss assumed on those cases with self-generated recycle of plutonium.

^mCharge for year 22.

ⁿValue does not include partially burned batches in the final core.

^pAlternate 3.40 and 3.20.

^qApproximately 1/3 of the core receives the higher burnup.

^rYear 17.5 taken as equilibrium charge year.

^tThis material would be recycled, but the mass flows do not show it.

^uOn-line refueling.

^vThe equilibrium core value reflects the annual charge.

^wMass flows show that bred U-233 and U-235 are recycled and that makeup U-235 is not recycled. However, values quoted in the table assume full recycle of U-235.

^xMass flows assume makeup U-235 makes 2 passes through the reactor and is then credited.

^yMass flows show U-233 being credited. However, values in table reflect full recycle.

^zBreeder is started with prebreeder discharge fuel which contains U-233 and all discharged U-235.

^{aa}Breeder is started with prebreeder discharge fuel which contains U-233 and only that U-235 which is produced by non-fissile absorptions in U-233.

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Table B.2. Fuel Utilization Summary for Fast Reactors

	LMFBR	LMFBR	LMFBR
Reactor power (MWe/MWt)	1000/2740	1000/2740	1000/2740
Breeding ratio	1.44	1.40	1.32
Capacity factor	0.70	0.70	0.70
Cycle length (yr)	1.0	1.0	1.25
Type of core	Heterogeneous	Heterogeneous	Homogeneous
Type of fuel			
Core	(Pu,U)O ₂	(Pu,U)O ₂	(Pu,U)O ₂
Axial blanket	UO ₂	UO ₂	UO ₂
Radial blanket	UO ₂	ThO ₂	UO ₂
Internal blanket	UO ₂	ThO ₂	---
Fissile charge (kg/yr), initial cycle/equil cycle ^a			
Core			
Fissile Pu	4742/1558	5079/1681	2558/1244
Total Pu	6132/2014	6569/2174	3272/1607
U-233	0/0	0/0	0/0
U-235	58/19	57/19	36/18
Fissile discharge (kg/yr), initial cycle/equil cycle ^a			
Core			
Fissile Pu	1477/1368	1571/1468	1214/1166
Total Pu	1950/1852	2075/1964	1632/1576
U-233 ^b	0/0	0/0	0/0
U-235	16/14	16/12	13/10
Axial blanket			
Fissile Pu	22/63	22/63	58/115
Total Pu	22/64	22/64	59/115
U-233 ^b	0/0	0/0	0/0
U-235	13/12	13/12	11/10
Radial blanket			
Fissile Pu	28/177	0/0	64/198
Total Pu	28/144	0/0	65/205
U-233 ^b	0/0	29/177	0/0
U-235	12/8	0/0	16/14
Internal blanket			
Fissile Pu	110/253	0/0	---
Total Pu	112/259	0/0	---
U-233 ^b	0/0	109/254	---
U-235	19/16	0/0	---

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Table B.2. Continued

	LMFBR	LMFBR	LMFBR
Reactor power (MWe/MWt)	1000/2740	1000/2740	1000/2740
Breeding Ratio	1.31	1.22	1.23
Capacity factor	0.70	0.70	0.70
Cycle length (yr)	1.25	1.25	1.50
Type of core	Homogeneous	Homogeneous	Homogeneous
Type of fuel			
Core	(Pu,U)O ₂	(Pu,Th)O ₂	(U3,U8)O ₂ ^c
Axial blanket	ThO ₂	ThO ₂	ThO ₂
Radial blanket	ThO ₂	ThO ₂	ThO ₂
Fissile charge (kg/yr), initial cycle/equil cycle ^a			
Core			
Fissile Pu	2613/1264	3277/1577	0/0
Total Pu	3342/1617	4192/2018	0/0
U-233 ^b	0/0	0/0	2269/1140
U-235	36/18	0/0	132/66
Fissile Discharge (kg/yr), initial cycle/equil cycle ^a			
Core			
Fissile Pu	1260/1172	779/964	264/463
Total Pu	1629/1585	1256/1433	273/495
U-233 ^b	0/0	269/451	814/599
U-235	13/10	0.2/1	66/67
Axial blanket			
Fissile Pu	0/0	0/0	0/0
Total Pu	0/0	0/0	0/0
U-233 ^b	61/114	60/113	49/103
U-235	0/0.1	0/0	0/0
Radial blanket			
Fissile Pu	0/0	0/0	0/0
Total Pu	0/0	0/0	0/0
U-233 ^b	60/184	58/177	41/139
U-235	0/0.1	0/0	0/0.1

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Table B.2. Continued

	FMSR	GCFR	GCFR
Reactor power (MWe/MWt)	1000/2778	1200/3290	1200/3290
Breeding ratio	1.7	1.51	1.54
Capacity factor	0.75	0.75	0.75
Cycle length (yr)	1.0	1.0	1.0
Type of core	---	Homogeneous	Homogeneous
Type of fuel			
Core	UO ₂	(Pu,U)O ₂	(Pu,U)O ₂
Axial blanket	---	ThO ₂	UO ₂
Radial blanket	---	ThO ₂	UO ₂
Fissile charge (kg/yr), initial cycle/equil cycle ^a			
Core			
Fissile Pu	0/0	4439/1512	4288/1452
Total Pu	0/0	5798/2096	5600/1897
U-233 ^b	0/0	0/0	0/0
U-235	8566/58	69/23	69/23
Fissile discharge (kg/Yr), initial cycle/equil cycle ^a			
Core			
Fissile Pu	/513	1466/1420	1424/1383
Total Pu	/594	2373/1819	1858/1872
U-233 ^b	0/0	0/0	0/0
U-235	115	19/13	19/13
Axial blanket			
Fissile Pu	---	0/0	101/273
Total Pu	---	0/0	103/280
U-233 ^b	---	105/273	0/0
U-235	---	0/0.1	26/22
Radial blanket			
Fissile Pu	---	0/0	89/251
Total Pu	---	0/0	89/253
U-233 ^b	---	87/237	0/0
U-235	---	0/0	36/33

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Table B.2. Continued

GCFR	
Reactor power (MWe/MWt)	1200/3290
Breeding ratio	1.40/1.35
Capacity factor	0.75
Cycle length (yr)	1.0
Type of core	Homogeneous
Type of fuel	
Core	(Pu,Th)O ₂
Axial blanket	ThO ₂
Radial blanket	ThO ₂
Fissile charge (kg/yr), initial cycle/equil cycle ^a	
Core	
Fissile Pu	7088/2329
Total Pu	9257/2143
U-233 ^b	0/0
U-235	0/0
Fissile discharge (kg/yr), initial cycle/equil cycle ^a	
Core	
Fissile Pu	2118/1654
Total Pu	2819/2377
U-233 ^b	249/590
U-235	0/0.6
Axial Blanket	
Fissile Pu	0/0
Total Pu	0/0
U-233 ^b	96/261
U-235	0/0
Radial Blanket	
Fissile Pu	0/0
Total Pu	0/0
U-233 ^b	741/189
U-235	0/0

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Table B.2. Footnotes

^aEquilibrium cycle refers to twenty-second year of reactor operation.

^bIncludes Pa-233.

^cU3,U8 is 10% denatured U-233 in U.

Table B.3. U₃O₈ Requirements of Various Reactors^a

Reactor/cycle ^b	Initial core charge (st/GWe)	Final core discharge (st/GWe)	Annual equilibrium ^c U ₃ O ₈ requirement (st/GWe)			30-Year lifetime ^d U ₃ O ₈ requirement (st/GWe)			Initial core U ⁵ /total U (ratio)	Equilibrium makeup U ⁵ /U (ratio)
			Charge	Discharge	Net	Charge	Discharge	Net		
PWR-LEU5(U)	407	232	183	39	144 ^e	5,680	1,340	4,340 ^e	0.0223	0.0301
PWR-LEU5(U)-X	470	322	159	20	139 ^e	5,031	861	4,170	0.0253	0.0430
PWR-Pu(U)-RPu	103	62 ^f	31	15 ^f	16 ^g	989	453 ^f	470 ^g	0.007	0.007
PWR-Pu(U)-X, RPu	103	55 ^f	18	7 ^f	11 ^g	598	226 ^f	332 ^g	0.007	0.007
PWR-Pu(Th)-RPu	0	0	0	0	0 ^h	0	0	0	0	0
PWR-MEU3(U, Th)-RUF	0	0	0	0	0	0	0	0 ^h	0	0
PWR-MEU5(U, Th)-RUF	664	303	176	67	109	5,767	2,242	3,525	0.20	0.20 ⁱ
PWR-LEU5, Pu(U)-RUPu ^j	407	177 ^{f, g}	144	32	112	4,584	1,092	3,492	0.0223	0.0301
PWR-LEU5, Pu(U)-RUPu, X ^j	470	262 ^{f, g, k}	138	17 ^{f, g}	121	4,461	741	3,720	0.0253	0.0430
PWR-LEU(U)-15%	407	317	154	18	136 ^e	4,846	818	4,028 ^e	0.0222	0.0414
PWR-LEU5(U)-30%	449	255	123	7	116 ^e	3,987	441	3,546 ^e	0.0243	0.0400
HWR-LEU5(U)	173	79 ^f	106	0 ^m	106	3,297	0 ^m	3,297	0.007	0.0120
HWR-MEU3(U, Th)-RUF	0	0	0	0	0	0	0	0 ^h	0	0
HWR-MEU5(U, Th)-RUF	615	91	103	48	55	3,555	1,464	2,091	0.20	0.20
HWR-MEU5(U, Th)-RUF-X	767	129	60	9	51	2,514	395	2,119	0.20	
HTGR-MEU5(U, Th)	342	143	135	11	124 ^e	4,235	446	3,789 ^e	0.20	0.20
HTGR-MEU5(U, Th)-RU5 ⁿ	452	290	189	53	135	5,887	1,797	4,090	0.20	0.20 ^o
HTGR-MEU5(U, Th)-RUF	329	135	103	12	91	3,281	460	2,821	0.20	0.20 ^o
HTGR-MEU5(Th)-RUF	512	137	76	14	62	2,670	519	2,151	0.93	0.93 ^o
HTGR-MEU3(Th)-RUF	0	0	0	0	0	0	0	0	0	0
HTGR-MEU3(U, Th)-RUF	0	0	0	0	0	0	0	0 ^h	0	0
HTGR-Pu(Th)-RPu	0	0	0	0	0	0	0	0	0	0
LWPR-MEU5(Th)-RU5-B ^p	612	379	233	63	170	7,341	2,206	5,138	q	0.93
LWPR-MEU5(U, Th)-RU5-B ⁿ	928	716	359	155	204	11,275	5,173	6,104	0.20	0.20
LWPR-MEU5(Th)-RU5-HCC, B ^q	772	q	313	117	196	9,849	3,393	6,456	0.93	0.93
LWPR-Pu(Th)-RPu-B ^q	0	0	0	0	0	0	0	0	0	0
LWPR-MEU5(U, Th)-RU5-TIM	1,555	1,189	490	307	183	14,775	9,476	5,299	0.20	0.20
LWBR-MEU3(Th)-RUF-SB	0	0	0	0	0	0	0	0	0	0
LWBR-MEU3(Th)-RUF-HCC	0	0	0	0	0	0	0	0	0	0
LWBR-MEU3(Th)-RUF, X-HCC	0	0	0	0	0	0	0	0	0	0
LWBR-MEU3(Th)-RUF-TIM	0	0	0	0	0	0	0	0	0	0
LMFBR-Pu(U)/U/U-RPu, HM	0	0	0	0	0	0	0	0 ^h	0	0
LMFBR-Pu(U)/Th/Th-RPu, HM	0	0	0	0	0	0	0	0 ^h	0	0
LMFBR-Pu(Th)Th/Th-RPu, HM	0	0	0	0	0	0	0	0	0	0
LMFBR-MEU3(U)/Th/Th-RUF, HM	0	0	0	0	0	0	0	0 ^h	0	0
LMFBR-Pu(U)/U/U-RPu, HT	0	0	0	0	0	0	0	0 ^h	0	0
LMFBR-Pu(U)/Th/Th-RPu, HT	0	0	0	0	0	0	0	0 ^h	0	0
FMSR-U5(U) ^t	1,848 ^t	q	29 ^u	17 ^u	12	2,690	480	2,210	0.045 ^x	0.007

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Table B.3. Footnotes

^a70% capacity factor assumed for a fuel in which the U-235 to total U fraction is r , the quantity of U_3O_8 required at some tail fraction, t , and some capacity factor, c , is:

$$\text{New } U_3O_8 \text{ requirement} = \text{stated requirement} \cdot \frac{(r - t)}{(r - 0.002)} \cdot \frac{0.00511}{(0.00711 - t)} \cdot \frac{c}{0.70} ; 1\% \text{ reprocessing, } 0.5\% \text{ conversion losses assumed; } 2 \text{ wt } \% \text{ tails assumed.}$$

^bAbbreviations defined in this Appendix.

^cAnnual equilibrium is defined as the average of 29 years of data. Initial core charge and final core discharge not included.

^d30-Year discharge and net values include a credit for final core discharge.

^eSince these are once-through cycles, only the charge value reflects actual U_3O_8 use.

^fNo credit given for U_3O_8 equivalent in recycle assemblies.

^gDischarge recycled uranium has less U-235 concentration than natural uranium.

^hAssume tails uranium used as a dilutant, consequently there is no U_3O_8 requirement.

ⁱMakeup assemblies are 20% enriched. Recycled assemblies require 93% enriched U-235 for topping.

^jSelf-generated recycle.

^kEstimate - final core data not supplied.

^mNo credit given when U-235 content is at or below tails.

ⁿThis reactor is optimized for a symbiotic system of reactors.

^oValue given is for makeup assemblies.

^pLight water breeder reactor Concept 1.

^qData not provided.

^rValues uncertain; mass flows show only partial U-235 recycle.

^sLight water breeder reactor Concept 6.

^tAverage for set of two plants.

^uIncludes initial fueled control assemblies.

^vNatural uranium makeup plus control assemblies.

^wRecovery from spent control assemblies only.

^xFirst plant in set.

Table B.4. Separative Work Requirements of Various Reactors^a
(10³ SWU/GWe)

Reactor/cycle ^b	Initial core charge	Final core discharge	Annual equilibrium requirement ^c			30-Year lifetime requirement ^d		
			Charge	Discharge	Net	Charge	Discharge	Net
PWR-LEU5(U)	210	82	110	4	106 ^e	3,408	192	3,216 ^e
PWR-LEU(U)-X	260	156	109	1	108 ^e	3,424	184	3,240 ^e
PWR-Pu(U)-RPu	0	0	0	0	0	0	0	0
PWR-Pu(U)-X, RPu	0	0	0	0	0	0	0	0
PWR-Pu(Th)-RPu	0	0	0	0	0	0	0	0
PWR-MEU3(U, Th)-RUF	0	0	0	0	0	0	0	0
PWR-MEU5(U, Th)-RUF	660	252	159	56	103	5,272	1,879	3,393
PWR-LEU5, Pu(U)-RUPu ^f	208	73	84	5	79	2,632	208	2,424
PWR-LEU5, Pu(U)-RUPu, X ^f	260	1	94	1	93	2,983	25	2,958
PWR-LEU5(U)-15%	208	142	105	0	105	3,255	142	3,113
PWR-LEU5(U)-30%	243	96	83	0	83	2,660	96	2,564
HWR-LEU5(U)	0	0	30	0	30	861	0	861
HWR-MEU3(U, Th)-RUF	0	0	0	0	0	0	0	0
HWR-MEU5(U, Th)-RUF	556	55	80	31	49	2,867	952	1,915
HWR-MEU5(U, Th)-RUF, X	693	85	51	4	47	2,167	208	1,959
HTGR-MEU5(U, Th)	309	111	122	6	116 ^e	3,848	285	3,563 ^e
HTGR-MEU5(U, Th)-RU5 ^g	408	244	172	41	130	5,347	1,437	3,910
HTGR-MEU5(U, Th)-RUF	297	102	94	7	87	2,848	291	2,557
HTGR-HEU5(Th)-RUF	508	218	63	1	62	2,217	256	1,960
HTGR-HEU3(Th)-RUF	0	0	0	0	0	0	0	0
HTGR-MEU3(U, Th)-RUF	0	0	0	0	0	0	0	0
HTGR-Pu(Th)-RPu	0	0	0	0	0	0	0	0
LWPR-HEU5(Th)-RU5-B ^h	610	373	233	63	170	7,367	2,200	5,167
LWPR-MEU5(U, Th)-RU5-B	839	632	325	132	193	10,276	4,458	5,818
LWPR-HEU5(Th)-RU5-HGC	i	i	i	i	i			
LWPR-Pu(Th)-RPu-B ^j	0	0	0	0	0	0	0	0
LWPR-MEU5(U, Th)-RU5-TIM	1,406	1,062	443	270	174	13,381	8,347	5,034
LWBR-HEU3(Th)-RUF-SB	0	0	0	0	0	0	0	0
LWBR-HEU3(Th)-RUF-HGC	0	0	0	0	0	0	0	0
LWBR-HEU3(Th)-RUF, X-HGC	0	0	0	0	0	0	0	0
LWBR-HEU3(Th)-RUF, TIM	0	0						
LMFBR-Pu(U)/U-RPu, HM	0	0	0	0	0	0	0	0
LMFBR-Pu(U)/Th/Th-RPu, HM	0	0	0	0	0	0	0	0
LMFBR-Pu(Th)/Th/Th-RPu, HM	0	0	0	0	0	0	0	0
LMFBR-MEU3(U)/Th/Th-RUF, HM	0	0	0	0	0	0	0	0
LMFBR-Pu(U)/U/U-RPu, HT	0	0	0	0	0	0	0	0
LMFBR-Pu(U)/Th/Th-RPu, HT	0	0	0	0	0	0	0	0
FMSR-U5(U) ^k	1,300 ^m	i	18 ⁿ	10 ^o	8	1,830	300	1,530

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Table B.4. Footnotes

^a70% capacity factor, 0.2 wt % fabrication loss, 1% reprocessing loss, and 0.5 wt % conversion loss.

^bAbbreviations defined in this Appendix.

^cAnnual equilibrium is defined as the average of 29 years of data. Initial core charge and final core discharge are not included.

^d30-Year discharge and net values include a credit for the final core.

^eSince these are once-through cycles, only the charge value reflects actual SWU use.

^fSelf-generated recycle.

^gThis reactor is optimized for a symbiotic system of reactors.

^hLight water breeder reactor Concept 1.

ⁱData not provided.

^jLight water breeder reactor Concept 6.

^kAverage for set of two plants.

^mIncludes initial fueled control assemblies.

ⁿNatural uranium makeup plus control assemblies.

^oRecovery from spent fuel assemblies only.

Table B.5. Initial Core, Final Core Mass Flows for Various Reactors^a

Reactor/cycle ^b	U-233 (kg/GWe) ^c		U-235 (kg/GWe)		Fissile Pu ^d (kg/GWe)		Heavy metal ^e (t/GWe)
	Initial core charge	Final core discharge	Initial core charge	Final core discharge	Initial core charge	Final core discharge	Initial charge
PWR-LEU5(U)	0	0	1759.4	1066.0	0	429.5	78
PWR-LEU5(U)-X	0	0	2010.2	1418.0	0	485.5	78
PWR-Pu(U)-RPu	0	0	564.7	392.1 ^f	1621.7	2971.6	81
PWR-Pu(U)-X, RPu	0	0	562.4	350.3 ^f	1783.3	4918.6	81
PWR-Pu(Th)-RPu	0	572.6	0	3.9	2488.7	3312.4	74
PWR-MEU3(U, Th)-RUF	1904.0	1608.8	27.9	171.1	0	170.7	74
PWR-MEU5(U, Th)-RUF	0	1085.2	2537.0	1222.0	0	156.3	74
PWR-LEU5, Pu(U)-RUPu ^g	0	0	1759.4	923.8 ^h	0	875.4	78
PWR-LEU5, Pu(U)-RUPu, X ^g	0	0	2010.2	1221.2 ⁱ	0	1242.5	78
PWR-LEU5(U)-15%	0	0	1759.4	1401.6	0	496.6	
PWR-LEU5(U)-30%	0	0	1919.2	1155.3	0	497.3	
HWR-LEU5(U)	0	0	946.7	567.0 ^f	0	367.4	132
HWR-MEU3(U, Th)-RUF	1878.3	1756.6	27.6	97.2	0	33.1	120
HWR-MEU5(U, Th)-RUF	0	1524.4	2443.6	383.4	0	29.6	120
HWR-MEU5(U, Th)-RUF, X	0	1523.0	3044.0	541.0	0	53.0	120
HTGR-MEU5(U, Th)	0	211.6	1359.0	581.1	0	109.3	31
HTGR-MEU5(U, Th)-RU5	0	406.3	1795.1	1161.9	0	160.6	37
HTGR-MEU5(U, Th)-RUF	0	459.2	1307.2	533.3	0	104.2	30
HTGR-MEU5(Th)-RUF	0	1178.9	2017.6	542.6	0	4.2	47
HTGR-MEU3(Th)-RUF	1905.0	2134.0	189.3	338.1	0	0.5	79
HTGR-MEU3(U, Th)-RUF	1362.9	1224.2	0	38.2	0	141.3	42
HTGR-Pu(Th)-RPu	0	422.2	0	5.1	1464.6	1003.7	25
LWPR-MEU5(Th)-RUF-SB	0	1992	3845	490	k	k	126
LWPR-MEU5(U, Th)-RU5-B	0	765.6	3687.1	2857.4	0	301.0	135
LWPR-MEU5(Th)-RU5-HGC, B ^j	0	k	3046.0	k	0	k	135
LWPR-Pu(Th)-RPu-B ^j	0	k	0	k	4464.0	k	151
LWPR-MEU5(U, Th)-RU5-TIM	0	904.7	6178.8	4733.3	0	140.9	264
LWBR-MEU3(Th)-RUF-SB	3085.9	4216.4	2552.9	541.5	0	0	171
LWBR-MEU3(Th)-RUF-HGC	1994.9	k	27.3	k	0	k	151
LWBR-MEU3(Th)-RUF, X-HGC	2291	k	31	k	0	0	151
LWBR-MEU3(U, Th)-RUF-HGC	2690	k	36	k	0	k	151
LWBR-MEU3(Th)-RUF-TIM	4945.5	k	66.7	k	0	k	336
LMFBR-Pu(U)/U/U-RPu, HM ^m	0	0	163.3	131.2	3197.7	3755.7	85
LMFBR-Pu(U)/Th/Th-RPu, HM ^m	0	769.5	44.4	28.0	3265.6	2972.3	80
LMFBR-Pu(Th)Th/Th-RPu, HM ^m	0	1647.1	0	1.3	4096.5	2723.3	88
LMFBR-MEU3(U)/Th/Th-RUF, HM ^m	3402.8	2837.2	193.1	200.7	0	1098.0	107
LMFBR-Pu(U)/U/U-RPu, HT	0	0	233.4	191.0	4741.5	5428.4	96
LMFBR-Pu(U)/Th/Th-RPu, HT	0	986	97	79	5080	4702	94
FMSR-U5(U) ⁿ	0	0	7490.0 ^o	k	0	k	192 ^o

Table B.5. Footnotes

^aAll values reported are for 70% capacity factor; 1% fabrication, 1% reprocessing, and 0.5% conversion losses included.

^bAbbreviations defined in this Appendix.

^cIncludes Pa-233 with discharged U-233.

^dPu-239 plus Pu-241.

^eMetric tons (t) of heavy metal (U + Pu + Th), including blankets where applicable.

^fAll discharge U-235 is contained in uranium rods which have a U-235 content below natural U.

^gSelf-generated recycle.

^h126.3 kg U-235 is contained in recycle fuel assemblies.

ⁱ65.9 kg U-235 is contained in recycle fuel assemblies.

^jLight water breeder reactor Concept 6.

^kData not supplied.

^mIncludes blanket material.

ⁿAverage for set of two plants.

^oIncludes initial fueled control assemblies.

Table B.6. Average Equilibrium Mass Flows for Various Reactors^a

Reactor/cycle ^b	U-233(kg/GWe-yr) ^c			U-235(kg/GWe-yr)			Fissile Pu ^d (kg/GWe-yr)			Heavy metal ^e (t/GWe)	
	Charge	Discharge	Net	Charge	Discharge	Net	Charge	Discharge	Net	Annual equilibrium charge ^f	
PWR-LEU5(U)	0	0	0	770	200	570 ^g	0	161	-161		25
PWR-LEU(U)-X	0	0	0	654	104	550 ^g	0	105	-105		15
PWR-Pu(U)-RPu	0	0	0	168	99 ^h	69	1045	691	353		25
PWR-Pu(U)-X, RPu	0	0	0	95	50 ^h	45	1067	586	481		15
PWR-Pu(Th)-RPu	0	252	-252	0	3	-3	1384	546	737		23
PWR-MEU3(U, Th)-RUF	724	421	303	35	38	-3	0	59	-59		23
PWR-MEU5(U, Th)-RUF	286	310	-24	707	273	433	0	53	-53		23
PWR-LEU5, Pu(U)-RUPu ⁱ	0	0	0	620	183	437	203	232	-30		25
PWR-LEU5, Pu(U)-RUPu, X ⁱ	0	0	0	567	98	478	142	160	-18		15
PWR-LEU5(U)-15%	0	0	0	636	100	536 ^g	0	114	-114		15
PWR-LEU5(U)-30%	0	0	0	512	50	462 ^g	0	90	-90		13
HWR-LEU5(U)	0	0	0	513	54 ^h	459 ^g	0	39	-139		41
HWR-MEU3(U, Th)-RUF	931	819	112	32	36	-4	0	23	-23		58
HWR-MEU5(U, Th)-RUF	649	712	-63	416	199	217	0	20	-20		60
HWR-MEU5(U, Th)-X, RUF	237	254	-17	268	43	225	0	0	-10		22
HTGR-MEU5(U, Th)	0	64	-64	537	46	490 ^g	0	25	-25		4.9
HTGR-MEU5(U, Th)-RU5 ^j	0	151	-151	749	213	536	0	51	-51		10.2
HTGR-MEU5(U, Th)-RUF	87	94	-7	409	49	360	0	23	-23		6.1
HTGR-MEU5(Th)-RUF	231	242	-10	298	73	224	0	1	-1		10.5
HTGR-MEU3(Th)-RUF	510	436	74	67	61	6	0	0.2	-0.2		17.5
HTGR-MEU3(U, Th)-RUF	554	302	252	3	14	-12	0	45	-45		13.4
HTGR-Pu(Th)-RPu	0	159	-159	0	3	-3	682	166	516		6.3
LWPR-MEU5(Th)-RU5-B ^k	0	296	-296	919	249	671	0	0	0		22
LWPR-MEU5(U, Th)-RU5-B	0	312	-312	1419	635	784	0	115	-115		43
LWPR-MEU5(Th)-HGC, B ^m	0	434	-434	1232	462	770	0	0	0		43
LWPR-Pu(Th)-RPu-B ^m	0	326	-326	0	5	-5	1280	619	662		37
LWPR-MEU5(U, Th)-RU5-TIM	0	376	-376	1803	1016	787	0	53	-83		82
LWBR-MEU3(Th)-RUF-SB ^k	1524	1493	31	310	301	9	0	0	0		64
LWBR-MEU3(Th)-RUF-HGC	1257	1187	70	69	77	-8	0	0	0		100
LWBR-MEU3(Th)-RUF, X-HGC	576	456	119	7	28	-21	0	0	0		37
LWBR-MEU3(U, Th)-RUF-HGC	696	491	205	9	30	-21	0	38	-88		37
LWBR-MEU3(Th)-RUF-TIM	1595	1581	14	59	70	-11	0	0	0		108
LMFBR-Pu(U)/U/U-RPu, HM ⁿ	0	0	0	52	34	18	1253	1455	-202		26
LMFBR-Pu(U)/Th/Th-RPu, HM ⁿ	0	283	-283	18	10	8	1260	1169	91		24
LWFBP-Pu(Th)Th/Th-RPu, HM ⁿ	0	715	-715	0	1	-1	1570	934	616		27
LMFBR-MEU3(U)/Th/Th-RUF, HM ⁿ	1128	930	298	65	66	-1	0	443	-443		27
LMFBR-Pu(U)/U/U-RPu, HT	0	0	0	54	51	3	1557	1802	-245		33
LMFBR-Pu(U)/Th/Th-RPu, HT	0	378	-378	32	24	8	1681	1497	184		32
FMSR-U5(U) ^o	0	0	0	147 ^p	44 ^q	103	0	560	-580		8.1 ^q

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Table B.6. Footnotes

^aAverage equilibrium is defined as the average mass flow during 29 years of operation. Initial core charge and final core discharge are not included. 1% fabrication, 1% reprocessing, and 0.5% conversion losses included; 70% capacity factor charge and discharge values include both makeup and recycled fuel.

^bAbbreviations defined in this Appendix.

^cIncludes Pa-233 with discharged U-233.

^dPu-239 plus Pu-241

^eMetric tons (t) of heavy metal (U + Pu + Th) including blankets where applicable.

^fActual annual charge, fabrication losses not included.

^gSince this is a once-through cycle, charge U-235 alone reflects actual reactor use.

^hU-235 concentration in discharge fuel is below natural uranium concentration (0.711 wt %).

ⁱSelf-generated recycle.

^jThis reactor is optimized for a symbiotic system of reactors.

^kLight water breeder reactor Concept 1.

^mLight water breeder reactor Concept 6.

ⁿValue given includes both core and blankets.

^oAverage for set of two plants.

^pNatural uranium makeup and control assemblies.

^qReuse of spent control assemblies as fuel assemblies.

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APPENDIX C

NASAP DATA BASE FOR U.S. ELECTRICAL UTILITY
INDUSTRY CONDITIONS

The data base used for assessments in the NASAP Final Report is presented in a series of tables, each accompanied by a brief description of the background. This data base evolved during the NASAP study and served as a mechanism for assuring consistency, as well as facilitating independent review of the underlying program assumptions.

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BASIS OF NASAP DATA BASE

Table C.1. Introduction Dates for Major Alternate
Reactor/Fuel-Cycle Systems

This table specifies the earliest introduction dates for the various reactor/fuel options. The table is based on recommendations of S. M. Stoller Corporation (SMSC), as contained in Volume IV of the NASAP report.

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Table C.1. Introduction Dates for Major
Alternative Reactor/Fuel-Cycle Systems

Reactor/fuel-cycle system	Earliest estimated date that first commercial plant could operate	
	Accelerated RD&D	Risk-balanced RD&D
<u>Light-water-reactor</u>		
<u>modifications</u>		
10-15% retrofittable	1985-1990	1985-1990
20-30% w/nonretrofittable	1990-2000	1995-2005
Light-water-reactor recycle	1990-2000	2000
<u>Advanced converters</u>		
HTGR-DC	2006	2014
HTGR-SC	2001	2005
HWR-LEU-OT	2001	2005
SSCR-OT	1999	2003
<u>Thermal breeder</u>		
LWBR-HGC	2001	2009
<u>Fast breeder reactors</u>		
LMFBR	2006	2014
GCFR	2014	2024

Table C.2. Projected Installed U.S. Nuclear Capacity

Projections of domestic nuclear generating capacity in the U.S. to be used for NASAP evaluations are shown for the years 1985 through 2025. These projections are based on two forecasting methods and cover two time periods, 1985-2000 and 2000-2025. Energy demand projections until 2000 were developed by the DOE Energy Information Administration. Projections for the years 2005-2025 were extrapolated by the DOE Office of Policy and Evaluation. Two demand cases were considered. For the high-trend case, the growth rate of the demand for electricity was assumed to be 3.7% to the year 2000, decreasing to 1.5% by 2025. Nuclear power generation was assumed to be 42% of total in 2000, increasing to 50% by 2025. For the low-trend case, the growth rate of electrical power was assumed to be 2% in 2000, decreasing to zero by 2025. Thirty-four percent of electricity was assumed generated by nuclear power throughout. A more complete discussion of the information in this table is given in Volume I of the NASAP Report.

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Table C.2. Projected Installed
U.S. Nuclear Capacity

Year	Capacity (GWe)		
	Low	Middle	High
1980	62	64	66
1985	100	111	122
1990	157	175	192
1995	200	238	275
2000	255	325	395
2005	275	378	480
2010	295	443	590
2015	305	498	690
2020	315	558	800
2025	320	615	910

Source: Department of Energy
(Energy Information
Administration, Office
of Policy and Evaluation).

Tables C.3. and C.4. Capital Investment Costs

The capital investment costs in Tables C.3 and C.4 include an owner's cost of approximately 7% of direct and indirect costs for nuclear plants and 8-1/2% for coal plants but do not include heavy-water inventory costs applicable to the HWR and SSCR systems. The interest during construction included in Table C.4 is based on the deflated average cost of money of 4.525% and cash flow for ten years lead time for design, licensing, and construction for nuclear plants and eight years for coal plants, equivalent to 20% of the sum of direct, indirect, and owner's costs for nuclear plants and 13% for coal plants. Plant costs were scaled using approximate scaling factors. To determine capital investment cost in dollars per kilowatt of capacity for other unit sizes, the following relationship is used:

$$\text{Cost, \$/kWe} = (\text{reference cost, \$/kWe}) \left(\frac{\text{size, MWe}}{\text{reference size, MWe}} \right)^n,$$

where $n = -0.55$ for nuclear plants and
 $n = -0.35$ for coal plants.

The exponent $n = -0.55$ for nuclear plants is based on data from United Engineers & Constructors, Inc.; the exponent $n = -0.35$ for coal plants has been derived empirically from ORNL's CONCEPT-5 computer code runs. The exponent for nuclear plants from CONCEPT-5 runs is $n = -0.48$.

All capital investment cost estimates given in Tables C.3 and C.4 have been normalized to an estimate of \$800/kWe for a 1000-MWe LWR, including owner's costs and interest during construction (\$665/kWe not including interest during construction).

The bases for various capital investment cost estimates are discussed briefly in the following paragraphs. The 1000-MWe normalized cost estimates are in general based on the recommendations of United Engineers & Constructors, Inc. (UE&C) for DOE energy alternative studies and also generally conform to investment cost estimates developed by the NASAP characterization contractors. However, there are a few exceptions also discussed in the following paragraphs.

The estimate of \$800/kWe for the 1000-MWe LWR is the most recent update by UE&C reflecting increased home office engineering costs that compares with

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\$780/kWe based on calculations made with the CONCEPT-5 computer code utilizing an earlier cost model developed by UE&C. The 800-MWe coal plant estimates are also based on the most recent update by UE&C and are slightly lower than CONCEPT-5 estimates. Capital investment cost estimates for SSCR plants are based on an estimate by Combustion Engineering, Inc. CE estimated that the additional cost of the NSSS, including the D₂O upgrader, is \$25 million for a 1260-MWe SSCR relative to the PWR and that there is no difference in the balance-of-plant cost. The total plant capital investment cost reported by CE for the SSCR is approximately 4% higher than a CE estimate for a 1270-MWe PWR plant.

The normalized capital investment cost estimate for HTGR plants based on UE&C data is 5% higher than the LWR cost estimate. Estimates by General Atomic Company are close to the reference value, after normalization.

The capital investment cost estimates for pressurized HWR plants utilizing slightly enriched uranium fuel reported by UE&C and by Combustion Engineering, Inc., are both approximately 8% higher than comparable LWR plant cost estimates. UE&C has also developed capital investment cost estimates for HWR plants utilizing natural uranium fuel which are approximately 12-15% higher than comparable LWR plant costs.

Reference capital investment cost estimates for GCFR plants are assumed to be 15% higher than for LWR plants. An estimate provided by General Atomic Company was in turn based on an estimate by UE&C. After scaling to 1000-MWe unit size and including 7% owner's cost, the GCFR cost estimate is approximately 10% higher than a comparable LWR cost estimate. The GCFR commercialization study reported by Helium Breeder Associates gives an estimate for a GCFR plant approximately 8% higher than for a comparable LWR plant. The UE&C estimate is approximately 12% higher than the LWR estimate, after normalization. However, it appears that design improvements to meet some of the uncertainties associated with the GCFR will increase estimated costs slightly.

Capital investment cost estimates for LMFBR plants are based on UE&C recommendations, which are reasonably well confirmed by a number of other sources.

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Capital investment cost estimates for LWBR plants are based on studies by Burns and Roe, Inc.

The capital investment cost for the fast mixed-spectrum reactor (FMSR) is assumed to be the same as for the LMFBR.

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Table C.3. Capital Investment Costs Not
Including Interest During Construction
(January 1978 \$/kWe)

Power plant type	Capital cost ^a		
	600 MWe	Reference 1000 MWe	1300 MWe
LWR	880	665	575
SSCR	915	690	595
HTGR	925	700	605
HWR (LEU fuel) ^b	945	715	620
GCFR	1015	765	660
LMFBR	1225	925	800
LWBR (high-gain converter)	905	685	590
LWBR (seed-blanket breeder)	925	700	605
FMSR	1225	925	800

^aIncludes 7% owner's cost during construction for nuclear plants.

^bHeavy water excluded.

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Table C.4. Capital Investment Costs^a Including
Interest During Construction
(January 1978 \$/kWe)

Nuclear				
Power plant type	Capital cost			Range (% of LWR)
	600 MWe	Reference 1000 MWe	1300 MWe	
LWR ^b	1060	800	690	
SSCR	1100	830	720	104-120
HTGR	1110	840	730	100-120
HWR (LEU fuel) ^b	1140	860	740	100-120
GCFR	1220	920	800	115-150
LMFBR	1470	1110	960	125-150
LWBR (high gain converter)	1090	820	710	103-110
LWBR (seed-blanket breeder)	1110	840	730	105-115
FMSR	1470	1110	960	125-150
Coal				
Power plant type	Capital cost			
	600 MWe	Reference 800 MWe	1000 MWe	
With scrubber (high-Btu coal)	630	570	530 $\pm$ 10%	
Without scrubber (low-Btu coal)	540	490	450 $\pm$ 10%	

^a Includes 7% owner's cost during construction for nuclear plants and 8.5% for coal plants.

^b Heavy water excluded.

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Table C.5. Heavy-Water Costs

The heavy-water costs contained in Table C.5 are based on current heavy-water sales prices and an initial heavy-water requirement of 0.75 t/MWe for the HWR and 0.19 t/MWe for the SSCR. If SSCRs and HWRs were to become accepted, new facilities would be required. The cost of D₂O from such facilities needs to be evaluated. An analysis by S. M. Stoller Corporation indicates that D₂O from new facilities may cost from \$110 to \$150 per pound. The unit price of D₂O used in HWRs is higher than for SSCRs, since pure D₂O is needed for the HWR.

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Table C.5. Heavy-Water Costs
(January 1978 \$/kWe)

Power plant type	Initial cost	Range
SSCR	33 ^a	<u>+20%</u>
HWR	160 ^b	<u>+20%</u>

^aBased on \$80/lb D₂O.

^bBased on \$97/lb D₂O.

Table C.6. Operation and Maintenance Costs

Nonfuel operation and maintenance (O&M) cost estimates for LWR and coal plants are based on calculations made with the OMCOST computer code used by ORNL. O&M cost estimates for HTGR, HWR, GCFR, and LMFBR plants are based on cost-estimating guidelines developed at ORNL for use by United Engineers & Constructors, Inc., in the Energy Economic Data Base (EEDB) for the DOE-ETN Program Planning and Analysis Division, which are in process of being documented. The O&M cost estimate for the SSCR plant is based on the LWR plant plus D₂O makeup of 0.4 kg/hr and an added allowance for D₂O upgrading and maintenance difficulties due to the presence of tritium. The O&M cost estimates for LWBR plants are assumed to be the same as for LWR plants. The O&M cost estimates for FMSR plants are assumed to be the same as for LMFBR plants.

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Table C.6. Operation and Maintenance Costs^a
(January 1978 \$/kWe per year)

Power plant type	Fixed	Variable ^b
LWR	13	1
SSCR	14	1
HTGR	12	1
HWR	16	1
GCFR	15	1
LMFBR	17	1
LWBR	13	1
FMSR	17	1
Coal		
With scrubber	10	12
Without scrubber	8	2

^a Based on 100-MWe reactor and 800-MWe coal plant sizes.

^b Must be multiplied by the capacity factor for given year.

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Table C.7. Fabrication Costs

The fabrication unit costs are based on recent ORNL estimates. All costs shown are for a developed industrial operation. First-of-a-kind costs or costs associated with a high-risk industrial operation are not included. The costs are estimated market prices in 1978 dollars. The costs are based on a so-called "typical industry" capitalization. The financial structure used is typical of a large chemical or petroleum company. A 14% return on equity and an 8.3% interest on borrowed money are assumed along with a capitalization ratio (equity/debt) of 65/35. An effective fixed charge rate on capital investment of 22.6% was estimated and used in the calculation of the unit costs. The plant sizes associated with the unit costs are included in the table. The first set of numbers is applicable to the first commercial plant built. A 15-year transition period is assumed until the high capacity industry costs are applicable. Costs for nonspiked plutonium fuels are based on a remotely operated but contact-maintained facility. Spiked and U-233 fuels were assumed to be fabricated in a remotely operated and maintained facility. No differentiation was made as to type of spikant.

In order to obtain costs at plant sizes other than the 2-t/day reference size plant, scale factors were established for conversion of these reference plant costs to other capacities. The equation used for estimating costs as a function of capacity is:

$$C_u = C_o \left(\frac{X_u}{X_o} \right)^Y,$$

where

C_u = cost of unknown plant in given category,

C_o = cost of reference plant in given category,

X_u = capacity of unknown plant,

X_o = capacity of reference plant,

Y = scaling factor for cost category.

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The procedure separates the fabrication cost components into four categories: investment in the facility, investment in equipment, operating costs, and material cost. The scaling factors, Y , for each category are:

$Y = 0.6$ for contact facility costs,

$Y = 0.8$ for remotely operated facility costs,

$Y = 0.7$ for equipment,

$Y = 1.0$ for expendable material and hardware,

$Y = 0.8$ for operating costs.

The LWR low-enriched uranium (LEU) high burnup and 15% U_3O_8 savings fuels are assumed to cost 10% more to fabricate than the standard LWR-LEU fuel. The fuel for the target once-through fuel cycle with its 30% U_3O_8 savings compared with the standard fuel cycle is assumed to cost about 20% more to fabricate than the standard fuel.

The LWBR, prebreeder, and high-gain converter fuel fabrication costs are preliminary estimates. The unit costs for single-fuel pellets are assumed to be the same as those for LWR fuels. Fuels using a duplex pellet are estimated to cost 50% more to fabricate than similar LWR fuels.

The FMSR fuel is assumed to be similar to that of the LMFBR. The unit costs are what would be expected for fabrication of low-enriched uranium LMFBR core fuel assemblies.

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Table C.7. Fabrication Costs
(January 1978 dollars per kilogram of heavy metal)

Reactor/ fuel type	C/HM atom ratio	Developed industry		High-capacity industry	
		Unit cost ^{a-d} (\$/kg)	Plant capacity (tHM/day)	Unit cost ^{a-d} (\$/kg)	Plant capacity (tHM/day)
PWR, SSCR					
LEU5/(U)					
Current		110	6	110	6
15% improved		140	2	120	6
30% improved		160	2	130	6
MEU5(Th)		140	2	120	4
MEU3(U)		630	2	540	4
MEU3(Th)		660	2	570	4
Pu(U)		430	2	370	4
Pu(Th)		440	2	380	4
Pu(U) spiked		670	2	580	4
Pu(Th) spiked		670	2	580	4
HWR					
Natural UO ₂		65	6	65	6
LEU5(U)		75	6	75	6
MEU5(Th)		100	2	80	6
MEU3(U)		530	2	450	4
MEU3(Th)		570	2	490	4
Pu(U)		360	2	310	4
Pu(Th)		370	2	320	4
Pu(U) spiked		570	2	490	4
Pu(Th) spiked		580	2	500	4
HTGR					
LEU5(U)T	450	740	1.0	670	2
MEU5(Th)OT	385	690	1.0	620	2
MEU5(Th)OT	348	640	1.0	580	2
MEU5(Th)	295	610	1.0	550	2
MEU3(Th)	240	1960	0.5	1230	2
MEU3(Th)	195	1130	2.0	970	4
Pu(Th)	375	1670	1.0	1460	2
HEU5(Th)	169	410	1.0	370	2
HEU3(Th)	169	960	1.0	840	2
HEU3(Th)	143	750	2.0	660	4

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Table C.7. (continued)

Reactor/ fuel type	C/HM atom ratio	Developed industry		High-capacity industry	
		Unit cost ^{a-d} (\$/kg)	Plant capacity (tHM/day)	Unit cost ^{a-d} (\$/kg)	Plant capacity (tHM/day)
FBR-oxide ^e					
U5(Th)		300	2	260	4
U3(Th)		1120	2	990	4
Pu(U)		650	2	580	4
Pu(Th)		660	2	590	4
Pu(U) spiked		1060	2	930	4
Pu(Th) spiked		1150	2	1010	4
UO ₂ radial blanket		140	2	130	4
ThO ₂ radial blanket		150	2	140	4
FBR-carbide ^e					
U3(Th)		1070	2	940	4
Pu(U)		630	2	560	4
Pu(Th)		660	2	580	4
Pu(U) spiked		1030	2	900	4
Pu(Th) spiked		1070	2	940	4
UC radial blanket		160	2	140	4
ThC radial blanket		180	2	160	4
FBR-metal ^e					
U3(Th)		1040	2	910	4
Pu(U)		610	2	540	4
Pu(Th)		650	2	580	4
Pu(U) spiked		960	2	850	4
Pu(Th) spiked		1040	2	910	4
U radial blanket		140	2	120	4
Th radial blanket		160	2	150	4
LWBR/prebreeder					
Duplex fuel					
U5(Th)		210	2	180	4
U3(Th)		950	2	850	4
Single fuel					
U5(Th)		140	2	120	4
U3(Th)		630	2	540	4
Pu(Th)		440	2	380	4
Pu(Th) spiked		670	2	580	4
Th		140	2	120	4

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Table C.7. (continued)

Reactor/ fuel type	C/HM atom ratio	Developed industry		High-capacity industry	
		Unit cost ^{a-d} (\$/kg)	Plant capacity (tHM/day)	Unit cost ^{a-d} (\$/kg)	Plant capacity (tHM/day)
FMSR					
LEU5(U)		280	2	240	4

^aThe first (developed industry) cost is for the year of introduction and does not include first-of-a-kind costs; a time span of 15 years is estimated from introduction pricing until high-capacity industry pricing prevails.

^bEffective fixed charge rate on initial capital investment is 22.6%. Fixed charge rate includes return on investment, return of investment, income taxes, property taxes, and interim replacement. Initial capital investment includes direct investment, capitalized interest during construction, and capitalized preoperational costs.

^cEffective full production days per year: contract operation, 260; remote operation, 240.

^dCost uncertainty: U 233 fuels, -10 to +10%; Pu fuels, -25 to +25%; U 233 fuels, -25 to +25%.

^eUnit cost for core fuel assemblies applies to total heavy-metal throughput for core plus axial blanket.

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Table C.8. Reprocessing Costs

Two sets of unit reprocessing costs are given in Table C.8 as representatives of two alternate philosophies on spent-fuel reprocessing, the so-called Allied General Nuclear Services (AGNS) and canyon-type facility costs.

The AGNS-type facility costs are for plants similar to the AGNS plant at Barnwell, South Carolina. These estimates assume current perceived regulations with semiremote maintenance. The direct investment cost of such a facility was estimated at \$1.0 billion with an annual operating cost of \$50 million.

The canyon-type facility costs are based on work by the Savannah River Laboratory (SRL). This facility, besides having fully remote maintenance, is designed in anticipation of much more stringent safeguards and operating criteria than required by current regulations. SRL personnel estimate that the capital investment required for a 1500-tHM/year plant is \$1.6 billion with annual operating costs of \$50 million.

HTGR fuel reprocessing investment and operating costs were obtained by adjusting the estimates in ORNL/TM-6522 in the same proportion as required for agreement of the ORNL LWR-PUREX estimate to the above value for AGNS- and canyon-type plants.

For other fuels and reprocessing methods, the capital cost estimates were obtained from the Alternate Fuel Cycle Evaluation Program. A \$50 million annual operating cost is used throughout. The unit costs were obtained by applying the same financial assumptions and cost estimating procedures as were discussed under Table C.7.

All costs are for a 1500-tHM/year plant size. In order to obtain costs at plant sizes other than this plant size, a cost scale factor as discussed under Table C.7 (applied to capital investment and annual operating costs) of 0.60 should be used for capacities down to 1200 tHM/year and 0.35 below 1200 tHM/year as recommended by ORNL.

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Table C.8. Unit Reprocessing Costs^a
(January 1978 \$)

Reactor/fuel type	Unit cost ^{b,c,d,e} (\$ per kilogram of heavy metal)	
	Semiremote AGNS-type facility	Fully remote canyon-type facility
LWR/SSCR/HWR		
(Pu,U)O ₂	250	370
(Pu,U)O ₂ , spiked	270	410
(MEU,Th)O ₂	270	410
FBR, core + axial blanket		
(Pu,U)O ₂	350	450
(Pu,U,Th)O ₂	390	490
(Pu,U)C	390	500
(Pu,U,Th)C	430	540
(Pu,U) metal	360	460
(Pu,U,Th) metal	400	500
FBR, radial blanket		
UO ₂	270	390
ThO ₂	290	430
UC	300	430
U metal	280	400
Th metal	300	440
HTGR		
C/HM ^f		
295 MEU5(Th)	330	500
195 MEU3(Th)	300	450
375 Pu(Th)	360	550
169 HEU5(Th)	290	440
143 HEU3(Th)	280	420

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Table C.8. Continued

Reactor/fuel type	Unit cost ^{b,c,d,e} (\$ per kilogram of heavy metal)	
	Semiremote AGNS-type facility	Fully remote Canyon-type facility
LWBR		
(U-233,Th)O ₂	270	410
(Pu,Th)O ₂	270	410
Duplex pellet fuels	400	600

^a Does not include spent-fuel shipping costs, waste shipping costs, or waste storage costs.

^b Developed industry costs. These costs do not include first-of-a-kind costs.

^c Effective fixed charge rate on initial capital investment is 22.6%. Fixed charge rate includes return on investment, return of investment, income taxes, property taxes, and interim replacement. Initial capital investment includes direct investment, capitalized interest during construction, and capitalized preoperational costs.

^d Reprocessing plant average annual throughput, 1500 metric tons per year.

^e Cost uncertainties: -25%, +25%.

^f Atom ratio.

Table C.9. Shipping and Waste Storage Costs

This table contains estimates of costs for spent-fuel shipping, reprocessing plant waste shipping, waste disposal, and spent-fuel disposal. The spent-fuel shipping costs are based on recent estimates at ORNL. This estimate updated cask and freight costs and used fuel assembly characteristics consistent with the current NASAP designs. The waste shipping and waste disposal costs are based on estimates given in report ORNL/TM-5976.

Waste includes all waste from the reprocessing plant (high-level waste, cladding hulls, transuranic waste, low-level nontransuranic waste). The estimates in ORNL/TM-5976 were modified to account for inflation between 1976 and 1978 and for an adjustment of the compaction factor for transuranic and low-level nontransuranic wastes to 10:1. The HWR cost estimates were also adjusted to the waste composition of the slightly enriched fuel cycle instead of the natural uranium case used previously. A 180-day cooling period is assumed for fuel shipped to the reprocessing plant.

The \$120/kg LWR spent-fuel permanent storage cost (or disposal cost) was rounded from \$117/kg given as the reference number in a recent DOE study. Comparable numbers for HWRs and HTGRs are not available. The HWR fuel permanent storage costs should be less than for LWR fuels since the fuel exposure is less and there is more heavy-metal mass per unit volume of element. The HTGR fuel, on the other hand, has a higher exposure and more volume per unit heavy-metal content than the LWR fuel. Until better numbers can be obtained, the HWR permanent fuel storage cost is assumed to be about one-half that of the LWR (\$60/kg) and the HTGR permanent storage cost about double that for the LWR (\$240/kg).

The FMSR shipping costs were assumed to be the same as for LMFBR fuel and the permanent storage cost to be that for LWR fuel.

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Table C.9. Shipping and Waste Storage Costs
(January 1978 dollars per kilogram
of heavy metal)

Reactor/fuel type	Spent-fuel shipping costs	Waste shipping costs	Waste storage costs	Permanent spent-fuel storage
LWR, SSCR, LWBR	15	10	50	120
HWR	10	5	25	60
FMSR	90			120
<u>HTGR</u>				
<u>C/HM</u>				
450 LEU5(U)OT	230			240
385 MEU5(Th)OT	200			240
348 MEU5(Th)OT	180			240
295 MEU5(Th)	160	30	60	
195 MEU3(Th)	100	30	60	
375 Pu(Th)	210	30	60	
169 HEU5(Th)	90	30	60	
143 HEU3(Th)	80	30	60	
<u>FBR</u>				
(U,Pu)O ₂	90	25	100	
(U,Th)O ₂	100	25	100	
(U,Pu)C	70	25	100	
(U,Th)C	90	25	100	
(U,Pu)Metal	60	25	100	
(U,Th)Metal	80	25	100	

Cost uncertainty

Shipping:	<u>+40%</u>
Waste storage:	<u>+50%</u>
Permanent storage:	<u>+50%</u>

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Table C.10. Total Nuclear Fuel-Cycle Back-End Cost

The total nuclear fuel-cycle back-end costs are tabulated in this table. The values shown here for recycle fuel are the sum of the reprocessing costs from Table C.8 and the shipping and waste storage costs from Table C.9. The throwaway costs are the sum of the spent-fuel shipping charge and the permanent spent-fuel storage charge.

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Table C.10. Total Nuclear Fuel-Cycle Back-End Cost^a
(January 1978 dollars per kilogram heavy metal)

Reactor/fuel type	Total recovery cost		Throwaway cost
	Semiremote AGNS-type facility	Fully remote canyon-type facility	
<u>LWR, SSCR</u>			135
(U,Pu)O ₂	330	450	
(U,Pu)O ₂ -spiked	350	490	
(MEU,Th)O ₂	350	490	
<u>HWR</u>			70
(U,Pu)O ₂	290	410	
(U,Pu)O ₂ -spiked	310	450	
(MEU,Th)O ₂	310	450	
<u>HTGR</u>			
<u>C/HM^b</u>			
450 LEU5(U)OT			470
385 MEU5(Th)OT			440
348 MEU5(Th)OT			420
295 MEU5(Th)	580	750	
195 MEU3(Th)	490	640	
375 Pu(Th)	690	880	
169 HEU5(Th)	470	620	
143 HEU3(Th)	450	590	
<u>FBR — core + axial blanket</u>			
(Pu,U)O ₂	560	660	
(Pu,U,Th)O ₂	610	710	
(Pu,U)C	580	690	
(Pu,U,Th)C	640	750	
(Pu,U)Metal	540	640	
(Pu,U,Th)Metal	600	700	
<u>FBR — radial blanket</u>			
UO ₂	480	600	
ThO ₂	510	650	
UC	490	620	
ThC	530	680	
U Metal	460	580	
Th Metal	500	640	

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Table C.10. (continued)

Reactor/fuel type	Total recovery cost		Throwaway cost
	Semiremote AGNS-type facility	Fully remote canyon-type facility	
<u>LWBR</u>			
(U-233,Th)O ₂	350	490	
(Pu,Th)O ₂	350	490	
Duplex pellet fuels	480	680	
<u>FMSR</u>			
LEU(U)			210

^a Total back-end costs for reprocessed fuel includes reprocessing cost, spent-fuel shipping cost, waste shipping cost, and waste storage cost. Throwaway cost includes spent-fuel shipping cost plus permanent spent-fuel storage cost.

^b Atom ratio.

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Table C.11. Enrichment Cost and Tails Composition

The post-1978 enrichment costs given in this table assume commercial pricing for enrichment. The \$100/SWU is based on DOE estimates. Current technology consists of the gaseous diffusion and centrifuge methods. Advanced systems consist of lasers, etc. The advanced enrichment processes are assumed to supply feed to the gaseous diffusion or centrifuge plants at an equivalent price of \$43 per pound of U_3O_8 and to operate with a tails composition of 0.05%.

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Table C.11. Enrichment Cost and Tails Composition

Time	Tails composition (weight fraction U-235)	Cost ^a
Current technology ^b		
1978	0.0020	\$75/SWU
Post 1978	0.0020	\$100/SWU
Advanced technology ^c		

^a January 1978 \$.

^b Gaseous diffusion and centrifuge.

^c It is assumed that advanced systems (lasers, etc.) will be used to reduce enrichment plant tails in the stockpile by enriching the tails to natural uranium. The advanced enrichment processes are assumed to supply feed to the gaseous diffusion or centrifuge plants at an equivalent price of \$43 per pound of U₃O₈ and to operate with a tails composition of 0.05%.

Table C.12. Money Costs for Power Plants

The composition of the cost of money for use in NASAP comparative analyses is given in Table C.12. All costs are specified in January 1, 1978, dollars so that inflation is removed from the comparative analyses. The money costs given in this table are representative of a typical U.S. electric utility. The decommissioning cost is based on a recent ORNL analysis. In this analysis a cost equal to 5% of the initial investment is accumulated in a sinking fund over the life of the plant. Both uninflated and actual costs (based on a 5.5%/year average inflation rate) are given, but only the uninflated (deflated) money costs are to be used. The heavy-water charge rate assumes that heavy water will be treated as a nondepreciable asset and subject to property taxes.

Table C.12. Money Costs for Power Plants^{a, b}

Item	Uninflated	Actual ^a
Debt interest, %	2.5	8.1
Equity return, %	7.0	12.9
Debt fraction	0.55	
Equity fraction	0.45	
Effective rate of return, %	4.5	10.3
Effective income tax rate, %	50	
Fixed charge rate on capital, ^c %	10.1	15.8
Heavy-water charge rate, ^c %	10.2	18.6

^a Normal utility practice, including inflation at 5.5% per year.

^b January 1978 \$.

^c Fixed charge rate breakdown:

Item	Power plant		D ₂ O	
	Uninflated (%)	Actual (%)	Uninflated (%)	Actual (%)
Return (effective interest rate)	4.525	10.26	4.5	10.3
Sinking fund depreciation	1.632	0.58		
Income tax	0.75	1.73	3.2	5.8
Property tax	2.50	2.50	2.5	2.5
Property insurance	0.25	0.25		
Interim replacement	0.35	0.35		
Decommissioning sinking fund	<u>0.08</u>	<u>0.14</u>		
Total fixed charge rate	10.1	15.8	10.2	18.6

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Table C.13. Additional Data

This table includes miscellaneous data needed in comparative NASAP analyses. The range of reference U_3O_8 prices reflects uranium produced from various grades of ore. The \$40/lb figure is approximately current market price, while the \$160/lb figure is for uranium produced from Chattanooga ohalco. The UF_6 conversion cost is approximately current market price. The fissile material storage charge is an estimate. This number is being refined in DOE's Alternate Fuel Cycle Evaluation Program. Out-of-reactor times for fuel recycle are estimated at from one to two years. The value of two years was chosen as an attainable figure. The estimated fissile material losses are attainable in current practice. These losses may be reduced in the future due to more stringent safeguards and accountability.

The reference capacity factor of 70% is consistent with those used in INFCE cost analyses.

The range of reference coal prices is approximately the current range for delivered coal in the eastern United States. The heat value for the coal and its heat rate for combustion were obtained from the same source as the data used to obtain the capital costs for coal-fired plants.

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Table C.13. Additional Data

Reference U ₃ O ₈ price, \$/lb	40, 100, 160 ^a
Cost of converting U ₃ O ₈ to UF ₆ , \$/kg U	4
Fissile storage charge after reprocessing, \$/g-year	2 (U-233 + fissile Pu)
Out-of-reactor time (fabrication lead time + reprocessing lag time), years	2
Fissile material losses, %	
Conversion	0.5
Fabrication	1.0
Reprocessing	1.0
Reference capacity factor, %	70
Coal assumptions:	
Reference value, \$/10 ⁶ Btu	0.50, 1.00, 1.50
Heat value, Btu/lb	
Eastern high sulfur	11,000
Western low sulfur	8,200
Heat rate, Btu/kWhr	
Eastern coal (with scrubbers)	9,500
Western coal (without scrubbers)	9,800

^aIn 1978 dollars reflecting uranium produced from various grades of ore.

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Table C.14. U.S. U_3O_8 Price Projections

This table shows the estimated price of U_3O_8 as a function of both year and cumulative demand for two demand scenarios and for high, low, and mid-range supply assumptions. The U_3O_8 price for the high-demand case and middle supply assumptions are given for both a no-net-import and 30%-import case.

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Table C.14. U.S. U₃O₈ Price Projection
(All prices in 1978 \$)

Year	Cumulative demand ^a	U ₃ O ₈ prices for supply assumptions ^b			
		Low	Middle	Middle ^c	High
<u>High demand</u>					
1978	0	20	20	20	20
1995	0.5	65	55	45	50
2003	1.0	90	80	55	70
2008	1.5	120	95	65	75
2013	2.0	160	100	75	80
2017	2.5	170	110	85	85
2021	3.0	180	125	90	85
2024	3.5	190	140	95	90
2027	4.0	200	160	100	90
2030	4.5	215	170	110	95
2033	5.0	230	180	125	100
2036	5.5	260	190	140	105
2039	6.0	260	200	160	110
<u>Low demand</u>					
1978	0	20	20		20
1997	0.5	55	50		45
2006	1.0	60	55		50
2015	1.5	80	60		55
2023	2.0	120	60		55
2032	2.5	160	65		55
2040	3.0	170	75		55

^a Millions of short tons of U₃O₈.

^b Intermediate prices can be calculated using linear interpolation.

^c Assuming a 30% level of imports.

Table C.15. Fissile Plutonium and U-233 Price

This table includes fissile material prices as a function of U_3O_8 ore price which were used by some of the NASAP contractors as baseline values in power generation cost calculations. The values shown are the economic indifference values for recycle in LWR reactors. The economic indifference value for a fissile material is that value which causes the power generation cost of interdependent fuel cycles to be the same.

The economic analysis in NASAP Volume V was based on a procedure where plutonium is valued at the cost of recovery from spent fuel. Other procedures for valuing plutonium are also possible.

The indifference value for fissile plutonium was estimated from fuel-cycle cost equations for LWRs operating on

1. a low-enriched uranium feed (LEU),
2. a plutonium-in-natural-uranium feed (Pu/U).

The LEU fuel cycle is assumed to sell its recovered plutonium to the Pu/U fuel cycle, which also recycles its plutonium. Recovered uranium from the LEU reactor is returned to the diffusion plant for reenrichment. No credit is taken for uranium recovered from the Pu/U fuel cycle.

The two fuel-cycle equations were used to establish both the plutonium price and the fuel-cycle cost. This fuel-cycle cost and plutonium value were then used with the derived fuel-cycle cost equation for an LWR fueled with de-natured U-233 in thorium (MEU3/Th) in order to establish a price for U-233.

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Table C.15. Fissile Plutonium and U-233
Indifference Value Based on
Recycle in LWRs^{a, b}

U ₃ O ₈ price ^c (\$/lb)	Pu(239 + 241) price ^c (\$/g)	U-233 price ^c (\$/g)
40	22	24
60	30	37
80	38	51
100	46	64
120	53	77
160	69	103
200	85	129
260	108	169

^aBasis:

Conversion cost, \$/kg U	4
Enrichment cost, \$/kg SWU	100
Fabrication costs, \$/kg HM	
LEU	110
Pu(U)	370
MEU3(Th)	570
Recovery costs, \$/kg HM	
(U,Pu)O ₂	330
(U,Pu,Th)O ₂	350
Thorium cost, \$/kg	38
Capacity factor, %	70

^b January 1978 \$.

^c Intermediate prices can be calculated using linear extrapolation.

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ACRONYMS

ACRS	Advisory Committee on Reactor Safety
AD	advanced design
ADR	accelerator-driven reactor
AEC	Atomic Energy Commission
AECCL	Atomic Energy of Canada Ltd.
AFCEP	Alternate Fuel Cycle Evaluation Program
AFR	away-from-reactor (storage)
AGNS	Allied General Nuclear Services
AI	Atomics International
AIS	advanced isotope separation
ALARA	as low as reasonably achievable
ANL	Argonne National Laboratory
ANL-W	Argonne National Laboratory — West
ANS	American Nuclear Society
AP&L	Arkansas Power and Light
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
AVLIS	atomic vapor laser isotope separation
AWBA	Advanced Water Breeder Applications
B	backfit design
B&W	Babcock & Wilcox

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BOC	beginning of cycle
BWR	boiling-water reactor
CACS	core auxiliary cooling system
CAHE	core auxiliary heat exchanger
CAMEL	Components and Materials Evaluations Loop
CANDU	Canadian Deuterium Uranium Reactor
CE	Combustion Engineering
CEA	Commissariat a l'Energie Atomique
CHF	critical heat flux
CIP	Cascade Improvement Program
CPDF	Centrifuge Plant Demonstration Facility
CRBRP	Clinch River Breeder Reactor Project
CRCTA	Composite-Reactor Component Test Assembly
CSTF	Containment Systems Test Facility
CUP	Cascade Upgrading Program
DC	direct cycle
DEH	direct electrical heating
DMSR	denatured molten-salt reactor
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
D-T	deuterium-tritium
DU	denatured uranium

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EBR	Experimental Breeder Reactor
ECCS	emergency core-cooling system
EHR	Experimental Hybrid Reactor
EOC	end of cycle
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
ETEC	Engineering Technology Engineering Center
FAB	fabrication option
FBR	fast breeder reactor
FCR	full core reserve
FCTT	Fuel Cladding Transient Test
FFTF	Fast-Flux Test Facility
FGRM	fission-gas release mechanism
FMSR	fast mixed-spectrum reactor
FP	fission products
FRG	Federal Republic of Germany
FSF	fuel storage facility
GAC	General Atomic Company
GCFR	gas-cooled fast reactor
GCRA	Gas-Cooled Reactor Associates
GE	General Electric
GESMO	Generic Environmental Statement for LWR Mixed Oxide Fuel

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GKR	gaseous-core reactor
HBA	Helium Breeders Associates
HBF	Hybrid Blanket Facility
HCDA	hypothetical core-disruptive accident
HEDL	Hanford Engineering Development Laboratory
HEF	Hot Experimental Facility
HEPA	high-efficiency particulate air (filter)
HEU	high-enriched uranium
HGC	high-gain converter
HM	heavy metal
HPFL	High-Performance Fuels Laboratory
HPP	hot pilot plant
HTGR	high-temperature gas-cooled reactor
HTSF	high-temperature sodium facility
HVAC	high-vacuum air cleaning
HWBR	heavy-water breeder reactor
HWR	heavy-water reactor
ID	inside diameter
IDC	interest during construction
IEEE	Institute of Electrical and Electronic Engineers
INEL	Idaho National Engineering Laboratory
IUREP	International Uranium Resource Evaluation Project

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LAFR	Linear Accelerator Fuel Regenerator
LASL	Los Alamos Scientific Laboratory
LEU	low-enriched uranium
LMEC	Liquid-Metal Engineering Center
LMFBR	Liquid-Metal Fast Breeder Reactor
LOCA	loss-of-coolant accident
LSFF	Large Sodium Fire Facility
LWBR	light-water breeder reactor
LWPR	light-water prebreeder reactor
LWR	light-water reactor
MEU	medium-enriched uranium
MFGCR	mixed-flow gaseous-core reactor
MIT	Massachusetts Institute of Technology
MLIS	molecular laser isotope separation
MOD	modified cycle
MOX	mixed oxide
MSBR	molten salt breeder reactor
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
NASAP	Nonproliferation Alternative Systems Assessment Program
NDT	nondestructive testing
NFS	Nuclear Fuels Services
NR	Division of Naval Reactors

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NRC	Nuclear Regulatory Commission
NRTS	National Reactor Testing Station
NSSS	nuclear steam supply system
NURE	National Uranium Resource Evaluation
OPERA	out-of-pile expulsion and reentry apparatus
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
OT	once-through
OTTO	once-through-then-out
PCI	pellet-clad interaction
PCL	power conversion loop
PCRV	prestressed concrete reactor vessel
PHWR	pressurized heavy-water reactor
PIE	postirradiation examination
PNL	Pacific Northwest Laboratory
PRLCDS	Proliferation-Resistant LMFBR Core Design Study
PSP	plasma separation process
PWR	pressurized-water reactor
RD&D	research, development, and demonstration
R&D	research and development
RF	radio frequency
ROI	return on investment

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SAR	Safety Analysis Report
SAREF	Safety Research Experimental Facility
SB	seed blanket
SBTF	Sodium Boiling Test Facility
SC	steam cycle
SGR	self-generated recycle
SLSF	Sodium Loop Safety Facility
SNM	special nuclear material
SPTF	Sodium Pump Test Facility
SSCR	spectral-shift-controlled reactor
st	short ton
SWU	separative work units
t	metric ton
TBP	tributyl phosphate
TD	theoretical density
THORS	Thermal-Hydraulic Out-of-Reactor Safety Facility
THR	Tokamak Fusion-Fission Hybrid Reactor
TMF-ENFP	Ternary Metal-Fueled Electronuclear Fuel Producer
TIM	Type 1 Module
TREAT	Transient Reactor Test Facility
TRU	transuranium

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UCO	uranium oxycarbide
UK	United Kingdom
WARD	Westinghouse Advanced Reactors Division
WCBR	Water-Cooled Breeder Reactor
WM	waste management option
X	Extended burnup
ZPPR	Zero-Power Plutonium Reactor

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GLOSSARY

Accelerator -- A device for imparting very high velocity to charged particles such as protons.

Advanced converter reactor -- Usually refers to a converter reactor other than a light-water reactor (LWR) and having better fuel utilization than an LWR.

Advanced isotope separation -- Refers to processes under development for the enrichment of U-235 that include selective photoexcitation by lasers and a plasma process based upon ion cyclotron resonance. A U.S. program goal is an SWU cost reduction of 50 to 75%.

Backfit -- The type of modification which can be made to equipment in current use.

Balance-of-plant -- Refers to the facilities at a nuclear power plant that are outside the reactor containment vessel.

Blanket -- A region containing fertile isotopes surrounding the nuclear reactor core.

Breeding ratio -- The ratio of the number of atoms of new fissile material produced in a breeder reactor to the number of atoms of fissile material consumed. This ratio is usually more than 1:1.

Breeder reactor -- A nuclear reactor that produces more fissile material than it consumes.

Burner -- A converter reactor which has no net production of any fissile species.

Centrifuge -- A rotating vessel that can be used for enrichment of uranium. The heavier isotopes of UF_6 gas tend to concentrate at the walls of the rotating centrifuge.

Cold -- An absence of significant levels of radiation.

Conversion ratio -- The ratio of the number of atoms of new fissile material produced in a converter reactor to the number of atoms of fissile material consumed. This ratio is usually less than 1:1.

Converter reactor -- A reactor that produces some fissile material but less than it consumes.

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Cooled spent fuel -- Spent fuel that has been stored until the most radioactive contaminants have decayed.

Coprocessing -- A proliferation-resistant nuclear fuel cycle in which the spent-fuel reprocessing step removes the fission products but does not separate the uranium and plutonium from each other.

Core -- The heart of the nuclear reactor where the fuel fissions and produces power.

Core disruptive accident -- A nuclear accident in which sufficient energy is generated to cause physical disruption of part or all of the reactor core.

Demonstration plant -- A plant designed using subscale components, generally smaller than 50 percent of commercial size. The principal function is to prove the design of systems and components.

Denatured uranium -- A mixture of uranium isotopes in which the concentration of fissile isotope(s) has been reduced by dilution with uranium of lower enrichment so that the resulting mixture is unsuited for weapons use. A concentration of <20% U-235 or 12% U-233 is usually assumed for denatured uranium.

Diffusion (gaseous) -- A technique for enrichment of uranium based on the fact that the lighter isotopes of a gas will diffuse through a porous barrier more rapidly than the heavier isotopes.

Doubling time -- The time required for a system of breeders to double the inventory of fissile fuel.

Electromagnetic enrichment -- A separation process based on the fact that an ion traveling perpendicular to a magnetic field will follow a circular path whose radius is proportional to the momentum of the ion. Ions formed from different isotopes of a given chemical species will follow paths of different radii, resulting in partial separation upon striking a collecting surface.

Enrichment -- The process of increasing the concentration of one isotope of a given element such as increasing the concentration of fissile U-235 in fertile U-238.

Fast breeder -- A reactor that operates with high-energy (fast) neutrons and produces more fissionable material than it consumes.

Fertile isotope -- An isotope that is not fissile itself but that is converted into a fissile isotope, either directly or after a short decay process following absorption of a neutron. For example, U-238 can capture a neutron

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to give U-239, which then decays to Np-239, which, in turn, decays to fissile Pu-239.

Fissile isotope -- An isotope that will split, or fission, into two (or more) lighter elements plus extra neutrons when it is absorbed by a neutron. The fission process and the fissile isotopes are the source of energy in nuclear weapons and nuclear reactors.

Fission -- The splitting of a nucleus, usually into two or more lighter elements. The total mass of the resulting particles is less than that of the original atom, with the difference being converted into energy.

Fluence -- A measure of the total number of neutrons passing through one square centimeter of a given target (flux times length of exposure).

Flux (neutron) -- A measure of the intensity of neutron radiation. It is the number of neutrons passing through one square centimeter of a given target in one second.

Fresh fuel -- Nuclear fuel ready for insertion into a power reactor.

Fuel assembly or bundle -- Typically 30 to 300 uranium-, plutonium-, or thorium-bearing rods mechanically held in a rigid assembly that can be loaded into a reactor core.

Fuel cycle -- The set of chemical and physical operations needed to prepare nuclear material for use in reactors and to dispose of or recycle the material after its removal from the reactor.

Fuel fabrication plant -- A facility where the nuclear material (e.g., enriched or natural uranium) is fabricated into fuel elements to be inserted into a reactor.

Fuel pin or rod -- A metal tube containing fissile fuel and/or fertile material. (See Fuel assembly.)

Full core reserve -- Fuel storage capacity for an entire reactor core loading.

Gaseous diffusion -- See Diffusion.

Heavy metals -- The fissile and fertile elements.

Heavy water (Symbol: D₂O) -- Water containing significantly more than the natural proportion of heavy hydrogen (deuterium) atoms to ordinary hydrogen atoms.

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Heavy-water reactor -- A nuclear reactor that uses heavy water for moderation and cooling.

Heterogeneous core -- A core where a variety of fuel assemblies are deployed. For example, heterogeneous fast reactors contain both fissile and fertile fuel assemblies within the core region.

High-enriched uranium -- Uranium that is enriched in U-235 or U-233 to above 20%. It can be used as a reactor fuel and is also a suitable material for a nuclear explosive.

Homogeneous core -- A core where only one type of fuel assembly is deployed.

Hot -- Containing significant levels of radiation.

Isotopes -- Atoms of the same chemical element whose nuclei contain different numbers of neutrons and, hence, have different masses, even though they are chemically identical. Isotopes are specified by their atomic mass number, that is, the total number of protons plus neutrons, and by a symbol denoting the chemical element (e.g., U-235 for uranium-235).

Lead plant -- A plant designed to demonstrate full-scale commercial-plant characteristics, economic and operating performance, and licensability.

Light-water reactor -- A nuclear reactor that uses ordinary water for moderation and cooling.

Liquid-metal fast breeder reactor -- A nuclear reactor that uses a liquid metal (e.g., sodium) for cooling, operates with high-energy (fast) neutrons, and produces more fissionable material than it consumes.

Low-enriched uranium -- Usually, uranium with a U-235 or U-233 enrichment of about 2 to 5%. However, LEU may also apply to uranium with a slightly higher or lower fissile fraction or to uranium with a U-233 fraction of several percent. LEU of about 3% enrichment is the current standard fuel for light-water reactors.

Medium-enriched uranium -- Uranium that is enriched above the level used for LWRs but less than highly enriched uranium.

Mixed oxide -- A mixture of uranium dioxide and plutonium dioxide.

Mixed-oxide fuel -- Nuclear reactor fuel composed of plutonium and uranium in oxide form. Prime candidate for LWR and LMFBR plutonium recycle fuels.

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Moderator -- A component (usually water, heavy water, or graphite) of some nuclear reactors that slows neutrons, thereby increasing their chances of being absorbed by a fissile nucleus.

Natural uranium -- Uranium as found in nature, containing about 0.7% of U-235, 99.3% of U-238, and a trace of U-234. It is used as the fuel in CANDU heavy-water reactors.

Once-through -- A nuclear fuel "cycle" in which spent nuclear fuel is stored indefinitely or disposed of after being used only once.

Proof-of-principle facility -- An experimental facility, the main function of which is to prove that the system-concept principles work.

Protactinium-233 (Pa-233) -- An isotope that is produced as the result of thorium irradiation and that decays to fissionable U-233. The U-233 content in thorium discharged from a reactor increases as a function of time due to this decay process.

Plutonium (Symbol: Pu) -- A heavy, radioactive man-made metallic element. Its most important isotope is fissionable Pu-239, produced by neutron irradiation of U-238. It is used for reactor fuel and in nuclear explosives.

Plutonium-239 (Pu-239) -- A fissile isotope created as a result of capture of a neutron by U-238. It is excellent material for nuclear weapons.

Plutonium-240 (Pu-240) -- A fissile isotope whose presence complicates the construction of nuclear explosives because of its high rate of spontaneous fission. It is produced in reactors when a Pu-239 atom absorbs a neutron instead of fissioning.

Pool storage -- Storage of radioactive fuel bundles under water in a specially designed cooling pool.

Plutonium dioxide (Symbol: PuO₂) -- A chemical form of plutonium used in mixed-oxide fuels.

Prebreeder -- A reactor fueled with either U-235 or plutonium operated to produce useful energy while also producing U-233.

PUREX -- The reprocessing method for spent fuel containing plutonium, uranium, and fission products.

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Reactor -- A facility that contains a controlled nuclear fission chain reaction. It may be designed to generate electrical power, to conduct research, or exclusively to produce fissile material for nuclear explosives.

Recycle -- The reuse of unburned uranium and plutonium in fresh fuel after separation from fission products in spent fuel at a reprocessing plant.

Repository -- A location containing wastes in long term storage or for ultimate disposal.

Reprocessing -- Chemical treatment of spent reactor fuel to separate the plutonium and uranium from the fission products and from each other.

Seed blanket -- A reactor core that includes a volume of material with a high content of fissile material (the seed) surrounded by a generally larger volume of material with a high content of fertile material (the blanket).

Separative work unit -- A measure of the effort required to separate uranium of a given U-235 content into two fractions, one having a higher percentage and one having a lower percentage of U-235.

Spent fuel -- Fuel elements that have been removed from the reactor because they contain too little fissile material and too high a concentration of radioactive fission products. They are both physically and radioactively hot.

Spiking -- A generic term implying the addition of radioactivity to enhance proliferation resistance.

Subcritical -- An assembly containing an insufficient quantity of fissile fuel to sustain a fission reaction.

Symbiotic -- The adjective used to describe a cycle where two or more systems are mutually supportive. For example, a fast breeder reactor may produce U-233 to supply denatured U-233 fuel for a converter reactor which in turn produces Pu-239 which is consumed by the breeder.

Tails (enrichment) -- Material that is lower in the concentration of the desired isotope than that being used as feed to the process.

Tandem fuel cycle -- A fuel cycle in which the fuel discharged from one reactor is charged into a different type of reactor without intermediate chemical processing.

Thermal neutrons -- Low-energy neutrons moving with the speed of natural thermal agitation.

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Thermal reactor -- A reactor in which the fission chain reaction is sustained by low-energy (thermal) neutrons.

THOREX -- The reprocessing method for spent fuel containing thorium, uranium, plutonium, and fission products.

Thorium-232 (Th-232) -- A fertile, naturally occurring isotope from which the fissile isotope U-233 can be bred.

Transmutation -- The transformation of one element into another by a nuclear reactor or by a series of reactions: for example, the transmutation of U-238 into Pu-239 by absorption of a neutron.

Transmuter -- A reactor that has as a primary purpose the fission of one isotope or element, such as plutonium, and the production of a second isotope, such as U-233.

Uranium-233 (U-233) -- A fissile isotope bred by fertile Th-232. It is similar in weapons quality to Pu-239.

Uranium-235 (U-235) -- The only naturally occurring fissile isotope. Natural uranium has 0.7% of U-235; LWRs use about 3% and weapons materials normally consist of 90% of this isotope.

Uranium-238 (U-238) -- A fertile isotope from which Pu-239 can be bred. It comprises 99.3% of natural uranium.

Uranium hexafluoride (UF₆) -- An easily volatilized compound of uranium used in uranium isotope enrichment.

Vitrification -- The solidification process in which high level waste is melted with a frit to form a glass.

Voloxidation -- A tritium removal process that involves the heating, oxidizing, and disintegrating of the spent fuel.

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