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GE Nuclear Energy

San Jose, California

Study of Pu Consumption in Advanced Light Water Reactors

Evaluation of GE Advanced Boiling Water Reactor Plants

prepared for
United States Department of Energy
San Francisco

MASTER

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PLUTONIUM DISPOSITION STUDY

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- (Data Requirement Chart A5 is in Appendix B)			
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(Data Requirement Charts A11 to A16, as noted in Appendix B, are in Section 10.1)			
4.5.1. Costs-Inside Complex	10.1	10.1	10.1
4.5.1.1 Pre-Operational Costs	10.1	10.1	10.1
4.5.1.2 Capital Costs	10.1	10.1	10.1
4.5.1.3 O&M Costs	10.1	10.1	10.1
4.5.1.4 Tritium Conversion	6.7		
4.5.1.5 Revenues	10.1	10.1	10.1
4.5.2 Cost-Outside Complex	10.1	10.1	10.1
4.5.3 Cash Flow	10.1	10.1	10.1
4.6 Schedules	10.2	10.2	10.2
4.6.1 Schedules -			
Inside Complex	10.2	10.2	10.2
4.6.1.1. Safety, Env, Reg.	10.2	10.2	10.2
4.6.1.2 R&D	10.2	10.2	10.2
4.6.1.3 Design	10.2	10.2	10.2
4.6.1.4 Construction	10.2	10.2	10.2
4.6.1.5 Preop. Checkout	10.2	10.2	10.2
4.6.1.6 Reactor Startup	10.2	10.2	10.2
4.6.1.7 Reactor Oper.	10.2	10.2	10.2
4.6.1.8 Tritium Conv.	6.5 and 6.6		
4.6.2 Schedules- Outside Complex	10.2	10.2	10.2

Compliance with the Statement of Work:

Statement of Work Section	Addressed in Section of Text
1.1 Neutronics	Section 2
1.2 Key Plant Performance Parameters	Section 4.3
1.3 Reactor Plant Layout:	Section 4
Plant Des. Layout	Section 4.1
Core Layout	Section 2
Core Design and Analysis	Section 2
Core Description	Section 2.5, 2.6, 2.7
Description of Fuel	Section 2, Section 4.2
Description of Control Rods	Section 4.2
Desc. of reactivity worths	Section 2.5, 2.6, 2.7
Description of control sys.	Section 4.2
Desc. Backup Safety	Section 4.2, 4.5
Desc. vessel, head, struc.	Section 4.2
1.4 Safety Margins	
Approach to Severe Accidents	Section 4.5, 2.8
Reactivity Coefficients, Normal	Section 2.5, 2.6, 2.7
Reactivity Coefficients, Transient	Section 2.8
Reactivity Reqs.	Section 2.5, 2.6, 2.7
Core coolability	Section 4.2, 4.5, 2.8
1.5 Tritium Production	Section 6
1.6 Electric Power Production	Section 4.2, 4.3
1.7 Plutonium Transformation Rate	Section 3.3
2.1 Deployment Strategy	Section 11
3.1 Fuel Design	Section 2
Fuel Cycle Diagram	Section 3.1-1, 3.5-2
Fuel Process, facilities	Section 3.5, 3.6
Spent Fuel Handling	Section 5
Spent Fuel Processing, Disposition	Section 5
Waste Management	Section 8
3.2 Alternatives: These alternatives are included as part of the Core Design in Section 2 and the associated changes to dispositioning in terms of fuel cycle, costs, schedules described throughout the document.	
3.3 Waste Stream	Section 8

**Statement of
Work
Section**

Addressed in Section of Text

4.1 Capital Cost Estimate	Section 10.1
4.2 Fuel Cycle and Waste Stream Costs	Section 10.1
4.3 Cost and Schedule for Tritium	Section 6.7
4.4 Project Cost Estimate	Section 10.1
4.5 Schedule	Section 10.2
5.1 Reactor Facility Complex Devt. Feed stock Reqt. Development	Section 11 Section 3.5 Section 9
5.2 Technology Needs	Section 9
6.0 "NEPA Documentation"	Section 8

1.0 Executive Summary

The ABWR is Extremely Well Suited for Plutonium Disposition: Enables Timely Disposition and a Broad Array of Core Design Options

Timely disposal of the weapons plutonium is of paramount importance to permanently safeguarding this material. GE's 1300 MWe Advanced Boiling Water Reactor (ABWR) has been designed to utilize full core loading of mixed uranium-plutonium oxide fuel. Because of its large core size, a single ABWR reactor is capable of disposing 100 metric tons of plutonium within 15 years of project inception in the spiking mode. The same amount of material could be disposed of in 25 years after the start of the project as spent fuel, again using a single reactor, while operating at 75 percent capacity factor. In either case, the design permits reuse of the stored spent fuel assemblies for electrical energy generation for the remaining life of the plant - for another 40 years. Up to 40 percent of the initial plutonium can also be completely destroyed using ABWRs, without reprocessing, either by utilizing six ABWRs over 25 years or by expanding the disposition time to 60 years, the design life of the plants and using two ABWRs. More complete destruction would require the development and testing of a plutonium-base fuel with a non-fertile matrix for an ABWR or use of an Advanced Liquid Metal Reactor (ALMR). The ABWR, in addition, is fully capable of meeting the tritium target production goals with already developed target technology.

The Proposed Concept Provides Safe, Licensable, Environmentally Sound Plutonium Disposition: Advanced Reactor Design and Automated Fuel Plant Enhance Safety and Lower Worker Exposure

The ABWR represents the distillation of over 30 years of experience gained world-wide in building and operating boiling water reactors and incorporates

ABWR FUEL CYCLE FOR Pu DISPOSITION

advanced features to enhance safety, minimize waste and reduce worker exposure. For example, the core is never uncovered nor any operator action required for 72 hours after any of the design basis accidents. In addition, the ABWR incorporates passive safety features to mitigate severe accidents. The Safety Analysis Report for the ABWR has already been completed and has undergone considerable safety review by the Nuclear Regulatory Commission (NRC). The ABWR is scheduled to be the first advanced light water reactor to receive a Final Design Approval (FDA) from the NRC in May, 1994. Two of these plants are currently being built in Japan. The conversion process from plutonium metal to mixed oxide fuel utilizes advanced automated technology, reducing waste and worker exposure. Evaluations reported in Section 8 of this study conclusively establish that the entire process could be accomplished while meeting all applicable environmental standards with a substantial margin.

ABWR KEY PARAMETERS	
MWt	3926
MWe	1300
HM/Bundle, Kg	180
Availability Factor (%)	87
Unplanned Scrams/Year	<1
Overnight Capital Cost (\$/kWe) ³	1400
Construction Schedule (Months)	48
Core Damage Probability (Per Year)	<10 ⁻⁶
Occupational Exposure (Man-Rem/Yr)	<100
Solid Radwaste (m ³ /Yr)	<100

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Provisions and Infrastructure for Safeguards already In-Place: Near Real-Time Special Nuclear Material Accountability, Proven NDT Assay Methods

Safeguards, much as in building a fuel fabrication plant or a reactor, require an infrastructure. An extensive domestic infrastructure is readily available. Existing provisions for safeguards to handle Special Nuclear Materials contained in U.S. Department of Energy Orders and Nuclear Regulatory Commission Regulations are sufficient to protect this material from theft or diversion. These provisions cover the entire range of processes and activities needed for disposal, from transport of metal, to conversion as oxide fuel. Nondestructive plutonium assay techniques, standards for material characterization and material accountability are readily available. The proposed approach to disposal provides timely, incremental increases in the resistance to diversion during each step of the way: initially in the destruction of the pit shape; then in the conversion to oxide as it takes more time to strip the oxygen; then in conversion to mixed oxide fuel and rod bundles as it takes even more time and equipment to divert fuel assemblies; and finally, after irradiation it would take extensive investments and shielded facilities to get back at the plutonium. The only additional issues in safeguards relate to system level considerations when plutonium disposal has to occur simultaneously, here in the U.S. and in the Commonwealth of Independent States (CIS). In Section 7, bilateral issues that arise under these conditions and the institutions that can be called on to serve the disposal process have been examined.

Fuel Cycle is Well Established: Proven Mechanically Mixed Process with State-of-the-Art Automation Features to Minimize Time to Bring Plant On-Line

The primary components of the fuel cycle are the conversion of metallic plutonium to its oxide, master-blending with uranium oxide powder, down-blending to the enrichments needed followed by pressing and sintering to the shape desired. All the processes required have been validated in the past in

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an industrial setting, while fabricating mixed oxide fuel for fast reactors. A number of alternate processes have been examined, however the mechanically mixed process incorporating a high level of automation has been selected consistent with the need for timely disposal and minimizing worker exposure. The level of investment needed in the fuel plant will depend upon the specific alternative chosen for plutonium disposition, the number of rods needed per year being the highest for the spiking case.

Process for Disposition Ready for Implementation: Proven Technologies No Developments Required

No additional development is required to implement the proposed concept for plutonium disposition. Detailed designs, ready for implementation, are available for the ABWR. Although there are no existing mixed oxide fuel fabrication facilities in the U.S., the entire process design, equipment and quality standards to which the fuel has to be produced are well known. Even though there is a high level of confidence in the fuel performance based on past experience, a lead test of mixed oxide fuel with gadolinia in a standard boiling water reactor or ABWR could be conducted, if desired.

High Level of Confidence in Estimated Costs: Cost Containment Designed into the ABWR and Supported by Plants under Construction

By disposing of the plutonium with a single ABWR or two ABWRs, if the disposal were to occur in the U.S. and CIS simultaneously, overnight capital costs have been held to a minimum. Significant reductions in the reactor and turbine island capital cost and, particularly, operation and maintenance costs have been designed into the ABWR in order to address the unfavorable cost histories of some previous generation light water reactor projects. The costs for the reactor, turbine island and the ex-reactor spent fuel storage pool could be relied upon to a high degree of confidence as two ABWRs are currently in construction in Japan and provide a good experience base with which the

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U.S. ABWR cost has been estimated. The power reactor approach to dispose of the plutonium allows this cost to be offset by revenue from the sale of electricity.

Cost containment for the fuel fabrication plant is provided by using proven technology. The basic chemistry and physics are well understood. The technology basis for equipment and facilities are also well developed. The integration of local computerized controls for automating the process steps at the level needed has not been executed in this country. However, this design integration step should be straight forward for the relatively simple fuel fabrication process selected.

Alternative	Overnight Capital Cost, \$M	Total Operating & Maintenance Costs,\$M (Years)
Spiking	2,918	1,579 (15)
Spent Fuel	2,889	2,410 (25)
Plutonium Destruction	10,900	7,029 (25)

High Level of Confidence in Keeping the Project on Schedule: ABWR Design Incorporates Factory Built Modular Components; Fuel Fabrication Plant Design Based on Proven Technology

The schedule for implementing this project is based on the experience derived from having built oxide fuel pellet lines and in building two ABWRs in Japan. The front-end time, prior to the start of the construction would be low as the ABWR has already undergone considerable licensing review by NRC. The extensive use of modular factory fabricated components in building the ABWR allows for better scheduling and control of the construction time. The use of already validated technology in converting plutonium metal to mixed oxide fuel, the use of existing infrastructures in implementing safeguards and

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the lack of new development required to implement this concept, all point to a high level of confidence that the project would be held to the proposed schedule. This schedule calls for the ABWR plant to go on-line, ready to dispose of the plutonium at full rated power in less than eight years from the start of the project. If located at a Government site where some of the site parameters are already available, the environmental impact studies could be reduced and the overall schedule improved by a year or more.

Infrastructure and Deployment: Institutional Model Exists; Independent Power Producers have Interest; Proven Reactor Technology Enhances Likelihood of Participation.

In the proposed approach for disposition, the turbine island complex will be owned by a utility or on Independent Power Producer with cost and revenue sharing arrangements. Institutional models for participation in such an arrangement with the reactor operated as a Government-Owned, Contractor Operated Facility exist. Preliminary discussions indicate that there is an interest for such an arrangement among the Independent Power Producer/Utility community.

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2.0

CORE DESIGN FOR PLUTONIUM UTILIZATION

In order to utilize plutonium in a boiling water reactor, a number of activities have to be undertaken but none is more important than a clear assessment of the capability of the reactor to handle mixed-oxide fuel while meeting all the stringent safety criteria. This assessment has been carried out in this section beginning with a brief evaluation of the principal differences in designing with plutonium rather than uranium fuel. Fortunately, GE's latest designs, the ABWR and the Simplified Boiling Water Reactor (SBWR) incorporate core design features which allow full core loading of mixed-oxide fuel. Following this, in Section 2.2, the mixed-oxide fuel performance data base in BWRs has been examined, which shows that this fuel should be able to operate as well as the conventional uranium fuel. In Section 2.3, the principal features of the two reactors SBWR and ABWR are compared for application to the disposition project. It is found that the ABWR is clearly superior to SBWR for this particular application, principally as a result of its larger core and the need to disposition large amounts of plutonium within a short time.

The specific core designs for the various alternatives are described in Sections 2.4 through 2.7, with the spiking alternative described in Section 2.5, the spent fuel alternative in Section 2.6 and the plutonium destruction option in Section 2.7. Full core loading of mixed-oxide fuel has been considered as the reference for each of these alternatives. Options under the various alternatives have also been indicated which will be pursued in future studies, if a particular option is judged to be attractive for a follow-on study. For both the spiking and spent fuel options, only one reactor is recommended with a single fuel design which fits both objectives. Destruction of plutonium in thermal reactors is difficult and to the extent this study was focused on this reactor type, this alternative has been taken to the limit allowed within the imposed constraints on disposition time and reactor type.

The Standard Safety Analysis Report for the ABWR has already been completed and an

entire range of transient events have been analyzed and described in that report.

In Section 2.8, the results of enveloping transients, evaluated for the core design judged to have the most limiting nuclear parameters among those proposed in Sections 2.5-2.7, are described. The fuel and the core is found to perform satisfactorily and within allowable limits.

2.1 DESIGN OF BWRs FOR PLUTONIUM UTILIZATION

This section of the report discusses the principal differences between designing the core and fuel of a BWR for utilization of plutonium fuel rather than standard uranium fuel. GE's advanced reactors, ABWR in particular, had originally been designed to handle a full mixed-oxide core. Standard BWR cores, with the irradiated fuel, contain a certain amount of plutonium as a result of the neutron capture of fertile material. Core management schemes have been developed to not only utilize the plutonium thus created, but actually to augment its production/consumption as methods of achieving fuel cycle efficiency. The principal difference involved in this study is really the degree to which the composition of BWR fuel and core consists of the various plutonium isotopes.

Despite the use of plutonium-bearing fuel in the design of the reactor, it should be remembered that the light water reactor is still a thermal reactor. In this respect, it emphasizes the thermal spectrum of the fissile material rather than its ability to operate in harder spectra.

There are two parts to the design of a BWR for plutonium utilization: lattice, or fuel design, and core configuration and management. Lattice design impacts spectrum changes, density changes, the resultant gadolinia loading and burnout rate of the fissile isotope, as well as the burn-in rate from fertile material.

BWR plutonium mixed-oxide fuel is designed to be as interchangeable as possible with the standard uranium fuel it replaces. The key criteria in designing for optimal plutonium

interchangeability with the standard designs are to:

- Minimize differences in reactivity versus burnup characteristics
- Maintain reasonable peak-to-average power factors
- Minimize effects on control blade reactivity worth

Experimental results from operating power stations and test facilities, and detailed evaluations with design models that have been compared with experiments have verified that the above objectives are achievable. The mixed-oxide fuel is designed to the same thermal, mechanical and hydraulic conditions and limits as the uranium fuel. The nuclear characteristics are, however, affected by the plutonium. Mixed-oxide fuel designs have been developed with the following nuclear characteristics relative to uranium fuel:

- Improved local and global power distributions
- Adequate control rod worth and cold shutdown margin
- Increased cold-to-hot operating reactivity swing
- Acceptable dynamic and safety characteristics

Interchangeability and high mixed-oxide fuel loading has been achieved in previous BWR designs by use of an mixed-oxide fuel island design in which the plutonium bearing rods are located in central regions of the lattice away from the control blades and the water gaps between assemblies. Location of mixed-oxide fuel in these lattice regions replaces the highest enrichment fuel rods in the BWR assembly, thereby maximizing uranium ore and separative work savings. The low enrichment uranium rods remain in the highest neutron importance regions at the edge of the assembly and effectively "shadow" the mixed-oxide fuel rods with respect to neighboring assemblies. This design also maximizes the plutonium concentration in each fuel rod to minimize the number of plutonium fuel rods and hence fabrication costs. Interchangeability has also been demonstrated for an all mixed-oxide fuel loading with a widespread use of gadolinia throughout the interior regions of the lattice.

This allows the burn rate of the interior plutonium bearing rods to be slowed so that the burnout rate of the fissile isotopes is more nearly that of the standard uranium fuel.

The basic ABWR lattice and control assembly is one of the key factors which makes possible the near direct substitution of plutonium for ^{235}U . One of the most important considerations in utilizing any appreciable amount of plutonium in the BWR core is maintenance of adequate rod worth when plutonium is inserted into the lattice. The plutonium isotopes have an absorption cross section approximately three times greater than ^{235}U , and thus plutonium, when placed adjacent to a control blade, will compete with the control blade and reduce control system reactivity worth. Because of the mechanical design of the ABWR assembly pitch, sufficient reactivity margin can be obtained in the hot, voided condition while still maintaining ample shutdown margin. The water gaps between assemblies also help to decouple the thermal neutron spectrum and minimize spectral interactions between fuel assemblies with large differences in fissile content and neutron absorption. This is especially important in the extended burnup designs such as the plutonium destruction option.

The interchangeability which has been designed into the lattices utilizing plutonium results in the ability to use core management techniques which have come into practice over years of operational experience. Neither fuel reliability nor overall reactor safety response would be expected to deviate appreciably from similar UO_2 loaded ABWR cores.

2.2 MIXED-OXIDE FUEL EXPERIENCE BASE FOR BWRs

Mixed-oxide fuel bundles have been designed utilizing the same principles outlined in this report and have undergone irradiation in a broad range of reactor environments. Some of the programs which have contributed to the BWR experience base with mixed-oxide fuel are summarized in Table 2.2-1. These programs range from those involving a few fuel rods to a full reload batch of mixed-oxide bundles at the Garigliano reactor.

The results of these irradiation programs have confirmed the high degree of

interchangeability between MO_2 and UO_2 bundles for operation in BWRs. Performance of mixed-oxide fuels has been found to be comparable to that of UO_2 fuel.

Table 2.2-1. GE BWR Mixed-oxide Fuel Experience Base

Program	Scope	Key Results
Euratom Sponsored Irradiation of Pu Fuel in Vallecitos BWR (1960-1966).	Pu fuel in 1/4 of a fuel assembly was irradiated.	Determination of isotopic content as a function of fuel burnup.
Plutonium Recycle Test Reactor (Hanford)	80 rods of hot pressed and 3 rods of cold-pressed pellets were irradiated to 11,000 MWd/MT.	Fuel subjected to specific powers in excess of 21 kW/ft to provide operating experience under severe environmental conditions
Garigliano MO ₂ Reload Fuel (1975-1981)	Reload fuel batch of 46 MO ₂ bundles (32 MO ₂ rods per bundle).	Bundles operated satisfactorily and reached a discharge exposure of >25,000 MWd/MT.
Big Rock Point-EEI Programs (19699-1974)	Irradiation of 32 MO ₂ rods in 16 UO ₂ carrier bundles and irradiation of 3 MO ₂ reload bundles.	Evaluation of annular and solid pellet MO ₂ fuel performance including destructive examinations. Peak pellet burnups to 36,000 MWd/MT.
Halden Plutonium Program	MO ₂ fuel rods with annular pellets were irradiated at 17 to 22 kW/ft.	Measurements of the effects of pellet-clad interaction showed acceptable behavior of the MO ₂ rods.
Kritz Program	A series of hot critical experiments was completed including power distribution and other measurements for Garigliano MO ₂ reload bundles.	Benchmarking of data to analytical predictions produced good agreement.
Quad Cities 1 EPRI Program (1974-1982)	4 MO ₂ bundles around the center control blade and 1 MO ₂ bundle on the core periphery underwent irradiation in a highly controlled situation with extensive non-destructive measurements.	Center bundles operated successfully for 8 years of core residence to burnups of 57,000 MWd/MT. Detailed benchmark data has been obtained on local power distributions, fuel isotopics and annular MO ₂ pellet mechanical integrity.

The EPRI sponsored program at Quad Cities 1 has included a substantial number of nuclear and mechanical performance measurements which provide benchmark data for verifying analytical models. Featured in the measurements are fuel rod isotopic distributions at low exposure (one cycle), nominal discharge burnup bundle rod-by-rod and axial gamma scans at cycle refueling outage to assess power/exposure distributions, and a

full range of fuel rod non-destructive tests and profilometry to verify mechanical integrity. The five test assemblies of this program have been operated satisfactorily for six years of core residence.

Based upon the design and operating experience accumulated to date on mixed-oxide fuel, it is expected that BWRs can safely and reliably operated with mixed-oxide bundles as well as UO_2 bundles. The mixed-oxide fuel bundle designs used in the analyses detailed in this report are representative of the best designs in the mixed-oxide fuel BWR data base.

2.3 GE'S ADVANCED BOILING WATER REACTORS

2.3.1 ABWR/SBWR and Downselection

GE has developed and is in the process of licensing two advanced boiling water reactor designs incorporating a number of proven features from more than three decades of BWR design and operating experience. These features simplify the design, reduce construction time and materials while improving safety margins and result in better economics. These two designs with capacities spanning a range of needs - the 600 MWe SBWR and the 1300 MWe ABWR - are readily adapted to the task of dispositioning excess plutonium in a safe, reliable and economic manner. The benefits of GE's advanced light water reactors are summarized in Table 2.3-1, which identifies the factors that contribute to the benefits. Some of the key performance characteristics for the two reactors are listed in Table 2.3-2.

The two reactors, the SBWR and the ABWR, were compared for their efficacy to disposition excess plutonium using a number of different criteria. These criteria include:

1. Ability to disposition large quantities of plutonium in a relatively short time frame, a key parameter in safeguarding plutonium,
2. Overall economics when considering either the spent fuel or spiking options, and
3. Flexibility in plant features to accommodate a full mixed-oxide core.

Table 2.3-1 BWR Benefits and Contributing Features

Benefit	Contributing Feature
Reduced capital cost	Predictable construction schedule Standard, prelicensed design Reduced building volume and quantities of materials Modular, prefabricated construction Passive approach to safety (SBWR) Cylindrical reinforced concrete containment vessel Rolling 4x10 construction schedule
Improved availability	Latest materials and water chemistry Outage length reduction <ul style="list-style-type: none"> - Better outage and maintenance planning - Reduced inservice inspection, less welds, automated - Improved equipment reliability - Automated service equipment - Minimum shuffle fuel loading - No reactor internal pumps (SBWR) - Simplified turbine island (SBWR) - Less Class 1E equipment (SBWR) - Fewer tech specs (SBWR) Scram reduction <ul style="list-style-type: none"> - Advanced C&I system - Self diagnostic capability - Less equipment and control panels Longer operating cycles Continued low fuel costs <ul style="list-style-type: none"> - High burnup level - Axial enrichment zones - Axial gadolinia zones - Uses latest BWR fuel designs
Enhanced safety	No uncovered core (design basis) No external recirculation loops Internal recirculation pumps No large nozzles below core Three separate mechanical and electrical ECCS divisions Three safety-grade RHR systems RCIC upgraded to safety-grade Redundant Control Rod Drive Insertion modes Improved seismic capability (RIPs, RCCV)
Passive severe accident mitigation	Inerted containment Lower drywell flooded Fission product retention Suppression pool Containment over-pressure protection Provision for external water supply
Reduced siting impact	Passive accident mitigation features No impact on public from severe accident More MWe per site
Less post-accident operator action	Automatic ECCS protection Redundant makeup water sources/power supplies No operator action needed for three days

Benefit	Contributing Feature
Less low-level radwaste	Advanced filtering technology Advanced materials Advanced water chemistry
Reduced worker radiation exposure	Reduced outage lengths Elimination of recirculation loops Less equipment maintenance Engineered features for servicing & maintenance Advanced low-cobalt materials Better water quality Two percent reactor water cleanup system Separated "clean" and "hot" areas No scram discharge volume
Enhanced fire protection	Reactor building divided into four areas physically separated by fire walls
Improved emergency procedures	Symptom and experience based
Improved C&I and control room	Enhanced man/machine interface More automation Mimic, menu-driven displays Digital safety systems Self diagnostics Fiber optic multiplexing
Decreased surveillance	Simplified systems Improved equipment reliability Self-testing and diagnostics Simplified tech specs
Improved operator support technology	Operator procedures optimized, computerized and displayed on CRT's Safety parameter display system integrated into design
Simplified plant operation	Touch screens System oriented controls Automation of operator action Menu-driven prompts keep operator in loop

Table 2.3-2 GE BWR Key Performance Characteristics

Characteristic	ABWR	SBWR
Net electric output (MWe)	1300	600
Construction schedule (months)	48	36
Overnight capital cost (\$/KWe)	1400	1700
Production cost (¢/KWh)	1.3	1.6
Fuel loading (Mt HM)	155	94
Average Fuel Burnup (GWd/t)	38	38
Availability factor (percent)	87	87
Refueling cycle (month)	18	24
Number of fuel bundles	872	732
Fraction refueled per cycle	0.32	0.36
Average Outage Days per year	29	<25

The comparisons between the two types of plants in regard to each of these criteria are given below:

Ability to Disposition Large Quantities of Fissile Plutonium: The amount of plutonium that needs to be dispositioned, 100 metric tons, is quite large there is sufficient fissile material here to power six ABWR plants for 20 years or 12 SBWR plants for the same duration. Timeliness is the key driver in safeguarding this plutonium as indicated by the requirement to disposition the plutonium within 25 years of the start of the project. The ABWR with its larger core is capable of dispositioning the entire inventory in less than eight years in the spiking mode. An SBWR or a similarly rated core, assuming that the core design could be optimized just as well, would take nearly twice this time or 16 years. For the spent fuel and plutonium destruction alternatives, the disparity becomes even more glaring as twice as many SBWR plants are required to achieve the same fuel exposure.

Overall Economics, Spiking Alternative: In the case of the spiking alternative, one might falsely construe an advantage for SBWR by relaxing the requirement relative to the shortest possible time for dispositioning and comparing the two plants based on their different dispositioning times. Since the fuel exposure levels are very low, it might be argued that plutonium can be dispositioned with an SBWR or similarly rated plant with an associated lower initial investment arising from its lower power rating. However, a detailed assessment, given below shows that once the plants are compared on an equal footing - that is, both plants are assumed to disposition the fuel over the same duration of 16 years - the ABWR is by far better suited for this task.

First, let us consider the difference in reactor plant capital costs for this is the only driver for considering an SBWR:

- The initial overnight capital cost for an ABWR is \$1.82 billion
- The initial overnight capital cost for an SBWR is \$1.02 billion

Or, the difference in the initial capital cost is \$0.8 billion. As detailed later, to meet the

post-irradiation dose level requirements, the bundles have to be irradiated in the energy cycle for approximately one month. Typically, refueling can be accomplished within a month, so that the capacity factor for this alternative is 50 percent. The SBWR requires 16 years to disposition the plutonium while the ABWR can do it in about 7.4 years at a capacity factor of 50 percent. If the ABWR operating time is increased so the plutonium is dispositioned in 16 years, the capacity factor will increase to about 75 percent.

The amount of electricity generated over the 16 years of operation will be:

- SBWR for 16 years at 50 percent capacity factor: 42 billion kilowatt-hours
- ABWR for 16 years at 75 percent capacity factor: 136 billion kilowatt-hours

The difference in the electricity generated, 94 billion kilowatt-hours, is worth \$5.64 billion dollars at the rate of six cents per kilowatt-hours. The return on the additional investment of \$0.8 billion dollars for the ABWR is at once apparent.

In addition to the comparison based on the revenue from electricity generated, it is instructive to compare other costs, both operating and investment costs, where the two plant systems could be expected to differ. First, it should be noted that with a bundle weight of 128 kilograms of heavy metal per assembly compared to 180 kilograms of heavy metal per assembly of an ABWR, the SBWR will require more numerous assemblies for dispositioning the same total quantity of plutonium. If the plutonium enrichments are assumed to be the same, the number of pins needed will be about 40 percent more for an SBWR compared to an ABWR.

From these discussions, it is clear that an ABWR is superior to an SBWR in the spiking mode, from the viewpoints of shortest time to dispositioning or from an economics viewpoint.

Overall Economics, Spent Fuel and Plutonium Destruction Alternatives: For the spent fuel and plutonium destruction alternatives, as stated earlier, the ABWR is by far superior for

plutonium dispositioning. For these alternatives, fuel exposure is a key driver and to achieve the same exposure, the number of SBWRs required will be twice that for ABWRs. In addition, the net electricity generated revenues from two SBWR are lower than for a single ABWR. Therefore, an ABWR has been chosen for these alternatives as well.

Flexibility to Accommodate Mixed-Oxide Fuel: Both types of reactors are inherently flexible in accommodating mixed-oxide fuel. However, there are certain design features inherent to the ABWR that makes it more attractive for mixed-oxide fuel usage. A major factor in dispositioning plutonium as mixed-oxide fuel in light water reactors is the ability to load the entire core with mixed-oxide fuel. Other designs have been developed in the past where mixed-oxide fuel has been used for partial core loadings. When the loading fraction is less than unity, the ability of the reactor to disposition the fuel quickly falls off rapidly. Similarly, when used in the spent fuel mode, when less than a full core is loaded with mixed-oxide fuel, complementary Urania fuel has to be bought and this leads to ineffective utilization of the plutonium, leading to less than desirable overall economics. Analyses documented later in this section have conclusively established the ability of ABWRs to utilize full core loads of mixed-oxide fuel, although SBWR might be capable of this as well.

For these reasons, the ABWR was selected for further detailed study.

2.3.2 Nuclear Methods And Qualification

This section describes the methods and data used by GE to model the heterogeneous boiling water reactor core with standard urania fuel and mixed-oxide (plutonia-bearing) fuel. The methods can be categorized into four main sections: (1) the three-dimensional nuclear model, (2) lattice physics methods, (3) the thermal-hydraulic model, and (4) model qualification.

The section on the three-dimensional nuclear model discusses the PANACEA BWR core simulator code. PANACEA is a three-dimensional, coupled nuclear and thermal-hydraulic,

time independent model of the BWR power distributions and thermal performance as a function of control rod position, fuel loading pattern, coolant flow, reactor pressure, and other operational and design variables. The program is capable of performing fuel cycle and thermal limits calculations. A special power and exposure iteration option is available for target exposure distributions and cycle length determinations.

The section on lattice physics methods discusses the TGBLA code. The report discusses the generation of few group neutron cross section by means of the neutron spectra calculations, the calculation of fuel rod power distributions for a fuel bundle, and the evaluation of the effects of fuel exposure. The output of TGBLA is used as input to PANACEA. The data flow between these codes in are shown in Figure 2.3-1.

The section on the thermal-hydraulic model discusses the void-quality correlation, the bypass region calculation, the total core energy balance, and the thermal limits calculations. The thermal-hydraulic model is coupled to the neutronics model of PANACEA. Iterations are performed on power and flow until a consistent solution is obtained by PANACEA for this neutronic and thermal-hydraulic coupling.

Model qualification is presented last. Extensive operating plant tracking results are presented. Comparisons of calculations are presented for gamma scan measurements taken at the end of six operating fuel cycles. Results are also presented on comparisons of cold critical measurements taken at two plants over seven cycles of operation.

This report presents the improvements made in the GE nuclear design methods. Additional information on these methods can be found in the previously reviewed and approved GE licensing topical reports.

2.3.2.1 Three-Dimensional Nuclear Model

The nuclear model is based on coarse nodal, one-group, static diffusion theory. The mesh

Figure 2.3-1 Fuel/Core Design Methods Data Flow

points are distributed approximately every two fast neutron mean free paths. Because of the short thermal neutron mean free path in a water reactor, most neutron diffusion takes place at high neutron energy. Therefore, the model is oriented toward accurate representation of fast neutron diffusion.

It is important to account for lattice fine structure in determining nuclear data for the coarse-mesh global model. Separability is assumed for lattice cells consisting of one or more fuel bundles. The nuclear input preparation is performed by detailed calculations for the lattice cells or fuel bypass. Coarse-mesh diffusion parameters are fit parametrically as a function of moderator density, exposure, control, and moderator density history. (Moderator density history accounts for spatial isotopic concentrations which are affected by the neutron spectrum during burnup; the spectrum, in turn, is sensitive to moderator density.) These parametric calculations produce isotopic compositions and three-group cross-section data homogenized over the lattice cell. The methods and procedures used in the lattice model are described in Section 2.3.2-2. A data reduction program (GELIB/SPP-AN) processes these data into correlations and tables used in the BWR Simulator.

The nuclear model includes representation of Doppler broadening as a function of effective average fuel temperature. Xenon poisoning is considered with the spatial xenon concentration calculated for equilibrium conditions or input specified for non-equilibrium conditions. Provisions are also made to account for the reactivity effect of crud deposited on fuel rods.

Convergence of inner flux iterations is achieved through the use of successive over-relaxation. Outer void loops are converged with an under-relaxation technique. A power exposure iteration option is available. Neutron flux is converted to fission power for thermal-hydraulic coupling.

Spatial isotopic concentrations are calculated for fuel cycle analysis. Simulated in-core instrument readings can also be calculated.

2.3.2.2 Detailed Lattice Physics Methods

This section describes the process of accounting for the lattice fine structure in determining nuclear data. This data is then used as input into the three-dimensional coarse-mesh core calculations documented in Section 2.3.2-1. Due to the differences in bundle design, control state, void condition and accumulated exposure, bundle nuclear properties in a core are actually nonuniform both in transverse and axial directions. As a result, the neutron behavior in each axial segment of a bundle is determined not only by the nuclear properties of that segment but also by the nuclear properties of adjacent segments of that bundle and neighboring bundles. The calculation of the influence of neighboring bundles is very difficult because each bundle can potentially undergo a wide spectrum of environments, depending on its location and core operating history. However, this influence is generally small. To model the bundle, an approximation is made to account for the influence of neighboring bundles in the core simulator and to generate homogenized cross sections. This approximation is made by assuming that the bundle is located in an infinite, periodic lattice along the transverse directions and is uniform and infinitely long along the axial direction.

As a result of the preceding approximation, a bundle can be simulated as an "isolated" two-dimensional heterogeneous system that may comprise fuel rods, water rods, burnable poison (gadolinia), in-channel water, channel, out-channel water, and control rod. The bundle modeling is then further divided into two stages: (1) the fuel rod cell and external region modeling by using transport theory methods, and (2) the coarse-mesh bundle modeling based on cell homogenization and diffusion theory methods.

The solution techniques begin with the generation of thermal broad-group neutron cross sections for all homogenized fuel rod cells and external regions in a bundle. In the thermal energy range, the rod-by-rod thermal spectra are calculated by a method similar to the THERMOS formalism. The major difference is that neutron leakage from rod to rod is taken into account.

The net leakage is determined by diffusion theory and is fed into the thermal spectrum

calculation. Iterations between diffusion theory and thermal spectrum calculations are carried out to determine accurate, spatially dependent, thermal cross sections.

The second step is to generate the cell and region-homogenized cross sections for the fast and epithermal broad groups. In the epithermal and fast energy range, the level-wise resonance integrals are calculated by an improved intermediate resonance approximation in which the intermediate resonance parameters are fuel-rod-temperature dependent. In addition, the fuel rod resonance escape probability is calculated according to a treatment by Mizuta.

The third step utilizes the cross sections generated in steps one and two in a two-dimensional, cross-mesh, broad-group, diffusion-theory calculation to determine the nodal flux and power distributions in the bundle.

The fourth and final step is to perform the nuclide depletion calculation. In the depletion calculation, 100 nuclides are treated, including 25 fissile and fertile nuclides and up to 48 fission products plus one pseudo fission product. An improved burnup integration scheme is employed which avoids numerical problems associated with the equations encountered in poison burnup calculations.

The preceding steps are repeated for each burn step until all given burn steps are completed.

The generation of thermal cross sections and diffusion coefficients includes the condensation of thermal cross-section library from 30 groups to 16 groups, the calculation of leakage-dependent region-wise neutron spectrum, the generation of thermal leakage-dependent region-wise neutron spectrum, the generation of thermal homogenized diffusion parameters, and P_1 - blackness theory for strong absorption mediums.

The conventional library for thermal spectrum calculations is composed of 30-group thermal cross sections. A properly condensed 16-group, rather than a 30-group, thermal library, is

sufficient for use to calculate the thermal spectrum correctly.

The fluxes for the group condensation of cross sections from 30 to 16 groups are calculated by using the THERMOS method for the averaged regular and the averaged gadolinia fuel rod cells. Each averaged cell is assumed to have two regions (fuel region and cladding-moderator homogenized region) in the radial direction and to be infinitely long in the axial direction.

The integral transport theory method has been widely utilized for calculating thermal neutron spectra in reactor lattices. The THERMOS computer code (based on this method) was introduced by Honeck in 1961. The integral transport method has continued to be used along with two improvements: (1) the collision probability is calculated more accurately, and (2) the reflecting boundary condition is generalized to the albedo (or leakage-dependent) boundary condition.

2.3.2.3 Thermal-Hydraulics Model

Power-void feedback of a BWR core makes the thermal-hydraulic representation of considerable importance to the nuclear calculation. Nuclear parameters and, therefore, the power distribution are sensitive to local steam void content.

Formulas from the 1967 ASME Steam Tables are used to evaluate coolant properties as a function of reactor core pressure and inlet enthalpy. All thermal-hydraulic variables are assumed to vary linearly between nodes, where the nodal representation is the same as that for the neutron flux; therefore, the thermal-hydraulic model is consistent with the difference equations used for solution of the nuclear model.

The flow distribution for characteristic parallel channels is determined by balancing core pressure drop. Characteristic channels are determined by combinations of total channel power, axial power shape, inlet orifice design, channel geometry (e.g., number of fuel rods, spacer design, lower and upper tie plate design,), and buildup of crud on the fuel rods. The

flow into individual channels is obtained by interpolation among characteristic channels of the same geometry and orifice type as a function of radial and axial power factors and crud thickness.

The axial enthalpy and quality distribution is determined for each channel by a nodal energy balance which considers fuel rod heat flux to the coolant, neutron moderation and gamma heating in the coolant and in the flow channel wall, heat transfer through the channel wall to the bypass region, and γ , (n,γ) , and (n,a) heating in control blades. The void distribution is given by a void-quality correlation which provides a physically oriented representation of sub-cooled and bulk boiling through a few empirical constants which describes the local flow structure.

Flow-power-void iteration is required to reach an equilibrium condition. Once this is obtained, the thermal-hydraulics model enables evaluation of the margin relative to thermal limits.

2.3.2.4 Model Qualification

The models in this report have been qualified against data obtained from operating Boiling Water Reactors. The qualification studies consist of: (1) simulation and tracking of nine operating cycles on three plants; (2) three-dimensional global power distribution comparisons with gamma scan measurements taken at the end of six cycles; and (3) cold critical measurements taken during seven cycles and two plants; and (4) burnup verification data at the end of one cycle.

In plant tracking calculations, the 3-D simulator (using cross sections generated from the lattice model) is used to simulate the behavior of a plant during operation. The reactor power, flow and pressure are input to the simulator, which then calculates the core k_{eff} and power distribution as a function of cycle exposure. The operating reactor is critical; hence, the calculated k_{eff} is compared to 1.0. Accurate and technically well-founded simulators should calculate a k_{eff} close to 1.0 and the difference between k_{eff} and 1.0 should not vary

appreciably from plant to plant or as a function of fuel exposure. Consistency of k_{eff} bias ensures that accurate cycle length estimates will be obtained in future core designs. Tracking calculations for Hatch 1 (Cycles 1 through 3), Quad Cities 1 (Cycles 1 through 5), and Millstone 1 (Cycle 7) were performed. The overall average is 1.002 with an RMS of 0.002, which is a quite narrow band considering the data are compiled over three plants and over nine cycles.

2.4 CORE AND FUEL DESIGN

2.4.1 ABWR Core Design Bases and Criteria

The three options considered in this study, (spiking, spent fuel and plutonium destruction) require two different fuel designs and three different core designs. Although the overall plutonium dispositioning of these options lies on a continuum from low utilization of the fissile material to reduction of the total amount of transuranic elements, the fuel and core designs associated with each point along the continuum are quite different from one another. These are explained in detail below. Production of tritium in the ABWR requires additional coupled fuel and core designs, which are also explained in Section 6.

The ABWR reactor core has been designed to accommodate any BWR fuel design. This provides the capability to utilize advanced fuel technology that is constantly evolving for the current fleet of BWRs, including the two ABWRs under construction at the Kashiwazaki Kariwa site (units 6 & 7). The ABWR core has symmetric bypass gap widths, similar to a C-lattice reactor; however, the bundle pitch is increased by 0.1 inches. The ABWR lattice has been given the designation N-lattice. The increased bundle pitch in the ABWR core results in increased bypass flow area which increases the hot to cold reactivity swing yielding improved cold shutdown margin. The hot to cold swing inherent in the ABWR core eliminates any impact on the thermal margin or energy performance of the core design from cold shutdown margin considerations. The increased bypass gap width also reduces the magnitude of the void reactivity coefficient which enhances the stability performance as well as reduces the impact of pressurization events. The nuclear characteristics of ABWR's

N-lattice make it an ideal reactor to utilize mixed-oxide fuels as well as standard urania fuel.

There are several advanced fuel product lines currently being offered by GE to the domestic and international BWR community. GE is also constantly developing advanced fuel technology for introduction into future fuel evolutions, including the use of mixed-oxide fuels. Detailed ABWR core designs have been prepared for DOE evaluation utilizing our most advanced 8x8 fuel designated as GE9. The GE9 designs presented here were chosen to fully meet the operating margin requirements typical of currently operating plants.

GE9 was developed primarily as a high exposure and high energy design that is capable of batch discharge exposures up to 38000 MWd/MT (although the optimum batch average discharge exposure is dependent upon specific fuel cycle cost parameters as well as other specific requirements which may result in a lower discharge exposure goal). The standard GE9 design has a peak pellet exposure limit of 60 GWd/MT. In order to take advantage of the high exposure mechanical design, GE9 has improved linear heat generation rate and reactivity margins (cold shutdown margin).

This section provides a description of the nominal design criteria used to establish design reference loading patterns, lattice designs and the fuel mechanical designs. A brief review of the major criteria used in BWR design, such as the linear heat generation rate, maximum average planar linear heat generation rate (MAPLHGR) and minimum critical power ratio is given in Appendix A. During implementation, the actual individual values of these criteria are often modified through communication with the utility staff to provide conformance to any specific requirements. As for a typical ABWR candidate application, a 10 percent margin requirement was applied to linear heat generation rate and a 15 percent margin requirement was applied to MCPR.

2.4.1.1 Lattice Design

GE customizes the nuclear design of the fuel bundles for each specific reactor application. This customization integrates the lattice designs with the core loading design to permit an

energy utilization plan and all the design limits discussed above to be met in an optimized manner.

The cycle energy and reactivity margin requirements determine the lattice enrichments and gadolinia loadings. These parameters control the hot and cold k-infinity characteristics and are selected such that cycle energy, hot excess reactivity and cold shutdown margins are met.

The thermal margin requirements are used to define the allowable local peaking and R-factor variations with exposure. The local peaking factor impacts the LHGR, while the R-factor, which is GE's designation for the subchannel factor, has a direct influence on the MCPR performance of the bundle.

Both the reactivity and thermal margins influence the axial zoning of the bundle. Although axial gadolinia zoning can be used to improve cold shutdown margin by means of a short shutdown zone near the top of the bundle, it is found that sufficient margin exists with the ABWR to utilize a simple single lattice bundle (no axial zoning- either enrichment or gadolinia). In addition, axial enrichment zoning is typically used in BWRs, even if only to provide natural uranium sections at the top and bottom axial nodes. These short sections of natural uranium are generally provided to help reduce fluence on the core plates as well as to provide some end-of-cycle power suppression in the upper end of the assembly. To improve plutonium disposition, however, these ends are not utilized. This results in slightly lower thermal margins at end-of-cycle, but since such margins are sizable in all these designs, the small decrease in the thermal margin is not considered to be of consequence.

The thermal mechanical limits on peak pellet exposure and maximum LHGR versus exposure are considered during the lattice design process. These limits are utilized to derive the lattice dependent MAPLHGR limits. The resultant MAPLHGR limit provides confidence that neither the UO_2 , MOX, or the $\text{MOX/Gd}_2\text{O}_3$ rods will exceed their respective LHGR versus exposure limits. Thermal mechanical limits for the mixed-oxide fuel are assumed to be the same as for UO_2 fuel.

A description of the physical configuration of the fuel lattice and the control blade configuration is given in Figure 2.4-2.

2.4.1.2 Mechanical Design

GE provides a prelicensed mechanical design for its utility customers and typically generates unique nuclear designs for each cycle of application. The prelicensing of the mechanical design, in the U.S., has been accomplished by confirming that the GE9 meets all of the licensing criteria approved by the NRC (Amendment 22 of GESTAR-II). For applications in other countries, GE will work with the individual regulatory agencies to demonstrate that GE9 meets all of their specific licensing requirements.

2.4.1.3 Reference Loading Pattern Design

The configuration of the ABWR core is presented in Figure 2.4-3. The ABWR core consists of 872 bundles operating at an average thermal power of 4.5 MW for a total core output of 3926 MWth. The orifice loss coefficient of the peripheral fuel locations is increased to preferentially force flow through the high power interior region. The specific design requirements are summarized in the following tables.

The GE design of a reference loading pattern complies with several design criteria to demonstrate: (1) the Technical Specification license limits are met throughout the cycle, (2) full power operability, (3) operational flexibility, and (4) fuel bundle and channel reliability. The nominal standard design criteria used by GE to establish a design reference loading pattern, and their bases, are described below. The specific values used for these criteria are summarized in Table 2.4-1 through Table 2.4-3. These standard criteria may be modified on a case by case basis during application.

	CHANNEL			FUEL ROD			PELLET	WATER ROD	
I.D.	A	B	C	D	E	F	G	H	I
IN	0.100	5.278	0.380	0.032	0.483	0.419	.411	1.340	1.260

	CONTROL ROD			BUNDLE LATTICE			CELL		
I.D.	J	K	L	M	N	O	P	Q	R
IN	1.55	4.902	0.328	0.640	0.157	0.158	0.331	0.331	12.20

	CONTROL ELEMENT		
I.D.	S	T	U
IN	0.220	0.027	0.045

Figure 2.4-2 ABWR Fuel Bundle & Control Blade Configuration

Figure 2.4-3 ABWR Core Configuration

Table 2.4-1. Performance Requirements

Thermal Power Rating, MWth	3926
Overall Capacity Factor, %	75
Exposure level @3 ft 2 yrs after discharge (rem/hr)	100

Table 2.4-2. Interface Requirements

Number of Bundles	872
Number of Control Blades	205
Control Blade Pitch, inches	12.2
Assembly Pitch, inches	6.1
Power Density, KW/l	50.6
Rated Core Flow (100%), Mlb/hr	115.1
Maximum Core Flow (111%), Mlb/hr	127.8

Table 2.4-3. Fundamental Fuel Design Requirements

Maximum Linear Heat Generation Rate, KW/ft	14.4
MCPR Operating Limit	1.25
Minimum Cold Shutdown Margin, % ΔK	1.0
Maximum Pellet Exposure, GWd/MT	60.0
Maximum Bundle Residence Time, years	7.0
MLHGR Design Margin, %	10
MCPR Design Margin, %	15
Dynamic Void Coefficient, %/°C	13
Doppler Coefficient %/°C	0.5

2.4.1.4 Minimum Critical Power Ratio Margin

The reference loading pattern is designed such that the core performance maintains the MCPR equal to or above the Operating Limit MCPR throughout the cycle, without derating the thermal power. Additional margin is also included to account for the prediction uncertainty of the analytical calculation and possible differences between calculations and actual core operation.

The MCPR operating limit presented in Table 2.4-3 was generated by performing detailed transient analyses for a range of core dynamic response factors. The operating limit is defined by the Load Rejection Without Bypass event and includes the effects of the turbine bypass capacity and the associated scram signals.

2.4.1.5 Linear Heat Generation Rate Margin

The basis of linear heat generation rate margin is similar to the MCPR. The reference loading pattern design also provides confidence the linear heat generation rate margin limit is met with adequate margin to account for the prediction uncertainties.

2.4.1.6 Cold Shutdown Margin

The core has to be designed such that the core can be brought to cold shutdown condition at any point of the cycle, even with the strongest control rod withdrawn. In addition to meeting this Technical Specification requirement, the GE reference loading pattern design provides extra margin to cover the known prediction uncertainties, such as the projected critical eigenvalues for Cycle N and the projected end of cycle exposure and its distribution in Cycle N-1 (for reloads).

2.4.1.7 Hot Excess Reactivities

Hot excess reactivity is the margin to criticality for start-up of the core design. Minimum

hot excess reactivity margins are considered in the reference loading pattern design to demonstrate the full power capability.

2.4.1.8 Maximum Fuel Duty Versus Burnup

A limit on fuel rod power as a function of fuel exposure is determined for each fuel design. This limit is specified in the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for each fuel type. This provides confidence that fuel rod thermal mechanical design and licensing limits are not exceeded during normal operation as well as in the event of an Anticipated Operational Occurrence. In the design of the reference loading pattern, margins to this limit are included to account for the known prediction uncertainties which are similar to the consideration for the LHGR.

2.4.1.9 Maximum Core Residence Time

A maximum value for the fuel bundle core residence time is used as an integral part of the fuel bundle thermal mechanical design basis. This maximum residence length is applied primarily in consideration of corrosion and hydriding of the fuel assembly component materials. Excessive corrosion and hydrogen embrittlement in the fuel assembly materials can lead to fuel assembly component failure. The projected cycle operating time is used with the bundle cumulative operating time in the process of selecting exposed bundles for use in the reference core loading.

2.4.1.10 Peak Fuel Pellet Exposure Limit

A maximum value for the individual fuel bundle peak pellet exposure is used as an integral part of the fuel rod thermal mechanical design basis. This limit is selected to be compatible with the planned operating power history of that fuel design and will allow individual batches of reload fuel to achieve their design basis batch average exposure.

2.4.1.11 Special Bundle Shuffle Recommendations

Longitudinal channel bowing results when channels are irradiated in a region of the core having a large gradient in fast neutron flux. This condition primarily exists in the core periphery, or in the nonperipheral control cells. The channel bowing, if large enough, may impact the control rod movement and reduce the expected channel lifetime.

To minimize the potential impact of channel bowing, the following bundle loading and shuffling recommendations are used in the reference core loading design for control cell cores:

- 1) Those bundles which resided in the nonperipheral control cell locations during the previous cycle should not again be used in a nonperipheral control cell location.**
- 2) A fuel bundle should not be shuffled from a peripheral core location to a central core location after one or more cycles on the periphery.**
- 3) Channels that reside in the periphery for more than one cycle should be located so that the channel sides adjoining water will be different from the previous cycle. (When possible channels should reside on the periphery for only one cycle).**

It should be noted that the above requirements usually will not have significant impact on core thermal performance or cycle energy. Should these recommendations become a significant impact on the core design, these restrictions can usually be relaxed to accommodate the core loading based on the balance of all design considerations or a different core loading can be developed.

2.4.1.12 Dynamic Void Coefficient

The dynamic void coefficient is a measure of the response of the reactor to changes in overall void fraction due to power, flow and pressure transients. This parameter is important in estimating system response to transient conditions and gaging reactor power/flow stability. The ABWR plant has been generically analyzed for the range of void coefficients from -2.5 cent/percent to -11.6 cent/percent for transient response and up to 9.0 cent/percent for stability. Values outside this range require special analysis to support establishment of safety limit critical power ratio and power/flow operating window.

2.4.1.13 Doppler Coefficient

The fuel Doppler coefficient is a measure of the response of the reactor to changes in fuel temperature due to power transients. This parameter is important in estimating system response to transient conditions. The ABWR plant has been generically analyzed for the range of Doppler coefficients from -0.216 cent/Kelvin to -0.63 cent/Kelvin for transient response. Values outside this range require special analysis to support establishment of safety limit critical power ratio.

2.5 CORE DESIGN AND PERFORMANCE FOR SPIKING ALTERNATIVES

The general approach utilized in the core and fuel nuclear design for the plutonium spiking option was to insure the bundle minimum exposure which provides a fission product inventory and bundle material activation sufficient to satisfy the dose rate requirements established by DOE. The design of the fuel cycle for this very unique application has as its primary goal the disposition of 100 metric tons of weapons grade plutonium in the shortest possible time. The economic consequences of this operational strategy are not considered.

The reference design presented in this section meets all operational and safety requirements normally imposed for commercial electric power generators. Alternatives which may prove superior in terms of disposition time or economics are discussed briefly.

2.5.1 Reference Spiking Option Core and Fuel Design

The reference spiking design presented here can disposition the required plutonium inventory in less than eight years operation of a single ABWR power plant. A summary of the important fuel cycle parameters and plutonium consumption rates is shown in Table 2.5-1.

Table 2.5-1. Key Parameter Summary, Spiking Case

Cycle Length, EFPD	34.8
Discharge Exposure, MWd/MT	2024
Reload Batch Size, bundles	380
Plant capacity factor (%)	50
Pu Loading kg/bundle	6.80
Pu Consumption rate MT/yr	13.53
MOX fuel rod usage, rods/year	120,000

2.5.1.1 Bundle Design

The bundle design for the spiking option contains plutonium in all sixty power producing rods. This design also has a relatively large number of gadolinia bearing burnable poison rods. This bundle design is particularly flexible. It can be utilized in fuel cycles with very low discharge exposures, as in this spiking option, to moderately high discharge exposures of up to 30 GWD/metric tons. The flat machining profile exhibited by this bundle design as a function of exposure would appear to allow the possibility of a reuse strategy after initial spiking.

The bundle axial and radial enrichment, plutonium and gadolinium distribution is given in Figure 2.5-1. Values of enrichment for uranium are read in hundredths of a percent. For instance, the "020" refers to 0.20 w/o U²³⁵. For the Spiking Option case, only depleted uranium (enrichment plant tails) is used. This allows the plutonium loading to be maximized while maintaining the required reactivity margins. The plutonia concentrations (of isotopic composition given previously) are similarly given in hundredths of weight percent. The

Figure 2.5-1 GE9 Bundle Design For The Reference Spiking Case

minimum plutonia concentration is 1.40 w/o and the maximum is 5.00 w/o PuO_2 . Use is made of only one gadolinia concentration (three w/o). There are a total of six pellet types required for this option, one of which is gadolinia bearing.

The infinite lattice local power peaking analysis is essential in determining the peak power producing rod. The distribution of relative power peaking (normalized to unity across the lattice) is shown at beginning of life for the 40 percent void case in Figure 2.5-2. The maximum infinite lattice relative power peaking factor is shown as a function of exposure for the 0 percent, 40 percent and 70 percent void history cases in Figure 2.5-3.

The exposure-dependent k_{∞} are given in Figure 2.5-4 for the uncontrolled lattice. This figure shows the k_{∞} for three void histories. As seen from Figure 2.5-4, the design of the lattice is quite different from that normally expected in a UO_2 lattice. The reactivity profile is very flat. This is accomplished by using a relatively large fissile loading together with a large number of gadolinia rods. This profile is required in order to maintain a sufficient excess reactivity during a very short cycle.

2.5.1.2 Core Design

A single nuclear design of fuel is loaded into the equilibrium cycle. The important fuel bundle parameters are summarized in Section 2.5.1.1. An inside-to-outside loading and shuffling scheme is utilized with a very large batch fraction. This bundle loading is diagramed in Figure 2.5-5. Only one quarter of the reactor core is illustrated in this view. This loading scheme precluded the use of control cells so that a conventional control blade sequence exchange was employed. The blades are exchanged at the one-fourth, one-half and three-quarter points in the cycle in order to assure an even exposure distribution among the bundles in this very short cycle.

The important parameters of the equilibrium cycle design are summarized in Table 2.5-2. Examination of the results reveals that all thermal and reactivity requirements are satisfied.

1.267	1.213	1.262	1.268	1.268	1.262	1.213	1.267
1.211	1.108	0.673	0.959	0.959	0.673	1.110	1.215
1.260	0.673	0.585	0.640	0.640	0.586	0.674	1.263
1.268	0.959	0.640			0.640	0.961	1.271
1.268	0.959	0.640			0.640	0.961	1.271
1.261	0.673	0.586	0.640	0.640	0.585	0.674	1.264
1.213	1.110	0.674	0.961	0.961	0.674	1.111	1.216
1.267	1.215	1.263	1.271	1.271	1.264	1.216	1.269

Figure 2.5-2 Beginning Of Life Lattice Power Distribution For The Reference Spiking Case

Figure 2.5-3 Lattice Power Peaking For The Reference Spiking Case

Figure 2.5-4 Hot Uncontrolled K_{∞} For The Reference Spiking Case

Figure 2.5-5. Equilibrium Cycle Loading Pattern

Table 2.5-2. Equilibrium Cycle Key Parameter Summary, Spiking Case

Cycle Length, EFPD	34.8
Cycle Energy, GWd	137
Cycle Exposure, MWd/MT	881.8
Core Mass, MT	155.2
Reload Batch Size, bundles	380
Maximum MAPRAT	0.91
Maximum CPRRAT	0.83
Maximum LHGR, KW/ft (LHGR limit = 14.4)	13.1
MCPR (OLMCPR = 1.25)	1.51
Minimum Cold Shutdown Margin (%)	8.6
Hot Excess Reactivity at BOC (%)	3.0

2.5.1.3 Core Thermal Margins

The critical power ratio and MAPLHGR thermal margin performance are plotted as a function of cycle exposure in Figure 2.5-6 and Figure 2.5-7. As explained in Section 2.4.1-8 operation within MAPLHGR limit assures the mechanical integrity of the fuel rods is maintained by limiting their power output in an appropriate manner throughout their lifetime. The MAPLHGR limits imposed on this spiking cycle with a very low discharge exposure are the same as the fuel licensed for up to 38 GWD/MT. Doing so allows the fuel bundles to be considered for reinsert to higher exposure at some future time after the spiking agenda is complete.

2.5.1.4 Core Reactivity Margins

The core minimum shutdown margin and excess reactivity are plotted as a function of cycle exposure in Figure 2.5-8 and Figure 2.5-9. The reactivity margins in this core are very high. The hot-to-cold reactivity window (the sum of hot excess reactivity and shutdown margin) is about 11 percent. This is much higher than typically observed in operation BWRs and is due primarily to the large gadolinium inventory in the core. There is a concern that this large hot-to-cold swing may lead to a large positive moderator temperature coefficient during startup. While a positive temperature coefficient is allowable, it may be an

Figure 2.5-6. Equilibrium Cycle Maximum MAPRAT vs Exposure

Figure 2.5-7. Equilibrium Cycle Maximum CPRRAT vs Exposure

Figure 2.5-8. Equilibrium Cycle Hot Excess Reactivity

Figure 2.5-9. Equilibrium Cycle Minimum Cold Shutdown Margin

operational concern if it is too large.

2.5.1.5 Core Reactivity Coefficients

The core dynamic void coefficient and Doppler coefficients of reactivity are plotted as a function of cycle exposure in Figure 2.5-10 and Figure 2.5-11. The dynamic void coefficient is well within the generic licensing basis of the ABWR. The Doppler coefficient is somewhat larger than the generic upper limit. The significance of this in terms of operational transient behavior is discussed in Section 2.8.

2.5.1.6 Core Performance Details

The control blade positions, axial power and exposure shapes, together with the important performance parameters are shown at five points in the cycle in Figure 2.5-12 through Figure 2.5-16.

2.5.1.7 Spent Fuel Radiation Dose Levels

The initial requirement of the Spiking Option for radiation dose level was based on satisfying 10CFR73.60 and 67. This requirement states that the radiation dose after a two year cool down period must be at least 100 rem/hr at a distance of three feet from all surfaces of the fuel bundle. Analysis of the radiation fields around a BWR fuel bundle show that due to the self shielding associated with the fuel and the upper and lower tie plates it is not possible to achieve this goal directly above and below the bundle without very large exposures. The dose requirement was modified so that the 100 rem/hr level is achieved around the majority of the bundle surface. In particular, this is taken to mean the radial direction over the active fuel length.

The dose field for the fuel bundle discharge with the lowest exposure is diagrammed in Figure 2.5-17. In this diagram a fuel bundle approximately 13 feet in length and six inches wide is represented in the center. Lines of constant dose rate are drawn around it. A

Figure 2.5-10. Equilibrium Cycle Dynamic Void Coefficient

Figure 2.5-11. Equilibrium Cycle Doppler Coefficient

Cycle Exposure, MWD/st	0.0
Cycle Energy, MWD	0.0
Number of Full Power Days	0.0
Core Average Void Fraction	0.435
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.270
Core Axial Power Peak	1.294
Maximum RAPLHGR	0.851
Maximum CPRRAT	0.797
Hot Excess Reactivity, %	3.0%
Cold Shutdown Margin	8.6%

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.5-12. Equilibrium Cycle Core Data Summary at 0 MWD/st

Cycle Exposure, MWD/st	200
Cycle Energy, MWD	34199
Number of Full Power Days	8.7
Core Average Void Fraction	0.433
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.236
Core Axial Power Peak	1.310
Maximum RAPHGR	0.880
Maximum CPRRAT	0.783
Hot Excess Reactivity, %	N/A
Cold Shutdown Margin	N/A

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.5-13. Equilibrium Cycle Core Data Summary at 200 MWD/st

Cycle Exposure, MWD/st	400
Cycle Energy, MWD	68397
Number of Full Power Days	17.4
Core Average Void Fraction	0.439
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.264
Core Axial Power Peak	1.305
Maximum RAPLHGR	0.871
Maximum CPRRAT	0.803
Hot Excess Reactivity, %	2.7%
Cold Shutdown Margin	9.2%

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power

Core Axial Average Exposure, GWD/st

Figure 2.5-14. Equilibrium Cycle Core Data Summary at 400 MWD/st

Cycle Exposure, MWD/st	600
Cycle Energy, MWD	102596
Number of Full Power Days	26.1
Core Average Void Fraction	0.412
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.278
Core Axial Power Peak	1.276
Maximum RAPLHGR	0.909
Maximum CPRRAT	0.826
Hot Excess Reactivity, %	N/A
Cold Shutdown Margin	N/A

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power
Core Axial Average Exposure, GWD/st
Figure 2.5-15. Equilibrium Cycle Core Data Summary at 600 MWD/st

Cycle Exposure, MWD/st	800
Cycle Energy, MWD	136795
Number of Full Power Days	34.8
Core Average Void Fraction	0.414
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.262
Core Axial Power Peak	1.336
Maximum RAPLHGR	0.895
Maximum CPRRAT	0.828
Hot Excess Reactivity, %	2.2
Cold Shutdown Margin	9.6

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power

Core Axial Average Exposure, GWD/st

Figure 2.5-16. Equilibrium Cycle Core Data Summary at 2880 MWD/st

Figure 2.5-17. Radiation Dose Field Around the Spiking Bundle (rem/hr)

rectangular box is drawn around the bundle representing the three foot limit. The area around the bundle in which the dose rate is greater than 100 rem/hr is quite large, extending well beyond three feet.

2.5.1.8 Spent Fuel Characteristics

The decay heat rate on a per bundle basis is shown in Figure 2.5-18. The average isotopic content of the fuel bundles for the reference spiking case is shown in Table 2.5-3. The isotopics are calculated five days after reactor shutdown.

2.5.2 Alternate Spiking Option Designs

The fuel bundle design used in the reference spiking case is a flexible one which allows a wide range of core and fuel cycle designs to be considered. It is in fact the same fuel bundle design used in the reference spent fuel case which is exposed to about 7300 MWD/metric tons. The flat reactivity profile exhibited by this bundle design as it is exposed also allows consideration of "reuse" strategies. In this way plutonium fuel could be spiked in a relatively short time, stored in the reactor and auxiliary storage pools, and then reinserted back into the core to produce electric power. The bundle may be exposed to about 28000 MWD/metric tons in a reuse scenario.

Another alternative fuel design is shown in Figure 2.5-19. This design incorporates both plutonium and enriched uranium in the same bundle. The mixed-oxide fuel pellets in this case contain a much larger plutonium concentration (up to 20 percent) than does the reference design. This large fissile loading is controlled by incorporating large concentrations of gadolinia (up to 8 percent) into the mixed-oxide fuel pellets. The primary advantage of this mixed or island design is that, due to the high plutonium concentrations, fewer mixed-oxide fuel rods need be manufactured to disposition a given quantity of

Figure 2.5-18 Decay Heat Load For The Spiking Case

Figure 2.5-19. – Alternative Pu Spiking Bundle Design

Table 2.5-3. Average Bundle Isotopic Content (grams per bundle)

Spiking Case

Th-232	tr	Kr-83	0.392	Eu-153	0.884
U-233	tr	Ru-101	9.152	Eu-154	0.036
U-234	1.668	Cd-113	0.127	Eu-155	0.328
U-235	327.6	Pm-148G	0.008	Gd-154	37.33
U-236	3.257	Pm-151	0.002	Gd-155	228.4
U-238	170500.	Ru-103	5.030	Gd-156	383.9
Pu-239	6057.	Sm-148	0.078	Gd-157	209.4
Pu-240	517.9	Xe-133	0.647	Gd-158	504.4
Pu-241	64.96	I-135	tr	Tb-159	33.17
Pu-242	0.977	Pr-143	1.711	Gd-160	378.8
Pa-231	tr	Rh-103	5.521	Nd-145	6.783
Np-237	1.247	Rh-105	0.022	Gd tails	0.199
Pu-238	0.284	Cs-133	13.40	Fiss.Prod. Balance	127.1
Np-239	1.619	Xe-135	tr		
Am-241	19.192	Nd-143	8.002		
Am-242	tr	Tc-99	8.162		
Am-243	0.022	Xe-131	6.789		
Th-228	tr	Pm-147	3.811		
Th-230	tr	Pm-148M	0.029		
Cm-242	1.244	Pm-149	0.023		
Cm-244	tr	Sm-147	0.109		
		Sm-149	0.972		
		Sm-150	1.879		
		Sm-151	1.372		
		Sm-152	1.680		

tr – less than 0.0005 grams/bundle

plutonium. However problems associated with the neutron absorption properties of plutonium and gadolinia have been discovered in designs such as this. While gadolinia is primarily a thermal neutron absorber, plutonium has large epithermal resonances. This can result in positive power coefficients of reactivity in some reactor operating regimes. This is an unacceptable condition which must be remedied. It may be possible to use other burnable absorbers, such as europium, which also have epithermal resonances. This was not explored in this study.

2.6 CORE DESIGN AND PERFORMANCE FOR SPENT FUEL ALTERNATIVES

The primary goal of the spent fuel option is to disposition 100 metric tons weapons grade plutonium with a single ABWR plant in 18 operating years. This specific energy output requires an average fuel discharge exposure of 7285 MWD/MT. The fuel bundle design is identical to that of the spiking option.

There are two ways in which one could choose to define "spent fuel". One way would be to base the definition on fuel exposure except that LWRs spent fuel spans a range of exposures. The exposures in BWR and PWR fuels in the 1960's are documented in a USAEC publication, "Design and Performance Data for BWR Fuel Elements and PWR Fuel Elements in Current Status and Future Technical Economic Potential of Light Water Reactors," issued in June 1967. In this compilation the BWR fuel exposures ranged between 2300 and 27000 MWD/MT while PWR fuel exposures ranged between 6600 to 31000 MWD/MT, with the average fuel exposure clustered around 12000 MWD/MT for both reactor types. Fuel cycle economics has dictated progressively longer exposures with the result, it is not uncommon to find light water reactor fuel exposures approaching 45000 MWD/MT. It should however be recognized that achieving today's design exposures, for 100 metric tons of plutonium fuel within the 25 years starting in 1993, would entail building eight ABWRs, all simultaneously, before the year 2001. This is an impractical objective and therefore, it is useful to base the definition of spent fuel on slightly different grounds.

An alternative definition of spent fuel would be to discharge the fuel that meets the

following the criteria: (a) it should be sufficiently radioactive that shielded facilities and sophisticated processing would be needed to get at the plutonium in the fuel; (b) it should be acceptable for the high level waste repository; (c) it should have had an exposure level that is greater than at least some of the spent fuel from commercial reactors, for instance from the 1960s; and finally, (d) the reactor should be capable of being operated at a 75 percent capacity factor in dispositioning the plutonium. This last requirement is very important in that from a reactor operator's perspective, once the fuel bundles have been operated in a "power production mode" and is acceptable at the HLW repository, the discharged fuel would be considered "spent fuel". It is this latter definition that has been adopted in this study, for the plutonium can be dispositioned with a single ABWR within 25 years, by operating in year long cycles with 75 percent capacity factor or higher, with a discharge exposure of 7285 MWD/MT. The rest of this section deals with the nuclear analysis that has been carried out for this reference design.

The reference design presented in this section meets all operational and safety requirements normally imposed for commercial electric power generators. Alternatives which may prove superior in terms of disposition time or economics are discussed briefly.

2.6.1 Reference Spent Fuel Option Core and Fuel Design

The reference spent fuel design presented here can disposition the required plutonium inventory in slightly less than eighteen years operation of a single ABWR power plant. A summary of the important fuel cycle parameters and plutonium consumption rates is shown in Table 2.6-1A.

Table 2.6-1A. Key Parameter Summary, Spent Fuel Case

Cycle Length, EFPD	125.5
Discharge Exposure, MWd/MT	7285
Reload Batch Size, Bundles	380
Plant capacity factor	75%
Pu Loading kg/bundle	6.80
Pu Consumption rate MT/yr	5.64
MOx fuel rod usage, rods/year	50000

2.6.1.1 Bundle Design

The bundle design for the spent fuel option contains plutonium in all sixty power producing rods. This design also has a relatively large number of gadolinia bearing burnable poison rods. This bundle design is particularly flexible. It can be utilized in fuel cycles with very low discharge exposures, as in this spiking option, to moderately high discharge exposures of up to 30 GWD/metric tons.

The bundle axial and radial enrichment, plutonium and gadolinium distribution is given in Figure 2.6-1. Values of enrichment for uranium are read in hundreds of a percent. For instance, the "071" refers to 0.71 w/o U^{235} . For the Spent Fuel case, only depleted uranium is used. This allows the plutonium loading to be maximized while maintaining the required reactivity margins. The plutonia concentrations (of isotopic composition given previously) are similarly given in hundreds of weight percent. The minimum plutonia concentration is 1.40 w/o and the maximum is 5.00 w/o PuO_2 . Use is made of only one gadolinia concentration (three w/o). There are a total of six pellet types required for this option, one of which is a gadolinia bearing rod.

The infinite lattice local power peaking analysis is essential in determining the peak power producing rod. The distribution of relative power peaking (normalized to unity across the lattice) is shown at beginning of life for the 40 percent void case in Figure 2.6-2. The maximum infinite lattice relative power peaking factor is shown as a function of exposure

Figure 2.6-1 GE9 Bundle Design For The Reference Spent Fuel Alternative

1.267	1.213	1.262	1.268	1.268	1.262	1.213	1.267
1.211	1.108	0.673	0.959	0.959	0.673	1.110	1.215
1.260	0.673	0.585	0.640	0.640	0.586	0.674	1.263
1.268	0.959	0.640			0.640	0.961	1.271
1.268	0.959	0.640			0.640	0.961	1.271
1.261	0.673	0.586	0.640	0.640	0.585	0.674	1.264
1.213	1.110	0.674	0.961	0.961	0.674	1.111	1.216
1.267	1.215	1.263	1.271	1.271	1.264	1.216	1.269

Figure 2.6-2 Beginning Of Life Lattice Power Distribution For The Spent Fuel Core

Figure 2.6-3. Lattice Power Peaking For The Reference Spent Fuel Core

Figure 2.6-4 Hot Uncontrolled K_{∞} For The Reference Spent Fuel Core

for the 0 percent, 40 percent and 70 percent void history cases in Figure 2.6-3.

The exposure-dependent k_{∞} are given in Figure 2.6-4 for the uncontrolled lattice. This figure shows the k_{∞} for three void histories. As seen from Figure 2.6-4, the design of the lattice is quite different from that normally expected in a UO_2 lattice. The reactivity profile is very flat. This is accomplished by using a relatively large fissile loading together with a large number of gadolinia rods. This profile is required in order to maintain a sufficient excess reactivity during a relatively short cycle.

2.6.1.2 Equilibrium Core Design

A single nuclear design of fuel is loaded into the equilibrium cycle. The important fuel bundle parameters are summarized in Section 2.6.1.1. An inside-to-outside loading and shuffling scheme was utilized with a very large batch fraction. This bundle loading is diagramed in Figure 2.6-5. This loading scheme precluded the use of control cell so that a conventional control blade sequence exchange was employed. The blades are exchanged at the one-fourth, one-half and three-quarter points in the cycle in order to assure an even exposure distribution among the bundles in this relatively short cycle.

The important parameters of the equilibrium cycle design are summarized in Table 2.6-1B. Examination of the results reveals that all thermal, reactivity and energy requirements are satisfied.

2.6.1.3 Core Thermal Margins

The critical power ratio and MAPLHGR thermal margin performance are plotted as a function of cycle exposure in Figure 2.6-6 and Figure 2.6-7. As explained in Section 2.4.1.8, operation within MAPLHGR limit assures the mechanical integrity of the fuel rods is maintained by limiting their power output in an appropriate manner throughout their lifetime. The MAPLHGR limits imposed on this spiking cycle with a very low discharge exposure are the same as the fuel licensed for up to 38 GWD/metric tons. Doing so allows

Table 2.6.1B. Equilibrium Cycle Key Parameter Summary

Cycle Length, EFPD	125.5
Cycle Energy, GWd	492
Cycle Exposure, MWd/MT	3174.7
Core Mass, MT	155.2
Reload Batch Size, Bundles	380
Maximum MAPRAT	0.89
Maximum CPRRAT	0.84
Maximum LHGR, KW/ft (LHGR limit = 14.4)	12.8
MCPR (OLMCPR = 1.25)	1.48
Minimum Cold Shutdown Margin	8.7%
Hot Excess Reactivity at BOC	2.1%

the fuel bundles to be considered for reinsert to higher exposure at some future time after the spent fuel agenda is complete.

2.6.1.4 Core Reactivity Margins

The core minimum shutdown margin and excess reactivity are plotted as a function of cycle exposure in Figure 2.6-8 and Figure 2.6-9. The reactivity margins in this core are very high. The hot-to-cold reactivity window (the sum of hot excess reactivity and shutdown margin) is about 11 percent. This is much higher than typically observed in operation BWRs and is due primarily to the large gadolinium inventory in the core. There is a concern that this large hot-to-cold swing may lead to a large positive moderator temperature coefficient during startup. While a positive temperature coefficient is allowable, it may be an operational concern if it is too large.

2.6.1.5 Core Reactivity Coefficients

The core dynamic void coefficient and Doppler coefficients of reactivity are plotted as a function of cycle exposure in Figure 2.6-10 and Figure 2.6-11. The dynamic void coefficient

Figure 2.6-6. Equilibrium Cycle Maximum MAPRAT vs Exposure

Figure 2.6-7. Equilibrium Cycle Maximum CPRRAT vs Exposure

Figure 2.6-8. Equilibrium Cycle Hot Excess Reactivity

Figure 2.6-9. Equilibrium Cycle Minimum Cold Shutdown Margin

Figure 2.6-10. Equilibrium Cycle Dynamic Void Coefficient

Figure 2.6-11. Equilibrium Cycle Doppler Coefficient

is well within the generic licensing basis of the ABWR. The Doppler coefficient is somewhat larger than the generic upper limit. The significance of this in terms of operational transient behavior is discussed in Section 2.8.

2.6.1.6 Core Performance Details

The control blade positions, axial power and exposure shapes, together with the important performance parameters are shown at five points in the cycle in Figure 2.6-12 through Figure 2.6-16. The core maps in these figures show the control blade patterns in the core expressed in terms of notches (which are 3-inch sections of blade) withdrawn from the top of the core. Those cells which have no numbers represent cells in which there are no blades inserted. The thermal limits and reactivity margins associated with the given exposure are noted in the summary included with each figure. As seen from these figures, all thermal and reactivity margins are met. The resulting core average power and exposure profile are also given.

2.6.1.7 Spent Fuel Characteristics

The decay heat rate on a per bundle basis is shown in Figure 2.6-17. The average isotopic content of the fuel bundles for the reference spiking case is shown in Table 2.6-2. The isotopics are calculated 5 days after reactor shutdown.

2.6.2 Alternate Spent Fuel Option Designs

The fuel bundle design used in the reference spent fuel case is a flexible one which allows a wide range of core and fuel cycle designs to be considered. It is in fact the same fuel bundle design used in the reference spiking case which is exposed to about 2024 MWD/metric tons. Fuel cycle designs with exposures ranging up to 30000 MWD/metric tons are possible with this bundle. The flat reactivity profile exhibited by this bundle design as it is exposed also allows consideration of "reuse" strategies. In this

Cycle Exposure, MWD/st	0.0
Cycle Energy, MWD	0.0
Number of Full Power Days	0.0
Core Average Void Fraction	0.4229
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.3430
Core Axial Power Peak	1.2493
Maximum RAPLHGR	0.8629
Maximum CPRRAT	0.8231
Hot Excess Reactivity, %	2.715
Cold Shutdown Margin	8.60

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.6–12. Equilibrium Cycle Core Data Summary at 0 MWD/st

Cycle Exposure, MWD/st	720
Cycle Energy, MWD	123115
Number of Full Power Days	31.4
Core Average Void Fraction	0.418
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.256
Core Axial Power Peak	1.243
Maximum RAPLHGR	0.840
Maximum CPRRAT	0.834
Hot Excess Reactivity, %	N/A
Cold Shutdown Margin	N/A

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power

Core Axial Average Exposure, GWD/st

Figure 2.6–13. Equilibrium Cycle Core Data Summary at 720 MWD/st

Cycle Exposure, MWD/st	1440
Cycle Energy, MWD	246230
Number of Full Power Days	62.8
Core Average Void Fraction	0.434
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.260
Core Axial Power Peak	1.279
Maximum RAPLHGR	0.861
Maximum CPRRAT	0.814
Hot Excess Reactivity, %	10.4%
Cold Shutdown Margin	1.75%

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power

Core Axial Average Exposure, GWD/st

Figure 2.6–14. Equilibrium Cycle Core Data Summary at 1440 MWD/st

Cycle Exposure, MWD/st	2160
Cycle Energy, MWD	369345
Number of Full Power Days	94.1
Core Average Void Fraction	0.408
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.257
Core Axial Power Peak	1.332
Maximum RAPLHGR	0.874
Maximum CPRRAT	0.846
Hot Excess Reactivity, %	N/A
Cold Shutdown Margin	N/A

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power

Core Axial Average Exposure, GWD/st

Figure 2.6–15. Equilibrium Cycle Core Data Summary at 2160 MWD/st

Cycle Exposure, MWD/st	2880
Cycle Energy, MWD	492460
Number of Full Power Days	125.4
Core Average Void Fraction	0.438
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.240
Core Axial Power Peak	1.300
Maximum RAPLHGR	0.852
Maximum CPRRAT	0.814
Hot Excess Reactivity, %	1.77
Cold Shutdown Margin	10.5

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power
 Figure 2.6-16. Equilibrium Cycle Core Data Summary at 2880 MWD/st
 Core Axial Average Exposure, GWD/st

Figure 2.6-17 Decay Heat Load For The Spent Fuel Case

Table 2.6-2. Average Bundle Isotopic Content (grams per bundle)

Spent Fuel Case

Th-232	tr	Kr-83	1.407	Eu-153	3.608
U-233	tr	Ru-101	32.95	Eu-154	0.433
U-234	1.564	Cd-113	0.464	Eu-155	0.900
U-235	290.6	Pm-148G	0.030	Gd-154	37.03
U-236	10.33	Pm-151	0.002	Gd-155	144.4
U-238	170000.	Ru-103	6.949	Gd-156	466.6
Pu-239	5255.	Sm-148	1.372	Gd-157	92.2
Pu-240	804.9	Xe-133	0.703	Gd-158	623.1
Pu-241	189.7	I-135	tr	Tb-159	4.35
Pu-242	9.627	Pr-143	1.891	Gd-160	393.6
Pa-231	tr	Rh-103	29.17	Nd-145	24.10
Np-237	4.438	Rh-105	0.024	Gd tails	0.528
Pu-238	2.485	Cs-133	48.90	Fiss.Prod. Balance	457.3
Np-239	1.814	Xe-135	tr		
Am-241	18.31	Nd-143	32.07		
Am-242	tr	Tc-99	29.28		
Am-243	0.516	Xe-131	24.76		
Th-228	tr	Pm-147	12.82		
Th-230	tr	Pm-148M	0.137		
Cm-242	3.033	Pm-149	0.027		
Cm-244	0.030	Sm-147	1.342		
		Sm-149	1.316		
		Sm-150	9.273		
		Sm-151	3.177		
		Sm-152	7.317		

tr – less than 0.0005 grams/bundle

way plutonium fuel could be exposed for a relatively short time, stored in the reactor and auxiliary storage pools, and then reinserted back into the core to produce electric power over a longer period. The bundle may be exposed to about 28000 MWD/metric tons in a reuse scenario.

Preliminary analyses have also been conducted where plutonium is dispositioned in two ABWRs over 18 years. The bundle exposure in this case (without the use of complementary uranium fuel) will be about 14,000 MWD/metric tons, however, reuse beyond this point will not be feasible as the reactivity is now in a region where the profile is no longer flat.

2.7 CORE DESIGN AND PERFORMANCE FOR PLUTONIUM DESTRUCTION ALTERNATIVES

The general approach utilized in the core and fuel nuclear design for the Plutonium Destruction option was to establish the bundle average enrichment to enable the fuel to attain the highest allowable batch average discharge exposure consistent with the thermal-mechanical limits imposed on the GE9 fuel product line which is approximately 38,000 MWd/metric tons. The batch size utilized in this design is 232 bundles in equilibrium (a batch fraction of 26 percent) which results in a batch average discharge exposure of 37.0 GWd/metric tons. The resulting maximum residence time for these bundles is, assuming the reference capacity factor of 75 percent, six years.

This discharge exposure is consistent with the upper range normally associated with uranium spent BWR fuel. The isotopic composition of the plutonium at this discharge exposure is actually more "denatured" for the Plutonium Destruction option than for standard UO_2 fuel. This is shown in Figure 2.7-1. Note that exposure here is given in terms of MWd/st.

The important features of the core design for the reference Plutonium Destruction Option are summarized in the following sections.

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2 of 6

Figure 2.7-1 Plutonium Isotopic Comparison

2.7.1 Reference Plutonium Destruction Option Core and Fuel Design

The reference plutonium destruction design presented here can disposition the required plutonium inventory in just over nineteen years operation of six ABWR power plants. A summary of the important fuel cycle parameters and plutonium consumption rates is shown in Table 2.7-1.

Table 2.7-1. Key Parameter Summary, Pu Destruction Case

Number of reactors	6
Cycle Length, EFPD	392.2
Discharge Exposure, MWd/metric tons	37081
Reload Batch Size, bundles	232
Plant capacity factor	75%
Pu Loading kg/bundle	5.33
Pu Consumption rate metric tons/yr	5.19
MOX fuel rod usage, rods/year	58000

2.7.1.1 Reference Bundle Design

The bundle design for the spent fuel option contains plutonium in all sixty power producing rods. This design also has a relatively large number of gadolinia bearing burnable poison rods. This bundle design closely resembles the reactivity characteristics normally associated with enriched uranium fuel.

The bundle axial and radial enrichment distribution is given in Figure 2.7-2 along with the gadolinia and plutonia concentration distribution. Values of enrichment for uranium are read in hundreds of a percent. For instance, the "071" refers to 0.71 w/o U^{235} . For the Plutonium Destruction case, only natural uranium is used. Two reasons for this are that a slight contribution from the U^{235} aids the reactivity coefficients and natural uranium is thought to be easier to use in the fuel fabrication facility, since quality control on the material is slightly higher for natural than for tails. The plutonia concentrations (of isotopic composition given previously) are similarly given in hundreds of weight percent. The

Figure 2.7-2 GE9 Bundle Design For The Reference Pu Destruction Alternative

minimum plutonia concentration is 1.00 w/o and the maximum is 4.20 w/o PuO_2 . Use is made of only one gadolinia concentration (1 w/o). There are a total of seven pellet types required for this option, three of which are gadolinia rods.

The infinite lattice radial power peaking is essential in determining the peak power producing rod. The distribution of relative power peaking (normalized to unity across the lattice) is shown at beginning of life for the 40 percent void case in Figure 2.7-3. The maximum infinite lattice relative power peaking factor is shown as a function of exposure for the 0 percent, 40 percent and 70 percent void history cases in Figure 2.7-4.

The exposure-dependent k_{∞} are given in Figure 2.7-5 for the uncontrolled lattice. This figure shows the k_{∞} for three void histories. As seen from Figure 2.7-5, the design of the lattice is such as to emulate a UO_2 lattice. This is accomplished by means of the widely dispersed gadolinia in the interior rods.

2.7.1.2 Equilibrium Core Design

The equilibrium core design philosophy for the Plutonium Destruction option was to simulate the reactivity distribution of an equilibrium UO_2 core in order provide simple operation. Also, the fuel fissile inventory was matched to the residence time of each batch in order to minimize any discharged plutonium inventory. As stated previously, the target discharge exposure is the maximum allowed by the thermal mechanical limits on the fuel. A single fuel nuclear design was utilized in an equilibrium batch of 232 bundles. The detailed core design layout is presented in Figure 2.7-6. The numbers shown in the beginning-of-equilibrium-cycle core map represent the relative number of the cycle since fuel loading. For instance, the number "1" refers to fresh fuel (loaded this cycle) and the number "4" refers to bundles which are about to start their fourth cycle.

A single nuclear design of fuel is loaded into the equilibrium cycle. The important fuel bundle parameters were summarized previously. A control cell core loading strategy that

1.302	1.273	1.287	1.308	1.309	1.288	1.274	1.305
1.273	0.788	0.706	0.760	0.761	0.706	0.789	1.276
1.287	0.706	0.697	0.763	0.763	0.697	0.707	1.290
1.308	0.760	0.763			0.763	0.761	1.311
1.309	0.761	0.763			0.764	0.761	1.312
1.288	0.706	0.697	0.763	0.764	0.697	0.707	1.291
1.288	0.789	0.707	0.761	0.761	0.707	0.789	1.277
1.305	1.276	1.290	1.311	1.312	1.291	1.277	1.307

Figure 2.7-3 Beginning Of Life Lattice Power Distribution For The Reference Pu Destruction Case

Figure 2.7-4 Lattice Power Peaking For The Reference Pu Destruction Case

Figure 2.7-5 Hot Uncontrolled K_{∞} For The Reference Pu Destruction Case

Cycle Loaded {
 1 = Fresh Fuel
 2 = Second Cycle
 3 = Third Cycle
 4 = Fourth Cycle

Core Mass = 156.075 MT

Figure 2.7-6 Equilibrium Cycle Loading Pattern

contains 37 control cells was utilized. Due to the improved hot to cold reactivity swing characteristics of the ABWR core, it was possible to design the fuel with a large cold shutdown margin and still maintain sufficient hot excess reactivity. The hot excess reactivity dictated the use of 37 control cells.

The important parameters of the equilibrium cycle design are summarized in Table 2.7-2. Examination of the results reveals that all thermal, reactivity and energy requirements are satisfied.

**Table 2.7-2. Equilibrium Cycle Key Parameter Summary,
Plutonium Destruction Case**

Cycle Length, EFPD	392
Cycle Energy, GWd	1,539
Cycle Exposure, MWd/metric tons	9861
Core Mass, metric tons	149.1
Reload Enrichment, w/o U-235	3.70
Reload Batch Size, bundles	232
Maximum MAPRAT	0.88
Maximum CPRRAT	0.77
Maximum LHGR, KW/ft (LHGR limit = 14.4)	13.0
MCPR (OLMCPR = 1.25)	1.63
Minimum Cold Shutdown Margin	1.9
Hot Excess Reactivity at BOC	0.5

2.7.1.3 Core Thermal Margins

The critical power ratio and MAPLHGR thermal margin performance are plotted as a function of cycle exposure in Figure 2.7-7 and Figure 2.7-8. As explained in Section 2.4.1-8, operation within MAPLHGR limit assures the mechanical integrity of the fuel rods is maintained by limiting their power output in an appropriate manner throughout their lifetime. The MAPLHGR limits imposed on this spiking cycle with a very low discharge exposure are the same as the fuel licensed for up to 38 GWD/metric tonnes. These results demonstrate ample margin to core thermal limits.

Figure 2.7-7 Equilibrium Cycle Maximum MAPRAT vs Exposure

Figure 2.7-8 Equilibrium Cycle Maximum CPRRAT vs Exposure

2.7.1.4 Reactivity Limit Summary

The reactivity performance of the plutonium destruction option design is summarized in Figure 2.7-9 and Figure 2.7-10. Due to the improved hot to cold reactivity swing of the ABWR N-lattice, there is abundant cold shutdown margin; therefore, there is little or no impact of the mixed-oxide fuel utilization on core design from cold shutdown margin considerations.

2.7.1.5 Reactivity Coefficients

The core dynamic void coefficient and Doppler coefficients of reactivity are plotted as a function of cycle exposure in Figure 2.7-11 and Figure 2.7-12. The dynamic void coefficient is somewhat larger than the upper generic limit for ABWR while the Doppler coefficient is within the generic limits. The significance of these results is discussed in detail in Section 2.8.

2.7.1.6 Core Performance Description

The core performance characteristics as a function of exposure through the cycle are given in Figure 2.7-13 through Figure 2.7-18. The core maps in these figures show the control blade patterns in the core expressed in terms of notches (which are 3-inch sections of blade) withdrawn from the top of the core. Those cells which have no numbers represent cells in which there are no blades inserted. The thermal limits and reactivity margins associated with the given exposure are noted in the summary included with each figure. As seen from these figures, all thermal and reactivity margins are met. The resulting core average power and exposure profile are also given. Typical of the ABWR, the power profile shifts towards the top of the core during the last quarter of the cycle. Since the reactor's core design itself provides sufficient margins, it is not necessary to axially grade the fuel assembly to accommodate the shift in power.

Figure 2.7-9 Equilibrium Cycle Hot Excess Reactivity

Figure 2.7-10 Equilibrium Cycle Minimum Cold Shutdown Margin

Figure 2.7-11 Equilibrium Cycle Dynamic Void Coefficient

Figure 2.7-12 Equilibrium Cycle Doppler Coefficient

Cycle Exposure, MWD/st	0.0
Cycle Energy, MWD	0.0
Number of Full Power Days	0.0
Core Average Void Fraction	0.4214
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.3476
Core Axial Power Peak	1.1928
Maximum RAPHGR	0.7830
Maximum CPRRAT	0.7374
Hot Excess Reactivity, %	0.83
Cold Shutdown Margin	5.18

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.7-13 Equilibrium Cycle Core Data Summary at 0 MWD/st

Cycle Exposure, MWD/st	2000.0
Cycle Energy, MWD	344082
Number of Full Power Days	87.6
Core Average Void Fraction	0.4301
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.3015
Core Axial Power Peak	1.2005
Maximum RAPLHGR	0.8272
Maximum CPRRAT	0.7119
Hot Excess Reactivity, %	N/A
Cold Shutdown Margin	N/A

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.7-14 Equilibrium Cycle Core Data Summary at 2000 MWD/st

Cycle Exposure, MWD/st	4000.0
Cycle Energy, MWD	688164
Number of Full Power Days	175.3
Core Average Void Fraction	0.4341
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.3581
Core Axial Power Peak	1.2133
Maximum RAPHGR	0.8725
Maximum CPRRAT	0.7166
Hot Excess Reactivity, %	1.46
Cold Shutdown Margin	4.95

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.7-15 Equilibrium Cycle Core Data Summary at 4000 MWD/st

Cycle Exposure, MWD/st	6000.0
Cycle Energy, MWD	1032245
Number of Full Power Days	262.9
Core Average Void Fraction	0.4460
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.3532
Core Axial Power Peak	1.2820
Maximum RAPLHGR	0.8823
Maximum CPRRAT	0.6797
Hot Excess Reactivity, %	N/A
Cold Shutdown Margin	N/A

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.7-16 Equilibrium Cycle Core Data Summary at 6000 MWD/st

Cycle Exposure, MWD/st	8000.0
Cycle Energy, MWD	1376327
Number of Full Power Days	350.6
Core Average Void Fraction	0.3819
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.3895
Core Axial Power Peak	1.2303
Maximum RAPLHGR	0.8666
Maximum CPRRAT	0.7663
Hot Excess Reactivity, %	0.58
Cold Shutdown Margin	4.59

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.7-17 Equilibrium Cycle Core Data Summary at 8000 MWD/st

Cycle Exposure, MWD/st	8946
Cycle Energy, MWD	1539000
Number of Full Power Days	392.0
Core Average Void Fraction	0.3738
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.3994
Core Axial Power Peak	1.2464
Maximum RAPLHGR	0.7967
Maximum CPRRAT	0.7524
Hot Excess Reactivity, %	0.0
Cold Shutdown Margin	3.90

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 2.7-18 Equilibrium Cycle Core Data Summary at End Of Cycle

Figure 2.7-19 Decay Heat Load For The Pu Destruction Case

2.7.1.7 Spent Fuel Characteristics

The decay heat rate on a per bundle basis is shown in Figure 2.7-19. The average isotopic content of the fuel bundles for the reference spiking case is shown in Table 2.7-3. The isotopics are calculated five days after reactor shutdown.

**Table 2.7-3. Average Bundle Isotopic Content (grams per bundle)
Plutonium Destruction Case**

Th-232	tr	Kr-83	7.463	Eu-153	24.808
U-233	tr	Ru-101	168.020	Eu-154	9.074
U-234	5.820	Cd-113	2.494	Eu-155	3.465
U-235	382.964	Pm-148G	0.085	Gd-154	18.709
U-236	144.963	Pm-151	0.002	Gd-155	0.094
U-238	168000	Ru-103	6.601	Gd-156	332.846
Pu-239	1414.876	Sm-148	24.027	Gd-157	0.103
Pu-240	1185.369	Xe-133	0.686	Gd-158	390.969
Pu-241	492.230	I-135	tr	Tb-159	14.568
Pu-242	223.414	Pr-143	1.936	Gd-160	210.204
Pa-231	tr	Rh-103	116.481	Nd-145	115.947
Np-237	27.778	Rh-105	0.023	Gd tails	3.704
Pu-238	31.542	Cs-133	222.921	Fiss.Prod.	
Np-239	2.321	Xe-135	tr	Balance	2378.898
Am-241	28.525	Nd-143	133.578		
Am-242	tr	Tc-99	139.838		
Am-243	43.571	Xe-131	90.899		
Th-228	tr	Pm-147	29.849		
Th-230	tr	Pm-148M	0.243		
Cm-242	10.085	Pm-149	0.035		
Cm-244	13.062	Sm-147	17.438		
		Sm-149	0.519		
		Sm-150	54.718		
		Sm-151	3.011		
		Sm-152	34.573		

tr - less than 0.0005 grams/bundle

2.7.1.8 Alternate Plutonium Destruction Option Designs

The limitation on the reference Plutonium Destruction Option is directly linked to the amount of fertile material present in the fuel. Fertile material, such as ^{238}U , acts as a parasitic absorber relative to the plutonium in the fuel. On the one hand, this parasitic absorption results in power being generated by isotopes which are not plutonium and therefore limit, for the same power density, the amount of plutonium which can be consumed. Also, ^{238}U undergoes neutron capture to ultimately produce additional plutonium. This further reduces the effectiveness of the LWR for destroying plutonium.

Use of a non-fertile fuel matrix, such as zirconia, solves this problem by allowing a high temperature, neutron-inert component to supply the binding material for the relatively dilute concentrations of plutonia required for this loading option. The zirconia has no appreciable cross-section for neutrons. However, the lack of parasitic absorptions in the epithermal energy region, which were previously provided by the ^{238}U , result in very positive void power coefficients, which are entirely untenable in a BWR. This form of fuel matrix has been used in the past, but additional absorber material is provided through chemical reactivity shim and other fixed control elements within the fuel lattice. Such use of control elements, while perhaps feasible, are beyond the scope and range of this study. Possible future effort might identify another matrix material which would provide sufficient epithermal absorption to mitigate the problem of positive void coefficients.

2.7.1.9 Development Requirements

The use of plutonia in light water reactors, in particular BWRs, has already been subjected to initial operational scrutiny, as previously mentioned. The plutonium destruction option does not deviate very far from this prior experience. The analytical tools required for design of the fuel and fuel for this option have already been benchmarked against critical experiment, plutonia bearing fuel being amongst those elements benchmarked. Although refinements to these existing tools may be desirable to provide DOE with additional required tracking and information, such effort does not fall under the category of

development requirements.

One of the unique features of this fuel design is the use of gadolinia within the plutonia rods. Although this has been accomplished on a laboratory scale, the large scale production of such gadolinia rods has not been accomplished. Although no development requirements are necessary for the core and fuel nuclear design, examination of this issue is a candidate for development in the area of fuel manufacture. Nuclear design would again play a role in the development of a Lead Use Assembly which might be used to validate the selected gadolinia/plutonia fuel fabrication technique.

2.8 TRANSIENT AND ACCIDENT ANALYSIS

The Standard Safety Analysis Report (SSAR) for the ABWR was submitted to the NRC in 1987. The NRC review is approaching completion. All major technical issues are resolved (SECY-89-153 and SECY-90-016) and approved by the NRC on June 26, 1990. Final design approval was expected but was rescheduled for 1994.

The ABWR was designed for use of generic fuel. Therefore, much of the available behavioral analyses for the ABWR, including the SSAR, are applicable to the present study. In that work, GE has evaluated the entire spectrum of events in nuclear safety and operational analysis areas to establish the most limiting or design basis events in a meaningful manner. The scope of the situations analyzed includes anticipated operational occurrences, off-design abnormal (unexpected) transients that induce system operational disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and finally, hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire control rod drive system). In the event analysis, all essential protection sequences were evaluated until all required safety actions were successfully completed. The event analysis identified front line safety systems and their essential auxiliaries, operator actions, and limits to satisfy the required safety actions. A partial list of the events examined is given below.

- **Events that Decrease Core Coolant Temperature**
 - **Loss of Feedwater Heating**
 - **Runout of One Feedwater Pump**
 - **Feedwater Controller Failure to Maximum Demand**
 - **Opening of One Bypass Valve**
 - **Opening of all Control and Bypass Valves**
 - **Inadvertent Opening of One Safety-Relief Valve**
 - **Inadvertent RHR Shutdown Cooling**

- **Events That Increase Reactor Pressure**
 - **Fast Closure of One Turbine Control Valve**
 - **Slow Closure of One Turbine Control Valve**
 - **Pressure Regulator Downscale Failure**
 - **Generator Load Rejection with Bypass On**
 - **Generator Load Rejection with Failure of One Bypass Valve**
 - **Generator Load Rejection with Failure of All Bypass Valves**
 - **Turbine Trip with Bypass On**
 - **Turbine Trip with Failure on One Bypass Valve**
 - **Turbine Trip with Failure of All Bypass Valves**
 - **Inadvertent MSIV Closure**
 - **Loss of Condenser Vacuum**
 - **Loss of AC Power**
 - **Loss of All Feedwater Flow**
 - **Feedwater Piping Break**
 - **Failure of RHR Shutdown**

- **Events That Decrease Reactor Coolant System Flow Rate**
 - **Trip of Three Reactor Internal Pumps**
 - **Trip of All Reactor Internal Pumps**
 - **Fast Runback of One Reactor Internal Pump**

- Fast Runback of All Reactor Internal Pumps
- Seizure of One Reactor Internal Pump
- One Pump Shaft Break

- Reactivity and Power Distribution Anomalies
 - Rod Withdrawal Error During Refueling
 - Rod Withdrawal Error During Startup
 - Rod Withdrawal Error at Power
 - Control Rod Misoperation
 - Abnormal Startup of One Reactor Internal Pump
 - Fast Runout of One Reactor Internal Pump
 - Misplaced Bundle Accident
 - Rod Ejection Accident
 - Control Rod Drop Accident

- Events that Increase Reactor Coolant Inventory
 - Inadvertent HPCF Startup

The ABWR response to transients is reported in Chapter 15 of the SSAR. Response to design basis accidents is reported in Chapter 6. Finally, compliance with the ASME code for overpressure events is reported in Chapter 5. In developing the ABWR two bases were used for the analysis -- design basis and licensing basis. For the design basis of the Nuclear Boiler GE chose the most conservative core design that was known at the time, including the possibility of the use of mixed-oxide fuel in the future. This was the so-called Core Z design used in the plant design development. In the licensing basis, reported in the SSAR for limiting transients, GE used a reference fuel design, typical of today's UO₂ fuel offerings, called Core A. Thus, as alternative mixed-oxide fuel designs used in this present study were developed, GE already had a guide as to which transients and accidents would be the most limiting, and whether there might be problems accommodating the designs.

Based on the alternatives developed in Sections 2.4-2.7, the core design with the most

limiting nuclear parameters was chosen for further analysis. Table 2.8-1 compares the nuclear dynamic parameters of Core A, Core Z, and those of the high burnup mixed-oxide core at the end of cycle, which is the most limiting point for pressurization transients.

Table 2.8-1. Comparison of Nuclear Dynamic Parameters

Parameter	Core A	Core Z	High Burnup Core
Void Coefficient, $\epsilon/\%$	-8.4	-11.6	-11.2
Doppler coefficient, $\epsilon/^\circ\text{C}$	-0.31	-0.43	-0.63

2.8.1 Transient Response

2.8.1.1 Determination of the Plant Operating Limit

The purpose of the transient response analysis is to set plant operating limits to avoid the possibility of departure from nucleate boiling for events expected to occur during the plant lifetime. For BWRs this has been measured by a parameter called Minimum Critical Power Ratio (MCPR), which is the ratio of the power at which a departure from nucleate boiling is expected to occur on the hottest rod in a fuel bundle to the bundle's current power level. Starting from the nominal situation, i.e. $\text{MCPR} = 1.0$, it is necessary to factor in statistical considerations which account for manufacturing and measurement tolerances, including uncertainties in the thermal/hydraulic correlations developed by test programs, to measure the departure from nucleate boiling. This sets what is called the Safety Limit MCPR (SLMCPR). For the ABWR, this has been calculated to be 1.07, meaning that if the most limiting fuel bundle in the core is predicted to have an MCPR of 1.07, there is a 0.1 percent probability that a rod will have experienced a departure from nucleate boiling. That is the acceptance criterion approved by the U.S. NRC for BWRs.

Next, all possible transients which can occur (generally due to a single equipment failure or single operator error) are analyzed to determine the most limiting one in terms of the change of critical power ratio which can occur. This so-called ΔCPR is added to the

SLMCPR to determine the plant Operating Limit MCPR (OLMCPR). The steady-state fuel performance analysis is then compared to the OLMCPR to determine the amount of operating margin.

In the prior studies of ABWR with Core Z and Core A, GE determined that the most limiting transient is the Load Rejection with Bypass Failure (LRWOBP) event. Even though this event involves multiple failures, a long-standing GE-USNRC agreement exists to consider this event as one of those in Chapter 15 to be compared to the MCPR safety criterion. The primary nuclear dynamic parameter driving this transient is the void coefficient, since it is the void collapse from the primary system pressurization which results from suddenly shutting of the steam heat sink which causes the power rise in the core. In the Chapter 15 analyses, pressurization transients were analyzed with GE's one-dimensional kinetics code, ODYN, which is capable of following the effects of the traveling pressure wave through the core as the event progresses. In this code, basic cross-sections from the three-dimensional steady-state core analysis are used, so it is not possible to apply conservative multipliers to the key parameters – Void Coefficient and Doppler Coefficient. Other transients were analyzed using the point model kinetics code REDY. While this does not compare all transients on an “apples-to-apples” basis, because of the background studies showing the pressurization transients to be limiting, the OLMCPR will be correctly computed. The net effect of the Chapter 15 analyses reported in the ABWR SSAR is that pressurization transients use the dynamic parameters derived from Core A, while the others use the more limiting Core Z. Table 2.8-2 shows the results of the three most limiting SSAR analyses all of which become pressurization-type transients.

Table 2.8-2. Results of Transient Analyses

Transient Event Description	<u>Delta Critical Power Ratio</u>	
	SSAR Result	High Burnup Core
Generator Load Rejection with Failure of all Bypass Valves	0.10	0.19
Fast Closure of One Turbine Control Valve	0.10	0.10
Feedwater Controller Failure - Maximum Demand	0.10	0.16

Comparing the parameters of Table 2.8-1 confirms that the LRWOBP event will be the limiting transient for the limiting mixed-oxide core, based on the background studies. Therefore, it was the key one analyzed to determine the OLMCPR and available operating margin for this study. Since two other events also showed comparable Δ critical power ratios for the UO_2 core, they were also repeated, using ODYN, for this study. Table 2.8-3 lists the input conditions used for the transient analyses. These are identical to those used in the SSAR.

Table 2.8-3 Input Parameters and Initial Conditions for System Response Transients

1. Thermal Power Level (MWt) Warranted Value Analysis Value	3926 4005
2. Steam Flow (kg/hr) Warranted Value Analysis Value	7.64x10 ⁶ 7.84x10 ⁶
3. Core Flow (kg/hr) Rated Maximum	52.2x10 ⁶ 59.0x10 ⁶
4. Feedwater Flow Rate (kg/sec) Warranted Value Analysis Value	2122 2179
5. Feedwater Temperature (Celsius)	217
6. Vessel Dome Pressure (kg/cm ² g)	73.1
7. Vessel Core Pressure (kg/cm ² g)	73.7
8. Turbine Bypass Capacity (%NBR)	33
9. Core Coolant Inlet Enthalpy (kcal/kg)	294.1
10. Turbine Inlet Pressure(kg/cm ² a)	69.9
11. Fuel Lattice	N
12. Core Leakage Flow (%)	11.67
13. MCPR Safety Limit	1.07
14. Nuclear Characteristics Used in ODYN Simulations	EOEC*
15. Number of Reactor Internal Pumps	10
16. Safety/Relief Valve Capacity (%NBR) at 80.5 kg/cm ² g	91.3
17. Relief Function Delay (sec)	0.4
18. Relief Function Opening Time (sec)	0.15
19. Safety Function Delay (sec)	0
20. Safety Function Opening Time (sec)	0.3

Table 2.8-3 (continued)

21. Set Points for Safety/Relief Valves Safety Function Relief Function	82.8, 83.5, 84.2, 84.9, 85.6 80.5, 81.2, 81.9, 82.6, 83.3, 88.4
22. Safety/Relief Valve Reclosure Setpoint - Both Modes (% of Setpoint) Maximum Safety Limit (used in analysis) Minimum Operational Limit	 98 93
23. High Flux Trip (%NBR)	127.5
24. High Pressure Scram Setpoint (kg/cm²g)	77.7
25. Vessel Level Trips (m above bottom of separator skirt) Level 8 (m) Level 4 (m) Level 3 (m) Level 2 (m)	 1.73 1.08 0.57 -0.75
26. APRM Simulated Thermal Power Trip (%NBR) Analysis Setpoint Time Constant (sec)	 117.3 7
27. Reactor Internal Pump Trip Delay (sec)	0.16
28. RIP Trip Inertia Time constant for Analysis (sec)	0.62
29. Total Steamline volume (m³)	113.2
30. Set Pressure of RIP Trip (kg/cm²g)	79.1

Table 2.8-4. Sequence of Events for the LRWOBP

Event	Time (seconds)
Turbine-Generator Detection of Loss of Electrical Load	-0.015
Turbine-Generator Load Rejection Sensing Devices Trip to Initiate Turbine control Valves Fast closure	0.0
Turbine Bypass Valves Fail to Operate	0.0
Fast Control Valve Closure Initiates Reactor Scram and Trip of Four RIPs	0.0
Turbine Control Valves Closed	0.07
Safety/Relief Valves Open Due to High Pressure	1.3
Safety/Relief Valves close	7.1
Safety/Relief Valves Open Again to Relieve Decay Heat	8.9
Safety/Relief Valves Close Again	>15.0 (est.)

Table 2.8-5. Sequence of Events for the Fast Closure of One Turbine Control Valve

Event	Time (seconds)
Simulate One Main Turbine Control Valve to close	0
Failed Turbine Control Valve Starts to Close	0.0
Turbine Bypass Valves Start to Open	2.7
Neutron Flux Reaches High Flux Scram Setpoint and Initiates a Reactor Scram	2.84
Water Level Reached Level 3 Setpoint. Four RIP's are Tripped	7.65

Table 2.8-6. Sequence of Events for the Runout of all Feedwater Pumps

Event	Time (Seconds)
Initiate Simulated runout of all Feedwater Pumps (130% at System Design Pressure of 74.9 kg/cm ² g on Feedwater Flow)	0
Level 8 Vessel Level Setpoint Initiates Trip of Main Turbine and Feedwater Pumps	18.45
Reactor Scram and Trip of Four RIP's are Actuated by Stop Valve Position Switches	18.46
Main Turbine Bypass Valves Opened Due to Turbine Trip	18.6
SRV's Open Due to High Pressure	20.2
Safety/Relief Valves Close	>25
Water Level Dropped to Low Water Level Setpoint (Level 2)	>40 (est.)
RCIC Flow into Vessel (not simulated)	>70 (est.)

Tables 2.8-4 to 2.8-6 show the key events occurring during the transients and Figures 2.8-1A to 2.8-3D graphically display the time variation of key plant parameters. The Δ CPRs from these events are given in Table 2.8-2. The most limiting event is the LRWOBP as expected; the limiting delta-CPR is calculated to be 0.19; therefore the OLMCPR would be set at 1.26 for this core design. Based on the steady-state results reported in Table 2.7-2, the minimum operating margin would be 23 percent (CPRRAT=0.77 with OLMCPR=1.26 and MCPR=1.63). GE has typically supplied fuel to currently operating BWRs with operating margins as little as seven percent; therefore this design is acceptable.

The size of the margin (23 percent) for MCPR is still significantly larger than the 15 percent margin requirements applied per section 2.4.1. This is the case even with the handicap of the additional 0.09 delta-CPR compared to the ABWR SSAR result for the LWROBP transient (Table 2.8-2). This significant margin has been maintained because this plutonium disposition study is based upon the GE9 fuel design. This fuel design shows significant performance advantages over the fuel design on which the ABWR SSAR is based, GE8.

Figure 2.8-1a Load Rejection With Bypass Failure

- 1 = LEVEL (INCH REF SEP-SKIRT)
- 2 = W R SENSED LEVEL (INCHES)
- 3 = N R SENSED LEVEL (INCHES)
- 4 = CORE INLET FLOW (%)
- 5 = PUMP FLOW 3 (%)

Figure 2.8-1b Load Rejection With Bypass Failure

- 1 = VESSEL PRESSURE RISE (PSI)
- 2 = STEAM LINE PRESSURE RISE (PSI)
- 3 = TURBINE PRESSURE RISE (PSI)
- 4 = RELIEF VALVE FLOW (%)
- 5 = BYPASS VALVE FLOW (%)
- 6 = TURBINE STEAM FLOW (%)

Figure 2.8-1c Load Rejection With Bypass Failure

Figure 2.8-1d Load Rejection With Bypass Failure

- 1 = NEUTRON FLUX
- 2 = PEAK FUEL CENTER TEMP
- 3 = AVERAGE SURFACE HEAT FLUX
- 4 = FEEDWATER FLOW
- 5 = VESSEL STEAM FLOW

Figure 2.8-2a Fast Closure of One Turbine Control Valve

- 1 = LEVEL (INCH REF SEP-SKIRT)
- 2 = W R SENSED LEVEL (INCHES)
- 3 = N R SENSED LEVEL (INCHES)
- 4 = CORE INLET FLOW (%)
- 5 = PUMP FLOW 3 (%)

Figure 2.8-2b Fast Closure of One Turbine Control Valve

- 1 = VESSEL PRESSURE RISE (PSI)
- 2 = STEAM LINE PRESSURE RISE (PSI)
- 3 = TURBINE PRESSURE RISE (PSI)
- 4 = RELIEF VALVE FLOW (%)
- 5 = BYPASS VALVE FLOW (%)
- 6 = TURBINE STEAM FLOW (%)

Figure 2.8-2c Fast Closure of One Turbine Control Valve

Figure 2.8-2d Fast Closure of One Turbine Control Valve

Figure 2.8-3a Run-up of All Feedwater Pumps

- 1 = LEVEL (INCH REF SEP-SKIRT)
- 2 = W R SENSED LEVEL (INCHES)
- 3 = N R SENSED LEVEL (INCHES)
- 4 = CORE INLET FLOW (%)
- 5 = PUMP FLOW 3 (%)

Figure 2.8-3b Run-up of All Feedwater Pumps

- 1 = VESSEL PRESSURE RISE (PSI)
- 2 = STEAM LINE PRESSURE RISE (PSI)
- 3 = TURBINE PRESSURE RISE (PSI)
- 4 = RELIEF VALVE FLOW (%)
- 5 = BYPASS VALVE FLOW (%)
- 6 = TURBINE STEAM FLOW (%)

Figure 2.8-3c Run of All Feedwater Pumps

Figure 2.8-3d Run-up of All Feedwater Pumps

The large MCPR margin for the high burnup mixed-oxide fuel is the result of using the GE9 design.

2.8.1.2 Stability

From a safety point of view, the ABWR has provided protection from reactor instabilities which might cause fuel damage. In BWRs density-wave oscillations have been observed under test and operating conditions as the reactor power and flow are lowered along the rated rod line. With this in mind, the ABWR has provided protection by excluding a zone of operation in the power-flow map (see Figure 2.8-4). During power ascension by pulling control rods, if the operator attempts to increase power above 25 percent with flow less than 40 percent (30 percent pump speed) an automatic rod block is issued by the control system which prevents further rod withdrawal until the minimum pump speed is achieved. During transients, if two or more RIPs are tripped and the flow becomes less than 40 percent with power greater than 25 percent, a Select Control Rod Run-In (SCRRI) command is issued by the control system to insert rods until the power becomes less than 25 percent.

The basis of the above protection is to prevent the plant operators from getting the plant into an operating state where instabilities have been known to occur. In spite of the above protection, however, the ABWR also monitors the local power through the LPRMs and will issue a scram if significant oscillations are detected.

The basis of the minimum pump speed limit has been determined from stability analyses demonstrating a 95 percent confidence that oscillations will not occur at or above this point. From many tests and analyses, GE has determined the primary drivers for instability. Once again, the void coefficient is the most significant parameter for the purpose of the mixed-oxide studies. Therefore, stability limits were checked to see if further restrictions in the operating map would be necessary.

Stability is calculated by the use of frequency-domain codes which linearize the basic dynamic equations for small perturbations and report the behavior of the reactor in terms

of a second-order feedback system. Commonly, a decay ratio is reported; this is the ratio of the second overshoot to the first overshoot of the reactor response to a step change in input. Thus a decay ratio <1.0 means the reactor is stable, and a decay ratio >1.0 means the reactor is unstable. In calibration of GE's methods to test data, a 20 percent conservatism is needed to achieve a 95 percent confidence of avoiding instability; thus it is required to predict the decay ratio <0.8 . The least stable point in the operating domain of Figure 2.8-4 is the intersection of the rated rod line with the minimum pump speed line. Thus, stability of the high burnup core was calculated, using the same procedures and methods documented in the SSAR for Core Z. The most limiting void coefficient for the mixed-oxide core was taken from the mid-cycle point (-12.9 cents/percent, per Figure 2.7-11).

The results are shown in Table 2.8-7. It can be seen that at the current minimum speed (30 percent), i.e. core flow of 40 percent the decay ratio = 0.86. Therefore, a second case was run with a slightly higher speed in order to determine how much further restriction would be necessary. Referring to Table 2.8-7, it can be seen that a nominal increase in required minimum pump speed will satisfy the stability criteria. This will not present any significant operating restriction.

Table 2.8-7 Stability Evaluations

	Core Decay Ratio	Hot Channel Decay Ratio
30% Speed/40% Flow	0.86	0.39
35% Speed/45% Flow	0.73	0.17
Design Criteria	<0.80	<0.60

2.8.1.3 Anticipated Transients Without Scram (ATWS)

ATWS events are a special class of transients also analyzed in Chapter 15 of the SSAR. Because of the low probability of occurrence, ATWS acceptance criteria are somewhat different than those for normal transients. These include:

Figure 2.8-4 Modified Power-Flow Operating Map For ABWR With MOX Fuel

- a. Core coolability, determined by demonstrating fuel temperatures <2200 Fahrenheit
- b. Nuclear Boiler peak pressure not to exceed service Level C of the ASME Boiler Code, i.e. 1500 psig.
- c. Containment pressure not to exceed service Level A, i.e. 45 psig

In the SSAR, ATWS analyses were reported showing significant margins to the acceptance limits. However, background studies for ABWR have shown that the peak pressure predicted for ATWS pressurization transients is within 20 psid of that for the ASME overpressure transient (reported below). Thus ATWS is not a limiting overpressure basis for ABWR.

2.8.2 Overpressure Analysis

In BWRs the number of safety/relief valves is normally set by requirements of the ASME code for overpressure protection. This analysis is a special transient in which the MSIVs are suddenly closed, the normal MSIV position scram is assumed to fail – the reactor scrams on high neutron flux. In addition no credit is given for safety/relief valves opening in the power-actuated relief mode; credit is given only for the spring mode. The acceptance criterion for this analysis is Nuclear Boiler peak pressure less than service Level B of the ASME code, i.e. less than 1375 psig. Since this is an overpressure transient, the void coefficient will once again play a significant role in the results. Therefore, this analysis was repeated for the limiting mixed-oxide core.

Results show that the peak vessel pressure is <1272 psig, well within the acceptance criterion. Even for a further delayed case using the second backup scram (high pressure scram) the result is 1284 psig.

2.8.3 Accident Analyses

Response to design basis accidents is reported in Chapter 6 of the SSAR for Loss-of-Coolant Accidents (LOCAs) and in Chapter 15 for Reactivity accidents. Reactivity accidents, such as Rod Drop or Rod Ejection, which are dominated by the fuel Doppler Coefficient, are eliminated in the ABWR by design features which preclude their happening. For the ABWR no analyses were reported in the SSAR; therefore no further consideration was given to this class of events for the present study.

For LOCAs, nuclear dynamic feedback plays an insignificant role, since the plant is presumed to scram at the start of the event. The only other core design dependent input is the decay heat level for the fuel after scram.

The decay heat level for the mixed-oxide is expected to be slightly less than a UO_2 core, because Pu^{239} has a lower decay heat power than U^{235} . In any case, the ABWR LOCA analysis is dominated by the inventory loss during the initial blowdown, and the results for a UO_2 core show no core uncover for any break size. Thus, the conclusion is that the mixed-oxide cores will not degrade the LOCA results.

3.0

FUEL FABRICATION AND FUEL CYCLE

In this section, the fuel cycle associated with the dispositioning process has been described. After an overview, the key process selections are given in Section 3.2. The plutonium stockpile reduction for the various alternatives is indicated in Section 3.3. The fuel fabrication requirements, process flow descriptions and the option to locate the fuel fabrication facility off the reactor site are discussed in Sections 3.4 through 3.7. The safeguards and security for plutonium handling and the fabrication plant are described in Section 3.8 while the safety and environmental aspects are discussed in Section 3.9. Finally, the development issues are outlined in Section 3.10 with the cost and schedule input given in Section 3.11

3.1 OVERVIEW FOR DISPOSITION OF WEAPONS-GRADE PLUTONIUM STOCKPILE

Today and in the near future, the most effective means of safeguarding weapons-grade plutonium to minimize the risk of it being diverted to remake weapons is to turn it into irradiated spent fuel. The key steps for dispositioning the plutonium with an ABWR and its fuel cycle are illustrated in Figure 3.1-1. The first step of transforming the high density metallic plutonium into low density plutonium oxide powder provides significant resistance to desensitization of diversion. Table 3.1-1 characterizes some differences between weapons-grade plutonium, and fresh and irradiated ABWR fuel. The plutonium in the fresh ABWR fuel would be an unattractive target for diversion. After use in an ABWR where some of the plutonium is destroyed and radioactive fission products have been generated, the irradiated ABWR fuel is highly radioactive. In the irradiated form the remaining plutonium would offer a high level of residence to diversion.

Table 3.1-1. Characteristics of Several Forms of Plutonium (Pu)

Characteristic	Weapons-Grade	Fresh ABWR Fuel	Irradiated ABWR Fuel
Physical Form	Metallic	Ceramic Pellet	Ceramic Pellet
Pu Density (in form)	19 gm/cc	0.1-2 gm/cc	<.1-2 gm/cc
Composition (%)			
Pu	>99	3	2
Other	<1	97	97
Fission Products	0	0	0
Radioactivity*(curies)	60	60	200,000

*Calculated for five kilograms of weapons-grade plutonium

The Technology Base for the Fuel Fabrication Plant: The fuel fabrication technology, like the ABWR, has seen many improvements based on world-wide applications in the past 30 years. A new fuel fabrication plant, which would be built in a complex with an ABWR, would utilize the most advanced and proven safeguards techniques, automated and mechanized equipment to run the processes to ensure highest quality and to minimize workers' radiation exposure, and optimized processes to minimize waste generation and handling. There is also the possibility of cooperative efforts of experts from around the world in establishing the detailed requirements and design for the fuel fabrication plant. In particular, cooperative efforts in the safeguards area from the beginning would ensure the highest level of visibility on monitoring the weapons-grade plutonium and keeping it secured. The possibility of implementing the best available technology in fuel fabrication, the resultant long-term safeguarding of the plutonium and the generation of billions of kilowatt-hours of electricity make the ABWR and its mixed-oxide fuel cycle the ideal approach for disposing of the weapons-grade plutonium.

3.2 KEY PROCESS SELECTIONS

Several key decisions were made in selecting the reference fuel cycle process and two of its key stages. These decisions include:

Figure 3.1-1 ABWR FUEL CYCLE FOR Pu DISPOSITION

- 1) the overall process approach,
- 2) the metal to oxide conversion approach,
- 3) and the approach to mixed urania, plutonia, gadolinia pellet fabrication.

The specific approach and the process details are described in Section 3.5. Following in this subsection are discussions of alternatives or considerations made.

3.2.1 OVERALL FABRICATION PROCESS: REFERENCE & ALTERNATIVE

Available Methods for Mixed-Oxide Fuel Fabrication

The uranium-plutonium mixed-oxide fuel needed to support the ABWR concepts described would have plutonium content of less than five weight percent. The current fabrication technology is capable of producing fuel having a plutonium content up to a maximum of about 30 weight percent. A wide selection of proven methods are available for fabricating such mixed-oxide fuel. The three most common and most developed fabrication methods are:

- mechanical blending
- co-precipitation
- sol-gel process

In selecting a preferred method, one of the primary requirements is to minimize worker occupational radiation exposure. The potential worker exposure is strongly affected by the radioactivity of the initial fuel feed material. With strict requirements on allowable worker exposure, the complexity and cost of the fuel fabrication facility would increase significantly as the radioactivity of the initial feed material increases. Table 3.9-1 shows the radioactivity calculated for one gram of weapons-grade plutonium and compares it to the radioactivity calculated for the plutonium that has been used to fabricate fuel pins and bundles for use in fast reactors like Fast Flux Test Facility. As indicated the radioactivity of weapons-grade plutonium is much lower than the radioactivity of the plutonium that has been used for fast reactor fuel.

Mechanical Blending/Co-precipitation Process: In the U.S. the mixed-oxide fuel fabrication experience has been primarily with plutonium typical of that used in fast reactors. Although this plutonium was not of the purity of weapons-grade plutonium, it was free of the highly radioactive fission product actinides that are present in spent light water reactor fuel. Because of the relatively clean plutonium used, mixed-oxide fuels have conventionally been fabricated using mechanical blending techniques in glove box type facilities and using manual operations and maintenance. Such facilities are best exemplified in the United States by the successful fabrication of the mixed-oxide fuel for the Fast Flux Test Facility. Both mechanical blending and co-precipitation processes were used in the fabrication of the Fast Flux Test Facility fuel. Similar pilot facilities using the mechanical blending technique in glove boxes have also been operated successfully throughout the world as part of various nations' nuclear technology development programs. An extensive literature is available on production of mixed-oxide fuel by this method and will not be repeated here. The mechanically mixed route has been chosen for this study and is described in more detail in Section 3.5.

Gel-Supported Precipitation Process (GSP): The other option is to utilize a fully shielded cell facility with capability for fully automated operations, maintenance, and repair. While facility costs are increased significantly, such a fully shielded and automated facility offers the following attractive option. Substantial work is currently being performed particularly in Europe and Japan to assess the feasibility of burning up the higher actinides in core to reduce the volume of high level waste that must be accommodated at waste repositories. For a fully shielded and automated facility where the radioactivity of the fuel is no longer a deciding factor, one now has the option of spiking the weapons plutonium with higher actinide waste as a means of disposal. The high gamma emission of the actinide enhances the proliferation resistance of the fuel. Review of the available literature indicates that within the European community, the preferred method for fabricating mixed-oxide fuel containing minor actinides is the GSP (Ref. 3-1) process. The main reasons for utilizing this particular process are the elimination of powder handling and the amenability of the process to remote operation. The main reasons against the GSP process are the high costs for the facilities, equipment, and operation and the quantities of liquid waste which must be treated

and solidified for disposal.

Selected as Reference Process: The selected reference fuel fabrication approach consists of the automated mechanical blending process. The automated process equipment would be setup in glove-boxes which would have built in shielding to minimize radiation exposure to workers. During normal process operations essentially no manual activities are planned. The glove-boxes would allow manual access for repair and periodic planned maintenance activities. Coupled with the low radioactivity of the weapons-grade plutonium this automated and shielded approach would provide acceptable radiation protection to the workers and would be the most cost effective. Alternatively, the technology to build a fully automated and fully shielded, fuel fabrication facility is available; however, this approach would be significantly more costly and would not be needed unless highly radioactive feed material were to be utilized. The reference process is described in Section 3.5.

3.2.2 Conversion of Metallic Plutonium to Plutonia

The preferred form of plutonium feed material into the fuel fabrication plant is plutonia, PuO_2 . It is assumed in this study that DOE will convert the metallic weapons-grade plutonium with composition essentially as indicated in Table 3.9-1 into plutonia ceramic grade and deliver it to the site of the fuel fabrication facility.

Plutonia made from the weapons-grade plutonium with the composition indicated in Table 3.9-1 is of sufficiently high purity that no further refining to further remove impurities is necessary. Besides the plutonia form, its key requirements will be an appropriate maximum particle size and a minimum powder activity.

There are several alternative methods for converting the plutonium into plutonia. They are

1. controlled burning of the metal to form the oxide and followed by a mechanical milling step,
2. controlled burning, followed by dissolution and precipitation process steps

3. controlled hydriding, dehydriding and oxidation process.

Conversion to plutonia by controlled burning and subsequent milling is expected to be acceptable for use. The activity of plutonia limits the mixed-oxide composition to a maximum of about five percent plutonia and would allow pellet densities in the low 90's range. This composition range is compatible with the reference spiking, spent fuel and plutonium destruction alternatives discussed in this study. It would not be compatible with the spiking Option 1 alternative discussed earlier in Section 2. This pellet density is compatible with the spiking alternative and will limit the maximum burnups of the spent fuel and the plutonium destruction alternatives. This process was selected because it would be the lowest cost approach and the plutonia may meet the fuel performance requirements for weapons-grade plutonium disposition. Further evaluations are needed to confirm these initial judgments.

If plutonia of higher activity or finer particle size are required, then the second or third alternative processes must be used. The second alternative of burning, dissolution and precipitation processes was used to meet the Fast Flux Test Facility fuel specification. These processes were routinely performed at the Plutonium Finishing Facility at Hanford, however, the facilities are no longer available. If this conversion approach was selected for implementation, a new facility would be required and it would be of significant cost.

The hydriding-dehydriding-oxidation would also provide plutonia with higher activity and or finer particle size. This process has been developed at Lawrence Livermore Laboratory. The hydriding step allows better control of the final oxidation process. Its application to date has been limited to laboratory tests scale. However, it promises to be a much simpler method for producing highly active plutonia compared to the dissolution/precipitation method. Again, if this conversion approach was selected, a new facility would be required and verification of process parameters and controls would be required.

3.2.3

Mixed Urania, Plutonia, Gadolinia Fabrication

The fuel bundles required for each of the three disposition options include both mixed-oxide and mixed-oxide/gadolinia fuel rods. Considerable experience, both foreign and domestic, is available for fabrication of the mixed-oxide fuel. The majority of this experience is with the powder blend process which was selected to fabricate the weapons-grade plutonium into fuel for the ABWR. Unfortunately, a similar experience base does not exist for the fabrication of mixed-oxide/gadolinia fuel. Urania-gadolinia fuel rods are commercially fabricated as fuel for light water reactors at a facility operated by GE in Wilmington, North Carolina. Development of this process indicated that gadolinia concentrations higher than seven percent may result in fabrication problems. Considerable material property work has been completed by the Japanese on plutonia-rare earth combinations (Ref.3-6, Ref. 3-7). Limited sample preparation information provided with these studies indicated no apparent problem with the fabrication of high density plutonia-rare earth samples. This material property sample information as well as the mixed oxide and urania-gadolinia fabrication information indicates that no significant problems should exist in the fabrication of mixed oxide-gadolinia fuel. Since the concentration of the fuel for the ABWR is in the 90 percent urania range, the urania feed will tend to smooth and mask some of the effects caused by plutonia and gadolinia additions. These effects should result in a process similar to that utilized for urania-gadolinia fabrication. Even though it appears that mixed-oxide/gadolinia fabrication will not be a problem, one key development area will be the demonstration of this process and the control of the fuel properties. In addition, the fuel certification techniques (analytical chemistry and nondestructive assay) must be developed and certified.

3.3

PLUTONIUM STOCKPILE REDUCTION

For this study three alternative scenarios for disposition of the weapons-grade plutonium have been developed. There are many more possible ways to use the ABWR, however, these three scenarios, which are summarized in Table 3.3-1, provide insight into the flexibility provided by the ABWR and its mixed-oxide fuel cycle. Case 1, the spiking alternative, characterizes the shortest time period scenarios for making the plutonium highly

diversion resistant and requires one ABWR. Case 2, the spent fuel alternative, uses one ABWR over a moderately longer period of time to first make the plutonium highly diversion resistant. Subsequent reuse of the plutonium fuel over the remaining reactor design lifetime to more fully utilize the energy available in the fuel to generate more electricity is possible for both these alternatives. Case 3, the plutonium destruction alternative, characterizes the greatest fraction of destruction of the plutonium possible within a 25-year constraint of a light water reactor fuel cycle period and uses six ABWRs. Figures 3.3-1 to 3.3-3 illustrate the timing for utilization of the weapons-grade plutonium by fabricating fresh fuel bundles, the periods of ABWR operation, and the accumulation of spent fuel bundles for each of the three scenarios.

Table 3.3-1. Scenarios for Disposing of 100 Metric Tons of Weapons-Grade Plutonium

Case	Number of ABWRs	Disposition Time Total Years/ Reactor Years	Electricity Generated kW-hr*	Number of Spent Fuel Bundles
1	1	15/7.4	43 billion**	14,700
2	1	24/18	145 billion**	14,700
3	6	25/18	900 billion	18,800

*Electricity generated during the period of plutonium disposition; additional electricity would be generated over the remaining design lifetime of the ABWR(s).

**Reuse of the 14,700 fuel bundles over the remaining 43 years of the ABWR design lifetime would generate an additional 367 billion kilowatt-hours of electricity for Case 2 and an even higher amount for Case 1.

The Spiking Alternative

The spiking alternative (Case 1 in Table 3.3-1) would disposition of the 100 metric tons of weapons-grade plutonium with one ABWR in about seven and a half years of reactor operation and about 15 years overall. The 15 years overall represents seven years needed to design, build and license the reactor for operation and the seven and a half years of

ANNUALIZED PLUTONIUM USAGE

3.10

SPENT CASE	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	5.2	1.2
	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	10.1	
	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	
PU DESTRUCTION CASE	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	9.3	

FIGURE 3.3 - 1
ANNUAL PLUTONIUM DELIVERY REQUIRED FOR
FUEL FABRICATION (100 MT TOTAL)

**PLANT CAPACITY
345 MT MOX**

REACTOR START UP

FIGURE 3.3.- 2

SPENT CASE	768	768	768	768	768	768	768	768	768	768	768	768	768	768	768	768	1650
	1920	1920	1920	1920	1920	1920	1920	1266									
	0	1104	1104	1104	1104	1104	1104	1104	1104	1104	1104	1104	1104	1104	1104	1104	1104

FIGURE 3.3 - 3

FUEL BUNDLES DISCHARGED FROM REACTOR FOR STORAGE

operation. The design of the fuel fabrication plant which is needed to supply the reactor would be initiated at the same time as the reactor design. However, its design, construction and licensing would be done in four years and fabrication of the fuel rods and bundles would begin three years before the initiation of reactor operation. Over the three year lead time and the approximately seven years of reactor operation, the fuel fabrication plant would manufacture about 14700 fuel bundles from the 100 metric tons of weapons-grade plutonium. Selected parameters of the GE-9 fuel bundle design are listed in Figure 2.5-1. Each of the bundles would be used for about 70 full power days in the reactor. Figure 2.5-17 shows the calculated radiation levels around a fuel bundle two years after removal from the reactor and shows that it would exceed 100 rem/hr. After removal from the reactor each fuel bundle would reside in the reactor spent fuel storage pool for two to three months after which it would be moved into a ex-reactor storage pool. The remote handling operations would minimize radiation exposure to workers from the highly radioactive bundles. Eventually the spiked fuel bundles would be moved to the site for permanent spent fuel storage in the high level waste repositioning.

During the seven and a half years at power, the ABWR would generate about 43 billion kilowatt-hours of electricity. After the short spiking mission, the ABWR would have about 52 years of design operating lifetime remaining. The U.S. Government could continue to operate the ABWR or sell it to a commercial operator who would continue to operate the plant.

For spiking, a promising scenerio, which has a fuel fabrication throughput requirement of 71 metric tons/year of mixed oxide fuel, would be to use one ABWR with a fuel bundle design utilizing both mixed-oxide and enriched uranium oxide fuel pins. This bundle design is described in Section 2.5.2, and would utilize about 20 mixed-oxide fuel pins along with 40 enriched uranium oxide fuel pins. The core concept needs further development as the current concept has been determined to have a positive void reactivity coefficient at low void fractions. If this concept can be acceptably developed it should be able to disposition the 100 metric tons of plutonium in five to seven years of reactor operation. In the following report sections on fuel fabrication, this scenario is presented as Option 1 for the

spiking alternative.

The Spent Fuel Alternative

The spent fuel alternative (Case 2 in Table 3.3-1) would disposition the 100 metric tons of weapons-grade plutonium with one ABWR in about 17 years of reactor operation and about 25 years overall. The fuel design for this alternative is the same as that for the spiking alternative. Compared to the spiking alternative where each fuel bundle would operate at full power in the reactor for about 70 days, for this alternative each fuel bundle would operate at full power in the reactor for about 275 days. Therefore, these spent fuel bundles would be more radioactive than those from the spiking alternative. This initial phase meets the study requirement to disposition the weapons-grade plutonium in less than 25 years. The spent fuel bundles would be stored initially in the reactor spent fuel storage pool and subsequently moved to ex-reactor storage pool. Following this 25 year period, these stored fuel bundles could be reused to more fully utilize the energy available in them. In this scenario, these fuel bundles could be reloaded into the ABWR and operate at full power for an additional 774 days. The total electricity generated over the initial 17 years disposition period, as indicated in Table 3.3-1, is about 145 billion kilowatt-hours. The 14,700 fuel bundles are sufficient to fuel the ABWR over its remaining design lifetime (another 43 years) and would generate an additional 367 billion kilowatt-hours of electricity. After full utilization, the spent fuel bundles would be stored in the reactor or ex-reactor spent fuel storage pools until they are eventually shipped to the high level waste repository.

The Plutonium Destruction Alternative

The plutonium destruction alternative (Case 3 in Table 3.3-1) would destroy the largest amount of the weapons-grade plutonium which would require six ABWRs in order to disposition the 100 metric tons of plutonium in the required time of 25 years. In this scenario the first seven years are again used for design, construction and licensing of a complex of six identical ABWRs. In the following 17 years of reactor operation each of the 18,800 fuel bundles will be used on average about 1492 days at full power in the reactor.

The fuel bundle design is only slightly different from the fuel design for the spiking and spent fuel alternatives and is given in Section 2.7.1.1. Each fuel bundle would be used to the maximum extent practical in an ABWR which results in the greater amount of plutonium destruction. Also, because of the resultant greater accumulation of fission products, each discharged bundle would be much more radioactive than the bundles from either the spiking or spent fuel alternatives.

The total electricity generated during the 17 years of operation, as indicated in Table 3.3-1, would be about 900,000 kilowatt-hours. After the plutonium destruction mission, each of the six ABWRs would have about 43 years of design operating lifetime remaining. The U.S. Government could continue to operate the six ABWRs or sell them to a commercial operator who would continue to operate the plants with uranium fuel.

For the plutonium destruction, a scenario, which has a fuel fabrication throughput requirement of 58 metric tons/year of mixed-oxide fuel, would be to use two ABWRs with all mixed-oxide fuel bundles designed to achieve the maximum practical burnup by once-through operation in the reactor. This scenario would take about 60 years of operation by each reactor to disposition of the 100 metric tons of weapons-grade plutonium. This scenario does not meet the 25 years disposition time requirement for this study; however, it is mentioned here because it gives the perspective on the best utilization of the 100 metric tons of weapons-grade plutonium. In the following report sections on fuel fabrication this scenario is presented as Option 1 for the plutonium destruction alternative.

3.4 ON-SITE FUEL FABRICATION REQUIREMENTS

The fabrication requirements for the fuel pins and bundles for each of the three plutonium disposition scenarios are summarized in Tables 3.4-1. (Note: detailed fuel pin designs are given in Section 2.0 on Core Design and Preferred BWR Concept for Plutonium Utilization.) Table 3.4-1 identifies and summarizes the amounts of the key strategic materials needed in each type of fuel bundle. The strategic materials are uranium, plutonium and gadolinium. The total amounts of each strategic material in each bundle

type is given. These materials are distributed among the 60 pins in each bundle. In some cases, there are as many as six different compositions among the 60 pins. For all cases, except for Option 1 for the spiking case, the 60 pins are mixed-oxide pins. For Option 1 of the spiking case, 20 of the pins in each fuel bundle are mixed-oxide and the remaining 40 pins in the bundle are enriched uranium dioxide pins.

Table 3.4-2 summarizes the annual throughput requirements for the fuel fabrication plant for the three plutonium disposition scenarios. In each scenario the fuel fabrication campaign is planned to begin three years before reactor operation begins and would end with the fabrication of the fuel bundles for the last planned reactor reload. The duration of the campaign needed to completely transform the 100 metric tons of weapons-grade plutonium into ABWR fuel bundles is also indicated in Table 3.4-2. Note, that the shortest fabrication campaign is about 10 years for the reference spiking scenario. The spiking Option 1 could be shorter if an acceptable core design is developed. Note, that the campaign for the reference plutonium destruction case is a relatively short 13 years. The annual throughput rate is driven by the need to fabricate the initial cores for six reactors in three years. At this capability, only ten more years are needed to fabricate the fuel bundles for the planned reactor operating time of 18 years.

3.5 FUEL FABRICATION PROCESS AND FACILITY DESCRIPTION

In this section, the fuel fabrication process and facility descriptions are described for the reference spent fuel alternative. The process would be the same for the other alternatives, however, the facility requirements for the two other alternatives would differ based on their different throughput requirements. These difference are covered in Section 3.6.

3.5.1 Plutonium Receiving Interface

The study assumes that DOE will supply the plutonium feed to the facility in the form of plutonium oxide meeting fuel fabrication specifications. Conversion of the plutonium metal from weapons disassembly to plutonium oxide is expected to be accomplished in an off-site

**Table 3.4-1. Summary of Design Parameters for the Fuel Bundles
for the Various ABWR Scenerios**

Design Parameters	Reference	Option 1
<u>Spiking Alternative</u>		
Fuel Bundle (kg):		
Uranium	169.4	163
Plutonium	6.8	10.6
Gadolinium	1.8	4.4
Fuel Pins (#):		
(U-Pu)O ₂	40	0
(U-Pu)O ₂ /Gd ₂ O ₃	20	20
UO ₂	0	40
<u>Spent Fuel Alternative</u>		
Fuel Bundle (kg):		
Uranium	169.4	
Plutonium	6.8	
Gadolinium	1.8	
Fuel Pins (#):		
(U-Pu)O ₂	40	
(U-Pu)O ₂ /Gd ₂ O ₃	20	
<u>Plutonium Destruction Alternative</u>		
Fuel Bundle (kg)		
Uranium	172.7	172.7
Plutonium	5.3	5.3
Gadolinium	1	1
Fuel Pins (#):		
(U-Pu)O ₂	28	28
(U-Pu)O ₂ /Gd ₂ O ₃	32	32

Table 3.4-2. Summary of Fuel Fabrication and Assembly Throughput Requirements for the Various ABWR Scenarios

Requirements	Reference	Option 1
<u>Spiking Alternative</u>		
Fabrication Requirements, (kg/yr)		
Uranium	255,000	57,000
Plutonium	10,100	13,600
Gadolinium	2,680	6,120
Assembly Requirements, (#/yr)		
MOX Fuel Pins	89,460	25,830
Bundles	1,491	1,392
Campaign Duration, (yrs)	10	7
<u>Spent Fuel Alternative</u>		
Fabrication Requirements, (kg/yr)		
Uranium	130,000	
Plutonium	5,200	
Gadolinium	1,360	
Assembly Requirements, (#/yr)		
MOX Fuel Pins	45,580	
Bundles	758	
Campaign Duration, (yrs)	19.5	
<u>Plutonium Destruction Alternative</u>		
Fabrication Requirements, (kg/yr)		
Uranium	303,000	56,000
Plutonium	9,300	1,700
Gadolinium	1,740	320
Assembly Requirements, (#/yr)		
MOX Fuel Pins	104,640	19,440
Bundles	1,744	324
Campaign Duration, (yrs)	13	60

access doors of sufficient design and construction to assure that only authorized entry is permitted. Closed circuit television surveillance cameras; personnel, vehicle and package examination and explosive detection devices; appropriate container identification (bar code readers, etc.) and other monitoring devices are provided to ensure that facility physical protection and material accountability goals are met.

After unloading and log-in of shipping containers and contents based on data provided on the shipping manifest, the containers are moved by controlled and monitored carts and cranes to a container opening area where the plutonium oxide containers are unloaded. The study assumes that each PuO_2 containers will contain three kilograms of material and is cylindrical in shape with dimensions of approximately four inches in diameter by six inches long.

Individual containers are weighed, assayed, and examined as necessary to identify containers and verify contents. Container identify, content, location and transfer data is recorded by the safeguard material accountability computer. Transfer to the plutonium storage vault is monitored by a combination of closed circuit television surveillance cameras and motion detection and monitoring devices and is recorded by the safeguard computer. The individual containers are loaded into transfer baskets and transferred by cart to the PuO_2 storage vault.

3.5.2 Plutonium Oxide Storage

Storage Vault Design: The plutonium oxide storage vault is designed to provide secure storage of a six-months supply of feed to the fuel fabrication facility. For the spent fuel base case, this is approximately 3000 kilograms of plutonium oxide.

The storage vault is designed as a chamber covered by a concrete slab at grade level with storage holes provided in the slab in a critically safe array. Baskets extend below the slab at each storage location to hold the containers. Each storage hole is provided with a shielding cover which can be removed or replaced by a telescoping crane over the vault.

The crane is also utilized to transfer baskets containing plutonium oxide containers to the storage locations. Figure 3.5-1 shows the vault design concept.

Transfer operations within the vault are designed to be totally remote. Adequate ventilation is provided by airflow around the storage baskets to remove the heat generated by the stored plutonium oxide. Ventilation air is filtered by high efficiency particulate air filters prior to release to the atmosphere.

Vault Safeguards and Security: The storage vault safeguards system utilizes a dual containment and surveillance system consisting of surveillance cameras and recorders and an infrared beam monitoring system to detect movement of containers or baskets in and out of the storage locations. The surveillance camera system will provide video tape records of movements within the vault. The infrared beam monitoring system consists of a network of infrared light sources and photosensors located along opposite sides of the vault which are aligned to provide intersecting beams above each of the storage locations. Accessing a storage hole activates a unique pair of photosensors, identifying the position to the safeguards computer, which determines whether an insertion or withdrawal has been authorized at that position. The system is equipped with tamper indicating switches as well as redundant components to ensure continuous operation during upset (power failure, etc.) conditions.

3.5.3 Fuel Fabrication Process Flow Description

The process selected for fabricating the weapons-grade plutonium into oxide fuel pellets has been utilized to fabricate both urania (Ref. 3-4) and mixed-oxide fuel (Ref. 3-5, 3-6, 3-7) for LWRs, mixed-oxide fuel (Ref. 3-8) for ALMRs and uranium nitride fuel (Ref. 3-9) for space reactor applications. The basic mill-slug-granulate process has been well developed for over 20 years. In this process the feed powders are blended together in the desired concentration, milled to improve homogeneity and prepressed and granulated to form free flowing powder for pellet pressing. The resulting pellets are sintered in a reducing atmosphere, ground to size, inspected and loaded into fuel rods. The fuel rods are sealed

Figure 3.5-1. PuO₂ Storage Concept

by welding, inspected and assembled into fuel bundles. The completed fuel bundles are stored at the fabrication facility until they are needed to provide initial or reload assemblies for a reactor. Several advancements in the fuel fabrication processing have been developed during the past ten years and some have been put into practice during the fabrication of urania fuels. These demonstrated process improvements were incorporated in the basic fuel fabrication process which is detailed as a flow diagram in Figure 3.5-2.

The fuel fabrication process will be identical for all three disposition options considered and the number of equipment items is variable to provide the required capacity for each option. The fuel fabrication facility will be separated into eight basic processing areas to prevent processing interference during normal and upset conditions. These areas include powder receiving, plutonia storage, mixed-oxide fabrication, mixed-oxide/gadolinia fabrication, wet scrap recycle/waste treatment, rod storage, bundle assembly/inspection and bundle storage. The facility design processing rates for the spent fuel, spiking and plutonium destructions options are summarized in Table 3.5-1. These hourly processing rates were determined based on annual fuel assembly requirements utilizing 24-hour operation for 300 days per year. The process operation was sized with 15 percent excess capacity and the operators were staffed on a five shift per week basis to provide adequate time for training and certification. Based on this type of staffing, the normalized throughput rates for fuel bundles in the fabrication facility are summarized by year in Figure 3.5-2.

The relatively low radioactivity levels associated with the weapons-grade plutonium will allow the fuel fabrication activities to be performed in a shielded glove box arrangement. The problem of reduced worker exposure limits will be handled by mechanizing all unit processing operations which will reduce the time the operator is in close proximity of the nuclear material. Also, localized shielding will be installed and lead glass and sheet added to the exterior of the glove boxes. All maintenance activities will be accomplished by hands-on techniques after nuclear material has been removed to shielded storage areas which will reduce exposure.

Table 3.5-1. Fabrication Processing Capacities

Fabrication Area	Spent Fuel	Spiking	Plutonium Destruction
Plutonium Storage (kg)	2600	5100	4700
MOX Fabrication (kg/hr)	16	31	25
MOX/GAD Fabrication (kg/hr)	8	15	29
WSR/Waste Treatment (kg/hr)	4	7	8
Rod Storage (#)	12,000	25,000	25,000
Bundle/Assembly/ Inspection (#/day)	10	10	10
Bundle Storage (#)	875	1750	5,000

The feed materials for the fuel fabrication process which utilizes powder blending are the oxide forms of the uranium, plutonium and gadolinium. The uranium (either natural or depleted) will be supplied as an active powder capable of sintering to high density. This plutonium must meet the requirements of ASTM C757, Nuclear-Grade Plutonium Dioxide Powder, Sinterable.

3.5.4 Pellet and Rod Fabrication

In the pellet fabrication process the feed oxides will be transferred from storage to geometrically safe holding bins in the blending/splitting area. Weight feeders will be utilized to load the appropriate amount of materials into a vibromill where the powders are blended and milled together. The resulting blend will be split into as many as eight increments and the subsequent eight increments cross blended to form a large uniform feed lot of powder. The blend splits will be stored in a geometrically safe arrangement and removed for continued processing one at a time. During reblending or cross blending, each blend split will be vibromilled to smooth blend differences and produce an agglomerated

powder for feed to pellet pressing. Small hydraulic pellets presses developed for glove box use will be utilized to press pellets. The throughput requirements are high enough that multi-cavity dies will be required on each of these presses. The green pellets will be loaded into molybdenum sintering boats with a capacity up to four kilograms each. Specially designed sinter furnaces with removable refractory sections will be utilized to sinter the mixed-oxide pellets (Ref. 3-10). These furnaces are designed with a capacity of eight kilograms/hour for mixed-oxide and four kilograms/hour for mixed-oxide fuel/gadolinia pellets. The oxygen atmosphere of the sintering furnace will be controlled to obtain fuel pellets of the desired stoichiometry. All of the sintered pellets will be passed through a center-less grinding station after inspections for sintering characteristics to control the pressing operations. The grinder will have an automatic control feed back loop which will automatically adjust the grinding to maintain the pellet within limits. The column of pellets which exit the grinding operation pass directly to the automated pellet inspection station. At the station, the pellets are inspected and sorted for geometric properties (diameter, length, density, surface finish, cracks and pits). After sampling for chemical characteristics, the pellets are placed into storage to await certification approval for fuel loading. The certified fuel pellets are aligned into columns where final column geometry data is obtained prior to loading into fuel cladding (tubes with one end plug welded in place).

Rod loading will be automatic but attended by an operator. The equipment will be properly shielded to protect the operator. Vendor supplied tubes with a serialized end plug welded on one end will be introduced into either the mixed-oxide fuel or mixed-oxide fuel/gadolinia area from the rod storage area. Certified pellets will be loaded into the tube and the fuel column weight will be checked against the design specification. Following tube loading, the rods are decontaminated and transferred to the final weld operation. Final welding will be done on an automated, operator attended tungsten inert gas welder. The equipment will be shielded to protect the operator. When the welding is completed the rods will be leak checked, inspected and sent to rod storage until required for bundle assembly. All in-process transport will be handled in shielded carts that will hold a critically safe quantity of rods.

Completed and in process fuel rods will be stored in rod storage racks. The rack design will be made to hold approximately 25000 mixed-oxide fuel and mixed-oxide fuel/gadolinia rods. The storage area will be shielded and adequately ventilated to remove decay heat. A critically safe quantity of rods will be stored in the rack bins. The trays will be moved by a stacker retriever on track rails. One stacker retriever is located on the ceramic/rod fabrication side of the facility and another is located on the bundle assembly area.

Certified rods would be removed from rod storage on the assembly side of the building. The rods are automatically transferred to the X-ray station where final weld integrity is checked. Following X-ray, in-specification rods are returned to the rod storage area. Out-of specification rods are returned to the quarantined section of the rod storage area. All rods will be actively and magnetically scanned for fissile content, pellet integrity, stacking and gadolinia content. Scanners and X-ray equipment will be operator attended. Equipment will be automated and shielded.

The bundle assembly will be performed adjacent to the X-ray and scanner operation. Selected channels or rods required for the assembly of one bundle are accumulated at a rod accumulation station. The station is limited to a safe batch of fuel rods or one bundle of fuel rods. The station will be operator attended and shielded to protect the operator. The quantity of rods required by type will be mechanically removed from the tray, inspected for integrity and transferred to an automated bundle assembly station. Rods will be transferred in random order. A serial number reading machine will read the serial number located on the shank of one end plug. The bundle assembly station is fixtured manually to accept bundle spacers. Following fixturing and serial number reading, the rods are automatically positioned into the proper bundle grid location. A road map of the rod locations are automatically transferred to a computerized Quality Assurance System for verification. The information is stored in the computer for further quality assurance traceability information and bundle certification.

A strong back at the assembly station tilts the completed bundle to a vertical position from which an automated crane will move the bundle to a shielded in process storage area

pending final inspection. This in process storage area is designed to hold a limited amount of critically safe fuel bundles.

Bundles are removed from the in-process storage area to a automatic bundle inspection station. The bundles are inspected for compliance to specification. The information from this inspection will be permanently stored in a computer along with other pertinent quality assurance information needed for bundle certification. The inspected bundle is automatically wrapped in a plastic sleeve for protection and transferred by a remote control crane to the bundle storage area.

The final bundle storage area design is incomplete at this time. As the mixed-oxide fuel Fabrication Facility will lead the reactor startup by three years, storage space is required for one initial core loads or 872 bundles. Final inspected bundles will be stored in critically safe storage containers and nested in shielded and ventilated storage pits. All operations associated with bundle storage will be automated and remote controlled. It is expected that the storage containers will serve as shipping container to the reactor.

3.5.5 Scrap Recycle

The wet scrap recycle area was designed to process up to 15 percent of the throughput of combined mixed-oxide and mixed-oxide fuel/gadolinia ceramic areas. It is noted that very low scrap rate is expected. The scrap feed will be primarily rejected quantities from rods, unsintered/sintered pellets and powders. The scrap will be crushed to provide feed material for the batch dissolvers which are phased in cycle for around the clock operation. The solution from the dissolver is filtered to remove any undissolved scrap and collected in geometrically safe slab tanks for storage. Feed batches are prepared for a Purex-type (Ref. 3-11) solvent extraction cycle. In this cycle, the solvent extraction columns remove the metallic corrosion products, americium and trace fission products. The nitrates of plutonium and uranium are back extracted, concentrated and stored for conversion to powder. In the conversion area the nitrates of plutonium and uranium are mixed with ammonium hydroxide to produce plutonium hydroxide and ammonium diuranate. The

colloidal slurry is feed to a fluidized bed type unit where it is converted to a fine plutonia-urania powder. The hot gases from the fluidized bed are feed to a scrubber system to remove the water vapor, nitrogen and excess fluidizing gases. The powder from the conversion area is reduced in a continuous furnace and stabilized with a carbon dioxide atmosphere to provide powder which can be added back into the ceramic blending operations. The wet scrap recycle will be designed with a miscellaneous waste treatment area to handle wastes which contain significant amounts of plutonium. The uranium and plutonium compounds will be removed by acid leaching and the undesirable compounds (chlorides and fluorides) eliminated prior to transfer of the solutions to solvent extraction area. This area will be equipped with kynar lined glove box to process chloride bearing scrap which primarily originates in the analytical chemistry area.

3.5.6 Waste Treatment

Since the fuel fabrication facility was designed so that no process liquids will be discharged to the environment, a liquid waste treatment area is required to collect the liquids and provide for chemical recycling. The main source of these liquids are several unit operating in the solvent extraction area. The acidic and basic waste solutions are collected separately and neutralized. The neutralized solutions are concentrated to about 40 percent dissolved salt in concentrated nitric acid. The vapors from the concentrator are feed to a fractionated tower where nitric acid is reclaimed for recycle within the plant. The solution from the concentrator is sent to a waste solidification area where the remaining acid is driven off and the metal nitrates are converted to oxides. These oxides are collected in waste containers suitable for shipment to a government waste repository.

The fuel fabrication area produces three basic types of gases which must be treated before discharge. The first of these gas streams are six percent H₂-N₂ which result from the various furnaces in the facility. All of these gases will be filtered hot to remove any potential uranium or plutonium particulate and routed to the building exhaust system for dilution and subsequent treatment. The second group of gases are primarily nitrogen with water vapor diluted with air supplied to fluidized the conversion beds in the wet scrap

recycle area. These gases are filtered hot to remove any particulate and routed to a gas scrubber. The scrubber is designed to chemically react with any of the oxides of nitrogen present in the gas. The final group of gas streams are generated by the scrap dissolver and waste treatment areas and contain appreciable amount of nitrogen oxides. These gases are sent to an absorber designed to remove 95 plus percent of the nitrogen oxides. The resulting liquid recovered from the absorber tower contains significant amount of nitric acid and will be recycled within the facility. The gases from the absorber tower are scrubber to remove any remaining oxides of nitrogen prior to discharge to the building exhaust system.

The solid waste treatment area will receive and process all radioactively contaminated waste materials generated during normal operation of the fuel fabrication facility. These wastes are expected to include used gloves, plastic bags, filters, sample containers and failed equipment items. The solid waste will be separated into several groups depending on the amount of plutonium it contains and the packaging required. Wastes which contains significant amounts of plutonium will be sent to the miscellaneous waste treatment area in wet scrap recycle area. All wastes will be compacted to reduce the volume and placed in waste containers for shipment to a government waste repository. Special cleaning and leaching procedures may be required to clean failed equipment items prior to packaging in waste containers for disposal.

3.5.7 Fuel Fabrication Facility

The fuel fabrication facility houses the mixed-oxide fuel fabrication process equipment and support systems as well as the plutonium oxide receiving and storage areas. The building also provides space for tritium target manufacturing for future conversion of the facility to tritium production. Figures 3.5-3 and 3.5-4 show the layout of the reference mixed-oxide fuel Fuel Fabrication and Storage Facility.

The portions of the building housing plutonium materials will be designed to DOE Order 64301A. Section 4.1 "Plutonium Processing and Handling Facilities". The plutonium

150M

3.30
110M

Figure 3.5-3. Mixed-Oxide Fuel Fabrication and Storage Building
20,000-105,000 Rods/Year Throughput

Figure 3.5-4. Mixed-Oxide Fuel Fabrication and Storage Building
2nd Floor

oxide storage facilities will be designed to DOE Order 64301A, Section 1305, "Plutonium Storage Facilities". The portions of the building structure which will normally or potentially contain plutonium or significant quantities of other radionuclides will be reinforced concrete structures designed to safety class 1 (Category 1) standards and maintain their confinement safety function in the event of a design basis accident.

Primary confinement of plutonium-containing materials will be provided by glove boxes, piping and process equipment. These components will be designed to nuclear quality assurance standards and maintain their confinement safety function through all normal, anticipated upset and design basis accidents the primary barrier is required to withstand. Secondary confinement systems including internal building structural barriers and associated secondary confinement ventilation systems will be designed to maintain their confinement function through normal, upset and applicable design basis accident events. Tertiary confinement provided by facility outer structures shall withstand all normal, upset and applicable design basis accident events (i.e., design basis earthquake and design basis tornado). Building ventilation systems will be designed to maintain flow from areas of low contamination potential to areas of higher potential. Exhaust ventilation systems will be provided with HEPA to minimize the release of plutonium and other hazardous material through the exhaust path.

The plutonium oxide storage facilities will be designed to ensure nuclear criticality safety during and following applicable design basis accidents including the design basis earthquake.

In addition to the process support systems such as chemical makeup, gas supply, waste processing and process off-gas systems; other process and building utility and service systems provided will include:

- Emergency power system
- Fire protection system
- Contaminated water collection and treatment system
- Chilled water system

- Cooling water system
- Compressed air system
- Hot water system
- Radioactive (hot) and non-radioactive (cold) maintenance facilities

The portion of the building housing non-nuclear materials and components and areas providing cold maintenance, material control, offices, medical and cafeteria facilities will be designed to uniform building code standards. Gas and chemical storage and electric equipment and HVAC chiller pads have been provided outside the building.

3.6 FUEL FABRICATION PROCESS ADAPTATIONS FOR DIFFERENT ALTERNATIVES

The primary difference between the three alternatives for disposition of weapons plutonium is the throughput required to produce the fuel for the reactors. In the spent fuel option, the fuel fabrication area must process about 140 metric tons of fuel each year. The fuel fabrication requirements for the spiking and plutonium destruction options require annual throughputs of about 270 and 315 metric tons respectively. This factor of slightly more than double the throughput rates associated with the spiking and plutonium destructions options will require larger feed powder storage, rod storage and bundle storage areas. The capacity of the ceramic fabrication area is increased by replicating equipment items. Not all areas were being fully utilized. The mixed-oxide fuel fabrication area sized for the spent fuel alternative must be increased by about 20 percent for the spiking and plutonium destruction alternatives. The mixed-oxide/gadolinia area must be increased by 40 percent for the spiking option and by 112 percent for the plutonium destruction option. The on-site storage area for fresh fuel assemblies for the initial core must be increased about 100 percent for the spiking alternative and about 150 percent for the plutonium destruction alternative.

The consideration for the construction of a MOX Fuel Fabrication Facility at the location of a reactor site has sound logistical basis. If for some economic reason a fuel plant at the reactor site might prove to be unacceptable, other alternatives for the fabrication of mixed-oxide fuel pins need to be considered. No commercial size mixed-oxide fabrications/facilities presently exist in the U.S. Some work could be done at Los Alamos National Laboratory but this would require facility scale-up and in the final analysis the facility could not probably support the volume requirements. Los Alamos National Laboratory could be used as a support facility for ceramic development work or prototype bundle fabrication. The Secured Automated Fabrication system for fabrication of mixed-oxide fuel pins developed in Hanford in the early 1980's. This facility never operated; however, it has adequate space to provide the pellet fabrication. The basic process was also the powder blending approach. An assessment is required to determine if the fuel rods and bundles assembly processes as well as the completed bundles storage can be accommodated in this facility.

An overview of the major mixed-oxide fuel fabrication facilities in Europe and Japan are shown on Table 3.7-1. No one facility appears to have the capacity to meet the pin fabrication requirements of this study. It is doubtful that even a combination of foreign operations could satisfy the scheduling logistics of the spiking case (100 metric tons plutonium in 7 1/2 years). If one such facility did exist with the required capacity requirements, it would be necessary to have it dedicated to the Plutonium Destruction Program. This approach would hardly be conceivable considering the national nuclear programs of the countries concerned. Shipping and storage logistics of plutonium mixed-oxide fuel pins and related components would require further study. Off-shore shipping of plutonium fuel could present a challenge from the public. The availability of satellite tracking systems for the movement of hazardous materials would certainly enhance the round trip shipments of fuel and pins if public opinion could be won over. The recent plan for the Japanese fabricators to have mixed-oxide fuel pins made in the Belgium MOX facility for their light water reactor programs might provide some insights on how to best

handle such transport arrangements. The planning for a large scale mixed-oxide fuel manufacturing facility in Japan to complement the Tokai Mura MOX facilities is still underway and details should be known later in 1993.

Table 3.7-1. Mixed-Oxide Fabrication Facilities

MIXED-OXIDE FABRICATION FACILITIES			
County/Facility	Status	Capacity	Comments
Belgium, Belgonucleaire	Operable	35	LWR/MOX
France, CF Ca	Operable	35	FBR/LWR/MOX
France, Melox	Under Construction	120	FBR/LWR/MOX, 1995, Planned Startup
France, FBRC	Operable	500	LWR/MOX, PWR
Germany, Siemens	Under Construction	120	FBR/LWR/MOX, 1993, Planned Startup
Japan, PNC	Operable	35	LWR/ATR/FBR/MOX
UK, Sellafield BNFL	Planned	58	LWR/MOX Commissioned 1993
UK, Sellafield BNFL	Stand By/Down	6	FBR/MOX
Source: Nuclear Engineering, Intl, Fuel Review 1992			

Of the facilities shown on Table 3.7-1, and depending on the timing, the Siemens plant in Germany or the French facility in Melox might best meet the requirements of the Plutonium Disposition Program. The planned capacity for the German plant is 120 tons of HM per year, starting in 1993 assuming licensing requirements are met. If there was space in the German/French manufacturing schedules, fuel pins could be made at one of these facilities. This plan assumes that all of the timetables to meet the initial reactor startup needs are properly programmed. If a new mixed-oxide fuel fabrication facility is planned in the U.S., it is important to consider the possible company location of plutonium to PuO_2 facility with the conversion to mixed-oxide fuel and loading into fuel pins, to avoid duplication of glove boxes, safety and licensing, and assaying at two different sites. These issues are

considered in more detail in Section 11.

3.8 SAFEGUARDS AND SECURITY

The safeguards and security measures to be implemented for the fuel fabrication plant will be developed to meet applicable U.S. and international regulations. Safeguards and security requirements will form an integral part of the design requirements for the fuel fabrication plant and would be maintained throughout the design, construction, operation and decommissioning phases. Multiple layers of safeguards and security to assure redundancy will be provided for the plutonia that is delivered to the fuel fabrication plant.

The layers of physical security would include:

- 1) the fuel fabrication plant will be within a double-fenced area that is within the fenced site area;
- 2) the construction requirements for the portion of the fuel fabrication facility that houses the plutonia storage area, the pellet and rod process, the fuel pin and bundle assembly areas, and the completed bundle storage area shall be similar to that for reactor containment building;
- 3) within the fabrication facility the initial plutonia storage area shall be within a vault which will have limited computer-controlled access;
- 4) in the batch processing area the plutonia will be controlled in glove-boxes;
- 5) after the fuel pellet loading step, the plutonia will be hermetically sealed into zircoloy tubes, which are subsequently clustered into bundles;
- 6) two fuel bundles would be loaded into a steel shipping container and moved by computer controlled crane into a shielded, vault-like storage area;
- 7) and, the site security force will monitor the area outside the fuel fabrication plant fences and will control access into and out of the double-fenced area. Access into and out of the actual fabrication process areas will also be limited.

The layers of safeguards would be provided by redundant capabilities to verify and monitor

the amounts and locations of plutonium on a near real-time basis. The safeguards would include:

- 1) an integrated computerized system with redundant capabilities would log-in each container of plutonium received by scanning the unique identification codes on each shipping container; collect and compare the verification data from each of the automated verification stations; monitor and record the movement of the containers of special nuclear materials into and out of each material balance area; monitor and record the weighing of each container into each material balance area;
- 2) initial verification of the contents of the shipping containers received will be done by passive gamma scan (Ref. 3-12) to establish its radioactive signature which will be compared against previously measured database of acceptable signatures, and diverse verification of container contents will be by the Californium-252 scanner technique (Ref. 3-12) which was developed for non-destructive assay of fast flux test facility fuel pins, and random samples of container contents;
- 3) assay and evaluation of decontaminated waste materials to determine the amounts of trace plutonium being lost with the waste materials;
- 4) material balance areas to include the initial plutonia storage area, the batch powder processing area, the pellet production area, the fuel pin assembly area, the fuel bundle assembly and storage area; and the scrap recycle area;
- 5) verification of fuel contents by chemical analysis of random samples taken from various stages in the fuel fabrication process;
- 6) verification of the contents of assembled fuel pins by the Californium-252 scanner technique;
- 7) quarterly material balance audits of the entire fuel fabrication facility;
- 8) and, space and resources for on-site independent inspectors to audit material balance and verification data for accuracy and timeliness, and to audit the methods of data collection and analysis.

The provisions of the DOE Orders for SNM accountability will be fully implemented, including all provisions related to the frequencies for physical inventory, checking of

standards, material characterization and software validation of computer codes used in material accounting.

3.9 SAFETY AND ENVIRONMENTAL ASPECTS

Over the past 30 years the requirements for the safety of the workers in a mixed-oxide fuel fabrication facility, and the possible exposure to the general public from radioactive releases from such a facility have become ever more stringent consistent with the importance of safety. In 1992 the DOE revised and issued its latest Radiological Control (RadCon) Manual. The fuel fabrication plant would be a government owned facility which would be operated under DOE guidance and is expected to meet the limits established in the RadCon manual.

The RadCon manual establishes administrative control levels for all DOE activities including exposure of occupational workers and radioactive releases from sites. For occupational worker exposure limits, approval by the Program Secretarial Official or designee is presently required prior to allowing a person to exceed two rem/yr effective dose equivalent. Further, DOE has established that a challenging administrative control limit be set annually following an appropriate review. The DOE recommends a limit of 0.5 rem/year effective dose equivalent and that a limit exceeding 1.5 rem/year effective dose equivalent is not considered challenging enough. In response to 40 CFR 61 subpart H, DOE has established that routine airborne releases from facilities be limited to site release rates of 0.01 rem/yr to members of the public. Building a new facility on a DOE site with other older facilities must take into account the release rates to the public from the existing facilities. As such, the new facility becomes part of the existing DOE Facility which, taken as a whole, is limited to the 0.01 rem/yr dose limit from airborne sources. These requirements are more restrictive than previously set; however, they have been set based on the learning that has occurred over the past 30 years and the advancements in providing protection from radioactive elements. These limits result in the narrowing of the design options for the fuel fabrication plant. In particular, these limits create an incentive for building an automated fuel fabrication plant to reduce worker handling of radioactive

materials, and utilizing shielding to reduce local dose rates. The fuel fabrication facility for the disposition of weapons-grade plutonium is planned to meet these requirements.

At this stage the fuel fabrication facility being described in Section 3.5 is only a concept; however, several factors inherent in the weapons-grade plutonium work in favor of being able to meet the RadCon limits in the final design. First, as indicated in Table 3.9-1 the radioactivity of the weapons-grade plutonium is significantly lower than the plutonium previously used. This is due to the very low Plutonium-240 and Plutonium-241 content in the weapons-grade plutonium. Plutonium-240 is the source of neutron radiation from spontaneous fissioning and the Plutonium-241 is the parent of the Americium-241 which is a major source of gamma radiation. As summarized in Table 3.9-1, the weapons-grade plutonium has about one-fourth the amount of plutonium-240 and about one-twentieth the amount of Plutonium-241 plus Americium-241 compared to typical reactor grade plutonium. Second, the fuel fabrication facility will be designed to make extensive use of automated processes to eliminate manual operations. The normal fuel fabrication process is planned to be fully automated with shielded glove-boxes. Preliminary analyses of the radiation dose rate on the exterior surface of a glove-box with typical shielding (one-half inch thick lead in walls and one inch thick lead-glass) and containing about eight kilograms of mixed-oxide fuel would be less than 10 micro-rem/hr. During normal operation no workers need be stationed adjacent to a glove-box. Process viewing can be accomplished by video techniques. Direct access to a glove-box is limited to times needed for unscheduled repair or scheduled maintenance activities. At such times the bulk of the radioactive material in the affected glove-box would be moved to a temporary storage area to minimize the local radiation source. The repair or maintenance activity would be completed as quickly as possible because of the manual access to the affected equipment in the glove-boxes. The fuel pin assembly and the fuel bundle assembly are also planned to be automated processes. When close-in access is required for repair or maintenance, portable shielding can be used in conjunction with moving fuel pellets, fuel pins, or fuel bundles away from the affected equipment to minimize radiation dose during repair or maintenance activities. The areas of the fuel fabrication facility containing the radioactive material will be separated by walls from non-radioactive areas and access to the radioactive areas will be controlled. Finally,

special attention will be paid to the two areas where large quantities of radioactive fuel are to be stored.

These areas are the plutonium receipt and storage area and the assembled fuel bundle storage area. For safeguards reasons the plutonium receipt area and storage area will be a vault which provides extremely good shielding. Movement of plutonium into and out of the vault will be accomplished by using a computer-controlled robot. Exposure in this area is therefore expected to be minimal. After completing bundle assembly, two bundles would be loaded into a steel storage and transfer container. This container would provide additional shielding as well as protection for the bundles. The container would be moved by a computer-controlled overhead crane into a shielded storage area. With these features, the annual exposure of the workers in the fuel fabrication area is expected to be less than 0.5 person-rem/year.

Table 3.9-1. Radioactivity (Curies/gm) of Plutonium

Isotope	Fast Reactor Plutonium				Weapons-Grade Plutonium			
	w/o	Gamma	Neutron	Total	w/o	Gamma	Neutron	Total
Pu-238	2	0.046		0.337				-
Pu-239	54	0.002		0.036	93.7	0.003		0.062
Pu-240	25	0.009	6.2e-9	0.067	5.7	0.002	1.4e-9	0.015
Pu-241	11	0.004		11.34	0.3	0.000		0.308
Pu-242	7	0.000						
AM-241	1	0.023		0.061	0.3	0.006		0.017
Total	100	0.084	6.2e-9	11.84	100	0.011	1.4e-9	0.402

*Total radioactivity also includes alpha and beta particle emissions.

By design, the fuel fabrication facility will have extremely small airborne release of radioactive elements. Coupled with the extremely small airborne release rate, the release dose rate is significantly reduced (compared to other plutonium fuel) because of the composition of the weapons-grade plutonium. The very low content of Plutonium-241 and

Americium-241 and the lack of Plutonium-238 reduces the dose rate (total of all particles emitted) by a factor of 30 compared to plutonium used in making fuel fabrication fast reactors. By design the interior volume of the fuel fabrication facility will be controlled to be at about one-half inch water negative pressure. The negative pressure condition would be established with a ventilation system which sucks air from the interior of the building through multiple layers of small particle filters and discharges the air via a stack to the atmosphere. Within the fuel fabrication facility, when the plutonium oxide and uranium oxide are in powder form they will be contained in hermetically sealed (welded) steel containers. These containers will only be opened after entrance into the sealed-off glove-box system for processing these powders into the pellet form. The sealed-off glove box system will be operated at a negative pressure (about a negative one-half inch of water) relative to the pressure in the fuel fabrication facility. The air in the glove-box system will be drawn through multiple filters and eventually discharged via the facility stack. At key stages in the batch processing like the mechanical blending stage, and the pellet grinding stage special vacuuming and dust collection features which have been developed over the years will be used to minimize the release of dust into the glove-box system. Another key stage is pellet sintering. At sintering temperatures, Americium-241 is volatilized and some of it diffuses out of the pellet and into the furnace gas. A cooling trap downstream of the furnace is added to collect the vaporized americium. The latest technology available for cleanup of the stack effluents to remove airborne radiation will be implemented with this design approach, the total airborne release of radioactive material out the stack of the fabrication facility is roughly estimated to be less than about 200 micro-curies/year.

Except for the wet scrap recycle area the fuel fabrication plant is planned to utilize a dry process. There is no water usage requirement for the main fuel fabrication processes. Water would be used in some support areas like cask cleaning activities and in emergency decontamination areas. Trace radioactive elements are expected when cleaning contaminated shipping casks/containers or when cleaning up anticipated small accidental releases. The liquids from the wet scrap recycle processes will be treated and the water evaporated off. The contaminated water from cask cleaning, etc. would also be treated, filtered and evaporated off. The liquid, solid waste would be collected and sent to the

repository. Other water used in the fabrication process would be for auxiliary functions like cooling of furnaces. This water would be monitored and recycled. If this water were to become contaminated, it would be treated, cleaned and evaporated off to remove the contamination. With the low requirements for water usage, and the planned cleanup and subsequent recycling of the water used in the radioactive area of the fuel fabrication plant, any water release is expected to be of drinking water quality.

3.10 FUEL DEVELOPMENT NEEDS

Simultaneously with the mixed-oxide fuel facility planning a few process development items will need to be addressed. The first item is the ceramic fabrication process of mixed-oxide/gadolinia fuel. The fabrication route and production experience of UO_2 and Gd_2O_3 are well known. Only a limited amount of information or production experience is available on mixed-oxide/gadolinia. It will be necessary to confirm the processing of this material. This study would evaluate all the steps of the process from blending through sintering and would utilize plutonia which was converted by the reference process. Depending on the arrangements made, this program could take anywhere from six months to one year to complete. A second item would be to fabricate a lead test assembly(s) for irradiation and verification of its performance in reactor. This program would parallel the mixed-oxide/gadolinia fuel process development and again could take up to a year to complete.

3.11 COST AND SCHEDULE INPUT

The equipment for the MOX Fuel Fabrication Facility has been specified to meet the process requirement and production volume needs of the spiking, spent fuel, and plutonium destruction cases. The annualized forecasts for the three cases are summarized in Section 10.

The major items for key systems to be purchased and installed in the plant is shown in Table 3.11-1.

Table 3.11-1. Equipment and Installation Costs

SYSTEM		EQUIPMENT COST \$K			INSTALLATION COST \$K		
		SPIKING	SPENT FUEL	PuDEST	SPIKING	SPENT FUEL	PuDEST
Ceramic Area MOX Fab		7821	6718	7821	4277	3789	4277
Ceramic Area MOX/GD Fab		7821	6230	1462	4268	3679	6193
Ceramic Area MOX Rod Fab		1462	1462	1462	1250	1250	1250
Ceramic Area MOX/GD Rod Fab		1426	731	12283	1250	625	1250
Waste Recovery Area		7450	6525	7450	5362	4536	5362
Bundle Assembly		2995	2995	2995	75	75	75
General Storage & Shipping		70	70	70	70	70	70
Instrument Laboratory		140	140	140	1	1	1
Chemical/Metallurgical Lab		1465	1465	1465	41	41	41
Cold Maintenance Area		555	555	555	12	12	12
Hot Maintenance Area		555	555	555	17	17	17
Manufacturing Laboratory		205	205	205	7	7	7
Nuclear Safety		1435	1435	1435	324	324	324
Material Control		1975	1975	1975	34	34	34
Men/Women's Change Room		60	50	60	2	2	2
Laundry		320	300	320	7	7	7
Waste Storage		125	125	125	13	13	13
Powder Receiving		145	130	145	34	34	34
PU02 Storage		750	500	750	58	48	2400
Rod Storage		430	175	430	22	15	22
Bundle Storage & Containers		850	750	850	38	24	38
Security		470	470	470	100	100	100
	Sub Total	38525	33561	43023	17261	14703	21528
Support And Utility Equipment		6475	5939	7477	2900	2500	3200
	Total \$K	45000	39500	50500	20521	17203	24728

References

- 3-1 Bambilla, G., P. Gerontopulos, A. G. Facchini, and D. Neri, "The SNAM Process for the Production of Ceramic Nuclear Fuel Microspheres," in Symposium on Sol-Gel Process and Reactor Fuel Cycles, Gatlinburg, Tennessee, May 4-7, 1970, CONF-700502, pp. 191-209
- 3-2 Fukushima, S., T. Ohmichi, and M. Handa, "Thermal conductivity of Near-Stoichiometric (U,Pu,Nd)₀₂ and (U,Pu,Eu)₀₂ Solid Solutions," Journal of Nuclear Material, Vol. 116 (1983), pp. 287-296
- 3-3 Fukushima, S., T. Ohmichi, A. Maeda, and M. Handa, "Thermal Conductivity of (Pu_{1-x}Nd_y)₀₂₋₆ and (Pu_{1-x}Y_y)_{02-y} Solid Solutions," Journal of Nuclear Material, Vol. 115 (1983), pp. 118-127
- 3-4 Mobley, W. N., "fuel Fabrication," Summary of Nuclear Fuels Seminar, Wrightsville Beach, North Carolina, May 14-17, 1972, pp. 4-1 thru 4-10
- 3-5 Hagie, L. T., et al., "Facilities Startup and Fabrication of Mixed Oxide Fuel for the Garigliano Reactor," NEDE-12457, General Electric Company, (October 1973)
- 3-6 Krellmann, J., "Plutonium Processing at the Siemens Hanau Fuel Fabrication Plant," Nuclear Technology, Vol. 102 (April 1993) pp. 18-28
- 3-7 Vanhellemont, G., "Fabrication of Plutonium Fuel," Nuclear Europe, Vol. 12 (1985) pp. 23-34
- 3-8 Ross, W. J., "Fabrication of FFTF Fuel Pins at Babcock & Wilcox," Journal American Nuclear Society, Vol. 21, June 1975, pp. 248-250
- 3-9 Matthews, R. B., "Ceramic Fuel Development for Space Reactors," Ceramic Bulletin,

Vol. 71, No. 1 (1992) pp. 96-101

3-10 Wadekamper, D. C., and M. L. Thompson, "Mixed Oxide Sintering Furnace Review, GEFR-00572, General Electric Company (July 1981)

3-11 Benedict, M., et al., "Plutonium and Other Actinide Elements," Nuclear Chemical Engineering, McGraw-Hill, New York, New York (1981) pp. 466-514

3-12 Goris, P., DeMarschman, "Fast Flux Test Facility (FFTF) Fuel Pin Non-Destructive Assay Measurements," Safeguarding Nuclear Materials Vol II, International Atomic Energy Agency, Vienna, 1976; pp. 599-601.

4.0 REACTOR COMPLEX

The total reactor complex includes all the facilities within the site boundary. The fabrication facility is described in Section 3, and the ex-reactor spent fuel storage facility in Section 5. This section covers the ABWR plant, and also the remaining site facilities. Section 4.1 provides the site layout. The ABWR plant description and parameters are provided in Section 4.2 and 4.3. Section 4.4 presents the safeguards and security, and Section 4.5, the safety and environmental aspects of the ABWR plant using mixed-oxide fuel. The development needs and the cost and schedule for the ABWR plant are included both in Section 4.6 and 4.7, and in the overall discussions in Section 9 and 10.

4.1 REACTOR COMPLEX OVERALL DESCRIPTION AND LAYOUT

Site layouts are shown on Figures 4.1-1 and 4.1-2. Figure 4.1-1 shows a typical site for a one-reactor case and Figure 4.1-2 shows a typical layout for a two-reactor case. The site layout for the six-reactor case is similar.

Although it is expected that the site will be located on an existing government owned reservation, DOE has instructed study participants to assume that general site conditions are in accordance with the EPRI Kenosha, Wisconsin Standard Hypothetical East/West Central Site for Nuclear Power Plants as defined in Oak Ridge National Laboratory TM-10071/R3, "Cost Estimate Guidelines for Advanced Nuclear Power Technologies", dated March, 1993. The site is typical of power generation facilities located in middle America which has access to water and rail transportation. Detailed site conditions are defined in Appendix F of this cost guidelines document.

Since the fuel fabrication facility if required to operate before reactor startup, this facility has been provided with perimeter safeguards fencing separate from the remainder of the reactor complex. This will allow completion of reactor construction while the fuel

Figure 4.1-1. Site layout One Reactor

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Centimeter

Inches

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3 of 6

fabrication facility produces the initial reactor core(s).

Major facilities within the physical security fencing for the reactor complex include the reactor-turbine(s), the spent fuel facility, a radwaste processing building, the cooling tower system(s) and ultimate heat sink(s), and various utility and support systems including a hot machine shop, fire protection tanks and pump house, water treatment facilities and electrical transformer area(s). Mechanical draft cooling towers have been shown in accordance with the above-referenced cost estimate guidance document (Appendix F, Section F.5).

Facilities shown within the fuel fabrication facility physical security fencing include the fuel fabrication building, a ventilation chiller system pad, and areas for storage of gases and process chemicals.

Facilities located outside the security fencing but inside the industrial security perimeter fence include an administration building, training center, warehouse, sewage plant, and area(s) for electrical switchgear for power transmission on and off-site.

Rail access has been provided to the reactors to allow rail shipment of spent fuel casks, construction materials, and major equipment components as required.

4.2 ADVANCED BOILING WATER REACTOR PLANT DESCRIPTION

The ABWR is a direct cycle BWR power plant. A simplified diagram showing overall plant operation is provided in Figure 4.2-1. Water is heated in the reactor core, producing saturated steam. The steam is separated from recirculation water, dried, and directed to the steam turbine-generator. The electrical output from the generator is approximately 1350 MWe. The condensate from the turbine is returned as feedwater to the reactor core.

Figure 4.2-1 ABWR Reactor Cooling Schematic

The ABWR Standard Plant includes the following:

- Reactor Building (including containment)
- Control Building
- Turbine Building
- Radwaste Building

In addition to these buildings, the ABWR plant includes the supporting facilities as shown in Figure 4.2-2. A cutaway of the ABWR is provided in Figure 4.2-3. The main additions to the standard plant site for plutonium disposition are the additions of (1) the Fuel Fabrication Building and (2) the Ex-Reactor Spent Fuel Storage Building, which are described separately in Chapters 3 and 5, respectively. There are a few modifications within the ABWR plant, and these are mainly in scale, such as provisions for additional on-site and off-site electrical power, service water, radwaste capacity, and safeguards. A modification to the reactor building is anticipated if the inclined fuel transfer system between the in-reactor and ex-reactor fuel storage pools is adopted, as discussed in Section 5.

The reactor building houses the containment, drywell, and major portions of the nuclear steam supply system, steam tunnel, refueling area, diesel generators, essential power, non-essential power, emergency core cooling systems, HVAC system, and other supporting systems. The reactor core is 5.2 meters in diameter by 3.7 meters high. It consists of 872 fuel assemblies and 205 control rods enclosed within the reactor pressure vessel (Figure 4.2-4). The core produces a thermal output of 3930 MWt. The power level is adjusted by positioning control rods within the core, and by changing the recirculation flow rate through the core. The ABWR employs bottom-entry and bottom-mounted control rods. This control rod drive configuration allows refueling without removal of control rods and drives, and allows testing with an open vessel prior to initial fuel loading or at each refueling operation. The electro-hydraulic fine motion control rod drives provide fine-motion control for rod positioning during normal operation and hydraulic pressure for scram insertion. The electric motor also provides increased reliability by providing a diverse means of rod motion as a backup to the hydraulic scram with motor run-in following scram.

NO.	FACILITY
1	REACTOR CONTAINMENT
2	REACTOR BUILDING
3	CONTROL BUILDING
4	MAIN STEAM/FEEDWATER TUNNEL
5	TURBINE BUILDING
6	SERVICE BUILDING
7	RADWASTE BUILDING
8	HOUSE BOILER
9	CONDENSATE STORAGE TANK
10	MAIN TRANSFORMER
11	NORMAL SWITCHGEAR
12	DIESEL OIL STORAGE TANK (3)
13	STACK
14	EQUIPMENT ENTRY LOCK
15	FIRE PROTECTION WATER STORAGE TANK (2)
16	FIRE PROTECTION PUMPHOUSE

Figure 4.2-2 ABWR Plant Buildings and Support Facilities

Figure 4.2-3

ABWR

Advanced Boiling Water Reactor

1	Reactor Pressure Vessel	18	HPCF Pump
2	Reactor Internal Pumps	19	RCIC Steam Turbine and Pump
3	Fine Motion Control Rod Drives	20	Diesel Generator
4	Main Steam Isolation Valves	21	Standby Gas Treatment Filter and Fans
5	Safety/Relief Valves	22	Spent Fuel Storage Pool
6	SRV Quenchers	23	Refueling Platform
7	Lower Drywell Equipment Platform	24	Shield Blocks
8	Horizontal Vents	25	Steam Dryer and Separator Storage Pool
9	Suppression Pool	26	Bridge Crane
10	Lower Drywell Flooder	27	Main Steam Lines
11	Reinforced Concrete Containment Vessel	28	Feedwater Lines
12	Lower Drywell Equipment Hatch	29	Main Control Room
13	Wetwell Personnel Lock	30	Turbine-Generator
14	Hydraulic Control Units	31	Moisture Separator Reheater
15	Control Rod Drive Hydraulic System Pumps	32	Combustion Turbine-Generator
16	RHR Heat Exchanger	33	Air Compressor and Dryers
17	RHR Pump	34	Switchyard

Figure 4.2-4

Advanced Boiling Water Reactor Assembly

- 1 Vent and Head Spray
- 2 Steam Dryer
- 3 Steam Outlet Flow Restrictor
- 4 Steam Separators
- 5 RPV Stabilizer
- 6 Feedwater Sparger
- 7 Shutdown Cooling Outlet
- 8 Low Pressure Flooder (LPFL) and Shutdown Cooling Sparger
- 9 High Pressure Core Flooder (HPCF) Sparger
- 10 HPCF Coupling
- 11 Top Guide
- 12 Fuel Assemblies
- 13 Core Shroud
- 14 Control Rod
- 15 Core Plate
- 16 In-Core Instrument Guide Tubes
- 17 Control Rod Guide Tubes
- 18 Core Differential Pressure Line
- 19 Reactor Internal Pumps (RIP)
- 20 Thermal Insulation
- 21 Control Rod Drive Housings
- 22 Fine Motion Control Rod Drives
- 23 RIP Motor Casing
- 24 Local Power Range Monitor

GE Nuclear Energy

All large nozzles in the reactor pressure vessel below the core are eliminated. The ABWR incorporates ten internal recirculation pumps located at the bottom inside the reactor pressure vessel. This reduces in-service inspection requirements, reduces personnel exposure, and reduces space requirements in the containment through elimination of the external recirculation piping. The use of the reactor internal pump feature allows for the optimization of the emergency core cooling systems and assures that the core will not be uncovered during any postulated pipe break.

The ABWR operates at essentially constant pressure. The steam pressure is about half the primary system pressure of a pressurized water reactor, while producing steam of approximately equal pressure and quality. The lower pressure and compact system is easier and less time-consuming to install. Also, the lower pressure operation permits application of fast-acting and effective emergency core cooling systems.

The ABWR plant is supported by the specialized functions of its auxiliary systems. Auxiliary systems used for normal plant operation include the reactor water cleanup system, the shutdown cooling function of residual heat removal system, the fuel building and containment pools cooling and filtering system, the reactor building cooling water system, and the radioactive waste treatment system. The systems used as backup (standby) or emergency systems include the standby liquid control system, the residual heat removal system's low pressure core flooders, containment spray and suppression pool cooling functions, the high pressure core flooders system, the reactor core isolation cooling system, the automatic depressurization system and the remote shutdown system.

The reactor building also provides the nuclear containment. The primary containment system design for this plant incorporates the drywell/pressure suppression feature of previous BWR containment designs into a dry containment type structure. In fulfilling its design basis as a fission product barrier, the primary containment is a low leakage structure even at the increased pressures that could follow a main steamline rupture or a fluid system line break.

The control building houses the control room, the computer facility, the cable tunnels, some of the plant essential switchgear, the reactor building cooling water system, and the essential HVAC system.

The turbine building houses all equipment associated with the main turbine-generator and other auxiliary equipment. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralization. The turbine-generator is equipped with an electrohydraulic control system and supervisory instruments to monitor performance. The gross electrical output of the turbine-generator is approximately 1400 MWe.

The radwaste building houses all equipment associated with the collection and processing of the liquid and solid radioactive waste generated by the plant. The liquid radwaste management system collects, monitors, and treats liquid radioactive wastes for return to the primary system whenever practicable. The solid radwaste management system provides for the safe handling, packaging, and short-term storage of radioactive solid and concentrated waste.

The following sections provide further description of these systems. The detailed parameters (size, temperature, pressure, flow, etc.) for these systems are provided below in Section 4.3.

4.2.1 Nuclear Boiler Systems

The nuclear boiler system includes a direct-cycle forced-circulation boiling water reactor that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear steam supply system for rated power conditions is shown in Figure 4.2.1-1. A description of the subsystems and selected components is contained in the following paragraphs:

87-592-01

Figure 4.2.1-1 ABWR Heat Balance at Rated Power

4.2.1.1 Reactor Pressure Vessel System

The reactor pressure vessel system consists of the reactor pressure vessel with the reactor internal pump casings; core and supporting structures; the steam separators and dryers; the control rod guide tubes; the spargers for feedwater, residual heat removal and core flooders system; the control rod drive housings; the in-core nuclear instrumentation guide tubes and housings; and other components (Figure 4.2-4). The reactor pressure vessel has an internal diameter of approximately 7.1 meters, an overall height of about 21 meters and a wall thickness of approximately 17 centimeters.

The main connections to the vessel include four main steamlines, six feedwater lines, cooling and power for the 10 reactor internal pumps, 205 control rod drives and 208 in-core nuclear instrument detectors, core flooders lines, residual heat removal lines, head spray and vent lines, and various process instrumentation sensing lines.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig (87.9 Kg/cm^2 at 302 Celsius). The nominal operating pressure in the steam space above the separators is 1040 psia. The vessel is fabricated of low alloy steel and is clad internally with stainless steel of Ni-Cr-Fe Alloy (except for the tophead, reactor internal pump motor casing, nozzles other than the steam outlet nozzle, and nozzle weld zones which are unclad).

The reactor core (Figure 4.2.1-2) is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the four approximately 640cm inside-diameter main steamlines.

A venturi-type flow restrictor with a capacity of 200 percent of rated steam line flow is a part of the reactor pressure vessel nozzle configuration for each main steamline. These restrictors limit the flow of steam from the reactor vessel before the main steamline

□ Fuel assemblies	872	○ Power range monitors	52 x 4
+ Control rods	205	■ Startup range neutron monitors	10
		△ Neutron sources	5

Figure 4.2.1-2 ABWR Core Layout

isolation valves are closed in case of a main steamline break outside the containment.

4.2.1.2 Fuel Assembly

The nuclear fuel assembly contains fissionable material which produces thermal power while maintaining structural integrity. The configuration of the fuel assembly (Figure 4.2.1-3) consists of 60 fuel rods and one water rod in an 8x8 square array, spacers, water rods, upper and lower tie plates, channel and channel fastener, all fabricated into a transportable, interchangeable assembly. The outer envelope of the fuel assembly is square with distinguishing features which provide support, identification, orientation and handling capabilities. The rods, spacers, and tie rods constitute the fuel bundle. The fuel channel encloses the fuel bundle and provides: 1) a barrier between two parallel coolant flow paths, one for flow inside the bundle and the other for flow in the bypass region between channels; 2) a bearing surface for the control rod; and 3) rigidity for the fuel bundle. The channel fastener attaches the channel to the fuel bundle, and along with the channel spacer buttons provides channel-to-channel spacing with resilient engagement.

4.2.1.3 Reactor Coolant Systems

Reactor Recirculation System: There are 10 reactor internal pumps which provide a continuous circulation path inside the reactor vessel for the core coolant flow. These pumps (Figure 4.2.1-4) each circulate approximately 5,800 tons per hour of water through the reactor core. The reactor internal pumps are located at the bottom of the vessel. The pump motors are enclosed in casings which are a part of the vessel. A break in the casing will result in a leak flow that is less than the emergency core cooling system design capacity so that full core coverage will be maintained. The internal pumps are a wet motor design with no shaft seals, thereby providing increased reliability, reduced maintenance requirements and decreased operational radiation exposure. The speed of the pumps is adjusted to control flow through the reactor by the recirculation flow control system.

Figure 4.2.1-3 Fuel Assembly Schematic

Figure 4.2.1-4 Section of Reactor Internal Pump (RIP)

4.2.1.4 Reactor Control and Backup Safety Systems

Control Rod Drive System: The control rod design is a sheathed cruciform array of stainless steel tubes filled with Boron Carbide (Figure 4.2.1-5).

When scram is initiated by the reactor protection system, the control rod drive system inserts the negative reactivity necessary to shut down the reactor. Each control rod is normally controlled by an electric motor unit (Figure 4.2.1-6). When a scram signal is received, high-pressure water stored in nitrogen charged accumulators forces the control rods into the core and the electric motor drives are signalled to drive the rods into the core. Thus, the hydraulic scram action is backed up by an electrically energized insertion of the control rods. During normal operation, control rods are positioned by means of control signals from the rod control and information system.

The fine motion control rod drive design incorporates an electro mechanical brake keyed to the motor shaft. The brake is normally engaged by passive spring force. It is disengaged when the spring load is overcome by the energized magnetic force. The braking torque between the motor shaft and the control rod drive spool piece is sufficient to prevent control rod ejection in the event of failure in the pressure retaining parts of the drive mechanism. The brake is designed so that its failure will not prevent the control rod from rapid insertion.

The control rod drive incorporates a centrifugal braking mechanism to limit rod ejection to a maximum distance of 0.5 feet in the event of failures in the pressure retaining parts of the drive mechanism. The brake is designed so that its failure will not prevent the control rod from rapid insertion (scram) when required. A nuclear excursion is prevented and thus the fuel barrier is protected.

Rod Control and Information System: The rod control and information system provides the means by which control rods are positioned from the control room for power control. The system operates the rod drive motors to change control rod position. For operation

87-226-02

Figure 4.2.1-5 ABWR Control Rod Assembly

Figure 4.2.1-6 Fine Motion Control Rod Drive (FMCRD) Schematic

in the normal gang movement mode, one gang of control rods can be manipulated at a time. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

Recirculation Flow Control System: During normal power operation, the speed of the reactor internal pumps is adjusted to control flow. Adjusting reactor internal pump speed changes the coolant flow rate through the core and thereby changes the core power level. The solid-state adjustable speed drives provide variable voltage, variable frequency electrical power to the reactor internal pump motors. In response to plant needs, the recirculation flow control system adjusts the adjustable speed drives power supply output to vary reactor internal pump speed, core flow and reactor power. The reactor internal pump has a low rotating inertia. Coupled with the solid state adjustable speed drives, the reactor internal pump can respond quickly to load transients and operator demands.

Standby Liquid Control System: The standby liquid control system provides a backup method to bring the nuclear reactor to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect of shutting down from rated power to cold shutdown condition.

The standby liquid control system is automatically initiated or can be manually initiated from the main control room to pump the neutron absorbing solution into the reactor.

The system includes a boron solution tank, a test water tank, two positive displacement pumps, motor-operated injection and pump suction valves, and associated local piping, valves and controls in the reactor building outside the primary containment. The liquid is piped into the reactor vessel through the high pressure core flood line downstream of the high pressure core flood inboard check valve inside the primary containment.

The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain

reaction in the fuel. The specified neutron absorber solution is sodium pentaborate. At all times, when it is possible to make the reactor critical, the standby liquid control system will be able to deliver enough sodium pentaborate solution into the reactor to assure reactor shutdown.

4.2.1.5 Main Steamline Systems

The main steamlines direct the flow of steam from the reactor vessel to the main turbine.

Nuclear System Pressure Relief System: A pressure relief system consisting of 18 safety/relief valves mounted on the main steamlines is provided to prevent excessive pressure inside the nuclear system as a result of operational transients or accidents. The safety/relief valves with spring type actuators each have a capacity of approximately 400 tons per hour and discharge into the suppression chamber.

Main Steamline Isolation Valves (MSIVs): Each main steamline is provided with two isolation valves in series, one on each side of the primary containment barrier. All pipelines that penetrate the containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities. Automatic isolation valves are provided in each main steamline. Each is powered by steam pressure, spring force and/or high pressure nitrogen gas. The closure period is three to four seconds. These valves fulfill the following objectives: 1) prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the reactor vessel resulting from either a major leak from the steam piping outside the containment or a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel; 2) limit the release of radioactive materials by isolating the reactor coolant pressure boundary in case of the detection of high steam line radiation.

4.2.1.6 Reactor Water Cleanup System

The reactor water cleanup system recirculates a portion of reactor coolant through a

filter-demineralizer to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions during plant heatup and provides clean water for the head spray nozzle.

The reactor water cleanup system is a closed-loop system of piping, circulation pumps, a regenerative heat exchanger, non-regenerative heat exchangers, reactor water pressure boundary isolation valves, and two filter-demineralizers.

During normal plant operation, the reactor water cleanup system takes suction from the single bottom drain on the reactor pressure vessel and the shutdown cooling suction line of residual heat removal Loop B, cools the water in the regenerative and non-regenerative heat exchangers, pumps it through the filter demineralizers, reheats it in the regenerative heat exchanger, and returns it to the reactor pressure vessel through the feedwater lines. The return can also be directed to the reactor pressure vessel head spray line as required for rapid reactor pressure vessel cool down. During plant heatup, excess water from reactor water swell can be directed to the main condenser, radwaste or the suppression pool.

4.2.1.7 Reactor Vessel Instrumentation

In addition to instrumentation for the nuclear safety systems and engineered safety features, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. This instrumentation monitors reactor vessel pressure, water level, coolant temperature, reactor core differential pressure, coolant flow rates, and reactor vessel head seal leakage.

The steam flow instrumentation is connected to the venturi-type steam nozzle of the reactor pressure vessel. The instrumentation provides high nozzle flow isolation signals in case of a main steam line break.

4.2.2 Emergency Core Cooling Systems

In the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant, three independent divisions of the emergency core cooling system are provided to maintain fuel cladding below the temperature limit as defined by 10CFR50.46. Each division contains one high pressure and one low pressure makeup system. The ADS discussed earlier forms part of the plant's emergency core cooling capabilities.

4.2.2.1 Residual Heat Removal System

The residual heat removal system is a system of pumps, heat exchangers, and piping that fulfills the following functions: 1) removes decay and sensible heat during and after plant shutdown; 2) injects water into the reactor vessel following a loss-of-coolant accident to reflood the core in conjunction with other core cooling systems; and 3) removes heat from the containment following a loss-of-coolant accident by cooling and recirculating the suppression pool water. The residual heat removal system pump motors are powered by the emergency diesel generators if normal power is not available. (See Figure 4.2.2-1 for a schematic diagram of the residual heat removal system.)

Low Pressure Core Flooder (LPCF): Low pressure core flooding is an emergency mode of the residual heat removal system, that acts in conjunction with other injection systems. LPCF uses the residual heat removal pumps to inject cooling water from the suppression pool into the pressure vessel. Low pressure core flooding operation provides the capability of core flooding at low vessel pressure following a LOCA in time to maintain cladding below the prescribed temperature limit. Depending upon the size of the break, the automatic depressurization system (ADS) may be required to reduce pressure in the vessel before the LPCF will inject water.

Automatic Depressurization System: The automatic depressurization system utilizes a subset of the safety/relief valves. In case high pressure core flooder or reactor core isolation system cannot maintain water level, the ADS rapidly reduces reactor vessel pressure in a

Figure 4.2.2-1 ABWR Emergency Core Cooling Systems (Low Pressure)

Figure 4.2.2-2 ABWR Emergency Core Cooling Systems (High Pressure)

loss-of-coolant accident, enabling the residual heat removal system in the low-pressure core flood mode to deliver cooling water to the reactor vessel.

The automatic depressurization system uses eight of the safety/relief valves that are part of the nuclear pressure relief system. The safety relief valves used for ADS are set to open on detection of appropriate low reactor water level and high drywell pressure signals. The automatic depressurization system will not be activated unless one or more residual heat removal or high pressure core flood pumps are operating. This is to ensure that adequate coolant will be available to maintain water level after depressurization.

Suppression Pool Cooling: The suppression pool cooling is placed in operation to: 1) limit the temperature of the water in the suppression pool following a design basis LOCA; 2) control the pool temperature during normal operation; and 3) reduce the pool temperature following a transient which results in operation of the safety/relief valves and/or the RCIC system. In this mode of operation, the residual heat removal main system pumps take suction from the suppression pool and pump the water through the residual heat removal heat exchangers where cooling takes place by transferring heat to the service water. The fluid is then discharged back to the suppression pool.

Wetwell/Drywell Spray: A spray system is provided for wetwell/drywell cooling in the suppression chamber and drywell air space. The wetwell/drywell spray can be initiated manually to reduce high containment pressure and provide primary containment atmospheric cooling following a LOCA. To initiate drywell spray, drywell pressure must exceed its high pressure setpoint, or the drywell spray valves can not be opened.

Shutdown Cooling: The shutdown cooling mode of residual heat removal removes residual heat after reactor shutdown in preparation for refueling or nuclear system servicing. This mode is initiated and shut down manually. The flow path is from the vessel through the reactor shutdown cooling lines, through the residual heat removal heat exchangers and back to the vessel (through the feedwater line on one loop and the low pressure core flood lines on the other two loops).

4.2.2.2 High Pressure Core Flooder System

High pressure core flooders are shown schematically in Figure 4.2.2-2. They are provided in two emergency core cooling divisions to maintain an adequate coolant inventory inside the reactor vessel to limit fuel cladding temperatures in the event of breaks in the reactor coolant boundary. The system is initiated by either high pressure in the drywell or low water level in the vessel. It operates independently of all other systems over the entire range of system operating pressures. Each division's pump has a capacity that varies from approximately 180 cubic meters per hour at a pump head of 890 meters to a capacity of 730 cubic meters per hour at a pump head of 190 meters. The high pressure core flooder system pump motors are powered by the diesel generators if normal power is not available. The system may also be used as a backup for the reactor core isolation cooling system.

4.2.2.3 Reactor Core Isolation Cooling System (RCIC)

The reactor core isolation cooling system provides makeup water to the reactor vessel when the vessel is isolated and is part of the emergency core cooling network. The reactor core isolation system uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel for required events. The reactor core isolation cooling system is shown schematically in Figure 4.2.2-2.

One emergency core cooling system division contains the reactor core isolation cooling system which consists of a steam-driven turbine which drives a pump assembly and the turbine and pump accessories. The system also includes piping, valves and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the main steam lines inside the primary containment and goes to the RCIC turbine. The turbine exhausts to the suppression pool. Makeup water is supplied from the condensate storage tank or the suppression pool with preferred source being the condensate storage tank. RCIC pump discharge lines include the main discharge line to the feedwater line which injects to the reactor vessel, a test return line to the suppression pool and a cooling

water supply line to auxiliary equipment.

Normally following a reactor scram, steam generation in the reactor core continues at a reduced rate due to the core fission product decay heat. The feedwater system supplies the makeup water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel drops due to continued steam generation by decay heat and loss through the safety relief valves. Upon reaching predetermined low level, the RCIC system is initiated automatically. RCIC may also be initiated manually to maintain reactor water level.

RCIC, with its steam driven turbine power, provides a diverse makeup source during loss of AC power events.

4.2.3 Containment and Environmental Control Systems

4.2.3.1 Primary Containment System

The primary containment system design for this plant incorporates the drywell/pressure suppression feature of previous BWR containment designs into a dry containment type structure. In fulfilling its design basis as a fission product barrier, the primary containment is a low leakage structure even at increased pressures that could follow a main steamline rupture or a fluid system line break.

The containment structure (Figure 4.2.3-1) consists of the following major components:

- 1) A drywell comprised of two volumes; a) an upper drywell volume surrounding the reactor pressure vessel (reactor pressure vessel) and housing the steam and feedwater lines and other connections of the primary coolant system, safety/relief valves and the

Figure 4.2.3-1 ABWR Containment Nomenclature

drywell HVAC coolers, and b) a lower drywell volume housing the reactor internal pumps, fine motion control rod drives, and under vessel components and servicing equipment. The upper drywell is a cylindrical, reinforced concrete structure with a movable steel head and a reinforced concrete diaphragm floor. The cylindrical reactor pressure vessel pedestal which is connected rigidly to the diaphragm floor separates the lower drywell from the wetwell. It is a prefabricated steel structure filled with concrete after erection. Ten drywell connecting vents, approximately one by two meters in cross-section, are built into the reactor pressure vessel pedestal and connect the upper and lower drywells. The drywell connecting vents are extended downward via 1.2 meters inside diameter steel pipes, each of which has three horizontal 0.7 meters diameter vent outlets into the suppression pool.

- 2) A wetwell consisting of an air volume and suppression pool filled with water to rapidly condense steam from a reactor vessel blowdown via the safety/relief valves or from a break in a major pipe inside the drywell through the vent system. The wetwell boundary is a cylindrical reinforced concrete wall which is continuous with the upper drywell boundary. A reinforced concrete mat foundation supports the entire containment system, and enclosed structures, systems and components, and extends to support the reactor building surrounding the containment.
- 3) The containment structure includes a steel liner to reduce fission product leakage to allowable levels. All normally wetted surfaces of the liner in the suppression pool are stainless steel. Penetrations through the liner for the drywell head, equipment hatches, personnel locks, piping, electrical and instrumentation lines are provided with seals and leak tight connections. The allowable leakage is 0.5 percent per day from all sources.

4.2.3.2 Leak Detection and Isolation System

The leak detection and isolation system consists of temperature, pressure, flow, and fission-product sensors with associated instrumentation, alarm, and isolation functions. This system detects and annunciates leakage and provides signals to close containment isolation valves, as required, in the following systems: 1) main steamlines; 2) reactor water cleanup

system; 3) residual heat removal system; 4) reactor core isolation cooling system; 5) feedwater system; 6) emergency core cooling systems; and 7) other miscellaneous systems.

Small leaks generally are detected by monitoring the air cooler condensate flow, radiation levels, equipment space temperature, and drain sump fill-up and pump-out rates. Large leaks are also detected by changes in reactor water level, drywell pressure, and changes in flow rates in process lines.

4.2.3.3 Atmospheric Control System

The atmospheric control system is provided to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during shutdown for refueling or equipment maintenance, and during limited periods of time to permit access for inspection at low reactor power.

4.2.3.4 Flammability Control System

The flammability control system is provided to control the potential buildup of hydrogen from design-basis radiolysis of water. A recombiner system is provided to control the concentration of hydrogen produced by radiolysis in the primary containment. The objective of the atmospheric control system and flammability control system is to preclude combustion of hydrogen and damage to essential equipment and structures.

4.2.3.5 Drywell Cooling System

The drywell cooling system is a recirculating system consisting of three fan coil units. Normally, two of the three fan coil units are in operation. Each fan coil unit consists of one cooling coil, a drain pan, and a centrifugal fan. Cooling water comes from the reactor cooling water system. The two sets of cooling coils are arranged in series. The return air passes over the first coil, which is cooled by the reactor cooling water system. Part of the

cooled air is then cooled by the second coil, which is cooled by HVAC normal cooling water (HNCW). This twice-cooled air is mixed with air which bypassed the second cooling coil. Condensate that drips from the coils is routed to the drain system via the leak detection system. Instrumentation is installed in the drain line to monitor cooler condensate flow and to provide input to the leak detection system.

The drywell cooling system supplies conditioned air to a common distribution header. The air/nitrogen is then ducted to areas within the drywell for equipment cooling. These areas consist of the drywell head area, upper drywell, lower drywell, shield wall annulus, and the wetwell air space. Gravity dampers and adjustable volume dampers control distribution of the air/nitrogen to the drywell space.

High drywell temperatures are alarmed in the main control room, alerting the operator to take appropriate corrective action. During loss of off-site power, when no LOCA signal exists, fan coil units restart automatically when power is available from the diesel generators. During loss of off-site power, chilled water from the HNCW system will not be available. Chilling will only be available from the reactor cooling water coils. The fan coil units are not operated during a LOCA.

4.2.3.6 Suppression Pool Temperature Monitoring System

The suppression pool temperature monitoring system is provided to monitor suppression pool temperature. Monitoring of suppression pool temperature is provided so that trends in suppression pool temperature can be established in sufficient time for proper cooling of the suppression pool water or for appropriate operator actions for suppression pool temperature control based upon symptom-based emergency operating procedures.

4.2.3.7 Suppression Pool Cleanup System

Suppression pool cleanup system provides a continuous purification of the suppression pool

water. The system removes impurities by filtration, adsorption, and ion exchange processes. The system consists of a recirculation loop with a pump and isolation valves. Suppression pool water passes through the fuel pool cooling and cleanup system filter/demineralizers for treatment. Treated water may be diverted to refill the reactor well and the upper pool during refueling outage or provide makeup water to the fuel pool and reactor cooling water surge tanks following a seismic event.

4.2.3.8 Secondary Containment

The secondary containment facilities consist of the reactor area of the reactor building and the standby gas treatment system. The reactor area of the reactor building completely surrounds the reactor containment vessel (primary containment). The interior is kept at sub-atmospheric pressure so that, even should there be leakage of radioactive substances from the reactor containment vessel, the direct discharge of radioactive material to the environs of the power station is prevented.

4.2.3.9 Standby Gas Treatment System

The standby gas treatment system minimizes exfiltration of contaminated air from the secondary containment to the environment following an accident or abnormal condition which could result in abnormally high airborne radiation in the reactor building. Because the fuel storage area is also in the secondary containment, it also can be exhausted to the standby gas treatment system.

This system consists of two demisters and fans and a single filtering device that includes a high-performance particle filter and a charcoal filter for iodine. In the event of an accident involving discharge of radioactive substances, the normal ventilation system will shutdown, and the radioactive substances which leaked in from the reactor containment vessel or elsewhere are passed through the filtering device and discharged through the 73 m high plant stack.

All safety-related components of the standby gas treatment system are operable during loss of off-site power.

4.2.4 Power Conversion Systems

4.2.4.1 Turbine Main Steam System

The main steam system (Figure 4.2.1-1 for a heat balance at rated power) delivers steam from the reactor to the turbine generator, the reheaters, the turbine bypass system, and the steam jet air ejectors from warmup to full-load operation. The main steam system also supplies the steam seal system and the auxiliary steam system when other sources are not available.

4.2.4.2 Main Turbine

The main turbine is a 1800 rpm, tandem compound six flow, reheat steam turbine with 52 inch last stage blades. The turbine generator is equipped with an electro-hydraulic control system and supervisory instruments to monitor performance. The gross electrical output of the turbine generator is approximately 1350 megawatts.

Turbine Control System: The turbine generator uses a digital monitoring and control system which, in coordination with the turbine steam bypass and pressure control system, controls the turbine speed, load, and flow for startup and normal operations. The control system operates the turbine stop valves, control valves, and combined intermediate valves. Turbine-generator supervisory instrumentation is provided for operational analysis and malfunction diagnosis. Automatic control functions are programmed to protect the turbine-generator through appropriate corrective actions.

Moisture Separator Reheater: Four horizontal cylindrical-shell, combined moisture separator/reheaters are installed in the steam path between the high and low pressure turbines. The moisture separator/reheaters serve to dry and reheat the high pressure

turbine steam exhaust before it enters the low pressure turbines. This improves cycle efficiency and reduces moisture-related erosion and corrosion in the low pressure turbines. Moisture is removed in chevron-type moisture separators, and is drained to the moisture separator drain tank and from there to the heater drain tank. The dry steam passes upward across the heater which is supplied with main steam. Finally, the reheated steam is routed to the combined intermediate valves which are located upstream of the low pressure turbines inlet nozzles.

Turbine Gland Steam System: The turbine gland steam system provides steam to the turbine glands and the turbine valve stems. The turbine gland steam system prevents leakage of air into or radioactive steam out of the turbine shaft and turbine valves. The gland steam condenser collects air and steam mixture, condenses the steam, and discharges the air leakage to the atmosphere via the main vent by a motor-driven blower.

Turbine Lubricating Oil System: The turbine lubricating oil system supplies oil to turbine-generator bearing lubrication lines and mainly consists of a lube oil tank, oil pumps, oil coolers, and oil purifier equipment.

Turbine Auxiliary Steam System: The turbine auxiliary steam system supplies steam to the steam jet air ejectors for condenser deaeration and to the turbine gland seal system which prevents radioactive steam leakage out of the turbine casing and atmospheric air leakage into the casing during turbine operation.

The house boiler steam is a backup to the reactor generated steam during operation and would be used only when reactor steam is unavailable or too radioactive.

Extraction System: Extraction steam from the high pressure turbine supplies the last stage of feedwater heating and extraction steam from the low pressure turbines supplies the first four stages. An additional low pressure extraction drained directly to the condenser protects the last stage buckets from erosion induced by water droplets.

4.2.4.3 Steam Bypass and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the flow permitted to pass to the turbine generator. The turbine bypass system has the capability to shed 40 percent of the turbine-generator rated load without reactor trip or operation of safety relief valves. The pressure regulation system provides main turbine control valve and bypass valve flow demands so as to maintain a nearly constant reactor pressure during normal operation. It also provides demands to the recirculation system to adjust power level by changing recirculation flow rate.

4.2.4.4 Generator

The generator is a direct-driven, three phase, 60 Hz, 27000 V, 1800 rpm, conduction cooled, synchronous generator rated at approximately 1600 MVA, at a 0.90 power factor, 75 psig hydrogen pressure and 0.60 short circuit ratio.

Generator Cooling System: The generator cooling system includes the hydrogen cooled rotor portion of the hydrogen gas cooling system and the water cooled stator portion of the turbine building cooling water system.

Generator Sealing Oil System: The generator sealing oil system prevents hydrogen gas from leaking from the generator. The sealing oil is vacuum-treated to maintain the gas purity.

Exciter: Main turbine-generator excitation power is provided by output of the acalternator-exciter. This output is rectified by the stationary silicon-diode rectifiers. The direct current output of the rectifier banks then is applied to the main generator field through generator collectors.

4.2.4.5 Heater, Drain and Vent System

The heater, drain and vent system permits efficient and dependable operation of the heat cycle balance of plant equipment and, particularly, the condensate and feedwater regenerative heaters. All process equipment drains and vents are collected and routed to the appropriate points in the cycle and flows are controlled for equipment protection.

4.2.4.6 Main Condenser

The main condenser is a multipressure three-shell deaerating type condenser. During plant operation, steam expanding through the low pressure turbines is directed downward into the main condenser and is condensed. The main condenser also serves as a heat sink for the turbine bypass system, emergency and high level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

4.2.4.7 Condensate and Feedwater System

The condensate and feedwater system provides a dependable supply of high-quality feedwater to the reactor at the required flow, pressure, and temperature. The condensate pumps take the deaerated condensate from the main condenser hotwell and deliver it through the steam jet air ejector condenser, the gland steam condenser, the off-gas condenser, the condensate demineralizer, and through three parallel strings of four low pressure feedwater heaters to the reactor feed pumps' suction. The two reactor feed pumps each have an approximate capacity of 4,600 m³/h. They each discharge through two stages of high pressure heaters (two parallel strings) to the reactor. Each reactor feedwater pump is driven by an adjustable speed synchronous motor. The drains from the high pressure heaters are pumped backward to the suction of the feed pumps.

The feedwater control system automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water within the vessel at predetermined levels. A

fault-tolerant triplicated, digital controller using conventional three-element control scheme, which provides control signals to adjustable speed drives for the feedwater pump motors, is used to accomplish this function.

4.2.4.8 Condensate Purification System

The plant is served by a 100 percent capacity condensate cleanup system, consisting of three hollow fiber filters followed by six deep-bed demineralizer vessels designed for parallel operation. One demineralizer vessel is a spare. The condensate cleanup system with instrumentation and automatic controls is designed to ensure a constant supply of high-quality water to the reactor.

The condensate filter facility continuously removes suspended solids by processing the full flow condensate through three, one-third capacity, hollow fiber filters. A fast acting full flow bypass valve opens on high pressure differential across the filter to protect against sudden loss of condensate flow.

The condensate demineralizers continuously remove dissolved solids to maintain reactor feedwater quality through five, one-fifth capacity, demineralizers and a sixth one-fifth capacity unit in manual standby. An emergency bypass line protects the equipment and a demineralizer resin handling and cleanup system is included.

4.2.5 Heat Rejection Systems

4.2.5.1 Service Water Systems

Reactor Service Water System: The reactor service water system provides cooling water during various operating modes, during shutdown and post-LOCA operations. The system removes heat from the reactor service water system and transfers it to the ultimate heat sink.

Turbine Service Water System: The turbine service water system supplies cooling water to the turbine cooling water system heat exchangers to transfer heat from the turbine cooling water system to the power cycle heat sink.

HVAC Cooling Water Systems: The HVAC normal cooling water system provides chilled water to the air supply cooling coils of the reactor building, to the heating/cooling coils in the drywell, and the control building electrical equipment room.

The HVAC emergency cooling water system provides chilled water to the cooling coils in the control building essential electrical equipment room, the main control room and the diesel generator electrical equipment areas. The safety-related chilled-water system is designed to meet the requirements of Criterion 19 of 10CFR50.

Ultimate Heat Sink: The ultimate heat sink is a spray pond which serves the safety-related functions of providing cooling water and acting as a heat sink for the reactor service water system during accident conditions. The spray pond also serves as a heat sink during normal operation by accepting the heat load of the reactor services water system.

4.2.5.2 Reactor Building Cooling Water System

The reactor building cooling water system provides cooling water to certain designated equipment located in the reactor building including the diesel generators, the residual heat removal and FPC heat exchangers, and the high pressure core flooders and residual heat removal pump motors. Adequate capacity and redundancy is provided in heat exchangers and pumps to ensure performance of the cooling system. Under loss of off-site power, emergency power for the system is available from the on-site emergency diesel generators. The closed loop design provides a barrier between radioactive systems and the service water discharged to the environment. Heat is removed from the closed loop by the reactor service water system. Radiation monitors are provided to detect contaminated leakage into the closed systems.

4.2.5.3 Circulating Water System

The circulating water system provides a continuous supply of cooling water to the main condenser to remove the heat rejected by the steam cycle. The circulating water system consists of the following components: screen house and intake screens, condenser water boxes and water box fill and drain subsystem, and related support facilities such as for water treatment and general maintenance.

The circulating water system is designed to deliver water to the main condenser within a temperature range of 40 to 100 degrees Fahrenheit. The 40 degree Fahrenheit minimum temperature is maintained, when needed, by warm water recirculation. Heat is removed from the circulating water by means of a cooling tower.

The cooling water is circulated by three fixed-speed motor-driven pumps. The pumps are arranged in parallel and discharged into a common header. The discharge of each pump is fitted with a butterfly valve. This arrangement permits isolation and maintenance of any one pump while the others remain in operation.

The circulating water system and condenser are designed to permit isolation of each set of the three series-connected tube-bundles to permit repair of leaks and cleaning of water boxes while operating at reduced power.

4.2.6 Station Electrical Systems

The on-site and off-site AC power distribution systems are shown in Figure 4.2.6-1.

4.2.6.1 Off-Site Electric Power System

The scope of the off-site electrical power system includes the entire system from the termination of the transmission lines coming into the switchyard to the termination of the bus duct at the terminals of the main generator and at the input terminals of the circuit

breakers for the medium voltage switchgear.

The main power transformer is a bank of three single phase transformers. One single phase installed spare transformer is provided.

A generator breaker capable of interrupting the maximum available fault current is provided. This allows the generator to be taken off line and the main grid to be utilized as a power source for the unit auxiliary transformers and their loads, both Class 1E and non-Class 1E. This is also the startup power source for the unit.

There are three unit auxiliary transformers, connected to supply power to three approximately equal load groups of equipment. The "Normal Preferred" power feed is from the unit auxiliary transformers so that there normally are no bus transfers required when the unit is tripped off the line.

One unit reserve auxiliary transformer is supplied to provide power via one winding for the emergency buses as an alternate to the "Normal Preferred" power. The other secondary winding supplies reserve power to the non-safety-related buses in the turbine building.

4.2.6.2 On-site Electric Power System

Three turbine building non-safety-related buses per load group and one reactor building safety-related bus per division receive power from the single unit auxiliary transformer assigned to each load group. Load groups A, B, and C line up with Divisions I, II, and III, respectively. One winding of the reserve auxiliary transformer may be utilized to supply reserve power to each of the non-safety-related buses either directly or indirectly through bus tie breakers. The three safety-related buses may be supplied power from the other winding of the reserve auxiliary transformer.

A combustion turbine generator supplies standby power to permanent non-safety-related loads in the turbine building. These loads are grouped on one of the 6.9 KV buses per load

group. A power supply bus is also provided from the combustion turbine to the three Class 1E medium voltage buses in the reactor building via breakers that are normally racked out for Divisions I and II and remote manually closed under administrative control for Division II.

In general, motors larger than 300 kilowatts are supplied from the 6.9 kilovolts buses. Motors 300 kilowatts or smaller but larger than 100 kilowatts are supplied power from 480V power center switchgear. Motors 100 kilowatts or smaller are supplied power from 480V motor control centers.

During normal plant operation all of the non-Class 1E buses and two of the Class 1E buses are supplied with power from the turbine generator through the unit auxiliary transformers. The third Class 1E bus is supplied from the reserve transformer. This third division is immediately available, without a bus transfer, if the normal preferred power is lost to the other two divisions.

Three diesel generator standby ac power supplies provide a separate on-site source of power for each Class 1E load group when normal or alternate preferred power supplies are not available. The transfer from the normal preferred or alternate preferred power supplies to the diesel generator is automatic. The transfer back to the normal preferred or the alternate preferred power source is a manual transfer.

The Division I, II, and III standby ac power supplies consist of an independent 6.9 kilovolts Class 1E diesel generator, one for each division. Each diesel generator may be connected to its respective 6.9 kilovolts Class 1E switchgear bus through a main circuit breaker located in the switchgear.

The standby AC power system is capable of providing the required power to safely shutdown the reactor after loss of preferred power and/or loss of coolant accident or to maintain the safe shutdown condition and operate the Class 1E auxiliaries necessary for plant safety during and after shutdown with any one of the three power load groups.

The plant 480V AC auxiliary power system distributes sufficient power for normal auxiliary and Class 1E 480 kilovolt plant loads. All Class 1E elements of the auxiliary power distribution system are supplied via the 6.9 kilovolt Class 1E switchgear and, therefore, are capable of being fed by the normal preferred, alternate preferred, standby or combustion turbine generator power supplies.

The 120V AC non-Class 1E instrumentation power system provides power for non-Class 1E control and instrumentation loads.

The Class 1E 120V AC instrument power system provides for Class 1E plant controls and instrumentation. The system is separated into Divisions I, II, and III with distribution panels fed from their respective divisional sources.

The 125V direct current power distribution system provides four independent and redundant on-site battery sources of power for operation of Class 1E direct current loads. The 125V direct current non-Class 1E power is supplied from three 125V direct current batteries located in the turbine building. A separate non-Class 1E 250V battery is provided to supply uninterruptible power to the plant computers and non-Class 1E direct current motors.

The safety system and logic control for reactor protection system and MSIV isolation logic derives its power from four uninterruptible 120V AC buses. The safety system and logic control for the emergency core cooling system derives its power from the four divisions of 125V direct current buses. The four buses provide the redundancy for various instrumentation, logic and trip circuits and solenoid valves.

4.2.6.3 Raceway System

The raceway system is a plant wide network comprised of metallic cable trays, metallic conduits and supports. Raceways are classified for carrying medium voltage power cables, low voltage power cables, control cables and low level signal/instrumentation cables. Divisional cables are routed in separate cable raceways for each division. Fiber optic

dataways are not restricted to raceway classifications, but would generally be run with control cables due to their common destinations.

4.2.7 Station Auxiliary Systems

4.2.7.1 Heating, Ventilating and Air Conditioning

The plant environmental control systems control temperature, pressure, humidity, and airborne contamination to ensure the integrity of the plant equipment, provide acceptable working conditions for plant personnel, and limit off-site releases of airborne contaminants.

The following environmental systems are provided: 1) the control room air conditioning system consisting of supply, recirculation/exhaust and makeup air cleanup units to ensure the habitability of the control room under normal and abnormal conditions of plant operation; 2) the reactor building secondary containment HVAC system maintains a negative pressure in the secondary containment under normal and abnormal operating conditions thereby isolating the environs from potential leak sources and it removes heat generated during normal plant operation, shutdown and refueling periods; 3) the drywell cooling system to remove heat from the drywell generated during normal plant operations including startup, reactor scrams, hot standby, shutdown, and refueling periods; 4) the power block pressure control supply and exhaust system to distribute air so that a negative pressure is maintained in the emergency core cooling equipment rooms, thereby isolating the potential airborne contamination in these rooms; 5) the electrical equipment supply and exhaust system to pressurize the electrical rooms allowing exfiltration of air to the battery rooms for exhaust to the outside atmosphere; 6) the power block exhaust system to maintain the refueling floor at a negative pressure with respect to the outside atmosphere to prevent the potential release of airborne contamination; 7) the diesel generator area air exhaust system to provide cooling during operation of the diesel generators (a tempered air supply system controls the thermal environment when the diesel generators are not operating); and 8) coolers in the steam tunnel and ECCS rooms to remove heat generated during operation of the equipment in these rooms.

The hot water heating system is a closed loop hot water supply to the various heating coils of the HVAC systems. The system includes two heat exchangers, a backup heat exchanger, surge and chemical addition tanks and associated equipment, controls and instrumentation.

4.2.7.2 Pneumatic Systems

Station Service Air System: The station service air system provides a continuous supply of compressed air of suitable quality and pressure for general plant use. The service air compressor discharges into the air receivers and the air is then distributed throughout the plant.

Instrument Air System: The instrument air system provides dry, oil-free, compressed air for valve actuators and for non-safety-related instrument control functions and for general instrumentation and valve services outside containment. All instrumentation and control systems located inside the containment are supplied with nitrogen gas during normal plant operation.

High Pressure Nitrogen Gas Supply System: Nitrogen gas is normally supplied by the Atmospheric Control System to meet the requirement on the main steam safety relief valve automatic depressurization and relief function accumulators, the main steam isolation valves, and instruments and pneumatic valves using nitrogen in the reactor building. When this supply of pressurized nitrogen is not available, the high pressure nitrogen gas supply system (HPIN) automatically maintains nitrogen pressure to this equipment. The HPIN system consists of high pressure nitrogen storage bottles with piping, valves, instruments, controls, and control panel.

Breathing Air System: The breathing air system includes air compressors, dryers, purifiers and a distribution network. This network makes breathing air available in all plant areas where operations or maintenance must be performed and high radioactivity could occur in the ambient air. Special connections are provided to assure that this air is only used for breathing apparatus.

4.2.7.3 Turbine Building Cooling Water System

The turbine building cooling water system is a single loop system that consists of one surge tank, one chemical addition tank, three pumps, three heat exchangers, and associated coolers, piping, valves, controls, and instrumentation. Heat is removed from the TCW system and transferred to the non-safety-related turbine service water system.

The turbine building cooling water system cools non-safety related loads including the HVAC normal cooling water chillers, the feedwater, condensate, and heater drain pump motor coolers, and various turbine auxiliaries (generator stator coolers, hydrogen coolers, seal oil coolers, and exciter coolers).

4.2.7.4 Fire Protection System

The fire protection system is designed to provide an adequate supply of water or chemicals to points throughout the plant where fire protection is required. Diversified fire-alarm and fire-suppression types are selected to suit the particular areas or hazards being protected. Chemical fire fighting systems are also provided as additions to or in lieu of the water fire fighting systems. Appropriate instrumentation and controls are provided for the operation of the fire detection, annunciation and fire fighting systems.

4.2.7.5 Makeup Water System

The makeup water system (purified) provides purified makeup water to the reactor coolant system and plant auxiliary systems.

The makeup water system (condensate) maintains the required water supply capacity for the reactor core isolation and high pressure core flooders systems and maintains the required level in the condenser hotwell. The system also stores and transfers water during fuel shipping cask loading, receives and stores the process effluent from the liquid radwaste system, and provides condensate to the control rod drive hydraulic system.

The makeup water system (condensate) consists of a condensate storage tank, three condensate transfer pumps, and the necessary controls and instrumentation. The condensate storage tank receives demineralized water from the condensate water makeup system and may also receive low conductivity water from the condensate return of the primary loop, from the radwaste disposal system and the condensate system in the turbine building.

4.2.7.6 Water Chemistry Systems

Sampling System: The process sampling system is furnished to provide process information that is required to monitor plant and equipment performance and changes to operating parameters. Representative liquid and gas samples are taken automatically and/or manually during plant operation for laboratory or on-line analyses.

Hydrogen Water Chemistry System: The hydrogen water chemistry system is used to mitigate the potential for intergranular stress corrosion cracking of sensitized austenitic stainless steel. The dissolved oxygen in the reactor water can be reduced to less than 20 ppb by the addition of hydrogen to the feedwater. The hydrogen water chemistry system is composed of hydrogen and oxygen systems. Hydrogen is injected into the feedwater and oxygen into the offgas system upstream of the recombiner.

4.2.7.7 House Boiler

The house boiler system consists of the house boilers, reboilers, feedwater components, boiler water treatment and control devices. The house boiler system supplies turbine gland steam and heating steam including the concentrating tanks and devices in the high conductivity waste equipment.

4.2.8 Protection, Control and Instrumentation Systems

In addition to the systems described below, the following previously described systems also

must be included in this category: the rod control and information system (4.2.1.4); the recirculation flow control system (4.2.1.4); the leak detection and isolation system (4.2.3.2); the suppression pool temperature monitoring system (4.2.3.6); the turbine control system (4.2.4.2); the steam bypass and pressure control system (4.2.4.3); and the feedwater control system (4.2.4.7).

4.2.8.1 Reactor Protection System

The reactor protection system is an overall complex of instrument channels, trip logics, trip actuators and scram logic circuitry that initiate rapid insertion of control rods (scram) to shut down the reactor. The reactor protection system also establishes reactor operating modes and provides status and control signals to other systems and annunciators. To accomplish its overall function, the reactor protection system interfaces with the essential multiplexing system, neutron monitoring system, process radiation monitoring system, control rod drive system, rod control and information system, reactor recirculation system, process computer, nuclear boiler system and other plant systems and equipment.

It acts in time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operational transients. The reactor protection system overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action even if a single failure occurs.

4.2.8.2 Neutron Monitoring System

The neutron monitoring system is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The startup range neutron monitors include fixed in-core sensors which provide indications during reactor startup and low power operation. Average power range monitors allow assessment of local and overall flux conditions during power range operation. The automatic traversing in-core probe system provides a means to calibrate the power range

monitors. The neutron monitoring system provides inputs to the rod control and information system to initiate rod blocks if preset flux limits or period limits for rod block are exceeded as well as inputs to the reactor protection system if other limits for scram are exceeded.

Those portions of the neutron monitoring system that input signals to the RPS qualify as a nuclear safety system. The startup range neutron monitors and the average power range monitors which monitor neutron flux via incore detectors provide scram logic inputs to the reactor protection system to initiate a scram in time to prevent excessive fuel clad damage as a result of over-power transients. The average power range monitors system also generates a simulated thermal power signal. Both upscale neutron flux and upscale simulated thermal power are conditions which provide scram logic signals.

4.2.8.3 Radiation Monitoring Systems

Process Radiation Monitoring System: Process radiation monitoring systems monitor radioactivity in process and effluent streams and activate appropriate alarms and controls.

The process radiation monitoring system indicates and records radiation levels associated with selected plant process streams and effluent paths leading to the environment. All effluents from the plant which are potentially radioactive are monitored.

Area Radiation Monitoring System: Area radiation monitoring systems alert occupants and the control room personnel of excessive gamma radiation levels at selected locations within the plant.

Containment Atmospheric Monitoring: The containment atmospheric monitoring system measures, alarms and records radiation levels and the oxygen concentration in the primary containment under post accident conditions. It is automatically put in service upon detection of LOCA conditions.

4.2.8.4 Process Computer

On-line process computers are provided to monitor and log process variables and make certain analytical computations.

4.2.8.5 Power Generation Control System

The power generation control system is a top level controller that monitors the overall plant conditions, issues control commands and adjusts the setpoints of lower level controllers to support automation of the normal plant startup, shutdown, and power range operations. It contains the algorithms for the automated control sequences associated with plant startup, shutdown, and normal power range operations. It issues reactor command signals to the automatic power regulator system. The reactor power change algorithms are implemented in the automatic power regulator system. The plant operator interfaces with the power generation control system through a series of breakpoint controls to initiate automated sequences from the operator control console (4.2.10.1). For selected operations that are not automated, the system prompts the operator to perform such operations. In the semi-automatic mode, the power generation control system provides guidance messages to the operator to carry out operations.

4.2.8.6 Automatic Power Regulator System

The primary objective of the automatic power regulator system is to control reactor power during reactor startup, power generation, and reactor shutdown, by appropriate commands to change rod positions, or to change reactor recirculation flow. The secondary objective of the automatic power regulator system is to control the turbine bypass and pressure control system during reactor heatup and depressurization (e.g., to control the reactor cooldown rate). The automatic power regulator system consists of redundant process controllers. Automatic power regulation is achieved by appropriate control algorithms for different phases of operation including approach to criticality, heatup, reactor power increase, automatic load following, reactor power decrease, and reactor depressurization and

cooldown. The automatic power regulation system receives input from the plant process computer (4.2.8.4), the power generation control system (4.2.8.5), the steam bypass and pressure control system (4.2.4.3), and the operator's control console (4.2.10.1). The output demand signals from the automatic power regulator system are to the rod control and information system (4.2.1.4) to position control rods, to the recirculation flow control system (4.2.1.4) to change reactor coolant recirculation flow, and to the steam bypass and pressure control system (4.2.4.3) for automatic load following operations. The normal mode of operation of the APR system is automatic. If any system or component conditions are abnormal, the power generation control system will be automatically switched into manual, any operations in progress will be stopped, and alarms will be activated to alert the operator. A failure of the automatic power regulator system will not prevent manual controls of reactor power, nor will it prevent safe shutdown of the reactor.

4.2.8.7 Multiplexing System

The multiplexing system provides redundant and distributed control and instrumentation data communications networks to support the monitoring and control of interfacing plant systems. The system provides all electrical devices and circuitry (such as multiplexing units, bus controllers, formatters and data busses) between sensors, display devices, controllers and actuators which are defined and provided by other plant systems. The multiplexing system also includes the associated data acquisition and communication software required to support its function of plant-wide data and control distribution.

4.2.8.8 Remote Shutdown System

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by use of controls and equipment that are available outside the control room.

Manual transfer devices are provided which override controls from the main control room and transfer controls to remote shutdown control. Control and process sensor signals are

interrupted by the transfer devices. All necessary power supply circuits are also transferred to other sources. Operation of the transfer devices causes an alarm in the main control room; access to the remote shutdown control panels (outside the main control room) is administratively and procedurally controlled. The system functions ahead of the plant multiplexing system: All controls and indications are hard-wired and will function regardless of the status of the multiplexing system. Instrumentation and controls on the remote shutdown panels include the following:

1. Controls and indications for operation of one HPCF loop to control reactor water level.
2. Controls and indications for operation of two residual heat removal loops to support shutdown cooling once reactor pressure has been reduced, and suppression pool cooling to control suppression pool temperature which may rise due to SRV operation.
3. Controls for operation of three safety/relief valves for maintaining and reducing reactor pressure.
4. Indications of reactor pressure vessel water level and pressure, and suppression pool temperature and level.
5. Controls and indications for operation of the RCW and RSW systems.
6. Controls and indications for electrical power distribution.
7. Controls for manually running and stopping two of the emergency diesel generators.

4.2.9 Radioactive Waste Systems

4.2.9.1 Liquid Radwaste Management System

The liquid radwaste management system collects, monitors, and treats liquid radioactive

wastes for return to the primary system whenever practicable. The ABWR is designed to operate as a closed system with no routine discharges of liquid effluents from the reactor coolant and associated systems. The radwaste processing equipment is located in the radwaste building. Processed waste volumes discharged to the environs are expected to be small. Any discharge is such that concentrations and quantities of radioactive material and other contaminants are in accord with applicable local, state, and federal regulations.

All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the plant. These wastes are transferred to collection tanks in the radwaste facility.

Waste processing is done on a batch basis. Each batch is sampled as necessary in the collection tanks to determine concentrations of radioactivity and other contamination. Equipment drains and other low-conductivity wastes are treated by filtration and demineralization and are transferred to the condensate storage tank for reuse. Laundry drain wastes and other detergent wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway and may be released from the plant on a batch basis. Protection against inadvertent release of liquid radioactive waste is provided by design redundancy, instrumentation for the detection and alarm of abnormal conditions, automatic isolation, and administrative controls.

Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum radiation exposure to personnel.

4.2.9.2 Solid Radwaste Management System

The solid radwaste management system provides for the safe handling, packaging, and short-term storage of radioactive solid and concentrated liquid wastes that are produced. Wet waste produced by this system is transferred to the solidification system where it is solidified and loaded into containers. Dry active waste is surveyed and disposed of whenever possible via the provisions of 10CFR20.302(a). The remaining combustible waste

is incinerated. Incinerator ash is loaded into containers. Non-combustible dry waste is compacted and loaded into containers. Waste is shipped in containers for off-site disposal.

4.2.9.3 Off-Gas System

The main condenser evacuation system removes the noncondensable gases from the main condenser and discharges them to the gaseous radwaste (off-gas) system. The evacuation system consists of two 100 percent capacity, multiple-element, multi-stage steam jet air ejectors with intercondensers, for normal station operation, and mechanical vacuum pumps for use during startup.

The off-gas system receives air and noncondensable gases from the steam jet air ejectors and processes the effluent for the decay and/or removal of gaseous and particulate radioactive isotopes. The system processes the off-gas and releases it to the environs through the plant stack, keeping the dose to the environment as low as practicable. The off-gas system also reduces the possibility of an explosion from the buildup of radiolytic hydrogen. This is accomplished by the recombination of the radiolytic hydrogen and oxygen under controlled conditions within a recombiner.

This strips the condensibles and reduces the volume. The remaining non-condensibles (principally air with traces of krypton and xenon) are passed through activated charcoal beds which provide a holdup volume to allow time for the krypton and xenon to decay. After this processing, the remaining gas is discharged through the plant stack.

The plant stack is located on the reactor building and rises to an elevation of 76 meters above grade level. The stack is a steel shell construction supported by an external steel tubular frame work. The stack vents the reactor building, turbine building, radwaste building, control and service buildings as well as discharge from the standby gas treatment and off-gas systems.

4.2.10 Main Control Room

The ABWR main control room, including the main control console, is shown in Figure 4.2.10-1. In the main control room arrangement, the main control console is located directly in front of the large display panel for optimum viewing efficiency by the plant operators seated at the main console. The shift supervisor's console is also placed in front of the large display panel, but at a somewhat greater distance than the main control console. The shift supervisor is, thus, in position behind the control console operators. This arrangement allows all control room personnel to view the contents of the large panel displays.

4.2.10.1 Main Control Console

The main control console comprises the work stations for the two control room plant operators. It is configured such that each operator is provided with controls and monitoring information necessary to perform their assigned tasks and allows the operators to view all of the displays on the large display panel from a seated position.

The main control console, in concert with the large display panel, provides the controls and displays required to operate the plant during normal plant operations, abnormal events and emergencies. These main control console controls and displays include the following: 1) On-screen control video display units for safety system monitoring and non-safety system control and monitoring which are driven by the plant process computer system. 2) A separate set of on-screen video display units for safety system control and monitoring and separate on-screen control video display units for non-safety system control and monitoring. The operation of these two sets of video display units is entirely independent of the process computer system. The first set of video display units and all equipment associated with their functions of safety system control and monitoring are divisionally separate and qualified to Class 1E standards. 3) Dedicated function switches used to support such functions as initiation of automated sequences of safety and non-safety system operations, manual scram, and reactor operating mode changes.

The main control console is also equipped with a limited set of dedicated displays for

Figure 4.2.10-1 ABWR Main Control Room

selected functions (e.g., the standby liquid control system and the synchronization of the main generator to the electrical grid).

In addition to the above equipment, the main control console is equipped with both intra-plant and external communications equipment and a laydown space is provided for hard copies of procedures and other documents required by the operators during performance of their duties.

4.2.10.2 Large Display Panel

The large display panel provides information on overall plant status with real-time data during all phases of plant operation. The information on the large display panel can be viewed from the main control console and the supervisor's console. The large display panel includes a fixed-position display, a variable display, and spatially dedicated alarm windows.

Fixed-Position Display: The fixed-position display provides key plant information for viewing by the entire control room staff. Those portions of the large display panel which present safety-related information are qualified to Class 1E standards. The information presented in the fixed-position display includes the critical plant parameter information, as defined by the Safety Parameter Display System requirements of NUREG-0737, Supplement 1, and the Type A post-accident monitoring instrumentation required by Regulatory Guide 1.97, Revision 3.

Large Variable Display: The large variable display is a video display unit which is driven by the plant process computer system. Any screen format resident in the process computer system can be shown on this large variable display.

Fixed-Position Alarms: Alarm tiles on the large display panel annunciate the key, plant-level alarm conditions that potentially affect plant availability or plant safety, or indicate the need of immediate operator action. (Alarms which are not indicated by fixed-position alarm

tiles on the large display panel, i.e., those alarms of nominally lower level importance such as those related to specific equipment status, are displayed to the control room operating staff via the main control console video display units.)

4.2.10.3 Supervisor's Console

A console is provided which is equipped with video display units on which any screen format resident in the process computer system and available to the operators at the main control console is also available to the shift supervisor.

4.2.11 Structures

4.2.11.1 Foundation Work

The analytical design and evaluation methods for the containment and reactor building walls, slabs and foundation mat and foundation soil are based on linear elastic finite-element analyses.

4.2.11.2 Reactor Building

The reactor building includes the containment, drywell and major portions of the nuclear steam supply system, steam tunnel, refueling area, diesel generators, essential power, non-essential power, emergency core cooling systems, HVAC and supporting systems.

4.2.11.3 Turbine Building

The turbine building houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.

4.2.11.4 Control Building

The control building includes the control room, the computer facility, the cable tunnels, some of the plant essential switchgear, some of the essential power, reactor building water system and the essential HVAC system.

4.2.11.5 Radwaste Building

The radwaste building houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

4.2.12 Fuel Storage and Handling

4.2.12.1 New Fuel Vault

The new fuel handling facilities are shown schematically in Figure 4.2.12-1. New fuel is stored in new-fuel storage racks in the new fuel storage vault located on the reactor refueling floor near the spent fuel storage pool work area to facilitate handling during fuel preparation. The new-fuel inspection stand is located close to the new-fuel storage vault to minimize fuel transport distance. The new fuel storage vault stores a 40 percent core load of new fuel assemblies.

4.2.12.2 Spent Fuel Storage Pool

Spent fuel removed from the reactor vessel is stored underwater while awaiting disposition. Spent fuel storage racks, which are used for this purpose, are located at the bottom of the fuel storage pool under sufficient water to provide radiological shielding. The pool water is processed through the fuel pool cooling and cleanup system to provide cooling to the spent fuel in storage and for maintenance of fuel pool water quality. The spent fuel capacity for a standard ABWR is equivalent to approximately 2.7 full core loadings. The standard ABWR refueling facilities are shown schematically in Figure 4.2.12-2. The additional fuel storage and handling facilities necessary for plutonium disposition are described in Section 5.4.

Figure 4.2.12-1 Simplified New Fuel Handling Facility Diagram

Figure 4.2.12-2 Simplified Refueling Facility Diagram

The spent fuel pool is a reinforced concrete structure with a stainless steel liner. The bottoms of all pool gates are sufficiently high to maintain the water level over the spent fuel storage racks and provide adequate shielding and cooling. All pool fill and drain lines enter the pool above the safe shielding water level. Redundant anti-siphon vacuum breakers are located at the high point of the pool circulation lines to preclude a pipe break from siphoning the water from the pool and jeopardizing the safe water level.

4.2.12.3 Fuel Storage Racks

The new and spent fuel storage racks are the same high density design. The new fuel racks can be used for either dry or submerged storage of fuel. The fuel storage racks are top entry racks providing storage for fuel assemblies (with channels) and fuel bundles (without channels). The upper tieplate of the fuel elements rests against the rack to provide lateral support. The lower tieplate sits in the bottom of the rack, which supports the weight of the fuel. The rack modules are arranged side-by-side with minimal clearance. The racks are designed to protect the fuel assemblies from damage caused by impacting from other fuel assemblies or other equipment, and to withstand all credible static and seismic loadings. They are designed to preclude the possibility of criticality. A full array in the loaded spent fuel rack is designed to be subcritical, by at least 5 percent delta-k. Neutron-absorbing material, as an integral part of the design, is employed to assure that the calculated k-eff, including biases and uncertainties, will not exceed 0.95 under all normal and abnormal conditions.

4.2.12.4 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system maintains acceptable levels of temperature and clarity and minimizes radioactivity levels in the water in the spent fuel pool, reactor well and dryer/separator pit on top of the containment. The system includes two heat exchangers, each capable of removing the decay heat generated from an average discharge of spent fuel, and two filter/demineralizers, each unit having the capacity to process the system flow or

greater to maintain the desired purity level. The heat exchangers and filter/demineralizers are located in the reactor building.

4.2.12.5 Fuel Servicing Equipment

The fuel servicing equipment includes the fuel preparation machines, the new fuel inspection stand, the channel bolt wrench, the channel handling tool, the fuel pool sipper, the general purpose grapple and the channel handling boom. The reactor building crane and the refueling platform are described separately below.

Two fuel preparation machines are mounted on the wall of the spent fuel pool and are used for stripping reusable channels from the spent fuel and for rechanneling of the new fuel. The machines are also used with the fuel inspection fixture to provide underwater inspection capability. Each fuel preparation machine consists of a work platform, a frame, and a movable carriage. The frame and moveable carriage are located below the normal water level in the spent fuel pool thus providing a water shield for the fuel assemblies being handled. The fuel preparation machine carriage has permanently installed up-travel-stop to prevent raising fuel above the safe water shield level.

The new fuel inspection stand serves as a support for the new fuel bundles undergoing inspection and provides a working platform for technicians engaged in performing the inspection. The new fuel inspection stand consists of a vertical guide column, a lift unit to position the work platform at any level, bearing seats and upper clamps to hold the fuel bundles in position.

The channel bolt wrench is a manually operated device approximately 3.76 meters in overall length that is used for removing and installing the channel fastener assembly while the fuel assembly is held in the fuel preparation machine. The channel bolt wrench has a socket which mates and captures the channel fastener capscrew.

The channel-handling tool is used in conjunction with the fuel preparation machine to

remove, install and transport fuel channels in the spent fuel pool. The tool is composed of a handling shaft, angle guides, and clamp arms which engage the fuel channel. The clamps are actuated (extend and retract) by manually rotating the lock/release knob.

The fuel pool vacuum sipper provides a means of identifying fuel suspected of having cladding failures. The fuel pool vacuum sipper consists of a fuel isolation container, fluid console, monitoring console with program controller and beta detector, and the inter-connecting tubing and cables. The suspected fuel assembly is placed in the isolation container. A partial vacuum is established in the gas volume above the fuel assembly. The fission product gas leakage is sensed by the beta detector and monitoring console.

The general purpose grapple is a handling tool used generally with the fuel. The grapple can be attached to the jib crane to handle fuel during channeling.

A channel handling boom with a spring-loaded balance reel is used to assist the operator in supporting a portion of the weight of the channel as it is removed from the fuel assembly. The boom is set between the fuel preparation machines. With the channel handling tool attached to the reel, the channel may be conveniently moved between the fuel preparation machines.

Refueling Interlocks: A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling and startup modes is provided to prevent an inadvertent criticality during refueling operation. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling platform, refueling platform hoists, fuel grapple, and control rod movement.

4.2.12.6 Reactor Building Crane

The reactor building crane provides heavy load lifting capability for the refueling floor. The reactor building crane consists of two crane girders and a trolley which carries two hoists. The runway track, which supports the crane girders, is supported from the reactor building

walls. The trolley travels laterally on the crane girders carrying the main and auxiliary hoists.

The reactor building crane is used to move all of the major components (reactor vessel head, shroud head and separator, dryer assembly and pool gates) as required by plant operations. The reactor building crane is used for handling new fuel from the reactor building entry hatch to the new fuel storage, the new fuel inspection stand and the spent fuel pool. It is also used for handling the spent fuel cask.

The NL1 spent fuel shipping cask, for example, weights about 100 tons loaded, is 204 inches long and has an outside diameter of about 88 inches. Tie-down lugs protrude about 4 inches radially beyond the body of the cask. The main hook of the reactor building crane has sufficient capacity to handle the weight. Based on the sizes of the equipment access hatch and the cask pit, there are no dimensional problems in handling a cask of that size.

4.2.12.7 Refueling Platform

The refueling platform is a gantry crane, which is used to transport fuel and reactor components to and from pool storage and the reactor vessel. The platform spans the spent fuel pool on bedded tracks in the refueling floor. A telescoping mast and grapple suspended from a trolley system is used to lift and orient fuel bundles for placement in the core or storage rack. Control of the platform is from an operator station on the refueling floor.

The refueling platform control computer provides memory of all the fuel and platform positions, directions for the travelable area and traveling paths, directions for the speed functions for all modes of travel and control of the fuel load. The computer controls automatic or manual refueling between storage and the reactor from the remote control room.

4.2.13 Maintenance Facilities

4.2.13.1 Control Rod Drive Maintenance Facility

The control rod drive maintenance facility is designed and equipped to accommodate maintenance of the fine motion control rod drive, provide decontamination of the fine motion control rod drive component, perform the acceptance tests and provide storage. The facility uses manual and/or remote operation to minimize the contamination of surrounding equipment during operation. The layout of the facility is designed so as to maximize the efficiency of the personnel thereby minimizing the number of workers.

4.2.13.2 Internal Pump Maintenance Facility

The reactor internal pump maintenance facility is located in the reactor building and is designed for performing maintenance work on the motor assembly and related parts. The facility is designed for one motor assembly including decontamination in assembled and disassembled states. The facility is equipped with all tools needed for inspection of motor parts and heat exchanger tube bundles. Reactor internal pump handling tools are stored outside the area.

4.3 ABWR PLANT PARAMETERS

This section provides the detailed parameters (size, temperature, pressure, flow, etc.) for the ABWR plant described above in Section 4.2.

4.3.1

Nuclear Boiler Systems

Parameter Description	Value
Thermal Power Level (MWt)	3926
Steam Flow (Kg/hr)	7.64E6
Core Flow (Kg/hr)	52.2E6
Feedwater Flow (Kg/hr)	7.64E6
Feedwater Temperature (C)	216
Vessel Dome Pressure (Kg/cm ²)	73.1
Turbine Inlet Pressure (Kg/cm ²)	69.25
Gross turbine Output (MWe)	1356

Reactor Pressure Vessel Parameters

Parameter Description	Value
Internal Diameter (m)	7.1
Wall Thickness (cm)	17
Overall Height (m)	21
Steam Flow Restrictor Nozzles	4 with max cap. of 3.82E6 Kg/hr
Feedwater Nozzles	6
Internal Pumps	10
Control Rod Drives	205
Rated Pressure (Kg/cm ²)	87.9
Rated Temperature (C)	302
Material	SS clad low alloy steel
Water Volume (m ³)	600

Fuel Assembly

Parameter Description	Standard ABWR	ABWR Modified for Plutonium Disposition
Type	8 x 8	8 x 8
Number of Assemblies	872	872
Fuel Rods Per Assembly	62	60
Water Rods Per Assembly	2	1
Fuel Rod Pitch (cm)	1.6	1.6
Fuel Rod Diameter (cm)	1.2	1.2
Cladding Wall Thickness (mm)	0.9	0.9
Effective Fuel Rod Length (m)	3.7	3.7

Recirculation System

Parameter Description	Value
Number of Internal Pumps	10
Pump Circulation Capacity (t/hr)	5800

Standby Liquid Control System Parameters

Parameter Description	Value
Number of Pumps	2
Pump Capacity (m ³ /hr)	11
Pump Head (m)	860
Neutron Absorber	Sodium Pentaborate
Borated Water Storage Capacity (m ³)	30
Minimum Reactivity Reduction Rate (dk/min)	0.001

Reactor Water Cleanup System Parameters

Parameter Description	Value
Number of Pumps	2
Number of Filter/Demineralizers	2
Pump Capacity (m ³ /hr)	80

Main Steam System

Parameter Description	Value
Number of Main Steamlines	4
Main Steamline Inside Diameter (m)	0.64
Number of Safety/Relief Valves	18
Capacity of Each Safety/Relief Valve (t/hr)	400
Number of Main-Steam Isolation Valves	8
MSIV Closure Time (sec)	3-4

4.3.2 Emergency Core Cooling Systems

Residual Heat Removal System Parameters

Parameter Description	Value
Number of Pumps	3
Pump Capacity (m ³ /hr)	950
Pump Head (m)	120
Number of Heat Exchangers	3

High Pressure Core Flooder System Parameters

Parameter Description	Value
Number of Pumps	2
Drive Capacity	Electric Motor
Pump Capacity (m ³ /hr)	180-730
Pump Head (m)	890-190

Reactor Isolation Cooling System Parameters

Parameter Description	Value
Number of Pumps	1
Drive Type	Steam Turbine
Pump Capacity (m ³ /hr)	180
Pump Head (m)	190-900

4.3.3 Containment and Environmental Control Systems

Reactor Containment System Parameters

Parameter Description	Value
Type	Pressure-Suppression Type
Shape	Cylindrical
Drywell Head Diameter (m)	10
Inner Diameter (m)	29
Overall Height (m)	36
Maximum Service Pressure, MSP (kg/cm ²)	3.16
Drywell Maximum Service Temp., MST (°C)	171
Suppression Chamber MST (°C)	104
Maximum Leak Rate at 0.9xMSP & Normal Temp (%/d)	0.4
Suppression Pool Water Volume (m ³)	3600

4.3.4 Power Conversion Systems

Main Turbine Parameters

Parameter Description	Value
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Type	Tandem Compound, Six flow, Reheat
Last Stage Blade Size (inches)	52
Speed (rpm)	1800
Rated Steam Flow (t/hr)	7600
Bypass Capacity (t/hr)	2500

Main Generator Parameters

Parameter Description	Value
Type	Direct Driven, 60 Hz, Synchronous
Output Voltage (V)	27000
Cooling	Conduction (Hydrogen)
Speed (rpm)	1800
Gross Electrical Output (MVA)	1600
Rated Power Factor	0.9

Condensate Storage Tank Parameters

Parameter Description	Value
Water Volume (m ³)	1700

Feedwater Pump Parameters

Parameter Description	Value
Number of Pumps	2
Drive Type	Motor Driven, Variable Speed
Pump Capacity (m ³ /hr)	4600

4.3.5 Station Electrical Systems

Offsite Power System Parameters

Parameter Description	Value
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Main Transformer	Bank of 3 Single-Phase Units Plus One Single-Phase Spare
Unit Auxiliary Transformers	3
Reserve Auxiliary Transformer	1

Onsite Power System Parameters

Parameter Description	Value
Buses Per Unit Auxiliary Transformer	3 non-safety related, and 3 safety related (one per div.)
Load Groups	3
Divisions	3

4.3.6 Station Auxiliary Systems

Heating, Ventilating and Air Conditioning System Parameters

Parameter Description	Value
Controls	Temperature, Pressure, Humidity, and Airborne Contamination

4.3.7 Control and Instrumentation System

Reactor Protection System Parameters

Parameter Description	Value
Scram Logic Scram Parameters	2 out of 4 High Reactor Pressure, Low Reactor Water Level, High Drywell Pressure, High Neutron Flux, Short Reactor Period, Rapid Drop in Core Flow, Closure of Main-Steam Isolation Valves, Closure of Turbine Main-Steam Stop Valve, Rapid Closure of Turbine Control Valve, High Radioactivity in Main-Steam Line, or Large Seismic Acceleration. Reactor Shutdown

Other Safety-Protection Circuit Parameters

Closure of Main-Steam Isolation Valves (MSIV's)

Parameter Description	Value
MSIV Closure Parameters	Low Reactor Water Level, High Radioactivity in Main-Steam Line, Low Pressure in Main-Steam Line, Large Flow Rate in Main-Steam Line, High Steamline Tunnel Temperature, or Low Condenser Vacuum

Closure of Emergency Ventilation System and Initiation of Standby Gas Treatment System

Parameter Description	Value
Initiation Parameters	High Drywell Pressure, Low Reactor Water Level, or High Radioactivity in Reactor Area of RB

Initiation of High-Pressure Core Flooder, Reactor Core Isolation Cooling, Low-Pressure Core Flooder and Emergency Diesel Generator Systems, and Closure of Isolation Valves Other Than MSIV's

Parameter Description	Value
Initiation Parameters	High Drywell Pressure, or Low Reactor Water Level

Neutron Monitoring System Parameters

Parameter Description	Value
Initiation Parameter	High Drywell Pressure, and Low Reactor Water Level

Neutron Monitoring System Parameters

Parameter Description	Value
Startup Range Neutron Monitor Channels	10
Startup Range Neutron Type	Compact Fission-Ionization Chamber
Power Range Neutron Monitor Channels	4
Power Range Monitor Incore Sensors	208
Power Range Neutron Monitor Type	Compact Fission-Ionization Chamber

4.3.8 Radioactive Waste System

Offgas Charcoal Bed Holdup System Parameters

Parameter Description	Value
Flow Capacity (m ³ /hr)	40
Minimum Xenon Holdup Time (days)	30
Minimum Krypton Holdup Time (hr)	40

Plant Stack Parameters

Parameter Description	Value
Flow Capacity (m ³ /hr)	550,000
Discharge Elevation (m above grade)	76
Inside Diameter (m)	2.4

4.3.9 Main Control Room

Control Panels

Panel	Description
Console Panels	Main Control Console Large Display Panel

4.3.10 Structures

Plant Structures

Parameter Description	ABWR	Plutonium Disposition
Structure	Reactor Building Turbine Building Control Building Radwaste Building Service Building	Reactor Building Turbine Building Control Building Radwaste Building Service Building Fuel Building

4.3.11 Fuel Storage and Handling

Fuel Storage Rack Module Parameters

Parameter Description	Value
Module Width (inches)	32
Module Depth (inches)	32
Module Height (inches)	169
Module Capacity (spaces)	25 (5x5 square)
Center-to-Center Between Spaces (inches)	6.3
Absorber Material	boral

Spent Fuel Storage Pool Parameters

Parameter Description	Value
Volume (m ³)	2750

Spent Fuel Storage Pool Heat Exchanger Parameters

Parameter Description	Value
Type	U-tube/shell
Orientation	Horizontal
Primary (Tubeside) Flow (m ³ /hr)	250
Primary Inlet Temperature (°C)	52 max
Secondary (Shellside) Flow (m ³ /hr)	280
Secondary Inlet Temperature (°C)	35 max
Heat Exchange Rate (MWt)	1.92
Source of Cooling	RCW System

Spent Fuel Storage Pool Filter/Demineralizer Parameters

Parameter Description	Value
Diameter (m)	1.5
Height (m)	4.66

Refueling Equipment Parameters

Parameter Description	Value
Fuel Servicing Equipment	150-ton Reactor Building Crane Fuel Handling Platform Fuel Inspection Stand 2 Fuel Preparation Machines Channel Handling Tool Channel Handling Boom Fuel Pool Sipper General Purpose Grapple

4.4 REACTOR COMPLEX AND ABWR PLANT SAFEGUARDS AND SECURITY

4.4.1 Site Safeguards and Security

The site safeguards and security systems are provided to deter, detect and delay unauthorized intruders into the site, reactor and fuel fabrication facilities and assure that diversion of nuclear materials does not occur. The systems provide both passive and active components to meet the safeguards and security objectives of the site and consist of perimeter and security fences; patrol roads and walkways; security intrusion detection, lighting and surveillance equipment, entry control points, guard houses and office centers, gates, and vehicle barriers. As noted in Section 4.1 separate systems have been provided for the reactor complex and the fuel fabrication facility.

Perimeter and Security Fences: The entire site is surrounded by a perimeter fence which provides industrial security against unauthorized entry. Additionally, both the reactor and fuel fabrication protected areas are enclosed by two security fences spaced 75 feet apart. A minimum clear zone of 20 feet is provided along each side of both security fence perimeters to facilitate intrusion detection and assessment. The isolation zones between the fences provides an area for security intrusion detection and observation. The fences are

constructed in accordance with DOE orders for security fences and include barbed topping and anchors below finished grade. The outer security fence is provided to minimize nuisance interference (e.g. wind-borne refuse, animals, or pedestrians) with the isolation zone intrusion alarm system. The inner fence is located a sufficient distance inside the perimeter to provide the requisite delay-facilitating response to an alarm and to allow for clear identification of the source of the alarm.

Patrol Roads and Walkways: All-weather patrol roads are provided along the inside of the security fencing surrounding the protected areas. Patrol roads have a lane width of at least 12 feet.

Security Intrusion Detection, Lighting, and Surveillance Equipment: Intrusion Alarm Systems (IASs) will be installed to provide timely detection and assessment of possible intrusion into the isolation zones. These systems will incorporate:

- Fence sensor systems to detect attempted penetrations or scaling of the fences
- State-of-the-art intrusion detection equipment to detect intruders within the isolation zones

The IASs will consist of a combination of two independent generic types of equipment in the isolation zone to alert the security personnel of possible unauthorized intrusion. This equipment will include narrow beam infrared and microwave systems located at appropriate points to give full coverage of the isolation zone. A fence-line sensor (e.g., Israeli taut wire) will be installed on the inner security fences. The IASs will initiate requisite signals that will activate automatic closed circuit television assessment in the area of the alarm and allow guard force response for rapid interception and apprehension of intruders. Appropriate lighting of the isolation and protected zones will be provided to allow visual detection of intruders by patrol guards or by remote closed circuit television viewing. All alarms generated by the IASs will be reported and logged at the appropriate plant central alarm station.

Entry Control Points: Entry control points are designed to provide surveillance of vehicular and pedestrian traffic that enters and exits the protected area (area within the double security fences). The number of entry control points are limited to establish and maintain the level of integrity required by the cognizant DOE Operations Office for secured areas. For both the reactor complex and the fuel fabrication facility, the main entry control points are placed in the isolation zones between the inner and outer security fences. This configuration provides a continuous isolation zone across the entry control points during low traffic periods.

The entry control points are structurally hardened, as necessary, to meet site-specific threat criteria. Entry control points design also facilitates ingress and egress of emergency vehicles fire protection equipment. Primary and auxiliary alarm and communication systems are provided between entry control points and a tactical response force communications center.

Emergency power supplies to support entry control points operation and communication systems are provided. Critical equipment requiring continuous operation will be supplied with battery-powered uninterrupted power. The systems will include emergency diesel generators which will be loaded sequentially and will be capable of accepting full load within 15 minutes of power outage and be capable of supporting full operation for a minimum of 24 hours.

Guard House and Office Centers: Separate Guard House/Office Centers are provided at the entry control points for the reactor complex and the fuel fabrication facility. As described above, these structures are located between the double security fences.

The building will be provided with hardened and non-hardened areas. Hardened areas within each building will provide space for a central alarm station, security system host computer, communications, equipment, emergency and uninterruptible power systems and a mechanical equipment room from emergency ventilation and other critical support systems. Hardened area will be seismic Category I structures which comply with nuclear safety codes.

Non-hardened areas will provide space for entry/exit turnstiles, control booths and card readers, guard station, area offices, men's and women's lockers, a lunch room, and utility rooms as required.

In addition to vehicle access control, the facility will provide controlled personnel access using walk-through turnstiles. Computer-controlled booths will be provided for identity verification, including biometrics tests (i.e., hand geometry) and badge presentation. Nuclear material, contraband material and metal detection equipment will also be provided.

Gates: Points of vehicle and pedestrian access to the protected areas provide the same level of physical protection as that provided at all other points along the security fencing. Gate hardware for security fencing will be installed in a manner that will mitigate tampering.

Motorized gates are provided at the plant primary access points. Motorized gate controls are located within guard stations. Motorized gates will be designed for manual operation during power outages.

Vehicle Barriers: Appropriate vehicle barriers and speed reducers will be installed outside the inner security fences at heavy traffic areas. Above-grade vehicle barriers will be designed to preclude intruder concealment. Speed reducers will be implemented at entry control points to slow approaching adversary vehicles to within vehicle barrier limits.

4.4.2 ABWR Plant Safeguards

The routine procedures for inventory control of the fuel at a standard nuclear power plant such as the ABWR are readily adaptable to safeguarding fuel containing plutonium.

First, the fresh fuel is received at the reactor building in uniquely numbered packages - the fuel bundles. The fuel remains as bundles throughout the residence in the reactor building. Second, the fuel bundles are moved into the reactor, operated in the reactor, and moved out of the reactor in a controlled manner based on the unique identification number of each

individual bundle. Third, the irradiated fuel bundles, which now contain plutonium from the transmutation of U_{238} are stored in a retrievable manner in the in-reactor spent fuel storage pool, again by number. The fuel bundles in a standard ABWR are thus located, both in fuel inventory control records and by direct inspection, at every stage in the reactor building from receipt of the fresh fuel bundles to release of the spent fuel bundles. These procedures will be basically the same but upgraded for the receipt, storage, and movement of fresh mixed-oxide fuel bundles, in order to implement more stringent requirements such as frequency and type of inspection, training of personnel, limited access, guard perimeters and patrols, etc. The procedures for safeguarding irradiated fuel, which is substantially more diversion resistant than fresh fuel, will be the same for both standard fuel and mixed-oxide fuel.

4.5 ABWR PLANT SAFETY AND ENVIRONMENTAL ASPECTS

This section addresses the five topics listed in Section 4.2 of the plutonium Disposition Study Requirements Document; namely:

- Safety margins
- Environmental margins
- Defense in depth
- Worker and public radiation exposure (ALARA)
- Probabilistic risk assessment

4.5.1 Safety Margins

The reactor complex portion of the facilities identified in this report is capable of meeting existing safety and environmental requirements. Each reactor option is licensable under NRC requirements and would satisfy the National Environmental Policy Act.

Since the reactor consists of the GE ABWR, the primary basis for the above statement of NRC licensability is the ABWR Standard Safety Analysis Report. This report has been

reviewed by the NRC; and shows conformance of the ABWR to NRC safety requirements. All that is necessary here is to indicate why the use of plutonium fuel will not pose a significant challenge to any requirement.

The ways in which the use of plutonium fuel could, in theory, affect safety requirements are:

- 1) **Fresh Fuel:** The radioactivity of plutonium and americium in fresh fuel may affect handling.
- 2) **Plutonium Core Dynamics:** The new plutonium cores have potentially different response during Chapter 15 transients.
- 3) **Chapter 12 Source Terms:** Source terms used in the ABWR SSAR Chapter 12, Radiation Protection, may differ in plutonium content from what would occur with a plutonium fueled design.
- 4) **Severe Accident Source Terms:** Source terms used in SSAR Chapter 19, Response to Severe Accident Policy Statement, may differ in plutonium content from what would occur for a plutonium fueled design.
- 5) **Spent Fuel:** Spent fuel characteristics in theory could differ from normal in some options.

Each of these issues is addressed in a section below.

Fresh Fuel

This issue is covered in the safety discussion of Section 3.9 for the fuel fabrication plant, where the problem will initially be faced. Handling at the reactor should be inherently less of a problem due to the smaller numbers of bundles present at one time. Accidents

involving fresh fuel are inherently less hazardous than spent fuel because the radioactive materials are in the form of non-volatile insoluble ceramics. There are no volatiles as is the case with spent fuel fission gases.

Plutonium Core Dynamics

The behavior of plutonium cores during transients is addressed in Section 2.8 where an assessment of limiting transients show that the requirements addressed in SSAR Chapters 5, 6, and 15 are met for all plutonium-disposition ABWR designs considered.

Chapter 12 Source Terms

The use of plutonium fuel in an ABWR can affect the source terms used in the SSAR Chapter 12, Radiation Protection, since there is proportionately more plutonium present. For example, the amount of Plutonium-239 present at the end of an 18 month cycle in a normal ABWR core is 775 kilograms. Compared to this, among the cores considered in this study the maximum plutonium content is 9200 kilograms (=10.55 kilograms of plutonium per bundle times 872 bundles). Thus, there is about 12 times as much plutonium as for a normal ABWR. Operational source terms in which plutonium (or its precursor Np239) are found would be expected to contain about 12 times as much plutonium as given in the source terms of Chapter 12 (Radiation Protection). It should be noted that these Chapter 12 source terms are highly conservative. In actual practice there may be no plutonium, or other insoluble fuel elements, released from the pins. In that case, the added amount of plutonium in the pins has no effect, since none is released.

Chapter 12 Radiation Protection of the SSAR addresses conformance of normal operations to radiation protection limits. Even if there were some release of insoluble such as plutonium from a pin, an increase of a factor of 12 in plutonium would not affect conformance of the reactor to these limits because the margins between the source term concentrations and the maximum permissible are so large. For example, the SSAR shows that a conservative estimate of the annual average airborne release of Np239 (Plutonium-

239 precursor) is $1.2\text{e-}02$ Curies/year at a concentration of $7.4\text{e-}16$ microcuries/cm³. This concentration is almost eight orders of magnitude below the 10CFR20 limit of $2\text{e-}8$ microcuries/cm³.

Severe Accident Source Terms

Similar to the case of SSAR Chapter 12 operational source terms, the severe accident source terms of Chapter 19 are such that all quantitative safety goals are met with large margins. Furthermore, plutonium does not contribute to these safety goal risks; hence, the presence of additional plutonium in the disposition option designs has no effect on Chapter 19 risks. Thus, the plutonium-disposition ABWR also meets the goals. Specifically these safety goals and the corresponding assessed values from Chapter 19 of the ABWR SSAR are as follows:

<u>Type of Goal</u>	<u>Goal Value</u>	<u>ABWR Value</u>
Individual Risk	$3.9\text{x}10^{-7}$	$1.8\text{x}10^{-11}$
Societal Risk	$1.7\text{x}10^{-6}$	$1.5\text{x}10^{-10}$
Radiation Dose Probability at 25 Rem	$1.0\text{x}10^{-6}$	$3.9\text{x}10^{-8}$

As indicated, the safety goals are achieved by the ABWR design with very large margins. The margins above are by factors of 25 to 20000. In the plutonium fueled ABWRs considered here, the amount of plutonium present is greater than the amount in the original ABWR by a factor of only 12. Thus, even if the entire dose in these assessments were due to plutonium, the assessed values could only increase by a factor of 12, which is less than the smallest margin of 25. Actually there is no contribution by plutonium at all to the risk values above. In fact, because of certain safety characteristics of the design, no plutonium is released in accidents. Hence, the addition of plutonium to the fuel has no effect on Chapter 19 source terms. These safety characteristics are described briefly below.

Two design characteristics of the ABWR responsible for eliminating plutonium from the source terms are 1) the filtered vented containment, and 2) the use of fusible plugs in the lower drywell (below the core). These fusible plugs cause flooding the drywell in case of over-temperature conditions, thus covering the core debris with a layer of water.

The filtered-vented containment feature prevents over-pressurization failure of the containment by venting it through a filter if excessive pressure is reached. Any plutonium in the vented atmosphere is filtered out.

In the more extreme accidents where the core melts through the reactor vessel, it will spread out across the large floor area of the drywell. The high temperatures will cause the fusible plugs to flood this core debris with the large volume of water in the wet well. Hence, the core is in an easily coolable condition. Furthermore, it has been shown (Ref.4.5-1) that as long as the core is covered by as little as one meter of water, no plutonium will be sparged into the containment atmosphere. Thus, even in these extreme conditions no plutonium will be released.

In these core debris scenarios, the need to vent containment is has been suppressed by use of a low-gas-generating type of concrete. Thus, in any case there is essentially no release of plutonium.

Another possible cause of containment failure, hydrogen ignition, is precluded by inerting of the containment.

Spent Fuel

Spent fuel is covered in the safety discussion for the spent fuel section of this report, Section 5.6.

4.5.2 Environmental Margins

The source terms which would demonstrate conformance of the plutonium disposition reactor to NEPA environmental requirements are derived from those given in Chapter 12 of the ABWR SSAR. As discussed in item 3) above, the levels of plutonium, even for these conservative source terms, is far below the maximum permitted. Hence, all the plutonium disposition option reactor designs will easily meet environmental requirements as far as plutonium involved in source terms. The environmental aspects of also adding tritium production to the reactor are discussed in sections 6 and 8 of this report.

4.5.3 Defense In Depth

Defense in depth has been incorporated into the design of ABWR in two ways. Firstly the traditional layers of defense have, in virtually each case, been reinforced by added redundancy or elimination of failure modes. Secondly, an extra layer has been added to the depth of the defense, namely the below core coolability and vented containment features. In a normal reactor, vessel melt-through or over-pressurization of containment lead to significant releases. These situations are highly mitigated in the case of the ABWR.

The first layer of defense in depth, power control reliability, has been enhanced by elimination of the scram discharge volume feature, a potential common cause failure mode. An enhanced degree of physical separation of control lines has the same effect.

The heat removal layer of defense in depth has been enhanced by adding loops to cooling and water supply systems.

4.5.4 Worker and Public Radiation Exposure (ALARA)

This topic is related to the environmental aspects of operational source terms discussed above. As explained, a plutonium fueled reactor would add little or no public exposure

beyond the very low levels of a normal ABWR. The ABWR reactor, as discussed in the SSAR Chapters 11 and 12 has radiation protection features that comply with NRC requirements. DOE requirements that go beyond those of the NRC are discussed in Section 8.

For the ABWR steps have been taken to markedly reduce occupational exposure levels below that experienced in current plants. This was achieved by building low dose maintainability into the design from the beginning.

Quantitative estimates of the ABWR reactor contribution to worker and public exposures are given in Subsection 8.1.7 of this report.

4.5.5 Probabilistic Risk Assessment

A Probabilistic Risk Assessment has been performed for the ABWR, and is reported in Chapter 19 of its SSAR. The use of plutonium fuel would not increase the risk at all because no plutonium is released in any of the accident source terms. The reasons for this are discussed in the paragraph, "Severe Accident Source Terms", under "Safety Aspects" above.

4.6 ABWR PLANT DEVELOPMENT NEEDS

The standard ABWR plant has been adopted for plutonium disposition, therefore, little, if any, is anticipated. Core design for mixed-oxide fuel and fuel assembly lead bundle and reuse confirmation tests, are discussed in Section 2. The transfer of fuel bundles between the in-reactor and ex-reactor spent fuel storage pools is discussed in Section 5. Other modifications of the ABWR plant for plutonium disposition, such as on-site electrical power and service (cooling) water systems, are expected to be issues of scale rather than development.

The cost and schedule for the standard ABWR plant are the baselines used to project application of the ABWR to plutonium disposition, and are presented in detail in the cost and schedule discussions in Section 10. The basis for the cost estimate was the capital costs estimate for an Nth-of-a-kind 1350 MWe (gross) ABWR plant conducted in the late 1980's, which was based on the extensive BWR experience and which used the DOE Energy Database Program Code of Accounts. That cost, in current dollars, is approximately 1400\$/kWe. Similarly, the schedules for construction and startup, four years after site preparation and one year, respectively, are consistent with the ABWR schedules being implemented currently. With two ABWRs currently under construction, it was assumed that Title I design could proceed immediately, and, since the ABWR SSAR is in place, that the time to start-of-site preparation would be determined by the time to issue the environmental record-of-decision, which was estimated to be approximately two years. Thus, the ABWR plant can be ready for full-power operation for plutonium disposition in less than eight years after the start of the program.

SECTION 5.0

SPENT FUEL MANAGEMENT

As an integral part of the plutonium consumption activities, supplemental storage of spent fuel will be required in an out of reactor (ex-reactor) storage facility due to the quantity of spent fuel generated over the allotted time period and due to the fact that final disposition to a mined geologic disposal system cannot be assumed available during the time period. A minimum storage period of 10 years is identified in the requirements document. Various options including re-use of short burn (spiked) fuel assemblies back to the reactor core for power production during the life of the reactor require fuel storage up to 50 years and the capability to retrievably store all spent fuel generated.

The initial selection of the type of ex-reactor storage facility (dry or wet storage) for the plutonium consumption options was critical. Only proven or high confidence technologies were considered. Design concepts focused on spent nuclear fuel storage designs that were licensed by the NRC. Several variables were considered for proper selection of the storage facility. A primary consideration was operational adequacy of the storage process to allow reuse of the spent fuel in the reactor core, in addition to cost considerations. Wet pool storage was selected. Section 5.4 provides additional discussion.

5.1 SPENT FUEL MANAGEMENT OVERVIEW

Management of spent fuel in an ex-reactor storage facility encompasses several aspects. Safe storage of spent fuel must incorporate radiological controls along with safeguards and security, safety, licensing, waste minimization, and environmental compliance. Radiological controls are addressed in Section 5.4. Safeguards and security is addressed in Section 5.5. Safety and Licensing is addressed in Section 6.1. Waste minimization is addressed in Section 8.3. Environmental compliance is addressed in Section 5.6.

5.2

SPENT FUEL GENERATED FROM VARIOUS ALTERNATIVES

The number of spent fuel bundles generated each year by each of the three disposition scenarios are summarized in Table 5.2-1. Note, that the reactor operating cycle for the spiking and spent fuel alternatives are less than one year in duration and that the operating cycle for the plutonium destruction alternative is greater than one year in duration. The discharged number of spent fuel bundles per year given in Table 5.2-1 is the equivalent number discharged on an annual basis. Also, for the spent fuel alternative, the first number indicates the initial operating phase and the second number indicates the reuse phase.

Table 5.2-1. Summary of Parameters for the Spent Fuel Bundles for the Various ABWR Scenarios

Spent Fuel Parameters	Reference	Option 1
<u>Spiking</u>		
Reactors (#)	1	1
Reactor Operations (yrs)	7.4	4.7
Discharged Bundles (#/yr)	1,920	1,877
Discharged Bundles (total)	14,706	9,320
Avg Bundle Burnup, MWD/Te at Discharge	2,000	2,000
<u>Spent Fuel</u>		
Reactors (#)	1	
Reactor Operations (yrs)	17/43	
Discharged Bundles (#/yr)	838/327	
Discharged Bundles (total)	14,706/14,706	
Avg Bundle Burnup, MWD/Te at Discharge	7,200/26,000	
<u>Plutonium Destruction</u>		
Reactors (#)	6	2
Reactor Operations (yrs)	19	58
Discharged Bundles (#/yr)	1,019	316
Discharged Bundles (total)	18,766	18,766
Avg Bundle Burnup, MWD/Te at Discharge	37,000	37,000

5.3

SPENT FUEL HANDLING REQUIREMENTS

The spent fuel handling requirements are driven by the requirement to dispose of the 100 metric tons of weapons-grade plutonium in less than 25 years. The extremely short reactor

operating cycle for the spiking option and the shorter than normal operating cycle for the spent fuel option lead to increased spent fuel storage requirements over that normally provided with an ABWR. The plutonium destruction case using multiple reactors is typical of a normal ABWR fuel cycle and does not require additional spent fuel storage capacity. The additional spent fuel storage facility must be able to dissipate the decay heat from these bundles and must meet subcriticality requirements.

High density storage rack designs which meet subcriticality requirements for BWR fuel bundles with up to five percent fissile enrichment are available. Preliminary analyses using the TGECLA code to calculate the k-infinity for infinite arrays of the various mixed-oxide fuel bundle designs developed for this study indicated that there should be no problem to provide a spent fuel handling system that will meet the subcriticality requirement. While the detailed analyses for spent fuel bundles from an ABWR have not yet been done, no feasibility issues are expected.

The decay heat associated with the spent fuel bundles of the various plutonium burning options must be dissipated by the spent fuel handling system. Preliminary decay heat calculations were done using the preliminary fuel bundle design described in Section 2.5 and the irradiation time was treated parametrically. The ORIGEN2 code was used to perform a point burnup analysis and decay evaluations. These results provide a reasonable estimate of the decay heat/bundle that will be required to be removed by the spent fuel storage system. The bundle decay heat is dependent on both the bundle power level, the irradiation time and the decay time. Only the bundle average power level is a constant in the various plutonium burning approaches whether spiking, spent fuel, or the maximum plutonium destruction alternative. For any specific bundle the power level is not a constant because the bundle power level depends on core position, fuel management, and control rod patterns. Figures 2.5-18 through 2.6-17, and 2.7-19 show the decay heat of the average bundle for the spiking, spent fuel and plutonium destruction alternatives, respectively. The decay heat in watts at selected decay times is shown in Table 5.3-1 for an average fuel bundle for the reference case for each of the plutonium disposition alternatives studied.

**Table 5.3-1. Calculated Decay Heat (watts)
for Average Power ABWR Bundle**

Decay Time	Spiked	Spent Fuel	Plutonium Destruction
2 Days	13520	16751	19907
10 Days	6882	9910	13091
30 Days	3686	6342	9402
120 Days	1488	3158	5497
1 Year	496	1159	2404
2 Years	218	471	1103
30 Years	115	137	277
60 Years	110	131	232

5.4 SPENT FUEL HANDLING DESIGN AND LAYOUT

Two technologies for spent fuel storage were assessed for utilization with the plutonium consumption options. This selection was based primarily on readily available information at the time of this assessment. A primary source of this information comes from the "Monitored Retrievable Storage Facility Conceptual Design Report" prepared for the U. S. Department of Energy (Ref.5-1). This report selected Vertical Concrete Cask Dry Storage for the reference design concept, therefore, this was the primary dry storage concept addressed in the evaluation. Also available was extensive information on the NUHOMS (*TM) Modular Storage System by Pacific Nuclear Services, Inc., utilizing horizontal dry storage technology. Wet transfer and storage of spent nuclear fuel assemblies has been licensed by the NRC at numerous commercial nuclear power facilities and has been used for many years throughout the United States and has extensive literature available. ANSI Standards were the primary source of information utilized to assess the wet storage option.

Wet pool storage was selected as the type of spent fuel storage most amenable to the needs and requirements for the plutonium consumption options. The primary justification for this selection is that there is some limited experience reusing spent fuel in the reactor core which is proposed as one of the spent fuel options. Dry storage has no such experience

along with some oxidation concerns when stored in air. Typical dry storage temperatures average 300-350 Celsius versus average wet pool temperatures around 100 degree Fahrenheit. Dry storage is also very limited by removal of decay heat. Some of the plutonium Consumption options will require storage of spent fuel aged only one month. Wet pool storage has enhanced capabilities of removing the quantities of decay heat that are anticipated. It also appears that wet pool storage for these quantities of spent fuel (approximately 2500-3000 metric tons heavy metal) is economically comparable to dry storage.

The wet storage approach utilizes a single, large building for storage of the spent fuel. It uses water for both radiation shielding and cooling of the spent fuel. The wet storage design concept uses underwater transfer of spent fuel to storage racks which are in water-filled storage pools. Transfer canals and storage pools are located inside a large, concrete structure. The design provides wet transfer of the fuel assemblies from the reactor pool utilizing the inclined fuel transfer systems which have been used with the BWR-6 reactors.

Many portions of the following text regarding water pool type storage of spent fuel are extracted from American National Standard ANSI/ANS-57.7 - 1988 (Ref.5-2) with permission of the publisher, the American Nuclear Society.

An independent spent fuel storage facility is a complex designed and constructed for the storage of spent fuel. An ex-reactor pool which is located on the site of another facility, as will be the case here, is permitted by ANSI/ANS 57.7 to share common utilities and services with the other facility and still be considered to be independent, provided that such sharing of utilities and services or physical connection does not: (a) increase the probability or consequences of an accident or malfunction involving components, structures or systems that are important confinement features, or (b) reduce the margin of safety as defined in the bases for any technical specifications of either facility.

The primary function is the interim protective, custodial storage of spent fuel. The facility is not intended to process or utilize the nuclear material. The facility is designed to

preserve the primary barrier of the fuel unit or assembly and provide an additional secondary barrier against the potential release of particulate radioactive materials to the immediate environs. The primary barrier normally is the fuel cladding for which protection against damage and degradation is provided by storage confinement, storage environment, fuel handling equipment, and decay heat removal system, as required.

Radioactivity control is provided by several secondary barriers. The pool water itself retains the radioactive particulate material which may be released from the spent fuel and provides personnel protection in the form of shielding from gamma radiation. Subcriticality of the spent fuel while in storage is ensured by the design features of the storage racks. Subcriticality of the spent fuel during normal handling is ensured by the design features of the handling system.

The pool water cleanup system is provided to remove dissolved and suspended radioactive material. The pool water cooling system is a closed system to preclude releasing radioactivity to the local environs. A pool water makeup system is provided to allow compensation for losses of pool water level.

In areas of the facility with the potential for significant release of airborne particulate radioactive material, the heating, ventilation and air conditioning system provides filtration of the effluent air.

5.4.1 Facility System Functions

The water pool storage facility is typically divided into the following systems:

Fuel Unit Storage and Cask Unloading Pools (Structural and Liner). The fuel unit cask and storage pools system includes the structures with which the functions of spent fuel cask unloading/loading, and storage of spent fuel take place. The design and configuration of these pools are closely interrelated with the cask handling, fuel handling, and fuel storage rack systems. It is proposed that this facility will not be provided with cask handling

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facilities. All transfers requiring cask handling will be conducted from the reactor fuel pool. The inclined transfer system will be utilized to transport material back to the reactor fuel pool from the ex-reactor pools.

Fuel Unit Storage Racks. The fuel unit storage rack system includes structures located in the fuel storage pool. They support and maintain the spent fuel assemblies in a subcritical geometry during storage. Commercially available high density storage racks will be procured.

Fuel Unit Pool Water Makeup, Cooling and Cleanup. The fuel pool water makeup, cooling and cleanup system includes equipment for the removal of decay heat by an external heat exchanger and for the removal of radioactive material and particulates from the pool water by circulation through filters and ion exchange units. The system also provides makeup water to the storage pools under normal and emergency conditions. The above system functions can be performed by physically separate systems.

Fuel Unit Handling. The fuel unit handling system includes the equipment for removal of the fuel assemblies from the inclined transfer system, transfer of the fuel units to the storage racks, inspection of racks and handling of damaged fuel units.

Heating, Ventilating and Air-Conditioning. The heating, ventilating and air-conditioning (HVAC) system includes all air moving and treatment equipment and associated controls required for the protection of the general public and installation personnel from potential release of airborne radioactive particulates, and maintenance of air quality suitable for operating personnel and equipment operation. HVAC systems for the ex-reactor pool will be in accordance with ANSI/ANS-57.2 (Ref.5.3) due to the short cooling/aging time in the reactor pool for some of the options.

Buildings. The building system includes the superstructures for the fuel storage pools and other structures required to house various components. The confinement building for the ex-reactor pool will also be in accordance with ANSI/ANS-57.2.

Radwaste Treatment. The radwaste treatment system includes equipment for handling, treatment, and preparation for disposal of liquid, gaseous and solid radioactive wastes.

Electrical Power, Instrumentation and Control (I&C), and Communications. The electrical power, I&C, and communications system includes electrical power, system control and status indication, lighting, communications, alarms, radiation monitoring, and access control.

For electrical power, the system provides for onsite power distribution and the onsite terminus of the offsite power supply. The system includes equipment required to distribute power to components and subsystems identified as follows. The offsite power subsystem terminates at the feeder breakers.

For control and status indication, the system includes equipment to provide the operator with system status and allow for proper operation of the pool storage facility.

For lighting, the system includes equipment to provide work area and security lighting under normal and abnormal conditions.

For communications, the system includes the equipment to provide for public address, evacuation alarm, communication between in-plant locations and communication with locations remote to the installation.

For alarms, radiation monitoring, and access control, the system includes the equipment to provide warning of abnormal conditions such as high radiation levels, airborne radiological contamination, or unauthorized access.

5.4.2 Facility Description

Figures 5.4-1 and 5.4-2 show the preconceptual layout and section of the ex-reactor spent fuel pool facility. The ex-reactor pool facility consists of three storage pool basins. The pools will be provided with refueling platforms identical to the fuel handling equipment associated with the reactor spent fuel pool. The two outboard basins will be designed to receive fuel transfers via the incline fuel transfer system utilized with BWR-6 fuel buildings. The center pool will be provided with a modified transfer system to receive new fuel for interim storage from the truck bay within the facility. The pools will be enclosed within a confinement structure.

Fuel Pools. The pools are constructed of two feet thick reinforced concrete to be seismically qualified. Each pool is 45x60 feet and 45 feet deep. Each pool is provided with a stainless steel liner with an annulus space or channels provided with leak detection capability. The pools are interconnected with vertical swing gates with pneumatic seals. A transfer canal is also provided so transfers can be made around the center pool when it is empty.

Fuel Storage Racks. The fuel storage racks are top entry racks, designed to maintain the spent fuel in space geometry that precludes the possibility of criticality under both normal and abnormal conditions. This is accomplished with the aid of neutron absorbing materials integral to the racks. The fuel racks are the same commercially available high density racks for BWR fuel that are used in the reactor pool. The racks are manufactured as modules in a 5x5 array measuring 32 inches square. This provides a center to center spacing of 6.3 inches. The racks will be mounted in the pools utilizing seismic restraints. Adequate flow of water by natural convection will preclude localized boiling.

Fuel Pool Water Makeup, Cooling and Cleanup. Each basin is provided with a separate makeup water line, pool water cleanup loop and a decay heat removal loop.

Figure 5.4-1 Ex-Reactor Fuel Storage Building

Each loop contains redundant components to preclude single point failures (i.e., two recirculation pumps, multiple filters, two heat exchangers, etc.). Also provided are cross connections between the different pool systems to allow backup or supplemental capacity from an adjacent pool.

Pool water cleanup is accomplished by recirculating the water through filters and a demineralizer. The pool volume is approximately 900,000 gallons which requires a recirculation rate of 1260 gpm to accomplish two volume changes per day to maintain adequate clarity for fuel handling activities. Each pool is provided with two heat exchangers. The heat exchangers are sized identical to the reactor pool heat exchangers (1.65×10^6 Kcal/hr). A single heat exchanger has adequate capacity for the spent fuel option. Both heat exchangers are required for the spiking option. An auxiliary heat exchanger must be borrowed from the middle pool during maximum anticipated heat load which only occurs after transferring a tritium core to the ex-reactor pool after a staggered 30/60/90 day cooling period. The heat sink for the heat exchangers is provided by the reactor cooling tower system.

Fuel Handling. The refueling platform is a bridge crane running at close to floor level in the pool area. It is identical to the refueling platforms used in the reactors and is provided with digital indexing capabilities to position and store the spent fuel assemblies. This computerized system will be modified with software to provide enhanced Nuclear Material Control and Accountability. Auxiliary systems to be incorporated include personnel card reader access to the system with password to identify and authorize use of the crane. Any material transfers would have to be entered into the accountability system prior to movement, thus providing real time inventory. Supervisory authorization from an independent location would be required to move an assembly to the disassembly stations for tritium target removal. A scale interlocked to the disassembly station would be used to verify that only the four tritium rods were removed during disassembly. This same software package would also be used for the storage of new fuel in the spent fuel facility.

The inclined fuel transfer system is shown in figure 5.4-3. This fuel transfer system transfers

The inclined fuel transfer system is shown in figure 5.4-3. This fuel transfer system transfers two fuel assemblies at a time between the reactor pool and the ex-reactor pool. A flapper valve and water injection are provided to prevent drainage of the elevated reactor pool to the ground level pool. An isolation valve is also provided for total separation when the transfer system is not in use. The mechanism is essentially a cable driven dumbwaiter traveling through an inclined transfer tube. The transfer tube must be heavily shielded along the entire length. Each pool has an enclosed section 10x20 feet connected with a swing gate. This allows the section to be drained for hands on maintenance of the mechanical transfer system. Normal operations have been shown to result in a transfer rate of four assemblies per hour in either direction.

A modified inclined transfer system will be installed to the middle pool from the truck bay within the facility to permit direct transfer of new fuel into the middle pool which may remain dry until required for spent fuel storage. New fuel storage is in accordance with ANSI/ANS 57.3 (Ref.5-4). Utilization of this transfer system will then require use of the fuel handling platform to move new fuel to a storage position which will result in all new fuel being entered directly into the computerized material accountability system. This modified transfer system will not be used for transfer of fuel out of the pool and can be locked out.

A unique fuel handling requirement is imposed during the tritium production option. Tritium target rods must be removed from the standard fuel assembly. This requires disassembly of the fuel assembly and removal and placement of the target rod into a target transfer canister for shipment. The work involves dechanneling the bundle, removing the bail and upper guide plate, removing the tritium target rods and placing them in a target transport canister, and then reassembling the bundle without the tritium targets. The throughput requirement for handling the tritium targets imposes the need for two fuel disassembly stations. One is located in each of the outboard pool basins (Fig. 5.4-1). The transport canister is sized identical to a fuel assembly and can then be transferred to the reactor pool for off-site shipment in a standard spent fuel shipping cask.

Figure 5.4-3 Fuel Transfer Arrangement

Heating, Ventilation and Air Conditioning: The HVAC systems are designed in accordance with ANSI/ANS 57.2 due to the short cooling/aging time in the reactor pool for some of the options. The confinement system is maintained at a negative pressure relative to atmospheric pressure. The system is designed to provide seven air-changes per hour, requiring a total flow rate of 90,000cfm. Supply air treatment and filtration is provided to reduce dust loading in the pool area. An exhaust filtration train is provided with HEPA filters and charcoal absorbers. This system is normally bypassed but is automatically brought on-line by activation of any of the continuous air monitors or radiation alarms in the pool basin area. This is similar to the reactor plant standby gas treatment system.

Buildings: The walls of the confinement building are constructed of reinforced concrete to provide seismic and tornado resistance. The roof over the pool area includes 4 inches of reinforced concrete. This qualified confinement structure is required because some of the options require transfer of fuel to the ex-reactor pool in less than one year.

The equipment cells provide a shielded area for pool water cleanup components that concentrate radioactive particulates removed from the water. The result of physical concentration is increasing dose rates from the equipment for durations between changes or regeneration of the media of filters or resin columns. The cells are provided with removable cover blocks for intermittent maintenance. A gantry crane is provided in this area for removal of cover blocks and removal of failed equipment to shielded containers for transport and disposal.

The pool basin and equipment cells are separated from the radwaste portion of the building by a full height concrete wall. This allows separation of the ventilation zones for the pool area which has the higher potential of radioactive airborne contamination.

Radwaste Treatment: The radwaste system is designed to collect, monitor and process liquid and solid radioactive wastes. The design will minimize exposure to operating and maintenance personnel. Floor drain and cell washdown solutions will be collected by gravity flow to a below grade collection tank. A separate radwaste collection tank is provided for

normal radioactive process waste streams such as filter backwash solution and spent resin discharge. The collection vessels are provided with mixing and sampling capability to categorize solutions prior to dispositioning them for disposal. All liquid radioactive waste including spent resin and filter sludges will be transferred to the reactor radwaste facility for dewatering and disposal. All solid radioactive waste will be packaged for shipment and disposal in accordance with current standards. Compaction will be provided to reduce volume of typical wastes (clothing, rags, wipes, plastic, etc.).

Electrical Power, Instrumentation and Control (I&C), and Communications: The electrical, I&C and communication systems are designed in accordance with commercial codes and standards. System components are designed to operate within the environmental conditions of their location (temperature, humidity, radiation levels, etc.).

A UPS system will be provided for uninterruptible emergency power to vital instrumentation, lighting, physical security and communication systems. A standby diesel generator is provided for backup power to the ventilation system and cooling loop pumps. Wiring and installation shall be in accordance with the National Electric Code.

Instrumentation is provided to detect and annunciate at high radiation levels. Area radiation monitors, continuous air monitors, stack monitors and radiation detectors at specific equipment are provided. A separate Criticality Alarm System will be provided. Because of the nature of the facility, automatic action and immediate operator action is not typically required, although system interlocks will be provided. Security systems for access control, motion detection and material control and accountability are provided.

Communication systems will include standard telephone service in addition to an internal multistation intercom or site telephone. Use of portable radios for normal communication will be minimized.

5.5

SAFEGUARDS AND SECURITY

It is stated in 10CFR72 (Ref.5-5) that a safeguards and security plan shall be provided and approved. The site plan will include the specific considerations for the ex-reactor pool. The following items will be addressed in the Site Safeguards and Security Plan. The ex-reactor spent fuel storage facility will be located within the site physical security and access control area. The facility is constructed of reinforced concrete without windows. Access to the facility will be limited to a minimal number of ingress and egress locations. Personnel access to the facility will be restricted and controlled by use of a card reader access control system as an integral part of the site security system. In addition, physical security of material stored in the pool basins is further controlled by the Material Control and Accountability system provided in conjunction with the fuel handling (refueling) platforms.

Also, if new fuel is stored as planned in the ex-reactor storage facility prior to completion of reactor construction, a security island in the middle of a minimal security construction area will be established with an additional cost for temporary barriers and access control systems required before the site systems are installed and activated.

5.6

SAFETY AND ENVIRONMENTAL ASPECTS

5.6.1

Safety and Licensing

The ex-reactor pool storage facility is designed a seismic category I facility. The confinement structure over the pools and the storage racks are also seismically qualified. The confinement building is also designed to withstand tornado winds and missiles. For criticality safety, Monte Carlo techniques are employed in the calculations performed to assure that k-effective does not exceed 0.95 under all normal and abnormal conditions. Fuel handling equipment is designed with excess capacity above design loads.

The spiked fuel option requires that spent fuel leave the reactor pool after aging only 30-120 days. Normal ex-reactor pool storage facilities are designed with the assumption that

spent fuel has aged at least one year in the reactor pool. In this fashion, it is normally assumed that the short-lived high specific activity radionuclides, particularly those of Iodine and Xenon are no longer present in significant quantities. It is this assumption that would normally allow the building housing the pool structures not to be designed as a confinement structure. It is therefore assumed that the pool enclosure building for the ex-reactor storage facility will be required to be qualified as a confinement structure with adequate off-gas treatment systems provided with the ventilation systems (in accordance with ANSI/ANS 57.2).

5.6.2 Environmental and Worker Safety Issues

In relation to other nuclear facilities, the spent fuel storage facility is a relatively 'clean', low impact facility. Fuel bundles are transferred to and placed in the pool storage area by computer-controlled equipment thus minimizing radiation exposure to humans. The supply air cleaning system is used to control infiltration of nuisance dusts into the facility and the exhaust system controls particulate pollutants exhausted from the facility. In addition, a pool water cleanup system is used to collect, concentrate, and remove impurities suspended in the pool water.

The spent fuel storage facility is planned to take fuel from a variety of reactor burn scenarios. The safety and environmental impacts were projected from the spent fuel option where 2,618 MT of material is discharged in 17.4 years at an average discharge burn of 7,000 MWD/MT. Fuel is allowed to 'cool' for a one year period in the reactor pool prior to transfer to the spent fuel storage facility.

Benchmark - Except for reactor pool storage, this type of storage experience is limited. The benchmark chosen for a comparison to this facility has never been built. It is the Monitored Retrievable Storage Facility (MRS) proposed in the mid-1980s for the State of Tennessee. This project, although now terminated, was the only recent similar project for which some NEPA documentation was generated. The primary documentation of use to this project was an Environmental Assessment (EA), DOE-R-0035-1.

The MRS environmental assessment used a 33,000 MWD/MT burnup for fuel and a fuel accumulation rate of 3,600 MT/yr. The spent fuel storage facility is projected to accumulate fuel at four percent of this rate over a 17.4-year period at slightly less burnup (i.e. smaller source term). The MRS fuel was assumed to have decayed one half-life for the radionuclides of interest, whereas the spent fuel is not considered decayed for the radionuclides of interest. The environmental source terms under these conditions were scaled to be consistent with the conditions of this project. The scaling factors were determined to be:

Reference Case

- Burnup - $7,000/33,000 = 0.21$
- Accumulation Rate - $2,618 \text{ MT}/17.4 \text{ yr} = 150 \text{ MT/yr}$ or 4 percent of MRS
- Decay - none for ^3H , ^{85}Kr , or ^{129}I

The comparison factor is therefore; $0.21 \times 0.04 \times 2 = 0.018$.

5.6.3 Gaseous Effluent Source Terms

Radionuclides released from the facility are quite limited due to the confinement of the fuel cladding and the pool water. Basically the only radionuclides available for release are longer-lived highly mobile radionuclides such as hydrogen gas and noble gases. All other radionuclides are either contained in the cladding or are collected in the water cleanup unit.

The gaseous effluents are collected by the ventilation system and are passed through HEPA filtration prior to discharge to the atmosphere. Exhausted gases will be redirected through a filtration train with charcoal absorbers in the event continuous effluent monitors indicate a problem. The following specifications were used for offsite assessment in this report:

- Stack - 30 meters in height with an inside diameter of 13" (0.77 m) at the top.
- Discharge volume of 35,000 scfm ($16.5 \text{ m}^3/\text{sec}$).
- Exit velocity of 35 m/sec.

The source terms developed for this facility are provided in Table 5.6.1. These values were calculated using the factors developed above. These source terms were then used to determine the incremental radiological impacts which are presented in Table 5.6.2.

Table 5.6.1. Spent Fuel Storage Facility Source Terms

Radionuclide	MRS Annual Release (Ci/Yr)	Spent Fuel Storage Annual Release (Ci/Yr)
^3H	290	5.2
^{85}Kr	9,600	172.8
^{129}I	0.03	0.00045

Table 5.6.2 Spent Fuel Storage Facility Environmental Radiological Impact

Individual Facility	Maximum Offsite Dose (mRem/Yr) 400 meters	Maximum Offsite Dose (mRem/Yr) 10 km
Inhalation	1.68×10^{-6}	1.45×10^{-4}
Ingestion	2.43×10^{-4}	1.91×10^{-2}
Ground Surface	1.51×10^{-7}	1.29×10^{-5}
Air Immersion	7.88×10^{-6}	6.77×10^{-4}
Total	2.45×10^{-4}	1.93×10^{-2}

5.6.4 Liquid Effluent Source Terms

The spent fuel storage facility has no planned liquid effluents. Liquid wastes are transferred to the reactor radwaste treatment building for treatment and preparation for disposal.

5.6.5 Occupational Safety

This facility has no outstanding occupational hazards, except for a criticality accident potential and work conducted near a deep pool of water. Only typical industrial hazards should be encountered in this facility.

Some occupational radiation exposure could be encountered. However, facility design, equipment limitations, and work controls are expected to keep radiation exposures as low as reasonably achievable (ALARA). During normal storage, fuel bundles are stored under a sufficient amount of water to decrease the external dose rate to significantly below one mrem/hour. During transfer, fuel bundles are moved at a less depth thus causing the exposure rates to approach two mrem/hr. Due to remote operations, personnel should seldom be exposed to this rate. Operation of the facility should not exceed about five person-rem/year.

5.6.6 Other Considerations

The construction and operation of the spent fuel storage facility may have impacts other than radiological. These impacts are addressed for the whole complex in more detail in Section 8.0. However, a few that are unique to the storage facility are addressed in this section.

5.6.6.1 Air Resources and Noise

Waste heat dissipated by the facility is expected to be less than five megawatts through the site cooling towers. No perceptible changes to the downwind environment are anticipated as a result of this heat energy dumped to the atmosphere.

5.6.6.2 Water Resources

Impacts to water resources are highly dependent on the site specific characterization. The proximity of the facility to surface waters largely determines the impact. The spent fuel storage area is expected to use about 2×10^6 gallons of water per year for operation. This is an insignificant quantity of water for most sites. Water is expected to be lost to evaporative processes in the water storage pools. There are no planned liquid effluents from this facility. Therefore, no pollutants from this facility should impact the quality of area water resources.

5.6.6.3 Waste Management

The operation of the spent fuel storage facility will generate a small volume of waste each year. The types and amounts are projected as follows:

Low-Level Radioactive Waste - A minimal amount of low-level radioactive waste is expected to be generated within this facility. The primary sources are maintenance activities and used protective clothing. Wastes generated in the water cleanup system is transferred to and handled by the reactor radwaste facility. This facility should not generate more than $10 \text{ m}^3/\text{yr}$ of low-level radioactive waste.

Hazardous Waste - A minimal amount of hazardous waste is projected to be generated by this facility. The primary source of this waste is lubricants and other chemical compounds. These wastes will be treated and disposed by an offsite vendor. The volume of this waste should be about $5 \text{ m}^3/\text{yr}$.

Solid and Sanitary Waste - Because of the limited staffing required for the facility, the generation of solid and sanitary waste is also quite limited. Solid waste should total about 25 m³/yr and sanitary waste should not exceed 1x10⁶ gal/yr.

5.7 SPENT FUEL HANDLING DEVELOPMENT REQUIREMENTS

There are no specific development requirements for the wet storage of spent fuel. This is a well demonstrated technology.

It is proposed that an ongoing testing program for re-use of spent fuel will be conducted. A few assemblies in each core may be re-used to collect data and provide demonstrated experience with re-use of fuel in the core. This will be initiated in the second year of operation. This demonstration program will include some destructive testing of a few fuel rods in a hot cell at another site to provide detailed metallurgical data. The fuel disassembly capabilities provided for fuel target removal can be utilized to remove specific fuel rods to support this testing program. Pool space is also available for additional inspection capabilities to support this demonstration program.

5.8 COST AND SCHEDULE

The cost and schedule information estimated for the ex-reactor pool has been scaled and extrapolated from the Monitored Retrievable Storage Facility Conceptual Design Report dated November 30, 1992. The conceptual design of the MRS wet pool storage option was compared to the preconceptual design of the ex-reactor pool storage facility. The detailed cost estimate for the wet storage mode was then reviewed and scaled to approximate the components, systems and structures required for the ex-reactor pool facility. Unit prices were either scaled, included at vendor price quotes when available, or extracted from the Oak Ridge National Laboratory Cost Estimate Guideline document (Ref. 5-6). The estimating guidelines were also used for unit manhour and unit labor costs. The cost estimate is summarized below. Design costs were scaled from the MRS estimate at 33 percent of construction costs.

The cost summaries are provided below:

Description	Cost (\$M)
Architectural and Engineering	18.2
Construction Management and Distributables	8.8
Construction and Procurement	55.0
Contingency (25 percent)	20.5
TOTAL	102.5

The construction schedule is estimated at 24 months.

5.0 REFERENCES

- 5-1 TSO.92.0323.0257 "Monitored Retrievable Storage Facility, Conceptual Design Report," TRW Environmental Safety Systems, Inc., November 30, 1992.**
- 5-2 ANSI/ANS 57.7 "Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)," American Nuclear Society, 1988.**
- 5-3 ANSI/ANS 57.2 "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," American Nuclear Society, 1983.**
- 5-4 ANSI/ANS 57.3 "Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants," American Nuclear Society, 1983.**
- 5-5 10CFR Ch. 1 (1-1-92 Edition) "Part 72 - Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Nuclear Regulatory Commission.**
- 5-6 ORNL/TM-10071/R3 "Cost Estimate Guidelines for Advanced Nuclear Power Technologies," J. G. Delene, C. R. Hudson II, December 1992.**

SECTION 6.0

TRITIUM PRODUCTION

In this section, the option to convert the reactor system to produce tritium has been evaluated. It is shown that tritium production goals can be met by the use of four lithium aluminate target rods per assembly. In Section 6.1, the rod design, fabrication and processing concepts are outlined. The equilibrium core design analysis for tritium production is given in Section 6.2, followed by an examination of the target rod performance and technology base in Section 6.3. The impact of tritium production on plant operations and effluents is examined in Section 6.4 while the considerations relative to conversion to a tritium core are given in Section 6.5. The support facility requirements and, cost and schedule are given in Sections 6.6 and 6.7 respectively. The development requirements for tritium production are given in Section 9.

The results of the evaluations relative to tritium clearly indicate that goal quantities of tritium can be produced with an ABWR using the existing data base for rod performance and a single cycle core, or by scaling the rod performance to higher exposures for use with an equilibrium core which should lower the fuel costs.

6.1 LITHIUM TARGET ROD DESIGN, FABRICATION AND PROCESSING CONCEPT

6.1.1 Design Approach

Review of the light water reactor lithium target developed by the Tritium Target Development Project (Ref. 6-1) (TTDP) and its performance during normal and off-normal operating conditions indicated that it is readily adaptable for application in the ABWR. Thus, the target rod design proposed is identical to the reference design developed by the TTDP except for minor dimensional changes required to interface with the standard ABWR fuel bundle design and elimination of the outer permeation barrier.

As a result of the extensive information available from the TTDP the approach used to establish the ABWR tritium production capability was:

- Scale the reference TTDP target rod design to fit the standard ABWR fuel bundle consistent with the design criteria and requirements established by the TTDP (Ref.6-2).
- Select a Li^6 enrichment and target rod placement appropriate to achieve goal tritium production with minimum impact on the ABWR core design.
- For the selected core design determine the Li^6 n,α rate, the target rod heat generation rate, etc. necessary to assess target rod performance and the annual tritium production capability.
- Confirm that contract quantities of tritium are produced and that the target design selected meets the performance requirements during normal operations and transient conditions based on test and analysis results from the TTDP.

6.1.2 Reference Design

The target rod, as illustrated in Figures 6.1-1 and 6.1-2, consists of cylindrical, annular LiAlO_2 pellets surrounded by a nickel-plated Zircaloy-4 getter to absorb and retain tritium during irradiation, thus maintaining a low tritium partial pressure in the free gas space. An inner zirconium liner, located in the central hole of the annular pellets, assists in absorption of tritium in the getter by chemically cracking T_2O released from the pellets to tritium gas. For convenience in assembly, the pellets, getters, and liners are packaged into 12.5-inch long units called "pencils". There are twelve pencils in the 163-inch long ABWR target rod.

The pencils are contained in a Type 316 stainless steel cladding tube. An aluminide coating is applied to the inner surface of the cladding to provide a barrier to permeation of tritium

Figure 6.1-1 ABWR Tritium Target Rod Cutaway

Figure 6.1-2 ABWR Tritium Target Rod

into the reactor coolant.

Table 6.1-1 summarizes the key target rod design parameters for the ABWR pre-conceptual design and the TTDP reference design.

Table 6.1-1. Target Rod Design Parameters

PARAMETER	ABWR	NPR REFERENCE
Outside Dia. (in.)	0.483	0.371
Barrier Coating	I.D. Only	I.D. & O.D.
O.D./(2 x Wall)	8.1	8.2
Rod Void Vol./LiAlO ₂ Vol.	0.8	1.3
Average GVR at EOL	77*	83
Average LHGR (kW/ft.)	0.9	0.4
Average Clad Temp. (°F)	541	610
Peak LOCA Clad Temp. (°F)	<600	1,700

* - In-reactor test data to gas-volume ratio = 116

As indicated in Table 6.1-1 the ABWR target rod is larger in diameter and has an aluminized barrier coating on the inside surface only. The ratio of diameter to wall thickness is essentially the same in both designs. The ABWR rod has a smaller void volume ratio which results in a higher gas pressure for the same gas-volume ratio. However, the clad hoop stress at end-of-life is still well below the unirradiated yield strength.

The key difference between the two designs is the peak clad temperature during the design basis loss of coolant accident (LOCA). This was the limiting factor in the NPR reference design whereas the ABWR target rod isn't subject to any significant temperature transients.

6.1.3 Fabrication

ABWR target rod fabrication involves: 1) producing the LiAlO₂ pellets, cladding, getter and

liner components, 2) assembling the pellets, liner and getter tubes into pencils, 3) loading the pencils into the cladding and 4) completing the rod final assembly and end cap welding.

The TTDP utilized commercial vendors to fabricate nickel-plated getter tubes, aluminized stainless steel cladding assemblies and the LiAlO_2 pellets utilized in the TTDP in-reactor tests. While additional work remains to qualify vendors for full scale production, a high level of confidence was established that commercial vendors could be qualified to fabricate all of the target rod components (Ref. 6-3). Thus, the basis selected for fabrication of the ABWR lithium targets is commercial vendor production of all target rod components and final assembly and inspection of the target rods in the fuel fabrication facility at the complex. Section 6.6.1 provides a description of the target rod fabrication process and the nature of the equipment and operations to be included within the complex for target rod assembly.

6.1.4 Processing

The recovery of tritium from irradiated light water reactor target rods involves puncturing the clad, heating the target rods in a vacuum furnace to release the tritium in the getters and pellets, purifying the product gases to remove helium and process impurities and separating the tritium from other hydrogen isotopes. The purification of the product gas and separation of hydrogen isotopes is well established technology within the DOE defense program. Thus, the TTDP focused on developing the process for extracting the tritium from irradiated light water reactor target rods. This activity was carried through laboratory scale testing and benchmarking of the analytical model needed to design a full scale tritium extraction process.

Section 6.6.2 describes the design and operating characteristics of a "stand alone" Target Processing Facility (RTF) that could be located within the complex. However, the recommended approach is to modify the Replacement Tritium Facility at the Savannah River Site (SRS) which has just been completed and has the capability to separate high purity tritium from the product gas extracted from the ABSR target rods. Under this

scenario the irradiated target rods would be shipped to the SRS in licensed light water reactor spent fuel casks (three to four shipments per year).

6.2 CORE DESIGN AND PERFORMANCE FOR TRITIUM PRODUCTION

An ABWR fuel bundle and core design is presented here for the production of 3.8 million curies of tritium per year. The design utilizes enriched uranium fuel rather than plutonium. The tritium is produced inside specially designed rods containing a lithium-6 target. Four of these target rods replace fuel rods in each fuel bundle in the core.

The basic concept calls for an in-reactor target residence time of about one year, after which time the target rods are removed from the fuel bundles and sent to a tritium extraction and processing facility. An alternative three year residence design is also discussed. This demands target pellet exposures substantially beyond the current experience base. Although it is believed that such exposures will ultimately prove quite feasible, alternate concepts such as target rod reconstitution and annual once-through core designs will be discussed.

The important features of the core design for the reference tritium production case are summarized in the following sections.

6.2.1 Reference Tritium Production Option Core and Fuel Design

The reference tritium production core and fuel design can produce 3.8 million curies of tritium per year in a single ABWR power plant. A summary of the important fuel cycle parameters and tritium production rates is shown in Table 6.2-1.

Table 6.2-1. Parameter Summary, Tritium Case

Number of Reactors	1
Cycle Length, EFPD	283.7
Discharge Exposure, MWd/MT	23400
Reload Batch Size	280
Average ²³⁵ U Enrichment	4.48%
Plant Capacity Factor	75%
Tritium Output, curies/yr	3.8x10 ⁶

6.2.1.1 Reference Bundle Design

The bundle design for the tritium production option contains four ⁶Li target rods and 56 UO₂ power producing rods. This design has a relatively small number of gadolinia bearing burnable poison rods since the ⁶Li target rods account for much of the neutron absorption. This bundle design closely resembles the reactivity characteristics normally associated with enriched uranium fuel. However the neutron absorption of the target rods nearly doubles the enrichment required to obtain the fuel discharge exposure.

The bundle axial and radial enrichment distribution is given in Figure 6.2-1 along with the gadolinia distribution. Values of enrichment for uranium are read in hundreds of a percent. For instance, the "160" refers to 1.60 w/o U²³⁵. For the tritium production case, natural uranium blankets are not used. This allows a maximum bundle average enrichment to overcome the ⁶Li target rod reactivity penalty and also flattens the axial power shape so as to reduce the exposure peaking in the target rods. There are a total of six pellet types used in this design, one of which is a gadolinia bearing pellet.

The infinite lattice radial power peaking is essential in determining the peak power producing rod. The distribution of relative power peaking (normalized to unity across the lattice) is shown at beginning of life for the 40 percent void case in Figure 6.2-2. The

Figure 6.2-1 Bundle Design Tritium Production

1.196	1.487	1.180	1.166	1.142	1.173	1.185	1.196
1.487		0.888	0.912	0.514	0.888		1.186
1.180	0.888	0.497	0.935	0.936	0.877	0.888	1.175
1.166	0.912	0.935			0.937	0.514	1.144
1.142	0.514	0.936			0.936	0.913	1.170
1.173	0.888	0.877	0.937	0.936	0.498	0.889	1.184
1.185		0.888	0.514	0.913	0.889		1.191
1.196	1.186	1.175	1.144	1.170	1.184	1.191	1.201

**Figure 6.2-2 Beginning of Life Lattice Power Distribution,
Tritium Case**

maximum infinite lattice relative power peaking factor is shown as a function of exposure for the 0 percent, 40 percent and 70 percent void history cases in Figure 6.2-3.

The exposure-dependent k_{∞} are given in Figure 6.2-4 for the uncontrolled lattice. This figure shows the k_{∞} for three void histories. As seen from, the design of the lattice is such as to emulate a low enrichment UO_2 lattice. This is accomplished by reducing the number and content of gadolinia rods.

6.2.1.2 Equilibrium Core Design

The equilibrium core design philosophy for the tritium production option was to simulate the reactivity distribution of a annual refueling equilibrium UO_2 core in order provide simple operation. A single fuel nuclear design was utilized in an equilibrium batch of 280 bundles. The detailed core design layout is presented in Figure 6.2-5. The numbers shown in the beginning-of-equilibrium-cycle core map represent the relative number of cycles since fuel loading. For instance, the number "1" refers to fresh fuel (loaded this cycle) and the number "4" refers to bundles which are about to start their fourth cycle.

A single nuclear design of fuel is loaded into the equilibrium cycle. The important fuel bundle parameters were summarized previously. A control cell core loading strategy that contains 37 control cells was utilized. Due to the improved hot to cold reactivity swing characteristics of the ABWR core, it was possible to design the fuel with a sufficient cold shutdown margin and still maintain sufficient hot excess reactivity.

The important parameters of the equilibrium cycle design are summarized in Table 6.2-2. Examination of the results reveals that all thermal and reactivity requirements are satisfied.

6.2.1.3 Core Thermal Margins

The critical power ratio and MAPLHGR thermal margin performance are plotted as a function of cycle exposure in Figure 6.2-6 and 6.2-7. As explained in Section 2.4.1-8,

Figure 6.2-3 Lattice Power Peaking, Tritium Case

Figure 6.2-4 Hot Uncontrolled K_{∞} , Tritium Case

Core Mass = 146.4 MT

Figure 6.2–5 Equilibrium Cycle Loading Pattern

Figure 6.2-6 Equilibrium Cycle Maximum MAPRAT vs Exposure

Figure 6.2-7 Equilibrium Cycle Maximum CPRRAT vs Exposure

operation within the MAPLHGR limit assures the mechanical integrity of the fuel rods is maintained by limiting their power output in an appropriate manner throughout their lifetime. The MAPLHGR limits imposed on this cycle with a relatively low discharge

**Table 6.2.2. Equilibrium Cycle Key Parameter Summary
Tritium Case**

Cycle Length, EFPD	283.7
Cycle Energy, GWd	1114
Cycle Exposure, MWd/MT	6903
Core Mass, MT	146.4
Reload Enrichment, w/o U-235	4.48
Reload Batch Size	280
Maximum MAPRAT	0.89
Maximum CPRRAT	0.84
Maximum LHGR, KW/ft (LHGR limit = 14.4)	12.8
MCPR (OLMCRP=1.25)	1.47
Minimum Cold Shutdown Margin	1.05
Hot Excess Reactivity at BOC	1.45

exposure are the same as the fuel licensed for up to 38 GWD/MT. These results demonstrate ample margin to core thermal limits.

6.2.1.4 Reactivity Limit Summary

The reactivity performance of the tritium production option design is summarized in Figure 6.2-8 and Figure 6.2-9. Due to the improved hot to cold reactivity swing of the ABWR N-lattice, there is sufficient cold shutdown margin; therefore, there is little or no impact of the core design from cold shutdown margin considerations.

Figure 6.2-8 Equilibrium Cycle Hot Excess Reactivity

Figure 6.2-9 Equilibrium Cycle Minimum Cold Shutdown Margin

6.2.1.5 Reactivity Coefficients

The core dynamic void coefficient and Doppler coefficients of reactivity are plotted as a function of cycle exposure in Figure 6.2-10 and Figure 6.2-11. Both the dynamic void coefficient and the Doppler coefficient are within the generic limits. No special transient or core hydraulic stability analysis are required.

6.2.1.6 Core Performance Description

The core performance characteristics as a function of exposure through the cycle are given in Figure 6.2-12 through Figure 6.2-18. The core maps in these figures show the control blade patterns in the core expressed in terms of notches (which are three-inch sections of blade) withdrawn from the top of the core. Those cells which have no numbers represent cells in which there are no blades inserted. The thermal limits and reactivity margins associated with the given exposure are noted in the summary included with each figure. As seen from these figures, all thermal and reactivity margins are met. The resulting core average power and exposure profile are also given. Since the reactor core design itself provides sufficient margins, it is not necessary to axial grade the fuel assembly to flatten or accommodate the shifts in power. However it may be advantageous to flatten the axial power shape in order to reduce the exposure peaking of the target rods.

6.2.2 Alternate Tritium Production Option Designs

The reference tritium production core design requires that the fuel bundles be resident in the core for at least three one-year cycles. Two options for inserting target rods are available with this core design:

- **Option 1: Single Cycle Target Exposure**

Fresh target rods are loaded into the bundles before each cycle of operation and unloaded and processed after each cycle.

Figure 6.2-10. Equilibrium Cycle Dynamic Void Coefficient

Figure 6.2-11. Equilibrium Cycle Doppler Coefficient

Cycle Exposure, MWD/st	0.0
Cycle Energy, MWD	0.0
Number of Full Power Days	0.0
Core Average Void Fraction	0.4466
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.307
Core Axial Power Peak	1.327
Maximum RAPLHGR	0.876
Maximum CPRRAT	0.840
Hot Excess Reactivity, %	1.42
Cold Shutdown Margin %	1.05

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 6.2–12 Equilibrium Cycle Core Data Summary at 0 MWD/st

Cycle Exposure, MWD/st	1000.0
Cycle Energy, MWD	161351
Number of Full Power Days	41.1
Core Average Void Fraction	0.446
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.290
Core Axial Power Peak	1.315
Maximum RAPLHGR	0.864
Maximum CPRRAT	0.811
Hot Excess Reactivity, %	1.50
Cold Shutdown Margin %	1.20

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 6.2–13 Equilibrium Cycle Core Data Summary at 1000 MWD/st

Cycle Exposure, MWD/st	2000.0
Cycle Energy, MWD	322702
Number of Full Power Days	82.2
Core Average Void Fraction	0.434
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.315
Core Axial Power Peak	1.292
Maximum RAPHGR	0.837
Maximum CPRRAT	0.804
Hot Excess Reactivity, %	1.57
Cold Shutdown Margin %	1.25

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 6.2-14 Equilibrium Cycle Core Data Summary at 2000 MWD/st

Cycle Exposure, MWD/st	3000.0
Cycle Energy, MWD	484054
Number of Full Power Days	123.3
Core Average Void Fraction	0.434
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.327
Core Axial Power Peak	1.269
Maximum RAPLHGR	0.821
Maximum CPRRAT	0.800
Hot Excess Reactivity, %	1.39
Cold Shutdown Margin %	1.31

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 6.2–15 Equilibrium Cycle Core Data Summary at 3000 MWD/st

Cycle Exposure, MWD/st	4000.0
Cycle Energy, MWD	645405
Number of Full Power Days	164.4
Core Average Void Fraction	0.449
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.323
Core Axial Power Peak	1.280
Maximum RAPLHGR	0.869
Maximum CPRRAT	0.790
Hot Excess Reactivity, %	1.30
Cold Shutdown Margin %	1.36

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Figure 6.2–16 Equilibrium Cycle Core Data Summary at 4000 MWD/st

Cycle Exposure, MWD/st	5000.0
Cycle Energy, MWD	806756
Number of Full Power Days	205.5
Core Average Void Fraction	0.456
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.285
Core Axial Power Peak	1.274
Maximum RAPLHGR	0.8825
Maximum CPRRAT	0.760
Hot Excess Reactivity, %	0.83
Cold Shutdown Margin	1.44

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power
 Core Axial Average Exposure, GWD/st
Figure 6.2–17 Equilibrium Cycle Core Data Summary at 5000 MWD/st

Cycle Exposure, MWD/st	6903
Cycle Energy, MWD	145702
Number of Full Power Days	283.7
Core Average Void Fraction	0.454
Core Flow, Mlb/hr	1.151
Maximum Channel Peaking	1.261
Core Axial Power Peak	1.244
Maximum RAPLHGR	0.793
Maximum CPRRAT	0.738
Hot Excess Reactivity, %	0.0
Cold Shutdown Margin	1.55

N N – Number of 3 inch increments that the control blade is withdrawn from fully inserted

Core Axial Average Relative Power
Figure 6.2-18 Equilibrium Cycle Core Data Summary at End of Cycle
 Core Axial Average Exposure, GWD/st

- **Option 2: Multi-cycle Target Exposure**

Target rods are loaded into the bundles prior to the first cycle and remain in the bundle until it's final discharge. Target rods obtain very high exposures and contain three times as much tritium and helium as Option 1.

In terms of core design and fuel cycle performance the two options are nearly equivalent. This is because the target rods are highly absorbant (nearly black) and do not deplete to an appreciable extent even with very high exposure. Thus their nuclear properties are nearly identical in both scenarios.

Option 1 has the advantage of being within the current target pellet burnup experience base. Thus it is considered the reference design for this study. Option 1 has the disadvantage that either a fresh core must be purchased each year or all bundles must be reconstituted with fresh targets between each cycle. Both prospects are likely to be expensive.

With Option 2, each year only about one third of the bundles are discharge to the refueling pool where the target rods will be removed from the bundles and sent to an off-site processing facility. Since each rod contains on average three times the amount of tritium all cost associated with manufacturing and processing them are reduced by a factor of three. However, the high exposures required for this option have not as yet been experimentally obtained. Therefore a lead test assembly program would be required prior to full scale implementation.

Table 6.2-3 shows target rod design parameters and performance results for each option.

Table 6.2-3. Tritium Core Design Options

	<u>Option 2</u>	<u>Option 1</u>
Target rod OD (in)	0.500	0.483
Target rod clad thickness (in)	0.040	0.030
Target pellet OD	0.379	0.390
Target pellet ID	0.299	0.240
⁶ Li enrichment	80%	50%
Tritium Output	36.9 MCi/yr	38.4 MCi/yr
⁶ Li n-α rates (per second per cm rod length)		
BOC Average	2.6x10 ¹³	2.7x10 ¹³
BOC Peak	4.9x10 ¹³	5.4x10 ¹³
MOC Average	2.7x10 ¹³	2.7x10 ¹³
MOC Peak	4.6x10 ¹³	5.1x10 ¹³
EOC Average	2.7x10 ¹³	2.7x10 ¹³
EOC Peak	4.6x10 ¹³	5.1x10 ¹³
³ H Inventory (Ci per rod)		
BOC Average	13359	0
BOC Peak	30441	0
MOC Average	18067	5280
MOC Peak	32958	7320
EOC Average	24067	11400
EOC Peak	38582	17040
End of Life Average	34118	11400
Target Pellet Exposure (GVR)		
End of Life Average	442	77
End of Life Peak	644	137

6.3 TARGET ROD PERFORMANCE AND TECHNOLOGY BASE

6.3.1 Normal Operations

Table 6.3-1 summarizes the key operating parameters for the average and peak target rods in the ABWR core during normal operation.

Table 6.3-1. Target Rod Operating Parameters

PARAMETER	AVERAGE ROD	PEAK ROD
Clad O.D. Temp. (°F)	541	541
Getter Temp. (°F)	585	599
Pellet I.D. Temp. (°F)	712	772
EOL Average GVR	77	112
EOL Helium Pressure (psi)	2,100	3,000
EOL Clad Hoop Stress (ksi)	14.8	21.5
EOL (H+T)/Zr in Getter	0.17	0.29
Cum. Tritium Perm. (Ci)	0.07	0.09
EOL Tritium Inventory (Ci)	11,400	17,000

The TTDP in-reactor tests operated at clad temperatures $\approx 50^\circ$ Fahrenheit higher than the ABWR target rods. However, the $n-\alpha$ reaction rates in the ABWR rods are higher than in the TTDP test rods. The net result is the ABWR rods have lower clad and getter temperatures than the in-reactor tests and comparable pellet temperatures.

The end-of-life gas-volume ratio for the ABWR rods is well within the range of the in-reactor test data.

The end-of-life helium pressure and clad stress in the average ABWR rod are comparable to the WC-1 test (1,900 vs 2,100 psi and 13.6 vs 14.8 ksi). The clad stress in the peak ABWR rod is higher but still well below the unirradiated yield strength for 316 stainless

steel. Further, the yield strength of the cladding will increase substantially during irradiation providing additional margin.

Finally, the getter loading at end-of-life, adjusted for hydrogen ingress based on in-reactor test experience, is well below the design specification limit of 0.7 and comparable to that observed in the WC-1 test.

As indicated in Section 6.1.1 the ABWR target rod was designed to remain within the criteria established by the TTDP. The above summary confirms that this goal was achieved and further that the normal operating conditions for the ABWR target rods are within the experience base established by the TTDP in-reactor tests.

6.3.2 Off-Normal and Accident Conditions

The TTDP conducted extensive analysis and testing to determine the response of the getter-barrier target design to light water reactor transient and accident conditions. While the TTDP analyses were based on a 1,250 MWe pressurized water reactor, the following fundamental conclusions related to target response also apply to the ABWR:

- Because of its low heat generation rate the target rod does not experience significant temperature increases during over power transients. For example, in PWR rod ejection accidents (Ref. 6-4) the target internal temperature was estimated to increase only 30° Fahrenheit.
- During a loss-of-coolant accident where the core is uncovered the target will be heated by thermal radiation from the surrounding hot fuel rods.
- The getter-barrier target can readily be designed to withstand a loss-of-coolant accident, including core uncover, without clad breach or bulging to an extent that would effect coolability.

- All other off-normal, transient and accident conditions addressed in a typical pressurized water reactor SAR were much less limiting in terms of target integrity than the design basis loss-of-coolant accident.

Based on the results of the TTDP the only transient or accident conditions that posed a concern to the integrity of the target cladding were those which involved core uncover. However as long as the core remained covered the target rods did not overheat because of their low heat generation rate. Since the ABWR core does not uncover during the largest loss-of-coolant accident, there is no concern for target integrity during off-normal and accident conditions.

6.3.3 Multi-Cycle Operation

As demonstrated in Section 6.2 the ABWR can produce the contract quantity of tritium by discharging all the target elements each year or one-third of them per year after a three year exposure. Since the available in-reactor performance data is limited to ≈ 1 year exposure, the full core annual discharge was selected as the reference case. This provides a highly conservative design approach that could be implemented with the first core if desired.

The one-third replacement, three year exposure mode more fully utilizes the burnup capabilities of the ABWR fuel and would be a much more economical tritium production cycle. A preliminary assessment of the potential to achieve three cycle exposure levels in the target rod indicates that it should be feasible using a slightly larger rod diameter (≈ 0.5 -inch) with higher lithium enrichment levels (≈ 80 percent) and the standard 0.040-inch pellet wall thickness. Further, the performance data necessary to support extending the target exposure levels could readily be obtained through irradiation of LTAs in any of the cores under consideration. However, since the contract quantity of tritium can be achieved within the existing target technology data base further optimization of the tritium production cycle was not pursued at this time.

The effects of lithium targets on the nuclear characteristics of the ABWR core were discussed in Section 6.2. This section addresses the loss of tritium from target rods to the coolant system, the buildup of tritium in the coolant system, and the losses of tritium to environmental pathways from the coolant. The tritium is assumed to be chemically combined with oxygen as T_2O or HTO molecules in the coolant. With this assumption, all calculations of the tritium behavior are straightforward. The assumption is easily justified, since all hydrogen present as gas in the reactor coolant system is recombined with oxygen in the off-gas treatment system. The water produced in the recombination reaction is re-injected into the reactor coolant system. Since the tritium behaves chemically like normal hydrogen (protium), there is likely to be only a very small fraction of tritium as elemental tritium in the coolant.

6.4.1**Bases for Tritium Target Release to Coolant**

The lithium targets lose tritium through the stainless steel cladding by diffusion related processes. This normal escape mechanism is controlled by the partial pressure of tritium in the target rod. There is also the possibility of a defect (hole) in the target cladding leading to depressurization of the rod.

6.4.1.1**Normal Permeation Losses**

The vast majority (99.99 percent) of the tritium produced is contained within the $LiAlO_2$ pellets and the getter. The partial pressure of free tritium in the gas space of the target rod during normal operation is only $\approx 10^{-4}$ atm. This partial pressure is determined by the dynamic balance among the process involved in release of tritium from the pellets, its absorption in the getter and liner and permeation through the clad. These processes have been modeled in the TKTARI code described in the literature (Ref. 6-5). This code, and an evolutionary upgrade used in this study, combine theory and validation experiments to give

a mechanistic calculation for the permeation release of the tritium from the targets during irradiation. For the design and irradiation parameters of the ABWR target the calculated tritium permeation (6.3 above) is 0.07Ci over the 284 day irradiation period per rod or 244Ci for the 3,488 rods in the core.

6.4.1.2 Losses from Failed Rods

No cladding failures occurred during the in-reactor testing and no failure mechanisms were identified other than fabrication flaws, external damage and gas pressure induced stress which is readily accommodated in the design. There was no mechanical interaction or chemical attack among the cladding, getter and pellets. However, the possibility of cladding breaches in the target rods cannot be dismissed.

While there is no failure experience for getter-barrier target rods, considerable data exist for the failure of fuel rods in light water reactors. The infant mortality rate for light water reactor fuel is generally accepted to be in the range of 1/10,000 rods/year (Reference 1). This number is used for the target rods with the recognition that it is conservative because of the less severe operating conditions of the target rods compared to typical LWR fuel rods. A second very conservative assumption is that 50 percent of the tritium produced leaks into the coolant from the failed rod.

Table 6.3-1 lists the tritium production for the average rod as 11,400Ci. On a statistical basis for 3,488 rods/core, an average of approximately one target rod is assumed to fail in every three core loadings and contribute 5,700Ci to the coolant.

6.4.1.3 Total Tritium Losses to the Reactor Coolant System

The total source term per cycle can be summarized as $244 + 0.3488(5,700) = 2,232\text{Ci}$. The coolant absorbs this at $2,232/284 = 7.86\text{Ci/day}$ during full power operation of 284 days per cycle.

6.4.2

Tritium Distribution in the Coolant and HVAC Systems

The water inventory into which the tritium can mix during operation includes in the reactor vessel, the feed and condensate system, the condensate storage tank, the rad waste system and the spent fuel pool. This total mass of coolant (1.2×10^7 pounds) is partially redistributed during refueling to include the dryer-separator pool. For simplicity of calculation the tritium is assumed to be "well-mixed" in this total inventory. The only difference in the power operation versus refueling periods as far as water loss is concerned is the spent fuel pool exposes 300m^2 to evaporation and the dryer separator pool and reactor cavity, which are filled during refueling, essentially doubles this evaporation area.

Evaporation from the pools is assumed to be controlled by the mass transfer coefficient for evaporation (K_e) and the pool-to-reactor building air humidity (water vapor concentrations). Using a literature value of 3000cm/hr for K_e (Ref.6-6) with the pool at $100^\circ\text{Fahrenheit}$, the evaporation losses were calculated for a reactor building HVAC flow of $100,000\text{cfm}$ with the flow assumed to be well-mixed in the building space. The inlet HVAC air was assumed to be $70^\circ\text{Fahrenheit}$ at 50 percent relative humidity. The resulting evaporation rates were $16,400\text{lb/day}$ during power operations and $31,300\text{lb/day}$ during refueling.

The ABWR is designed as a closed system with no routine liquid discharges to the environment. However, based on experience it is expected that $\approx 10,000$ gallon/day ($83,000\text{lb/day}$) of makeup water will be required to compensate for pool evaporation and miscellaneous steam leaks. Based on the pool evaporation rate indicated above the $10,000$ gal/day makeup rate implies that $\approx 66,600$ lb/day are lost through steam leakage, primarily in the turbine building, during power operation. The loss of water from the discharge of solidified rad waste is negligible in the overall tritium balance.

To account for the gradual buildup of tritium in the coolant system, an unsteady-state mass balance of tritium in the system was modeled in differential equation form and integrated for several cycles of power operation and refueling. The overall loss rate included radioactive decay but was dominated by the water loss terms. The tritium was attached to

the water in the model and no separation process was assumed during pool evaporation. The tritium inventory is plotted as a function of time in Figure 6.4-1. As can be seen the tritium reaches a repeating level in just two complete cycles. The effective half life of tritium in the reactor coolant system is only 0.324 years. The average coolant inventory over each cycle reaches a steady level of 1,010Ci or 8.1×10^{-5} Ci/lb. The model also provides a tabulation of tritium source terms for worker exposure and site dose calculations.

6.4.3 Summary of Airborne Tritium Releases

Tritium losses to the HVAC system are again assumed to be well-mixed in the building space to determine the building airborne concentrations. These concentrations are also time-averaged over both the power and refueling periods of each cycle. Table 6.4-1 summarizes the pertinent calculated results needed for the dose calculations.

6.4.4 Safety and Environmental Issues

The addition of the tritium production option to the ABWR is not projected to have a significant impact to public or worker health and safety. In fact, no impact is projected to the following resources:

- Land/Geologic Resources
- Air Resources and Noise
- Water Resources
- Land Use, Recreational, and Visual Environment
- Biotic Resources
- Cultural Resources
- Socioeconomics
- Decontamination and Decommissioning

Figure 6.4-1 Buildup of Tritium in Reactor Coolant Water System

Table 6.4.1 Summary of Tritium Concentrations and Source Terms*

Liquid Phase	1,009 Ci average cycle inventory 8.1×10^{-5} Ci/lb average cycle concentration 8.3×10^{-5} Ci/lb average @ power concentration 7.4×10^{-5} Ci/lb average refueling concentration
Gas Phase	<p>Turbine Building (TB) 1,557 Ci lost**/cycle @ power 0 Ci lost/cycle refueling 6.7×10^{-7} μCi/cc in TB @ power 0. μCi/cc in TB refueling</p> <p>Reactor Building (RB) 384 Ci lost/cycle @ power 232 Ci lost/cycle refueling 3.3×10^{-7} μCi/cc in RB @ power 6.1×10^{-7} μCi/cc in RB refueling</p>

* - Steady operation (>2 cycles) with 284 days of effective full power followed by 94 days refueling.

** - "Ci lost" are time-integrated values that do not include decay after exiting the stack.

The tritium production option may have an incremental impact on the following parameters.

6.4.4.1 Environmental

The tritium released into the reactor and turbine buildings is projected to have a chemical form of water (HTO or T₂O) and a physical form of water vapor (gas phase). The water vapor containing tritium could either be collected by the ventilation system and exhausted to the environment through the elevated plant stack or could condense on building surfaces and be collected in a liquid sump. The liquid would then be treated in the radioactive waste system with a portion being disposed as low-level radioactive waste. The balance of the liquid is recycled back into the primary coolant.

The environmental source terms for tritium at the reactor complex are:

Reactor Building during Power - 384 Ci/yr

Reactor Building during Refueling - 232 Ci/yr

Turbine Building during Power - 1557 Ci/yr

Turbine Building during Refueling - 0 Ci/yr

The annual tritium release is conservatively assumed to be the sum of the above releases, or 2,170 Ci. All tritium in the vapor phase is assumed to be collected by the building ventilation and exhausted from the plant stack. This source term produces radiological impacts that are provided in Table 6.4-2.

Table 6.4-2. Reactor Facility Incremental Tritium Impact

Individual Facility	Maximum Offsite Dose (mRem/Yr) 400 meters	Maximum Offsite Dose (mRem/Yr) 10 km
Inhalation	7.72×10^{-20}	8.21×10^{-1}
Ingestion	2.58×10^{-19}	2.74×10^{-1}
Ground Surface	0	0
Air Immersion	0	0
Total	3.36×10^{-19}	3.56×10^{-1}

6.4.4.2 Occupational Dose

During the time period between tritium release into the building vapor space and the time it is collected and exhausted, it is available for worker exposure. Steam leaks in the turbine building and water evaporation in the reactor building are the tritium source terms. Ventilation systems are the collection and loss terms. After considering these, the steady-state airborne concentrations are projected to be:

Reactor Building during Power	- Airborne concentration of $3.3 \times 10^{-7} \mu\text{Ci/cc}$
Reactor Building during Refueling	- Airborne concentration of $6.1 \times 10^{-7} \mu\text{Ci/cc}$
Turbine Building during Power	- Airborne concentration of $6.7 \times 10^{-7} \mu\text{Ci/cc}$
Turbine Building during Refueling	- Airborne concentration of $0.0 \mu\text{Ci/cc}$

The time-weighted concentration levels are:

$$.75(3.3 \times 10^{-7} \mu\text{Ci/cc}) + .25(6.1 \times 10^{-7} \mu\text{Ci/cc}) = 4.0 \times 10^{-7} \mu\text{Ci/cc for the reactor bldg.}$$

$$.75(6.7 \times 10^{-7} \mu\text{Ci/cc}) + .25(0.0 \mu\text{Ci/cc}) = 5.0 \times 10^{-7} \mu\text{Ci/cc for the turbine bldg.}$$

These levels are an average over the 378 day cycle and are also an average over the room volumes. It is possible that portions of the reactor and turbine buildings could experience concentration levels higher than the periodic averages of 4.0×10^{-7} and $5.0 \times 10^{-7} \mu\text{Ci/cc}$, respectively. DOE 5480.11 requires worker respiratory protection for tritium concentrations exceeding $2 \times 10^{-6} \mu\text{Ci/cc}$ (10 percent of the derived air concentration). For tritium, this means a positive-pressure protective device because air filtration devices are not effective for tritium. However, the average concentration in the turbine building is projected to be 25 percent of that concentration requiring respiratory protection.

6.4.4.3 Transportation

The addition of tritium production to the reactor complex will cause an increase in the transportation of radioactive materials. This is true whether the tritium processing facility is located within or external to the complex since the product and the wastes would not remain within the complex. However, shipping the target rods which contain tritium in a solid form should reduce the risk of accidental release compared to shipment as a gas.

6.4.4.4 Waste Management

Low-Level Radioactive Waste - The tritium production option would not result in the

generation of a significant additional volume of low-level radioactive waste. The additional volume is projected to be approximately 10 m³/yr. This additional volume is primarily from activities involved in removing target rods from fuel bundles after irradiation.

Hazardous Waste - The tritium production option would not result in the generation of additional hazardous waste in the reactor complex. Waste generated in the manufacture of the target rods is minimal as described in Section 6.6. The waste generated in the processing of the target rods is also summarized in Section 6.6 but is not included here since it is not within the complex.

Solid and Sanitary Waste - An insignificant amount of additional solid and sanitary waste would be generated by the addition of the tritium production option.

6.5 CONVERSION TO TRITIUM PRODUCTION

The reference tritium production case is based on utilizing a standard, low enriched uranium core containing four lithium target rods per bundle. While tritium production can readily be incorporated in plutonium disposition cores it was not considered practical in this limited study due to the large number of core design permutations under evaluation and the secondary importance assigned by DOE to the tritium production mission. Further, DOE would need to specify a number of details related to the timing and duration of tritium production relative to the available plutonium supply and the nature of the plutonium mission (e.g., spiking, spent fuel or burn) in order to make such an evaluation meaningful.

The key criteria utilized in selecting the reference tritium production mode were:

- Stay within the existing technology base.
- Provide maximum flexibility to initiate tritium production with minimum lead time and as soon as possible after start of operations.
- Minimize the construction of facilities that are not required unless tritium is produced.

- Provide the capability to optimize tritium production to reduce costs and/or include it with the plutonium disposition mode if desired.

A low enriched uranium core operating on an annual cycle meets all of these criteria. It is well within the existing fuel and target technology base. It can be fabricated by a commercial fuel vendor and be available at start up or stored until needed. In the ABWR a full core refueling can be completed during a normal outage (fuel handling time ≈ 10 days). Thus, conversion to tritium production can be accomplished very quickly regardless of the previous operating mode.

The proposed target fabrication approach, (vendor production of target rod components and assembly in the mixed-oxide fuel fabrication plant) eliminates the need for a separate target fabrication facility and staff. Further, modification of the replacement tritium facility at SRS eliminates the need to construct and staff a separate target processing facility.

Finally, the standard ABWR uranium fuel has the capability to support an optimized tritium production cycle as discussed previously.

If the tritium production requirement is more a "capability" than a "hard demand" for material then a lower cost but less timely option is available. This approach is based on "fully developed options" that have low up-front costs but can be implemented with a high degree of cost, schedule and technical certainty. For example the tritium production core could be designed, the reload analysis approved and the procurement of commercial UO_2 fuel negotiated and placed on hold. In addition, the modifications to the replacement tritium facility at SRS could be designed, procurement packages prepared and any long lead items purchased. This option would require about three years to tritium availability but has minimum up front investment.

6.6 TRITIUM SUPPORT FACILITIES

6.6.1 Target Fabrication Facility

A simplified process flow diagram for target rod fabrication is shown in Figure 6.6-1. It involves 1) production of LiAlO_2 pellets, cladding, getter and liner components, 2) assembling the pellets, liner and getter tubes into pencils, 3) loading the pencils into the cladding and 4) completing the rod final assembly and end cap welding.

The pellet fabrication process begins with enriched lithium carbonate and aluminum oxide which are blended and spray dried. After drying, the powder is calcined in a furnace and a dry binder added using a blender. Multicavity hydraulic presses press the pellets and a belt-feed continuous furnace sinters the pellets. The pellets are inspected and sampled and centerless ground to final size. The pellets are then assembled with getter tubes and liners into pencils and stored until required for loading into the cladding tube assemblies.

The barrier-coated 316 stainless steel cladding is fabricated by a pack aluminizing process in which a mixture of aluminum alloy powder, alumina powder, and an ammonium chloride activator is blended in a predetermined mixture. The prepared pack is loaded into a clad tube which has the lower end cap already welded in place. A group of packed clad tubes is then placed in a retort and heated under conditions that produce the desired barrier coating. Eddy current and air-gauge inspection techniques are used to assure uniformity of the barrier coating.

The nickel-plated zircaloy getter tubes are fabricated using commercial electro-plating technology followed by vacuum annealing. The inner liner is a roll-formed tubular shape of 0.004-inch thick zirconium metal. Metal shapes of this geometry are routinely fabricated in industry.

The final rod assembly consists of loading subassemblies in air on a work bench. Loaded rods are inserted manually into a portable evacuation/drying magazine. Each magazine

Figure 6.6-1. Target Fabrication Process Flow Diagram

holds a one-day throughput. The internal atmosphere is replaced with dry, high purity helium. Each magazine is allowed to soak until the components are within acceptable limits for moisture and oxygen. The magazine would then be connected to a welding enclosure and flushed with high purity helium. The springs and top end caps are inserted into the tubes and the end caps welded. When all the end welds are complete the magazine is uncoupled from the welding enclosure and the rods removed and placed in the helium leak-test chamber. Following final inspection the rods are placed in containers and moved to the bundle assembly area where target rods, fuel rods and associated hardware are assembled into a fuel bundle for shipment to the reactor.

As discussed previously a high level of confidence has been established that commercial vendors can be qualified to fabricate all of the target rod components. Thus, the only target fabrication operations to be included within the complex are:

- Target rod component inspection and storage
- Pencil assembly
- Rod loading
- Helium back fill
- End closure assembly and welding
- Final rod inspection and loading in the fuel bundle

Based on the design developed for the LWNPR support facilities (Ref.6-7) $\approx 15,000$ target rods per year can be fabricated in an industrial-type, single-story, 15 meters by 30 m building. The ABWR would require only 25 percent of this throughput. However, the need to handle four meter rods horizontally requires ≈ 10 meters of working length. Allowing space for component storage, inspection, helium back fill and welding equipment a total of $\approx 300 \text{ m}^2$ should be more than adequate for target rod assembly.

The equipment required for target rod assembly includes:

- Two five meter long assembly tables

- Five evacuation/drying magazines
- Helium fill station equipped for pump-down and back-fill
- Helium glove box with two automated GTAW machines
- Helium leak testing and weld radiography equipment
- Miscellaneous inspection tools

The target fabrication operation could be located in a stand alone building or be incorporated in the fuel fabrication facility. The later option was selected because:

- Target rod assembly involves the same type of activities required for fuel rod assembly hence the operating staff should be shared.
- Co-location is expected to be less expensive due to shared support capabilities such as security, radiography, film processing, etc.

The target rod assembly operation is a normal industrial activity. There are no special safety, licensing, worker health and safety, effluent or waste issues associated with this activity. The only wastes generated that are potentially regulated under the Resource Conservation and Recovery Act (RCRA) are the spent radiography film processing chemicals. These wastes would be collected and packaged as required for disposal by a commercial chemical waste contractor. Finally, sharing radiography film processing and other laboratory type facilities with the fuel fabrication operation provides the opportunity to minimize the quantity of chemical wastes generated.

The LWNPR fuel and target fabrication facilities, which were designed for an annual throughput of $\approx 55,000$ fuel rods and $\approx 15,000$ target rods, generated the following quantity of film processing wastes (see Table 6.1-1).

This is the same order of magnitude of fuel rod fabrication required for a number of the plutonium disposition scenarios. However, in the tritium production mode the quantities would be much less.

Table 6.6.1. Film Processing Chemical Wastes (kg/yr)

Constituent	Fuel Fabrication	Target Fabrication
Potassium Sulfite	235	50
Hydroquinone	95	20
Potassium Hydroxide	45	10
Acetic Acid	34	7
Aldehyde Bisulfate	30	6
Glutaraldehyde	12	3
Ammonium Thiosulfate	780	180
Sodium Acetate	160	35
Sulfuric Acid	45	10
Ammonium Sulfate	45	10
Silver Oxide	28	7
Detergents	5	2

The only safeguards and security associated with target fabrication is related to the enriched lithium pellets and protection of any classified information related to the fabrication process. The protection and accountability requirements associated with enriched lithium which is classified as "Other Nuclear Materials" (Category 4, Attractiveness Level E) under DOE Order 5533.3a, are less stringent than fissile material and are similar to those associated with precious metals in the DOE system. Co-location of the target and fuel fabrication activities assures adequate protection of the enriched lithium and classified information and provides the infrastructure necessary to support the accountability requirements for ONM.

6.6.2 Target Pod Processing

The target processing facility receives irradiated lithium targets from the reactor, extracts and purifies the tritium and loads it into containers for off-site shipment. The facility and process are designed to minimize tritium releases to the environment and to minimize the

generation of tritiated waste.

The following description applies to a "stand alone" target processing facility based on the design developed for the LWNPR. However, the recommended approach is to modify the replacement tritium facility at SRS rather than build a new target processing facility within the complex.

6.6.2.1 Facility Description

The target processing facility consists of two attached buildings. One houses the processes, laboratory, and other activities associated with handling tritium and other radioactive materials. The other contains nonradioactive support activities, including offices, lunch room, nonradioactive maintenance, and nonradioactive ventilation systems.

The process building consists of a reinforced concrete structure constructed to withstand the design basis earthquake, tornado and wind loadings. The process building includes a hot cell for receiving, storing, and processing irradiated target rods.

Most of the process operations and laboratory analyses are carried out in gloveboxes to assure tritium containment. The hot cell contains two vacuum furnaces for extracting tritium from the irradiated targets. The furnaces are separated from the rest of the hot cell by airlocks.

The support building is a single-story structure of conventional construction.

6.6.2.2 Design Considerations

Most of the process lines, valves, vessels, pumps, and tanks are constructed of 304 or 304L stainless steel. Type 316 stainless steel may be used in the furnace but is not considered necessary in the remainder of the process system. Exceptions to the use of stainless steel are the palladium/silver diffuser tubes, stellite valve seats, and the aluminum thermal cycling

absorption process (TCAP) column. An alternate design of a stainless steel TCAP unit is available and can be considered in later design stages.

The stainless steel system will be designed to meet appropriate DOE Order 6430.1A requirements. Generally, piping and vessels will be designed for 100 lbf/in² (gage) internal pressure at an appropriate temperature.

The process system, including the gloveboxes, will be designed to withstand the design basis earthquake unless it can be shown that portions of the system can fail without significant adverse effects to the workers, off-site population, or the environment.

The process will be controlled by a distributed control system from a central control room. The system features redundant electronic equipment and data highways (cabling) to transmit process data, equipment status (valve positions, etc.) and control commands to and from the control consoles. The control system will also monitor instrumentation for failure and will alarm a failure. Alarms and process status will also be displayed locally, but the process will be controlled only from the central control room.

Control can be either by "automatic" sequencing or by manual, step-wise sequencing. Software programs are provide for each process unit operations. These can be displayed for easy reference for manual operation or will be followed by the control computer in the "automatic" operating mode.

The process is controlled within specified process limits. If the process conditions move outside the limits, the computer control system will alarm the condition and take automatic action if required. The distributed control system will be backed up by independent hard-wired controls for critical safety limits.

The process system is designed to shut down if power is lost. Automatic valves will shut, heaters and pumps will turn off, and hydride beds are protected from overheating due to tritium decay by loss-of-coolant tanks. These tanks are not insulated and will dissipate

tritium decay heat to the glovebox atmosphere at a limiting temperature, which limits the pressure to the loss-of-coolant design value.

6.6.2.3 Process Description

The targets are heated in the extraction furnace to drive off the tritium. This gas is collected and fed through a heated decomposer bed to remove reactive impurities. The remaining gas is passed through another bed to separate inert gas from hydrogen isotopes. The tritium is separated from the other hydrogen isotopes in the TCAP unit. Process off-gas streams and the blanket gas from the gloveboxes are treated to recover tritium before release via the stack. A simplified flow diagram for the process is shown in Figure 6.6-2

6.6.2.4 Extraction Furnace System

Target rods are received from the reactor in a standard light water reactor spent fuel shipping cask. During shipment the target rods are supported in a handling container that has the same external configuration as the ABWR fuel bundle. Each container holds 49 target rods and there are 18 to 24 containers in the shipping cask depending on the cask used.

The target rods are unloaded from the handling containers and placed in magazines which support and position them within the extraction furnace. The magazines are then moved to the furnace cell through an "airlock" port which is designed to be purged or evacuated and back filled with nitrogen to remove most of the oxygen.

After the magazine is loaded into the furnace the furnace is closed and leak checked and the vacuum pumps are turned on with the flow path to the furnace evacuation hold tank. The rods are then punctured by lasers through optical ports in the furnace wall and the furnace heated to operating temperature.

Figure 6.6-2. Tritium Extraction Process Block Diagram

After extraction is complete, the hold tank is valved off and an argon cooling flow admitted to the furnace. When the temperature of the target rods is below 50 Celsius, the puncture holes are sealed by de-focusing the lasers and the argon is evacuated to the off-gas treatment system. The furnace is then back filled with nitrogen to cell pressure, opened, and the magazine of spent target rods replaced with the next batch to be processed. The spent target rods are packaged for disposal and the magazine is reused for subsequent batches.

The TTDP (Ref.6-8) confirmed the feasibility of the laser puncture-vacuum furnace extraction method and benchmarked the analytical model for design of a full scale extraction system. Model predictions indicate that a single furnace with $\approx 1 \text{ ft}^2$ crosssectional area operating at a 70 percent utilization factor can process the annual discharge of target rods. However, for redundancy and surge capability it is proposed to install two extraction furnaces.

6.6.2.5 Gas Separation

The process gas from the extraction hold tank is first pumped through a heated uranium bed to decompose the oxides of tritium and hydrogen. An in-line moisture meter, capable of detecting a few tens of ppm water vapor, is used to monitor performance of the decomposer. When the bed is depleted, it is removed and replaced.

Gas from the decomposer is cooled and pumped through the primary uranium hydride bed to collect the hydrogen isotopes. The effluent gases are sent through a palladium/silver sacrificial bed to remove impurities that could damage the diffuser then pumped through a multi-stage diffuser to recover the remaining hydrogen isotopes.

The gases that flow through the diffuser tubes, primarily helium, are collected in one of two tanks which can be sampled to determine tritium content. These gases are sent to the off-gas cleanup system where they are further treated, if necessary, or sent to stack discharge. The hydrogen isotopes, which diffuse through the tubes, are collected directly on the secondary uranium hydride beds.

The primary and secondary hydride beds are then heated to release the stored hydrogen isotopes which are pumped to hydride storage beds in the gas collection system.

6.6.2.6 Gas Collection System

The gas collection system serves as a way-station for tritium-containing gases that must be sent to the isotopic separation system. The major sources of hydrogen isotopes are the primary and secondary hydride beds in the gas separation system. Other sources of hydrogen isotopes include the hydride beds in the off-gas treatment and the load-out system. LOC tanks are provided for all hydride storage beds.

The gas collection system is also used when evacuating the process system for maintenance. If these gases contain hydrogen oxide or other gases which need to be decomposed they are pumped to a uranium bed decomposer and then to a cooled hydride bed. If the stream contains other gases, it will be sent to the multi-stage diffuser in the gas separation system.

6.6.2.7 Isotopic Separation System

Hydrogen isotopes stored in the gas collection system are released by heating the hydride beds then pumped to the thermal cycling absorption process (TCAP) feed tank. The TCAP cycles the gas between two beds of palladium-coated kieselguhr. One bed is heated to drive off the gases while the other is cooled to promote gas absorption. The palladium-coated kieselguhr preferentially retains the tritium and thus separates it from the other hydrogen isotopes.

A second TCAP unit is provided as backup to the first unit and to separate hydrogen isotopes from gases recovered from other parts of the process where the ratio of tritium to protium (^1H) may be substantially different than in the main process gas.

When the pressure in the TCAP product tanks reaches the desired level, the gas is assayed and pumped to the load-out tanks. The raffinate material is sent to the off-gas cleanup

system for removal of residual tritium and is then discharged to the stack via the purge cleanup system.

6.6.2.8 Product Load-Out System

Product is collected in one of two tanks in the load-out system. The other tank is used as a feed tank for the load-out pumps. Two load-out positions are provided. However, more may be added if needed.

To make a transfer a canister is placed the load-out glovebox, connected, and the load-out line and canister evacuated to the off-gas cleanup system. After satisfactory evacuation and leak checking tritium is pumped into the canister. The transfer canister is then removed from the glove box, checked for contamination, and sent to a calorimeter for a precise measurement of tritium content. After the canister is removed, the loading line is evacuated to the gas collection system.

6.6.2.9 Off Gas Treatment

A schematic of the process ventilation and off-gas systems is shown in Figure 6.6-3. As described previously portions of the process system are periodically evacuated to the off-gas treatment system. The gases involved include pure tritium and mixtures of argon and nitrogen with hydrogen isotopes.

Elemental tritium is sent directly to a uranium hydride bed. Gases containing oxides of hydrogen are sent through a uranium decomposer. Mixtures of hydrogen isotopes with argon or nitrogen are sent to a multi-stage diffuser unit. As described previously, the hydrogen isotopes removed are collected on a hydride bed and periodically transferred to the gas collection system.

The other gases are collected in two tanks and analyzed for tritium contamination. If acceptable, the gases are sent to the purge cleanup system and discharged to the stack. If

LEGEND

BACKUP SYSTEM - - - -

Figure 6.6-3 Ventilation and Offgas Cleanup System

tritium concentration is high, the gas is recycled through the diffuser.

6.6.2.10 Blanket Gas Cleanup System

There are two blanket gas cleanup loops, one serving the gloveboxes and the other the extraction furnace cell. These are backed up by a reserve loop which is connected individually to each glovebox and the cell. All three loops operate similarly. All of the gloveboxes use nitrogen as a blanket gas.

Nitrogen blanket gas flows from the gloveboxes to a header. Two blowers, one operating and the other on stand-by, are provided in each cleanup loop. Hydrogen and dry air are bled into the loop downstream of the blowers to enhance the oxidation reaction in the catalyst bed. The gas is then heated and passed through a split catalyst bed which oxidizes hydrogen isotopes on a palladium catalyst and removes excess oxygen on a copper catalyst. The gases are then cooled and sent to zeolite beds, containing 3 A° molecular sieve material which absorbs the water. The copper catalyst is regenerated as needed by passing hot hydrogen over the bed to convert copper oxide to copper and water.

Two of the three zeolite beds are operated in series or parallel depending on the amount of cleanup desired per pass and the gas flow rate. The third zeolite bed is maintained in reserve or is being regenerated.

Normally the tritium concentration in the blanket gas is well below $0.1 \mu\text{Ci}/\text{cm}^3$. However, upon indication of high tritium concentration ($>1 \mu\text{Ci}/\text{cm}^3$) the subject glovebox would be isolated to prevent the spread of contamination and connected to the reserve cleanup loop.

6.6.2.11 Zeolite Bed Regeneration System

Zeolite beds are regenerated by heating and passing the vapors through one of two uranium bed decomposers. The off-gases from the decomposer are collected in tanks and routed to the purge cleanup system, back to the blanket cleanup system, or to the off-gas treatment

system depending on tritium concentration.

6.6.2.12 Purge Gas Cleanup System

Purge gases from the process, the blanket cleanup system, gas bubblers, airlocks, etc. are sent through the purge gas cleanup system before discharging to the stack.

The system consists of redundant pumps which feed the gases into two zeolite beds containing 4A° molecular sieve material. This results in at least a factor of 10 reduction in the tritium concentration before the gases are discharged to the stack. A third zeolite bed is provided to permit regenerating one bed while two are operating. The moisture and tritium concentration of the discharged gases are measured in-line. Stack monitors provide flow and radioactivity (including tritium) readings of the exit gases.

6.6.2.13 Maintenance Considerations

Much of the smaller process equipment and instrumentation are maintained through use of gloves in the gloveboxes. A glovebox requiring maintenance is isolated and the blanket gas is sent to the reserve blanket cleanup loop to reduce tritium concentration to well below $0.1 \mu\text{Ci}/\text{cm}^3$. The system to be maintained is first isolated and evacuated to the gas collection system. It is then back flushed with argon and evacuated to the off-gas treatment system. The process system is then opened and the equipment or instrument replaced.

For larger equipment which requires open glovebox work the glovebox is isolated and the blanket gas cleaned to well below $0.1 \mu\text{Ci}/\text{cm}^3$. A greenhouse is built around the glovebox panel to be opened. The process system is isolated, evacuated to the gas collection system, back flushed with argon, and again evacuated to the off-gas treatment system. Personnel are required to wear plastic suits with a fresh air supply when opening or working on open gloveboxes. After the glovebox panel is removed, an air flow velocity of 200 ft/min is established at the opening into the glovebox and out the airlock and airlock hood. Savannah River experience has shown that outflow of tritium can be prevented at air

velocities above 150 ft/min.

6.6.2.14 Radioactive Particulate Releases

No particulate radioactive releases are expected from the target processing facility. The nature of the materials handled plus the thorough treatment of off-gas streams will assure that no radioactive particulates will be released.

6.6.2.15 Gaseous Radioactive Releases

The only gaseous radioactive release from the target processing facility is expected to be tritium. Tritium will be released as elemental, T_2 or HT, or as tritiated water, T_2O or HTO. The total estimated release of tritium is less than 1,000 Ci/yr. Less than 10 Ci of the release will be as water. To meet DCG at the point of release (end of stack), the release as water must be less than 150 Ci/yr for stack flow of 100,000 ft³/min. As elemental tritium, the release can be as high as 2×10^7 Ci/yr.

6.6.2.16 Nonradioactive Gaseous Releases

The gases released via the stack will be air or gases from the process. The process gases include argon, helium, and nitrogen. These process gases will represent less than 1 percent of the total stack flow.

6.6.2.17 Liquid Waste Generation

No radioactive liquid wastes are released from the target processing facility. All radioactive liquids generated by the process will be solidified in the plant and handled as solid waste. Nonradioactive liquid wastes consist of non-process water, non-process steam condensate, and sanitary wastes. The non-process water is estimated at 3 Mgal/yr. The sanitary waste is estimated at 1 Mgal/yr.

6.6.2.18 Solid Waste Generation

Both radioactive and nonradioactive solid wastes are generated. All of the radioactive solid waste contains some tritium. Only the spent target rods and spent uranium beds contain significant radioactivity from radioactive isotopes other than tritium. The nonradioactive solid waste will consist of miscellaneous trash and scrap materials. The estimated radioactive waste volumes are presented in Table 6.6-2.

6.6.2.19 RCRA and Mixed Waste Generation

No RCRA waste or mixed waste has been identified.

6.6.2.20 Plant Operation

The materials used to operate the target processing facility are discussed below. The quantities are rough, order-of-magnitude estimates of annual requirements.

Electricity, steam, and other fuel will be used to operate the target processing facility. The estimate of annual requirements is contained in Table 6.6-3 .

The chemicals required to operate the target processing facility consist of material used in processing and off-gas treatment equipment. Argon and nitrogen are needed to support the process and the process laboratory. Liquid nitrogen is also used for cooling. An estimated 10,000 standard ft³ of argon is used per year. An estimated 250,000 L of liquid nitrogen is used per year.

The estimated labor force to operate the target processing facility is based on a 5 d/wk, 3 shift/d operation. Total workers are about 130. Radiation zone workers are about 75 of that total. The annual operating cost of the target processing facility is estimated to be \$20 million.

**Table 6.6-2. Estimated Annual Radioactive Solid Waste
Volumes From Tritium Target Processing**

Waste Description	Estimated Volume For Disposal	Estimate Tritium Content (Ci)
Solidified liquids:		
Low tritiated	125 M ³	2,000
Intermediate	5 M ³	50,000
High tritiated	<5 M ³	
Catalyst/zeolite	(250 kg)	12,000
Spent uranium beds	(500 kg)	<5,000
Boxed trash	150 M ³	500
Miscellaneous	<1 M ³	2,000
Spent targets	8 M ³	
Spent Targets		Other Isotopes (Ci)
Total		<3.0x10 ⁶
14 _C		2.0x10 ¹
51 _{Cr}		1.5x10 ⁶
54 _{Mn}		1.0x10 ⁵
55 _{Fe}		4.5x10 ⁵
58 _{Co}		3.0x10 ⁵
60 _{Co}		1.5x10 ⁵
63 _{Ni}		1.5x10 ⁴
95 _{ZrNb}		2.5x10 ⁵

Table 6.6.3. Annual Energy Use Estimates for Tritium Target Processing

Electricity	2,000,000 kWh
Steam	10,000,000 lb
Diesel/gasoline	10,000 gal
Natural gas	none

Water in the target processing facility is primarily used for non-process purposes. Sanitary water use is about 1 Mgal/yr. Another 3 Mgal water is used to meet other needs.

6.7 COST AND SCHEDULE

The basis for the cost of converting to tritium production is as follows:

- The reference configuration for tritium production is:
 - Low enriched UO_2 fuel for the first core
 - Target assembly in the MOX fabrication plant
 - Target processing in the modified replacement tritium facility at SRS
- No changes to the reactor (other than the core) or the ex-reactor pool are necessary.
- Reactor operating costs are the same for the tritium production or plutonium disposition modes and are not included as a "tritium conversion" cost.
- Conversion to tritium production can be completed during a normal refueling outage.
- Fuel fabrication costs beyond the first core are not charged to tritium production on the basis that one cycle (378 days) provides sufficient time to change the MOX fuel production to a support tritium production mission.
- Modifications to the replacement tritium facility and the incremental operating costs associated with these modifications are considered a tritium conversion cost. However, the normal operating cost of the replacement tritium facility is not included on the basis that it would already be operating for processing weapons returns.

6.7.1 Capital and Pre-Operational Costs

Target Assembly - The 300 m² space required for target rod assembly was included in the cost of the MOX fabrication plant. Target assembly would also share radiography and laboratory facilities with the fuel fabrication activity. The equipment required for target assembly is minimal and was described in Section 6.6. The estimated cost of this equipment is \$200K plus \$50K for installation and check out. The only pre-operational costs are associated with procedure preparation and training which are estimated to be three months at full staffing level or \$180K.

Target Processing - Section 6.6 described the process required to extract tritium from the ABWR target rods and perform the necessary purification and isotopic separations to obtain the desired product quality. It also included a description of the support systems needed to minimize worker exposure and environmental releases. Except for the head end extraction operation, all of these process capabilities are included in the replacement tritium facility which has just been completed at SRS. The replacement tritium facility also includes a 10x30m expansion space immediately adjacent to its main processing area.

Since the gas purification and isotopic separation systems plus their associated support systems (blanket gas, purge gas, etc) represent the majority of the cost of a LWR target processing facility, the recommended approach is to add target extraction capability to the replacement tritium facility rather than build another new complete facility.

Figures 6.7-1 and 6.7-2 show the plan view and elevation of the target extraction facility that would be located adjacent to the expansion area on the north side of the replacement tritium facility. As indicated the building consists of a 16x18x9m high steel framed structure enclosing a 9x9x5.5m high hot cell and a rail car bay for receipt of standard LWR spent fuel casks. The rail car bay is equipped with a 100 ton crane and a below-grade cask transfer tunnel that provides access to the top loading spent fuel casks through the hot cell floor.

FIGURE 6.7 - 1. TRITIUM EXTRACTION CELL - PLAN VIEW

FIGURE 6.7 - 2. TRITIUM EXTRACTION CELL - ELEVATION

The hot cell is divided into an air-cell and a nitrogen-cell by a metal partition. The partition is less than full height with a closed top making the furnace portion of the cell a sealed nitrogen-filled box. This permits the in-cell crane to operate over the nitrogen box which is equipped with access ports for furnace removal. The nitrogen box can also be equipped with a wall or ceiling mounted light duty electro-mechanical manipulator if needed. The partition also contains an air lock for transfer of furnace magazines containing target rods from the air cell to the nitrogen cell and back.

The process gas, blanket gas and the argon purge streams described in Section 6.6 would be connected to the corresponding replacement tritium facility gas systems for processing and tritium recovery. The extraction furnace system cost includes the valves, gas receiver tanks and vacuum pumps which would be located in the replacement tritium facility expansion area.

The air cell includes storage holes for 125 percent of the annual reactor discharge of target rods, an unloading station for the transfer cans used to ship target rods from the complex to SRS and a furnace magazine loading station. The nitrogen cell includes two vacuum furnaces, their associated support equipment and the transfer mechanism for loading and unloading the furnaces.

6.7.2 Operating Costs

The operating costs for tritium production includes:

- Reactor fuel supply (first core only).
- Target rod fabrication (material and labor).
- Target rod removal in spent fuel pool.
- Target rod shipping.
- Tritium extraction (material and labor).
- Waste disposal.

The cost of low enriched UO_2 fuel, including shipping, is \$105 million. The cost of target rod fabrication based on the TTDP estimates (Ref.6-9) is \$7.7million/year. The cost of the additional staff required to remove the target rods from the fuel bundles and prepare them for shipment is between \$760,000 and \$1 million depending on whether the starting point is the spiking or spent fuel option. The estimate for target rod shipping (\$160,000/year) is based on vendor information for leasing an IF-300 cask. The cost of waste disposal was based on the estimated volumes listed in Section 6.6

Table 6.7-1 summarizes the costs for conversion to tritium production based on the reference concept described above. The acquisition cost for a stand alone target processing capability based on the TTDP in 1/93 dollars is $\approx$ \$200 million. While a stand alone target processing facility is somewhat less than the combined cost of the replacement tritium facility (\$230 million) and the hot cell addition (\$41 million) it is almost five times the cost of the hot cell addition alone.

6.7.3 Schedule

The target assembly area will be completed as part of the mixed-oxide fuel fabrication plant. The hot cell addition to the Replacement Tritium Facility at Savannah River site is not a completed structure and can readily be completed by the time the reactor is operational leaving a year for start up testing and the operational readiness review.

Table 6.7-1. Tritium Conversion Costs
(\$ Thousands 1993)

Item	Target Fabrication	Target Process	Reactor/ Spent Fuel Pool
ACQUISITION COSTS			
Pre-Operational	180	6,000	
Design & Procurement	200	21,200	
Construction Direct	50	3,100	
Construction Indirect	-	2,600	
Contingency	100	6,100	
Project Management	-	2,000	
Subtotal	530	41,000	
OPERATIONS COST - Per Year			
Process Materials	7,000	300	
Labor	700	1,100	690 to 910
Support Materials	100	200	70 to 90
Utilities	-	50	
Target Rod Shipping	-	160	
Subtotal	7,800	1,810	760 to 1,000
OTHER COSTS - One Time			
Target Shipping Cans	-	500	
First Core	-	-	105,000
Decontamination & Decommissioning	-	3,000	
Subtotal	-	3,500	105,000

6.0 REFERENCES

- 6-1 PNL-8142, "Tritium Target Development Project Executive Summary Topical Report", W. J. Apley, September 1992.
- 6-2 WHC-SP-0840, "Topical Report: NPLWR Tritium Target Design", J.W. Weber, September 1992.
- 6-3 PNL-8128, "NPLWR Tritium Target Fabrication Topical Report - Volume 1", J.P. Pilger, et. al, September 1992.
- 6-4 RSA-0101, "Assessment of 10 Percent Core Performance During Reactivity Insertion Accidents Through End-of-Life", B.E. Schmidt, et. al, Pacific Northwest Laboratories, October 1991.
- 6-5 WHC-SP-0684, "TKTARI, A Computer Code for Predicting Tritium Target Rod Performance," D.R. Wilson, 1991
- 6-6 "Handbook of Chemical Property Estimation Methods," W.J. Lyman, W.F. Reehl, and D.H. Roseblatt, 1992.
- 6-7 WHC-EP-0261, "Light Water Reactor Support Facilities Description - New Production Reactor," A.B. Carlson, et.al. June 1989
- 6-8 PN-8127, "A Predictive Model for LWR Tritium Extraction", R.M. Burnside, et.al, September 1992
- 6-9 WHC-EP-0483, "WNP-1 NPR Option June 1992 Update Estimate of Cost & Schedule," November 1991

7.0 SAFEGUARDS

7.1 LONG-TERM SAFEGUARDING

In the context of the present study, it is necessary to distinguish between two related but different definitions that can be ascribed to the word "Safeguards." In one case, we are referring to the permanent long-term disposition for plutonium derived from dismantling the weapons. Here, we are concerned with the relative efficacies of different disposition options for long-term safeguarding of plutonium. In the second case, we are referring to the safeguards provisions that should be implemented during the various stages in a given disposition process to prevent theft or diversion. Although, much of this section is concerned with this second definition of safeguards, it is useful to briefly examine the long-term disposition advocated in this study.

7.1.1 Permanent, Long-Term Safeguarding of Plutonium from Dismantled Weapons

Permanently safeguarding the plutonium from weapons is the paramount reason for this study. Dispositioning the plutonium as irradiated fuel assemblies by using this material as fuel in a light water reactor represents one of the most permanent, long-term, readily implemented means of safeguarding this material.

The literature on safeguards, both at the system level and in its functional details, is extremely large (Ref.7.1-7.3). The conclusions of the Office of Technology Assessment (Ref.7.1) is of particular interest in this regard. This study concludes that massive spiking with a material such as radioactive cobalt is neither cost effective nor useful in the restraint of proliferation; that the concept of 'denaturing' plutonium where Pu^{239} is converted into higher atomic mass plutonium isotopes is fallacious; that storage and transport in dilute mixed-oxide form does present a certain level of resistance as it requires time-consuming chemical separation activities to reclaim the plutonium. The report advocates real-time material

control and accountability, citing work done in this area at Los Alamos National Laboratory and Lawrence Livermore National Laboratory. Overall, the technology assessment study concludes that the use of fissile material in dilute form as fuel in LWR's poses the lowest risk, so long as appropriate safeguards are implemented to counter diversion.

7.1.2 Diversion Resistance Characteristics of the Various Alternatives

Among the alternatives discussed in this study, namely dispositioning by spiking or as spent fuel or after plutonium destruction, there is little to choose one alternative over the other from a purely permanent safeguards point of view. Recovering plutonium from any of these irradiated assemblies will be time and capital intensive and IAEA-type monitoring programs will easily detect such diversion. The spiking alternative is attractive in that it will result in timely dispositioning, however, by choosing the spike and reuse option it is possible to disposition plutonium and later obtain the energy benefits. In the case of the plutonium destruction alternative, unless the amount of plutonium left over is less than a few kilograms and spread through a large number of assemblies, it is no more diversion resistant than any of the other alternatives. This destruction fraction of plutonium has to be greater than 99 percent of the initial value and further, such destruction should be complete. In other words satisfying this limit by converting plutonium to other isotopes of plutonium may not provide sufficient grounds for removal of such assemblies from continued surveillance. Isotopes of plutonium with an atomic mass higher than 239 have considerable fission cross-section for fast neutrons. Further, generally speaking, high plutonium destruction alternatives have been found to involve new types of fuel or unproven technology, increasing the risk that dispositioning will not be carried out in a timely fashion.

7.1.3 Diversion Resistance Characteristics through Various Process Steps

In addition to the high diversion resistance characteristic of the final irradiated fuel assemblies, it is useful to consider the level to which diversion resistance is imparted as the plutonium is taken through the various process steps. The first of these, where the

plutonium is taken out of the weapon and the pit shape destroyed, affords the first level of resistance. In effect then, redeployment will mean the material will have to be recast and worked to a weapons-usable shape which will take time and sophisticated equipment. Such diversion and efforts towards redeployment will be detected quickly under the proposed safeguards implementation structure since the fuel fabrication process includes on-line material accountability assessment.

The second process step is conversion to oxide. This imparts a higher level of resistance as the oxide has to be reduced to metal before it could be worked to the required shape for redeployment. Down blending with uranium oxide and sintering increases the resistance further as more material will have to be processed in order to obtain the same (critical) quantity of plutonium. In the case of sintered pellets containing gadolinia, the resistance is still further increased as even small amounts of gadolinium could interfere with the efficiency of the fission process in the weapon. Assembly of pellets into rods and then into fuel bundle increases the resistance as it will then require heavy machinery to transport an entire fuel bundle. Finally, the irradiated material will have the highest level of diversion resistance as it will require a high level of infrastructure, investment and considerable time to reclaim the plutonium.

7.2 SAFEGUARDS IMPLEMENTATION - SYSTEM LEVEL CONSIDERATIONS

We now turn our attention to the definition of Safeguards in the sense of provisions that need to be implemented at various stages of the processes required to disposition the plutonium as irradiated fuel assemblies, to guard against theft or diversion. Curiously, diversion of the material by the nation-state to produce weapons is not as large an issue in that even after the first round of dismantlement of the weapons, a large stockpile will still be left in the hands of the CIS, or at least Russia. Nevertheless, there are certain reciprocity issues that have to be taken into account. Before considering the functional details of Safeguards implementation, certain system level considerations will be outlined first.

The conventional provisions for safeguarding special nuclear materials have to be modified in the present context for a number of reasons:

- 1. The quantity of fissile material involved that has to be processed within the short time allowed for dispositioning, particularly in the spiking alternative, is by far larger than the typical quantities dealt with in conventional safeguards.**
- 2. The quality of the fissile material, coming as it does from dismantled weapons, would require little or no rework, if diverted for deployment as weapons.**
- 3. This would be the first time that dismantlement of nuclear weapons and disposition of plutonium would have been undertaken in a concerted way and therefore requires a consideration of broad bilateral and (in future) potentially multilateral issues and agreements. Further, in as much as this dispositioning could be the forerunner of future weapons reductions, any problems in this area - whether they be technical or institutional - could lead to major setbacks for future efforts.**
- 4. Safeguards implementation in this project is inextricably tied to specific deployment options, for example as to whether dispositioning would occur bilaterally in the U.S. and CIS or even involve a third party as a neutral site.**
- 5. Unique to this project, safeguards should be effective as measured against the backdrop of possible social and political changes in the CIS. In this sense, it is important that the material be dispositioned with speed.**
- 6. If dispositioning occurred at both the U.S. and CIS, issues of transparency and reciprocity in safeguards have to be taken into account.**

Fortunately, the technology and processes recommended by this study are well known with the result we can fully utilize an extensive, already existing safeguards infrastructure. This includes, for instance, physical protection, material characterization and material

accountability provisions of the U.S. DOE Orders and NRC regulations when working with Special Nuclear Materials.

7.2.1 Institutions Available for Implementing Safeguards Provisions

It is useful at this point to start with the institutions which might be called on to provide the safeguards provisions that might be applied to the dispositioning project. Three different institutions could be distinguished:

1. Domestic Safeguards as applied to Special Nuclear Materials are principally enforced per NRC regulations on the commercial side. For Special Nuclear Materials controlled by government agencies or in DOE facilities which are exempt from licensing and regulation by NRC, DOE Orders apply for the handling, processing, transport and utilization of these materials.
2. International Safeguards, principally enforced per international treaties such as the Non-Proliferation Treaty through the IAEA.
3. Bilateral Safeguards agreements between the U.S. and CIS, which are yet to be negotiated.

Conventional safeguards is primarily administered through the first two institutions. To the extent that safeguards provisions under these two institutional arrangements are well documented, the technical problems associated with the third will be evaluated first and brief descriptions of the provisions of the other two institutions and their applicability will be described later. It should be noted that the provisions of international safeguards, as applied through IAEA or a similar arrangement, will still be required after discharge of irradiated assemblies from the reactor to the extent that any spent fuel assembly can be processed to yield weapons grade plutonium.

7.2.2

Bilateral Safeguards Agreements Between the U.S. and CIS

At present no bilateral agreements exist between the U.S. and CIS for the safeguards provisions to be applied in dispositioning.

The first step in dispositioning is the dismantlement of the weapon, removal of the pit and the destruction of its shape. There are a number of issues in this area, related to the levels of transparency and reciprocity and the manner in which each country would ascertain that the plutonium that is made available to this project did in fact come from a dismantled weapon. In particular, as of this writing, it appears that this process - the recovery and destruction of the original pit shape from the weapon - will not be made transparent. The details relative to this area will be worked out by a designated U.S. agency or interagency and therefore, will not be discussed any further in this report.

The first key issue that needs to be examined is the extent to which the U.S. provisions for safeguards embodied in the U.S. institutions such as NRC regulations and DOE Orders are made part of the bilateral safeguards. It is desirable that the entire U.S. safeguards provisions be made part of the bilateral safeguards, however, this requires some additional study. This will entail the transfer and/or the supply of procedures and machinery that are used in materials characterization and accountability such as assay techniques. Detailed studies on the potential for theft, diversion or redeployment would be needed if the currently accepted U.S. safeguards provisions are not incorporated *in toto* in the bilateral agreement. Evaluation and qualification of the CIS safeguards practices to have them equivalent to the U.S. DOE Orders is a possibility.

The next safeguards issue that needs to be tackled bilaterally occurs when dispositioning is carried out, presumably using similar processes, at both the CIS and U.S. venues. Reciprocity agreements at the detailed technical level need to be worked out and these are disposition-process dependent. For example, even if we assume that the bilateral agreements for the rate of plutonium metal being made available at each site are being met, the issue still arises as to the rate at which the dispositioning process is occurring at each

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location. For, it is unlikely that the rate of dispositioning at both locations will be exactly or even be very nearly the same given that some plant equipment may become unavailable for unavoidable reasons. If this happens to be the case, some provisions must be evolved in the bilateral agreement on such factors as allowable lead or lag in process throughputs between the two locations and specifications for hold points, if any. A major (expected) hold point will be fuel bundle loading into the reactor as recovery of plutonium from an irradiated fuel bundle requires very large investments and infrastructure. It is conceivable therefore, that the lag/lead allowable at this point would be the minimum acceptable to both parties. These types of issues become much more complex if dispositioning occurs in the two countries using entirely different processes.

Additional factors that need to be considered in bilateral safeguards include provisions for transparency and bilateral responsibilities for material transfer/accountability provisions. When material is transferred from one stage to the next and more importantly if material has to be routed in any manner other than that postulated under the baseline configuration, it is likely that the provisions will have to include bilateral permissions. In addition, it is desirable that reciprocal arrangements be made available to independently verify and concur with the safeguards design and standards, both during initial design, construction and checkout of the machinery, and later for procedures in the MOX plant for special nuclear materials characterization and accountability during the disposition operation. Stated differently, it is desirable for U.S. personnel to be present to verify the dispositioning operation in key stations in the CIS dispositioning process, as part of the operating crew. In addition, it is desirable for the U.S. to independently verify the standards used for material accountability at specific intervals. The need for this reciprocity is not necessarily to prevent diversion of the SNM by the state but more to ensure that this material is not stolen or diverted to third parties by disgruntled employees.

A final area of bilateral agreement relates to transport of special nuclear materials. At present there is no infrastructure for transport of special nuclear materials between the CIS and U.S. If the plutonium from CIS were to be dispositioned in the U.S., a transportation safeguards infrastructure has to be developed, and once again, it is possible to model these

provisions on already existing infrastructure for transport within the U.S.

7.2.3 Domestic Safeguards

Domestic safeguards provisions for handling special nuclear materials have been very successful to-date in preventing theft or diversion. The processes envisioned in the proposed dispositioning scheme - plutonium metal to oxide to MOX powder to MOX pellets to fuel bundles to reactor loading - are all well covered by existing DOE Orders for special nuclear materials in government-owned or DOE operated facilities which are exempt from licensing by NRC and on the commercial side by NRC regulations. DOE Order 5633.2A dated September 1992, for instance specifies the Responsibilities and Authorities for the Control and Accountability of Nuclear Materials. There are two areas where additional work appears to be needed. These arise as a result of the exceptions to conventional safeguards noted at the beginning of this section:

- a. The admissible level of discrepancies in material accountability arising from the natural scatter in measurement techniques needs further scrutiny in view of the large amounts of fissile material flow-through involved. Although it is unlikely that diversion will take place by the removal of very small quantities of material over a long period of time, this cannot be discounted. No new assay techniques with a lower scatter are needed but the results of material accountability and the statistics of the control measurements should be correlated more precisely and any errors that result in material accountability which are within the statistically admissible region should be tracked by trend analysis. If the results indicate that the unaccounted material is dispersed in a normal fashion over a period of time, no action may be warranted, however, if the unaccounted material is always biased (towards a possible removal of material from the stream), additional actions may be warranted.
- b. A second area where additional work is needed relates to the use of conventional assay techniques. A number of different assay techniques are available to determine the quantity of plutonium in a given setting. Some of them depend not on a positive

interrogation for Pu^{239} but on the signature of other isotopes which decay more rapidly such as Americium. The quantity of Pu^{239} is then computed knowing the ratio of the various isotopes in the batch. As noted in the exceptions at the beginning of this section, the plutonium material is almost all Pu^{239} . Therefore, these "indirect" methods should be discarded in the assay techniques. The Non Destructive Test assay method using Cf^{252} developed at Los Alamos National Laboratory and successfully used to assay MOX fuel pins fabricated at Westinghouse Hanford for the Fast Flux Test Facility would appear to be the most suitable.

Although a complete and thorough review of the domestic safeguards provisions with specific reference to its application to this study could not be conducted because of time limitations, from a preliminary review it is concluded that available domestic safeguards provisions are sufficient to meet all the requirements to protect this material against diversion.

7.2.4 International Safeguards

International safeguards are administered through the IAEA and these would be required at least to the extent that spent fuel is not being reprocessed to obtain weapons grade plutonium after the material has been dispositioned. The database to-date indicates that obtaining plutonium for weapons purposes by reprocessing commercial spent fuel has generally not been attempted by any of the nation states. This is simply a result of the fact that reprocessing to obtain plutonium from commercial spent fuel or in this case, spiked fuel, involves an extensive infrastructure and investment and the resulting plutonium is generally considered less than desirable from a purity standpoint for weapons use. It is however possible to irradiate special assemblies for short periods of time and then to obtain, by processing, weapons grade plutonium.

The IAEA safeguards parallel domestic safeguards in a number of respects, particularly as it concerns provisions related to material characterization, accountability and physical security. There has been good cooperation between the U.S. agencies such as DOE and

NRC and the IAEA to bring to bear the latest in available technology to international Safeguard systems design and implementation. There are two key features where IAEA Safeguards differ. First, it is a voluntary system in that the nation-states declare the sites where special nuclear materials may be involved and open them to inspection and further these states are voluntary parties to the NPT. The inspection efforts are principally aimed at detecting diversion if it occurs from these sites rather than preventing diversion and IAEA has no enforcement powers. The second key feature relates to the fact that it is a two-tier system, applies as it does to those who currently possess nuclear weapons and those who do not. This feature would normally not be of concern, except that the nation-states of the CIS who will participate in this disposition project might claim the status of a nuclear weapons-state and therefore claim exemptions from IAEA safeguards accorded to this tier.

The entire issue of IAEA Safeguards and whether they apply needs to be examined. Of the experts consulted, there was no consensus as to whether IAEA safeguards provisions applied to the present study or even as to whether or not IAEA should be involved in the process. This entire area needs to be examined further, specifically to assess if a combination of domestic and bilateral safeguards are sufficient for this project.

7.3 SAFEGUARDS IMPLEMENTATION - FUNCTIONAL LEVEL CONSIDERATIONS

For purposes of this study, it has been assumed that the provisions of domestic safeguards as enforced by DOE orders and NRC regulations will be implemented. Starting with the arrival of plutonium metal where the pit shape has already been destroyed, the plutonium metal will be transported to the site where it will be converted into oxide if the conversion plant is not co-located at the pit removal site. Safeguards provisions for transport will be coordinated through applicable Transportation Safeguards Orders applied by the DOE, Albuquerque office.

7.3.1

Safeguards Implementation in the Conversion Process

Receiving, storing, handling, and processing of plutonium from its metal form to oxide through the mixed-oxide fuel fabrication plant, to its loading into the fuel bundles will follow the provisions contained in a number of DOE Orders and/or NRC regulations. These provisions specify physical security, material characterization and accountability provisions, setting up of material balance areas, and frequencies for destructive and nondestructive assays. The latest technology for Non-Destructive Test assay, and on-line computer control for material accountability will be implemented. These safeguards provisions in the mixed-oxide fuel fabrication plant are discussed in more detail in Section 3. The mixed-oxide plant itself will be highly automated but not to the levels which are required to handle spent fuel for reprocessing. In other words, the plant is automated for the most part providing a high level of safeguards and reducing worker exposure, however, if non-routine maintenance has to be carried out, this can be done quickly with access by trained personnel. A completely automated factory with extra-heavy shielding capable of handling irradiated fuel, where even the non-routine maintenance is done by remote control is not necessarily the best from a safeguards viewpoint for the present project. A completely remotely operated facility with heavy shielding for mixed-oxide fuel production has three drawbacks: (a) it will take more time to build and may not meet the timeliness objective; (b) it may not be able to satisfy the need for transparency; and (c) it can be used to reprocess spent fuel by the nation-state to recover plutonium at a later date.

7.3.2

Safeguards Implementation in the Reactor Plant

Safeguards implementation in the reactor plant itself is straightforward and is discussed in Section 4. It is difficult to remove a bundle without heavy machinery. Further, the loading and unloading of fuel bundles is accomplished with computer control. The control software will be modified as needed to implement the level of safeguards provisions needed in the reactor plant, particularly in regard to the movement of arriving fresh bundles so that once they are placed into the pool on initial arrival, they cannot be moved without sufficient

permissions. Redundancy will be established in counting the number of fresh bundles under storage and the entire reactor complex area will be under a high level of physical security.

7.3.3 Safeguards Implementation for the Complex Perimeter

Reactor complex perimeter security is another important element of safeguards and is also discussed in detail in Section 4. Briefly, the perimeter safeguards will include procedures and physical security systems to deter and detect unauthorized intruders and assure that the diversion of special nuclear materials does not occur. Both passive and active components will be incorporated including perimeter and security fences; entry control points; guard houses; vehicle barriers, etc. In addition, the latest technology will be consulted to interrogate outgoing and incoming packages for nuclear materials.

7.4 DEVELOPMENT REQUIREMENTS

With the proposed dispositioning scheme, no new additional safeguards technology is needed as the domestic infrastructure covers the entire range of handling and processing plutonium to mixed-oxide fuel to irradiated bundles. However areas, though not falling under the conventional definition of Development, have been identified here and discussed more fully in Section 9 that will enhance safeguards implementation. These areas are:

- a. Simulation studies of the process flow to identify points of vulnerability, requirements relative to reciprocity and transparency, lead/lag requirements for simultaneous dispositioning in two different sites, statistical sampling requirements, etc.
- b. Literature survey, obtaining information on European/CIS MOX plants and documenting latest advances in Safeguards implementation.
- c. Software requirements for movement of material in the MOX plant or in the reactor building requiring bilateral permissions.

Processing plutonium into fuel generally involves movement of the nuclear material through various stages in a batchwise fashion. Movement from one stage to the next generally entails passage of one or more containers containing the Special Nuclear Material through a tunnel or passageway connecting the adjacent stages. In the past it had been necessary to manually record the data pertinent to such a movement including the container number, the empty weight of the containers, the loaded weight prior to transfer to the new stage, the weight after discharge of the material to the new stage, and the weight on return of the empty container to the old stage etc. The recording process could be automated by having bar coded containers and code reading machines in the transfer tunnel. Such automated recording processes would help in on-line accountability. Similar active and passive interrogation techniques of packages for Special Nuclear Material have been developed at Los Alamos National Laboratory which could be implemented to prevent diversion.

Other elements of technology to be incorporated to enhance safeguards will be evolved as more detailed design and process descriptions are implemented in Phase II of this project.

The best overall safeguards to prevent diversion or redeployment of plutonium as weapons is to choose a dispositioning option that is already well proven, where the underlying institutions already exist and no new ones have to be invented, where the attributes of timeliness, reciprocity and transparency could be implemented. The proposed dispositioning scheme fulfills all these requirements and is therefore thought to represent the best route to achieving safeguards of the weapons plutonium.

Alternative routes to dispositioning either by glassifying the plutonium and burying, or by mixing with fission products and burying do not have the data base for permanent long-term safeguarding of the weapons plutonium comparable to spent fuel assemblies from commercial reactors. The destruction of the pit shape constitutes the first level of

safeguards. Beyond this point, safeguards is assured by rapid processing through the various steps that incrementally increase the resistance along the way.

Safeguards implementation, similar to building a mixed-oxide fuel fabrication facility or an ABWR, requires an infrastructure. Such an infrastructure embodying the division of responsibilities, activities and specification of safeguards provisions is already in place for the proposed dispositioning approach. While a domestic safeguard infrastructure is already in place, a comparable infrastructure may not to be in place in the CIS. Issues relative to bilateral safeguards agreements have been pointed out. Dispositioning the plutonium as irradiated fuel assemblies in an ABWR through its use as mixed-oxide fuel, represents the most timely, safe and proven means of achieving long-terms safeguards of this material.

7.0 REFERENCES

- 7.14 Nuclear Proliferation and Safeguards, Congress of the United States, Office of Technology Assessment, Praeger Publishers, New York, 1977
- 7.2 Safeguarding Nuclear Materials, Vols. I and II, Proceedings of a Symposium held in October 1975, IAEA, Vienna, 1976.
- 7.3 Measurement Technology for Safeguards and Material Control, NBS Special Publication 582, U.S. Department of Commerce, National Bureau of Standards, Proceedings from ANS Topical Meeting held in November 1979 in South Carolina, June 1980.

SECTION 8.0

SAFETY, ENVIRONMENTAL, AND WASTE MANAGEMENT ISSUES FOR THE ENTIRE COMPLEX

This section provides a summary of projected impacts from the Plutonium Disposition Complex (PDC) as a whole. Descriptions of the individual facilities that make up the complex components are provided in Sections 3.0, 4.0, 5.0, and 6.0.

It also provides early identification of the impacts to worker health and safety and the environment that may result from complex construction and operation. The goal is to provide a brief preliminary assessment of these potential impacts so that a comparison between plutonium disposition options can be conducted. The limited scope of this study did not permit the comprehensive assessment of the impacts that would be required in an environmental impact statement.

In order to have a benchmark for determining the margin of operational safety for the complex, applicable regulations are presented. The general criteria for the study were that the complex should be licensable by the NRC. However, because this would be a DOE project, DOE and other applicable regulatory agency standards and codes that are more restrictive than those in the Code of Federal Regulations (CFR), Title 10, were considered. The following is a listing of pertinent regulatory agencies and limitations considered:

Occupational Safety - *Department of Energy and Department of Labor* - Construction and operation of the complex must meet occupational protection standards specified in DOE Orders, 29 CFR, and/or applicable State regulations.

Occupational Radiation Safety - *Nuclear Regulatory Commission* - The plutonium destruction complex must meet the requirements of 10 CFR 20 (Standards for Protection Against Radiation) in order to qualify for licensing by the NRC. The standards applicable to this facility are:

- **Subpart B - Radiation Protection Programs**
 - Operational radiation protection program established.
 - Occupational radiation doses as low as reasonably achievable (ALARA).
- **Subpart C - Radiation Dose Standards**
 - 5 Rems/yr total effective dose equivalent (TEDE).
 - 50 Rems/yr to individual organ (sum of dose equivalent and committed dose equivalent).
 - 15 Rems/yr eye dose equivalent (DE).
 - 50 Rems/yr DE to skin and extremities.

Department of Energy - The plutonium destruction complex must also meet the health, safety, and environmental protection standards of the DOE 5400 series of orders. These Order requirements must be met in order to obtain approval to operate through the DOE readiness review process. The standards applicable to this facility are:

- **DOE 5480.11 - Radiation Dose Standards**
 - 5 Rem annual effective dose equivalent (AEDE).
 - 15 Rems/yr DE to the lens of the eye.
 - 50 Rems/yr DE to extremities.
 - 50 Rems/yr DE to individual organ or tissue.

In addition, DOE and commercial facilities (NRC licensees) were previously designed and operated with occupational radiation dose limits ranging from 1 to 5 rem/yr. However, with the issuance of the DOE Radiological Control (RadCon) Manual in 1992, the DOE inferred a design criterion of 0.5 Rem effective dose equivalent (EDE) per year. Further, despite maintaining a 5 rem/yr EDE dose limit, DOE has imposed some rather restrictive administrative control levels significantly below the dose limit [e.g., Program Secretarial Official (PSO)] approval to exceed 2 rem/yr EDE].

Environmental Protection

Nuclear Regulatory Commission

- Subpart D - Radiation Dose for Individual Members of the Public
 - 0.1 Rem/yr (excluding sanitary sewer disposal).
 - Subpart K - Sanitary sewerage release not to exceed Table 3, Appendix B limits.

Department of Energy

- 40 CFR 61 H
 - Site dose limit of 0.01 rem/yr EDE to members of the public from all DOE airborne sources.
 - DOE Order 5400.5
 - 0.1 rem/yr EDE to offsite individuals from all DOE sources.
 - Pollutants in liquid discharges are driven by the ALARA policy and implemented with Best Available Technology (BAT) criteria.

Waste Management - The construction and operation of the complex will generate a volume of waste each year. The various waste types to be generated include high-level radioactive waste, low-level radioactive waste, transuranic waste, hazardous waste, and solid and sanitary wastes. Waste activities are regulated by a variety of government agencies. These typically are:

- High-level radioactive waste - Department of Energy and/or Nuclear Regulatory Commission.
- Low-level radioactive waste - Department of Energy, Nuclear Regulatory Commission or State (if an agreement state).
- Transuranic waste - Department of Energy.

- Hazardous waste - Environmental Protection Agency or cognizant State agency (if RCRA program approved).
- Solid and sanitary wastes - Local health agency standards.

8.1 OVERALL SAFETY AND ENVIRONMENTAL ISSUES SUMMARY

This section provides the projected occupational and environmental safety impacts of the complex and compares those impacts to current regulatory limits and to similar prior projects. The information provided demonstrates that the complex is safe to construct and operate and that with respect to health, safety, and environmental protection, the Reactor Facility is licensable by the Nuclear Regulatory Commission (NRC). In addition, this section demonstrates the complex would also meet those standards where the Department of Energy is more restrictive than the NRC standards.

The National Environmental Policy Act (NEPA) requires identification and assessment of impacts to the human environment from proposed projects such as the plutonium disposition project. This section provides preliminary identification and assessment of these impacts for planning purposes only. This project is expected to eventually require an environmental impact statement (EIS). This section presents specific information that would be involved in preparing an EIS. Section 8.2 provides information on the projected schedule of NEPA activities. Section 8.3 is a discussion of the waste minimization characteristics incorporated into the design and operation of the complex.

The NEPA process requires a significant amount of information to be collected or developed for a proposed site's environment. For a project of this magnitude and type, the following site resources must be characterized prior to determining the project's impact to these resources:

- Land/Geologic Resources
- Air Resources and Noise
- Water Resources

- **Land Use, Recreational, and Visual Environment**
- **Biotic Resources**
- **Cultural Resources**
- **Radiological Impacts**
- **Nonradioactive Hazardous Materials**
- **Socioeconomics**
- **Transportation**
- **Waste Management**
- **Decontamination and Decommissioning**

Some of these potentially impacted resources can be discussed in useful detail for a generic site. However, the impacts to some resources such as biotic, cultural, and socioeconomic resources and transportation systems are highly site dependent. More detailed assessments of impacts to these resources must be made after specific sites are proposed.

Because siting for this project has not been established at this date, this preliminary environmental assessment was performed for a generic site. More specific and accurate impacts to the human environment can be projected for specific site(s) during preparation of an EIS. However, this assessment is satisfactory for comparing impacts from various project alternatives.

In a few cases, detailed site specific information has been developed over several years for similar proposed DOE projects. These are sites located at Hanford in Washington State, the Idaho National Engineering Laboratory in the State of Idaho, and at Savannah River in the State of South Carolina. In these cases, the environment has been well characterized. This information is available for use in preparing an environmental impact statement for the proposed project with little more than updates and minor modifications. Consideration of other potential sites may require the collection and/or determination of existing environmental characteristics which would entail significant additional resource and time commitments.

This preliminary assessment studies potential impacts from each of the specific nuclear facilities that together make up the PDC. The individual facilities are the Fuel Fabrication Facility, the Advanced Boiling Water Reactor Facility with and without the tritium production option, and the Spent Fuel Storage Facility. Details of each facility is discussed in their respective sections. This section provides a summary of the projected impacts as a whole.

8.1.1 Land/Geologic Resources

The construction of the complex including one or more reactors may have a limited impact to the land/geologic resources of the site. However, the impact is highly dependent upon the specific site chosen. It is possible to choose a Site with an insignificant impact to these resources.

Depending upon the method for dissipating the waste heat into the environment, the impact to surrounding land from complex operation may or may not be significant. Water vapor from cooling towers may contain small amounts of salts and other trace chemical compounds. One realistic scenario is for salt to build up in the soil to levels that could cause deleterious effects on the area vegetation. The actual affect would depend upon the specific site chosen and the susceptibility of local vegetation to slight salt concentration increases.

8.1.2 Air Resources and Noise

Preconstruction and construction activities are expected to temporarily degrade the ambient air quality in the immediate vicinity of the site. During dry periods, fugitive dust levels may be exceeded on a short term basis in the area of construction from land disturbance and heavy vehicle traffic. Dust mitigation measures may be implemented during certain construction periods. During operation of the facility, no significant quantities of fugitive dust emissions are anticipated from any stationary source.

The facility is expected to generate about 2,500 megawatts. The impacts of waste heat discharge can be mitigated through the selection of appropriate technology such that no perceptible changes to the environment would be encountered.

The projected air quality impacts from project construction and operation include all pollutants (e.g., dust, CO₂, NOX, etc.) that could become airborne as a result of project activities with the exception of radioactive materials, which are included in the Radiological Impacts Section. Some other examples include noise impacts, ground vibration, and airblast overpressurization from blasting. The actual impact depends upon the type of biological systems inhabiting the area and the distance from the complex.

During plant operation, the waste heat facility may produce vapor plumes that are visible for significant distances. The plumes would inject about 2,900 m³ per hour of water into the atmosphere in the form of water vapor. Depending upon meteorology, the plumes could impact the close-in area with a drizzle-like condition, ground fog, and ground icing.

Also during operation, the three emergency or backup power generators would be test run intermittently. These would emit about 1400 lbs of SO₂, 980 lbs of NO_x, and 34 lbs of particulates per year.

8.1.3 Water Resources

Impacts to water resources are highly dependent on the specific site. The proximity of the facility to surface waters and the specific source of water used largely determines the impact. During facility construction, it is possible that water runoff from the construction site could contain a high suspended solids content that could impact clarity of surface waters. Depending on the site, mitigation efforts could be warranted.

The complex is expected to withdraw from the local environment about 2.4×10^7 gallons of water per year for operation. The breakdown for specific facility use is provided in Table 8.1. This may or may not be a significant quantity of water, depending on the specific site.

Water is expected to be lost to evaporation and process uses (e.g., steam, sanitary use, etc.). There are no planned routine liquid effluents from this facility other than cooling water or cooling tower blowdown. Therefore, no pollutants from this facility should impact the quality of area water resources. However, the complex is expected to be permitted to release liquids during unusual conditions. These liquids could contain trace pollutants below levels specified in the facility NPDES permit.

Table 8.1. Plutonium Complex Annual Water Use

Individual Facility	Volume Withdrawn From Environment (m³/yr)	Volume Returned to Environment (m³/yr)
Fuel Fabrication Facility	5.5x10 ⁴	1.46x10 ⁴
1 ABWR	2.35x10 ⁷	3.79x10 ⁶
Spent Fuel Storage	8.3x10 ³	2x10 ³
Total	2.35x10⁷	3.8x10⁶
Tritium Production (Incremental)	0	0

Depending upon the ultimate heat sink chosen, area water features could be thermally impacted. Continuous and/or intermittent thermal discharges to rivers, streams, or lakes could increase water temperature enough to affect aquatic systems. The actual impact is dependent upon the specific site chosen.

8.1.4 Land Use, Recreational, and Visual Environment

The largest projected impact from this facility will be the clearing of land (if the site contains trees or large rocks) and subsequent loss of this land to production and ecological processes. The impacts to land use, recreation, and the visual environment will depend upon the final method of waste heat disposal and existing site uses. This facility will require about 185 acres of land, the total impact being dependent upon the specific site chosen.

8.1.5 Biotic Resources

Construction and operation of the complex could have some impact on local terrestrial and aquatic biotic systems. The actual impact on these systems is highly dependent upon the specific site chosen. Any impact to biological species such as endangered or threatened, must be considered. The extent and magnitude of these impacts are highly site-specific.

8.1.6 Cultural

Construction and operation of the reactor plant could have some impact on local cultural resources. The actual impact on these is highly dependent upon the specific site chosen.

8.1.7 Radiological Impacts

This section details the projected environmental and occupational radiological impacts to the public and to facility workers. The various reactor designs are expected to have consistent environmental impacts during the construction phase of the project. However, a major design criterion of the ABWR has been the minimization of radiological impacts during operation, and decontamination and decommissioning activities. This section provides detailed information on the radiological impacts and demonstrates that this design results in significantly reduced radiological impacts.

8.1.7.1 Considerations

Prior DOE and commercial facilities (NRC licensees) were designed and operated with individual occupational radiation dose limits ranging from 1 to 5 rem/yr. However, with the issuance of the DOE Radiological Control (RadCon) Manual in 1992, the DOE inferred a design criterion of 0.5 Rem EDE per year. Further, despite maintaining a 5 Rem/yr EDE dose limit, DOE has imposed some rather restrictive administrative control levels significantly below the dose limit. From a radiological control standpoint, the design criterion of 0.5 Rem/yr EDE to individual workers is a significant challenge, however, it is

achievable for the ABWR.

Also, Department of Energy facilities are subject to more restrictive effluent concentration limits in many cases than are licensees regulated by the Nuclear Regulatory Commission. For instance, 10CFR20.1301 specifies a dose limit of 0.1 rem/year for members of the public from a licensed facility and Table II airborne concentration limits are based upon an individual radionuclide delivering 0.05 rem/yr. However, DOE facilities must comply with a site dose limit of 0.01 rem/yr to members of the public from all airborne radionuclides (40CFR61, subpart H).

Siting the complex within an existing DOE site must therefore take into account the current dose delivered to the public by existing facilities. In such a case, the complex becomes part of the DOE Facility which, taken as a whole, is limited to the 0.01 rem EDE dose standard from airborne sources. DOE Order 5400.5 also specifies a limit of 0.1 rem/yr EDE to offsite individuals that pertains to radiation dose accumulated from all DOE sources. This Order also specifies that pollutants in liquid discharges will be driven by the DOE ALARA policy which is implemented with Best Available Technology (BAT) criteria.

8.1.7.2 Assessment

Radiological impacts of the complex are divided into occupational and public, or environmental. The reason for the division is that different protection philosophies and dose standards apply to these different groups.

Occupational - Occupational doses to complex workers were calculated using external dose rates and projected exposure times. The doses for the ABWR, which is the predominant contributor to the complex occupational dose, are in annual effective dose equivalents (AEDE) and are presented in Table 8.2. These doses are an annual average based on an 18 month refueling cycle. That is, the doses accumulated during the refueling and maintenance outage actually occur every other year. Any internal deposition of radionuclides would contribute to a committed dose to individual workers and could

Table 8.2. ABWR Occupational Dose Assessment

Plant Area	Typical BWR		ABWR		
	Man-Hours Per Year	Man-Rem	Man-Hours Per Year	Man-Rem	Comment
Drywell					
MSIV	9,000	72.0	2,000	3.0	A,B
SRV, Pumps,	1,450	24.7	1,150	11.2	A,C
Valves	1,500	25.5	200	3.0	C,D
CRD	200	12.0	200	8.5	B,C
LPRM/TIP	2,400	38.8	1,200	7.8	A,E
ISI	5,500	93.5	1,300	9.1	A,E
Other					
Refueling					
Access/Reassembly	4,500	13.5	2,000	2.0	A,D
Refueling	4,400	11.0	2,000	1.0	A,D
Fuel Sipping	950	3.2	0	0.0	D
Jet Pump Work	600	3.0	0	0.0	C
Other	2,400	4.5	1,000	1.5	D
Reactor Building					
RHR/RWCU	1,550	25.6	650	7.8	B,D,E
CRD	600	3.0	200	1.0	C
Instruments	1,000	5.0	1,000	5.0	
Other	5,000	16.3	3,000	7.5	B,D
Radwaste	6,700	36.9	3,500	5.3	B,D,E
Turbine Building					
Valves	1,150	10.9	1,000	3.0	B,D
Turbine	18,250	5.5	15,000	2.3	B,D
Condensate	2,000	15.0	1,000	2.0	B,D,E
Other	13,150	1.3	10,000	1.0	B,D
Work At Power	5,000	33.0	3,500	15.8	D,E
Total	87,300	444	49,900	98.0	

A = Reduced radiation. No recirculation lines.

B = Improved maintenance procedures.

C = Fewer components requiring maintenance.

D = Reduced radiation levels due to improved fuels, materials selection, and water chemistry.

E = Automation resulting in fewer man-hours or less personnel exposure by remoting.

increase the dose values slightly. However, no airborne contaminant problems are projected with the ABWR design.

Committed effective dose equivalents for the tritium production option were calculated with personnel exposure times and projected workplace tritium concentrations.

Radiation doses for workers are projected and totaled for all facilities in Table 8.3. Both of these tables pertain to steady-state operating conditions. The relationship between projected occupational dose and facility operating time is demonstrated in Figures 8.1-1 to 8.1-3. The radiation dose accumulated by workers over the facility operating life is provided in Figures 8.1-4 to 8.1-6.

Environmental - The environmental impacts from released radiation were assessed using projected facility source terms at a generic DOE site. This site is assumed to have a distance of 400 meters from the gaseous effluent point (stack) to the protected area fence (or fenceline). In reality, the complex would most likely be located on a DOE site with a significant distance (e.g. 2-12 kilometers) to the closest public access. Also, calculations show that facility elevated emissions would generally not contact the ground for some distance beyond the 400 meters chosen as the fenceline distance. Therefore, the impacts to an offsite receptor at 10 kilometers were also provided. Table 8.4 contains the characteristics of the gaseous effluent points for each facility.

In calculating the impacts to the postulated 400 meter and 10 kilometer receptors, adverse meteorological conditions were assumed. The highest dose at each location under a variety of meteorological conditions were selected as the impact for each facility. These were then summed in Table 8.5 to present the total radiological offsite impact for the complex. It must be noted that the dose values presented should be considered the maximum at those two receptor locations, but may not be the maximum possible along the plume centerline. Determination of maximum receptor locations and values can be more accurately determined when specific candidate site(s) are chosen.

Figure 8.1-1 Complex Occupational Dose, Spiking Case

Figure 8.1-2 Complex Occupational Dose, Spent Fuel Case

Figure 8.1-3 Complex Occupational Dose, Plutonium Destruction Case

Figure 8.1-4 Complex Occupational Radiation Dose, Spiking Case

Figure 8.1-5 Complex Occupational Radiation Dose, Spent Fuel Case

Figure 8.1-6 Complex Occupational Radiation Dose, Plutonium Destruction Case

Table 8.3. Plutonium Complex Occupational Radiological Impact

Individual Facility	Occupational Dose (Person-Rem/Yr)
Fuel Fabrication Facility	0.5
1 ABWR	98
Spent Fuel Storage	3
Total	101.5
Tritium Production (Incremental Dose)	2.4

Table 8.4. Gaseous Effluent Point Characterization

Stack Exhaust Characteristics	MOX Fuel Facility	Reactor Complex	Spent Fuel Storage
Stack Height	30 meters	76 meters	30 meters
Stack Inside Diameter (Top)	0.66 meters	2.4 meters	0.77 meters
Stack Flow Rate	11.8 m ³ /sec	153 m ³ /sec	16.5 m ³ /sec
Gas Exhaust Velocity	35 m/sec	33.8 m/sec	35 m/sec
Exhaust Gas Temperature	80°F (26.7°C)	106°F (41.1°C)	80°F (26.7°C)

Table 8.5. Plutonium Complex Offsite Radiological Impact

Individual Facility	Maximum Offsite Dose (mRem/Yr) 400 meters	Maximum Offsite Dose (mRem/Yr) 10 km
Fuel Fabrication Facility	2.1×10^{-1}	6.8×10^{-3}
1 ABWR	1.78×10^{-19}	1.28×10^{-1}
Spent Fuel Storage	2.45×10^{-4}	1.93×10^{-2}
Total	2.1×10^{-1}	1.54×10^{-1}
Tritium Production (Incremental Dose)	3.36×10^{-19}	3.56×10^{-1}

The offsite doses were calculated using the Radiological Safety Analysis Computer Program (RSAC). The projected offsite dose terms were calculated with the following source term assumptions:

- Fuel Fabrication Facility - All radiological releases were Pu-239.
- Advanced Boiling Water Reactor - All radionuclide constituents were considered except ^{16}N . This radionuclide is expected to be decayed a minimum of 25 half-lives prior to reaching the facility boundary. Therefore, the ^{16}N source term is assumed to be 0.
- Spent Fuel Storage - Only ^3H , ^{85}Kr , and ^{129}I are available for release.
- Tritium Releases - All tritium assumed to be water vapor (HTO or T_2O).

8.1.8 Socioeconomics

The construction and operation of the ABWR could have a socioeconomic impact upon on the region. The impacts could range from insignificant to significant, depending upon the specific site chosen.

8.1.9 Transportation

The construction and operation of the ABWR could have an impact upon on the regional transportation system. The impacts could range from insignificant to significant, depending upon the specific site chosen.

8.1.10 Waste Management

The construction and operation of the ABWR will generate a volume of waste each year. The various waste types to be generated include high-level radioactive waste, low-level radioactive waste, transuranic waste, hazardous waste, and solid and sanitary wastes.

8.1.10.1 High-level Radioactive Waste

The operation of nuclear reactor(s) at the site will produce spent fuel that is classified as high-level radioactive waste. The disposition of this waste is the responsibility of the DOE. The volume of this waste generated annually is dependent upon the plutonium disposition option chosen.

8.1.10.2 Transuranic Waste

Operation of the fuel fabrication facility will generate a small volume of transuranic waste each year. Transuranic wastes will be packaged, possibly compacted, certified, and transported to WIPP for disposal in the State of New Mexico. For this report, the transuranic waste volume is included as low-level radioactive waste.

8.1.10.3 Low-Level Radioactive Waste

Operation of the complex facilities will generate a relatively small volume of low-level radioactive waste each year. The reactor waste will primarily consist of concentrated waste from the reactor water cleanup system and waste generated during maintenance activities (e.g. protective clothing). Fuel fabrication waste will primarily be low-level radioactive waste, however, some transuranic wastes will also be generated. The spent fuel storage facility will generate a very small volume of low-level radioactive waste that will be handled by the reactor waste facility.

Low-level radioactive waste generated at the complex will be compacted, packaged, and transported to a DOE facility for disposal.

8.1.10.4 Hazardous Waste

Complex construction and operation will involve generation of nonradioactive hazardous wastes. Some examples are used oils, lubricants, hydraulic fluids, and various chemical compounds. The complex operator is responsible for complying with all RCRA

requirements for these wastes. All hazardous wastes will either be rendered non-hazardous at the Site or will be shipped to an offsite contractor for treatment and disposal.

8.1.10.5 Solid and Sanitary Waste

The construction and operation of the complex will generate a volume of solid and sanitary waste each year. Solid waste consists of garbage, trash, and items normally disposed at a community landfill. It will be disposed at or near the site. Sanitary waste will be disposed either in site septic systems or will enter existing municipal systems for treatment and disposal.

Table 8.6 presents a summary of each type of waste generated at the complex per year. Figures 8.1-7 - 8.1-9 provide a plot of the low-level radioactive waste volumes generated by the complex each year over the facility operating life. One plot is provided for each of the plutonium disposition scenarios. Figures 8.1-10 - 8.1-12 are graphic illustrations of the total accumulated low-level radioactive waste generated over the operating life of the complex.

Table 8.6. Plutonium Complex Waste Generation

Individual Facility	Low-Level Radioactive Waste (m ³ /Yr)*	Hazardous Waste (m ³ /Yr)	Solid and Sanitary Wastes (/Yr)
Fuel Fabrication Facility	60	100	30 m ³ 2.1x10 ⁶ gal.
1 ABWR	100	290	3,100 m ³ 1.6x10 ⁶ gal.
Spent Fuel Storage	10	5	25 m ³ 1x10 ⁶ gal.
Total	170	395	3155 m³ 4.7x10⁶ gal.
Tritium Production	10	0	0

*Approximately 1500 kg/M³ for compacted low level radioactive waste density of approximately 1.5 g/cc.

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Figure 8.1-7 Complex Low Level Waste Generation, Spiking Case

Figure 8.1-8 Complex Low Level Waste Generation, Spent Fuel Case

Figure 8.1-9 Complex Low Level Waste Generation, Plutonium Destruction Case

Figure 8.1-10 Complex Low Level Waste Generation, Spiking Case

Figure 8.1-11 Complex Low Level Waste Generation, Spent Fuel Case

**Figure 8.1-12 Complex Cumulative Low-Level Waste Generation,
Plutonium Destruction Case**

8.1.11 Decontamination and Decommissioning

The method(s) and timetable for decontaminating and decommissioning the complex will be determined as the plant reaches the end of its useful life. Various options are available on the decontaminating and decommissioning scheduling. The process could begin relatively quickly after permanent shutdown, or an interim storage period may be justified to allow radiation levels in the reactor facility to decrease. The plant has certain design features to aid in the decontamination effort. Some examples are:

1. Provisions for draining, flushing, and decontaminating equipment and piping
2. Design of equipment to minimize the buildup of radioactive material and to facilitate flushing of corrosion product traps.
3. Shielding that provides protection during both maintenance or repairs and during decommissioning operations.
4. Provision of means and adequate space for utilization of movable shielding.
5. Separation of more highly radioactive equipment from less radioactive equipment and provisions of separate shielded compartments for adjacent items of radioactive equipment.
6. Provision of access hatches for installation or removal of plant components.
7. Provisions of design features such as the RWCS and the condensate demineralizer to minimize corrosion product buildup.

Dismantling the reactor buildings would basically have the same impacts on the environment as the original site preparation and facility construction. Vehicle traffic, noise, and dust levels would be temporarily increased.

8.1.12 Summary

The information presented in this section demonstrates that each facility in the proposed complex is safe to construct and operate and would pose risks that do not exceed levels consistent with a similar industry. A review of the impacts against applicable standards demonstrate a wide safety margin for complex construction and operation. Impacts appear to be consistent with those projected and encountered on similar projects. The radiological impacts, both occupational and environmental are as totalled in Table 8.7.

Table 8.7. Plutonium Complex Radiological Impact Summary

Individual Facility	Occupational Dose (Person-Rem/Yr)	Maximum Offsite Dose (mRem/Yr) 400 meters	Maximum Offsite Dose (mRem/Yr) 10 km
Fuel Fabrication Facility	0.5	2.1×10^{-1}	6.8×10^{-3}
1 ABWR	98	1.78×10^{-19}	1.28×10^{-1}
Spent Fuel Storage	3	2.45×10^{-4}	1.93×10^{-2}
Total	101.5	2.1×10^{-1}	1.54×10^{-1}
Tritium Production (Incremental Dose)	2.4	3.36×10^{-19}	3.56×10^{-1}

8.2 SCHEDULE FOR NEPA ACTIVITIES

The process for obtaining a project Record of Decision and approval to proceed through the National Environmental Policy Act (NEPA) requires significant time and resources. This section provides a discussion of what can be expected with respect to the NEPA process cost and schedule.

If the complex is to be licensed by the Nuclear Regulatory Commission, the requirements of Section 102(2)(C) of NEPA and 10 CFR Part 51 (Licensing and Regulatory Policy and Procedures for Environmental Protection) must be followed.

- **NEPA Requirements**
 - Publish a notice of intent to prepare an Environmental Impact Statement (EIS).
 - Prepare Environmental Report per 10CFR51.20.
 - Prepare Draft EIS (DEIS) per 10CFR51.22.
 - The DEIS provides a determination of:
 - The environmental impact of the proposed action.
 - Any adverse environmental effects which cannot be avoided should the proposal be implemented.
 - Alternatives to the proposed action.
 - The relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity.
 - Any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented.
 - Distribute DEIS for comment.
 - Resolve DEIS comments.
 - Issue Final EIS (FEIS).
- **NEPA Activity Timetable** - This section identifies the NEPA minimum time requirements and the more likely timetable that has evolved since implementation of the Act.
 - **Minimum NEPA Timetable** - The National Environmental Policy Act (NEPA) contains three basic requirements that have a direct impact on the Plutonium disposition complex. The first is that an environmental impact statement (EIS) would have to be prepared to identify and assess any significant impact the project may have on the environment. Considering past history with similar facilities, an EIS would have to be prepared for either the complex as a whole or for each individual facility.

The second primary requirement is a 90 day public comment period after publication of the Draft EIS in the Federal Register. During this period, no decision on the proposed action can be made or recorded.

The third requirement is a 30 day comment period after publication of the final EIS in the Federal Register. During this period, no decision on the proposed action can be made or recorded.

- **Historical NEPA Timetable** - Historical experience shows that a significantly longer schedule can be expected to reach a record of decision. The more realistic schedule is as follows:

Notice of intent to prepare EIS	No time requirement
Scoping meetings conducted	30 days
Prepare Draft EIS	No time requirement
Draft EIS comment period	90 days
Comment resolution and response	120 days
Final EIS released	No time requirement
Record of decision	30 days from FEIS release
Design and construction	

The total time period from start to obtaining a record of decision is typically 2 years.

* Bold indicates time estimates, not statutory requirements.

8.3 WASTE MINIMIZATION

Minimization of the generation of low-level radioactive waste, and of the volume to be shipped to the waste disposal sites, has been a key objective in the design of the ABWR plant. The advanced but proven ABWR technologies will be utilized by the plutonium disposition reactor plant, including the mixed oxide fuel fabrication facility and the ex-

reactor spent fuel storage facility, in order to minimize the input to the waste disposal sites in accordance with DOE Order 5820.2A, September 26, 1988.

8.3.1 ABWR Plant Radwaste Minimization

The ABWR plant has been designed to reduce inputs to waste treatment facilities, to utilize advanced, proven technologies for waste treatment, and to minimize the volume of the waste. Specifically, the radwaste volume has been minimized as shown below:

	<u>Year</u>
• Current U.S. Operating Plants	~ 233m ³
• Latest Japanese Plants	~ 200m ³
• ABWR Goal	~ 100m ³

The reduction in inputs to the waste treatment facilities is accomplished through the use of improved condensate treatment systems using hollow fiber filters and deep beds with no resin regeneration, by using high quality condensers and alloy steels, by improved housekeeping and operating practice, and by improved availability of equipment. The use of hollow fiber filters is schematically illustrated in Figure 8.3.1. Solidification with cement-glass, incineration of dry waste, and compaction of non-combustible wastes all lead to a minimization of waste from the reactor plant.

Significant system simplifications of the ABWR radwaste offgas system which eliminated desiccant dryers leads to less equipment and thereby less volume in eventual disposal. Additional radwaste minimization measures will be implemented as new technologies evolve and operating experience with the ABWR plants currently being built in Japan is evaluated.

Figure 8.3-1 Hollow Fiber Filters for Reduction in Radwaste

- *Used in Japanese BWRs since 1986*
- *Sub-micron filtration*
- *Large filtration area in a compact unit*

Reduced crud/occupational exposure
Reduced input to radwaste system

8.3.2 Fuel Fabrication Plant Radwaste Minimization

Techniques that minimize the generation and treatment of radioactive wastes from the fuel fabrication plant will be implemented and in many cases these are similar to the techniques to be utilized in the reactor. Improved housekeeping and operating practices and techniques will reduce waste generation. Treatment of water and gases which may contain radioactive elements can be processed with techniques similar to those in the ABWR. Incineration of appropriate combustible wastes, compaction of non-combustible wastes and solidification of wastes in cement-glass will all be utilized.

The areas in the fuel fabrication plant that historically have tended to generate difficult to handle or large quantities of waste are the chemical analysis laboratory and the scrap recycle areas. Utilization of accurate non-destructive assay techniques will reduce the number of the chemical analyses required and thereby reduce the amount of contaminated chemicals that must be recovered. In the scrap fuel area, the amount of waste generated will be minimal because the amount of scrap fuel will be very low. Since the prime objective is to dispose of the plutonium and not to obtain the maximum burnup or lifetime possible with the fabricated mixed-oxide fuel, almost all fuel pellets made will be accepted for irradiation. Utilizing the latest fabrication techniques and with a broad fuel specification no scrap fuel should be generated. The major source of fuel, which may need recycling, would be generated during an equipment or process upset. Most equipment or process upsets should not generate scrap fuel because the planned on-line computer monitoring and control of the equipment or process should initiate actions to prevent damage to the fuel being fabricated. Additionally, if needed, special treatments, like resintering if pellet density were found to be low, can be done to recover the affected fuel.

8.3.3 Ex-Reactor Spent Fuel Storage Radwaste Minimization

Several items incorporated in the design of the ex-reactor pool will contribute toward waste minimization. Solid waste compaction is provided to typical facility waste (clothing, rags, wipes, plastic). Inlet filters on the supply air system reduce the amount of dust within the

facility. This reduces the dust loading on HEPA filter reducing the change-out requirements, thus reducing the volume of radioactive waste. The reduction of dust also reduces the amount of sludge and silt in the water pools, also reducing the amount of radioactive sludge to be removed from the pool cleanup filtration system. The use of demineralized water with chemical adjustment for pH reduces the degradation of the fuel cladding, greatly reducing the potential release of radioactive materials to the pool water as a result of a breached/leaking fuel assembly. Equipment is provided for detection and confinement of a leaking fuel assembly which reduces levels of contamination of the pool water. The pool cleanup system of filters and demineralizer maintain pool water clarity such that fuel handling activities can be performed at safe depths underwater to reduce personnel exposures.

9.0

DEVELOPMENT REQUIREMENTS

9.1 SUMMARY OF DEVELOPMENT REQUIREMENTS

The proposed means of dispositioning plutonium, by conversion to mixed-oxide fuel and its utilization in an ABWR, entails no new technology and requires a minimum of development while embodying already developed methods for implementing verifiable safeguards provisions. The entire process from the removal of pits from weapons through conversion to oxide, fabrication of mixed-oxide fuel, preparation of fuel assemblies and irradiation in the ABWR, to final disposition as irradiated fuel can be executed in a timely fashion precisely because of the very low level of development required to implement such a concept. Most all the elements that are required for the implementation of this concept have previously been developed and tested in a large scale/industrial setting and few additional areas have been identified that require research and development in the conventional sense of this term. For example, mixed-oxide fuel fabrication facilities exist in a number of countries (Ref.9.1) and in the U.S., mixed-oxide fuel has been fabricated in large quantities as part of the Liquid Metal Reactor Programs (Ref.9.2). The use of mixed-oxide fuel in BWRs was tested more than 20 years ago at Big Rock Point (Ref.9.3) and this study included a U.S. AEC sponsored integrated safeguards experiment to verify that sufficient redundancy could be incorporated into the fuel fabrication program to verify accountability provisions. Mixed-oxide fuel fabrication plants employing recycled plutonium which present a significantly greater challenge are currently in operation in other countries (Ref.9.4), so the level of development required for converting weapons plutonium to mixed-oxide fuel with none of the fission products generally present in recycled plutonium poses much less of a challenge. In the case of the reactor, two ABWRs are currently under construction and no significant changes to the plant system are contemplated. The only exception to the layout of the ABWR Reactor/Turbine Island is the interface with an additional ex-reactor spent fuel storage pool. Therefore, the specific areas listed below, identified as requiring development, are primarily related to confirmatory testing and analyses rather than conventional research and development. In some cases, specific tasks

are listed here not so much because they require any new breakthroughs but because they are considered to be on the critical path for implementing this project and as such it would be useful to get the confirmatory results at an early date.

The development requirements given here have already been briefly identified in the respective sections where they arise, however, they are described in more detail here, together with an estimate for schedule and cost.

9.1.1. Safeguards

The proposed method of dispositioning plutonium, by irradiating it as a fuel, perhaps constitutes the only method of safeguarding plutonium that has a proven experience base. Nevertheless, there are some key aspects of the present scenario which differ from conventional safeguards and require additional consideration. These primarily relate to the need for safeguarding very large quantities of plutonium which in turn might require more stringent measures in material accountability and the need for a high level of transparency and reciprocity in case dispositioning occurs simultaneously in the U.S. and CIS. No new hardware development, for example in assaying, is anticipated at this time; however, the following analyses/planning for safeguards is listed as part of development:

1. **Dispositioning Process Simulation:** Computer simulation of the dispositioning process from the pit to irradiated bundle, time sequencing the material flow, hold-up points, incorporating downtime for maintenance and inventory control, checking material characterization and accountability, simulating unavailability of specific plant equipment and statistical errors in assays to identify needs for redundancy, to detail levels of transparency and components of reciprocity that need to be agreed upon if dispositioning takes place in two countries.

The output of this computer simulation is expected to provide the following information for the detailed design:

- a) **Statistical sampling requirements in regard to accountability**
- b) **Redundancy needs for material accountability**
- c) **Storage requirements at various points in the process**
- d) **Points of maximum vulnerability, that is points where the loss of material might be hardest to detect because of margins of error in assay and transit times**
- e) **Possible impact of levels of transparency and frequency of complete physical inventory. Complete physical inventory with the process lines in the off condition would tend to increase the downtime but is expected to provide the best verification for accountability.**
- f) **Detail areas relative to lead/lag requirements and reciprocity in bilateral agreements if dispositioning were to occur in the U.S. and CIS simultaneously, and if the dispositioning rates were to differ due to unavailability of process equipment.**

This activity is expected to take six months and cost \$300K.

2. **Literature Survey and Site Visits to Document Latest Advances:** This task will involve a survey of Safeguards implementation at operating plants and a survey of literature and equipment for implementing safeguards and their applicability to this project. In particular, the safeguards implemented in the weapons program for guarding fissile material will be examined for potential application to this program. This activity is expected to take nine months and cost \$150K.
3. **Software Requirements:** Software requirements for implementing safeguards including reciprocal arrangements for release of material from one station to the next, from pits to irradiated assemblies. In handling plutonium bearing material and its movement through the process cycle, it is anticipated that much of the process will be automated. In implementing safeguards provisions, it will occasionally be necessary for movement of the material in a fashion other than the specified process flow steps pointing in the forward direction. An example would be where a previous

process step has to be repeated or where the material has to be removed from a downstream station and reintroduced at an earlier station for recycling. These are generally recognized to be vulnerable points for diversion and it might be necessary for two or more persons to authorize such a movement. These conditions (sign-offs) have to be incorporated in the software of the automated equipment. This development task will identify potentially such areas and the type of software needed for such actions. This activity is expected to take 12 months and cost \$240K.

9.1.2 Fuel Cycle

Three tasks have been identified in the fuel processing and fabrication area. These are:

1. **Confirmatory Tests/Analysis for Rapid Conversion of Plutonium to PuO₂:** In order to enhance safeguards, it is desirable to convert the metallic plutonium to plutonium-oxide in a rapid and cost effective manner. Although a number of processes are available to do this, a direct controlled oxidation as developed by Lawrence Livermore National Laboratory, appears to be very promising. The activities encompass initial literature survey, analysis and process specifications and laboratory scale testing to confirm the advantages of this process. This will take 12 months and is estimated to cost \$750K.

2. **Confirmatory Tests for MOX Pellet Fabrication:** Only one area of MOX pellet fabrication is expected to require confirmatory testing. This relates to the possible evaporation of americium in high americium fuel during the sintering process. Features will need to be incorporated to trap this americium and ensure that all the environmental regulations are met. A simulated sintering run of depleted uranium fuel with either americium or another element to simulate americium will need to be conducted. Several sintering furnaces which have been used with plutonium fuel are available in many national laboratories and one of these will be used for this test.

To design, procure material and establish interfaces is expected to take six months.

The sintering tests could be conducted in three months followed by examinations in three months. The cost for the first activity is estimated to be \$300K, the sintering test and post-irradiation examinations are expected to cost another \$300K.

3. **Confirm Fabrication Procedures for (U,Pu,Gd) Oxide Pellets:** Urania fuel with gadolinia (burnable poison) is now routinely fabricated by GE. The addition of plutonium to (urania-gadolinia) to fabricate (urania-plutonia-gadolinia) is not expected to pose any problems. Control of oxygen has been the primary concern in the production of (urania-gadolinia) pellets. Given that plutonium-oxide could exist in a wide band of stoichiometries, it should be less difficult to sinter (U,Pu,Gd) oxide. However, it is necessary to confirm this by test. Sintering furnaces which could utilize plutonium are available in a number of national laboratories.

The initial phase, to design the test, establish interfaces and procure necessary materials is expected to take six months and cost \$300K. The sintering test and post-sintering examinations are expected to take six more months and an additional \$300K.

9.1.3. Reactor System and Core Analysis

Little, if any, development is needed in the reactor system area since the plant specifications are the same as for the standard ABWR product. The only changes pertain to the use of MOX fuel instead of the standard Urania fuel and the addition of a secondary storage pool for discharged assemblies. The development requirements therefore relate to these two areas:

1. **Develop Core Analysis Methods and Optimize Core Designs:** A number of options have been incorporated in core design to provide maximum flexibility during implementation. The primary focus during this phase of the study was on full mixed-oxide cores as a means of both dispositioning the plutonium quickly and to derive the energy benefit from this fuel. Additional core analysis would be required both

to complete cycle-by-cycle analysis as well as to further optimize this design.

The first phase of core analysis to further optimize options is expected to take three months and cost \$150K while the second phase of core analysis to complete cycle-by-cycle analysis is expected to take 12 months and cost \$500K.

2. **Develop Approach for Bundle Reuse and Outline Confirmatory Tests for Reuse:** A key option that has been proposed pertains to quick dispositioning followed by extended use of the discharged fuel to utilize the plutonium for energy production. Although bundles from spent fuel pools have been reused after nearly eight years of storage, the proposed scenario envisions reuse after very long periods of storage, approaching 20 or more years. It is necessary in these cases to define the quality assurance and confirmatory test steps that may be needed including aspects related to core management, spent fuel pool water chemistry control and qualification steps prior to reuse of the bundle. In addition, a plan for confirming the possibility of reuse by lead tests on BWR discharged bundles is possible.

A number of activities have to be conducted each of which is expected to take the specified durations with the associated costs:

- Outline reuse verification plan: 2 months, \$75K
- Evaluate core management aspects: 6 months, \$150K
- Spent fuel pool water chemistry studies: 6 months, \$100K
- Specify confirmatory steps prior to reuse: 4 months, \$75K
- Test plan for reuse with existing BWR bundle in storage: 4 months, \$100K
- Conduct reuse confirmatory test and needed post-irradiation examinations:
Up to 12 months, \$ 450K

However, many of these activities will be conducted in parallel with a total schedule time for this task of 24 months.

3. **Complete Core Analysis for Very High Plutonium Content ($\sim 20\%$) Cores:** A partial mixed-oxide core design with high plutonium fractions, approaching 20 percent, has been proposed for very quick dispositioning under the spiking alternative. These rods have a high gadolinia loading as well, to counteract the high plutonium content and balance the reactivity. This particular core design requires a minimum mixed-oxide throughput for dispositioning in the quickest time. Additional nuclear analysis are however required to meet all the design criteria. In addition, under severe accident conditions, the possibility of gadolinia separation from plutonium and its impact needs to be examined further. It is expected that the analyses and results from high plutonium cores such as liquid metal reactors will be used to support the severe accident analysis scenario for this option.

Optimization of the high plutonium core and preliminary analyses/evaluations to describe its behavior under severe accident conditions, based on available data is expected to take six months and cost \$250K.

If the high (20 percent) plutonium content option is not chosen, this development element does not arise. The full mixed-oxide cores used for the other options are judged at this time not to require any additional effort because of the low plutonium enrichment levels.

4. **Lead MOX Bundle Test in a BWR Reactor to Confirm Performance:** Although mixed-oxide fuel has been tested and found to perform satisfactorily in BWRs, a lead mixed-oxide assembly test is desirable. No critical experiments are planned at this time as the core designs are well within the envelope of existing data. The lead mixed-oxide bundle test is expected to provide sufficient data to confirm the nuclear analysis.

The lead mixed-oxide bundle test is expected to take up to 12 months to design and establish interfaces with the test reactor facility, up to 36 months to test with another 12 months for completion of post-irradiation examinations. The total cost for this

activity is estimated at \$1500K, with initial results becoming available after 36 months.

9.1.4 Secondary Spent Fuel Pool

Detailed Design of Secondary Pool and Transfer Tube: The design of the ex-reactor spent fuel pool is based on already existing NRC Licensed designs for the transfer tube arrangement currently in use. Further detailed design will be needed to implement this concept and to mate this arrangement with the standard ABWR design. Again, this is not an area requiring development in the conventional sense, but needs to be addressed to assure it is not the critical path.

The review of details of the transfer tube and secondary pool storage design is expected to take 6 months and cost \$250K.

9.1.5 Tritium Option

1. **Completion of Target Rod Development Program:** The Tritium Target Development Program (TTDP) was terminated before all the activities required for deployment of an LWR tritium production target were complete. However, as part of the TTDP closeout activity a program completion strategy (Ref.9.5) was developed which included the following three phases:

- Phase I Completion of the target system development, lead test assembly fabrication and evaluation of deployment options
- Phase II Preparation for deploying the selection option
- Phase III Deployment of the selected production option

In the context of the plutonium disposition study, only the following parts of Phase I are

considered development since the core design and plant evaluation activities in Phase I and all of Phases II and III are included within the project costs for the plutonium disposition complex.

- Finalizing the target component fabrication parameters and specifications and conducting vendor qualification testing to verify that the required quality is maintained at the desired production rates.
- Complete target rod design criteria and performance modeling activities.
- Constructing and operating a prototype full-length extraction furnace using deuterium-loaded getters for optimizing process parameters.
- Fabricating, irradiating and examining one or more lead test assemblies.

Cost and Schedule: It is estimated that the above activities will require three years to complete (including the lead test assemblies irradiation and examination) at a total cost of \$35 million. This is less than the \$45 million listed in the TTDP completion strategy due to inclusion of the planning, design and evaluation activities within the overall project cost for the plutonium disposition complex.

2. **Complete Core Design for a Single Cycle Design:** During the current phase, the equilibrium cycle for tritium production has been defined. This approach will require additional target rod development as only a third of the rods in the core produce the requisite annual production levels and the tritium production in the individual rods exceed the currently available database. Two alternatives exist, one to remove the target rods from each of the core assemblies at the end of each cycle to be replaced by new target rods, and another to employ a single cycle core. In the latter case, a new core of fuel will be needed at the end of each cycle. The equilibrium core analysis is needed as a starting point to prescribe the type of fuel enrichments that are needed for a single cycle core. Again, no additional

development in the conventional sense is needed, but further analysis will have to be performed to define this single cycle core, if this option should be chosen. The additional analysis to define the single cycle tritium core is expected to take six months and cost \$125K.

3. **Automated Bundle Disassembly Hardware:** The ABWR fuel bundle is designed to be disassembled remotely using existing manual tooling. However, this operation was intended for occasional use rather than a high volume production activity. While experience indicates that a single crew can process four bundles per shift, which is more than adequate, it is proposed to develop a semi-automated disassembly machine to minimize worker exposure, reduce operating costs and minimize the potential for errors and/or damage to target rods during disassembly. This would involve construction and testing of a full size prototype which would require six months and \$500K.

9.1.6 Infrastructural

Although little or no development is needed in the technology areas, a number of infrastructural areas require further scrutiny for the efficient deployment of this proposed method for dispositioning plutonium.

Utility Participation: The turbine island is identified to be owned by a Utility. Potential participation by an utility should be firmed up with special emphasis on the interfaces between the Reactor and Turbine Island during design, construction, operation and post-disposition operations which arise from the use of mixed-oxide fuel. Specifically, this could relate to the additional costs in operating the Turbine island arising from increased security/safeguards arrangements in the balance of the complex.

International Deployment: A number of areas require further analyses potentially resulting in formal bilateral agreements. These include levels of transparency and reciprocal arrangements in safeguards verification, resource sharing in implementation, applicable

environmental regulations and handling of any delays in the dispositioning as it is unlikely that dispositioning will take place at the exact same rate at two different sites. These are specific to the proposed dispositioning method for plutonium and are in addition to other numerous agreements that doubtless will be negotiated in executing a project such as this in two countries.

Infrastructure development costs and schedules for the two areas listed here will be developed once a formal request for implementation of the dispositioning project is issued.

9.2 KEY CHALLENGES

The following items, taken from the list of activities under 9.1 are thought to constitute the key challenges in implementing the proposed method of plutonium disposition:

1. Safeguards simulation to identify key drivers for bilateral technical agreements reactive to safeguards.
2. Software requirements for safeguards implementation.
3. Confirmation tests for rapid conversion of plutonium to PuO_2
4. Confirmation of fabrication process for (U, Pu, Gd) Oxide
5. Develop approach and confirm reuse of MOX bundles after long term storage
6. Lead MOX assembly test in BWR
7. Completion of target development program for tritium option
8. Infrastructural development issues

9.3 COST AND SCHEDULE FOR DEVELOPMENT

The cost and schedule for the various development activities listed earlier are summarized in Section 10.

9.0**REFERENCES**

- 9.1 Nuclear Technology, Vol. 102, April 1993, H. MacLeod and G. Yates, "Development of Mixed-Oxide Fuel Manufacture in the United Kingdom"; J. Kellerman, "Plutonium Processing at the Siemens Hanau Fuel Fabrication Plant"; P. Deramaix, D. Haas and J. Van De Velde, In-Pile Performance of Mixed-Oxide Fuel with Particular Emphasis on MIMAS Fuel".
- 9.2 T.T. Claudson and R.E. Bardsley, "The FFTF Fuel Story," JSD:-SA-1959, October 1979
- 9.3 GE-NEDC-10747, Plutonium Utilization in Boiling Water Reactors, Phase II, November 1972.
- 9.4 M. Zendel, IAEA, Table I in "Experiences and Trends in Safeguarding Plutonium Mixed-Oxide Fuel Fabrication Plants," JNMM, February 1993.
- 9.5 W.J. Apley, "Tritium Target Development Project Executive Summary Topical Report," PNL-8142, Appendix G, September 1992

SECTION 10

COST AND SCHEDULE

10.1 COST ESTIMATES

10.1.1 Cost Estimate Scope

Cost estimate development for the study was based on the DOE Plutonium Disposition Study Requirements Document dated February 19, 1993 and the DOE Scope of Work document, attachment I of the DOE contract DE-ACO3-93SF19681, titled "Evaluation of the Potential for Reactor Technologies to Consume Excess Plutonium".

Cost estimates required by these documents include development and pre-operational costs, capital costs, operating and maintenance (including fuel) costs, costs for conversion to tritium production, costs for calculation of revenues from the sale of steam or electric power, costs for facilities outside the reactor complex, and cash expenditure data with time.

The estimate format and scope follow the DOE Cost Estimate Guidelines (Ref. 10-1). All the values presented here are in constant 1992 dollars. The format and scope for the costs for calculation of revenues from sale of steam or electric power follow the directions from Oak Ridge National Laboratory for the plutonium disposition study (Ref.10-2).

10.1.2 Estimate Assumptions

The major assumptions made in developing the cost estimates included the following:

- The reactor site is the Electric Power Research Institute (EPRI) standard hypothetical east/west central site for nuclear power plants at Kenosha, Wisconsin per Appendix F of the DOE Cost Estimate Guidelines.
- The Plutonium Disposition Reactor (PDR) will be owned and operated by DOE

to produce steam and the electricity generating Energy Conversion Area (ECA) of the plant will be owned and operated by a third party.

- The ABWR plant deployed for the project will be essentially identical to the design and configuration of the commercial standard ABWR design.
- The MOX fuel fabrication facility design will employ remote plutonium processing operations within glove box enclosures with hands-on maintenance capability.
- The engineering for the first-of-a-kind plant has been completed in Japan, and design to U.S. standards will be started by the time the project Title I design starts.
- Substantial environmental and safety analyses have been completed at the time of the start of project design.
- Funding is available as required to support uninterrupted design, testing, construction, installation, checkout, and plant startup.
- No unique nuclear or conventional licensing restrictions or delays will affect plant design, construction or operation approval.
- The use of factory assembled modules is utilized to the maximum extent to reduce field construction costs.

10.1.3 Preoperational Costs

10.1.3.1 Estimating Approach to Preoperational Costs

The approach to estimating development costs was to assign reasonable costs to each

defined development program based on the defined scope of development activities and the expected schedule for each program. Scope definition was based on discussions with GE senior technical specialists as well as development laboratory personnel to define required program manpower, hardware, and facility requirements.

Pre-operational costs for pre-Title I engineering; safety and environmental regulatory activities; test operations and startup; plant acceptance; operations, maintenance and test procedures; and management / administration were estimated using historical data from similar DOE and commercial nuclear projects.

10.1.3.2 Preoperational Cost Results

Preoperational costs including safety and environmental submittals, plant mechanical completion and acceptance testing, and preoperational tests and startup were requested in the DOE requirements document for this study. However, as stated in the referenced DOE Cost Estimate Guidelines, the Field Supervision and Field Office Services Account includes costs for plant startup and test (Account 934). Also, the definition of the Owner's cost account 941 for management, engineering, integration and QA/QC, includes costs for licensing and account 944 includes costs for staff training and startup. Therefore, even though these costs are presented separately in this study as preoperational costs, they should not be considered costs which are additive to estimated capital costs.

As stated in Section 9.0 of the report, costs requested by DOE for research and development consist only of confirmatory development costs. No research programs are necessary to provide data for unproven concepts. Table 10.1-1 summarizes the development programs required, their expected durations, and the estimated costs for each program.

Estimated preoperational costs (including development costs) for the DOE Plutonium Disposition Reactor complex as well as the estimated preoperational costs for the third party energy conversion area are shown in Table 10.1-2.

Table 10.1-1 ABWR Plutonium Disposition Reactor Development Requirements

ITEM	EST. COST	EST. DURATION
SAFEGUARDS		
Computer simulation of disposition process	\$300K	Six months
Literature survey applicable to project	\$150K	Nine months
Software development	\$240K	Twelve months
Subtotal	\$690K	Twelve months
FUEL CYCLE		
Pu to PuO ₂ conversion confirmation	\$750K	Twelve months
MOX pellet fabrication confirmation	\$600K	Six months
U,Pu,Gd oxide pellet fab confirmation	\$600K	Twelve months
Subtotal	\$1.95M	Twelve months
REACTOR SYSTEM AND CORE ANALYSIS		
Develop core analysis methods and optimize core design	\$650K	Fifteen months
Develop bundle reuse option	\$950K	24 months
High content Pu core analysis	\$250	Six months
Lead MOX in-reactor bundle test	\$1500K	Five years
Subtotal	\$3.35M	Five years
SECONDARY SPENT FUEL POOL		
Transfer tube/fuel pool studies	\$250K	Six months
Subtotal	\$250K	Six months
TRITIUM OPTION		
Completion of Target Rod Development	\$35M	Three years
Complete core design for single cycle	\$125K	Six months
Develop automated bundle disassembly	\$500K	Six months
Subtotal	\$35.625M	Three years
TOTALS	\$41.8M	FIVE YEARS

TABLE 10.1-2 PREOPERATIONAL COSTS (\$M)

FACILITY/ACTIVITY	REF.SPENT FUEL	SPKING REF.CASE	DESTRUCT. 6 REACTORS	DESTRUCT. 2 REACTORS
Plutonium Disposition Reactor (PDR)				
Development	\$4.29	\$4.29	\$4.29	\$4.29
Pre-Title I Design & Engineering (Spent Fuel Storage)	\$1	\$1	\$1	\$1
Safety/Environmental Submittals	\$13	\$13	\$13	\$13
Other Preoperational Costs (1)	\$25	\$25	\$84	\$40
MOX Fuel Fabrication Facility				
Development	\$1.95	\$1.95	\$1.95	\$1.95
Pre-Title I Design & Engineering	\$5	\$5	\$5	\$5
Safety/Environmental Submittals	\$7	\$7	\$7	\$7
Other Preoperational Costs (1)	\$26	\$26	\$26	\$26
Energy Conversion Area (ECA)				
Development	-	-	-	-
Pre-Title I Design & Engineering	-	-	-	-
Safety/Environmental Submittals	Included with Reactor	Included with Reactor	Included with Reactor	Included with Reactor
Other Preoperational Costs (1)	\$10	\$10	\$35	\$15
TOTALS (\$M)	\$93	\$93	\$177	\$113

NOTE (1): Other prep. costs include mechanical completion, acceptance testing, preop. testing, startup, O&M/testing procedures, mgmt & admin.

10.1.4 Capital Costs

10.1.4.1 Capital Cost Estimate Approach and Bases

The estimating approach to developing capital costs for the study was to establish a base case capital cost estimate for the spent fuel reference case and use this estimate as a basis to parametrically adjust capital costs for the major cases presented in the report.

Capital costs for the base case reactor complex (excluding the fuel fabrication facility, the spent fuel storage facility, and the facilities required for conversion to tritium production) were developed from the GE ABWR capital cost estimate conducted in the late 1980's in the standard DOE format using the DOE Energy Economic Data Base (EEDB) program code of accounts.

This GE estimate was for an nth-of-a-kind (NOAK) 1350 MWe ABWR plant. The estimate was based on historical costs from past boiling water reactor experience and the ABWR design as it existed at that time. Equipment costs, craft man-hour requirements and quantities of site materials were estimated in 1987 dollars at the three-digit EEDB account level using unit rates taken from the DOE Phase IX Update (1987) Report for the EEDB Program (EEDB-IX), report DOE/NE-0091.

The direct capital costs in the GE estimate were escalated to January 1, 1992 dollars using escalation rates from Table 2.5 of the DOE Cost Estimate Guidelines. Since the original GE estimate was for an NOAK plant, capital costs for the nuclear steam supply system (NSSS) and other factory equipment and site labor were adjusted to reflect a third- of-a-kind, first commercial U.S. plant. This designation reflects the experience gained in the design and construction of the two ABWR Japanese units currently under construction. Equipment, material and labor costs were broken out per the DOE Cost Estimate Guidelines for the spent fuel base case.

Direct capital costs were then added for the fuel fabrication and spent fuel storage facilities

as well as direct costs for changes to the complex support and utility systems required for the reference spent fuel case. Direct cost estimates for the new facilities were based on the pre-conceptual designs developed for these facilities. Costs for these new facilities were based primarily on historical costs or detailed estimates for equipment and structures from similar projects. Where details of equipment or facility design were not developed, proven cost estimating techniques were used.

Indirect costs for construction services, AE home office engineering services, and field office supervision and services (accounts 91 through 93) were developed from algorithms developed by Bechtel for the Advanced Liquid Metal Reactor (ALMR) nuclear facilities based on in-house historical engineering, procurement and construction experience. Owner's expenses (account 94) were estimated as a percentage of direct and other indirect costs. Reactor manufacturer's home office engineering/services is included in the reactor plant equipment account (account 22A).

Capital cost contingency has been included in accordance with the DOE Cost Estimate Guidelines for the major estimate cost items. The total weighted average contingency is less than 17 percent of the total base construction cost.

Capital costs for study cases other than the spent fuel reference case were developed by parametrically adjusting the base case estimate based on the known equipment and facilities changes from the base case.

10.1.4.2 Capital Cost Results

The spent fuel base case estimate detail defined to the two-digit EEDB account level is shown in Table 10.1-3. The table format follows the DOE Cost Estimate Guidelines and shows factory equipment, site labor hours and site labor and materials costs separately for each account item. Note that accounts have been added to the Structures and Improvements and Reactor Plant Equipment accounts, (21 & 22 accounts, respectively) for the mixed-oxide fuel fabrication/plutonium oxide storage and separate spent fuel storage

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TABLE 10.1-3
ONE REACTOR - BASE CASE
SPENT FUEL (FUEL RE - USE)

ESTIMATE BASIS / QUALIFICATIONS

PRE - CONCEPTUAL LEVEL SCOPE & ESTIMATE

1350 MW GE ABWR / PU DISPOSITION FACILITIES			
(DOE / EDRB BASIS - JANUARY 1992 \$ / PRE - CONCEPTUAL			
GE & SAIC / BECHTEL ESTIMATE - APRIL 1993			
FACTORY	SITE	SITE	TOTAL
EQUIP	LABOR	LABOR	1993 COST
\$1,000s	HOURS	\$1,000s	\$1,000s

ARWR - PLUTONIUM
DISPOSITION STUDY
DESCRIPTION

ESTIMATE SUMMARY

SHEET 1 OF 1

EDRB 2 - DIGIT EST DETAIL

FOR % ECA %

ITEM		1 AND 2 AND 3		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0		0	
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structures and equipment. Also, adjustments have been made to the original ABWR estimate to allow for additional yardwork, perimeter safeguards and security, and ultimate heat sink structures associated with the plutonium disposition mission of the reactor. Utility, electrical, and service systems associated with the fuel fabrication and spent fuel facilities have been included in the 21 and 22 accounts. Costs have been estimated to the overnight cost level as directed by DOE.

Table 10.1-4 is the two-digit EEDB account summary of the four major cases estimated for the study.

Total overnight costs range from approximately \$2.9 billion for the spiking and spent fuel reference cases to \$4.5 billion for the two reactor plutonium destruction case and \$10.9 billion for the six reactor plutonium destruction case.

For the single reactor cases, costs added to the base construction cost of the escalated ABWR commercial standard plant estimate range from about \$650 million for the spent fuel and spiking cases to approximately \$2.1 and \$7.6 billion for two and six reactors, respectively, for the plutonium destruction cases, excluding contingency. Enriched UO_2 bundles for the initial fuel loading are not required for any of the cases.

10.1.4.3 Bulk Commodity Quantities

Per the DOE Cost Estimate Guidelines and requirements documents, bulk commodity quantities were developed as shown on Table 10.1-5. The quantities shown for the nuclear island were extracted from estimates recently developed for a commercial ABWR study. Quantities for the energy conversion area portion of the plant were derived from historical data from past boiling water reactor projects. Commodities for the fuel fabrication and ex-reactor spent fuel storage facilities were estimated from pre-conceptual design information by factoring from quantities developed from similar facilities.

Table 10.1-4
Capital Cost Estimate Summaries

08 - May - 83		CAPITAL COST ESTIMATE											
ABWR OPTION - PLANT DISPOSITION STUDY; ALTERNATIVES PLANT NOMINAL MWe:		BASIS : GE - ABWR 1350 EEDB - IX ESTIMATE		BASIS : GE - ABWR 1350 ESC / ADJ ESTIMATE		(BASE) 1350 BASE SPENT		1350 BASE SPIKE		8100 BASE DESTRUCTN		2700 BASE DESTRUCTN	
EEDB ACCT		1987 \$ \$1,000s		1992 \$ \$1,000s		1992 \$ \$1,000s		1992 \$ \$1,000s		1992 \$ \$1,000s		1992 \$ \$1,000s	
SCOPE / DESCRIPTION		SPENT FUEL		SPENDING		DESTRUCTN 18 YRS		DESTRUCTN 60 YRS					
20 LAND & LAND RIGHTS		EXCL		EXCL		EXCL		EXCL		EXCL		EXCL	
A YARDWORK (GE)		\$27,383	\$31,224	\$31,224	\$31,200	\$100,000	\$45,000						
B POWER PLNT STRCTRS / ... (GE)		\$182,328	\$210,113	\$210,113	\$210,100	\$1,100,000	\$380,000						
C FUEL FAB / Pu OXIDE FACIL (B)		\$0	\$0	\$180,000	\$180,000	\$168,000	\$158,000						
D SPENT FUEL STORAGE (SAIC)		\$0	\$0	\$13,000	\$13,000	\$0	\$0						
E SF GROS/SCRTY/PERIM FACIL (B)		\$0	\$0	\$16,000	\$16,000	\$51,000	\$20,000						
F ULTIMATE HEAT SINK / ASSOC (B)		\$0	\$0	\$10,000	\$10,000	\$51,000	\$17,000						
21 STRUCTURES & IMPROVEMENTS		\$209,721	\$241,337	\$440,337	\$448,300	\$1,488,000	\$630,000						
A REACTOR PLANT EQPT (GE)		\$347,548	\$488,838	\$488,838	\$488,800	\$2,500,000	\$800,000						
B FUEL FAB EQPT (GE)		\$0	\$0	\$58,700	\$58,500	\$75,200	\$34,500						
C SPENT FUEL STRG EQPT (SAIC)		\$0	\$0	\$48,000	\$48,000	\$0	\$0						
D													
22 REACTOR PLANT EQUIPMENT		\$347,548	\$488,838	\$572,538	\$581,300	\$2,575,200	\$834,500						
A POWER PLANT T/G EQPT (GE)		\$205,083	\$251,823	\$251,823	\$251,800	\$1,350,000	\$480,000						
B CHANGES FOR FUEL FAB / ETC		\$0	\$0	INCL ABV									
23 TURBINE PLANT EQUIPMENT		\$205,083	\$251,823	\$251,823	\$251,800	\$1,350,000	\$480,000						
A POWER PLANT ELECTRICAL (GE)		\$88,804	\$88,807	\$88,807	\$88,800	\$480,000	\$170,000						
B CHANGES FOR FUEL FAB / ETC		\$0	\$0	INCL ABV									
C													
24 ELECTRICAL PLANT EQUIPMENT		\$88,804	\$88,807	\$88,807	\$88,800	\$480,000	\$170,000						
A POWER PLANT MISC (GE)		\$48,704	\$80,374	\$80,374	\$80,400	\$200,000	\$110,000						
B CHANGES FOR FUEL FAB / ETC		\$0	\$0	INCL ABV									
C													
25 MISC PLANT EQUIPMENT		\$48,704	\$80,374	\$80,374	\$80,400	\$200,000	\$110,000						
A POWER PLANT HT FUJCTN (GE)		\$54,155	\$87,320	\$87,320	\$87,800	\$300,000	\$120,000						
B CHANGES FOR FUEL FAB / ETC		\$0	\$0	\$0									
C													
26 MN CONSR HEAT REJECTION SYS		\$54,155	\$87,320	\$87,320	\$87,800	\$300,000	\$120,000						
2 TOTAL DIRECT COST		\$832,719	\$1,178,000	\$1,481,000	\$1,488,900	\$6,363,200	\$2,444,800						
81 CONSTRUCTION SERVICES		\$184,824	\$248,000	\$274,000	\$277,000	\$1,108,000	\$448,000						
82 AE HOME OFFICE ENGNRG / SRVCS		\$80,751	\$112,000	\$335,000	\$338,000	\$800,000	\$435,000						
83 FIELD OFFICE SUPRV / SRVCS		\$83,088	\$105,000	\$123,000	\$124,000	\$300,000	\$200,000						
84 OWNER'S EXPENSES		EXCL	\$187,000	\$288,000	\$288,100	\$854,800	\$388,800						
85 RM HOME OFFICE ENGNRG / SRVCS		(W / 22)	(W / 22)										
8 TOTAL INDIRECT COSTS		\$388,473	\$883,000	\$888,000	\$1,008,100	\$3,084,800	\$1,484,800						
2+8 BASE CONSTRUCTION COST		\$1,201,199	\$1,842,000	\$2,478,000	\$2,504,000	\$9,448,000	\$3,929,600						
CONTINGENCY		EXCL	EXCL	\$410,000	\$414,000	\$1,482,000	\$581,000						
EEDB TOTAL OVERNIGHT COST \$1,000s		EXCL	EXCL	\$2,888,000	\$2,918,000	\$10,930,000	\$4,510,600						
UNIT COST \$ / KWe :		N/A	N/A	\$2,140	\$2,181	\$1,348	\$1,887						

TABLE 10.1-5
ABWR PLUTONIUM DISPOSITIONING REACTOR COMPLEX
BULK COMMODITY LIST*

PLUTONIUM DISPOSITIONING REACTOR (PDR)				FUEL FAB. FACIL.		SPENT FUEL FACIL.		TOTALS	
COMMODITY	UNITS	NI	ECA	SUBTOTAL	NUCL.	INDUST.	NUCL.	INDUST.	INDUST.
Formwork & Decking	SF	1,050,000	800,000	1,850,000	600,000	270,000	93,000	0	1,070,000
	TN	7,000	2,000	9,000	550	230	40	0	2,230
Reinforcing Steel	TN	27,000	5,000	32,000	5,200	2,200	1,200	0	7,200
Embedded Steel	TN	2,200	600	2,800	110	40	250	0	640
Concrete	CY	110,000	100,000	210,000	45,000	20,000	5,500	0	120,000
Large Pipe (>2.5 inch)	LF	64,000	185,000	249,000	7,000	63,000	0	3,800	251,800
	LF	80,000	82,000	162,000	9,000	81,000	0	3,600	166,600
Wire and Cable	LF	2,400,000	3,000,000	5,400,000	160,000	2,740,000	0	91,000	5,831,000
Cable Tray	LF	45,000	50,000	95,000	2,400	45,000	0	1200	76,200

***NOTE: QUANTITIES SHOWN ARE ESTIMATED BASED ON NUCLEAR ISLAND TAKEOFFS, ESTIMATES OF ECA QUANTITIES FOR PAST BWR'S, AND PRE-CONCEPTUAL ALLOWANCES FOR FUEL FAB AND SPENT FUEL STORAGE FACILITIES.**

10.1.5 Operating and Maintenance Costs

10.1.5.1 Operating and Maintenance Cost Estimating Approach

Reactor complex and energy conversion area turbine generation operating and maintenance costs were developed by adjusting costs reported in recent U.S. Council on Energy Awareness (USCEA) and EPRI studies for ABWR plutonium disposition operations. Operating and maintenance costs for the fuel fabrication facility were derived from historical data from operation of similar facilities and from DOE cost guidelines.

Decontamination and decommissioning (D&D) costs were estimated based on USCEA/EPRI studies for the reactor complex. The D&D costs for the fuel fabrication facility were developed by adjusting costs from historical decontamination and decommissioning projects for similar alpha-contaminated facilities.

10.1.5.2 Operating and Maintenance Costs Results

Operating and maintenance cost results are shown in Table 10.1-6.

10.1.6 Costs for Conversion to Tritium Production

Costs required to convert the reactor complex to a tritium production facility are covered in Section 6.7.

10.1.7 Costs to Calculate Steam Revenue to DOE and Electric Revenue to the Energy Conversion Area (ECA) Owner

10.1.7.1 Estimating Approach

Since a basic assumption of the study is that the DOE will own the reactor nuclear island and generate steam for use by a utility owner of the turbine generator/energy conversion

TABLE 10.1-6 OPERATING AND MAINTENANCE COSTS (\$M)

FACILITY/ACTIVITY	REF.SPENT FUEL	SPKING REF.CASE	DESTRUCTION 6 REACTORS	DESTRUCTION 2 REACTORS
Plutonium Disposition Reactor (PDR)				
Operations and Maintenance (1)	\$32M / yr	\$32M / yr	\$180M / yr	\$62M / yr
Decontamination and Decommissioning (2)	\$240M (\$12.2M / yr)	\$240M (\$12.2M / yr)	\$1000M (\$55.1M / yr)	\$400M (\$20.6M / yr)
MOX Fuel Fabrication Facility				
Operations and Maintenance (1)	\$59.6M / yr	\$93M / yr	\$103M / yr	\$33M / yr
Decontamination and Decommissioning (2)	\$50M (\$2.1M / yr)	\$50M (\$2.1M / yr)	\$50M (\$2.1M / yr)	\$50M (\$2.1M / yr)
Energy Conservation Area (ECA)				
Operations and Maintenance (1)	\$24M / yr	\$24M / yr	\$110M / yr	\$44M / yr

NOTES: (1) O&M cost includes on-site staff, maintenance materials, supplies & expenses, off-site technical support, pensions & benefits, regulatory fees, insurance fees, and other G & A. MOX Fuel Fabrication Facility O&M costs include fuel assembly hardware with zero cost for plutonium and uranium.

(2) Annual decontamination and decommissioning costs were based on USCEA/EPRI studies for the reactor complex and historical data from similar D&D projects for the fuel fabrication facility.

area, cost estimates were separated into DOE costs and energy conversion area owner costs. Costs were developed as required by Oak Ridge National Laboratory to calculate steam revenue to DOE and electric revenue to the energy conversion area owner as defined in the memo of April 5, 1993, titled "Calculation of Steam Revenues for Plutonium Disposition Reactors (PDR)", by C.R. Hudson (Reference 10-2). The data requested in Table 3 of the referenced document was developed from the plutonium disposition reactor/energy conversion area information available.

10.1.7.2 Results

Costs for determining revenues for sale of steam and electric power are shown on Table 10.1-7. Electrical capacities shown were derived from the current ABWR electrical single line diagram adjusted for the addition of facilities required for plutonium disposition. Values shown are for a single reactor case. For multiple reactors, the values for gross and net electrical capacity output and total complex annual electrical load should be adjusted by direct proportioning as noted in the Table. Energy conversion area overnight capital costs were derived by separating plutonium disposition reactor/energy conversion area costs by engineering judgement as shown on Table 10.1-3 and annual operating and maintenance costs and interim capital replacement assumptions were derived by adjusting values reported in USCEA/EPRI studies to the ABWR plutonium dispositioning reactor operating scenarios as shown on Table 10.1-6.

It is assumed that the spent fuel disposal fee, which is not included in these costs, will be accounted for in the revenue calculations by Oak Ridge National Laboratory.

10.1.8 Costs for Facilities Outside the Reactor Complex

With the exception of the off-site costs required to convert the reactor complex to a tritium production facility (Section 6.7), there were no other off-site costs identified in the study that are required to support the ABWR disposition reactor complex.

**Table 10.1-7. ABWR Plutonium Disposition Study
Data/Costs Required for Calculation of Revenues***

Operational Dates For Each Unit/Facility	See Schedules - Section 10.2															
**Gross Capacity Output of ECA in MWe (Net of ECA House Loads)	1356 Mwe															
**Net Capacity Output of PDR/ECA Plant in MWe	1297 MWe															
PDR/ECA Plant Annual Capacity Factors	<table><tr><td></td><td colspan="2">Spent</td><td colspan="2"><u>Destruction</u></td></tr><tr><td></td><td>Fuel</td><td>Spiking</td><td>Ref 6 Retr</td><td>Option 2 Retr</td></tr><tr><td>ALTERNATIVE FACTOR - %</td><td>75</td><td>50</td><td>75</td><td>75</td></tr></table>		Spent		<u>Destruction</u>			Fuel	Spiking	Ref 6 Retr	Option 2 Retr	ALTERNATIVE FACTOR - %	75	50	75	75
	Spent		<u>Destruction</u>													
	Fuel	Spiking	Ref 6 Retr	Option 2 Retr												
ALTERNATIVE FACTOR - %	75	50	75	75												
**Total PDR Annual Electrical Energy Requirements in MWHrs (Excluding ECA House Loads)	190X10 ³ MWHrs															
ECA Overnight Capital Cost	See Table 10.1-4															
***ECA Annual Operating/Maintenance Expenses	See Table 10.1-6															
ECA Interim Capital Replacement Assumptions	Assume approximately 0.5% of ECA total overnight capital costs															

*Per the requirements for vendor input (Table 3) of the April 5, 1993 Oak Ridge National Laboratory memo titled "Calculation of Steam Revenues for Plutonium Disposition Reactors (PDRs)"(Ref. 10-2).

**For the two and six reactor cases, multiply these values by 2 and 6 respectively

***Spent fuel disposal fee not included

10.1.9

Cash Flows

Cash flow data as required by Figures A11 through A 15 of the attachment to the DOE Requirements Document were developed as defined. Tables 10.1-8 through 10.1-15 present this data. Cash flow for revenues from the sale of electric power or steam was not included in these tables as these cash flows are to be developed by Oak Ridge National Laboratory per the Oak Ridge National Laboratory letter referenced in Section 10.1.7.

10.2 SCHEDULES

10.2.1 Scope, Approach and Summary

Per the DOE requirements document schedules were developed to show the major activities involved in bringing the reactor complex on-line to disposition the 100 metric tons of plutonium and to produce electricity. The major activities identified are:

- Safety/Environmental/Regulatory Activities
- Research & Development (R&D)
- Design
- Construction
- Preoperational Check-Out and Startup Testing
- Reactor Complex Operations
- Tritium Conversion and Tritium Production
- Facilities Outside the Reactor Complex

The reactor schedules shown are based on previously developed schedules for commercial clients as well as actual experience from the design and construction of the two ABWR units in Japan. The detailed reactor construction schedules were developed from EPRI ABWR guidelines and utilized results of previous Simplified Boiling Water Reactor (SBWR) studies which examined construction tasks and established critical path activities.

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TABLE 10.1 - 8

**ONE REACTOR -
OVERALL PROJECT
CASH FLOW ESTIMATE**

**1350 MWe ABWR - Pu DISPOSITION STUDY :
BASE CASE 1 SPENT FUEL**

**R&D / CAPITAL / O&M COSTS - TOTAL \$
CASH FLOW (M\$) ***

YEARS FROM PROJECT START (Rx TITLE 1)	TOTAL COST CASH FLOW (\$ MILLIONS)	TOTAL COST CUMMU- LATIVE
1	75	75
2	212	287
3	300	587
4	450	1,037
5	690	1,727
6	800	2,527
7	430	2,957
8	198	3,155
9	118	3,273
10	116	3,389
11	116	3,505
12	116	3,621
13	116	3,737
14	116	3,853
15	116	3,969
16	116	4,085
17	116	4,201
18	116	4,317
19	116	4,433
20	116	4,549
21	116	4,665
22	116	4,781
23	116	4,897
24	116	5,013
25	100	5,113
26	100	5,213
27	70	5,283
28	28	5,311
29	0	5,311
30		
31		
32		
33		

> JANUARY 1992 -
* OVERNIGHT * COST LEVEL
(EXCLUDES ESCALATION / AFUDC)

MILESTONE SCHEDULE

- START TITLE I - REACTOR & START CON-
CEPTUAL DESIGN FUEL FAB (PRE-TITLE I)

- FUEL FAB - START OPERATIONS :
FUEL FAB PLANT OPERATES FOR
19.3 YRS @ \$ 59.6 M/YR = ~ \$1150M
- REACTOR / ECA - COMM OPER :
REACTOR OPERATIONS 17.3 YRS
@ \$32 M / YR = ~ \$554 M
ECA (CONVERSION AREA) OPER
17.3 YRS @ \$24 M = ~ \$416 M

DECON/DECOMM FUEL FAB
FACIL - \$50M - OVER 3 YRS
\$50 M

DECON/DECOMM REACTOR
FACIL - \$240M - OVER 3 YRS
\$240 M

- REACTOR OPERATION COMPLETE
- DECON / DECOMM OPERATIONS

TOTAL R&D / PRE-TITLE 1 DESIGN - \$12 M
TOTAL CAPITAL COST - \$2,889 M
(INCL PRE-OP)
TOTAL O&M / D & D - \$2,410 M

* OVERNIGHT * COST ESTIMATE
TOTAL - ABWR PU DISP. : \$5,311 M

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TABLE 10.1 - 9

**ONE REACTOR -
R&D & PRE-OPERATION
CASH FLOW ESTIMATE**

**1350 MWe ABWR - Pu DISPOSITION STUDY:
R & D / PRE-TITLE I EXPENDITURES &
PRE - OPERATIONAL COSTS**

(SPENT FUEL

ONE REACTOR PLANT CASH FLOW (M\$) *

QUARTERS	YR	R&D & PRE-OP ESTIMATES	
FROM		CASH	CUMMU-
START OF		FLOW	LATIVE
Rx CONSTR		(\$ MILLIONS)	
-12	0	3.9	3.9
-11		3.9	7.8
-10		3.3	11.1
-9		3.3	14.4
-8	1	1.4	15.8
-7		1.4	17.2
-6		1.7	18.9
-5		1.5	20.4
-4	2	1.2	21.6
-3		1.4	23.0
-2		1.3	24.3
-1		1.4	25.7
1	3	0.3	26.0
2		0.3	26.3
3		0.2	26.5
4		4.3	30.8
5	4	5.2	36.0
6		6.2	42.2
7		6.1	48.3
8		4.1	52.4
9	5	0.9	53.3
10		0.5	53.8
11		2.0	55.8
12		2.5	58.3
13	6	3.0	61.3
14		4.0	65.3
15		4.0	69.3
16		4.0	73.3
17	7	5.0	78.3
18		6.0	84.3
19		5.0	89.3
20		2.0	91.3
21	8	1.0	92.3
22		0.5	92.8
23		0.3	93.1
24		0.2	93.3
25	9		
26			
27			
28			
29			
30			

> INCLUDES R&D COSTS \$12.3 M PLUS
SAFETY / ENVIRONMENTAL / OTHER
PRE - OP COSTS \$81.0 M

MILESTONE SCHEDULE (MONTHS)

START R&D / TITLE I REACTOR DESIGN (-36)
& FUEL FAB CONCEPTUAL DSGN(PRE-TITLE I)

START TITLE II - Rx DETAIL DESIGN (-24)
& FUEL FAB DETAIL DESIGN

START SITE WORK / FUEL FAB PLANT (-10)

START CNTMINT / RB STRUCTURAL CONC.(+1)

FUEL FAB PLANT COMPLETE (+18)

Rx START PRE-OP TESTS & START-UP(+31)

UNIT 1 POWER PLANT COMPLETE -
START UNIT 1 FUEL LOAD (+48)

UNIT 1 COMMERCIAL OPERATION (+57)

* CASH FLOW ESTIMATE \$93.3M INCL :
DEVELOPMENT REQUIREMENTS - \$12.3 M
(INCL PRE-TITLE I ENGNRG)
SAFETY / ENVIRON / LICENSING - \$20.0 M
OTHER PRE-OPER - \$61.0 M
NOTE : \$81.0 M ALSO INCL W / CAPITAL COST

SIMILAR C/FLOW CURVE FOR OTHER
ONE REACTOR CASE (SPIKE)

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TABLE 10.1 - 10

1350 MWe ABWR - Pu DISPOSITION STUDY :
BASE CASE 1 SPENT FUEL

ONE REACTOR
CAPITAL COSTS
CASH FLOW ESTIMATE

ONE REACTOR PLANT CASH FLOW (M\$) *

QUARTERS FROM START OF Rx CONSTR	YR	CAPITAL COST CASH FLOW (\$ MILLIONS)	CAPITAL COST CUMMU- LATIVE
-12	0	5	5
-11		10	15
-10		20	35
-9		30	65
-8	1	40	105
-7		50	155
-6		60	215
-5		60	275
-4	2	70	345
-3		70	415
-2		80	495
-1		80	575
1	3	90	665
2		100	765
3		120	885
4		140	1,025
5	4	150	1,175
6		160	1,335
7		170	1,505
8		190	1,695
9	5	200	1,895
10		200	2,095
11		180	2,275
12		160	2,435
13	6	140	2,575
14		100	2,675
15		80	2,755
16		50	2,805
17	7	30	2,835
18		20	2,855
19		20	2,875
20		12	2,887.0
21	8	1.0	2,888.0
22		0.5	2,888.5
23		0.3	2,888.8
24		0.2	2,889.0
25	9		
26			
27			
28			
29			
30			

> * OVERNIGHT * COST LEVEL

MILESTONE SCHEDULE (MONTHS)

START R&D / TITLE I REACTOR DESIGN (-36)
& FUEL FAB CONCEPTUAL DESIGN

START TITLE II - Rx DETAIL DESIGN (-24)
& FUEL FAB DETAIL DESIGN

START SITE WORK / FUEL FAB PLANT (-10)

START CNTMNT / RB STRUCTURAL CONC.(+1)

FUEL FAB PLANT COMPLETE (+18)

Rx START PRE-OP TESTS & START-UP(+31)

UNIT 1 POWER PLANT COMPLETE -
START UNIT 1 FUEL LOAD (+48)

UNIT 1 COMMERCIAL OPERATION (+57)

* BASE CASE 1 INCLUDES :
EPC OVERNIGHT COSTS - \$2,889 M
(PRE - OPER ~ \$81 M - INCL W / EPC \$)

SIMILAR C/FLOW CURVE FOR
ONE REACTOR CASE (SPIKE) -
TOTAL: \$2,918 M

TABLE 10.1 - 11

ONE REACTOR O & M AND De&De COSTS CASH FLOW ESTIMATE

CASH FLOW ESTIMATE

CASH FLOW ESTIMATE

CASH FLOW ESTIMATE

> ' OVERNIGHT ' COST LEVEL - PLUS

COMPLETE FUEL FAB STARTUP -
& OPER FUEL FAB 19.5 YRS

FUEL FAB PLANT OPERATES FOR 19.3 YRS @ ~ 59.6M / YR = \$ 1150	1150
--	------

REACTOR OPERATIONS FOR 17.3 YRS
@ \$32 M / YR = ~ \$554 M - 1704

ECA CONVERSION AREA OPER
17.3 YRS @ \$24 M = ~ \$416 M - 2120

DECON/DECOMM FUEL FAB
FACIL - \$50M - OVER 3 YRS - 2170

DECON/DECOMM REACTOR
FACIL - \$240M - OVER 3 YRS - 2410

REACTOR OPERATION COMPLETE

TOTAL O&M / DECON / DECOMM -
OVERNIGHT COST EST \$2,410 M

* SIMILAR CASH FLOW CURVE
FOR ONE REACTOR CASE (SPIKE)

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TABLE 10.1 - 12

2700 MWe ABWR - Pu DISPOSITION STUDY :
CASE 5 Pu DESTRUCTION (60 YEARS) *

**TWO REACTORS
CAPITAL COSTS
CASH FLOW ESTIMATE**

TWO REACTOR PLANT CASH FLOW (M\$) *

QUARTERS FROM	YR	CAPITAL COST	CAPITAL COST
START OF Rx CONSTR		CASH FLOW	CUMMU- LATIVE
(\$ MILLIONS)			
-12	0	5	5
-11		15	20
-10		25	45
-9		40	85
-8	1	60	145
-7		60	205
-6		80	285
-5		80	365
-4	2	100	465
-3		100	565
-2		100	665
-1		120	785
1	3	140	925
2		140	1,065
3		160	1,225
4		180	1,405
5	4	190	1,595
6		200	1,795
7		220	2,015
8		220	2,235
9	5	250	2,485
10		270	2,755
11		250	3,005
12		230	3,235
13	6	220	3,455
14		200	3,655
15		180	3,835
16		160	3,995
17	7	120	4,115
18		110	4,225
19		100	4,325
20		80	4,405
21	8	60	4,465
22		33	4,498.0
23		1.0	4,499.0
24		0.5	4,499.5
25	9	0.3	4,499.8
26		0.2	4,500.0
27			
28			
29			
30			

> * OVERNIGHT * COST LEVEL

MILESTONE SCHEDULE (MONTHS)

START R&D / TITLE I REACTOR DESIGN (-36)
& FUEL FAB CONCEPTUAL DESIGN

START TITLE II - Rx DETAIL DESIGN (-24)
& FUEL FAB DETAIL DESIGN

START SITE WORK / FUEL FAB PLANT (-10)

START CNTMNT / RB STRUCTURAL CONC.(+1)

FUEL FAB PLANT COMPLETE (+18)

Rx START PRE-OP TESTS & START-UP(+31)

UNIT 1 POWER PLANT COMPLETE -
START UNIT 1 FUEL LOAD (+48)

UNIT 1 COMMERCIAL OPERATION (+57)

UNIT 2 C/O (+63)

* CASE 5 INCLUDES :
EPC OVERNIGHT COSTS - \$4,500 M INCL
PRE - OP COSTS (AS A % OF OWNERS COSTS)

EXCLUDES : R&D AND O&M / D&D COSTS

11-May-93

TABLE 10.1 - 13

8100 MWe ABWR - Pu DISPOSITION STUDY :
CASE 4 Pu DESTRUCTION (18 YEARS)

SIX REACTORS
CAPITAL COSTS
CASH FLOW ESTIMATE

SIX REACTOR PLANT CASH FLOW (M\$)

QUARTERS FROM START OF Rx CONSTR	YR	CAPITAL COST CASH FLOW	CAPITAL COST CUMMU- LATIVE
(\$ MILLIONS)			
-12	0	10	10
-11		20	30
-10		40	70
-9		60	130
-8	1	80	210
-7		100	310
-6		100	410
-5		120	530
-4	2	150	680
-3		180	860
-2		190	1,050
-1		200	1,250
1	3	200	1,450
2		220	1,670
3		220	1,890
4		240	2,130
5	4	240	2,370
6		260	2,630
7		260	2,890
8		280	3,170
9	5	280	3,450
10		300	3,750
11		330	4,080
12		380	4,460
13	6	380	4,840
14		400	5,240
15		450	5,690
16		500	6,190
17	7	550	6,740
18		600	7,340
19		550	7,890
20		480	8,370
21	8	440	8,810
22		390	9,200
23		360	9,560
24		330	9,890
25	9	280	10,170
26		230	10,400
27		180	10,580
28		140	10,720
29	10	100	10,820
30		77	10,897
FINAL YEAR		3	10,900

> * OVERNIGHT * COST LEVEL

MILESTONE SCHEDULE (MONTHS)

START R&D / TITLE I REACTOR DESIGN (-36)
 & FUEL FAB CONCEPTUAL DESIGN

START TITLE II - Rx DETAIL DESIGN (-24)
 & FUEL FAB DETAIL DESIGN

START SITE WORK / FUEL FAB PLANT (-10)

START CNTMNT / RB STRUCTURAL CONC.(+1)

FUEL FAB PLANT COMPLETE (+18)

Rx START PRE-OP TESTS & START-UP(+31)

UNIT 1 POWER PLANT COMPLETE -
 START UNIT 1 FUEL LOAD (+48)

UNIT 1 COMMERCIAL OPERATION (+57)

UNIT 2 C/O (+63)

UNIT 3 C/O (+69)

UNIT 4 C/O (+75)

UNIT 5 C/O (+81)

UNIT 6 C/O (+87)

EPC OVERNIGHT COSTS - \$10,900 M INCL
 PRE - OP COSTS (% OF OWNERS COSTS)

EXCLUDES : R&D AND O&M / D&D COSTS

11-May-93

TABLE 10.1 - 14

**ONE REACTOR -
OVERALL PROJECT
CASH FLOW ESTIMATE**

1350 MWe ABWR - Pu DISPOSITION STUDY:

BASE CASE SPIKING (7.4 YEARS)

R&D / CAPITAL / O&M COSTS - TOTAL \$ *

CASH FLOW (M\$) *

YEARS FROM PROJECT START (Rx TITLE 1)	TOTAL COST CASH FLOW (\$ MILLIONS)	TOTAL COST CUMMU- LATIVE
1	75	75
2	212	287
3	300	587
4	460	1,047
5	730	1,777
6	842	2,619
7	463	3,082
8	180	3,262
9	150	3,412
10	148	3,560
11	148	3,708
12	148	3,856
13	148	4,004
14	136	4,140
15	56	4,196
16	115	4,311
17	100	4,411
18	70	4,481
19	28	4,509
20	0	4,509
21		
22		
23		
24		
25		
26		
27		
28		
29		
30		
31		
32		
33		

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> * OVERNIGHT * COST LEVEL
(EXCLUDES ESCALATION / AFUDC)

MILESTONE SCHEDULE

- START TITLE I - REACTOR & START CON-
CEPTUAL DESIGN FUEL FAB (PRE-TITLE I)

- FUEL FAB - START OPERATIONS :
FUEL FAB PLANT OPERATES FOR
9.4 YRS @ \$ 93 M/YR = ~ \$874 M
- REACTOR / ECA - COMM OPER :
REACTOR OPERATIONS 7.4 YRS
@ \$32 M / YR = ~ \$237 M
ECA (CONVERSION AREA) OPER
7.4 YRS @ \$24 M = ~ \$178 M

- REACTOR OPERATION COMPLETE (7.4 YRS)
- DECON / DECOMM OPERATIONS

DECON/DECOMM FUEL FAB
FACIL - \$50M - OVER 3 YRS
\$50 M

DECON/DECOMM REACTOR
FACIL - \$240M - OVER 3 YRS
\$240 M

TOTAL R&D / PRE-TITLE 1 DESIGN - \$12 M

TOTAL CAPITAL COST - \$2,918 M

(INCL PRE-OP)

TOTAL O&M / D & D - \$1,579 M

* OVERNIGHT * COST ESTIMATE

TOTAL - ABWR Pu DISP. : \$4,509 M

12-May-93

TABLE 10.1 - 15

SIX REACTORS - OVERALL PROJECT CASH FLOW ESTIMATE

6 x 1350 MWe ABWR - Pu DISPOSITION STUDY :

BASE CASE DESTRUCTION *

R&D / CAPITAL / O&M COSTS - TOTAL \$

CASH FLOW (M\$) *

YEARS	TOTAL	TOTAL
FROM	COST	COST
PROJECT	CASH	CUMMU-
START	FLOW	LATIVE
(Rx TITLE 1)	(\$ MILLIONS)	
1	140	140
2	400	540
3	720	1,260
4	880	2,140
5	1,090	3,230
6	1,350	4,580
7	1,830	6,410
8	2,360	8,770
9	1,720	10,490
10	1,140	11,630
11	580	12,210
12	390	12,600
13	390	12,990
14	390	13,380
15	390	13,770
16	390	14,160
17	390	14,550
18	390	14,940
19	310	15,250
20	310	15,560
21	300	15,860
22	290	16,150
23	290	16,440
24	290	16,730
25	210	16,940
26	510	17,450
27	300	17,750
28	191	17,941
29	0	17,941
30		
31		
32		
33		
34		
35		
36		
37		
38		

JANUARY 1992 -

* OVERNIGHT * COST LEVEL
(EXCLUDES ESCALATION / AFUDC)

MILESTONE SCHEDULE

- START TITLE 1 - REACTOR & START CON-
CEPTUAL DESIGN FUEL FAB (PRE-TITLE 1)

- FUEL FAB - START OPERATIONS :
FUEL FAB PLANT OPERATES FOR ~
13 YRS @ \$ 103 M/YR = ~ \$1339 M
- REACTOR / ECA - 1st @ COMM OPER :
REACTOR OPERATIONS AVG ~ 16 YRS
@ AVG \$180 M / YR = ~ \$2880 M
ADD UNITS 2 - 6 RX COMM OPER
@ 6 MONTH INTERVALS
ECA (CONVERSION AREA) OPER
16 YRS @ AVG \$110 M = ~ \$1760 M

- DECON/DECOMM FUEL FAB
FACIL - \$50 M - OVER 3 YRS
\$50 M
DECON/DECOMM REACTOR
FACIL - \$1000 M - OVER 3 YRS
\$1000 M

- REACTOR OPERATION COMPLETE
- STRT Rx DECON / DECOMM OPERATIONS

TOTAL R&D / PRE-TITLE 1 DESIGN - \$12 M
TOTAL CAPITAL COST - \$10,900 M
(INCL PRE-OP)

TOTAL O&M / D & D - \$7,029 M

* OVERNIGHT * COST ESTIMATE

TOTAL - ABWR Pu DISP. - \$17,941 M

The fuel fabrication facility schedules shown was developed from previous GE experience designing and building fuel fabrication plants with appropriate adjustments incorporated due to the introduction of plutonium into the process. European and Japanese experience in building mixed-oxide fuel fabrication plants have been incorporated in the schedules shown.

Figures 10.2-1 and 10.2-2 present the overall project schedules for a single reactor case and a two reactor case. The six reactor case schedule will be essentially the same as the two reactor case with six month intervals between startup and operation of successive units (Section 10.2-6).

Because of the advanced status of the ABWR (i.e. two plants currently under construction), Title I design for the plutonium disposition reactor can begin immediately following contract award. Further, because the SSAR for the ABWR is already available, it is assumed that the critical path to start of site preparation will be issue of the Record of Decision (ROD) for the complex in approximately two years. Reactor construction duration is approximately 48 months with initial fuel load at the end of the seventh year. Fuel fabrication facility startup will be approximately three years before reactor fuel load to allow for fabrication of the first reactor core.

The two reactor case shows essentially the same schedule for the second reactor with fuel loading and startup proceeding immediately upon completion of startup of unit 1.

The six reactor case shows the relationship of the construction / startup schedules for each unit and shows that startup of the sixth reactor is approximately 87 months after start of construction of the first unit.

10.2.2 Schedule Assumptions

The schedules were developed using the following assumptions:

- The ABWR plant deployed for the project will be essentially identical to the

Figure 10.2-1 ABWR Pu Disposition Overall Project Schedule- One Reactor Case

Figure 10.2-2 ABWR Pu Disposition Overall Project Schedule- Two Reactor Case

design and configuration of the commercial, uranium burning ABWR reactors

- **Engineering for the first-of-a-kind plant has been completed in Japan and design to U.S. standards will be started by the time the project starts reactor design.**
- **NRC approval of the ABWR design will be substantially complete at the time of start of safety / environmental document preparation**
- **The reactor will not be licensed by NRC but will be designed to meet or exceed NRC standards**
- **The reactor will be sited at an existing DOE site which has been previously extensively evaluated for environmental impact and nuclear safety**
- **There are no major research and development programs required to delay detailed design and construction**
- **Reactor component fabrication and construction will maximize the use of modular construction techniques**
- **Construction field labor will utilize rolling 4-10 schedules**
- **All critical resources are available for the project when needed including required funding**
- **The fuel fabrication facility design will utilize existing technology for mixed-oxide fuel fabrication and include remote plutonium processing operations within glove box enclosures with hands-on maintenance capability**
- **No unusual soil or subsurface conditions are present at the site**

10.2.3 Safety/Environmental/Regulatory

Safety, environmental, and regulatory durations were developed from historical data from similar projects. The effects of the advanced stage of ABWR licensing submittals to the NRC were incorporated in estimated times for SAR approvals. Estimated durations of environmental compliance activities were based on detailed milestone schedules previously developed for the New Production Reactor project.

10.2.4 Research and Development

Durations shown for research and development are based on the scope of development activities described in Section 9.0 and shown on Table 10.1-1.

10.2.5 Design

As noted above, it is assumed that U.S. ABWR Title I design will be started when the plutonium disposition project begins and the project detailed design will utilize the design work done for the Japanese ABWR reactors. With this extensive experience base, Title I design can start immediately and is expected to be complete within the first year of the project. Title II design will follow and is assumed to be approximately 80 percent complete at start of construction.

For the two reactor case, the design duration will be essentially the same as that shown for the single reactor case since the two reactors will be identical except for orientation on the site. Detailed design drawings, specifications, etc. will be the same for both reactors.

It is also assumed that the design duration for the six reactor case will be the same as that for the one reactor case.

After EIS approval, site preparation will begin. Construction of the fuel fabrication facility will follow in about four months with reactor construction beginning at the end of the third year after start of Title I. It is assumed that long lead procurement items such as the reactor vessel will be ordered early in the project. It is also assumed that modular construction techniques employing factory assembled components is maximized to achieve early installation in the field. Also, as noted above, a rolling 4-10 construction schedule has been assumed to achieve the 48 month construction duration.

The EPRI ABWR Requirements Document, Volume II, Chapter 6, dated June, 1990 prescribes a reference construction schedule for a first-of-a-kind 1350 MWe ABWR of 48 months. However, if extensive modularization similar to plans for the SBWR were applied, the schedule could reasonably and significantly be shortened. It is probable that lessons learned from the Japanese construction experience and a more detailed ABWR modular construction scheme will be developed in the near future and that a corresponding construction schedule of about 39 months will evolve. Despite this potentially shorter schedule, the first unit built in the U.S. will require schedule contingency to cover resolution of module interfaces and constructibility problems as well as revitalization of the nuclear industry. A contingency of about nine months has been considered appropriate resulting in the proposed construction schedule of 48 months. A more detailed construction schedule showing major construction activities was developed to verify the construction schedule proposed and is shown in Figure 10.2-3.

Start of construction of the second reactor is shown approximately seven months after start of construction of the first unit. This allows time to gain "learning curve" experience from construction of the first reactor and also allows the main excavation work and basemat construction to be completed on the first unit before starting this work for the second unit. Common useage of field equipment and materials such as field erection cranes and concrete forms can be employed. This logic results in a construction schedule of approximately 49 months for reactor number two allowing startup of this reactor about six months after

- BASIC CRITERIA RELATING TO CONSTRUCTION SCHEDULE**
- ASSUMES EXISTING DESIGN
 - ENGINEERING TO ADVANCED STAGE (NOT AT START OF STRUCT. CONCRETE)
 - NO REGULATORY/PERMITTING RESTRAINTS
 - EXTENSIVE MODULAR CONSTRUCTION
 - MECA SPLICES INSTEAD OF CADWELDS
 - REMOVAL, VI SITE LOCATION
 - ROLLING 4-HOUR 170 HR. WEEK
 - NO LABOR AVAILABILITY OR CASH FLOW RESTRAINTS

Figure 10.2-3. ABWR Pu Disposition Study
Detailed Construction Schedule for
Single & 2 Unit Plants

startup of reactor number one.

However, the six unit construction schedule requires allowances for construction and startup sequencing in addition to the effects of learning from previous unit construction and startup. Figure 10.2-4 shows the schedule with startup of the sixth unit approximately 87 months after start of construction of the first unit. Seven months have again been allowed between start of construction to allow for excavation/basemat construction and six months between successive reactor startups. This schedule requires extensive manpower and material resources to attain the milestones shown. Peak site manpower would exceed 7000 manual workers per shift for the nuclear island alone.

10.2.7 Preoperational Checkout and Startup Testing

Preoperational testing and startup durations are based on previous schedules developed in detail for the ABWR. Startup of a conventional BWR is normally six months from fuel load to commercial operation. Allowing for new reactor contingencies, eight months has been allocated for this startup period for the first ABWR unit.

10.2.8 Reactor Complex Operations

Startup of the mixed-oxide fuel fabrication facility is shown beginning after four years from start of facility design. Fuel fabrication operations at full capacity will begin approximately nine months later. This allows about 2 1/2 years of production to fabricate the first reactor core loading.

Full-power reactor operations of a single reactor is shown to begin approximately 57 months after start of construction. This allows a period of about 17.3 years of reactor operation to disposition plutonium within the allowable 25 years prescribed in the DOE requirements document.

As shown on Figures 10.2-2 and 10.2-4, startup sequencing of multiple reactors results in

SHORTEST 6-UNIT CONSTR. SCHEDULE IS BASED ON:

- 7 MONTHS BETWEEN START OF UNITS EXCAVATION LIMITATION
- 8 MONTHS BETWEEN COMPLETION OF UNITS (STARTUP LIMITATION)

NOTE - RESERVATIONS REGARDING PRACTICAL/ECONOMIC CONSIDERATIONS!

Figure 10.2-4. ABWR Pu Disposition Study
Construction Schedule for 6 Unit Plant

operations beginning in six month sequences. This yields operating periods of 16.8 years for the second reactor, 16.3 for a third reactor, etc. resulting in an operating period of a sixth reactor of about 14.8 years.

10.2.9 Tritium Conversion and Tritium Production

Schedules for conversion of the reactor complex to produce tritium are covered in Section 6.7.

10.2.10 Facilities Outside the Reactor Complex

There are no major facility construction or modification projects required except for the tritium conversion case, covered in Section 6.7.

10.0 REFERENCES

- 10-1 Delene, J. G. and C. R. Hudson II, "Cost Estimate Guidelines for Advanced Nuclear Power Technologies," ORNL/TM-10071/R3, March 1993**
- 10-2 Hudson, C. R., "Calculation of Steam revenues for Plutonium Disposition Reactors (PDR)," Oak Ridge National Laboratory Memo 5 April 1993**

11.0

IMPLEMENTATION AND DEPLOYMENT

11.1 INTRODUCTION

The purpose of this section is to:

- Summarize the considerations and status of the infrastructure required for a plutonium disposition program, (many of which have been separately discussed in other parts of the report), and
- Review the potential deployment options for the program, related to the extent that the infrastructure exists for each part of the program.

The infrastructure considerations which were evaluated for a Plutonium Disposition Program are:

- A. Organizational, and related to this were the agreements, authorizations, procedures, laws, regulations, and standards under which the organizational institutions would operate.
- B. Available technology or required development
- C. Physical resources including labor, materials, and existing facilities.

The aspects of the Plutonium Disposition Program for which the infrastructure considerations were evaluated are (Figure 11-1):

- 1) Transportation
- 2) Plutonium conversion and fuel fabrication
- 3) Plutonium consumption and Reactor Operations
- 4) Spent Fuel, Waste disposal, and Program closeout

Figure 11-1. Process Flow For Destruction of Weapons Plutonium

The organizational and technology aspects of the program have been extensively discussed in other sections of the report.

The organization infrastructure is extensive and exists. As an overview, in the U.S., in addition to the DOE, the organizational complex includes other U.S. Government administrative and technical agencies such as DOE, DOT, State, Commerce, Treasury, The Administration, and congress. Other involved U.S. based organizations or interests include U.S. corporations, technical and public interest groups including the Press, and the public at large. In addition to U.S. and corresponding CIS Government Agencies, Bureaus, technical organizations and emerging commercial interests, a bilateral U.S. and CIS Plutonium Disposition Program would and could include the IAEA, International Arms Control Commissions, and to the extent that it may be advantageous to implementation of the program, third nation or party government interests and corporations.

Selection of an approach for implementation and deployment of any system for destruction of weapons plutonium will be strongly influenced by international government decisions, some made at the highest levels of government. In addition, the methods for financing or sharing costs of the projects, in the CIS and the U.S. and other participant countries, could impact how implementation and deployment can best be accomplished. Because these issues are yet to be identified and resolved, the discussion that follows, which is based on past experience, is subject to modification. The objective is to identify and discuss some of the known factors influencing implementation and make recommendations while fully recognizing that larger political and financial factors could overshadow some of the issues discussed here.

The study concludes there is only one practical, low risk, near-term alternative for plutonium conversion that provides the safest, most economical, and truly safeguarded mutually and concurrent disposition of weapons plutonium in the CIS and U.S. That alternative is the use of the Advanced BWR design comparable to that already under construction in Japan.

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Alternates to this approach exist, but they appear less attractive. In many instances, significant technical, political, and economic risks and consequences, or entirely unacceptable conditions result when an ABWR is not used.

11.2 DEPLOYMENT

The deployment evaluation which follows used the following criteria in order of importance to evaluate alternatives:

- Ability of the facility to satisfy mission requirements of capacity and functionality
- Minimized risks of political or administrative bottlenecks or delays
- Minimized distribution and transportation of Special Nuclear Material
- Technical maturity of the alternative

The requirements for the commercial design certification, and for the construction and issuance of an operating permit for the ABWR are defined. Authorized and certified in Japan, the ABWR certification process documentation has been submitted and final certification for U.S. commercial operation is scheduled to be completed in the next two years. This will assist with the plutonium disposition program which would be implemented as a non-commercial, DOE defense project so that the organizational delays associated with commercial project could be avoided. The DOE Major Systems Acquisition (MSA) processes would be utilized. The environmental section of this report summarizes those requirements and that process.

Commercial resources for the detailed design, supply of material and equipment, and construction all exist. The commodities in the U.S. exist reliably. The commodities for a CIS project might be less reliable but could be managed and/or made available by an international team applied to a CIS project.

The infrastructure issues, therefore, are limited to site selection and operation.

Consideration was given in this evaluation to the use of an existing light water reactor or completion and upgrade of an unfinished light water reactor site and facility. Positive aspects identified were that the facilities have the potential to be made available quickly, the plant site already exists, capital investment requirements might be reduced, and government ownership could be limited.

While use of an existing light water reactor or the completion and upgrade of incomplete LWR facilities might appear as an attractive potential for lowering the PDS reactor capital cost part of a plutonium disposition system, significant technical and non-technical factors impede this approach. Potential negative factors relating to the use of existing light water reactors are detailed below.

1. The existing light water reactor design may not technically support fully incorporating all the available safety features which are now available in the U.S. nuclear industry.
2. The extensive design and safety reviews and DOE MSA process for a retrofit design and for an operating licensing for the retrofit and completion of an existing light water reactor to the equivalent or same safety standards as are available in an ABWR could incur cost escalations and/or schedule delays exceeding any potential cost advantage.
3. The nature of attempting to retrofit an older design to achieve capabilities not originally "designed-in" usually requires design compromises, and/or incurs deficiencies, and design and construction cost escalations which also could easily exceed the potential expected savings. This is therefore a higher technical, schedule and cost risk approach than proceeding with design and construction of a known and dedicated plant specifically designed and proven for the safety features and goals desired.
4. The locations of existing light water reactors are inconsistent with an infrastructure and political acceptance deployment strategy which would strongly suggest a PDS

reactor site that already and currently has plutonium operating and management experience and future acceptance. Existing light water reactor sites would also probably incur a distributed deployment scheme with the plutonium conversion and or fuel fabrication occurring at some other location, thus incurring additional transportation risks to the deployment scheme. Alternately, same-site operation of the fuel plant would require plutonium conversion or plutonium fuel fabrication operations at a site not currently compatible, acceptable, or familiar with those operations.

5. The conversion of an existing utility LWR facility would require significant commercial motivation and potentially lengthy commercial negotiations for utility acceptance, and may also be incompatible with the ongoing commercial plans of the utility or owner for that site.
6. The existing or converted LWR site lacks the ABWR's designed-in ability to operate with mixed-oxide fuel cores due to its evolution from the BWR 6 design.

The existing site of the current or unfinished light water reactor would also involve compromises which may add to the risk. Any delta in costs in a LWR versus ALWR comparison should be evaluated in the context of overall project costs along with other elements, i.e., fuel plant, Operation and Maintenance, storage and transportation of the fuel, and a dedicated site selection. Existing facilities requiring distributed locations also run counter to the primary safeguards objective of the plutonium disposition program. Therefore, although the ALWR may at first appear more costly, the uncertainty in a compromise "retrofit" could involve more risk and uncertainty than proceeding with a known ALWR. The ABWR also has built-in advantage for operating with mixed-oxide fuel. Large size capacity for plutonium disposition allow the mixed-oxide fuel disposition goal to be met in the shortest time.

Review of the ALWR against the preceding criteria indicate a number of significant advantages to its deployment as the preferred choice. Many of these advantages are listed

below.

1. While existing LWR facilities are safe designs that have been updated and retrofitted to meet changing NRC requirements, the ABWR design under construction and under final design review and certification for the US, has the most advanced safety capabilities available, especially by virtue of its having been designed to meet requirements of the Severe Accident rule making.
2. Although the ABWR was not specifically designed for mixed-oxide fuel, this study showed that some ABWR core design features provided additional capacity and safety margins for operating with mixed-oxide fuel cores. The ABWR design evolved from its BWR 6 predecessor which was designed for such operation. Thus, full mixed-oxide fuel core operation is easily supported within the even more restrictive ABWR safety criteria.
3. The ABWR offers the nearest term, safest, highest capacity PDS reactor project, with the least technical, cost, and schedule risk of any potential alternate.
4. The ABWR study showed that significant flexibility exists in the design of mixed-oxide core configurations to accommodate different electric production goals. Therefore, the ABWR would easily accommodate mission changes dictated by subsequent changes that might occur in the political, technical, or budgetary criteria for the program and for the mission of the facility.
5. A new plant could be located at a Government site. This would meet the previously identified study criterion of minimizing risks of political or administrative bottlenecks or delays. This approach also meets the criterion of minimizing distribution and transportation of Special Nuclear Material.
6. The ABWR provides a high level of confidence in keeping the project on schedule. The front-end time, prior to the start of construction should be lower for this method

compared to any other as the ABWR has already undergone considerable licensing review by NRC. The extensive use of modular factory fabricated components in building the ABWR allows for better scheduling and control of the construction time. If located at a Government site where some of the site parameters are available ready, the environmental impact studies could be concluded for the entire complex in 18 months.

The ALWR-based approach is, therefore, the most viable of the alternatives, then, if one wants to implement the program with a fast start and assured schedule. The ABWR of the ALWRs is clearly the lead plant due to its commercial status in Japan and its licensing review status in the U.S.

To implement plutonium destruction, there are four essential activities; conversion, utilization, transportation and disposal. These are shown diagrammatically in Figure 11-1. Conversion and utilization can be effectively implemented in several ways, but transportation and disposal must use existing or planned international government facilities and organizations, and there is little opportunity or need for innovation. Alternatives for these four key areas are discussed in the following sections.

11.3 CONVERSION

The fuel cycle (conversion process) was described in Section 3. It is fully supported by available, known and proven technology, commodities, constructors, equipment vendors, and operators all exist. Safeguards criteria, safety and licensing, environmental, and design criteria exist and can be met. Facilities for limited confirmatory testing described in Section 3.10 are available within the GE Nuclear Energy commercial complex. Process selection and optimization, facility site selection, and cost tradeoffs become the key consideration for deployment.

One of the key factors in selecting sites or facilities for the conversion process is the need to preclude theft of the plutonium or diversion of the conversion facilities. In this context

theft has been defined (Ref.1-2) as an attempt by a sub-national group to acquire special nuclear material and diversion as the activities needed to implement a decision by a government or sub-national group to use fuel cycle facilities or special nuclear material to make a weapon.

The barriers to theft and diversion generally fall into the following categories:

- Physical security,
- Special nuclear material accountability,
- Technical measures (e.g. process design features, radioactivity level, dilution, material form, etc.) and
- External (international) oversight.

The effectiveness of the first three barriers is more related to the design of the conversion process and facilities and the performance of the operating organization than the site selected. The effectiveness of external oversight is related to the special nuclear material accountability system, the technical measures employed and the cooperation of the host country which could well be site dependent.

Other important goals which also influence the selection of sites or facilities for conversion include:

- The entire inventory of excess weapons plutonium should be converted to the most theft or diversion resistant state as soon as possible (Timeliness),
- The entire conversion process should be easily monitored by other interested parties (Transparency),
- The facilities and sites selected for conversion should support the desired reciprocal response by the other party (Reciprocity) and,

- The facilities and sites selected for conversion should be resistant to changes in national policy (Recidivism).

Co-location of all the steps in the conversion process is generally considered an advantage with respect to theft since the end product (mixed-oxide fuel subassemblies) is a less attractive target and is more easily protected and accounted for than the intermediate process materials. In this regard the ultimate co-location strategy would be to integrate conversion with weapons disassembly. However, this may not be an optimum strategy considering the other goals listed above. For example:

- Co-location of conversion and weapons disassembly may not support the reciprocity or recidivism goals.
- Since new facilities would be required in the U.S., co-location may not support the timeliness goal if acceptable alternatives are available.
- The most attractive targets are the complete weapons which must be shipped in any scenario. The plutonium metal from weapons disassembly can be shipped in the same safe, secure transports used to ship the weapons. Thus shipping plutonium metal would probably not significantly increase the overall risk of theft since the number of shipments would less than double and these shipments are only one step in the overall weapons disposal process.

There is no significant mixed-oxide fuel fabrication capability in the U.S. today although there are two modern Safety Class I plutonium handling facilities (FMEF at Hanford and TA-55 at Los Alamos) that have the potential for conversion to mixed-oxide fuel fabrication. Combined these two facilities have approximately 16,000 square meters of Safety Class 1 processing area. Since approximately 14,000 square meters of Safety Class 1 space is all that is needed, the other 2,000 square meters could be used for the conversion process. Depending on the scenario assumed conversion of one or both of these facilities may represent an attractive option that should be pursued during the next phase of this

program.

The Japanese, Germans, British, Belgians, French and Russians all have planned or operational facilities capable of producing mixed-oxide fuel for an ABWR. Table 11-1 lists the characteristics of some of these facilities (Ref. 11-3). Of the existing or near term facilities only the new German plant, which is complete, and the Russian plant, which is under construction, have adequate throughput by themselves to meet the needs for all the scenarios examined. The planned British and French facilities would also have adequate throughput when available.

Table 11-1. General Plant Characteristics

	PPFF	PPFF	Belgonuclaire	Siemens MOX II	Siemens MOX II
Start of Operation	1972	1987	1973	1980	(1993)
Near Real Time Accountancy	No	Yes	No	No	Yes
Present Capacity (tMOX/y)	10	5	40	35	(110)
Main Fuel Product	ATR	FBR	LWR	LWR	LWR
Routine Frequency of Inspection	Once Per Month	Once Per Month	Continuous	Continuous	Continuous

The other issue that must be examined with respect to any of the existing facilities is the current plans for their use. The German plant, which was constructed to modern safety, safeguards and security requirements was intended for recycle of their existing light water reactor spent fuel. However, the current administration of the state in which the plant is located (Hesse) is opposed from a policy standpoint to recycle of light water reactor spent fuel and has blocked startup of the plant. Since this is a political policy issue rather than a safety or licensing concern, the state administration's view might be different if the mission of the plant was supporting nuclear disarmament rather than light water reactor fuel recycle and there was strong multinational support for use of the plant in this mode.

The potential for use of the other existing or planned overseas mixed-oxide fuel fabrication capability has not been examined in any detail due to the limited time of this study. However, a number of the companies involved have and continue to work together in joint ventures, etc. In addition, there have been discussions between the German fuel manufacturer and the Russians relative to making available the German mixed-oxide fuel fabrication experience. Thus, it is quite conceivable that given the necessary support of their respective governments the existing and planned mixed-oxide fuel fabrication capability needed to convert the excess weapons plutonium to mixed-oxide fuel could be made available in a timely and economic manner.

While new conversion facilities offer the potential to optimize the process with respect to the various goals (Safeguards & Security, Timeliness, Transparency, Reciprocity and Recidivism) the time and cost involved makes it imperative to more thoroughly evaluate utilization of existing and planned facilities. Therefore it is recommended that the next phase aggressively pursue discussions with the owners of these facilities and the appropriate government agencies relative to how these facilities might be used in support of nuclear disarmament.

11.4 UTILIZATION

The infrastructure for the ABWR is fully in place. The licensing and environmental qualification requirement are defined. The DOE MSA process would be used and could parallel the equivalent commercial process which is on-going. No development testing specific to the reactor has been identified. Most all the elements that are required for the implementation of this concept has previously been developed and tested in a large scale/industrial setting and few additional areas have been identified that require research and development in the conventional sense of this term. The specific areas listed in Section 9, identified as requiring development, are primarily related to confirmatory testing and analyses rather than conventional research and development. There may be some need to perform testing of features related to designing and handling the targets for tritium production. Further work is needed to define this area of design. Thus, site selection

becomes the key deployment issue.

As mentioned previously, the most direct option for utilizing the mixed-oxide fuel is to insert it in existing commercial LWRs. The political and procedural problems with this approach are large. The conversion or diversion of an operating light water reactor for plutonium conversion would probably require significant initiative and negotiation between DOE and the utility. Conversion of a completed, operating light water reactor is a remote option and not considered. Light water reactor owners have been identified that are willing to accept the inconvenience and risks that are associated with upgrading and expanding operational and security procedures and training staff to support their normal commercial missions with large scale mixed-oxide fuel operations. Outside the U.S., there are LWRs that will be operating with recycled mixed-oxide fuel. Although these facilities might technically be better prepared to accept the mixed-oxide fuel from weapons material, there are other institutional issues related to security and accountability. Two nations actively pursuing use of mixed-oxide fuel are France and Japan. Use of such third nation for plutonium disposition might be attractive but existing facilities may also preclude operating in a tritium production. These conclusions are based on our general understanding of the LWR owner community and their reluctance to complicate their operations for any reason other than safety. A specific survey of owners has not been conducted.

It is conceivable that with appropriate government incentives, a BWR owner might find reason to accept the complications. To realistically investigate this option, an outline of a government program and possible incentives for the owner would have to be developed. This was not possible to do within the time of the current study but is proposed for the next phase.

The primary alternative to this direct approach is to build new ABWRs specifically for destruction of the plutonium. Evaluations have shown that as few as one ABWR could implement an expedient contamination of the mixed-oxide fuel while also delivering electric power with a capacity factor acceptable to a utility grid. In this approach, the Hanford N-Reactor and the operational approach for the planned New Production Reactor, provide

appropriate experience. In these examples, the government builds and owns the nuclear island (NI).

A GOCO agreement with a utility or IPP would allow at least a partial if not total commercial construction financing and partial or total ownership for the balance of plant ECA which would then sell the electric power. There are obviously many sharing arrangements that the Government and GOCO/IPP could enter into to share financial and construction responsibility for the ECA. There would have to be financial incentives to compensate the utility for the complications, uncertainty, and inefficiencies that are likely to be associated with such a program. Using this alternative for a facility sited in the U.S. leaves only two questions; where to site the facility so that GOCO operation is consistent with the electric production cycle for the plutonium disposition mission, especially in the early years when a short-term spiking cycle might be desirable; and is it suitable for operation with mixed-oxide fuel rods or fuel bundles as input ?

This model is probably not appropriate for the Russian facilities and may not be suitable if a third party foreign site were considered. Those two cases will be discussed following discussion of U.S. siting and co-located fuel assembly.

The technical issue of tritium production is discussed in Section 6.0 but the implication to implementation has not been completed and could have an impact on some options.

The siting decision is theoretically dominated by the regional need for electric power and the availability of suitable sites for the government-owned reactor. Since these issues have been recently considered within the NPR program and Savannah River was selected, this site should have strong preference. However, there have been significant changes within the administration and Congress since the NPR siting decision and therefore, realistic alternatives must be re-evaluated. These include Idaho National Engineering, Hanford, NTS, Pantex and Oak Ridge National Laboratory. Pantex has the technical advantage of being the only site at which all operations could be co-located. The overall safeguarding and accountability activities would be significantly reduced by selecting this site since

transportation of clean plutonium would not be necessary. The projected need for electric power in the south and southeast is also compatible with Pantex or Savannah River as a nuclear power site. Cooling water availability at Pantex could be an issue because of competition with the farming community for a limited amount of water. The NTS site would also require unique engineering solutions for cooling. Lake Mead is a potential source of cooling water, and air cooling could be considered, but would introduce development issues for the turbines and additional cost issues.

Full evaluation of the government sites could include consideration of separate siting of fuel rod production from the final fuel assembly facility with final assembly occurring at the reactor site. This offers options on space requirement and existing space in available facilities. This is different than current practice but there is no technical reason for not considering this option. Locating the fuel assembly facilities with the reactor, apart from a remote conversion facility responsible for output of fuel rods (rather than full assemblies) could be more acceptable to a site such as Pantex. Facilities for fuel rod assembly into fuel bundles will require safeguards security for the plutonium, for control of criticality, and may require special attention to ventilation and fire protection but otherwise are not particularly special.

An alternative to the government site is to identify a utility with a suitable site and interest in participating in a government sponsored nuclear program to build an ALWR reactor on their site. Conceptually, the relationship between the government and utility may not be much different than that required for dispositioning the plutonium in existing reactors. This approach could be extended to foreign utilities' sites, such as Mexico, that are clearly in need of electric power and financial support. Extension of the approach to Russia, the Ukraine, or Kazakhstan is also possible. In these foreign countries, the utilities are likely to be government owned and operated and therefore, the significance of utility participation is not as distinctive as in the case of the U.S. Because of IAEA and non-proliferation treaty commitments, the utilities might have to lease or cede the site to the US in exchange for "cheap" power and economic participation (99 year lease). Within the U.S. it is not obvious that there are any utility sites that are more attractive than the government sites already

identified. Thus, the challenge of identifying a suitable new utility-owned site is significant and thus, the direct approach of confining the U.S. project to Government-owned sites is considered preferable. The probability of a suitable PDS/ECA GOCO split is highly likely, and GE expects to conduct discussions with U.S. utility IPP/Operating subsidiaries that have expressed strong interest as potential partners in such an arrangement.

In the case of the foreign countries, this appears to be a very attractive approach. This could include government-to-government agreements that would result in a power generating facility that uses mixed-oxide fuel, with the facility and fuel largely underwritten by the U.S. or G-7 governments, and with the overall cost reduced by sale of electricity within the country. In the case of Mexico, most of the mixed-oxide fuel would come from the U.S., while in the CIS states, the mixed-oxide fuel would most likely come from Russia since they have a mixed-oxide fuel facility well along in construction. The CIS states would benefit in many ways from such a program. The most advanced and safest reactor technology would be introduced to these countries. Conversion of their industries so that they could supply additional plant construction to replace aging and less safe nuclear facilities could be implemented through this initial weapons destruction program. Again, G-7 financial support could be solicited to implement this type of program.

There are financing concepts that use the plutonium inventory as collateral and would potentially provide another basis for the CIS to obtain hard currency. Upgrading of not only utility operations, but the equipment supply industries to U.S. standards would benefit all eastern Europe and the Baltic states from the standpoint of risk reduction. The status of ABWR design, construction and NRC certification is sufficiently mature that transfer of the technology to CIS state industries can be accomplished more rapidly than any other alternative.

Similar technology transfer for retrofit/upgrade of the unfinished CIS mixed-oxide fuel facility would be similarly accomplished. An international participation in the design and construction and acceptance testing of CIS facilities would promote international cooperation and ensure high technical standards are maintained. International participation

also would ensure sufficient infrastructure resources were available for the project where they might not be reliably available from within the CIS. Significant cadres of permanently assigned U.S. or international cadres of technical persons could be assigned to engineering, operation and maintenance, and training at CIS facilities, supported by US funding.

An attractive hybrid of this concept of international participation could involve the parallel construction of ABWRs in Japan and the CIS. Since the ABWR can handle a full core of mixed-oxide fuel and K-6,7, the latest GE furnished Japanese ABWRs, are scheduled to enter service in late 1996, mid-1997, this scenario could provide the quickest start date for the utilization process. International agreements could also be established for the use of GE designed MOX fuel, fabricated initially in a completed CIS MOX plant, which might also be the quickest path for the conversion process, and later the fuel for either ABWR could be furnished from either the CIS plant or a Japanese mixed-oxide fuel fabrication facility which would come on line later. Implementation of this technology within the CIS provides a vehicle for investment in CIS infrastructure and its economy. Implementation of this scenario also puts the latest reactor design into use in the CIS, takes CIS plutonium out of circulation, utilizes and employs the talents of CIS nuclear scientists, and promotes international cooperation among the U.S., Japan, and the CIS.

11.5 TRANSPORTATION

Off-site transportation of nuclear materials required for the ABWR plutonium dispositioning concept includes weapons grade plutonium pits to a plutonium metal to plutonium oxide conversion facility, transportation of plutonium oxide to the MOX fuel fabrication facility, transportation of fuel assemblies to the reactor site, transportation of irradiated fuel assemblies and low level nuclear waste to off-site disposal facilities, and transportation of tritium targets or tritium canisters to off-site storage during the tritium production mode of operation. These types of shipments introduce no unique issues and the necessary equipment and practices are readily available.

Transportation of plutonium pits and plutonium oxide canisters will be accomplished with

the currently existing fleet of approximately 70 special armored trucks, known as Safe Secure Tractor-trailers, or SSTs. These trucks are equipped with radio and satellite equipment to enable government officials to track the truck's location at all times. The trucks are escorted by heavily armed guards during nuclear materials shipments.

Transportation of irradiated fuel assemblies and low level nuclear waste to off-site disposal facilities will be accomplished using the normal truck or rail cars designed for commercial nuclear fuel casks and containers.

Tritium target or canister transport will utilize specially designed trucks similar to those currently used at the Savannah River Plant tritium production facilities.

A similar secure shipping system should be introduced into the CIS or between US, CIS, and an international third site fuel fabrication facility. If an option is selected that requires international shipment of plutonium, it would most likely be in the form of fuel rods or assemblies. There are no physical problems with shipping these articles and appropriate safety, safeguards and accountability practices can be established. These should have the objective of achieving the same level of protection that is achieved for SST shipments within the U.S.

11.6 DISPOSAL AND DECOMMISSION

The irradiated fuel assemblies are essentially the same as standard light water reactor fuel assemblies and would be stored and disposed of the same as currently planned for the standard LWR. Although the plutonium burnup may be less than from the standard light water reactor fuel cycle, decay heat will be somewhat reduced but the radiation hazard will be comparable and will continue to require heavy shielding during storage and shipping ensuring diversion control. The fuel in both the U.S. and CIS could be stored indefinitely at the reactor site and could be placed ultimately in permanent fuel disposal sites. Fuel irradiated in the CIS facilities would follow the same path except that plutonium recycling into LWR or LMR fuel may be used to further reduce the plutonium inventory. There is

significant merit in proceeding on this path in the U.S. also, but only for the long term.

Spent fuel and other waste disposal requirements are clearly defined within existing NRC and DOE regulations. Ultimate disposal of the spent fuel and other wastes from the complex, comparable in kind to the larger nuclear spent fuel and other related high level wastes from commercial and government sources also awaiting ultimate disposal, will remain a broader issue of national policy and political agreements related to the Yucca Mountain operation, or alternate disposition such as reprocessing. As such, the methods, technology and facility for disposal do exist, subject possibly to economic trade-off studies and the political decision process. The small increase from the plutonium disposition complex as compared with existing quantities can easily be justified considering the strategic safeguards and security objective of this program.

The requirements and commercial capability for ultimate decommission of the complex are known. Comparable decommissioning projects have been done, i.e., Shippingport, to name one, and would benefit from the experience of many such projects which will have previously been done by the time the complex needs to be decommissioned.

11.7 RECOMMENDATION

After extensive study and weighing all the alternatives against the criteria, GE recommends deployment of the Advanced BWR design comparable to that already under construction in Japan.

Because the ABWR is well advanced in its design and construction experience and because it also includes enhanced safety characteristics, it can be implemented expeditiously in any of the possible options. Its large size and design compatibility with full mixed-oxide fuel cores reduces the number of facilities needed to rapidly dispose of the plutonium. It can be equally well implemented in the CIS, the U.S. and other foreign countries if desired.

Fuel production facilities do not appear to be as readily available for producing the fuel at a rate needed to rapidly convert plutonium metal into fuel assemblies. Final decisions related to fuel conversion might be necessary to keep it off the critical path. Implementing a solution which constructs a new ABWR can be completed on a schedule compatible with building a new mixed-oxide fuel facility and producing mixed-oxide fuel for a spiking cycle. The high cost of new facilities also makes it important to consider in more detail the possibility of using the German and Russian facilities or cooperative agreements with planned British or Japanese facilities. So little is really known about the Russian facility that it is premature to make any firm recommendations in this area. The mixed-oxide fuel plant and ABWR efforts need to be started about the same time and therefore, need to be given similar levels of attention in continuing studies.

Converting FMEF at Hanford and/or adapting TA-55 at Los Alamos to provide plutonium conversion and mixed-oxide fuel rod or bundle fabrication capability is an attractive option that should be investigated further. These are both modern Safety Class I plutonium handling facilities.

Because of the effort already invested in establishing Savannah River as a suitable site for NPR, it appears to be the site that would be preferred for an ABWR size plant although most of the other government sites could also accommodate the ABWR if there were reasons to site it elsewhere such as the more favorable market for electric production market for western site as compared to that "currently" existing in the Savannah River region. There also is a need to proceed with the building of a new plutonium conversion facility in the U.S. and CIS if the most rapid approach to destruction of the plutonium is to be implemented. Negotiations and modification of "potential" alternates in other countries, would be expected to almost certainly slow the process.

It appears most desirable to place the conversion adjacent to the Pantex site to eliminate any additional shipping of plutonium metal over public roads and take advantage of the experienced work force and community that are familiar with this material. The facility should produce fuel assemblies but could stop production at fuel rods if there were

sensitivity within the community to the production of fuel assemblies. Alternately, if obstacles to a Pantex conversion plant were to arise, the alternate SRS site would be a very acceptable alternative.

The institutional model for the utilization phase, using Independent Power Producers exists. In the proposed dispositioning approach, the turbine island complex would be owned by an Independent Power Producer with cost and revenue sharing arrangements. Institutional models for such an arrangement with the reactor and mixed oxide fuel plants to be operated as a Government-Owned, Contractor Operated Facility exist. A preliminary survey indicates that there is an interest for such an arrangement among the Independent Power Producer Community. The use of a low risk approach such as suggested here is likely to increase the possibility of implementing such an arrangement.

Consideration should be given to an independent Program Management and Integration contract to implement the MSA and project qualification, and approval phase of the contract so as to streamline the governmental process and reduce potential bottlenecks.

Since destruction of weapons plutonium must proceed in the CIS as rapidly as in the U.S., there is a need to move forward there on the same schedule. The availability of the partially complete high capacity mixed-oxide fuel plant in Russia dictates that these facilities be investigated further to establish their acceptability for use in the joint program. Since U.S. or G-7 money most probably will be used to modify and complete these facilities, it will be necessary to determine if they can meet international safety, safeguards and environmental standards. Efforts to complete these facilities and to proceed to converting CIS weapons plutonium to mixed-oxide fuel should be given highest priority. Solutions to this problem area could involve European experience and even existing equipment or facilities.

Establishing the route to utilizing the mixed-oxide fuel product in the CIS is somewhat more complicated than in the U.S. because of the potential for use in a variety of reactors. Introduction of the ABWR into the CIS, with its high level of approved safety, has the most

overall benefit to the world and should continue to be investigated. This approach might have a higher cost than using existing CIS PWRs or RBMK-1000s, but there are safety and environmental benefits in addition to the primary objective of plutonium destruction from proceeding with this approach. International agreements for construction of the GE ABWR in the CIS, and the use GE ABWR fuel from U.S., German, British, or Japanese MOX facilities should be evaluated. This approach deserves further review in the next phase of this study.

11.0 REFERENCE

- 11-1 DOE/NE-001 "Nuclear Proliferation and Civilian Nuclear Power-Report for the Nonproliferation Alternative Systems Assessment Program, Executive Summary" June 1980**
- 11-2 INFCE/PC/2/H "International Nuclear Fuel Cycle Evaluation - Reprocessing, Plutonium Handling, Recycle, Report of Working Group 4," IAEA, January 1980**
- 11-3 JNMM "Experiences and Trends in Safeguarding Plutonium Mixed Oxide (MOX) Fuel Fabrication Plants", M. Zendel, IAEA, February, 1993**

APPENDIX A: KEY BWR OPERATIONAL LIMITS DEFINITIONS

A.1 LINEAR HEAT GENERATION RATE

The **LINEAR HEAT GENERATION RATE (LHGR)** is a measure of the heat generation rate of a fuel rod in a fuel assembly at an axial location. Limits on LHGR are specified to assure that fuel design limits will not be exceeded anywhere in the core during normal operation including anticipated operational occurrences.

Fuel design evaluations have been performed and demonstrate that the 1% plastic strain fuel design limit is not exceeded during continuous operation with LHGR's up to the operating limit. The analysis also includes allowances for short term transient operation above the operating limit to account for anticipated operational occurrences, plus an allowance for densification power spiking.

A.2 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The **AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)** is a measure of the average linear heat generation rate of all the fuel rods in a fuel assembly at an axial location. Limits on APLHGR are specified to assure that the fuel design limits will not be exceeded during anticipated operational occurrences and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) will not exceed the limits specified in 10 CFR 50.46.

Cycle design calculations often refer to the ratio of the highest value of APLHGR in the core (MAPLHGR, maximum APLHGR) divided by the APLHGR limit. This ratio is called MAPRAT or RAPLHGR, and must always remain below 1.0 throughout the cycle or the APLHGR limit will be exceeded.

A.3 MINIMUM CRITICAL POWER RATIO

The **MINIMUM CRITICAL POWER RATIO (MCPR)** is a measure of the operating fuel assembly power relative to the fuel assembly power that would result in the onset of boiling transition. The Safety Limit MCPR (SLMCPR) is set such that 99.9% of the fuel rods will avoid boiling transition if the limit is not violated. The operating limit MCPR

(OLMCPR) is established to assure that no fuel damage results during anticipated operational occurrences. Although fuel damage would not necessarily occur if a fuel rod actually experienced boiling transition, the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomena that is readily detected during the testing of various bundle designs. Based on this experimental data, correlations have been developed that are used to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., pressure, mass flux, subcooling, etc.). Since plant operating conditions and bundle power levels are relatively easily monitored and determined, monitoring MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur. Cycle design calculations often refer to the ratio of the OLMCPR divided by the smallest CPR calculated for the whole core. This ratio is called CPRRAT and must always remain below 1.0 throughout the cycle or the operating limit CPR will be exceeded.

Appendix B: Engineering, Technical and Cost Data

The engineering, technical and cost data requested in the attachment to the Requirements Document are provided in this Appendix. To avoid duplication, references are provided to the Tables or Figures in the main body of the text when appropriate.

A1. Occupational Radiation Exposure (person-rem) for the reactor complex vs. time

The charts describing the occupational exposure, both yearly as well as the accumulated values, are shown in Figs. 8.1-1 through 8.1-6 in the main body of the text.

Y

YEARS SINCE START OF WORK	SPIKING OPTION (MWT)	SPENT FUEL OPTION (MWT)	PU DESTRUCTION OPTION (MWT)
0	0	0	0
7	3926	3926	3926
7.5	3926	3926	3926
8	3926	3926	7850
8.5	3926	3926	11800
9	3926	3926	15700
9.5	3926	3926	19600
10	3926	3926	23600
15	3926	3926	23600
25	3926	3926	23600
67	3926	3926	23600

A2 MWT INSTALLED CAPACITY VS TIME

YEARS SINCE START OF WORK	SPIKING		SPENT FUEL		PU DESTRUCTION	
	MT/YEAR	CUMULATIVE	MT/YEAR	CUMULATIVE	MT/YEAR	CUMULATIVE
0	0	100	0	100.0	0	100.0
4	10.1	89.9	5.2	94.8	9.3	90.7
5	10.1	79.8	5.2	89.6	9.3	81.4
6	10.1	69.7	5.2	84.4	9.3	72.1
7	10.1	59.6	5.2	79.2	9.3	62.8
8	10.1	49.5	5.2	74.0	9.3	53.5
9	10.1	39.4	5.2	68.8	9.3	44.2
10	10.1	29.3	5.2	63.6	9.3	34.9
11	10.1	19.2	5.2	58.4	9.3	25.6
12	10.1	9.1	5.2	53.2	9.3	16.3
13	9.1	0	5.2	48.0	9.3	7.0
14			5.2	42.8	7.0	0
15			5.2	37.6		
16			5.2	32.4		
17			5.2	27.2		
18			5.2	22.0		
19			5.2	16.8		
20			5.2	11.6		
21			5.2	6.4		
22			5.2	1.2		
23			1.2	0		

**A3 WEAPONS PU STOCK PILE REDUCTION TO THE REACTOR COMPLEX
(WHEN SUPPLIED AS PU OXIDE)**

YEARS SINCE START OF WORK	SPIKING		SPENT FUEL		PU DESTRUCTION	
	INCREMENT	CUMM	INCREMENT	CUMM	INCREMENT	CUMM
	10 ⁹ KW HRS		10 ⁹ KW HRS		10 ⁹ KW HRS	
0	0	0	0	0	0	0
7	0	0	0	0	0	0
7.5	2.847	2.847	4.271	4.271	4.271	4.271
8	2.847	5.694	4.271	8.542	8.542	12.813
8.5	2.847	8.541	4.271	12.813	12.813	25.626
9	2.847	11.388	4.271	17.084	17.084	42.707
9.5	2.847	14.235	4.271	21.355	21.355	64.062
10	2.847	17.082	4.271	25.626	25.626	89.688
11	5.694	22.776	8.541	34.167	51.252	140.94
12	5.694	28.47	8.541	42.708	51.252	192.192
13	5.694	34.164	8.541	51.249	51.252	243.444
14	5.694	39.858	8.541	59.79	51.252	294.696
15	5.694	45.552	8.541	68.331	51.252	345.948
20	42.71	88.262	42.71	111.041	256.26	602.208
25	42.71	130.972	42.71	153.751	256.26	858.468
50	213.53	344.502	213.53	367.281	1281.3	2139.768
67	145.19	489.692	145.2	512.481	871.284	3011.052

A4 KW HRS OF ELECTRICITY PRODUCED VS TIME

A5 Kg. of Strategic Materials Required

All the materials needed for dispositioning are available on the commercial market. Table below provides required quantities for Uranium, Plutonium (to be dispositioned) and Gadolinium which may be construed as strategic materials. For the tritium case, the Li^6 required per year is also reported.

Material	Kg/ Bundle	Total Required, MT
<u>Spiking Case</u>		
Uranium	169.4	2492.0
Plutonium	6.8	100.0
Gadolinium	1.8	26.5
<u>Spent Fuel Case</u>		
Uranium	169.4	2492.0
Plutonium	6.8	100.0
Gadolinium	1.8	26.5
<u>Pu Destruction Case</u>		
Uranium	172.7	3259.0
Plutonium	5.3	100.0
Gadolinium	1.0	18.9
<u>Tritium Case</u>		
Lithium ⁶ Required per Bundle:		0.923 kg
Lithium ⁶ Required per Core Load (per year):		80.72 kg

A6 Input to the Reactor Kg. Isotopics of Fresh Reactor Fuel Vs. Time

Each bundle design utilizes rods of various Pu enrichments. The bundle designs are specified in the text as indicated below:

Spiking Case	Figure 2.5-1
Spent Fuel Case	Figure 2.6-1
Pu Destruction Case	Figure 2.7-1

The yearly plutonium loading into the reactor are as follows:

Spiking Case	13.53 MT
Spent Fuel	5.64 MT
Pu Destruction	5.19 MT

In all cases, the initial Pu feed material isotopics is assumed to be that given in Table 3.9-1.

A7 On-Site in Process and Spent Fuel Storage Requirements

Sufficient storage space has been built into the fuel fabrication complex for storage of fuel from the time of initial receipt to the time they are stored as fuel bundles. The storage requirements are based on Figs. 3.3-1 (Plutonium receiving as a function of time at the fuel plant) and Fig.3.3-2 (MT of MOX fuel throughput as a function of time). In both these charts, the time for initial reactor start up should be taken as 7.5 years after the start of work.

The discharged MOX fuel bundles from the reactor plant are shown in Fig. 3.3 -3 which are used for determining the size of the additional ex-reactor storage pool. In Fig. 3.3-3, the reactor start up should be taken as 7.5 years after the start of work.

A8 High Level Waste Generated

Both high level and low level waste generation have been calculated. Information relative to the low level waste is given in Figures 8.1-7 through 8.1-12

The high level waste generated consists entirely of irradiated bundles. The total bundle weight including the heavy metal and bundle hardware is approximately 270 kg. The bundle volume is approximately 0.085 cubic meters. The high level waste weights and volumes given below are based on these individual bundle weight and volume.

Spiking Case:

HLW Generated per Year: 518.4 MT, 163.2 cubic meters
Total HLW Generated: 3971 MT, 1250 cubic meters
Total increases linearly with time, starting 7.5 years after start of work, as the reactor plant comes on-line.

Spent Fuel Case:

HLW Generated per Year: 226.3 MT, 71.3 cubic meters
Total HLW Generated: 3971 MT, 1250 cubic meters
Total increases linearly with time, starting 7.5 years after start of work, as the reactor plant comes on-line

Pu Destruction Case

HLW Generated per Year: 275.2 MT, 86.7 cubic meters
Total HLW Generated: 5066 MT, 1596 cubic meters
Total increases nearly linearly with time, starting 7.5 years after start of work, as the reactor plants come on-line.

ACTINIDE	ACTINIDE OUTPUT (KGS/YR)																	
	YEAR AFTER REACTOR START UP																	
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
Th 232	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
U 233	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
U 234	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	1.281	2.752
U 235	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	251.597	540.54
U 236	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	2.501	5.374
U 238	130944	130944	130944	130944	130944	130944	130944	130944	130944	130944	130944	130944	130944	130944	130944	130944	130944	281325
Pu 239	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	4651.776	9994.05
Pu 240	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	397.747	854.535
Pu 241	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	49.889	107.184
Pu 242	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	0.75	1.612
Pu 243	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Pu 244	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	0.958	2.058
Pu 245	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.218	0.469
Pu 246	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	1.243	2.871
Am 241	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	14.739	31.667
Am 242	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Am 243	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.017	0.038
Th 238	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Th 239	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Cm 242	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	0.955	2.053
Cm 244	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

A 9 ACTINIDE OUTPUT VS TIME - SPENT FUEL CASE

ACTINIDE	ACTINIDE CONTENT (KGS/YR)									
	YEAR AFTER REACTOR START UP									
	1	2	3	4	5	6	7	8		
Th 232	0	0	0	0	0	0	0	0	0	0
U 233	0	0	0	0	0	0	0	0	0	0
U 234	3.003	3.003	3.003	3.003	3.003	3.003	3.003	3.003	1.98	
U 235	557.952	557.952	557.952	557.952	557.952	557.952	557.952	557.952	367.9	
U 236	19.834	19.834	19.834	19.834	19.834	19.834	19.834	19.834	13.078	
U 238	326400	326400	326400	326400	326400	326400	326400	326400	215220	
Pu 239	10089.6	10089.6	10089.6	10089.6	10089.6	10089.6	10089.6	10089.6	6652.83	
Pu 240	1545.408	1545.408	1545.408	1545.408	1545.408	1545.408	1545.408	1545.408	1019.003	
Pu 241	364.224	364.224	364.224	364.224	364.224	364.224	364.224	364.224	240.16	
Pu 242	18.484	18.484	18.484	18.484	18.484	18.484	18.484	18.484	12.188	
Pa 231	0	0	0	0	0	0	0	0	0	
Np 237	8.521	8.521	8.521	8.521	8.521	8.521	8.521	8.521	5.619	
Pu 238	4.771	4.771	4.771	4.771	4.771	4.771	4.771	4.771	3.146	
Np 239	3.483	3.483	3.483	3.483	3.483	3.483	3.483	3.483	2.297	
Am 241	35.155	35.155	35.155	35.155	35.155	35.155	35.155	35.155	23.18	
Am 242	0	0	0	0	0	0	0	0	0	
Am 243	0.991	0.991	0.991	0.991	0.991	0.991	0.991	0.991	0.653	
Th 228	0	0	0	0	0	0	0	0	0	
Th 230	0	0	0	0	0	0	0	0	0	
Cm 242	5.823	5.823	5.823	5.823	5.823	5.823	5.823	5.823	3.84	
Cm 244	0.058	0.058	0.058	0.058	0.058	0.058	0.058	0.058	0.038	

A 9 ACTINIDE OUTPUT VS TIME - SPIKING CASE

AVERAGE ACTINIDE CONTENT (KGS/YR)																			
YEAR AFTER REACTOR START UP																			
ACTINIDE	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	
Th 232	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
U 233	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
U 234	0	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	6.425	
U 235	0	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	422.792	
U 236	0	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	160.039	
U 238	0	185472	185472	185472	185472	185472	185472	185472	185472	185472	185472	185472	185472	185472	185472	185472	185472	185472	
Pu 239	0	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	1562.023	
Pu 240	0	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	1308.647	
Pu 241	0	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	543.422	
Pu 242	0	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	246.649	
Pa 231	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
Np 237	0	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	30.667	
Pu 238	0	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	34.822	
Np 239	0	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	2.562	
Am 241	0	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	31.492	
Am 242	0	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	0.552	
Am 243	0	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	48.102	
Th 228	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
Th 230	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	
Cm 242	0	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	11.134	
Cm 244	0	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	14.42	

R12

A 9 ACTINIDE OUT PUT VS TIME - PU DESTRUCTION CASE

		FISSION PRODUCTS (KGS/YR)																	
		YEAR AFTER REACTOR START UP																	
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
	SPENT CASE	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	1520.64	3267.00
	SPIKE CASE	4788.48	4788.48	4788.48	4788.48	4788.48	4788.48	4788.48	3157.40	0	0	0	0	0	0	0	0	0	0
	IPU DESTRUCTION CASE	0	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14	5033.14

A 10 FISSION PRODUCT OUTPUT VS TIME - ALL CASES

A11 - A16: Cost Data

Cost Data, covering the data requirements in A11-A16, are presented in Section 10.

**DATE
FILMED**

9 / 16 / 93

END