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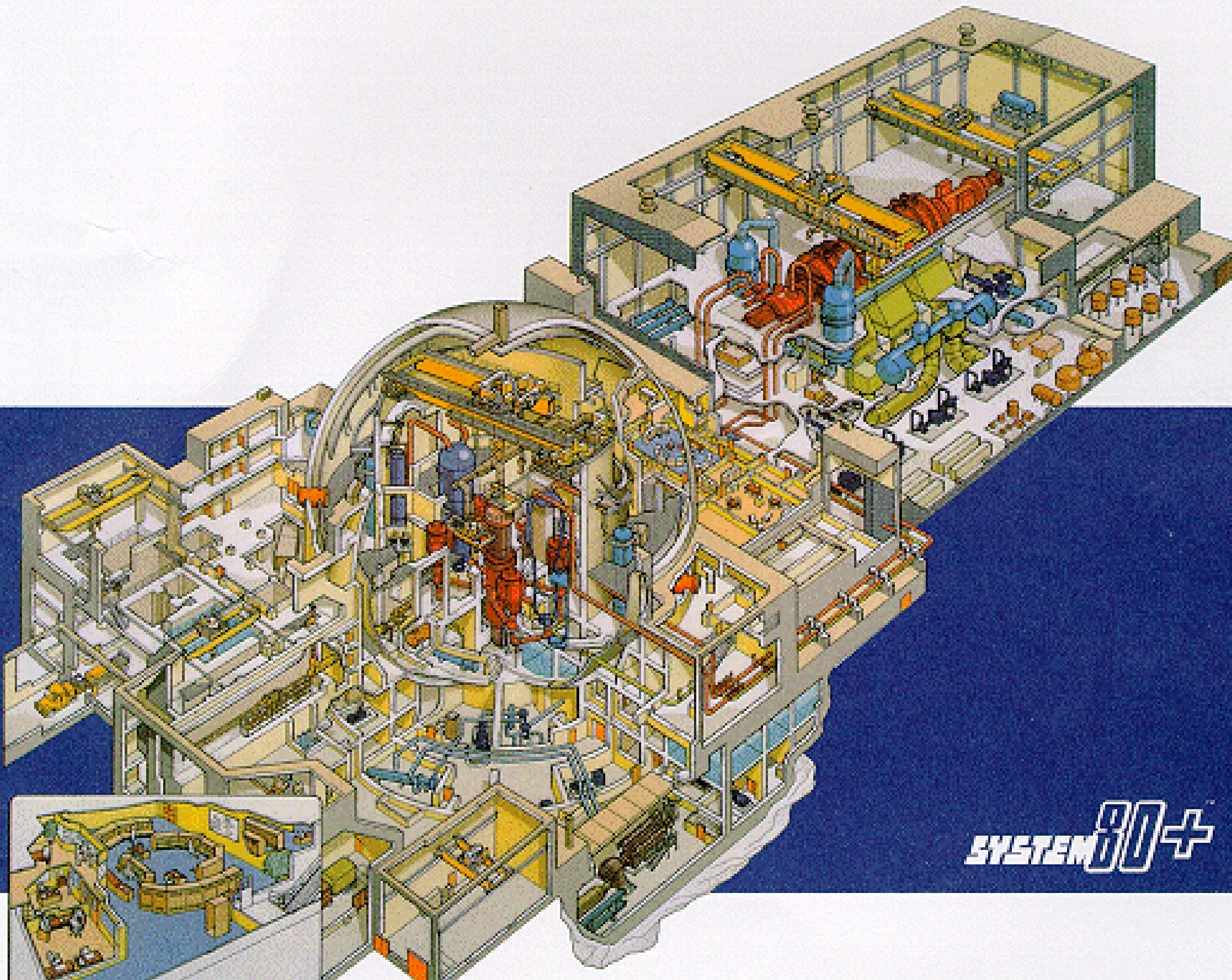
DOE Plutonium Disposition Study
Pu Consumption in ALWRs

Contract No. DE-AC03-93 SF19682



A Final Report

by

ABB-Combustion Engineering
Windsor, Connecticut

May 15, 1993

ABB

ASEA BROWN BOVERI

DOE/SF/19682--T1-Vol.2

DOE PLUTONIUM DISPOSITION STUDY

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MASTER

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ABB Combustion Engineering
Windsor, CT 06095

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LIST OF ACRONYMS AND TERMINOLOGY

Generally, terms and abbreviations are defined at the time they are first introduced in the following report. The more commonly found terms and abbreviations which are appropriate to the System 80+™ station design and Plutonium Disposition Study are compiled below as a convenient reference for the reviewer. In the interest of practical application, note very possible scientific term or abbreviation is listed.

ABB-CE	Asea Brown Boveri-Combustion Engineering
ADV	Atmospheric Dump Valve
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
APR	All Plutonium Reactor
APS	Alternate Protection System
ATWS	Anticipated Transient Without Scram
AVS	Annulus Ventilation System
BNWL, BNL	Battelle Northwest Laboratories
BPR	Burnable Poison Rod
CAS	Central Alarm Station
CBC	Critical Boron Concentration
CCTV	Closed Circuit Television
CCW(S)	Component Cooling Water (System)
CEA	Control Element Assembly
CEDM	Control Element Drive Mechanism
CEG	Cost Estimating Guidelines
CIS	Commonwealth of Independent States
CM	Corrective Maintenance
CPC	Core Power Calculator
CRT	Cathode Ray Tube
CS	Containment Spray (System)
CSAS	Containment Spray Actuation Signal
CSB	Core Support Barrel
CVCS	Chemical and Volume Control System
D&D	Decontamination & Decommissioning
DE&S	Duke Engineering and Services
DIAS	Discrete Indication and Alarm System
DIT	Discrete Integral Transport
DNB	Departure from Nucleate Boiling
D-O	Destruction Deployment Option
DPS	Data Processing System
DVI	Direct Vessel Injection
ECA	Energy Conversion Area
EFPD	Effective Full Power Days
EFW(S)	Emergency Feedwater (System)
EFWST	Emergency Feedwater Storage Tank
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
ESFAS	Emergency Safety Features Actuation Signal

LIST OF ACRONYMS AND TERMINOLOGY (Continued)

ESW(S)	Essential Service Water (System)
EWG	Electric Wholesale Generator
FEA	Fuel Element Assembly
FMEF	Fuels and Material Examination Facility
FPF	Fuel Pin Fabrication
FRS	Fuel Receiving and Storage
FTC	Fuel Temperature Coefficient
FTFF	Fuel and Target Fabrication Facility
GVR	Gas-to-Volume Ratio
GWD	Gigawatt-Days
GWMS	Gaseous Waste Management System
HACTS	Head Area Cable Tray Structure
HEPA	High Efficiency Particulate Air (Filter)
HJTC	Heated Junction Thermocouple
HPSI	High Pressure Safety Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
ICI	In-Core Instrumentation
ID	Intrusion Detection
INPO	Institute of Nuclear Power Operations
IPP	Independent Power Producer
IPSO	Integrated Process Status Overview
IRWST	In-Containment Refueling Water Storage Tank
ITAAC	Inspection, Test, Analysis Acceptance Criteria
LCO	Limiting Conditions of Operation
LDB	Licensing Design Basis
LOCA	Loss of Coolant Accident
LTOP	Low Temperature Over Pressurization
LWMS	Liquid Waste Management System
MAA	Material Access Area
MFIV	Main Feedwater Isolation Valve
MOPS	Moisture Preseparators
MOX	Mixed Oxide
MRS	Material Receiving and Storage
MSIV	Main Steam Isolation Valve
MSSA	Master Safeguards and Security Agreement
MST	Multiple Stud Tensioner
MT, MTU	Metric Ton (Uranium)
MTC	Moderator Temperature Coefficient
MWD	Megawatt-Days
MW(t)	Megawatt-Thermal
NEPA	National Environmental Policy Act
OBE	Operating Basis Earthquake
O&M	Operation and Maintenance
ONM	Other Nuclear Materials

LIST OF ACRONYMS AND TERMINOLOGY (Continued)

PBRC	Plutonium Burner Reactor Complex
PBRF	Plutonium Burner Reactor Facility
PDR	Plutonium Disposition Reactor
PDS	Plutonium Disposition Study
PM	Preventive Maintenance
PRA	Probabilistic Risk Assessment
PRF	Permeation Reduction Factor
PSN	Project Summary Network
PVNGS	Palo Verde Nuclear Generating Station
RCB	Reactor Containment Building
RCM	Reliability Centered Maintenance
RCP	Reactor Coolant Pump
RCGV	Reactor Coolant Gas Vent
RCS	Reactor Coolant System
RD	Requirements Document
RDT	Reactor Drain Tank
RPCS	Reactor Power Cutback System
RPS	Reactor Protection System
RSPT	Reed Switch Position Transmitter
SAF	Secure Automated Fabrication
SAS	Secondary Alarm Station
SBO	Station Black Out
SC	Shutdown Cooling (System)
SCRUPS	Special Cross-Under Pipe Separators
SCV	Steel Containment Vessel
SDS	Safety Depressurization System
SF-O	Spent Fuel Deployment Option
SF-1	Spent Fuel Deployment Option - 1 Reactor
SF-2	Spent Fuel Deployment Option - 2 Reactors
SFS	Spent Fuel Storage
SG	Steam Generator
SGR	Self Generated Recycle
SI(S)	Safety Injection (System)
SIT	Safety Injection Tank
S-O	Spiking Deployment Option
SMB	Safety Margin Basis
SNM	Special Nuclear Materials
SRS	Savannah River Site
SSE	Safe Shutdown Earthquake
SSSP	Site Safeguards and Security Plan
SWEC	Stone and Webster Engineering Corporation
SWMS	Solid Waste Management System
T&Q	Training and Qualification (Program)
TTDP	Tritium Target Development Program
UGS	Upper Guide Structure
URD	Utility Requirements Document

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LIST OF ACRONYMS AND TERMINOLOGY (Continued)

WANO	World Association of Nuclear Operators
WNP, WNPP	Washington Nuclear Power Project

III. PLUTONIUM FUEL CYCLE

A. SYSTEM 80+ PLUTONIUM BURNER DESIGN PARAMETERS

1. Reference System 80+ Reactor Design

The System 80+ standard PWR design is used as the reference design for the plutonium burner concept evaluated in this study. The System 80+ design has several advantages for this application, which include the following:

- System 80+ was specifically designed for maximum fuel management flexibility and can accommodate plutonium fuel loadings up to and including all-plutonium-reactor (APR) operation with relatively minor modifications.

- The System 80+ design is based on the proven System 80 design in operation at the Palo Verde Nuclear Generating Station (PVNGS). The evolutionary improvements in the System 80+ design are based on extensive plant operating experience, industry and regulatory feedback, and integrated design analyses using probabilistic risk assessment (PRA).

- The System 80+ design conforms with the EPRI Utility Requirements for Evolutionary Advanced Light Water Reactors.

- The System 80 units currently under construction in Korea include numerous evolutionary features of the System 80+ design (e.g., ring-forged reactor vessel, greater design margins for major components, improvements to safety systems) and represent an active program of procurement, manufacturing and construction.

- The System 80+ reference design described in CESSAR-DC has completed extensive review by the NRC covering all regulatory requirements for new plant designs. The design successfully addresses all current US regulations and policies, and is scheduled for Final Design Approval in 1994.

Basic technical characteristics of the System 80+ design are included in the Technical Description in Section II of this report. The reference System 80+ design described in CESSAR-DC has a core power rating of 3914 MWt (reference UO_2 core design) and a corresponding thermal rating of the nuclear steam supply system (NSSS) of 3931 MWt, which includes the thermal input of the reactor coolant pumps. The reference System 80+ NSSS components consistent with this power rating are maintained for the plutonium burner design. However, the core power rating is reduced for the fuel cycle applications in this study (i.e., core power of 3800 MWt for plutonium burning fuel cycles, and core power of 3410 MWt for the tritium production fuel cycle). In these applications the core power is limited in order to maintain the same level of core thermal margin as the reference UO_2 fuel cycle.

The pertinent characteristics of the System 80+ reactor which provide for plutonium burning fuel cycles are unique design features of the fuel assembly, control element assemblies (CEAs) and the reactor internals which increase control rod coverage of the core. Although the reader is referred to a more general description in Section II of this report, these features are briefly summarized below:

a. The core is comprised of 241 fuel assemblies, each assembly having a 16x16 fuel rod array with five large structural guide tubes (each guide tube occupies 2x2 fuel lattice locations), as shown in Figure III.A-1. The four outer guide tubes are for CEA fingers (or elements), while the center guide tube is for in-core instrumentation. The in-core instruments are bottom-entry and therefore do not interfere with the upper internals design for CEA guidance.

b. The control element assemblies have either 4- or 12-element arrangements, as illustrated in Figure III.A-2. The large CEA element design (for the 2x2 guide tube) provides a higher degree of mechanical ruggedness and increased absorber surface area per element than in PWR designs where the control rod fingers occupy a single fuel rod lattice location. The 12-element CEA mechanical design with B_4C neutron absorber is further shown by Figure III.A-3.

c. The 12-element CEA has the unique characteristic of inserting into five adjacent fuel assemblies, as illustrated by Figure III.A-4. This characteristic is made possible by the upper guide structure design of the reactor internals which provides continuous guidance for each individual CEA element into the fuel assembly guide tube, while providing adequate flow area for primary coolant exiting the core. The upper guide structure, illustrated in Figure III.A-5, is a rugged, all-welded structure and protects each CEA element from flow forces and dynamic loads associated with seismic events and design basis accidents.

d. The CEA pattern for the reference System 80+ design, shown in Figure III.A-6, consists of forty-eight (48) full-strength 12-element CEAs, twenty (20) full-strength 4-element CEAs, and 25 part-strength 4-element CEAs, or a total complement of ninety-three (93) CEAs. The pattern using 12-element CEAs enables coverage of adjacent fuel assemblies by CEAs, so that a large portion of the fuel assemblies (213 of 241 assemblies) contain either four or two CEA elements. This provides a high degree of core shutdown worth through distribution of CEA elements over the core. The 12-element CEAs are used in shutdown banks. The 4-element full strength CEAs are used in regulating banks. The 4-element part-strength CEAs (which contain Inconel absorber) are provided for rodded maneuvering.

System 80+ is designed to accommodate plutonium fuel in the form of PuO_2-UO_2 mixed-oxide (MOX). The mechanical characteristics of MOX fuel are similar to those of UO_2 fuel. The nuclear and irradiation characteristics of MOX fuel for lower fissile plutonium loadings characteristic of commercial LWR fuel reprocessing are established based on early evaluation (e.g., the US Generic Environmental Statement on Mixed Oxide Fuel in LWRs issued in 1974) furthered by the experience in commercial fuel reprocessing outside the US.

The System 80+ reference design with the CEA pattern shown in Figure III.A-6 can accommodate MOX fuel loadings up to the level of self-generated recycle (SGR) without modification. SGR is defined as the amount of plutonium generated by the reference UO_2 fuel cycle. This would allow approximately one-third of the feed fuel assemblies to contain MOX fuel, while the remaining feed assemblies would contain UO_2 fuel. Design modifications to accommodate higher loadings of MOX fuel, including all-plutonium-reactor (APR) operation, are described below.

2. Design Modifications for APR Operation

Utilization of commercial MOX fuel at the SGR and APR levels has been extensively investigated for the System 80 design (Refs. III-1 through III-5). The early design studies showed that design modifications are required for PWR systems to accommodate large loadings of MOX fuel. These modifications include additional control rods to provide required shutdown margin, equipment modifications to accommodate higher soluble boron concentrations, core and spent fuel cooling equipment sized to accommodate the higher decay heat loads associated with irradiated MOX fuel, design of the reactor vessel and internals to tolerate a greater flux of high energy neutrons than arises in uranium fueled operation, modifications to the radwaste systems to accommodate higher tritium activity in the primary coolant, and design of fuel storage and fuel handling facilities to safely accommodate MOX fuel.

Table III.A-1 summarizes the basic impact of APR operation on PWR plant system design requirements. The System 80 design was specifically developed to accommodate MOX fuel loadings up to and including APR. Consequently, design requirements for APR operation were incorporated in the basic systems of the System 80 NSSS, or design provision made which facilitate modifications for APR operation. These system features to enable APR operation have been preserved in the evolutionary System 80+ design. The summary below describes physical effects of MOX fuel operations at the APR level and the accommodation of these effects in the System 80+ APR design.

a. Irradiated MOX fuel exhibits higher long-term decay heat generation rates and longer decay times than irradiated UO_2 fuel. Typical decay heat loads for APR operation are higher by approximately twenty percent than for UO_2 operation one day after shutdown and continue to diminish more slowly with time. This higher heat load must be accommodated in the design of plant cooling systems. For the System 80+ design, the higher heat loads are accommodated in the following systems:

   - Shutdown Cooling System (SCS)
   - Containment Spray System (CSS)
   - Spent Fuel Pool Cooling System (SFPCS)
   - Component Cooling Water System (CCWS)

b. Higher soluble boron concentrations are required in the primary coolant due to lower reactivity worth of B^{10} with MOX cores. For APR operation the required soluble boron concentrations are approximately doubled relative to UO_2

operation. For the System 80+ design, the higher soluble boron requirements for APR operation are accommodated by increasing the size and processing capacities of the Chemical and Volume Control System (CVCS), and by increasing the soluble boron concentration in the Safety Injection System (SIS) and In-containment Refueling Water Storage Tank (IRWST).

c. The tritium concentration in the primary coolant is substantially higher for APR operation than for UO_2 operation. This results from the higher operating concentrations of soluble boron causing increased tritium production by the $B^{10}(n, 2\alpha)H^3$ reaction. The resulting tritium buildup in the primary coolant is approximately seventy percent higher for APR operation in comparison to UO_2 operation. For the System 80+ design the higher tritium levels are accommodated in the design and operation of the liquid and gaseous radwaste systems, and provision of a tritium removal system for APR operation.

d. The rate of high energy (>1 MeV) neutron irradiation of the reactor vessel and internals is increased by approximately six percent for APR operation in comparison to UO_2 operation. This is due to an increase in the number of prompt neutrons emitted in plutonium fission and a slightly higher average energy of the fission neutrons. The higher neutron fluence levels are accommodated by design and materials controls of the System 80+ reactor vessel and internals.

e. Gamma emission rates are higher by approximately twenty percent for APR operation compared to UO_2 operation. This leads to correspondingly higher heating rates which are accommodated by the design of the reactor internals.

f. Radioactive decay of plutonium isotopes (and small quantities of americium) in fresh MOX fuel requires provision of shielding in the fuel receipt, handling and inspection area.

g. The relative individual control rod worth is reduced by twenty-five to thirty percent for APR operation in comparison to UO_2 or SGR operation. The control rod requirements for APR operation are accommodated in the System 80+ design by incorporating an extended CEA complement. The extended CEA complement is achieved starting with the reference ninety-three (93) CEA pattern, shown in Figure III.A-6, and modifying the CEA pattern by utilizing the eight (8) spare CEA nozzles provided in the reference System 80+ design, and by utilizing full-strength (B_4C absorber) CEAs in all locations. The resulting extended CEA pattern for APR operation is shown in Figure III.A-7. This pattern provides coverage of 221 of 241 fuel assembly locations by the full-strength CEAs. Because of the high shutdown worth of the reference System 80+ CEA pattern, and the modifications to increase the number and strength of CEAs, the extended CEA pattern provides the necessary shutdown requirements for APR operation. Core maneuvering is more restricted for APR operation due to the elimination of part-strength CEAs and rodded operating restrictions associated with shutdown worth and safety margins. Normal operating capabilities for startup, shutdown, power operations, and power level changes are not significantly affected, however.

3. Safety Implications of APR Operation

The evaluation of commercial MOX fuel utilization for the System 80 design included fuel management and safety analyses for fuel cycles transitioning from UO_2 operation to equilibrium SGR or equilibrium APR operation. Table III.A-2 gives the characteristics of comparative equilibrium cycles for UO_2, SGR and APR operation. The safety related physics characteristics for these fuel cycles are summarized in Table III.A-3.

The parameters in Table III.A-3 show trends in the core physics characteristics with higher loadings of plutonium. These trends are expected based on the nuclear properties of Pu^{239} in comparison to U^{235}. A major effect of increased plutonium loadings is stronger thermal absorption in the fuel which alters various core physics parameters. In particular, the reactivity worth of soluble boron and control rods are reduced, and the prompt neutron lifetime (ℓ^*) is reduced. The delayed neutron fraction (β_{eff}) is also reduced with increased plutonium loading. The change in these parameters is relatively small from UO_2 to SGR operation (since U^{235} reactions are predominant) and greater for APR operation (where Pu^{239} reactions are predominant). Consequently, the required soluble boron concentrations are approximately doubled for APR operation in comparison to UO_2 or SGR operation, and the extended CEA complement is required for APR operation.

Moderator temperature coefficient (MTC) and fuel temperature coefficient (FTC) are affected to a lesser extent with higher plutonium loadings. MTC is more negative at beginning-of-cycle (BOC) conditions for SGR or APR operation. For end-of-cycle (EOC) conditions the MTC for APR operation is comparable to that for UO_2 operation, while the MTC for SGR operation is more negative. FTC becomes slightly less negative with the higher plutonium loadings.

Basic safety implications of the core physics characteristics for APR operation are summarized below.

a. The effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (ℓ^*), which are important to short term power transients, are decreased for APR operation. While this result in itself would appear to have an adverse effect upon short period transients such as a rod ejection accident, the overall consequence is mitigated by the lowered reactivity worth of the ejected rod and a reduced sensitivity of the core power distribution to local reactivity perturbations. These mitigating effects are a consequence of the strong thermal absorption properties which reduce the thermal diffusion length of the MOX fuel lattice.

   CEA ejection analyses previously performed for SGR and APR operations of the System 80 design at full power and hot zero power initial conditions have shown acceptable consequences in all cases (i.e., comparable to results expected for UO_2 operation). For the System 80 and System 80+ designs the control rods allowed to be inserted in the core when the reactor is critical are of the 4-element type. The insertable reactivity worths of 4-element full-strength CEAs are small in comparison to β_{eff}, so that the core power transient is small in comparison to the local power transient. The core power transients associated with the CEA ejection events for APR operation were, in fact, predicted to self-limiting below the power conditions which would be expected to result in a reactor trip, despite the lower values of β_{eff}, ℓ^*, and fuel temperature coefficient. The more favorable results analyzed for APR operation are a consequence of a less adverse initial power distribution, reduced ejected CEA worth, and reduced response of the core power distribution to the reactivity insertion.

b. For events with decrease in primary coolant temperature, the negative moderator temperature coefficient associated with UO_2, SGR or APR operation results in a positive reactivity insertion. The positive reactivity insertion results in a power increase transient which is opposed by the negative fuel temperature coefficient and may, for larger cooldown events, result in a reactor trip. The extended CEA pattern for APR operation is provided to offset the reduced individual CEA worth in order to provide adequate scram worth for the most limiting cooldown events. It is noted that cooldown events are more limiting near end-of-cycle (EOC) for the equilibrium cycles due to the more negative MTC values at EOC, as shown in Table III.A-3. The CEA worth increases as a function of burnup for plutonium fuel cycles (as shown in Section III.F), thus providing higher scram worth for the most limiting postulated cooldown events near end-of-life.

c. For events associated with reduced reactor coolant flow or reduced heat removal the consequences are characterized by a decreased margin to departure from nucleate boiling (DNB). The plutonium content of the fuel does not affect the consequences of such events to any significant degree.

d. The consequences for loss of coolant accidents (small LOCA or large LOCA) are not expected to be significantly affected by the plutonium content of the fuel. A potential difference for APR operation is in the requirement to prevent post-LOCA boric acid build-up during the long-term emergency cooling. APR operation requires a higher concentration of soluble boron in the safety injection system and the in-containment refueling water storage tank (IRWST) for System 80+. The reference design boron concentrations in these systems for APR and SGR operations are 6200 ppm and 4400 ppm, respectively. Analyses performed for APR operation of the reference System 80 design indicate that operator response time to provide hot-leg injection flow during long-term cooling based on standard procedures (i.e., several hours after the event) is sufficient for the most limiting postulated large LOCA.

4. <u>Design Features for Utilizing Weapons Grade Plutonium</u>

The System 80+ design for utilizing weapons grade plutonium is based on the reference design modified for APR operation. Specifically, the extended CEA pattern and plant system requirements as described in Section III.A.2 are implemented in the design. Additional features are provided based upon consideration of the higher fissile content of weapons-grade plutonium (versus plutonium from commercial reprocessing) and specific mission requirements in the DOE Plutonium Disposition Study Requirements Document. Major requirements include:

- Disposition of 100 MT of weapons-grade plutonium within a period of 25 years after October 1993;

- Design for three fuel cycle alternatives for plutonium disposition, which are designated Plutonium Spiking, Spent Fuel, and Plutonium Destruction (the requirements for each alternative are described separately in Sections III.C, III.D and III.E below);

- In all cases, the design should produce electric power and be capable of producing tritium. Recommended changes to optimize the design for tritium production should be included.

These requirements lead to several practical considerations, reflected in the design objectives for this study:

- The reactor design should be capable of accommodating large loadings of weapons-grade plutonium. This would favor a large core size and the capability for APR operation utilizing weapons-grade feed plutonium in order to accomplish the plutonium burning mission with realistic constraints on capital investment.

- The reference reactor design and features for APR operation should be based to the maximum extent on proven technology, proven operating experience of the reference design, and assurance of licensability based on substantial completion of NRC licensing review of the reference design as a new plant design. These considerations are essential in order to realistically meet the schedule for design, construction, startup and disposition of 100 MT of weapons-grade plutonium within the schedule period from October 1993 to October 2018.

- The reference design should have the flexibility to accommodate the required fuel cycle alternatives and the requirement for tritium production operation without major in-service modification of plant systems and reactor design features. The design differences for these modes of operation should be limited to fuel assembly design details and core operating power level.

The additional System 80+ nuclear design features which address the requirements and design objectives for utilizing weapons-grade plutonium are described below.

a. Mixed-Oxide Fuel Design

Table III.A-4 shows relative concentrations of plutonium discharge isotopes for a reference 18-month UO_2 fuel cycle of the System 80+ design (average discharge burnup of approximately 48 GWD/MTU). This provides a basis of comparison of differences of feed fuel for the weapons-grade plutonium burner versus a "commercial-grade" plutonium burner (i.e., using reprocessed plutonium from UO_2 discharge fuel). Secondly, it provides a basis for comparing the discharge plutonium isotope ratios for the weapons-grade plutonium burner (i.e., Spent Fuel Alternative) with those of the reference UO_2 fuel cycle characteristic of the System 80+ design.

The feed fuel concentrations of plutonium isotopes, particularly Pu^{239} and Pu^{240}, are a principal consideration in the utilization of weapons-grade plutonium in the nuclear design. The effects of basic differences between the weapons-grade and commercial-grade plutonium on the safety-related physics parameters were evaluated for the fuel cycle alternatives developed in this study and are summarized in Section III.F. The results indicate that safety-related characteristics of APR operation do not change significantly for utilization of weapon-grade plutonium fuel compared to use of reprocessed plutonium from commercial LWRs.

The specifications for the weapons-grade plutonium are expected to vary relative to the values in Section III.F. These may include variations in the concentration of the Pu^{239} and Pu^{240} isotopes, presence of small concentrations of Pu^{241}, Pu^{242}, Am^{241}, etc. These variations are of less significance to the core nuclear design than to the fuel fabrication process, however, and can be accommodated without significant modification of the core and fuel cycle designs described below for the System 80+ plutonium burner.

The fuel design used (except for the Plutonium Destruction Alternative) is mixed-oxide (MOX), consistent with reference System 80+ design for commercial APR operation. Based on the design objective of providing as high as practical loading of weapons-grade plutonium in the core, the APR design utilizes MOX feed fuel in the form of PuO_2-UO_2-Er_3O_2, with the following characteristics:

- Weapons-grade plutonium comprising approximately 7 wt% of the heavy metal (HM);
- Uranium tails (0.2 wt% U^{235} tails assay) comprising approximately 93 wt% of the HM;
- Erbium burnable poison admixed in the form of natural Er_3O_2 in the metal-oxide with typical concentrations of 1-2 wt% of the MOX fuel.

The loading of approximately 7 wt% weapons-grade plutonium in the System 80+ APR design enables 100 MT of the material to be loaded in approximately fifteen (15) full cores. The use of uranium tails and erbium burnable poison facilitates the nuclear characteristics for reactivity and power distribution control with the high plutonium fissile loading, in an analogous fashion to design applications for higher burnup, higher enrichment UO_2 fuel cycles.

The use of uranium tails in the fuel is desirable in order to minimize additional fissile content in the fuel (i.e., essentially eliminate the effects of U^{235}). It is also desirable from the standpoint of reducing the uranium tails inventoried at

DOE uranium separation facilities. The presence of U^{238} prolongs the depletion of Pu^{239} over lifetime due to its fertile characteristic (i.e., conversion to Pu^{239} by neutron absorption reactions). The presence of U^{238} provides beneficial effects on the nuclear design characteristics, however, including partially offsetting the low β_{eff} of Pu^{239} and providing for a more gradual change in core physics parameters over lifetime. The isotope characteristics of the plutonium in discharge MOX fuel at end-of-life (Spent Fuel Alternative) were evaluated to be similar to those of plutonium in discharge UO_2 fuel, as shown in Table III.A-4. In particular, the relative concentration of Pu^{240} in the discharge plutonium is approximately twenty-three percent in both cases.

The use of erbium as a burnable poison in the MOX fuel is an innovative design application for the plutonium burner, which provides substantial benefits for accommodating high concentrations of Pu^{239}. Erbium is a rare earth, similar in chemical and metallurgical properties to gadolinium. Like gadolinium, erbium is comprised of several natural occurring isotopes. The natural abundancies and depletion chain of erbium are illustrated in Figure III.A-8. Er^{167} is the primary neutron absorber. The energy-dependent neutron absorption properties of Er^{167} include a large double resonance in the vicinity 0.5 ev, as shown in Figure III.A-9. This enhances the thermal neutron absorption of erbium (i.e., providing a non-1/v absorption characteristic), and provides the additional characteristic of improving the negative fuel temperature and moderator temperature coefficients due to the location of the resonance at the high end of the thermal energy spectrum. In contrast to gadolinium, erbium has a slower depletion characteristic as a burnable poison, releasing reactivity gradually over a longer period of fuel burnup.

Erbium has been extensively used in TRIGA (Ref. III-6) to provide a more negative fuel temperature coefficient for the high enrichment uranium fuel. ABB-CE has more recently developed the application of erbium as a burnable poison for PWRs, in the form of Er_2O_3 admixed with enriched UO_2. This application was developed as an optimized burnable poison design for 18- and 24-month UO_2 fuel cycles (i.e., the cycle lengths currently in operation for all US ABB-CE plants). For extended UO_2 cycle lengths the erbium burnable poison design shows major advantages of improving thermal margins (reducing power peaking over long cycle lengths by distribution of the required burnable poison over a large number of fuel rod locations) and providing a negative moderator coefficient at beginning-of-cycle (enabling high total loading of erbium to control excess reactivity with higher UO_2 enrichments). The ABB-CE erbium burnable poison design has completed irradiation demonstrations in two operating ABB-CE plants and is scheduled for full batch implementation by 1994. The design has been generically approved by the NRC for Er_2O_3 concentrations up to 2.5 wt% in enriched UO_2 (Ref. III-7).

The application of erbium burnable poison offers key benefits for the System 80+ plutonium burner design, analogous to the benefits provided for longer UO_2 fuel cycles. These include the following:

- The admixture of Er_2O_3 in MOX is analogous to its use in UO_2 fuel and provides the capability to accommodate high fissile plutonium loading. Since the erbium poison is admixed homogeneously in the fuel it provides the ability to control a large amount of excess reactivity, while precluding the possibility of loss of this reactivity control by any mechanism, including misoperation or mechanical disassembly of the fuel.

- The 0.5 ev neutron absorption resonance of Er^{167} overlaps significantly with the 0.3 ev resonance of Pu^{239}, as shown by Figure III.A-9. This enhances the neutron absorption worth of erbium burnable poison in comparison to use of purely 1/v absorbers, such as B^{10}, which have significantly diminished reactivity worth in the presence of a high loading of Pu^{239}. Consequently, the required reactivity holddown for 7 wt% loadings of weapons-grade plutonium in the MOX is provided with low concentration of Er_2O_3 (each wt% of Er_2O_3 corresponds to approximately 6%$\Delta\rho$ reactivity holddown at full power conditions).

- The long-term reactivity control characteristics and ability to vary the distribution of the erbium concentration over the fuel lattice provide a high degree of flexibility for control of power distribution over lifetime, in order to minimize peaking factors and provide a high degree of thermal operating margin.

b. Non-Fertile Fuel Design

A non-fertile fuel design has been evaluated for the Plutonium Destruction Alternative (see Section III.E). The concept evaluated utilizes a ceramic pellet design of similar geometry characteristics to the MOX or UO_2 pellet designs, but consisting of Al_2O_3-PuO_2-Er_2O_3. In this design, the loading of weapons-grade plutonium is unchanged relative to that of the MOX fuel design described above. Al_2O_3 is provided as a non-fertile diluent, replacing the UO_2 tails in the MOX design. Because the U^{238} is eliminated by this design, the rate of destruction of Pu^{239} would be achieved at a higher rate with fuel burnup, with significantly lower achievable levels of Pu^{239} in the discharge fuel than for the MOX design. The selection of Al_2O_3 for this purpose is based on its high melting temperature and thermal conductivity characteristics, and extensive PWR experience in long-term core-irradiation applications (e.g., Al_2O_3-B_4C burnable poison rods used in ABB-CE cores since the 1970's). The loading of Er_2O_3 for the non-fertile fuel design is significantly higher than the level for the MOX design (e.g., the equivalent of using 4 wt% Er_2O_3 in the MOX pellet design). The higher erbium loading is required to compensate for the removal of U^{238} reaction rates in the non-fertile design, and to provide a more negative fuel temperature coefficient in the absence of U^{238}. The reduction of β_{eff} due to the elimination of U^{238} is not compensated for in the non-fertile fuel design.

c. Fuel Assembly Design

The fuel assembly design for both the MOX and non-fertile fuel designs described above is based on the reference System 80+ 16x16 fuel assembly design. The analyses of fuel depletion for the Spent Fuel Alternative and Plutonium Destruction Alternative showed that it is desirable to include a limited number of Al_2O_3-B_4C burnable poison rods in the fuel lattice. The reference fuel assembly designs for the System 80+ plutonium burner concept are based on the use of either PuO_2-UO_2-Er_3O_2 MOX fuel rods or Al_2O_3-PuO_2-Er_2O_3 non-fertile fuel rods (Plutonium Destruction Alternative). Fuel assembly design arrangements for the System 80+ Plutonium Burner core design are shown in Figure III.A-10. The basic fuel assembly types shown in this figure are designated 0-shim, and 12-shim arrangements.

The 0-shim fuel assembly arrangement contains 236 fuel rods, which is the maximum number of fuel rod locations provided in the standard System 80+ 16x16 assembly design.

The 12-shim fuel assembly arrangement incorporates twelve Al_2O_3-B_4C burnable poison rods in the fuel lattice. Each 12-shim fuel assembly contains 224 fuel rods and 12 non-fuel burnable poison rods. The Al_2O_3-B_4C burnable poison rods are located in a standard arrangement used in ABB-CE UO_2 fuel assemblies, as shown in Figure III.A-10. Unlike standard ABB-CE application, which have the burnable poison rods permanently fixed in the fuel lattice, 12-shim design for the plutonium burner application has the Al_2O_3-B_4C burnable poison rods contained in non-structural guide tubes within the fuel assembly (each non-structural guide tube occupies 1x1 lattice locations). The burnable poison rods are designed to be insertable/removable by removing the upper end fitting of the fuel assembly in order to access the burnable poison rods. Such operations would be required infrequently, however, and would not be on the critical path of fuel cycle operations. Removing and replacing the upper end fitting of the ABB-CE fuel assembly design is a simple operation, but requires use of special tools in a controlled area of the spent fuel pool. Therefore, mishandling of the burnable poison rods would be precluded during normal core loading and offloading operations.

Table III.A-5 includes a summary of fuel assembly design parameters and fuel cycle characteristics for the MOX fuel design which is applied for the Plutonium Spiking and Spent Fuel Alternatives. Table III.A-6 provides a similar summary for the non-fertile fuel design which is applied for the Plutonium Destruction Alternative. The System 80+ Plutonium Burner fuel cycles represented in these tables use 0-shim and 12-shim fuel assembly designs in 160 and 81 core locations, respectively. The inclusion of Al_2O_3-B_4C burnable poison rods in the fuel cycle design serves the following purposes:

- The Al_2O_3-B_4C burnable poison rods supplement the long-term reactivity holddown of the erbium burnable poison and facilitate the design for a gradual, negative rundown characteristic of the fuel k_∞ with burnup;

- The Al_2O_3-B_4C burnable poison rods can be selectively removed prior to fuel load in later cycles (e.g., fourth annual cycle for Spent Fuel Alternative or Plutonium Destruction Alternative) in order to remove the residual reactivity holddown. This feature adds flexibility for fuel management and achieving cycle length near end-of-life;

- Target rods for tritium production can be substituted for the Al_2O_3-B_4C in any operating cycle (except near end-of-life) in order to provide tritium production capability. This capability would exist in all cases (as specified by the DOE Requirements). The evaluation of tritium production (see Section III.G) indicates that substitution of the tritium production target rods, which contain Li^6, can be accommodated at different times in life due to similarity of the reactivity holddown characteristic relative to the Al_2O_3-B_4C burnable poison design (note that both Li^6 and B^{10} have a 1/v thermal neutron absorption characteristic).

To meet contract quantity tritium production requirements set forth in DOE guidance, multiple System 80+ Plutonium Burner units would be required to meet the required tritium production rate capability using the core designs described in Table III.A-5 or III.A-6. However, Table III.A-7 describes a Tritium Production core design which provides the capability for meeting the specified tritium production rate with a single System 80+ Plutonium Burner unit. This design uses a 32-shim assembly arrangement, as shown in Figure III.A-11, for accommodating either Al_2O_3-B_4C burnable poison rods or target rods for tritium production. The tritium production capability is described in more detail in Section III.G.

d. Core Thermal Rating

Table III.A-8 summarizes the core thermal parameters for the System 80+ Plutonium Burner design in three modes of power operation. The core designs for which these modes of power operation apply are described below:

- <u>UO_2 Fuel Cycle.</u> This mode of power operation applies for the reference System 80+ UO_2 fuel cycle design, which is an 18-month cycle length design using Er_2O_3-UO_2 burnable poison. Other UO_2 fuel cycle designs with cycle lengths ranging from 12-months to 24-months are also available for this mode of power operation. The core power level is 3914 MWth, consistent with the reference System 80+ design described in CESSAR-DC.

- <u>Plutonium Disposition.</u> This mode of power operation applies for the Plutonium Spiking and Spent Fuel Alternatives described in Sections III.C and III.D, respectively, using the PuO_2-UO_2-Er_3O_2 MOX core design features described in Table III.A-5. Alternatively, this mode of operation applies for the Plutonium Destruction Alternative describe in Section III.E, using the Al_2O_3-PuO_2-Er_2O_3 non-fertile core design features described in Table III.A-6. In this mode of power operation the core power level is limited to 3800 MWth in order to maintain the same core thermal operating margins as in the reference System 80+ design, accounting for the displacement of fuel rod locations by Al_2O_3-B_4C burnable poison rods or target rods.

- Tritium Production. This mode of power operation applies for the single-unit Tritium Production core design described in Table III.A-7. The core and fuel cycle design is based on PuO_2-UO_2-Er_3O_2 MOX fuel, with the capability to accommodate 32 target rods per fuel assembly. (An alternate Tritium Production design using enriched UO_2 fuel in lieu of MOX fuel is also possible, but was not evaluated for this study.) The core power rating for this mode of operation is limited to 3410 MWth in order to maintain the same core thermal operating margins as in the reference System 80+ design, accounting for the displacement of fuel rod locations by target rods or Al_2O_3-B_4C burnable poison rods.

5. References

III.A-1 "Assessment of PWR Plutonium Burners for Nuclear Energy Centers," Technical Information Center, US Energy and Research Development Administration, C000-2786, June 1976.

III.A-2 A. J. Frankel, P.C. Rohr and N. L. Shapiro, "PWR Plutonium Burners for Nuclear Energy Centers," ABB Combustion Engineering, TIS-4847, ANS/CNA Joint Meeting, Toronto, June 1976.

III.A-3 R. L. Hellens, "Problems in Recycle of Plutonium in Pressurized Water Reactors," ABB Combustion Engineering, TIS-3283, ANS Winter Meeting, Miami, October 1971.

III.A-4 R. L. Hellens and N. L. Shapiro, "Plutonium Fuel Management Options in Large Pressurized Water Reactors," ABB Combustion Engineering TIS-3779, ANS Winter Meeting, San Francisco, November 1973.

III.A-5 C. K. Anderson and R. H. Klinetob, "Plutonium Burning Light Water Reactor Concept," TIS-7016, ANS Topical Meeting on Technical Bases for Nuclear Fuel Cycle Policy, September 1981.

III.A-6 "TRIGA LEU Shrouded Fuel Cluster Design", UZR-14A, IAEA TECDOC-233.

III.A-7 "Methodology for Core Designs Containing Erbium Burnable Absorbers," CENPD-383-P, October 1990, and CENPD-382-P, Supplement 1-P, February 1992.

TABLE III.A-1

APR IMPACT ON PWR SYSTEM REQUIREMENTS

System or Component	System Requirements Changes
Plant Cooling System	(1) Increased Core Decay Heat Removal Capacity for Plant Cooldown and Safety (2) Accommodation of Increased Long Term Decay for Spent Mixed-Oxide Fuel
Chemical and Volume Control System (CVCS)	(1) Increased Maximum Soluble Boron Concentrations in Primary System and CVCS Components (2) Increased Capacities for CVCS Processing and Waste Water Holdup
Safety Injection Systems	(1) Increased Maximum Soluble Boron concentration in IRWST and Safety Injection Tanks
Control Element Assembly (CEA) Complement	(1) Increased Number of CEAs to Accommodate Reduced Individual CEA Worth
Fresh Fuel Handling and Storage Facility	(1) Shielding of Gamma and Neutron Sources from Fresh Mixed-Oxide Fuel
Spent Fuel Storage Facility	(1) Increased Storage Capacity due to Lower Average Discharge Burnup and Potentially Longer Storage Time (2) Accommodation of Altered Reactivity Characteristics of Mixed-Oxide Fuel in Conjunction of Uranium-Oxide Fuel
Radwaste System	(1) Addition of Tritium Removal System to Accommodate Higher Tritium Production Rate in Primary Coolant

TABLE III.A-2

MIXED-OXIDE FUEL CYCLE CHARACTERISTICS

	Equilibrium Cycle UO_2	Equilibrium Cycle SGR	Equilibrium Cycle APR
Cycle Length MWD/(MWd/t(metal))	11,400	11,400	11,400
Average UO_2 Feed Enrichment	3.29	3.62	--
Average Mixed Oxide Feed Enrichment (w/o Fissile Pu)	--	3.05	4.57
Number of UO_2 Assemblies	241	157	0
Number of Mixed Oxide Assemblies	0	84	241
Core Plutonium Inventory (Total Pu)			
Beginning-of-Cycle	421.2 Kg	2228.1 Kg	8439.1Kg
End-of-Cycle	740.0 Kg	2148.9 Kg	7824.4 Kg
Core Plutonium Inventory (Fissile Pu)			
Beginning-of-Cycle	336.9 Kg	1205.6 Kg	4279.9 Kg
End-of-Cycle	561.8 Kg	1233.7 Kg	3829.0 Kg

TABLE III.A-3

SAFETY RELATED PHYSICS CHARACTERISTICS FOR MIXED-OXIDE CYCLES

	Equilibrium Cycle UO_2	Equilibrium Cycle SGR	Equilibrium Cycle APR
Beginning of Cycle Reactivity (CEAs Withdrawn, No Dissolved Boron), ρ			
Hot Standby	0.137	0.122	0.083
Full Power, No Xenon	0.121	0.103	0.064
Full Power, Equilibrium Xenon	0.101	0.081	0.055
Dissolved Boron Requirements			
PPM Dissolved Boron for Criticality - CEAs Withdrawn			
BOC Hot Standby	1589	1820	3189
BOC Full Power, No Xenon	1400	1539	2450
BOC Full Power, Equilibrium Xenon	1170	1208	2100
Requirement for Refueling (5% Subcritical)	1955	2383	4203
Inverse Boron Worth (PPM/% $\Delta\rho$)			
Full Power BOC	116	149	383
Full Power BOC	101	130	331
Moderator Temperature Coefficient (10^{-4} $\Delta\rho$/°F)			
Full Power BOC	-0.59	-0.95	-1.00
Full Power EOC	-3.24	-3.73	-3.10

TABLE III.A-3 (Cont'd)
SAFETY RELATED PHYSICS CHARACTERISTICS FOR MIXED-OXIDE CYCLES

	Equilibrium Cycle UO_2	Equilibrium Cycle SGR	Equilibrium Cycle APR
Fuel Temperature Coefficient (10^{-5} $\Delta\rho/°F$)			
Full Power BOC	-1.24	-1.08	-1.01
Full Power EOC	-1.25	-1.17	-1.09
Neutron Kinetics Parameters			
Prompt Neutron Lifetime (μsec)			
Beginning-of-Cycle	21.3	17.0	6.8
End-of-Cycle	24.8	19.5	7.9
Effective Delayed Neutron Fraction			
Beginning-of-Cycle	0.00625	0.00567	0.00442
End-of-Cycle	0.00546	0.00518	0.00447
Available Control Rod Worth			
Total (%$\Delta\rho$)	13.8	13.5	12.6[a]
Net[c] (%$\Delta\rho$)	10.2	9.9	9.8[a]

(a) APR core with extended CEA complement

TABLE III.A-4

SYSTEM 80+ UO2 EQUILIBRIUM CYCLE

CORE POWER, MWt	3914
NUMBER FUEL ASSY	241
FEED BATCH ASSY	80
FEED ENRICHMENT, wt%	4.20
FEED U, MTU	34.98
CYCLE LENGTH, months	18
CYCLE LENGTH, EFPD	432
AVG CAP FACTOR, %	79
AVG DISCHG BU, GWD/T	47.8
DISCHG Pu, kg	389.8
DISCHG Pu238/Pu	0.018
DISCHG Pu239/Pu	0.527
DISCHG Pu240/Pu	0.232
DISCHG Pu241/Pu	0.154
DISCHG Pu242/Pu	0.070

Table III.A-5

System 80+ Pu Burner MOX Core Design Characteristics

Power Level	
Core	3800 MW(th)
Power Density	95.5 kW/liter
Average Linear Power(1)	17.7 kW/m (5.40 kW/ft)
Maximum Linear Power(1)	41.7 kW/m (12.7 kW/ft)
Core Dimensions	
Active Core Length	3.81 m (150 in)
Equivalent Core Diameter	3.65 m (143.6 in)
Fuel Assemblies	
Number	241
Dimensions	202.7 mm x 202.7 mm (7.98 in x 7.98 in)
Array	16x16
0-Shim Assembly	
Number Fuel Rods	236
12-Shim Assembly	
Number Fuel Rods	224
BPR Guide Tubes(2)	12
BPR Guide Tube(2)	
Outside Diameter	11.2 mm (0.440 in)
Thickness	0.91 mm (0.032 in)
Material	Zircaloy-4
Fuel Rods	
Outside Diameter	9.7 mm (0.382 in)
Cladding Thickness	0.64 mm (0.025 in)
Fuel Sintered Pellet Material	UO_2-PuO_2-Er_2O_3
Cladding Material	Zircaloy-4
Lumped Burnable Poison Rods (BPR)	
Number per 12-Shim Assembly	12
BPR Outside Diameter	8.7 mm (0.344 in)
Cladding Thickness	0.64 mm (0.025 in)
BPR Absorber Material	Al_2O_3-B_4C
BPR Cladding Material	Zircaloy-4

(1) Based on 0.975 average energy deposition fraction in the fuel.

(2) Non-structural guide tubes allow removal of BPRs for later cycles.

Table III.A-5 (Cont.)

System 80+ Pu Burner MOX Core Design Characteristics

Control Element Assemblies (CEAs)	
Number CEAs in Core	101
- 12-element Assemblies	48
- 4-element Assemblies	53
CEA Rod Outside Diameter	20.7 mm (0.816 in)
Cladding Thickness	0.89 mm (0.035 in)
CEA Absorber (all CEAs)	B_4C / Feltmetal and Reduced Diameter B_4C
Cladding Material	Inconel 625
Feed Fuel Batch	
Number of Assemblies	
0-Shim	81
12-Shim	160
Active Fuel Length	3.81 m (150 in)
Number of Fuel Rods	54956
Heavy Metal Feed	98.75 MTHM
Uranium (tails) Feed	92.08 MTU
Plutonium Total Feed	6.67 MTPu
Total Pu in HM	6.75 wt%
Uranium (tails) Feed Isotopes	99.8% U-238, 0.2% U-235
Plutonium Feed Isotopes	93.5% Pu-239, 6.5% Pu-240
Fissile Pu Feed	6.24 MTPu
Fissile Pu in HM	6.32 wt%
Mixed-Oxide (MOX) Composition	UO_2-PuO_2-Er_2O_3
Average Erbium in MOX	1.6 wt% Er_2O_3 in MOX pellets
BPRs in Feed Fuel Batch	
Number of Burnable Poison Rods	1920
Active Poison Length	3.45 m (136 in)
Average B-10 Loading in Poison	0.0102 g/cm (0.026 g/in)
Pu Spiking Alternative Fuel Cycle	
Average Capacity Factor	0.43
Cycle Length	3-months (39 EFPD)
Average Discharge Burnup	1500 MWD/MTHM
Spent Fuel Alternative Fuel Cycle	
Average Capacity Factor	0.75
Cycle Length	12-months (274 EFPD)
Number of Irradiation Cycles	4
Average Discharge Burnup	42,200 MWD/MTHM
Average Pu-240 in Discharge	23% of Total Pu Inventory

Table III.A-6

System 80+ Pu Burner Non-fertile Core Design Characteristics

Power Level	
Core	3800 MW(th)
Power Density	95.5 kW/liter
Average Linear Power[1]	17.7 kW/m (5.40 kW/ft)
Maximum Linear Power[1]	41.7 kW/m (12.7 kW/ft)
Core Dimensions	
Active Core Length	3.81 m (150 in)
Equivalent Core Diameter	3.65 m (143.6 in)
Fuel Assemblies	
Number	241
Dimensions	202.7 mm x 202.7 mm (7.98 in x 7.98 in)
Array	16x16
0-Shim Assembly	
Number Fuel Rods	236
12-Shim Assembly	
Number Fuel Rods	224
BPR Guide Tubes[2]	12
BPR Guide Tube[2]	
Outside Diameter	11.2 mm (0.440 in)
Thickness	0.91 mm (0.032 in)
Material	Zircaloy-4
Fuel Rods	
Outside Diameter	9.7 mm (0.382 in)
Cladding Thickness	0.64 mm (0.025 in)
Fuel Sintered Pellet Material	Al_2O_3-PuO_2-Er_2O_3
Cladding Material	Zircaloy-4
Lumped Burnable Poison Rods (BPR)	
Number per 12-Shim Assembly	12
BPR Outside Diameter	8.7 mm (0.344 in)
Cladding Thickness	0.64 mm (0.025 in)
BPR Absorber Material	Al_2O_3-B_4C
BPR Cladding Material	Inconel

(1) Based on 0.975 average energy deposition fraction in the fuel.

(2) Non-structural guide tubes allow removal of BPRs for later cycles.

Table III.A-6 (Cont.)

System 80+ Pu Burner Non-fertile Core Design Characteristics

Control Element Assemblies (CEAs)	
Number CEAs in Core	101
- 12-element Assemblies	48
- 4-element Assemblies	53
CEA Rod Outside Diameter	20.7 mm (0.816 in)
Cladding Thickness	0.89 mm (0.035 in)
CEA Absorber (all CEAs)	B_4C / Feltmetal and Reduced Diameter B_4C
Cladding Material	Inconel 625
Feed Fuel Batch	
Number of Assemblies	241
0-Shim	81
12-Shim	160
Active Fuel Length	3.81 m (150 in)
Number of Fuel Rods	54956
Fuel Composition	Al_2O_3-PuO_2-Er_2O_3
Plutonium Total Feed	6.67 MTPu
Plutonium Feed Isotopes	93.5% Pu-239, 6.5% Pu-240
Fissile Pu Feed	6.24 MTPu
Erbium Total loading (approx.)	4.6 MT Er_2O_3
BPRs in Feed Fuel Batch	
Number of Burnable Poison Rods	1920
Active Poison Length	3.45 m (136 in)
Pu Destruction Alternative Fuel Cycle	
Average Capacity Factor	0.75
Irradiation Cycle Length	12-months (274 EFPD)
Number of Cycles	4
Average Discharge Burnup	1096 EFPD

Table III.A-7

System 80+ Tritium Production Core Design Characteristics

Power Level	
Core	3410 MW(th)
Power Density	83.2 kW/liter
Average Linear Power[1]	17.75 kW/m (5.41 kW/ft)
Maximum Linear Power[1]	41.7 kW/m (12.7 kW/ft)
Core Dimensions	
Active Core Length	3.81 m (150 in)
Equivalent Core Diameter	3.65 m (143.6 in)
Fuel Assemblies	
Number	241
Dimensions	202.7 mm x 202.7 mm (7.98 in x 7.98 in)
Array	16x16
32-Shim Assembly	
Number Fuel Rods	204
TR Guide Tubes[2]	32
TR Guide Tube[2]	
Outside Diameter	11.2 mm (0.440 in)
Thickness	0.91 mm (0.032 in)
Material	Zircaloy-4
Fuel Rods	
Outside Diameter	9.7 mm (0.382 in)
Cladding Thickness	0.64 mm (0.025 in)
Fuel Sintered Pellet Material	UO_2-PuO_2-Er_2O_3
Cladding Material	Zircaloy-4
Target Rods (TRs)	
Number TRs in Core[3]	7712
Number TRs per Assembly	32
Target Rod Outside Diameter	8.7 mm (0.344 in)

[1] Based on 0.975 average energy deposition fraction in the fuel.

[2] Non-structural guide tubes allow insertion/removal of TRs.

[3] Burnable Poison Rods (BPRs) can be substituted for TRs if fuel is not to be used for production in any cycle.

Table III.A-7 (Cont.)

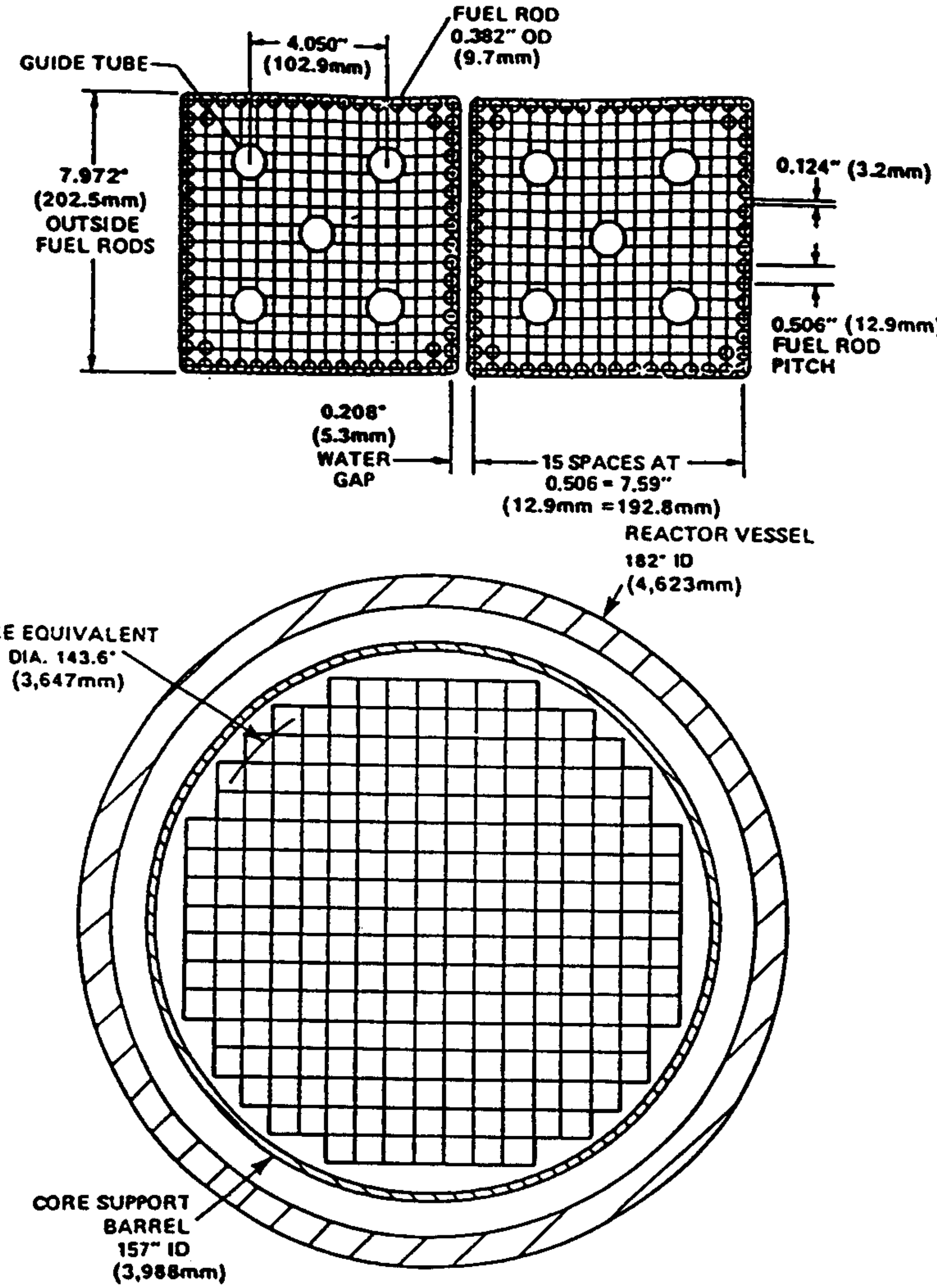
System 80+ Tritium Production Core Design Characteristics

Control Element Assemblies (CEAs)	
Number CEAs in Core	101
- 12-element Assemblies	48
- 4-element Assemblies	53
CEA Rod Outside Diameter	20.7 mm (0.816 in)
Cladding Thickness	0.89 mm (0.035 in)
CEA Absorber (all CEAs)	B_4C / Feltmetal and Reduced Diameter B_4C
Cladding Material	Inconel 625
Feed Fuel Batch	
Number of Assemblies	241 (Full Core)
Active Fuel Length	3.81 m (150 in)
Number of Fuel Rods	49164
Heavy Metal Feed	89.04 MTHM
Uranium Metal Feed	82.37 MTU
Plutonium Metal Feed	6.67 MTPu
Uranium Feed Isotopes	99.8% U-238, 0.2% U-235
Plutonium Feed Isotopes	93.5% Pu-239, 6.5% Pu-240
Pu-239 Concentration	7.00wt% Pu-239 in HM
Mixed-Oxide (MOX) Composition	UO_2-PuO_2-Er_2O_3
Average Erbium in MOX	1.2 wt% Er_2O_3 in MOX pellets
Core Operating Cycles	
Average Capacity Factor	0.75
Cycle Length	12-months (274 EFPD)
Number of Cycles	4
Average Discharge Burnup	42,200 MWD/MTHM
Average Pu-240 in Discharge	23% of Total Pu Inventory

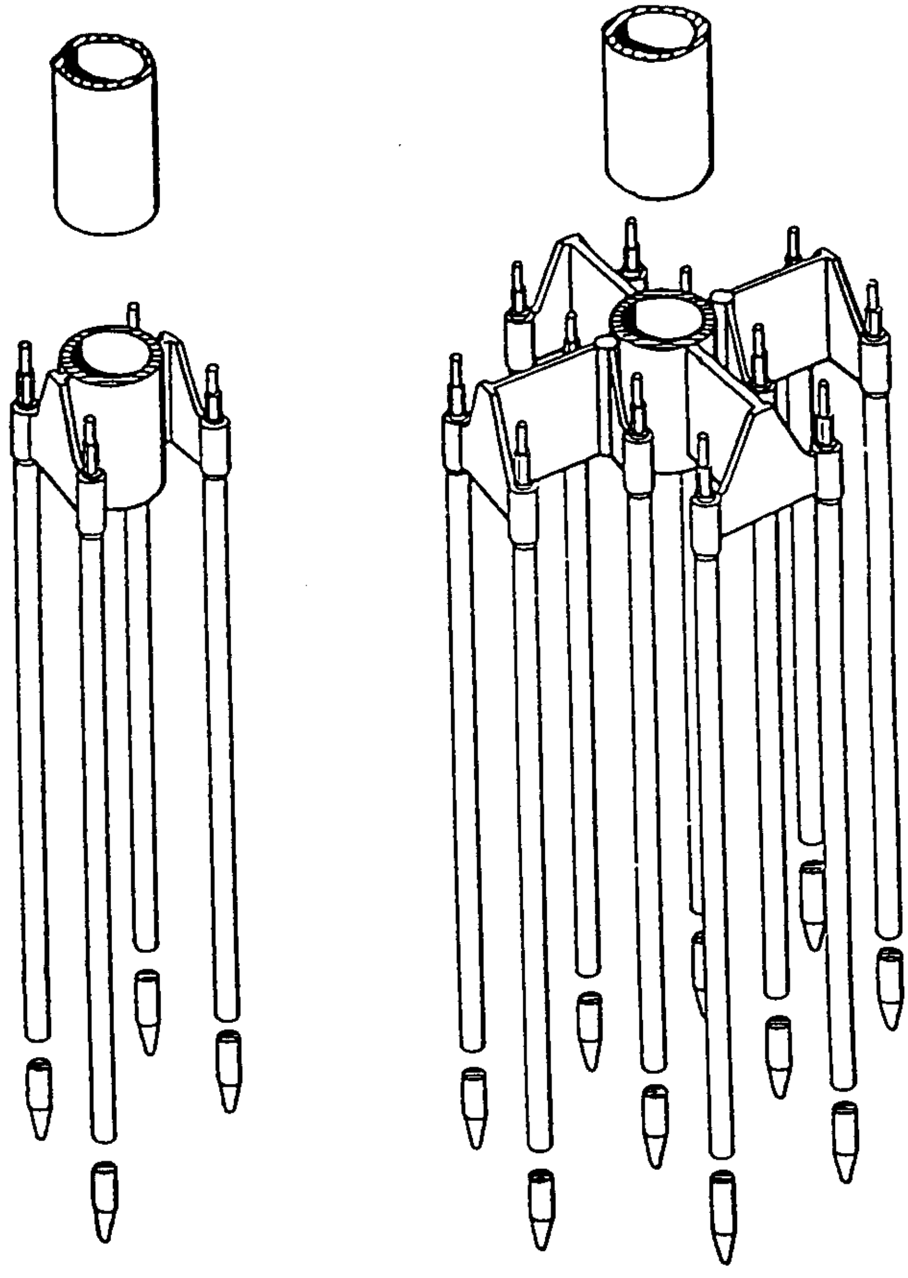
Table III.A-8

Thermal Output Data for System 80+ Plutonium Burner

Parameter	UO_2	Pu-Bnr	H^3-Prod
Core Thermal Output, MWth	3914	3800	3410
NSSS Thermal Output, MWth	3931	3817	3427
Percentage Reference NSSS Power	100%	97.10%	87.18%
Hot Leg Temperature, °F	611.	609.5	604.
Steam Pressure at SG outlet, psia	1012.	1014.	1023.4
Total Steam Flow, Mlbm/hr	17.66	17.08	15.15
Minimum Steam Quality	.9975	.9975	.9975
Feedwater Temperature, °F	450	447	437



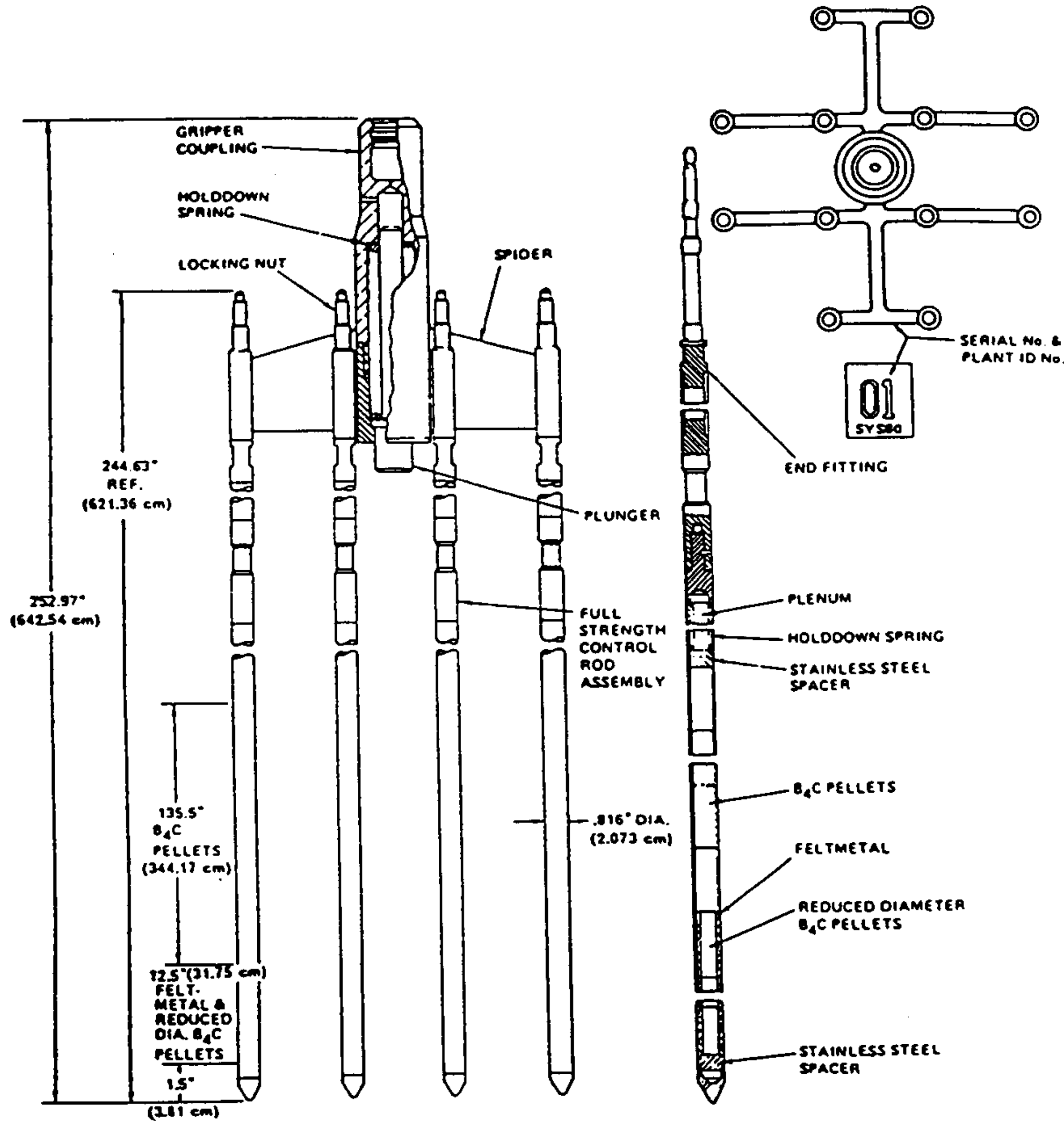
SYSTEM 80+™	REACTOR CORE CROSS SECTION	FIGURE III.A-1



4 FINGER CEA

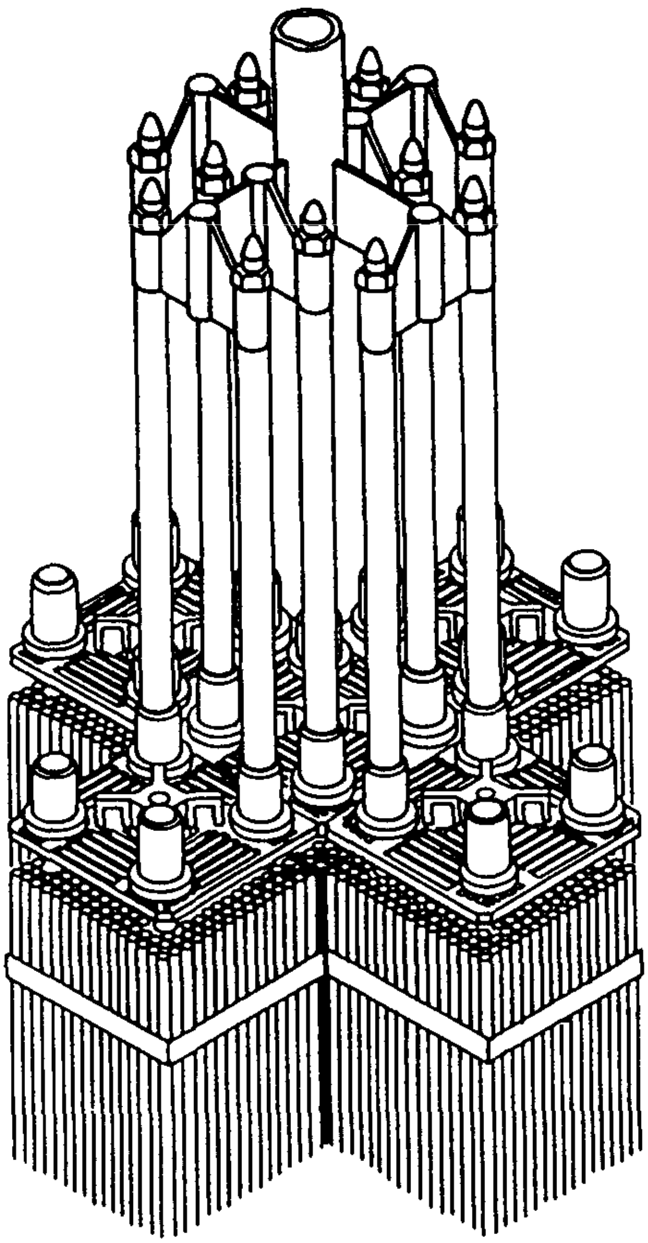
12 FINGER CEA

SYSTEM 80+™	4 AND 12 FINGER CONTROL ELEMENT ASSEMBLIES	FIGURE III.A-2

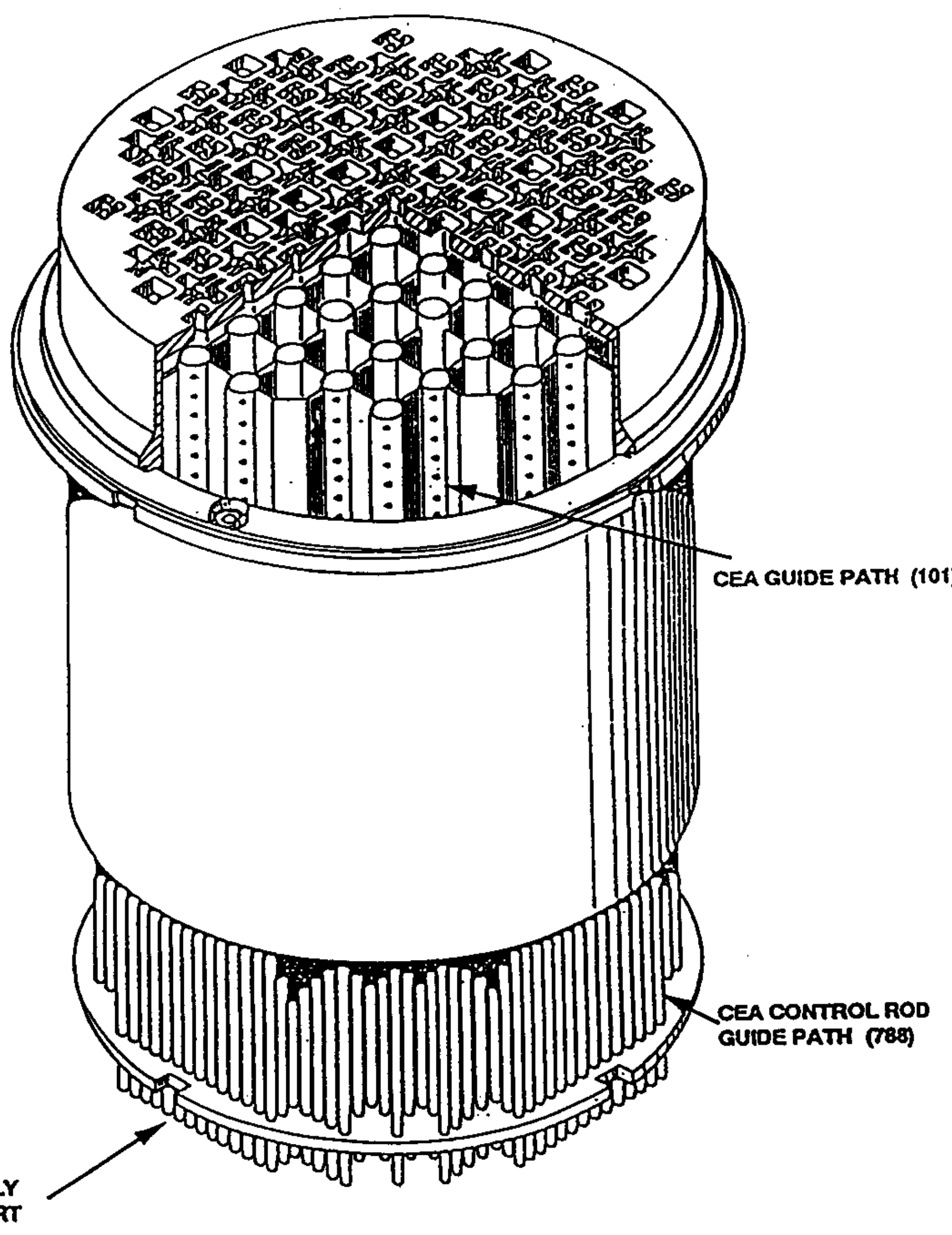


SYSTEM 80+™	FULL STRENGTH CONTROL ELEMENT ASSEMBLY (12 FINGER)	FIGURE III.A-3

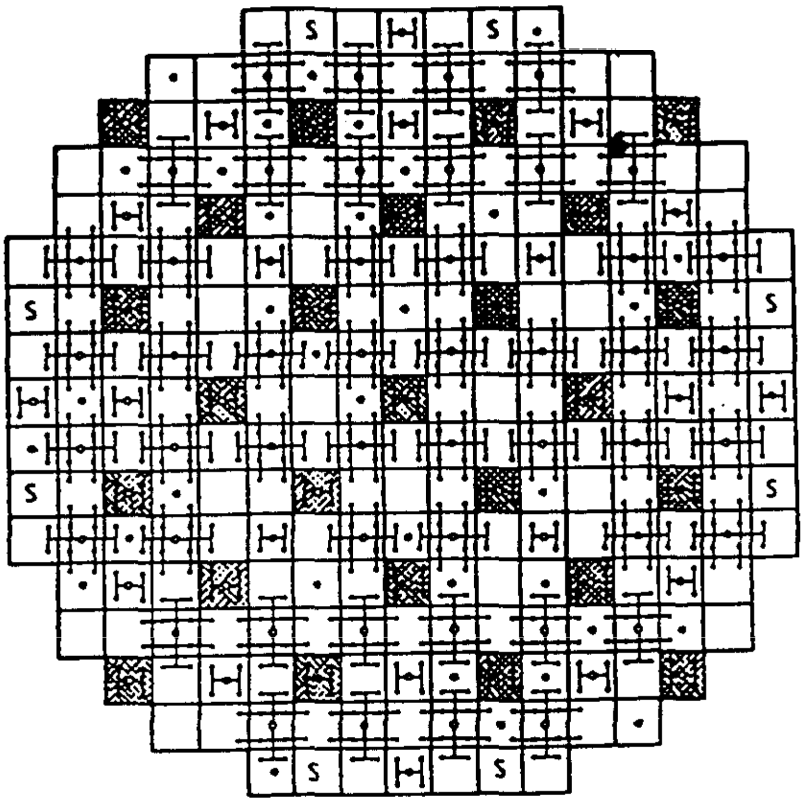
UPPER GUIDE STRUCTURE AND FUEL ASSEMBLY DETAILS
OMITTED FOR VISUALIZATION PURPOSES



SYSTEM 80+™	TWELVE-FINGERED CEA INSERTED IN FIVE FUEL ASSEMBLIES	FIGURE III.A-4



SYSTEM 80+™	UPPER GUIDE STRUCTURE ASSEMBLY	FIGURE III.A-5



12 ELEMENT FULL STRENGTH CEA (48)

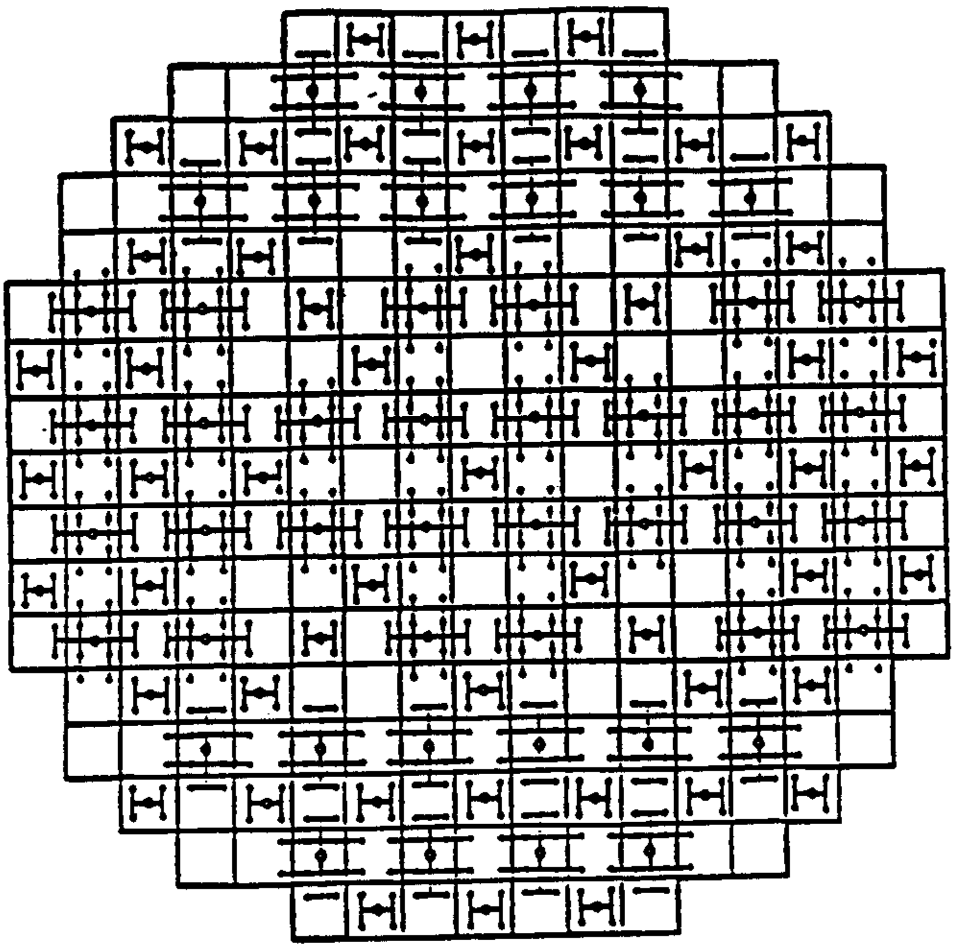
4 ELEMENT FULL STRENGTH CEA (20)

4 ELEMENT PART STRENGTH CEA (25)

S DENOTES SPARE CEA LOCATIONS FOR OPEN-MARKET PLUTONIUM RECYCLE (8)

DENOTES LOCATIONS OF FIXED RHODIUM IN-CORE NEUTRON DETECTOR STRINGS (61)

SYSTEM 80+™	CONTROL ELEMENT ASSEMBLY AND IN-CORE INSTRUMENT LOCATIONS	FIGURE III.A-6



12 ELEMENT FULL STRENGTH CEA (48)

4 ELEMENT FULL STRENGTH CEA (53)

FIGURE III.A-8

ERBIUM ABUNDANCES AND DEPLETION CHAIN

$$\begin{tikzcd} 33.4\% & 22.9\% & 27.1\% & & 14.9\% & \\ ER^{166} \arrow[r, dashed] & ER^{167} \arrow[r, dashed] & ER^{168} \arrow[r, dashed] & ER^{169} \arrow[r] \arrow[d] & ER^{170} \arrow[r] & ER^{171} \arrow[d] \\ & & & TM^{169} \arrow[r] & TM^{170} \arrow[r] & TM^{171} \end{tikzcd}$$

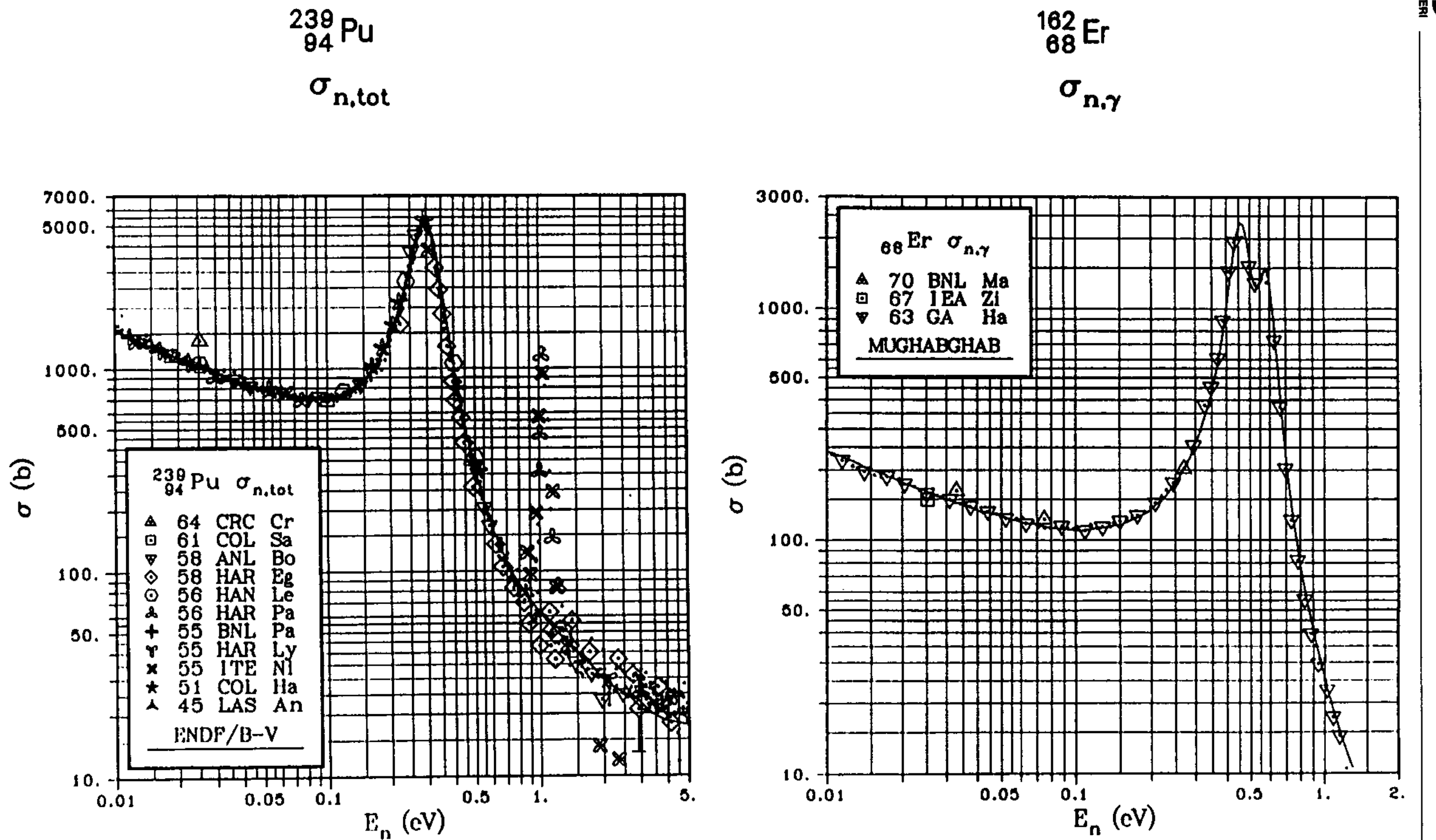


FIGURE III.A-9
COMPARISON OF Er^{162} AND Pu^{239} NEUTRON CROSS CHARACTERISTICS

FIGURE III.A-10
SYSTEM 80 + PLUTONIUM TRITIUM PRODUCTION FUEL ASSEMBLY LAYOUT

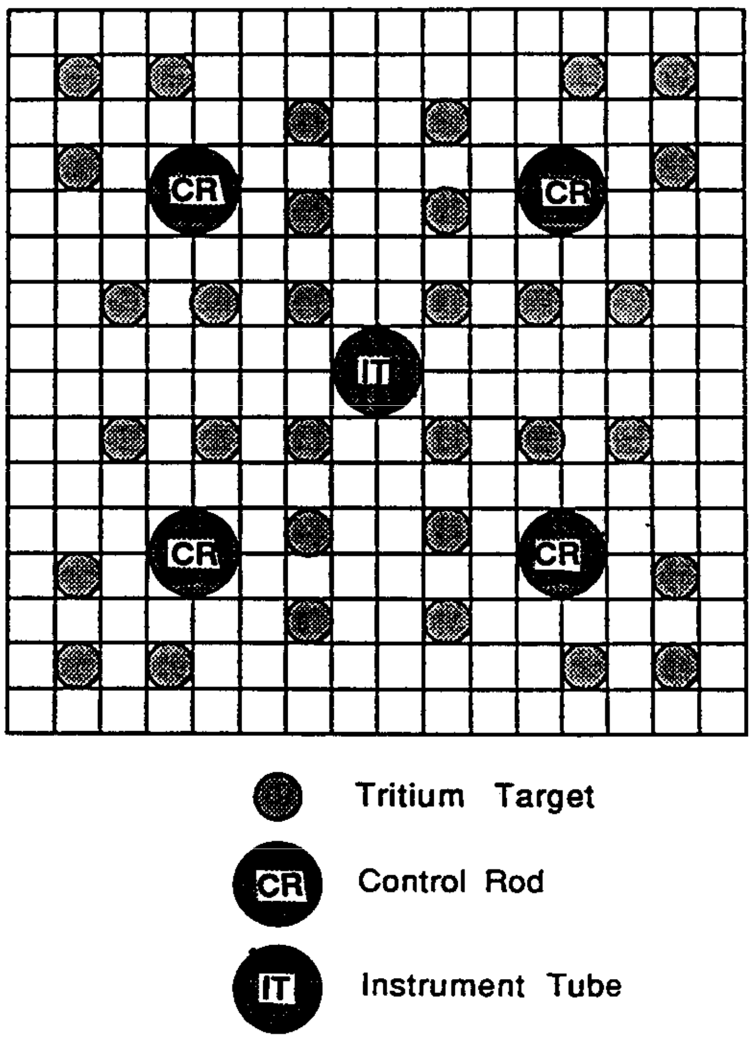
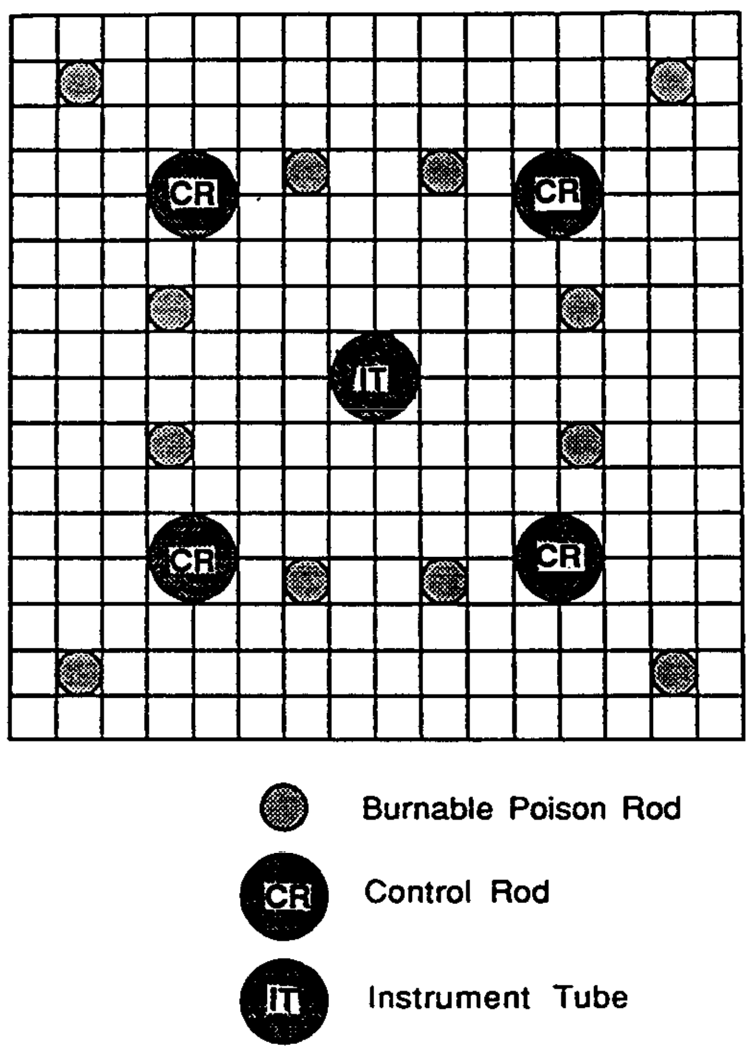


FIGURE III.A-11
SYSTEM 80 + PLUTONIUM BURNER FUEL ASSEMBLY LAYOUT



B. COMPUTER CODES

1. General

This section presents an overview description of principal computer codes for nuclear design and safety analysis of the System 80+ Standard Plant Design as described in CESSAR-DC. The conceptual analyses for the weapons-grade, all-plutonium-reactor fuel cycle alternatives in this study are based principally on the Discrete Integral Transport (DIT) assembly transport theory code which is a basic component of ABB-CE's reactor physics methodology. The DIT analyses are fundamental to nuclear design applications, including fuel management, core power distribution, transient and safety analyses. Consistent with established design practice, the fuel cycle concepts presented in this study may be developed and analyzed in greater detail using the nuclear design and safety analysis computer codes described below.

2. Principal Codes for Nuclear Design and Safety Analysis

A brief summary is given below for principal nuclear design and safety analysis used by ABB-CE, categorized by design application. The codes described below are supported by numerous processing and editing codes which automate the design process.

a. Fuel Management Codes

These codes are used to perform scoping, enrichment setting and final design fuel management calculations in 2-D or 3-D diffusion theory and in both coarse and fine mesh. In addition these codes are used extensively in the calculation of safety data for thermal hydraulic and system analysis.

ROCS

The ROCS (Reactor Operation and Control Simulator) code is a coarse mesh, two-energy group neutronics code which allows the user to model all aspects of reactor operations from startup to refueling. Because of its structure and advanced mathematical formulations, ROCS is a more cost effective design tool for fuel management and core follow than fine mesh codes. ROCS is a nodal code which can be used in either two or three dimensions. Both neutronic and thermal-hydraulic effects are accounted for, thereby allowing the simulation of physics tests, load following, soluble boron rundown, control movement, as well as end-of-cycle power coastdown.

MC

The MC code is used to calculate fine-mesh (pin-wise) power and burnup distributions. Fine mesh analysis is performed by the MC code through the application of the nodal imbedded method to individual assemblies using inter-assembly currents calculated by the coarse-mesh ROCS program. Capabilities also include fine-mesh fuel depletion and in-core instrument modeling. Data files written by MC contain fine-mesh fluence, burnup, and power information.

b. Cross Section Codes

These codes are used to calculate and verify cross sections for ROCS, HERMITE and MC.

DIT

The DIT (Discrete Integral Transport) code is the principal code for cross section generation. When used in conjunction with CESAW and MCXSEC, few-group cross sections generated by DIT can be directly input into either coarse-mesh or fine-mesh diffusion theory programs. This state-of-the-art code includes three major components. First, a spectrum calculation using an 85, or an optional 41-group ENDF/B-IV based multigroup library, is typically performed for a number of characteristic cell types: asymptotic fuel, fuel with poison rod neighbors and fuel with water hole neighbors. Coupling between cell types is accounted for through interface neutron currents. Following the spectrum calculation, a few-group assembly calculation is performed which explicitly accounts for the coupling between individual cells. Finally, a pin-by-pin depletion is performed. Where necessary, pins can be spatially subdivided for this depletion calculation.

The DIT code is also used for fuel assembly depletion analyses including calculation of detailed isotopics and reaction rates, and lattice physics parameters for all operating conditions and times in life.

CESAW

The CESAW code produces HARMONY-like tablesets for ROCS and HERMITE using properly formatted cross section data stored on permanent files. The ability of CESAW to generate special purpose tables (such as those used to represent poison rod cells or accounting for thermal feedback) eliminates hand preparation of poison rod tablesets.

MCXSEC

MCXSEC is a code used to prepare cross section input files for MC from DIT produced files. It works off of the same DIT as CESAW so that consistency between coarse and fine mesh cross sections is maintained.

c. DIT Data Library

The base DIT 85-group data library for PWR core analysis contains cross section tables derived from the ENDF/B-IV database. The cross sections are collapsed to 85 energy groups. This library was used for the conceptual analysis of the plutonium burner.

An ENDF/B-VI based library under development as planned upgrade to the DIT library will be available for detailed follow-on analyses. The DIT cross section library is prepared using the NJOY code.

The NJOY nuclear data processing code is a comprehensive code system for producing point use and multipgroup neutron and photon cross sections from ENDF/B-VI nuclear data. The ABB-CE version of the code is derived from NJOY89 developed by Los Alamos National Laboratory. ABB-CE modifications include addition of the RABBLE module to prepare resonance cross sections for DIT, addition of the LIBPRE module to prepare a library of infinities dilute cross sections and scattering matrices, resonance cross section tables, fission yields, depletion chains and neutron emission spectra for DIT. Plotting and file merge features have also been added.

d. Axial Shape Analysis and Space-Time Codes

The principal code in this category is HERMITE, as described below:

HERMITE

HERMITE is a space-time kinetics code used for analysis of design and off-design transients in large pressurized water reactors. The multi-dimensional, few groups, time-dependent neutron diffusion equation is solved by either the nodal expansion method (NEM) in 1-D, 2-D or 3-D or the space-time factorization method (FIESTA) in 1-D. Included are the feedback effects of fuel and coolant temperatures, coolant density and control rod motion. The heat conduction equation in the pellet, gap and clad is solved by a finite difference method. Continuity and energy conservation equations for the coolant are also solved. The momentum conservation equations are solved for a three dimensional open channel flow model. Nuclide concentrations are calculated using depletion equations. Xenon and iodine concentrations are calculated using either depletion or equilibrium equations. Fuel management capabilities are included. Quasi-steady state solutions to off-nominal conditions can also be obtained. HERMITE is also available in a one-dimensional only version.

1-D HERMITE provides the ability to perform one-dimensional space time loss of flow and space time scram worth calculations. The HERMITE code is also capable of a variety of static and space-time calculations in two (x-y) and three dimensions. HERMITE three-dimensional, open-channel calculation are used in steam line break analysis. HERMITE can also be used to perform 2-D space time asymmetric steam generator analysis.

e. Radiation Physics and Criticality Codes

Principal codes used for radiation physics analysis and for criticality analysis for fresh and spent fuel are as follows:

DOT 4.3

DOT 4.3 is a multi-group, discrete ordinates transport code. DOT determines the flux or fluence of particles throughout a one- or two-dimensional geometric system due to sources either generated as a result of particle interaction with the medium or incident upon the system from extraneous sources. The principle application is to deep-penetration transport of neutrons and photons. Criticality problems can be solved. In addition, the ABB-CE version generates fine-mesh albedo data for the MC code.

KENO IV

KENO IV is a transport theory code which calculates the reactivity of a system containing fissionable material using 3D Monte Carlo methods. KENO IV determines other neutron characteristics such as generation time, leakage, absorption, and flux.

ORIGEN II

The ORIGEN II code calculates the fission product and transmission sources and isotopes for irradiated fuel as a function of operating history and decay time.

f. Heat Transfer and Fluid Flow Codes

The codes in this group solve problems related to heat transfer, fluid flow, fuel performance and form the basis for calculations in the Transients and Setpoints Codes.

CETOP-D

The CETOP-D code calculates the thermal margin of a PWR for steady state operation. It differs from the TORC design model by its simplified geometric modeling of the core and faster calculation algorithm. The CETOP-D model of a reactor core must be benchmarked to a similar TORC model before it is used to perform thermal-hydraulic analyses of the core. Application of the CETOP-D code results in substantial reductions in execution time when compared to TORC.

TORC

The TORC code determines the thermal margin of a PWR core for steady state operation. The code solves the conservation equations for a three-dimensional representation of an open lattice core to determine the local coolant conditions at all points within the core. The code uses the local coolant conditions in conjunction with a suitable critical heat flux correlation to determine the minimum value of the departure from nucleate boiling ratio (DNBR) for the core.

HRISE

The HRISE code calculates the departure from nucleate boiling ratio (DNBR) from various critical heat flux correlations for rod bundles and heated tubes. The MacBeth, Bowring and Biasi built-in correlations have a wide range of validity. The code models a single closed channel with non-uniform axial heat flux. Input nodal flow factors enable the user to model cross flow effects.

g. Fuel Performance Analysis Codes

FATES3B

The FATES code calculates the radial and axial steady state temperature distribution through a single fuel rod using specified values of the rod linear heat rate and coolant flow rate. The effects of fission gas release, fuel swelling, densification and relocation, and clad creep are treated.

FATES4

Fuel mechanical performance related to PCI. Calculates stress and strain distributions throughout the fuel and cladding during power transients. Models fuel pellet and cladding elastic and inelastic interactions during the transient.

h. Thermal Hydraulic Design Codes

CEPOOL

Thermal-hydraulic analysis of fuel in spent fuel pool. Determines coolant temperatures within the spent fuel pool to verify design criteria. The code solves the steady state mass momentum and energy equations that describe the flow network in a spent fuel pool. The code calculates the steady state coolant temperatures and flow rates within each flow path to determine if boiling will occur, and the length of boiling region within each cell.

GUIDO

A thermal-hydraulic code used to calculate the flow rate required to preclude bulk boiling in a guide tube. The results provide radial temperature distributions in control rod and coolant annulus at various axial locations.

i. Plant Transient Analysis Codes

This group of codes is used to perform the non-LOCA transient analyses to verify acceptable performance and to determine or verify selected COLSS and CPCS database constants for those plants that use COLSS and CPCS for monitoring and protection. In addition to these codes, the non-LOCA transient analyses may also require the use of the HERMITE code, and the HRISE, TORC (used to set up CETOP models) and CETOP-D codes.

CESEC-III

The CESEC-III code is a highly flexible analytical tool for simulating symmetric and asymmetric plant responses to non-LOCA events. The code models safety-related control and plant protection systems, high and low pressure safety injection pumps and tanks and the effects of core temperature tilts. Superheating is allowed in the pressurizer model which permits complete inhomogeneity and does not require that the phases be in thermal equilibrium. The reactor vessel upper head model allows phase separation during depressurization but requires thermal equilibrium of the phases. Heat transfer between primary coolant and primary system metal components is modeled in detail. The CESEC-III code simulates four reactor coolant pumps with wide flexibility in pump on/off schemes. CESEC-III also provides for various critical flow correlations for the calculation of mass flow through valves and leaks. CESEC-III selects heat transfer correlations depending on local fluid conditions and the direction of heat flow. CESEC-III is used in licensing analyses and has been extensively benchmarked against plant startup and operational data. It is also used in support of operator training and emergency procedure guidelines. Automatic data transferral from CESEC-III to the TORC/CETOP and HRISE codes is also featured.

STRIKIN-II (Transient Analysis Version)

STRIKIN-II is used for hot channel heatup calculations in the safety analyses. The code is used to calculate the transient DNBR, coolant enthalpy, and fuel temperatures in the hot rod in the hot assembly.

The non-LOCA transient analysis version of the STRIKIN-II code was derived from the LOCA version and has been maintained separately. This version is used for the analysis of the CEA ejection accident and includes a point kinetics neutronics model as well as other enhancements.

The STRIKIN-II code solves the one-dimensional (axial) conservation equations and the equations of state for the fluid with provisions for local fluid expansion.

In a fuel rod, the STRIKIN-II code solves (radially) the one-dimensional cylindrical heat conduction equation for each axial region along the rod. The conduction model explicitly represents the gas gap region and dynamically calculates the gap conductance in each axial region. A volume averaged temperature is calculated for each radial node. The STRIKIN-II code uniquely determines the heat transfer regime at the clad/coolant boundary for the updated temperature distribution.

CENTS

CENTS is an interactive, faster than real time computer code for simulation of the Nuclear Steam Supply System and related systems. It calculates the behavior of a PWR for normal and abnormal conditions including accidents. It is a flexible tool for PWR analysis which gives the user complete control over the simulation through convenient input and output options.

CENTS is an adaptation of design computer codes to provide PWR simulation capabilities. It is based on detailed first-principles models for single and two-phase fluids. Use of nonequilibrium, nonhomogeneous models allows a full range of fluid conditions to be represented, including forced circulation, natural circulation, and coolant voiding. The code provides a comprehensive set of interactions between the analyst, the reactor control systems and the reactor. This allows simulation of multiple failures and the effects of correct and incorrect operator actions. Examples of simulation runs with CENTS are steady state, power change, pump trip, loss of load, loss of feedwater, steam line break, feedwater line break, steam generator tube rupture, anticipated transients without scram, rod ejection, loss of coolant accidents, anticipated operational transients, and malfunctions of components, control systems or portions of control systems.

j. LOCA Codes

This group of codes is used for large and small break loss of coolant accident (LOCA) analysis, post-LOCA long term cooling analysis and blowdown loads. Also included in this group are RELAP5.

All codes described below are recognized by the NRC as part of ABB-CE's LOCA licensing analysis capability.

STRIKIN-II (LOCA VERSION): Hot Rod Heatup for LOCA

The STRIKIN-II code (LOCA Version) is an NRC approved computer program that calculates the transient clad temperatures of the hot rod during blowdown, refill, and reflood. It solves the one-dimensional (axial) conservation of energy equation and the equations of state for the fluid with provisions for local fluid expansion. This code is used to perform a closed-channel heat transfer analysis of the hottest fuel rod. The fluid mass-flow rate and enthalpy are specified at the inlet end of the channel which is the entrance during the period of forward flow through the core. This method of analysis maximizes fuel rod heatup since no credit is taken for crossflow between coolant channels.

The primary outputs from the STRIKIN-II code are the peak cladding temperature (PCT) and maximum cladding oxidation. For small breaks STRIKIN-II is used to evaluate fuel rod temperatures during the initial period of the blowdown.

CEFLASH-4A/FII: Blowdown Thermal Hydraulics for Large Break LOCA

CEFLASH-4A/FII is an NRC-approved computer program that is used to calculate the thermal hydraulic response of the reactor coolant system during the blowdown phase of a large break loss of coolant accident (LOCA). The code is applicable to any PWR loop arrangement. It is used extensively for licensing ABB-CE designed two- and three-loop plants and has been documented for use for three- and four-loop W-type reactors. The Fully Implicit Iterative (FII) solution technique makes CEFLASH-4A/FII a fast running code that produces results whose precision is comparable to or better than other evaluation model codes.

The CEFLASH-4A/FII code is a multinode-multiflow path code that models the NSSS as a series of volume nodes connected by flow paths. The equations of conservation of mass and energy are solved for the nodes at each time step. The static pressure in each node is determined at each time step using an equation of state assuming the fluid within each node is in thermodynamic equilibrium. The flow paths connect the volume nodes at specified elevations. The conservation of momentum equation is solved for each flow path assuming that the fluid within each flow path is homogeneous and in thermodynamic equilibrium. The code represents the fluid properties associated with single and two-phase conditions (subcooled and saturated water, two-phase steam-water mixtures, and saturated and superheated steam).

CEFLASH-4AS: Blowdown Hydraulics for Small Break LOCA

CEFLASH-4AS is a version of CEFLASH-4A/FII, described herein, which has been extensively modified for application to analysis of blowdown hydraulics during small break LOCAs. It is an NRC approved code. The primary difference which distinguishes CEFLASH-4AS from CEFLASH-4A/FII is the heterogeneous description of the fluid within each node and the flow paths.

COMPERC-II/LB: Refill/Reflood Thermal Hydraulics

COMPERC-II/LB is an NRC-approved digital computer program that is used in the thermal hydraulic analysis of the refill/reflood period of a large break LOCA. It is applicable to any 2-, 3- or 4-loop PWR arrangement. The code models the NSSS with a detailed reactor vessel model with a steam flow resistance network that accounts for the RCS piping, steam generators and reactor coolant pumps. The FLECHT-based reflood heat transfer model is applicable to 14x14, 15x15, 16x16, and 17x17 fuel assemblies.

COMPERC-II/SB: Reflood Hydraulics for Small Break LOCA

COMPERC-II/SB, an NRC-approved computer code for small break LOCAs, is a modified version of the large break version that is used to evaluate the reflood hydraulics during a small break LOCA. COMPERC-II/SB provides the transient two-phase level and pressure, or the FLECHT heat transfer coefficients for the PARCH/EM code.

PARCH/REM: Steam Cooling Heat Transfer for Large Break LOCA

PARCH/REM is an NRC approved computer program that calculates steam cooling heat transfer coefficients for the hot fuel rod heatup calculation. HCROSS is used in conjunction with PARCH/REM to define flow diversion caused by local hot channel blockage as well as to determine subsequent flow recovery above the blockage.

PARCH/EM: Hot Rod Heatup Model for Small Break LOCA

A version of the NRC-approved PARCH/EM code is used for analysis of small break LOCA accidents. Its primary function for small breaks is to evaluate the fuel rod temperatures after the end of forced convection, that is, during pool boiling. It performs a closed-channel heat transfer analysis of the hottest fuel rod. The rate of boil-off of steam from the two-phase region and the local steam temperatures are calculated using hot rod properties.

CEFLASH-4B: Blowdown Hydraulics for Loads on Inner Vessel Internals

The CEFLASH-4B computer code predicts the transient reactor pressure vessel pressure, flow distribution and coolant properties during the subcooled and saturated portion of the blowdown period of a LOCA. The pressure distributions produced by CEFLASH-4B are used to calculate loads within the pressure vessel. The resulting loads are used in stress analyses of hardware inside the pressure vessel. The NRC has approved the code for analyzing blowdown transients.

BORON: Post-LOCA Core Boric Acid Concentration

BORON is an NRC-approved computer program that calculates the boric acid concentration in the reactor vessel and sump after a LOCA. It is used to determine if boric acid precipitation could prevent cooling of the fuel rods.

CELDA: Post-LOCA Core Cooling

CELDA is an NRC-approved computer program that is used to describe the blowdown and refill behavior of the primary system after a small break LOCA. It models heat generation in the core, steam generator heat transfer, wall heat transfer, phase separation, and critical flow.

CEPAC: Post-LOCA Steam Generator Secondary Temperature

CEPAC is an NRC-approved computer code that calculates system cooldown following a LOCA. It is used to compute the steam generator secondary temperature and feed water consumption after a LOCA until shutdown cooling conditions are reached.

NATFLOW: Post-LOCA Core Natural Circulation

NATFLOW is an NRC-approved code that calculates the natural circulation flow rate in the core and the primary system temperature after a LOCA. It is used to find the steady state conditions in the reactor coolant system while the steam generators act as a heat sink.

RELAP5/Mod3 Code:

RELAP5/MOD3 is the latest in a series of best estimate computer programs written by the Idaho National Engineering Laboratory (INEL) to produce best estimate transient simulations of a pressurized water reactor and associated systems. The code provides modeling capability for a wide range of transients including postulated accidents such as a small or large break LOCA as well as operational transients such as anticipated transient without scram (ATWS), loss-of-offsite power, loss-of feedwater, loss-of-flow, etcetera. In addition the code is applicable to both separate effects and integral experiments.

RELAP5/MOD3 provides thermal-hydraulic analysis capability for a fluid mixture with water, steam, one non-condensible fluid, and a non-volatile solute. The fluid and energy flow paths are approximated by one-dimensional stream tube and conduction models. A generic modeling approach is utilized to permit modeling as much of a system as is needed for the simulation. Supplementary primary system models are provided for the core neutronics, pumps, valves, steam generators, etcetera. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensors, and feedwater systems. The code also contains separator and jet pump models which have allowed it to be used to model boiling water reactor systems.

3. Licensing Approval

This licensing application of the codes presented in this section for the System 80+ standard plant design are described in detail in CESSAR-DC. Selected topical reports are identified below.

- "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983.
- "HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients," CENPD-188-A, March 1976.
- "Methodology for Core Designs Containing Erbium Burnable Absorbers", CENPD-382-P, October 1990 and CENPD-382-P, Supplement 1-P, February 1992.
- "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," Combustion Engineering, Inc., CENPD-161-P-A, April 1986.
- "CETOP-D Code Structure and Modeling Methods for SONGS 2 and 3," CEN-160-S-P, Rev. 1, September, 1981.
- "C-E Fuel Evaluation Model Topical Report," Combustion Engineering, Inc., CENPD-139-P, CENPD-139 Rev. 01, CENPD-139 Supplement 1, Rev. 01 (Non-Proprietary), July 1974.
- "Improvements to Fuel Evaluation Model," Combustion Engineering, Inc., CEN-161-P(A), August 1989; and CEN-161-P(B) Supplement 1-P, April 1986.
- "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107-P, April 1974.
- "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," Combustion Engineering, Inc., CENPD-135P, Supplement 2, December 1974, Supplement 4, August 1976.
- "Calculation Methods for the C-E Large Break LOCA Evaluation Model," CENPD-132, Supplement 1, December 1974.
- "C-E Method for Control Element Assembly Ejection Analysis," CENPD-190-A, January 1976.

C. PLUTONIUM SPIKING

1. Pu Spiking: Study Requirements

The DOE Requirements Document specifies the following objectives and assumptions used for the Plutonium Spiking Alternative:

- The goal is to transform, as quickly as possible, 100MT of weapons grade Pu to a material that could not be returned to a weapons-usable form unless several economic and engineering barriers were overcome. These barriers, at minimum, would include:
  - The necessity for remote handling (hot cell);
  - Large scale separation facilities;
  - Large capital investment to develop infrastructure.
- The material created should be compatible with planned requirements for qualification and placement of the transformed Pu in high level waste repository.
- The fuel assembly, or equivalent, gamma radiation levels are greater than 100 rem/hr at three feet after a two year cooldown period following Plutonium Spiking.

Further guidance from DOE indicates that the key objective for the Pu Spiking alternative is Speed, and that the controlling factors to implement this alternative include reactor size, number of units, fuel (heavy metal) throughput, and maturity of technology.

2. Plutonium Spiking: Fuel Cycle Analysis

a. Base Concept Description

The concept of Plutonium Spiking with the System 80+ Plutonium Burner was developed consistent with the requirements and guidance stated above. The base concept accomplishes Pu spiking by short-term irradiation of weapons grade plutonium in a single reactor unit. The controlling factors for this base concept are discussed below:

Reactor Size. The unit core size is large, consisting of 241 fuel assemblies, with a core thermal rating of 3800 MWth. The reactor accommodates all-plutonium-reactor (APR) operation based on the use of PuO_2-UO_2-Er_2O_3 MOX fuel. The APR core design characteristics for the Plutonium Spiking Alternative are described in Section III.A.4 and in Table III.A-5.

Number of Units. Based on the evaluation of controlling factors for speed, the practical speed of Pu spiking is determined to be more limited by fuel fabrication capacity than by the reactor capacity for a single System 80+ unit. Therefore a single reactor unit is used for the base concept.

Fuel Throughput. The fuel cycle for Pu Spiking is described in Table III.C-1. The fuel cycle consists of loading and irradiation of a feed core for a single irradiation cycle. The discharge core is offloaded and stored in the spent fuel pool of the reactor complex. The rate of fuel throughput is maximized by the large loading of weapons-grade plutonium in each feed core (i.e., 6.67 MT plutonium metal) and by the short irradiation cycle (39 EFPD irradiation per cycle, with an average cycle time, including refueling, of 3 months). The spent fuel pool for the single System 80+ Plutonium Burner unit is sized to accommodate 15 full cores (i.e., all the required cores for irradiation of 100 MT weapons-grade plutonium).

The fuel cycle schedule for accomplishing Pu spiking for 15 full cores (100 MT weapons-grade plutonium) is described in Table III.C-2. The corresponding total time of plant operations from start to completion of Pu spiking is 45 months, with an average capacity factor of 0.43. The project schedule for the Plutonium Spiking Alternative is based on the assumption of project initiation in October 1993, as specified in the DOE Requirement Document. Major project milestones associated with this schedule are as follow:

Milestone	Date	Month
Project Initiation	10/1993	0
First Concrete	10/1995	24
Initial Fuel Load	04/2000	78
Begin Pu Spiking Operations	10/2000	84
Complete Pu Spiking Operations	07/2004	129

Maturity of Technology. The System 80+ reactor systems technology is fully mature and proven in operating reactors, as described in Section III.A. More limited PWR experience exists for MOX fuel operations. A fuel demonstration program for the MOX fuel in this design application would be performed in parallel with plant construction, and fully completed prior to fuel load.

Other Considerations. The MOX fuel cycle used for the Plutonium Spiking Alternative is compatible with the fuel cycle used for the Spent Fuel Alternative (see Section III.D). Therefore, the spiked fuel stored in the spent fuel pool would be reusable in the System 80+ reactor to generate power and meet the plutonium isotope transformation requirements of the Spent Fuel Alternative. The MOX fuel cycle used for the Plutonium Spiking Alternative is also compatible with conversion of the plant operation to provide tritium production, as described in Section III.G.

b. Fuel Cycle Length

The fuel cycle length of 39 effective full power days (EFPD) for the Plutonium Spiking Alternative is based on a short average refueling interval of three months. This operating cycle is judged to be the shortest practical for a large operating nuclear plant. Assuming an average operating capacity factor of 0.80 during the 39 EFPD power operation, the average planned outage time for each cycle is approximately 42 days. The overall capacity factor, including planned outages, is 0.43. In practice, the outages could be distributed as three refueling-only outages and one refueling and maintenance outages per year, consistent with the 17 day breaker-to-breaker refueling capability of the System 80+ design. On this basis it would practical to plan refueling-only outage lengths of 30 days, and allow an refueling and maintenance outage of approximately 80 days during the Pu spiking operations.

The 39 EFPD cycle length also assures that the dose rate of each discharge fuel assembly exceeds the requirement stated in Section III.C.1 above. A gamma dose rate scoping calculation was performed using the ORIGEN2 code for irradiation of the reference MOX fuel assembly to 30 EFPD, followed by two years decay. On the basis of the gamma source calculation, the dose in air at a distance of three feet from the side of the fuel assembly is estimated to exceed 10^5 rem/hr, or three orders of magnitude greater than the minimum 10^2 rem/hr specified by the DOE Requirements Document.

c. Fuel Cycle Data

The cycle-dependent physics parameters for the Plutonium Spiking Alternative are identical to those for beginning-of-life (BOL) conditions of the Spent Fuel Alternative, as described in Section III.D.

Due to the short 39 EFPD cycle length of the Plutonium Spiking Alternative, the plutonium transformation over cycle is very small, as shown by Table III.C-3 which gives the kilogram mass of actinide isotopes in core at beginning and end of the irradiation cycle.

Additional technical information and data for the Plutonium Spiking Alternative is provided in Section III.K.

Table III.C-1

System 80+ Plutonium Burner Fuel Cycle Characteristics
Plutonium Spiking Alternative

Pu Spiking Alternative Fuel Cycle	
Core Power Level	3800 MW(th)
Average Capacity Factor	0.43
Cycle Length	3-months (39 EFPD)
Average Discharge Burnup	1500 MWD/MTHM
Feed Fuel Type	MOX
Feed Fuel Batch	
Number of Assemblies	241
0-Shim	81
12-Shim	160
Active Fuel Length	3.81 m (150 in)
Number of Fuel Rods	54956
Fuel Composition	UO_2-PuO_2-Er_2O_3
Average Erbium in MOX	1.6 wt% Er_2O_3 in MOX pellets
Heavy Metal Feed	98.75 MTHM
Uranium (tails) Feed	92.08 MTU
Plutonium Total Feed	6.67 MTPu
Total Pu in HM	6.75 wt%
Uranium (tails) Feed Isotopes	99.8% U-238, 0.2% U-235
Plutonium Feed Isotopes	93.5% Pu-239, 6.5% Pu-240
Fissile Pu Feed	6.24 MTPu
Fissile Pu in HM	6.32 wt%
BPRs in Feed Fuel Batch	
Number of Burnable Poison Rods	1920
Active Poison Length	3.45 m (136 in)
Average B-10 Loading in Poison	0.0102 g/cm (0.026 g/in)

Table III.C-2

Fuel Cycle Operating Schedule
Plutonium Spiking Alternative (S-O)

Number of Reactor Units: 1
Core Power Rating: 3800 MWt

Cycle Length:	Months	EFPD	Cap Factor
Cyc 1	3	39	.43

First Core Startup Test Period: 6 months
Number of Feed Cores for Mission: 15

Operating Cycles		Scheduled Start of Cycle (Yr/Mo)
Feed Core	Cycle	Unit 1
1	1	2000/04
2	1	2001/01
3	1	2001/04
4	1	2001/07
...		
15	1	2004/04

Table III.C-3

Fuel Cycle Actinide Inventory (Metric tonnes)

Plutonium Spiking Alternative

MONTHS EFPD	0.0 0.0	3.0 39.0
U235	1.8414E-01	1.7633E-01
U236	0.0000E+00	6.4259E-04
U238	9.1879E+01	8.9321E+01
NP237	0.0000E+00	3.3283E-04
PU238	0.0000E+00	5.0580E-06
PU239	6.2367E+00	5.8988E+00
PU240	4.3356E-01	4.9023E-01
PU241	0.0000E+00	1.0242E-02
PU242	0.0000E+00	3.2473E-05
AM241	0.0000E+00	1.9742E-05
AM243	0.0000E+00	1.1688E-07
CM242	0.0000E+00	1.13503-07
CM244	0.0000E+00	5.0612E-10
TOTAL HM	9.8733E+01	9.5897E+01
TOTAL U	9.2063E+01	8.9498E+01
TOTAL PU	6.6703E+00	6.3993E+00
TOTAL AM+CM	0.0000E+00	1.9973E-05

PU ISOTOPE FRACTION

PU-238/PU	0.0000	0.0000
PU-239/PU	0.935	0.922
PU-240/PU	0.065	0.077
PU-241/PU	0.0000	0.002
PU-242/PU	0.0000	0.0000

DESTRUCTION FRACTION

PU-239	0.000	0.054
TOTAL PU	0.000	0.041

D. SPENT FUEL

1. Spent Fuel Alternative: Study Requirements

The DOE Requirements Document specifies the following objectives and assumptions used for the Spent Fuel Alternative:

- The alternative represents a plant that effectively burns Pu while producing electrical power with the added capability to produce tritium. The alternative should be optimized to achieve a cost effective transformation of weapons-grade Pu to a form similar to the spent fuel normally produced by the reactor concept.

- The goal is to transform, as economically as possible, 100MT of weapons grade Pu to a material that could not be returned to a weapons-usable form unless several economic and engineering barriers were overcome. These barriers, at minimum, would include:

  - The necessity for remote handling (hot cell);
  - Large scale separation facilities;
  - Large capital investment to develop infrastructure.

- The material created should be compatible with planned requirements for qualification and placement of the transformed Pu in high level waste repository.

- The capacity factor for the Spent Fuel Alternative is assumed to be 75% (annual average after 18 months initial startup and operation).

- The start date for proceeding with the engineering of the complex is assumed to be October 1993, with the objective of completing the disposal of 100 MT of Pu within 25 years of the start date.

Further guidance from DOE indicates that the key objective for the Spent Fuel alternative is Economy, and that the controlling factors to implement this alternative include basic attributes such as electrical output, and economic trade-off factors such as economy of scale versus experience factor, cost versus schedule, etc.

2. Spent Fuel Alternative: Fuel Cycle Analysis

a. Base Concept Description

The concept of the Spent Fuel Alternative with the System 80+ Plutonium Burner was developed consistent with the requirements and guidance stated above. The fuel cycle developed for this alternative supports the favorable economic attributes of a 1300 MWe System 80+ unit operating with an all-plutonium-reactor (APR) fuel cycle. The major favorable attributes include 1) the economy of scale of the large APR reactor unit which has the potential for major reduction of the capital and O&M costs required for disposition of 100

MT of weapons-grade plutonium, and 2) savings in fuel costs due to the high Pu loading which has the potential to minimize fabrication costs and allows use of uranium tails with no uranium enrichment. The characteristics of base concept are discussed below:

Reactor Size. The unit core size is large, consisting of 241 fuel assemblies, with a core thermal rating of 3800 MWth. The reactor accommodates all-plutonium-reactor (APR) operation based on the use of PuO_2-UO_2-Er_2O_3 MOX fuel. The APR core design characteristics for the Spent Fuel Alternative are described in Section III.A.4 and in Table III.A-5.

Number of Units. The base concept presented for the Spent Fuel Alternative is based on a four-unit reactor complex which satisfies the 25 year schedule constraint specified as a study objective. Additional concepts are presented based on the capability for completion of Spent Fuel plutonium disposition mission by operation of a single-unit or two-unit complex over a longer schedule. The basic fuel cycle is the same in all cases.

Fuel Throughput. The fuel cycle for the Spent Fuel Alternative is described in Table III.D-1. The fuel cycle consists of loading and irradiation of a feed core for a total of four annual irradiation cycles (a total of 1096 effective full power days). All irradiated fuel offloaded from the core (either discharge fuel or intermediate to irradiation cycles is stored in the spent fuel pool of the reactor complex. The Each MOX feed core contains 6.67 MT plutonium metal and approximately 92 MT uranium metal in the form of U tails (0.2wt% assay of U^{235}). Fifteen (15) full feed cores are therefore sufficient to accommodate 100 MT of weapons-grade plutonium. After completion of fours years reactor power operations, the discharge fuel has transformed plutonium isotope characteristics similar to those of spent fuel normally produced by operation of UO_2 fuel cycles for the reference reactor design. The discharge fuel is therefore of suitable form for disposition at a high level waste depository in a similar manner as commercial spent fuel assemblies.

The fuel cycle schedule for accomplishing the disposition for 15 full cores (100 MT weapons-grade plutonium) in a period of 25 years is described in Table III.D-2a. The corresponding total time from start to completion of reactor power operations for the Spent Fuel Alternative is 18 years (216 months), with an average capacity factor of 0.75. The construction and startup schedules for the reactor four units are separated by one year. The project schedule for the Spent Fuel Alternative is based on the assumption of project initiation in October 1993, as specified in the DOE Requirement Document, with completion of all power operations for Pu Spent Fuel disposition by October 2018. Major project milestones associated with this schedule are as follow:

Milestone (Four-unit Complex)	Date	Month
Project Initiation	10/1993	0
First Concrete	10/1995	24
Initial Fuel Load (Lead Unit)	04/2000	78

Begin Power Operations	10/2000	84
Complete Power Operations	10/2018	300

Optimum Economy. The optimum economy for the disposition of 100 MT weapons-grade plutonium as Spent Fuel is expected to result by minimizing the required number of reactor units, which has the effect of reducing capital and O&M costs. As described previously, the large plant System 80+ design is based on mature and proven technology. The design capability for APR operation permits the accommodation of 100 MT of Pu in 15 full cores. The fifteen cores may be operated with greater economy in a reactor complex consisting of a either single System 80+ unit or two System 80+ units. The single-unit concept is expected to represent the optimum Spent Fuel concept measured by economy, and is based on the System 80+ plant design life of 60 years. Tables III.D.2b and III.D.2c summarize the Spent Fuel operation schedule for the single-unit and two-unit reactor concept, respectively. The corresponding project schedule milestones are as follow:

Milestone (Single-unit Complex)	Date	Month
Project Initiation	10/1993	0
First Concrete	10/1995	24
Initial Fuel Load	04/2000	78
Begin Power Operations	10/2000	84
Complete Power Operations	10/2060	804

Milestone (Two-unit Complex)	Date	Month
Project Initiation	10/1993	0
First Concrete	10/1995	24
Initial Fuel Load (Lead Unit)	04/2000	78
Begin Power Operations	10/2000	84
Complete Power Operations	10/2031	456

The required time to perform all power operations for Pu Spent Fuel is 60 years for the single-unit concept and 31 years for the two-unit concept.

Other Considerations. The MOX fuel cycle used for the Spent Fuel Alternative is compatible with the fuel cycle used for the Pu Spiking Alternative (see Section III.C). Operating strategies are possible which would would achieve both Pu Spiking and Spent Fuel. A practical strategy would extend the cycle length for Pu Spiking to one-year, so the the first irradiation cycle for Spent Fuel would accomplish Pu Spiking. In this manner the full 100 MT of weapons-grade plutonium could be spiked prior to proceeding with the intermediate irradiation cycles for the Spent Fuel Alternative.

The MOX fuel cycle used for the Spent Fuel Alternative is compatible with tritium production, as described in Section III.G.

b. Fuel Cycle Data

The fuel cycle actinide inventory for the Spent Fuel Alternative is summarized in Table III.D-3. This data applies for the MOX fuel cycle operation for the base four-unit concept, and for the single-unit and two-unit concepts. The Pu isotope fractions show the transformation of plutonium during the fuel cycle. At discharge (1096 EFPD) the relative fractions of Pu^{239} and Pu^{240} are approximately 63% and 23%, respectively. This compares with relative fractions of approximately 53% and 23% for the discharge fuel of a reference UO_2 fuel cycle as described in Section III.A.

The plutonium destruction fraction data in Table III.D-3 also indicates that approximately 51% of the initial Pu^{239} inventory is destroyed at discharge for the Spent Fuel Alternative. The total fraction of Pu destroyed at discharge is approximately 27%, however, due to the buildup of Pu^{240}, Pu^{241}, and Pu^{242} with burnup. The change in Pu^{239} and Pu^{240} with burnup as a fraction of the initial Pu inventory is illustrated by Figure III.D-1.

Cycle-dependent physics parameters including critical boron concentrations, control rod worths, reactivity coefficients, and other safety-related parameters for the Spent Fuel Alternative are provided in Section III.F.

Additional technical information and data for the Spent Fuel Alternative is provided in Section III.K.

Table III.D-1

System 80+ Plutonium Burner Fuel Cycle Characteristics
Spent Fuel Alternative

Spent Fuel Alternative Fuel Cycle	
Core Power Level	3800 MW(th)
Average Capacity Factor	0.75
Irradiation Cycle Length	12-months (274 EFPD)
Number of Cycles	4
Average Discharge Burnup	42.2 GWT/MTHM (1096 EFPD)
Feed Fuel Type	MOX
Feed Fuel Batch	
Number of Assemblies	241
0-Shim	81
12-Shim	160
Active Fuel Length	3.81 m (150 in)
Number of Fuel Rods	54956
Fuel Composition	UO_2-PuO_2-Er_2O_3
Average Erbium in MOX	1.6 wt% Er_2O_3 in MOX pellets
Heavy Metal Feed	98.75 MTHM
Uranium (tails) Feed	92.08 MTU
Plutonium Total Feed	6.67 MTPu
Total Pu in HM	6.75 wt%
Uranium (tails) Feed Isotopes	99.8% U-238, 0.2% U-235
Plutonium Feed Isotopes	93.5% Pu-239, 6.5% Pu-240
Fissile Pu Feed	6.24 MTPu
Fissile Pu in HM	6.32 wt%
BPRs in Feed Fuel Batch	
Number of Burnable Poison Rods	1920
Active Poison Length	3.45 m (136 in)
Average B-10 Loading in Poison	0.0102 g/cm (0.026 g/in)

Table III.D-2a

Fuel Cycle Operating Schedule
Spent Fuel Alternative (SF-O)

Number of Reactor Units: 4
Core Power Rating: 3800 MWt

Cycle Length:	Months	EFPD	Cap Factor
Cyc 1	12	274	.75
Cyc 2	12	274	.75
Cyc 3	12	274	.75
Cyc 4	12	274	.75

First Core Startup Test Period: 6 months
Number of Feed Cores for Mission: 15

Operating Cycles		Scheduled Start of Cycle (Yr/Mo)			
Feed Core	Cycle	Unit 1	Unit 2	Unit 3	Unit 4
1	1	2000/04			
	2	2001/10			
	3	2002/10			
	4	2003/10			
2	1		2001/10		
	2		2002/10		
	3		2003/10		
	4		2004/10		
3	1			2002/10	
	2			2003/10	
	3			2004/10	
	4			2005/10	
4	1				2003/10
	2				2004/10
	3				2005/10
	4				2006/10
...					
15	1			2014/10	
	2			2015/10	
	3			2016/10	
	4			2017/10	

Table III.D-2b

Fuel Cycle Operating Schedule
Spent Fuel Alternative (SF-1)

Number of Reactor Units: 1
Core Power Rating: 3800 MWt

Cycle Length:	Months	EFPD	Cap Factor
Cyc 1	12	292	.80
Cyc 2	12	292	.80
Cyc 3	12	292	.80
Cyc 4	12	292	.80

First Core Startup Test Period: 6 months
Number of Feed Cores for Mission: 15

Operating Cycles		Scheduled Start of Cycle (Yr/Mo)			
Feed Core	Cycle	Unit 1	Unit 2	Unit 3	Unit 4
1	1	2000/04			
	2	2015/10			
	3	2030/10			
	4	2045/10			
2	1	2001/10			
	2	2016/10			
	3	2031/10			
	4	2046/10			
3	1	2002/10			
	2	2017/10			
	3	2032/10			
	4	2047/10			
4	1	2003/10			
	2	2018/10			
	3	2033/10			
	4	2048/10			
...					
15	1	2014/10			
	2	2029/10			
	3	2044/10			
	4	2059/10			

Table III.D-2c

Fuel Cycle Operating Schedule
Spent Fuel Alternative (SF-2)

Number of Reactor Units: 2
Core Power Rating: 3800 MWt

Cycle Length:	Months	EFPD	Cap Factor
Cyc 1	12	292	.80
Cyc 2	12	292	.80
Cyc 3	12	292	.80
Cyc 4	12	292	.80

First Core Startup Test Period: 6 months
Number of Feed Cores for Mission: 15

Operating Cycles		Scheduled Start of Cycle (Yr/Mo)			
Feed Core	Cycle	Unit 1	Unit 2	Unit 3	Unit 4
1	1	2000/04			
	2	2001/10			
	3	2015/10			
	4	2016/10			
2	1		2001/10		
	2		2002/10		
	3		2016/10		
	4		2017/10		
3	1	2002/10			
	2	2003/10			
	3	2017/10			
	4	2018/10			
4	1		2003/10		
	2		2004/10		
	3		2018/10		
	4		2019/10		
...					
14	1		2013/10		
	2		2014/10		
	3		2028/10		
	4		2029/10		
15	1	2014/10			
	2	2015/10			
	3	2029/10			
	4	2030/10			

Table III.D-3

Fuel Cycle Actinide Inventory (Metric tonnes)

Spent Fuel Alternative

MONTHS EFPD	0.0 0.0	12.0 274.0	24.0 494.0	36.0 822.0	48.0 1096.0
U235	1.8414E-01	1.6198E-01	1.4130E-01	1.2165E-01	1.0395E-01
U236	0.0000E+00	5.4947E-03	1.0256E-02	1.4398E-02	1.7788E-02
U238	9.1879E+01	9.1293E+01	9.0701E+01	9.0090E+01	8.9488E+01
NP237	0.0000E+00	1.3388E-04	4.8512E-04	1.0275E-03	1.6726E-03
PU238	0.0000E+00	4.5221E-06	3.6721E-05	1.2340E-04	2.7469E-04
PU239	6.2367E+00	5.3194E+00	4.4842E+00	3.7181E+00	3.0635E+00
PU240	4.3356E-01	6.9670E-01	8.8812E-01	1.0229E+00	1.1021E+00
PU241	0.0000E+00	1.9000E-01	3.6334E-01	5.0725E-01	6.1072E-01
AM241	0.0000E+00	5.5972E-03	2.1992E-02	4.8553E-02	8.1952E-02
AM243	0.0000E+00	3.0387E-03	1.0589E-02	2.0352E-02	2.9674E-02
CM242	0.0000E+00	3.7457E-04	2.8795E-03	9.2802E-03	1.9920E-02
CM244	0.0000E+00	2.6212E-04	1.9930E-03	6.3764E-03	1.3559E-02
TOTAL HM	9.8733E+01	2.2039E-05	3.2674E-04	1.5908E-03	4.5152E-03
TOTAL U	9.2063E+01	9.7676E+01	9.6627E+01	9.5562E+01	9.4538E+01
TOTAL PU	6.6703E+00	6.2117E+01	5.7577E+00	5.2969E+00	4.8585E+00
TOTAL AM+CM	0.0000E+00	3.6974E-02	1.5788E-02	3.7599E-02	6.7668E-02

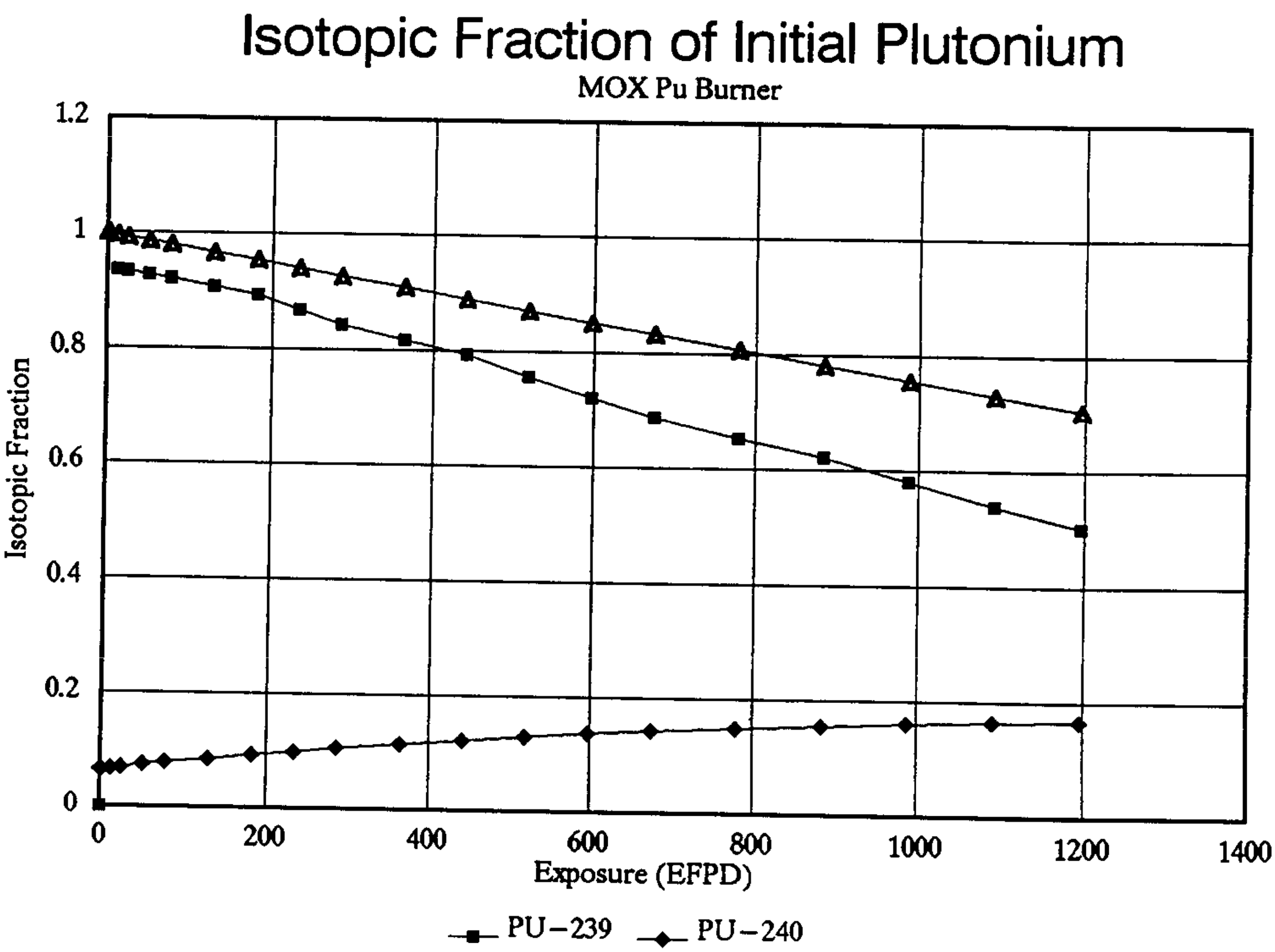
PU ISOTOPE FRACTION

PU238/PU	0.000	0.000	0.000	0.000	0.000
PU239/PU	0.935	0.856	0.779	0.702	0.631
PU240/PU	0.065	0.112	0.154	0.193	0.227
PU241/PU	0.000	0.031	0.063	0.096	0.126
PU242/PU	0.000	0.001	0.004	0.009	0.017

DESTRUCTION FRACTION

PU239/PU	0.000	0.000	0.000	0.000	0.000
TOTAL PU	0.000	0.069	0.137	0.206	0.272

FIGURE III.D-1



E. PLUTONIUM DESTRUCTION

1. Pu Destruction Alternative: Study Requirements

The DOE Requirements Document specifies the following objectives and assumptions used for the Plutonium Destruction Alternative:

- The alternative represents a design that is optimized to enhance the extent of the destruction of the 100 MT of weapons-grade plutonium.
- The goal is to transmute or fission Pu to other elements to the maximum extent possible, such that the end product contains little or no plutonium.
- The material created should be compatible with planned requirements for qualification and placement in high level waste repository.
- The capacity factor for the Pu Destruction Alternative is assumed to be 75% (annual average after 18 months initial startup and operation).
- The start date for proceeding with the engineering of the complex is assumed to be October 1993, with the objective of completing the disposition of 100 MT of Pu within 25 years of the start date.

Further guidance from DOE indicates that the key objective for the Pu Destruction alternative is Extent, and that the controlling factors to implement this alternative include the definition of Pu destruction, the fuel cycle design and risks, amount of development required, and burn cycle length.

2. Pu Destruction Alternative: Fuel Cycle Analysis

a. Base Concept Description

The concept of the Pu Destruction Alternative with the System 80+ Plutonium Burner was developed consistent with the requirements and guidance stated above. The fuel cycle concept developed for this alternative is based on APR operation with a non-fertile plutonium-oxide fuel form, consistent with the objective that the end product plutonium be reduced to the maximum extent possible for the PWR reactor. The capability of the System 80+ design to accommodate APR operation is a major advantage for this application. The schedule and cost uncertainty in successful development and deployment of a non-fertile plutonium fuel type is the greatest disadvantage. The characteristics of base concept are discussed below:

Reactor Size. The unit core size is large, consisting of 241 fuel assemblies, with a core thermal rating of 3800 MWth. The reactor accommodates all-plutonium-reactor (APR) operation based on the use of PuO_2-Al_2O_3-Er_2O_3 non-fertile fuel. The APR core design characteristics for the Pu Destruction Alternative are described in Section III.A.4 and in Table III.A-6.

Number of Units. The base concept presented for the Pu Destruction Alternative is based on a four-unit reactor complex which satisfies the 25 year schedule constraint specified as a study objective. Additional concepts are presented based on operation of a single-unit or two-unit complex over a longer schedule. The basic fuel cycle is the same in all cases.

Fuel Throughput. The fuel cycle for the Pu Destruction Alternative is described in Table III.E-1. The fuel cycle consists of loading and irradiation of a feed core for a total of four annual irradiation cycles (a total of 1096 effective full power days). All irradiated fuel offloaded from the core (either discharge fuel or intermediate to irradiation cycles is stored in the spent fuel pool of the reactor complex. The Each non-fertile feed core for this concept contains 6.67 MT plutonium metal in oxide form. A relatively high loading of erbium (i.e., the equivalent of 4 wt% Er_2O_3 for the MOX design) provides a negative Doppler characteristic in the absence of U^{238} and acts as a burnable poison to assist in the control excess reactivity. Fifteen (15) full feed cores are therefore sufficient to accommodate 100 MT of weapons-grade plutonium. After completion of fours years reactor power operations, the discharge fuel has fissioned or transmuted a major portion (approximately 83%) of the fissile Pu^{239} in the weapons-grade feed fuel. Further burnup of the fuel is deemed impractical on the basis that calculated safety-related characteristics become unstable beyond 48 months of fuel burnup (e.g, the calculated MTC trends rapidly in the positive direction) due to diminishing fuel/water ratio. Because of the relatively high percentage destruction of Pu^{239} and the similarity of the relative isotope fractions of the remaining Pu to that of commercial spent PWR fuel, it is considered more practical to dispose of the spent fuel at a high level waste repository than to pursue reprocessing and further destruction.

The fuel cycle schedule for accomplishing the disposition for 15 full cores (100 MT weapons-grade plutonium) in a period of 25 years is described in Table III.E-2. The corresponding total time from start to completion of reactor power operations for the Pu Destruction Alternative is 18 years (216 months), with an average capacity factor of 0.75. The construction and startup schedules for the reactor four units are separated by one year. The project schedule for the Pu Destruction Alternative is based on the assumption of project initiation in October 1993, as specified in the DOE Requirement Document, with completion of all power operations for Pu Destruction disposition by October 2018. Major project milestones associated with this schedule are as follow:

Milestone (Four-unit Complex)	Date	Month
Project Initiation	10/1993	0
First Concrete	10/1995	24
Initial Fuel Load (Lead Unit)	04/2000	78
Begin Power Operations	10/2000	84
Complete Power Operations	10/2018	300

Risk Factors. The non-fertile plutonium fuel introduces a number of risk factors that may not be readily resolved in a concept study. The first risk factor is related to lack of proveness of the non-fertile fuel concept for PWR applications. Although the constituent materials in the PuO_2-Al_2O_3-Er_2O_3 fuel concept are individually proven in PWR applications, fuel irradiation and burnup-dependent behavior is not available. Deployment of a non-fertile fuel design concept would require a substantially greater amount of development and in-reactor testing than MOX for the PWR application. A second risk factor is related to less favorable safety-related physics parameters, and trends of these parameters with burnup which diverge relative to the charactertistics of UO_2 or MOX fuel cycles (see Section III.F). This introduces uncertainty in the successful development and licensing of a non-fertile fuel cycle concept. The uncertainties related to development, testing and licensing are considered to place the Pu Destruction Alternative at a disadvantage relative to the Spent Fuel Alternative based on MOX fuel.

b. Fuel Cycle Data

The fuel cycle actinide inventory for the Pu Destruction Alternative is summarized in Table III.E-3. This data applies for the non-fertile fuel cycle operation for the base four-unit concept, and for the single-unit and two-unit concepts. The Pu isotope fractions show the transformation of plutonium during the fuel cycle. At discharge (1096 EFPD) the relative fractions of Pu^{239} and Pu^{240} are approximately 41% and 32%, respectively. This compares with relative fractions of approximately 53% and 23% for the discharge fuel of a reference UO_2 fuel cycle as described in Section III.A.

The plutonium destruction fraction data in Table III.D-3 also indicates that approximately 83% of the initial Pu^{239} inventory is destroyed at discharge for the Pu Destruction Alternative. The total fraction of Pu destroyed at discharge is approximately 61%, however, due to the buildup of Pu^{240}, Pu^{241}, and Pu^{242} with burnup. The change in Pu^{239} and Pu^{240} with burnup as a fraction of the initial Pu inventory is illustrated by Figure III.E-1.

Cycle-dependent physics parameters including critical boron concentrations, control rod worths, reactivity coefficients, and other safety-related parameters for the Pu Destruction Alternative are provided in Section III.F.

Additional technical information and data for the Pu Destruction Alternative is provided in Section III.K.

Table III.E-1

System 80+ Plutonium Burner Fuel Cycle Characteristics
Pu Destruction Alternative

Pu Destruction Alternative Fuel Cycle	
Core Power Level	3800 MW(th)
Average Capacity Factor	0.75
Irradiation Cycle Length	12-months (274 EFPD)
Number of Cycles	4
Average Discharge Burnup	1096 EFPD
Feed Fuel Type	Non-fertile Pu Oxide
Feed Fuel Batch	
Number of Assemblies	241
0-Shim	81
12-Shim	160
Active Fuel Length	3.81 m (150 in)
Number of Fuel Rods	54956
Fuel Composition	Al_2O_3-PuO_2-Er_2O_3
Plutonium Total Feed	6.67 MTPu
Plutonium Feed Isotopes	93.5% Pu-239, 6.5% Pu-240
Fissile Pu Feed	6.24 MTPu
Erbium Total loading (approx.)	4.6 MT Er_2O_3
BPRs in Feed Fuel Batch	
Number of Burnable Poison Rods	1920
Active Poison Length	3.45 m (136 in)

Table III.E-2

Fuel Cycle Operating Schedule
Pu Destruction Alternative

Number of Reactor Units: 4
Core Power Rating: 3800 MWt

Cycle Length:	Months	EFPD	Cap Factor
Cyc 1	12	274	.75
Cyc 2	12	274	.75
Cyc 3	12	274	.75
Cyc 4	12	274	.75

First Core Startup Test Period: 6 months
Number of Feed Cores for Mission: 15

Operating Cycles		Scheduled Start of Cycle (Yr/Mo)			
Feed Core	Cycle	Unit 1	Unit 2	Unit 3	Unit 4
1	1	2000/04			
	2	2001/10			
	3	2002/10			
	4	2003/10			
2	1		2001/10		
	2		2002/10		
	3		2003/10		
	4		2004/10		
3	1			2002/10	
	2			2003/10	
	3			2004/10	
	4			2005/10	
4	1				2003/10
	2				2004/10
	3				2005/10
	4				2006/10
...					
15	1			2014/10	
	2			2015/10	
	3			2016/10	
	4			2017/10	

Table III.E-3

Fuel Cycle Actinide Inventory (Metric tonnes)

Plutonium Destruction Alternative

MONTHS EFPD	0.0 0.0	12.0 274.0	24.0 494.0	36.0 822.0	48.0 1096.0
U235	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
U236	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
U238	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
NP237	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
PU238	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
PU239	6.2369E+00	4.7197E+00	3.3308E+00	2.0743E+00	1.0603E+00
PU240	4.3357E-01	6.8860E-01	8.4159E-01	8.9655E-01	8.3968E-01
PU241	0.0000E+00	2.2314E-01	4.1325E-01	5.3665E-01	5.6161E-01
PU242	0.0000E+00	7.5777E-03	3.1340E-02	7.3906E-02	1.3542E-01
AM241	0.0000E+00	3.5292E-03	1.1779E-02	2.0593E-02	2.5187E-02
AM243	0.0000E+00	5.8035E-04	4.6179E-03	1.5516E-02	3.5166E-02
CM242	0.0000E+00	3.4076E-04	2.6743E-03	8.7850E-03	1.9119E-02
CM244	0.0000E+00	3.8910E-05	6.2100E-04	3.1864E-03	9.7576E-03
TOTAL HM	6.6705E+00	5.6435E+00	4.6367E+00	3.6295E+00	2.6862E+00
TOTAL U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL PU	6.6705E+00	5.6390E+00	4.6170E+00	3.5814E+00	2.5970E+00
TOTAL AM+CM	0.0000E+00	4.4892E-03	1.9692E-02	4.8080E-02	8.9230E-02

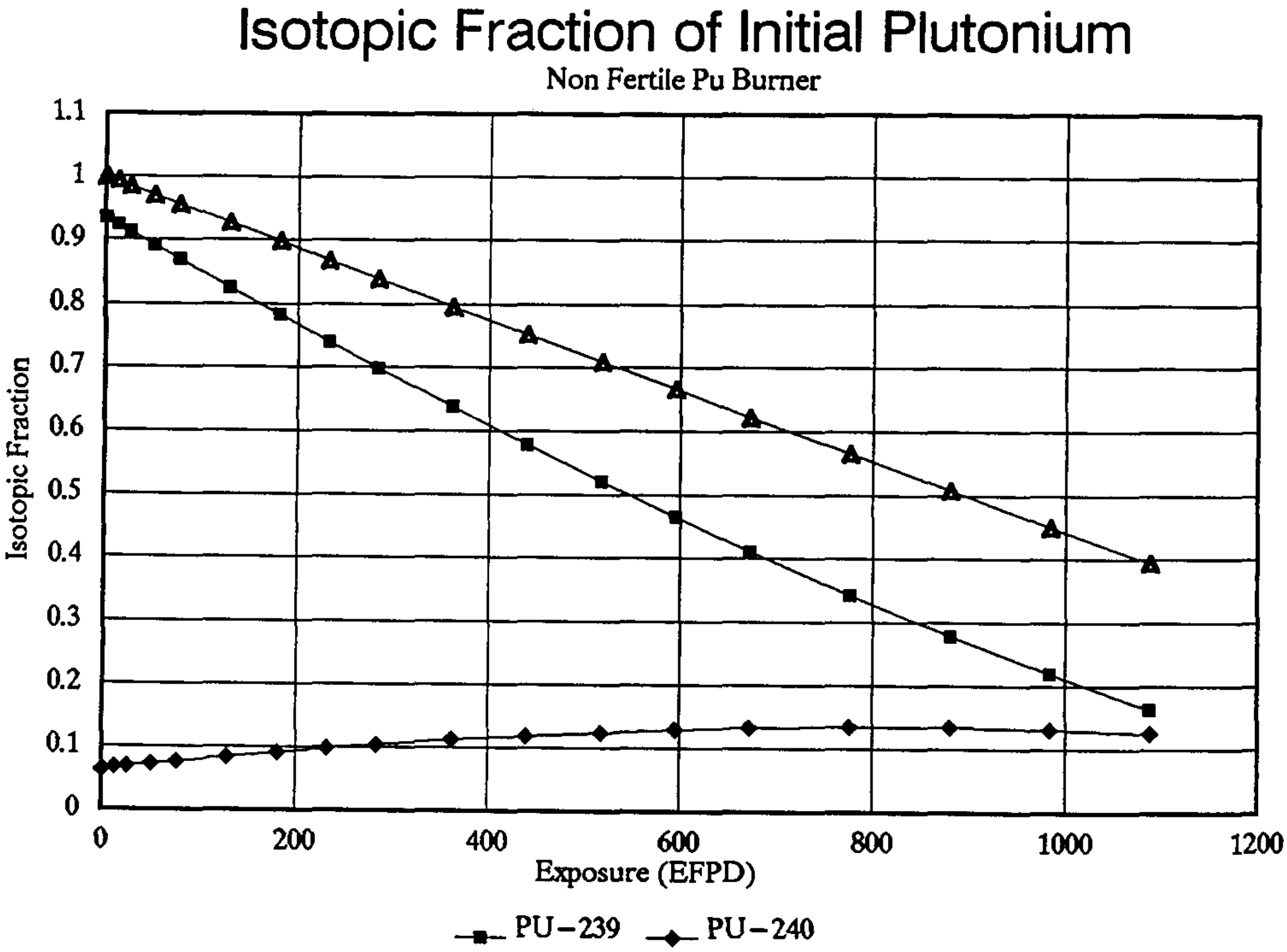
PU ISOTOPE FRACTION

PU238/PU	0.000	0.000	0.000	0.000	0.000
PU239/PU	0.935	0.837	0.721	0.579	0.408
PU240/PU	0.065	0.122	0.182	0.250	0.323
PU241/PU	0.000	0.040	0.090	0.150	0.216
PU242/PU	0.000	0.001	0.007	0.021	0.052

DESTRUCTION FRACTION

PU239/PU	0.000	0.243	0.466	0.667	0.830
TOTAL PU	0.000	0.155	0.308	0.463	0.611

FIGURE III.E-1



F. PLUTONIUM BURNER PHYSICS SAFETY CHARACTERISTICS

This section provides a summary of safety-related physics parameters for the System 80+ Plutonium Burner based on the evaluations of fuel cycle concepts for the Spent Fuel Alternative and the Pu Destruction Alternative utilizing weapons-grade plutonium fuel. The fuel cycle characteristics for these alternatives are described in Sections III.D and III.E, respectively. Safety related parameters are also compared with those for commercial fuel cycles based on UO_2, self-generated recycle (SGR), and all-plutonium-reactor (APR).

1. Spent Fuel Alternative: Physics Characteristics

Basic cycle-dependent physics characteristics of the MOX fuel cycle concept evaluated for the Spent Fuel Alternative are shown by the following figures:

- Figure III.F-1: Critical boron vs. burnup
- Figure III.F-2: Inverse boron worth vs. burnup
- Figure III.F-3: Core CEA worths
- Figure III.F-4: MTC vs. burnup
- Figure III.F-5: FTC vs. burnup

The magnitude and burnup trend of the parameters shown for the MOX concept are similar to those of APR cycles based commercial-grade recycled plutonium, as were discussed in Section III.A-3. In particular, the values of parameters near end-of-life (EOL) approach values characteristic of commercial UO_2 cycles.

2. Pu Destruction Alternative: Physics Characteristics

The corresponding cycle-dependent physics characteristics of the non-fertile fuel cycle concept evaluated for the Pu Destruction Alternative are shown by the figures listed below:

- Figure III.F-6: Critical boron vs. burnup
- Figure III.F-7: Inverse boron worth vs. burnup
- Figure III.F-8: Core CEA worths
- Figure III.F-9: MTC vs. burnup
- Figure III.F-10: FTC vs. burnup

The magnitude and burnup trend of the parameters shown for the non-fertile concept differ in a number of respects from those of APR cycles based commercial-grade recycled plutonium. The significance and implications of these differences are addressed in the section below.

3. Comparison of Parameters for MOX and Non-fertile Concepts

Table III.F-1 gives a comparison of physics parameter for the MOX and non-fertile concepts using weapons-grade plutonium and for commercial fuel cycles based on UO_2 operation and APR plutonium recycle. Specific parameters are discussed below based on the comparisons provided.

a. Critical Boron Concentration (CBC)

Calculated values of CBC at full power over cycle are shown in Figures III.F-1 and III.F-6 for the MOX and non-fertile cycles. The values of inverse boron worth (IBW) are similarly shown in Figures III.F-2 and III.F-7.

For the MOX cycle, a relatively high CBC exists at BOL, consistent with the reduced soluble boron worth shown by the IBW. The CBC decreases over cycle, and indicates that a significant amount of excess reactivity remains at EOL (1096 EFPD). The excess reactivity shown at EOL is favorable from the viewpoint of providing flexibility for optimizing the design and cycle length of the MOX concept.

For the non-fertile fuel cycle, the CBC is relatively lower through cycle, indicative of a lower amount of excess reactivity. The CBC trend shown for the non-fertile cycle is lower at BOL, increases at middle-of-life (MOL), and decreases rapidly near EOL. The lower CBC at BOL is a consequence, in part, of the high required erbium loading which is necessary to provide a negative Doppler coefficient for this concept. The depletion rate of the higher erbium concentration results in a net increase in reactivity with burnup, resulting in a higher CBC at MOL. The drop off of CBC near EOL results from the rapid diminishment of Pu^{239} concentration in comparison to fission products and other neutron absorbing materials in the lattice. The deleterious effect of erbium depletion for the non-fertile concept may be compensated by optimizing the design of the insertable burnable poisons (e.g., to include a strong burnup-independent component such as hafnium in the poison rods), and removal of the insertable poisons in the last irradiation cycle. The high poison requirements and sensitivity of the cycle-dependent reactivity to plutonium depletion indicate, however, that the design and cycle length optimization for the non-fertile concept would be significantly more difficult than for the MOX concept.

Critical boron concentrations for calculated operating conditions at BOL are compared in Table III.F-1. The concentrations of natural soluble boron shown for the MOX and non-fertile cycles are consistent with the amount of excess reactivity at BOL. Since the overall excess reactivity of the MOX cycle is higher than required for cycle length, it is expected that the CBC values and the refueling boron concentration for an optimized design can be reduced to values near those for the commercial APR cycle. The lower CBC values for the non-fertile cycle reflect lower excess reactivity of the cycle.

b. Control Rod Worth

Calculated values of core reactivity worth of the control element assemblies (CEAs) over cycle are shown for conditions of hot-full-power (HFP), 300°F zero-power, and 68°F zero-power in Figures III.F-3 and III.F-8 for the MOX and non-fertile cycles. The CEA worth values are based on the extended CEA pattern for APR operation as described in Section III.A.2. On the basis of the calculated results, the available CEA shutdown worth is sufficient for normal operations and safety-related requirements of both the MOX and non-fertile cycles. The cycle-dependent behavior of CEA worth shows continuous increase in worth from BOL to EOL for both the MOX and non-fertile cases, consistent with the expected trend based on depletion of plutonium.

c. Moderator Temperature Coefficient (MTC)

Calculated values of MTC over cycle are shown for in Figures III.F-4 and III.F-9 for the MOX and non-fertile cycles. These curves show the magnitude and trend of MTC at a constant soluble boron concentration.

For the MOX cycle, MTC is less negative at BOL and trends to a more negative value at EOL. Overall, the MTC characteristic for the MOX cycle is favorable relative to that of UO_2 cycles. The high fissile plutonium content (supplemented by the effect of erbium) provides a more negative MTC at BOL than in UO_2 fuel cycles. The MTC at EOL is similar in comparison to UO_2 cycles due to depletion of plutonium. Calculations for full-power conditions and zero-power conditions, xenon-free conditions further show that MTC is negative for all critical conditions over the MOX cycle.

For the non-fertile cycle, the MTC trend with burnup is in the positive direction, opposite to that of MOX or UO_2 cycles. This trend may be explained by the high rate of increase in H/Pu ratio with burnup due to plutonium destruction. As a result, the positive trend in MTC accelerates with burnup. This characteristic may prove restrictive on the design and cycle length for the non-fertile concept.

d. Fuel Temperature Coefficient (FTC)

Calculated values of FTC over cycle are shown in Figures III.F-5 and III.F-6 for the MOX and non-fertile cycles. For the MOX cycle, the negative FTC magnitude is comparable to that of UO_2 cycles, with little variation over cycle. For the non-fertile cycle, the negative FTC is substantially smaller in magnitude at BOL (i.e, one-half the value for MOX or UO_2 cycles) and diminishes with burnup. The small magnitude of FTC for the non-fertile concept has potential detrimental safety implications which may prove restrictive on the design and cycle length capabilility for this concept.

e. Delayed Neutron Fraction

Comparisons of delayed neutron fraction (β_{eff}) and prompt neutron lifetime (ℓ^*) are given for BOL and EOL in Table III.F-1. For the MOX cycle, the values of β_{eff} are in the range of .003, which is lower than for the commercial APR due to the high Pu^{239} concentration in combination with U^{238}. Based on evaluations for commercial APR cycle, the lower β_{eff} for the MOX concept is expected to be acceptable for safety-related performance (e.g., CEA ejection accident). For the non-fertile cycle, the β_{eff} value at BOL is in the range of .002 which is characteristic of Pu^{239} as the only fission nuclide. The lower β_{eff} has more adverse potential implications for the non-fertile concept since it is accompanied by a significantly lower FTC, and therefore may be restrictive on the design.

Table III.F-1

Comparison of Safety Related Physics Parameters

PARAMETER	COMMERCIAL UO2 EQ. CYCLE	COMMERCIAL APR EQ. CYCLE	WEAPONS-GRADE MOX APR	WEAPONS-GRADE NON-FERT APR
MTC (delta-rho/deg F)				
Full Power, BOL	-5.90E-05	-1.00E-05	-5.30E-05	-2.07E-04
Full Power, EOL	-3.24E-04	-3.10E-04	-2.89E-04	-7.20E-05
FTC (delta-rho/deg F)				
Full Power, BOL	-1.24E-05	-1.01E-05	-1.37E-05	-6.50E-06
Full Power, EOL	-1.25E-05	-1.09E-05	-1.39E-05	-3.90E-06
Dissolved Boron (ppm)				
CBC at BOC, Unrodded				
Hot Standby	1589	3189	3839	1860
Full Power, no Xe	1400	2450	3220	1474
Full Power, Eq. Xe	1170	2100	2896	1138
Refueling (5% subcrit)	1955	4203	4996	3312
IBW (ppm/delt-rho)				
Full Power, BOL	116	383	390	318
Full Power, EOL	101	331	277	126
CEA (%delt-rho)				
Full Power, BOL	13.8	12.6	12.8	12.7
Eff. Delayed N. Fraction				
BOL	0.00625	0.00442	0.00308	0.00209
EOL	0.00546	0.00447	0.00364	0.00318
Prompt N. Lifetime (sec)				
BOL	2.13E-05	6.80E-06	6.68E-06	8.19E-06
EOL	2.48E-05	7.90E-06	9.44E-06	2.08E-05

FIGURE III.F-1

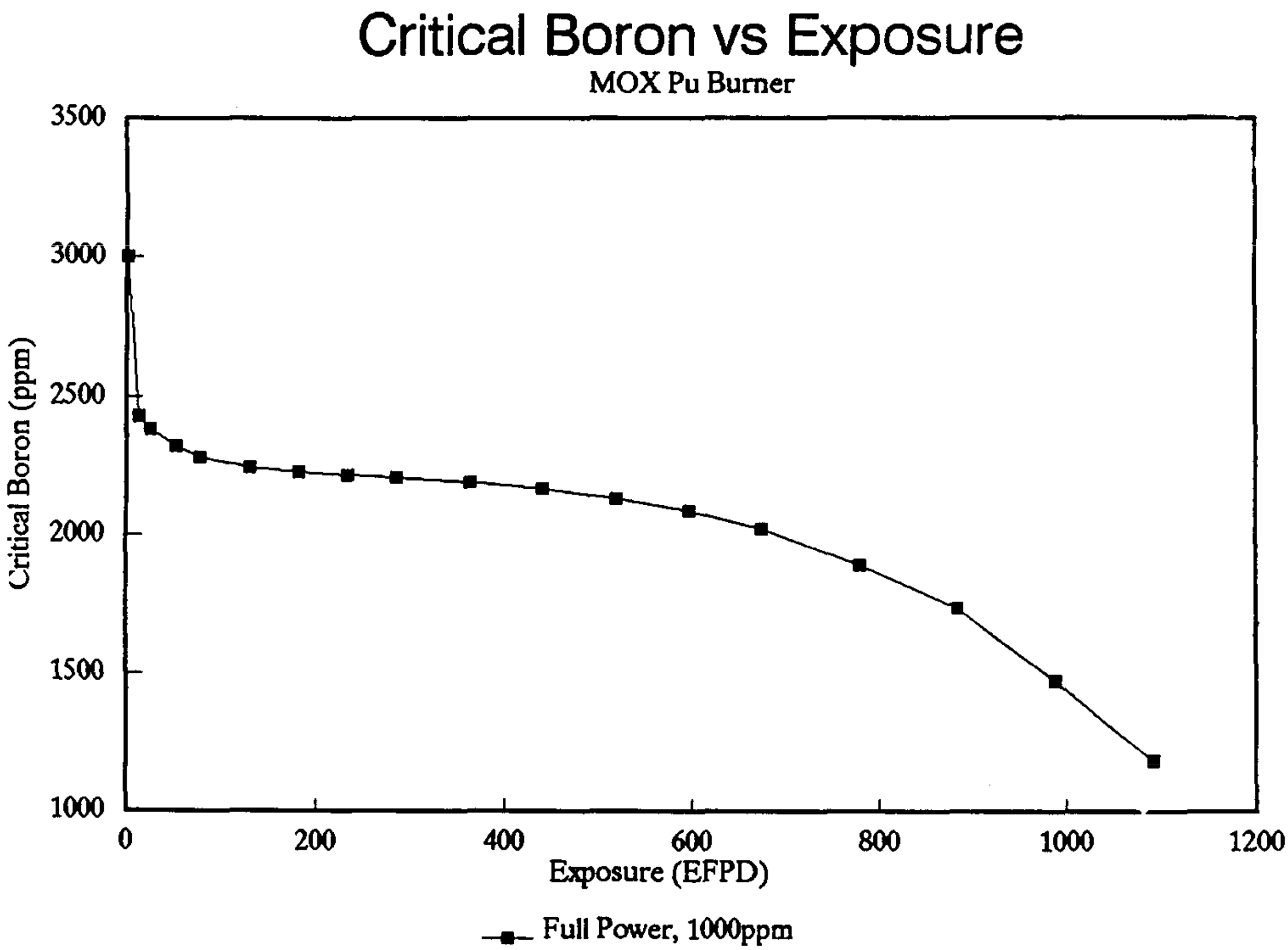


FIGURE III.F-2

INVERSE BORON WORTH VS BURNUP

MOX Pu Burner

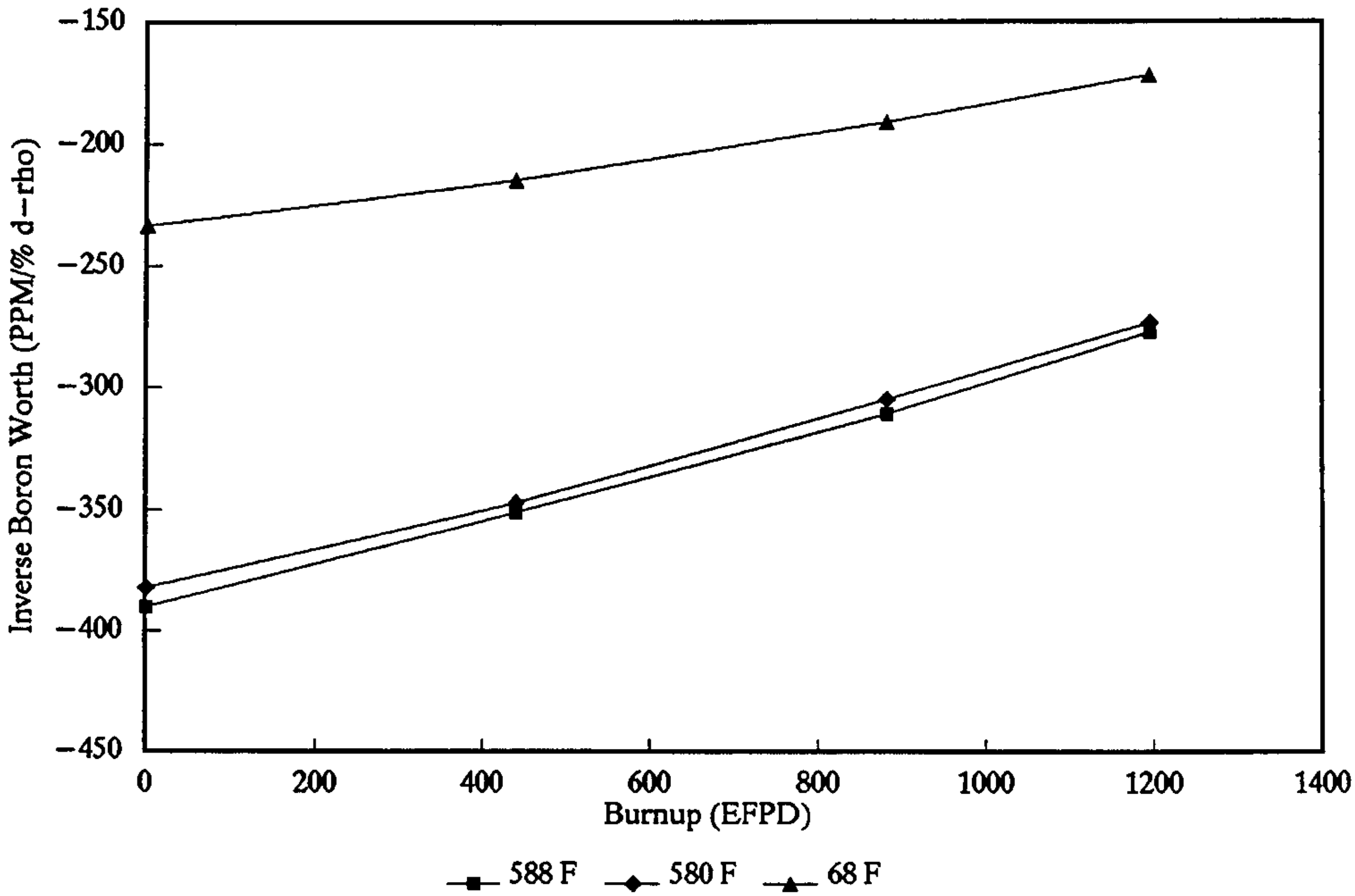


FIGURE III.F-3

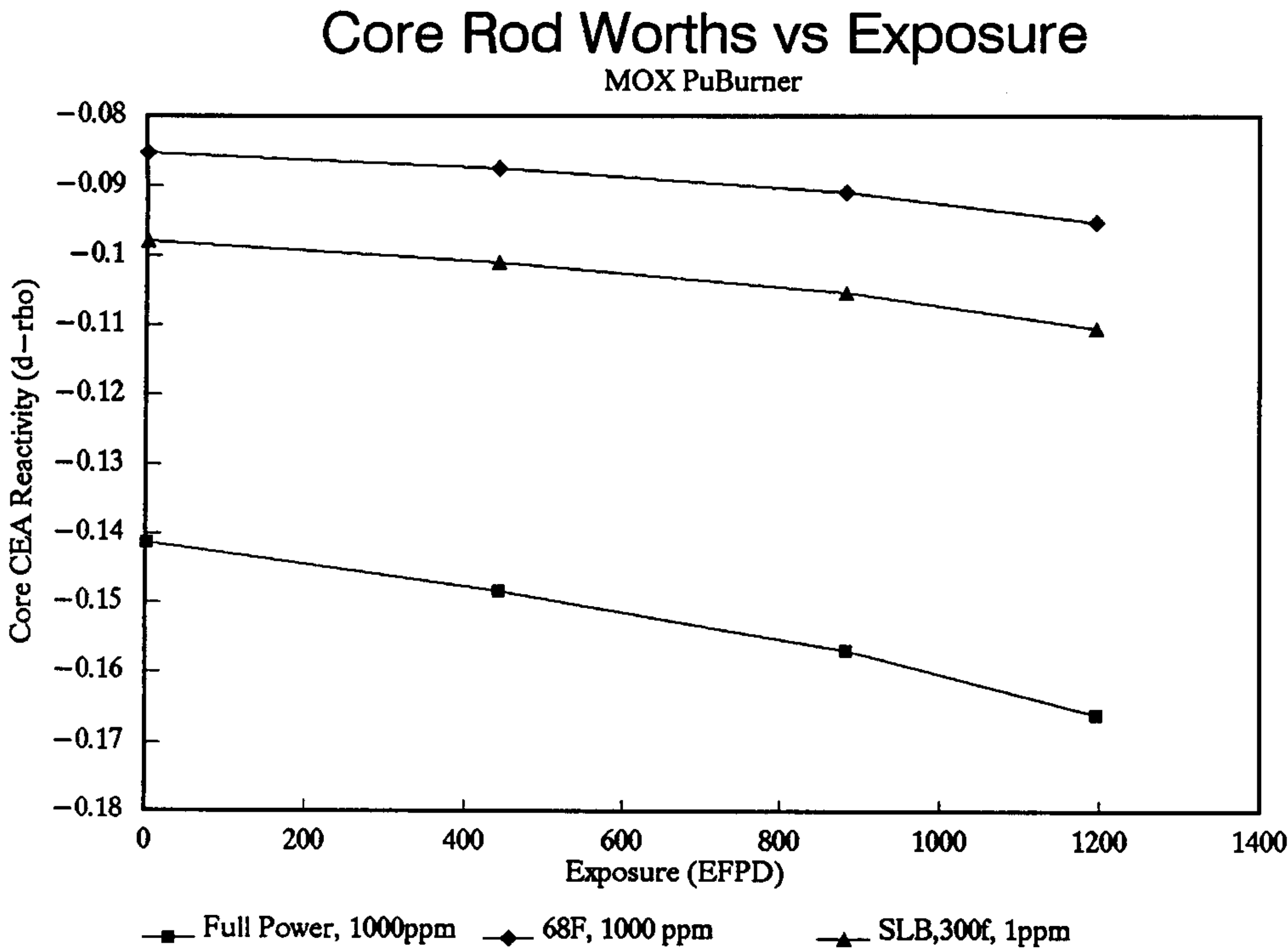


FIGURE III.F-4

MTC VS BURNUP

MOX Pu Burner

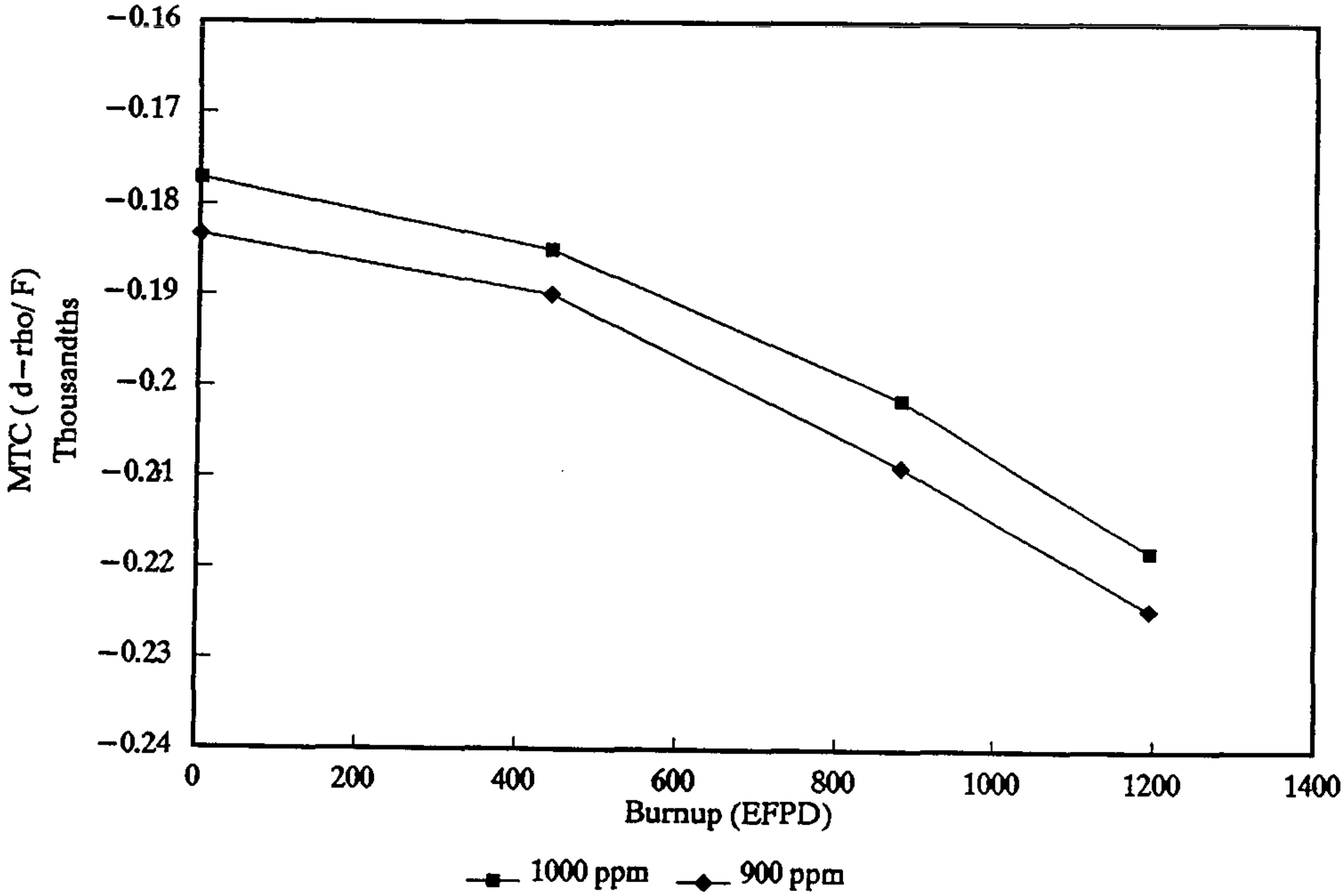


FIGURE III.F-5

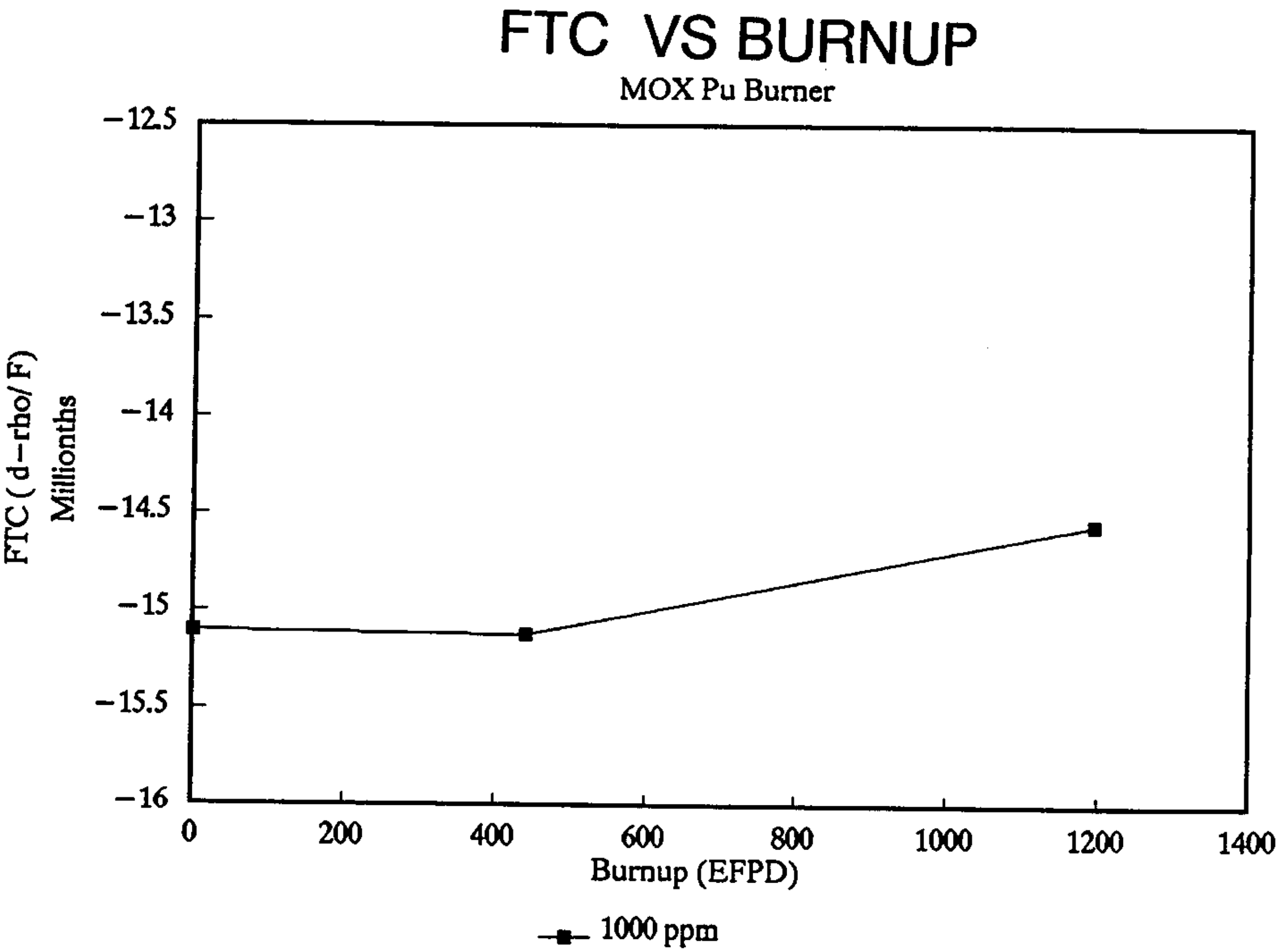


FIGURE III.F-6

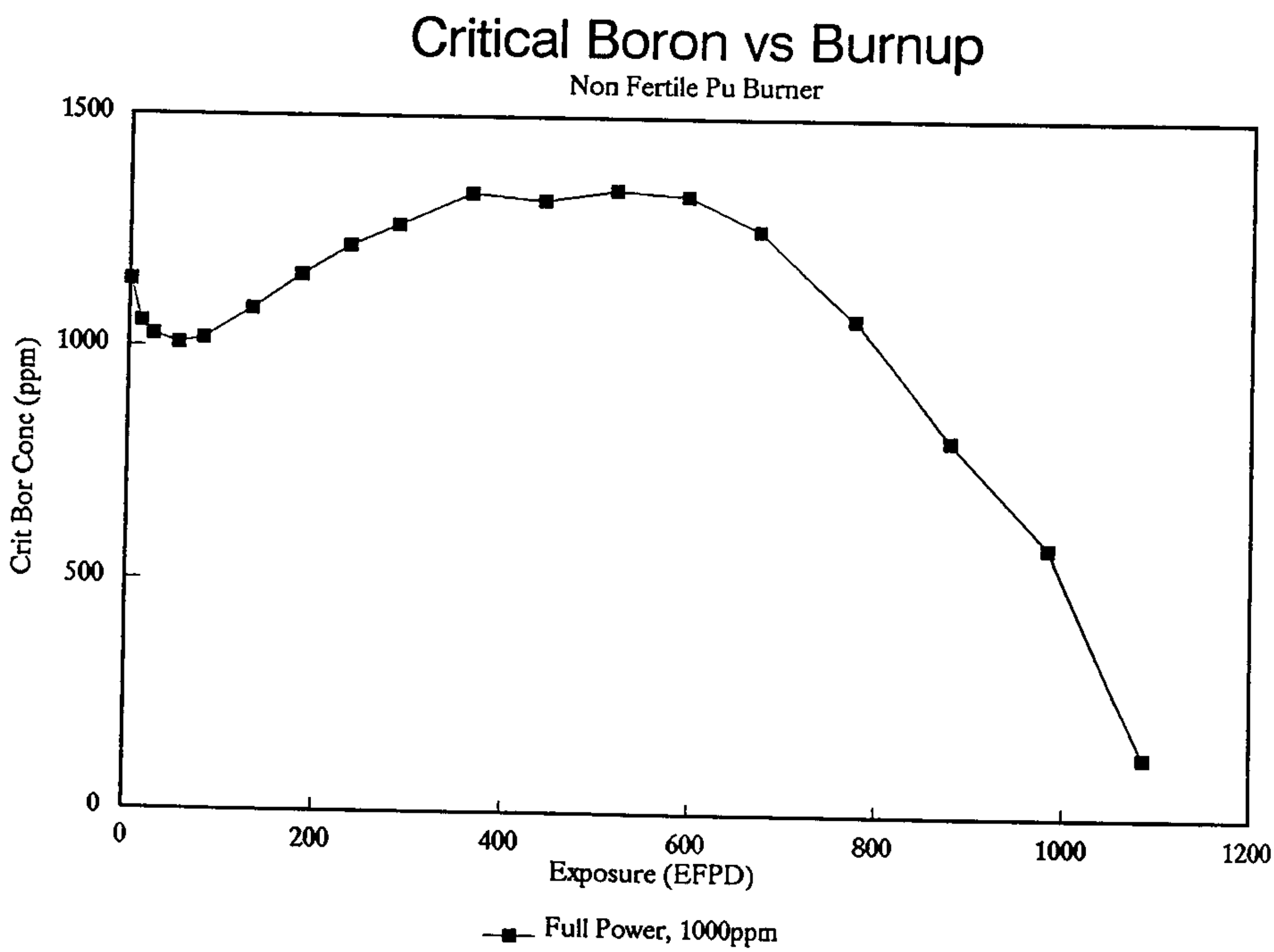
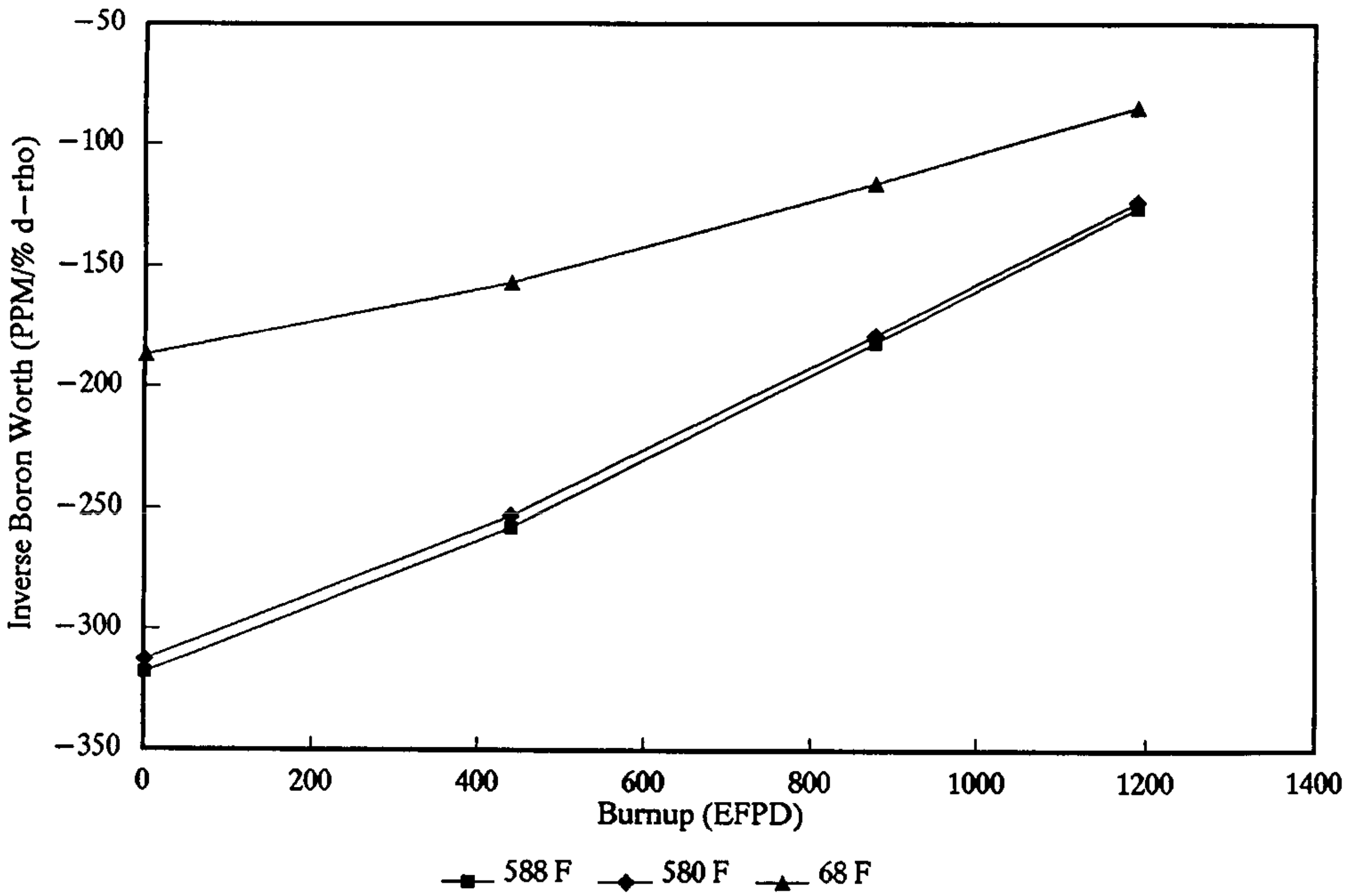


FIGURE III.F-7

INVERSE BORON WORTH VS BURNUP

Non Fertile Pu Burner



ASEA BROWN BOVERI

FIGURE III.F-8

Core CEA Worths

Non Fertile Pu Burner

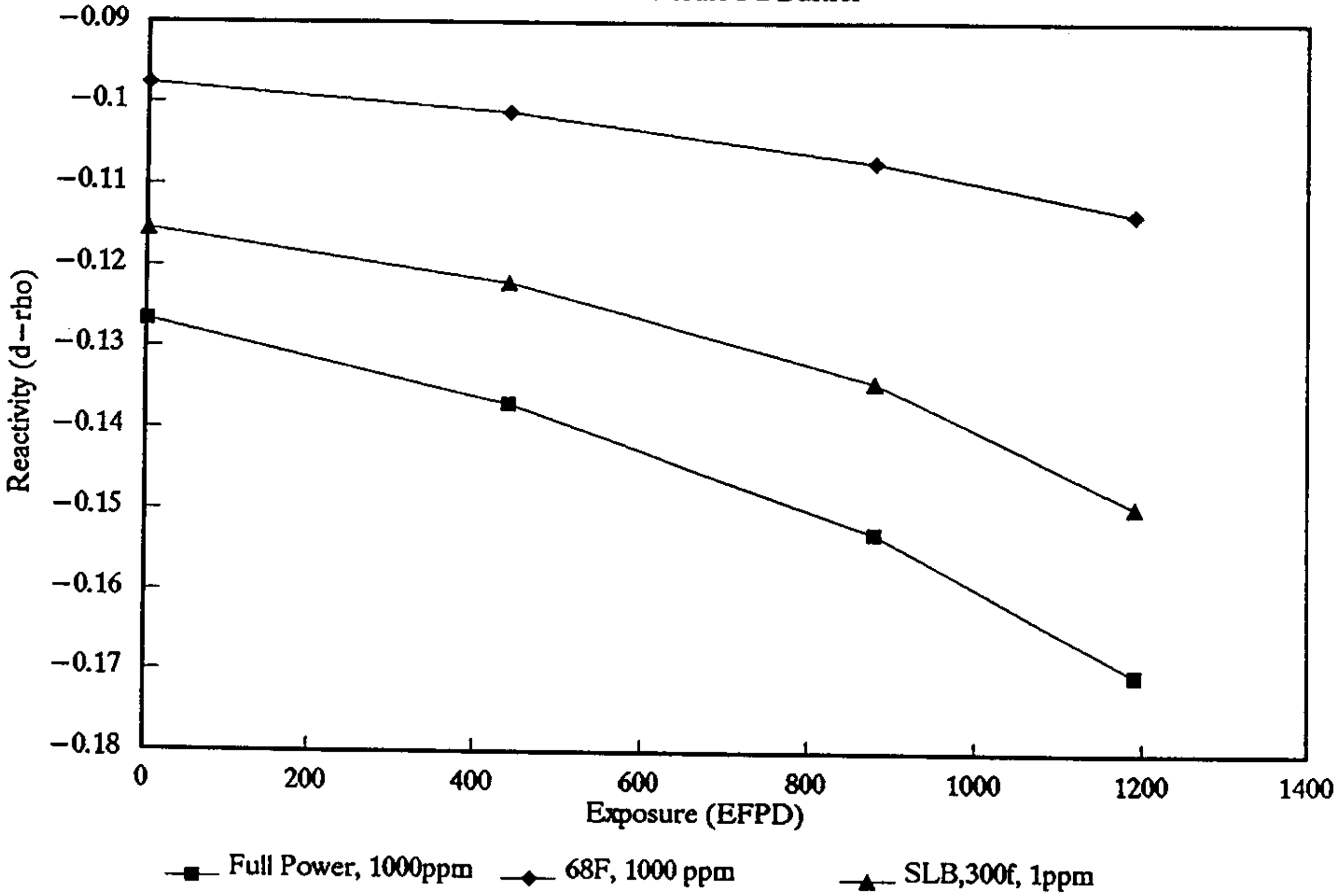


FIGURE III.F-9

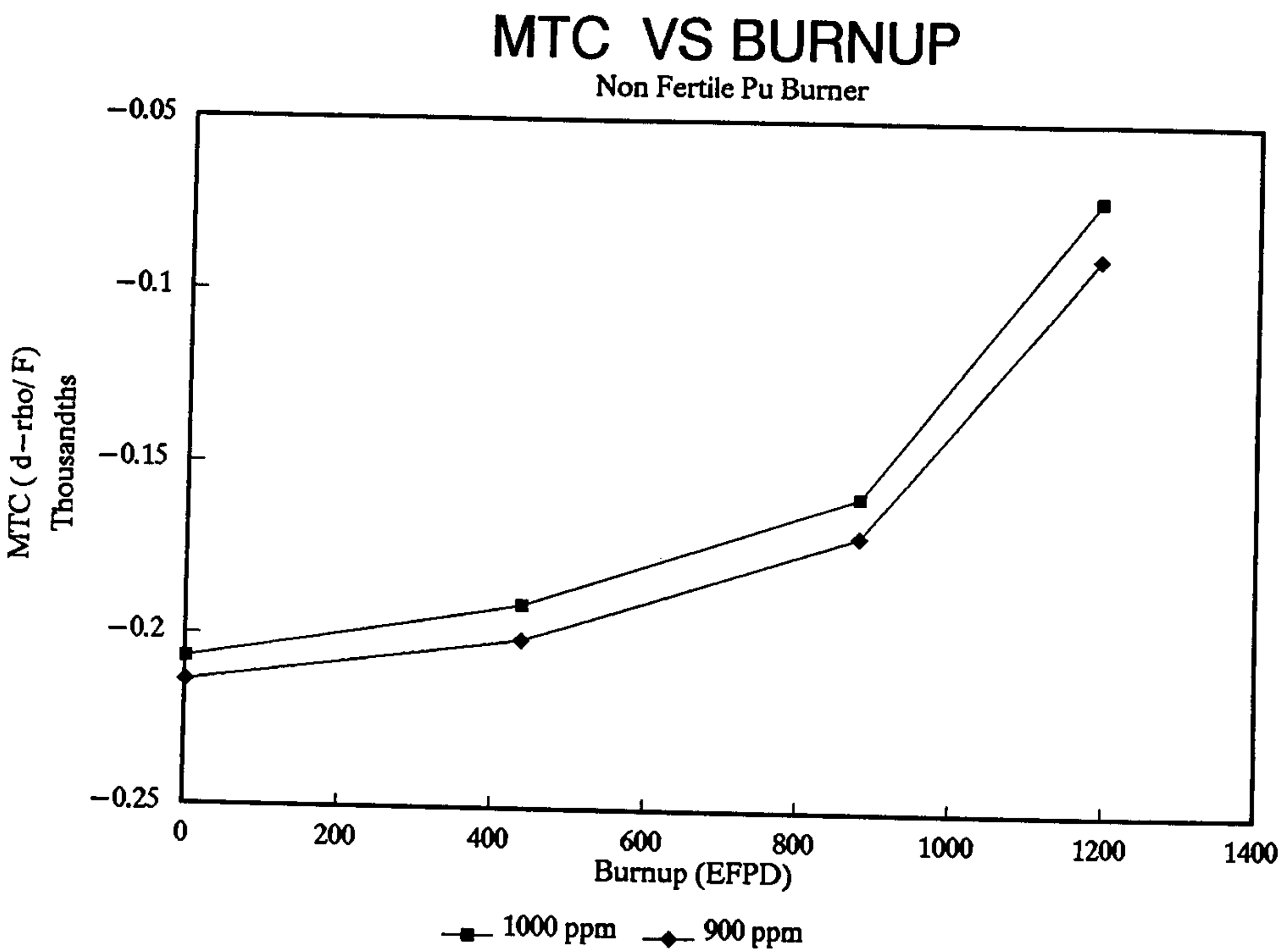
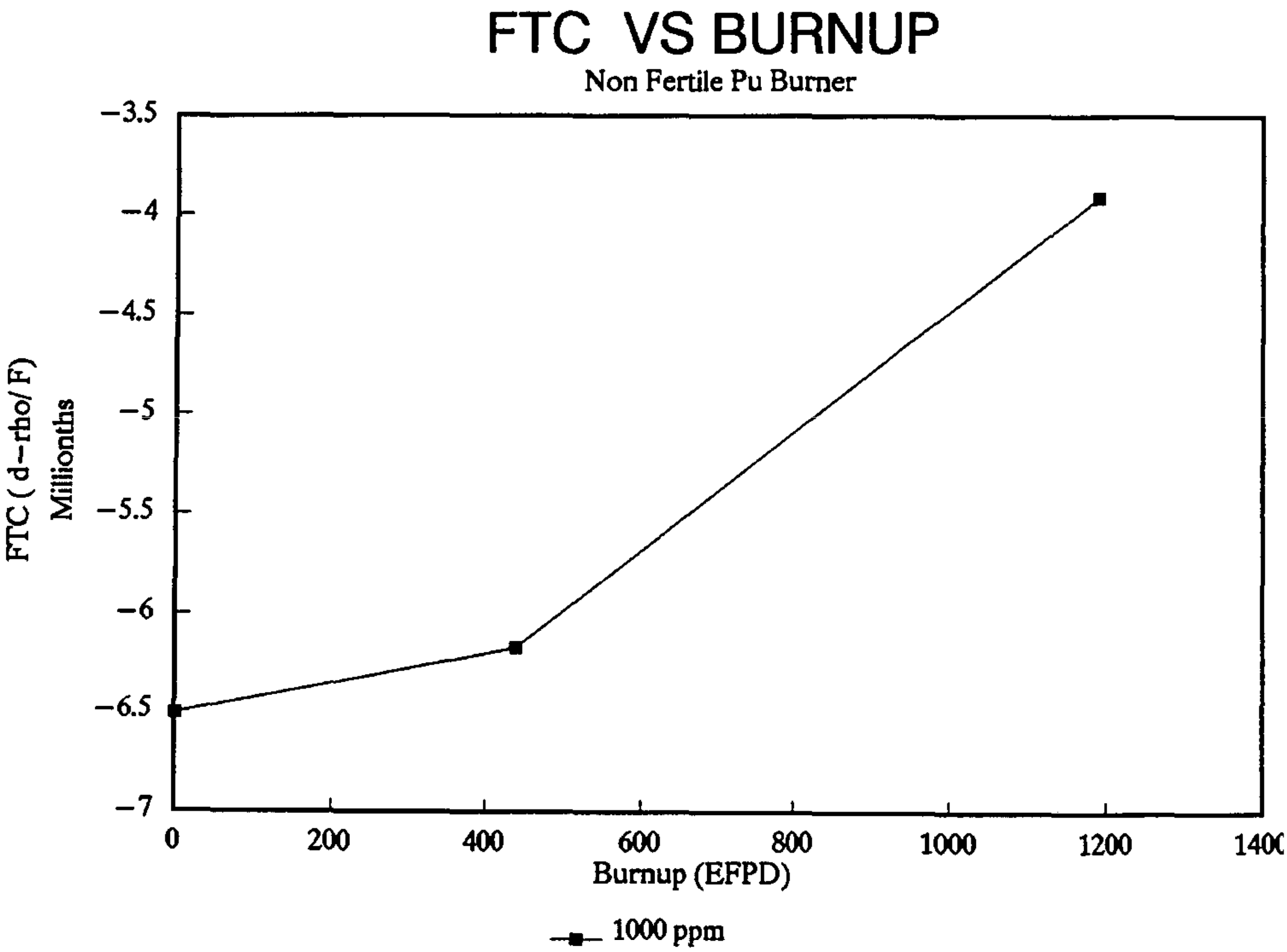


FIGURE III.F-10



G. TRITIUM PRODUCTION CAPABILITY

1. Summary

As part of the Plutonium Disposition Study, the feasibility of producing tritium in the plutonium burning reactor was evaluated. While not a specific requirement in the Requirements Document, an objective is to produce tritium with as few changes as possible to the plutonium burning design in order to minimize the impact of tritium production on the reactor. The required design changes are specified, and these changes have been evaluated to determine the impact on fuel and plant design, reactor performance, and reactor safety.

Tritium production requirements are expected to be achieved with small modifications to the plutonium core. The modifications include the removal of burnable poison pins and twenty fuel pins in each fuel assembly, replacing them with 32 tritium producing targets. Reactor power has been reduced slightly to meet thermal margin requirements. These modifications allow for the production of greater than contract quantities of tritium, while at the same time preserving the total mass of plutonium denatured in either a spiking or normal power production cycle. However, since power is reduced, the total number of plutonium atoms which can be destroyed in a tritium producing core is approximately 1% less at the same fuel exposure than in a dedicated plutonium burning mission for one cycle of operation.

It is expected that the tritium producing design can be operated using the same fuel management options as the plutonium burning core without compromising other safety or operational parameters. To produce contract quantities of tritium, however, a one batch core must be used. Assessments have been made of the impact on reactor control systems, core thermal performance and transient performance, and it is expected that the performance in these areas is within the design envelope of the plutonium burning only design. Assessments of other plant operations, such as refueling and balance-of-plant operations, are also made, with the indication being that the impact of tritium production operations will be small.

Support facilities that would be required specifically for a tritium mission are identified. Under the auspices of the Light Water Tritium Target Development Program, most of the support facility requirements and technical issues have been addressed. Premature termination of the program did not allow all target facility development items to be completed. Target fabrication and tritium extraction facilities are discussed, and development needs identified.

2. Introduction

The objectives for the Plutonium Disposition Study were specified by DOE in the Requirements Document released on January 21, 1993 (Reference 1). A requirement specified in the Requirements Document was that the reactor complex have the option of producing a specified quantity of tritium. For the System 80+ Plutonium Burner, the tritium production mission has been configured so that weapons-grade plutonium is burned as fuel during the tritium production mission.

3. Tritium Production Objectives

A objective of the Plutonium Disposition study is to determine the feasibility of producing contract quantities of tritium in a light water reactor whose primary purpose is the destruction of weapons grade plutonium. This study addresses the production of tritium while also burning weapons-grade plutonium. The effect of the tritium production mission on the quantities of plutonium destroyed are addressed. The goal of the tritium design is to produce tritium with as little impact to the plutonium burning process as possible. This study investigates the impact of operating a plutonium burning tritium production core versus operating a plutonium burning only reactor.

Work objectives include neutronic analysis to determine the assembly and target design parameters required to produce contract quantities of tritium. Changes in the plutonium burning only design required to produce tritium are analyzed for their effect on plant operations and safety. Impacts on reactor control systems, core thermal performance and transients, and fuel management options are identified and assessed. The effects on operational and environmental issues are addressed.

In addition to plant operations, discussion of the current state of light water tritium target development is made. The additional facilities and operations required for target fabrication and tritium extraction are described. Additional target development needs are addressed.

4. Tritium Production Assembly Design Description

The development of a fuel assembly design which will produce tritium as well as burn plutonium in a light water reactor is an evolution of a plutonium burning assembly developed by CE. The plutonium burning assembly is a genesis from the commercial System 80+ design. The plutonium burning version has been designated as the System 80+. The evolution to an assembly which both burns plutonium and produces tritium is designated as System 80+PT.

The System 80+PT assembly design is based on the System 80+P design, with only minor modifications to the mechanical design. The mechanical modifications to the design are needed to accommodate the tritium production mission. Since the design modifications are minor, and most effects of the modifications are expected to be within the operational and safety envelope of the System 80+P design, all performance and safety effects of the modifications are related directly to the System 80+P design. The intent is to show that the System 80+P and the System 80+PT can operate within the same design envelope. However, modifications to the fuel cycle are required to meet production requirements.

Reference Plutonium Burning Design

The reference System 80+ Plutonium Burner assembly design is described in detail under other task reports of this project. A brief summary is provided here to allow for comparison to the tritium production design.

Table III.G-1 gives a summary of the major System 80+ Plutonium Burner design parameters. The parameters provided are for the maximum power generation option. In the maximum power generation mode, a three batch core on a 1-year cycle is used. If tritium production is not desired, this core design is the most desirable, and is therefore the design to which comparisons are made.

The design core operating power is 3800 MWth. This power is obtained with a loading of 6.9 w/o weapons-grade plutonium in heavy metal and an average erbium oxide loading of less than 2 w/o in MOX. Each assembly in the core has either zero or twelve burnable poison rod (BPR) locations. Figure III.G-1 shows the assembly layout with twelve BPR locations.

The reference System 80+ Plutonium Burner design is expected to meet all required neutronic and thermal hydraulic safety margins necessary for the licensing of a commercial reactor core. The additional control locations available in the System 80+ design provide sufficient rod worths and control margins in the reactor for all-plutonium-reactor operations. Since the basic assembly and balance-of-plant design is the same as the System 80+, and the average linear heat generation rate is the same as the System 80+, thermal margin and transient performance are expected to be within the design envelope of the commercial System 80+ design. For a more complete discussion of the System 80+ Plutonium Burner design, see Section III.A.

Plutonium Burning-Tritium Producing Assembly Design

The tritium production option has been assessed for the System 80+ reactor design. The conceptual design of the System 80+ Tritium Production design allows for the continued destruction of plutonium while at the same time producing desired tritium quantities. The design characteristics of the fuel assembly have been slightly modified, however, such that plutonium burning and power production are no longer optimized. The deviations in design are sufficiently minor that major plant design modifications are not required.

The System 80+ Tritium Production design is based on the System 80+ Plutonium Burner described above. The mechanical modifications involve the removal of the 12 BPRs from the assembly, replacement of 20 fuel rods with non-structural guide tubes, and inserting target rods containing lithium in their place. The total number of target rods per assembly is 32.

Thirty-two target rods per assembly are required to produce the desired tritium quantities. The quantity of tritium which can be produced per target rod is limited by target rod design considerations. To meet production requirements set forth in DOE guidance[1] while staying within the established target performance envelope, more targets are required in a fuel assembly than the available burnable poison locations. A detailed discussion of the target design is provided below.

To meet production requirements, the reactor fuel cycle must also be altered. Analysis has shown that greater than contract quantities of tritium are produced in

[1] Letter from J. A. Delos Santos to D. F. Newman dated March 19, 1993

the System 80+ Tritium Production core when a one batch core is operated on a 1-year cycle, with fresh fuel being loaded every cycle. Multi-batch cores have not been analyzed for this study. Tritium can also be made using multi-batch cores or longer cycles, however some tritium production capability is likely to be lost.

Since the desired goal is to remain within the same safety design envelope as the System 80+ Plutonium Burner assembly design, maintaining the same average linear heat generation rate (LHGR) is a reasonable method of accomplishing this objective. The removal of 20 fuel pins per assembly to accommodate the target rods requires that the total core power be reduced in order to maintain this average LHGR. The System 80+ Tritium Production core is therefore proposed to operate at a reduced power of 3410 MWth.

Table III.G-1 lists the major design features of the tritium production core. Comparison to the System 80+ Plutonium Burner parameters indicates additional minor changes which were made to further optimize the tritium production assembly design. Figure III.G-2 shows the assembly pin layout in the System 80+ Tritium Production core including the additional target pins.

Fuel Assembly Loading

The System 80+ Tritium Production conceptual core has been analyzed using an averaged fuel assembly design. Full core calculations have not yet been performed, so the effects of variations in axial or radial enrichment patterns have not yet been determined. It is anticipated that continued work would include parametric calculations which will lead to design optimization.

The System 80+ Tritium Production fuel rods contain a single enrichment of 7.38 w/o PuO_2 and 0.5 w/o Er_2O_3 in MOX. The fuel and poison loadings in the System 80+ Tritium Production core are marginally different from the System 80+ Plutonium Burner. The PuO_2 concentration has been increased slightly from 6.9 w/o to 7.38 w/o to maintain the same total core mass of plutonium as in the System 80+ Plutonium Burner. This additional concentration is required because of the removal of fuel pins to accommodate the required targets. By increasing the plutonium concentration in fuel and maintaining the same fissile content as the System 80+ Plutonium Burner, the effective amount of weapons-grade plutonium denatured per year in either a spiking or normal power generation mode remains constant.

The neutronic design for the System 80+ Tritium Production assemblies was performed using the WIMS-E neutronics code. The WIMS-E code model developed uses a two-dimensional, integral transport methodology to calculate reactivity, temperature coefficients, and tritium production capabilities of the tritium production assembly. Figure III.G-3 shows reactivity plotted against burnup for the tritium production core, and Figures III.G-4 and III.G-5 show the moderator temperature coefficients and doppler coefficients, respectively, with burnup.

The calculated moderator temperature coefficients for the System 80+ Plutonium Burner core are also provided in Figure III.G-4 for comparison. Comparison indicates that the System 80+ Plutonium Burner and the System 80+ Tritium Production cores are neutronically similar, with the coefficient values within acceptable ranges of each other. The core would therefore be expected to operate in similar manners during moderator temperature transients. Fuel temperature coefficients were not available for comparison to the System 80+ Tritium Production, however the coefficients shown in Figure III.G-5 are comparable to coefficients in commercial reactor cores, which are in the range of -2.0×10^{-5} to -1.0×10^{-5} $\Delta\rho/°F$.

There will be a slight difference in the total number of plutonium atoms destroyed per year between the two cores since the total core power in the System 80+PT is lower than the System 80+ Plutonium Burner. Figure III.G-6 shows the plutonium isotopic destruction in the System 80+ Tritium Production and the System 80+ Plutonium Burner. Comparison indicates that the difference in the number of plutonium atoms destroyed at the same exposure is only about 1% less than in the System 80+ Plutonium Burner.

The erbium loading in the tritium production assembly is smaller than in the System 80+ Plutonium Burner. This is largely due to the reactivity effects of the increased core poison loading held in the target rods. The tritium target rods effectively act as burnable poisons in the tritium core, and the total poison loading of target rods is greater than the BPRs in the System 80+ Plutonium Burner. The erbium loading must therefore be lowered to meet total core reactivity requirements.

Parametric studies on fuel design parameters have not yet been performed for the System 80+ Tritium Production. Studies performed would investigate how variations in design parameters such as fuel, target and poison loadings impact performance, safety and production parameters. The results of these parametric studies would be used to further optimize the tritium production assembly. These parametric studies are not appropriate to the level of this study. Full core calculations have not been performed, and it is expected that the results of full core calculations will be required to optimize the design. It is expected that these studies will allow the design to be further optimized to enhance plutonium destruction as well as tritium production. As an example, core design optimization will likely permit the core power to be increased during the tritium production mode.

Target Design

The System 80+ Tritium Production core design utilizes tritium producing targets located in guide tubes. The targets are similar in design to the targets which have been developed as part of the Light Water NPR Tritium Target Development Program (TTDP). Design parameters for fabrication and tritium extraction from the System 80+ Tritium Production targets are expected to be well within the development and performance parameters of the targets designed for the TTDP.

Figure III.G-7 shows a cut-away of the target structure. Table III.G-2 lists the dimensions and composition of the target components. The major functional components of the target are the pellet, the liner, the getter, the barrier coated clad and the non-structural guide tube. The pellet is lithium aluminate. Lithium-6 in the target pellet absorbs neutrons to form tritium and helium as shown in the reaction below:

$$Li^6 + n \rightarrow H^3 + He^4$$

The zirconium liner effectively dissociates any THO or T_2O which forms into its constituent atoms. The getter is made of nickel coated zirconium. The zirconium absorbs tritium atoms in the target and keeps them immobile. The nickel coating prevents oxide layers from building and prohibiting the passage of tritium into the getter. The stainless steel cladding encapsulates the target components to contain the tritium, and the aluminum barrier coating is an effective permeation barrier to tritium.

The target guide tube function is two fold. The first function is merely to hold the target in place in the core, and to allow easy removal and replacement of the targets. It is essentially a tube into which the target is slid. The second purpose is to protect the target from the effect of transients. In design basis transients such as LOCAs, the guide tube thermally isolates the target from the potentially high radiative heat transfer from the fuel pins. In extreme cases, the guide tube may reduce the maximum temperature of the target clad by 400°F when compared to the same design basis event (DBE) using targets without guide tubes.

The target mechanical design is governed by the expected neutronic and mechanical performance requirements in the core. The target performance envelope is defined primarily by two considerations: the gas-to-volume ratio (GVR) in the target and the potential target internal gas pressure during a transient. This performance envelope controls many aspects of the fuel assembly design because it governs the quantities of tritium that can be made per target, and therefore the number of targets needed to produce a given quantity of tritium.

The GVR criteria is imposed because the evolution of excess gaseous products in a given target volume may cause the target pellet to disintegrate. Relocation of the target pellet poison material after disintegration could cause detrimental reactivity effects in the core, possibly leading to excessive localized power peaking. The maximum GVR of the maximum exposure target in the core must therefore be lower than a specified limit to ensure that no target pellets suffer disintegration from internal gas effects.

The GVR limit for the target design has been set conservatively low. Indications from the TTDP are that higher limits could be set, however the premature termination of the TTDP did not allow this to be verified. In addition to the conservative design limit, the exposure calculation used for design calculations assumes a conservatively high average total peaking factor of 2.0 for the entire irradiation period of the target. It would not be expected that any one target would experience total peaking to this extent for the entire cycle. This is a conservative power peaking assumption, thereby adding additional conservatism in the target design.

The assumed target internal gas pressure used for target design is conservatively high. High internal gas pressures could result from a DBE such as a large break LOCA. During such accidents, the target temperature rises, increasing internal target gas pressure. Additionally, at higher temperatures some tritium desorbs from the getter and target pellet, increasing the internal gas pressure further. The cladding yield strength also decreases with increasing temperature.

Indications from the TTDP are that the desorbtion of tritium from the target materials occurs slowly relative to the time length of a DBE. Design basis transients would only produce high temperatures for a few minutes at most, but for target design purposes it is assumed that 100% of the gas is immediately released from the getter and target pellet. The targets are then designed not to breach even during this maximum pressure loading. This design basis is even more stringent than the design basis for fuel pins, for which a limited number of fuel pins are permitted to rupture during design basis events.

Since the target performance envelope used in this study is considered to be very conservative, there is a high degree of confidence that target integrity will be maintained during all operating and transient conditions. The target is considered extremely robust, and is likely to maintain integrity at least as long, if not longer, than typical fuel rods during a DBE. Activities performed under the TTDP have increased the understanding of some target phenomena, and the confidence in the performance capabilities of the target is very high. Further design development of the targets is still required. However this development will likely allow for a better understanding of the target performance and a reduction in the conservatism now used in the design.

Having assumed the 100% gas release to the target free volume, the maximum target cladding temperature for which the design yield stress will not be exceeded is 1300°F. Again, the assumption is made that the transient occurs at EOC, when the gas inventory is greatest, and that the peak target has experienced an average 2.0 peaking factor during the cycle. It must also be remembered that the target guide tube is protecting the target clad, and that the guide tube temperature may be very much higher than 1300°F.

The LWR tritium target design is very flexible within the bounds of the performance envelope. Continued development of the tritium target technology would serve to allow for a decrease in some of the conservatism used in the target design. In addition, further analysis work on the System 80+ Tritium Production core design would allow refinement of reactivity and power peaking factors which may allow further reductions in conservatism and thereby allow for optimization of the target and assembly design.

5. System 80+PT Operational Features

Tritium Production Capability

The proposed tritium production assembly design is expected to make greater than contract quantities of tritium per year as required in the guidance document. This tritium requirement is made using a one-cycle, one-batch core configuration. The design is reasonably versatile in that variation in production requirements can be made by altering lithium loadings and erbium concentrations. Data is not presented here, however design iterations have shown that this assembly design is capable of making greater than contract quantities of tritium. Design optimization for any of these quantities can be achieved without difficulty.

Fuel Management Options

The System 80+PT design can be operated with fuel management and cycle length strategies similar to the System 80+P. Functionally, the tritium targets in the System 80+PT core displace other poisons (BPs, soluble boron or erbium) which are present in the System 80+P design. With design iteration and optimization, it is expected that a tritium production core could meet any reasonable cycle length or loading scheme requirements.

The System 80+PT core will only produce greater than contract quantities of tritium under certain cycle conditions, however. The System 80+PT is designed to exceed contract production requirements for tritium in a one batch core on an annual cycle. The design contains sufficient reactivity to provide 274 EFPDs per cycle. The tritium assembly design presented will produce greater than contract quantities of tritium only if a one batch core is used, and if the core is loaded with fresh fuel every cycle. The one batch option is the only core loading option which has been analyzed at this point. Conversion to a three batch core or to longer cycle lengths in a tritium production mode is feasible with additional design effort, however production penalties will likely be incurred unless plutonium concentrations are altered.

Conceptual core loading patterns have not yet been developed for the System 80+PT core. It is likely that when full core design analysis is performed, fuel bundles with slightly different plutonium, erbium, or lithium concentrations will be developed to obtain the desired core power profiles.

Burnup reactivity control in the System 80+PT core is accomplished by a combination of target depletion, erbium depletion and soluble boron. The System 80+PT core will use soluble boron enriched in boron-10 for reactivity control. The use of enriched boron maintains consistency with the boron enrichment planned for use in the System 80+P design. Since maximum flexibility between the tritium mission and non-tritium missions is required, the soluble boron requirements for the two concepts are the same.

6. Safety Impacts

Impact on Reactor Control Systems

No specific analysis has been performed to determine control rod worths in the tritium production core. The reactor control systems designed for the System 80+P are expected to be sufficient for use with the System 80+PT core. The fissile content is the same, and effort has been made to maintain the same reactivity and safety margins for the tritium core as the System 80+P core. As has been shown previously, the core temperature coefficients throughout the cycle are also similar. No radical design changes have been made which would create a challenge to the control system during normal or transient operations, therefore it is expected that no modifications or upgrades to the System 80+P design would be needed. In any case, a complete analysis of the control rod worths will be required before a tritium production core is implemented.

Core Thermal Performance

Preliminary thermal hydraulic analysis of the System 80+PT assembly has been performed to ensure that the modified fuel assembly and the target designs are within the design envelope of the System 80+P core. At this time, only normal operating conditions have been assessed, however this assessment coupled with assessments made for the Light Water NPR program are used to infer the results that would be expected from a full transient analysis.

Fuel Analysis

The System 80+PT fuel pin design is the same as the System 80+P design. However, the number of fuel pins, and the power in each pin has been altered. To show that the thermal margins have not been compromised, thermal analysis of the fuel pins has been performed.

At this time, analysis of fuel thermal margins is limited to the calculation of the departure from nucleate boiling (DNBR) ratio. The calculated DNBR is compared to the minimum allowable DNBR for the design. Since the DNBR criteria ensures that the cladding temperature remains close to the coolant temperature, no additional criteria for cladding temperature is required for normal operation and DBEs.

The design limit DNBR that is used depends strongly on the DNB correlation and analytical methods used. For the purpose of this analysis, the non-proprietary B&W-2 correlation is used. The design DNBR limit for 15 x 15 and 17 x 17 pin fuels is approximately 1.35. To apply the correlation to the CE 16 x 16 pin configuration, the limit was conservatively increased to 1.45 to account for any undetermined uncertainties in the application of the correlation. Additional conservatism is added to determine a design goal. This design goal maintains the margin between the DNBR goal and the DNBR limit at values similar to commercial light water cores. For purposes of conceptual design of the System 80+PT core, a minimum DNBR goal of 2.21 is assumed to maintain the same margins as commercial core designs.

The thermal-hydraulic performance the System 80+PT fuel has been performed using the VIPRE-01 sub-channel code (Reference 4). The hydraulic design is set to match the available parameters for the System 80+PT design. Pie-shaped, 1/8 sections of symmetry of the hot fuel assembly have been considered for evaluating the limiting design conditions.

A DNBR analysis was performed for the System 80+PT design. This was for the design operating condition of 102% overpower (3478 MW), 2205 psia system pressure, 558°F inlet temperature, and 95% core inlet mass flow rate (160.7 Mlbm/hr). Since full core power profiles are not yet available, total core peaking is based on an estimated core average axial peaking (a symmetrical chopped cosine), the expected maximum radial peaking at any location in the core, and a conservative local peaking factor. A total peaking of 2.35 was therefore assumed.

The DNBR analysis for the design operating conditions indicate a calculated DNBR of 2.66. Comparing to the goal of 2.21, this indicates that there will be sufficient thermal margin in the System 80+PT design. Further analyses will be required to establish specific margins, including using correlations optimized for CE fuel designs.

Fuel temperatures and other related fuel performance parameters for steady-state operation have not yet been thoroughly analyzed for the System 80+PT. Estimates for preliminary light water NPR fuel pin designs were performed and reported in Reference 3. Since the System 80+PT design has not yet been analyzed in detail, a comparison is made to the results of the NPR designs.

The estimated steady state fuel temperatures for the 16 x 16 System 80+PT fuel are a core average fuel temperature of 1330°F, with a peak fuel temperature of 3540°F. The peak occurs at the centerline at the design operating condition. These temperatures compare favorably with other plant designs.

Target Analysis

Total target peaking for the DNBR analysis was set to 2.15. This value is more conservative than the power factor of 2.0 used for the production calculations in order to account for and conservatively bound any uncertainties which have not been identified. The target power used is 8.46 kW/rod, which is conservatively based on previous target heat rate calculations performed, and takes into account reactor design differences. The analysis values chosen are expected to conservatively bound the System 80+PT core target design.

The estimated target temperatures for the System 80+PT target design are a core average target temperature of 755°F, with a peak target temperature of 980°F.

Pressure Losses

The core pressure loss for the 0.382 in OD fuel pin design has been estimated by the VIPRE-01 sub-channel code using a Blasius relationship for rough tube friction factors. The equation coefficients are defined for the specific relative roughness. The roughness of drawn tubing was assumed to be 5×10^{-6} ft. It was also assumed that six grid spacers were equally spaced within the active zone. The grid spacer pressure loss coefficients were assumed to be 1.20.

The active zone total pressure loss for the 0.382 in OD fuel pin design was estimated to 22.4 psi for the core active zone at a core flow of 160.7 Mlbm/hr.

Reactor Transient Performance

The reactor response to transients is driven by both the neutronic and thermal design parameters of the reactor system. No detailed calculation of transient scenarios has been performed on the System 80+PT core configuration to date. From other basic design information, however, insight can be gained as to the expected transient response of the tritium core in relation to the System 80+P core.

Reactor power during a transient is driven by the core response to upset condition before a scram is initiated. The core response is governed mainly by the changes in the coolant or fuel temperature levels. The assembly moderator temperature and fuel doppler coefficients (Figures III.G-4 and 5) give an indication of the core response rates. As previously discussed, the System 80+PT moderator coefficients are similar in magnitude to System 80+P or commercial reactor coefficients. Since it is likely that fuel temperature coefficients will also be similar, it is therefore expected that the neutronic response to temperature transients will also be similar.

Because of the modest fuel temperatures and adequate MDNBR margins present in the System 80+PT core when compared with existing commercial plant designs, no major differences are expected in transient performance. The 0.382 in OD design shows margin to DNB comparable to commercial cores at the design operating condition. However, detailed analyses remains to be performed to confirm this observation before a final design is produced.

7. Impact On Plant Operations

Refueling Operations

Little difference is expected between refueling operations for the System 80+ core concept and the System 80+ concept. Fuel assemblies will contain 32 targets in the tritium core, 20 of which displace fuel rods in the System 80+P. The total plutonium content per assembly is the same, and the weight of the assemblies will not be significantly different. The assembly envelope therefore will not be changed, thus manipulation of the assemblies will not be effected. The cycle length will remain at one year and all assemblies will be off-loaded each cycle. Consequently there will be no change in the method of shuffling assemblies.

In addition to the refueling operations, however, provision must be made for removal and replacement (if desired for multi-cycle cores) of the targets in the irradiated assemblies and for transport of the irradiated targets to an extraction facility. Target removal should be no more difficult than removal of BPRAs from assemblies, as is routinely done at commercial reactor sites. The removal of the target has been made as simple as possible by placing the targets in guide tubes and attaching them to a spider or baseplate assembly. Target replacement would occur if multi-batch cycles are planned during tritium production missions. If a tritium mission is no longer required, yet it is desired to burn the fuel further, BPRAs could be simply be placed in the target guide tubes since the target guide tubes and the BPR tubes are identical.

Target Handling

Consideration must also be given to the handling of irradiated targets, and the transportation of the targets to the extraction facility. Proper shielding will be required for target storage and shipping since the irradiated stainless steel clad will be highly activated after neutron exposure. This is likely to require the development of storage or shipping casks for the targets if current designs are unsuitable. Protection against tritium releases to the environment during storage and transport must also be considered. Since the tritium desorbtion rate from the targets is very slow, this is not considered a technical challenge.

Maintenance

The production of tritium in a light water reactor is not expected to affect normal maintenance activities to any significant extent.

Routine Releases

An Environmental Impact Statement (EIS) was prepared for the NPR program that (Reference 5) evaluated potential tritium releases from a light water tritium producing reactor. The EIS states that environmental releases of tritium from a light water tritium mission in a light water reactor would have an upper limit projection of tritium releases to the environment of 20,000 curies, as compared to approximately 900 curies from a commercial light water plant. The projected environmental exposures resulting from these upper bound releases were well within prescribed dose limits.

The EIS was written at a time when goal tritium production was much higher, and before much of the TTDP work was completed. Actual releases and exposures are expected to be significantly lower for the System 80+PT design than the upper limit projections in the EIS for the following reasons:

- There would be far fewer target rods irradiated in the System 80+PT core than assumed in the TTDP since the required tritium production is lower. With fewer target rods, it would be expected that there would be fewer target failures, and hence, lower tritium releases.

- The upper limit release estimate for the EIS assumed failure of two target rods per year at EOC. The failed rods were assumed to release their full inventory of tritium. TTDP work has indicated that the targets are far more robust than originally expected, and that two per year is very conservative. Although the average of two failed rods per year was conservatively consistent with fuel rod failure experience, it is highly unlikely that target rod failures would occur at the same rate. As noted previously, the target rods have been designed far more robustly than fuel rods. They have about one-tenth the heat rate, the guide tube protects the targets from debris and fretting, and the stainless steel cladding is much stronger than the zircaloy cladding used for fuel tubes.

- TTDP analysis indicates that when target rods do fail, only one-fourth to one-half of the inventory would be released on the average from a failed target rod.

- Tritium permeation factors from the target rods for the upper limit projections used in the EIS are based on the use of a diffusion barrier which provides a permeation reduction factor (PRF) on the order of 100-200. Results of later TTDP laboratory work indicated that a PRF of 300-1000 or more may be achievable.

- The tritium concentration and release estimates for NPR EIS purposes are calculated for an equilibrium cycle following many years of operation. First year concentrations and releases would be from one-tenth to one-half the equilibrium values, and if an equilibrium tritium mission is not used, the values would be further decreased.

Using a qualitative evaluation of the reductions in conservatism used in the EIS, and considering the above factors, the release value for the System 80+PT should have an upper bound limit of less than 15% of the EIS estimate, or approximately 3,000 curies. Again, this should be compared with release estimates from commercial reactors, which is approximately 900 curies.

Impact On The Balance-Of-Plant

Impact to the balance-of-plant would be limited to the possible inclusion of a detritiation facility. Work performed for the TTDP concluded that, based on the expected doses from tritium and the capital cost of a detritiation facility, a detritiation facility was not cost beneficial for the light water NPR (Reference 6). Since this conclusion was reached for a reactor which produced far more tritium, and for which the release estimates were overly conservative, it is expected that a detritiation facility would not be required for the System 80+PT concept. There are therefore no further additional impacts expected in comparison to the System 80+P.

8. Support Facilities

Target Fabrication

Most of the target components are expected to be procured from commercial sources. Under the TTDP, commercial vendors were contracted to provide barrier coated cladding and nickel plated getter materials (Reference 7). While these components can be procured from commercial sources, DOE facilities must be available to perform some fabrication work because certain design features of the targets must be protected. It is anticipated that the lithium blending, pressing, and sintering processes will be required to be performed on a DOE site, and that final assembly of the targets using these components will also be performed on a DOE site.

The technology required to fabricate some of the target components was developed under the TTDP, however some development items will still need to be addressed before the targets could go into nuclear service. The program was successful in procuring aluminum barrier coated stainless steel cladding tubes which meet the requirements for the program, however it was not shown that the cladding tubes could be fabricated in large quantities with consistently acceptable coatings. An NDE technique was developed for evaluating the quality of the coating in the tubes. In addition, nickel coated getter tubes were successfully procured from commercial sources. Since the program was terminated before either of these processes could be nuclear qualified, additional development will be required to qualify the vendors.

End cap welding techniques which maintain the integrity of the barrier coating were also developed during the TTDP. A technique was developed for the initial end cap, but development of the final end cap weld was not completed.

Most of the major technical difficulties concerning target components were addressed as part of the TTDP. Some additional development would be required before production targets could be fabricated. In addition to the above mentioned items, other development needs include determining the optimal pellet pressing methodology.

Once the components are procured, the actual operation of a target fabrication facility is a straight forward process similar to a fuel fabrication facility. A general process flow is outlined in Figure III.G-8. The process is as follows. Lithium aluminate powder is pressed into pellets and sintered. The pellets and inner liner are encapsulated in short getter tubes to form target "pencils". The pencils are then inserted into the target cladding, and the end cap is welded on. Targets are then attached to a base plate and the target assemblies are taken to the fuel bundle assembly area for insertion into the target guide tubes in the fuel assemblies.

Tritium Recovery Facility

Tritium recovery from irradiated targets will require operation in a hot cell to protect operators from the gamma dose of the irradiated stainless steel cladding. The hot cell is required to be reasonably large because of the operations which must be performed and the equipment which must be used to extract the tritium. The cell must accommodate storage of the thirteen foot long target rods and waste materials, the target preparation areas, and the extraction furnaces. The size of the facility is ultimately dependant on the size of the equipment needed for the process design.

A simplified process flow diagram is shown in Figure III.G-9. The target rods are received at the hot cell facility in transport casks. When removed from the casks, they will be stored on racks in the hot cell until required in the preparation area. The preparation area will be vacuum sealed since target pre-puncturing or cutting will occur in the area, and the gases which will escape during pre-puncture need to be recovered. The targets will then be moved to a vacuum furnace where they are heated to extract the tritium from the target structural materials. It is expected that 99.5% of the tritium in the targets will be extracted. Tritium and helium extracted from the target will be vacuum pumped to a storage area. The remaining target structural materials with the residual tritium will be stored or disposed of as radioactive waste.

Most of the physical phenomena associated with the extraction processes were characterized under the TTDP, however development was not completed (Reference 8). A conclusion of the TTDP was that the targets should be pre-punctured before being inserted into the furnaces, either by using a small puncture or by slicing the target in sections. Pre-puncture was preferred to stress rupturing under high temperature since it precluded target material dislocation. The method of pre-puncture to deploy was not determined in the TTDP however, and depends partially on the furnace design parameters desired. If the targets are sliced in half for example, the furnaces could be sized to accept half the length of the target instead of having to be large enough to contain the entire thirteen foot length.

To extract the gases from the target components, vacuum furnaces would be used. The furnaces would elevate the target temperatures, and the vacuum extraction would recover the gases released in the furnace. The vacuum furnaces required for the heating and tritium extraction are expected to be procured commercially. The physical recovery mechanisms investigated were permeation of gas through the target clad, desorbtion of gas from the target components, and conductance of material through the puncture hole. Estimates of the required temperatures and recovery times have been made and are available.

Permeation of tritium through the cladding was determined to be a minor factor to the recovery process because the rate is very low. Conductance of the gas through a small puncture hole was investigated, and it was found that the location of the hole and the size of the hole was not a large factor affecting the time of migration of gas out of the target after the initial depressurization was complete.

Additional development is required on several other recovery mechanisms. These include the desorbtion of tritium from the getter and target pellet. All development work to date has been done using scale size targets and non-radioactive materials (hydrogen and deuterium). While it has been postulated what the differences between the hydrogen and deuterium desorbtion and tritium desorbtion might be, tritium desorbtion from any component has not been demonstrated. Desorbtion from the lithium aluminate has not yet been investigated since this requires radioactive tests.

All desorbtion work under the TTDP was performed using individual components of the target assembly. The simultaneous recovery of tritium from all target components has not been demonstrated. All of the results to date have been extrapolated to obtain expected desorbtion times and efficiencies for tritium, but these extrapolations have significant uncertainties.

Additional development work is required to obtain data for recovery of tritium. The TTDP recommended that pilot scale radioactive tests be completed, and that full scale non-radioactive tests be performed. After this work is complete, the specification of actual extraction plant parameters can be completed.

9. Conclusions

The concept evaluated uses an all-plutonium fueled reactor core for the tritium production mission. The capability is shown to meet the objective tritium production requirement with a single System 80+ reactor unit. Alternative use of an enriched UO_2 core for this mission with a single System 80+ reactor unit is also feasible, although not specifically analyzed.

From the analysis performed to date, it has been concluded that the production of goal quantities of tritium in a reactor core designed to destroy weapons-grade plutonium is feasible and can be considered as an option for meeting tritium requirements. The design of such a core is expected to perform within the operational and safety envelope of a core dedicated to the destruction of plutonium. The only effect to the plutonium destruction mission is that the total number of plutonium atoms destroyed is reduced about 1% less for fuel at the same exposure. The total mass of plutonium denatured in a spiking or power generation mode is the same.

There is a high confidence that the proposed target design will perform well. The mechanical design is flexible to suite changing production requirements. The physical parameters indicate a robust mechanical design which should perform well under all operating conditions. Further development pertaining to target mechanical design should allow design conservatism to be reduced while maintaining adequate safety margins.

A significant amount of target fabrication and tritium extraction technique development was performed under the TTDP. Further development work is required, however the TTDP has demonstrated the technical feasibility of all of the major processes involved. It is not expected that support facility development would inhibit a light water tritium production mission.

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TABLE III.G-1

Summary of System 80+ Plutonium Burner and System 80+ Tritium Production Design Parameters

Design Parameter	Plutonium Burner	Tritium Production
Power Level		
Core	3800 MW(th)	3410 MW(th)
Average Linear Power[1]	17.77 kW/m (5.42 kW/ft)	16.46 Kw/m (5.40 Kw/ft)
Core Dimensions		
Active Core Length	3.81 m (150 in)	3.81 m (150 in)
Equivalent Core Diameter	3.65 m (143.6 in)	3.65 m (143.6 in)
Fuel Assemblies		
Number	241	241
Dimensions	202.7 mm x 202.7 mm (7.98 in x 7.98 in)	202.7 mm x 202.7 mm (7.98 in x 7.98 in)
Array	16 x 16	16 x 16
Fuel Rods per Assembly	224/236	204
Fuel Rods in Core	54,956	49,164
Fuel Rods		
Outside Diameter	9.7 mm (0.382 in)	9.7 mm (0.382 in)
Cladding Thickness	0.64 mm (0.025 in)	0.64 mm (0.025 in)
Fuel Pellet Material	UO_2-PuO_2-Er_2O_3	UO_2-PuO_2-Er_2O_3
Plutonium in MOX	6.9% in HM	7.38% in HM
Erbia in MOX	<2 w/o	0.5 w/o
Cladding Material	Zircaloy-4	Zircaloy-4
Guide Tubes[2]		
Number in Core	1920	7712
Number per Assembly	12/0	32
Outside Diameter	11.2 mm (0.440 in)	11.2 mm (0.440 in)
Thickness	0.91 mm (0.032 in)	0.91 mm (0.032 in)
Material	Zircaloy-4	Zircaloy-4
Burnable Poison Rod Assemblies (BPRA)		
Number BPRAs in Core	241	
BPRs per Assembly	12/0	
BPRs in Core	1920	
BPR Outside Diameter	8.7 mm (0.344 in)	
Cladding Thickness	0.64 mm (0.025 in)	
BPR Absorber	Al_2O_3-B_4C	
BPR Cladding Material	Zircaloy-4	

1 Based on 0.975 average energy deposition fraction in the fuel.

2 Non-structural guide tubes are added to accommodate BPRAs and/or TTAs in all fuel assemblies. These guide tubes should be differentiated from the structural guide tubes in control locations.

TABLE III.G-1 (Continued)

Summary of System 80+ Plutonium Burner and System 80+ Tritium Production Design Parameters

Design Parameter	Plutonium Burner	Tritium Production
Tritium Target Assemblies (TTA)		
Number TTAs in Core		241
Number TTs per Assembly		32
Number TTs in Core		7712
TTA Outside Diameter		8.7 mm (0.344 in)
Cladding Thickness		0.76 mm (0.030 in)
TTA Absorber		$LiAlO_2$
TTA Cladding Material		SS-316
Control Element Assemblies (CEA)		
Number CEAs in Core	101	101
12-element Assemblies	48	48
4-element Assemblies	53	53
CEA Rod OD	20.7 mm (0.816 in)	20.7 mm (0.816 in)
Cladding Thickness	0.89 mm (0.035 in)	0.89 mm (0.035 in)
CEA Absorber	B_4C/Feltmetal and	B_4C/Feltmetal and
Reduced Diam. B_4C	Reduced Diam. B_4C	Cladding Material
	Inconel 625	Inconel 625

TABLE III.G-2

Target Design for the System 80+ Tritium Production Core

Component	O.D. (in)	I.D. (in)	Thickness	Material
Guide Tube	0.440	0.408	0.016	Zircaloy-4
Target Clad (incl. barrier)	0.344	0.284	0.030	SS-316
Barrier	-	-	0.003	Aluminum
Getter	0.267	0.245	0.011	Ni Plated Zirconium
Target Pellet	0.240	0.136	0.052	$LiAlO_2$
Liner	-	-	0.003	Zirconium

ABB
ASEA BROWN BOVERI

FIGURE III.G-1
SYSTEM 80 + PLUTONIUM BURNER FUEL ASSEMBLY LAYOUT

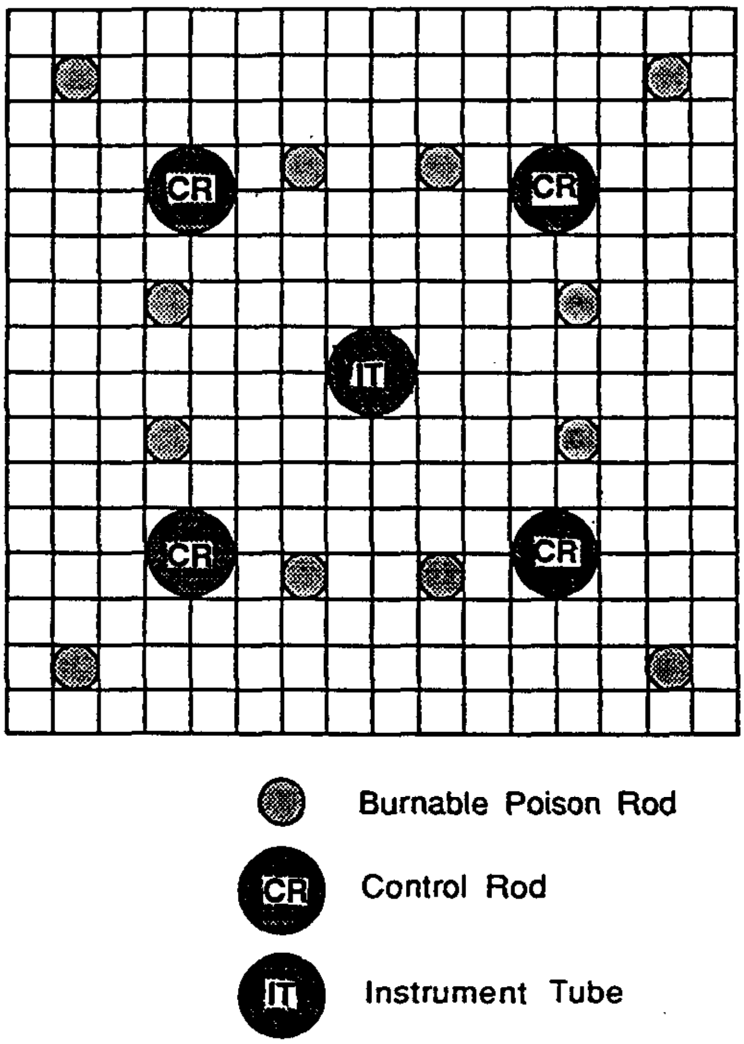


FIGURE III.G-2
SYSTEM 80 + TRITIUM PRODUCTION FUEL ASSEMBLY LAYOUT

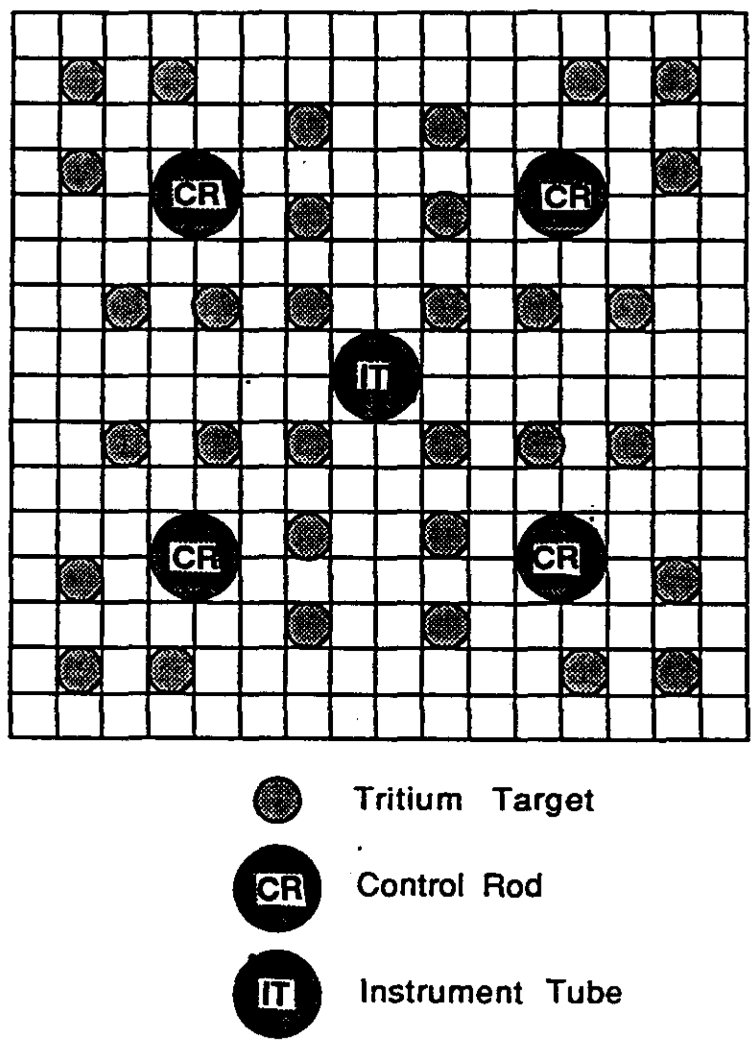
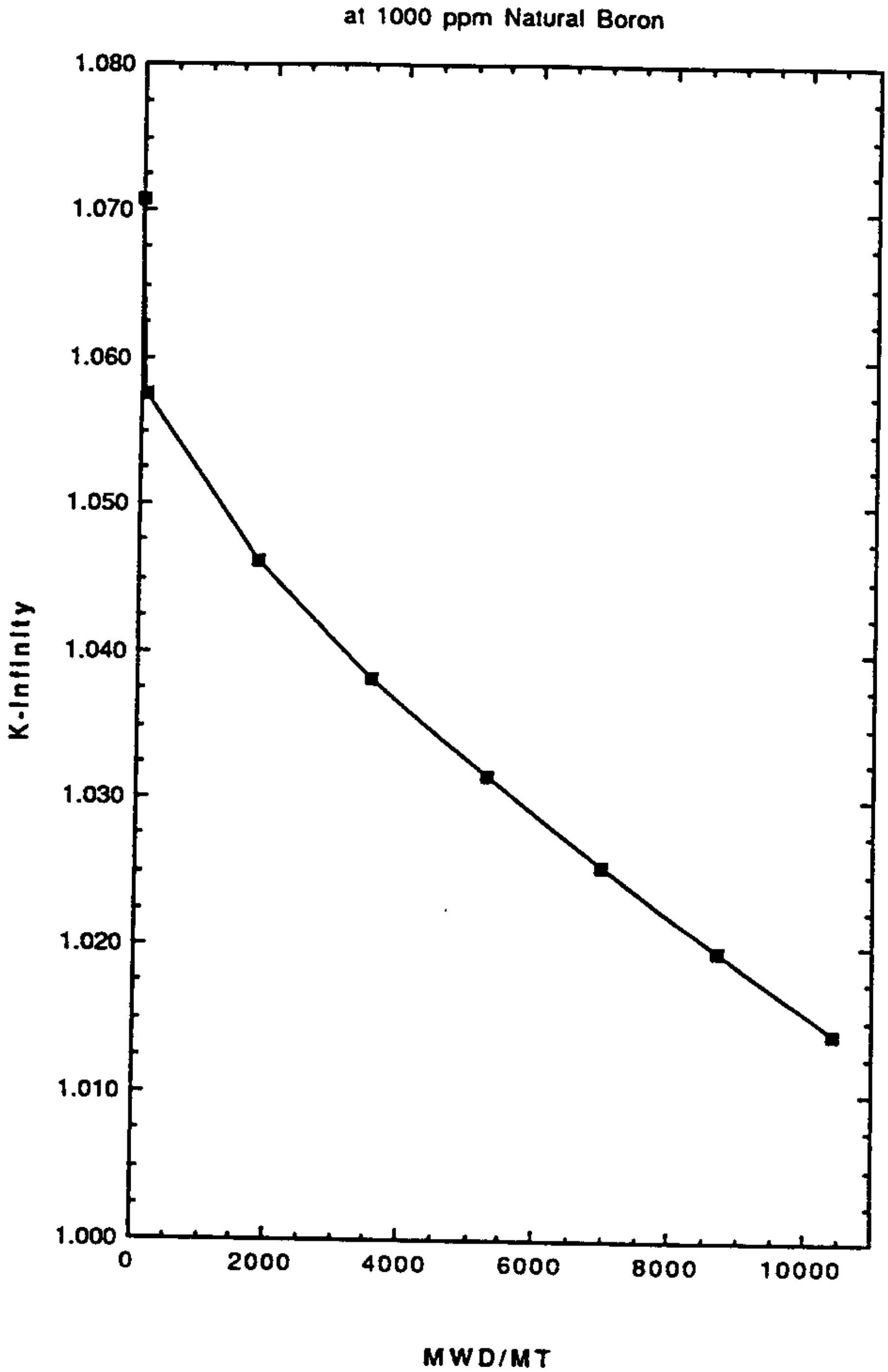


FIGURE III.G-3
REACTIVITY BURNUP CURVE FOR SYSTEM 80 + TRITIUM PRODUCTION FUEL



ASEA BROWN BOVERI

FIGURE II.G-4
MODERATOR TEMPERATURE COEFFICIENT CURVES FOR SYSTEM 80+ PLUTONIUM BURNER AND SYSTEM 80+ TRIVIUM PRODUCTION FUEL

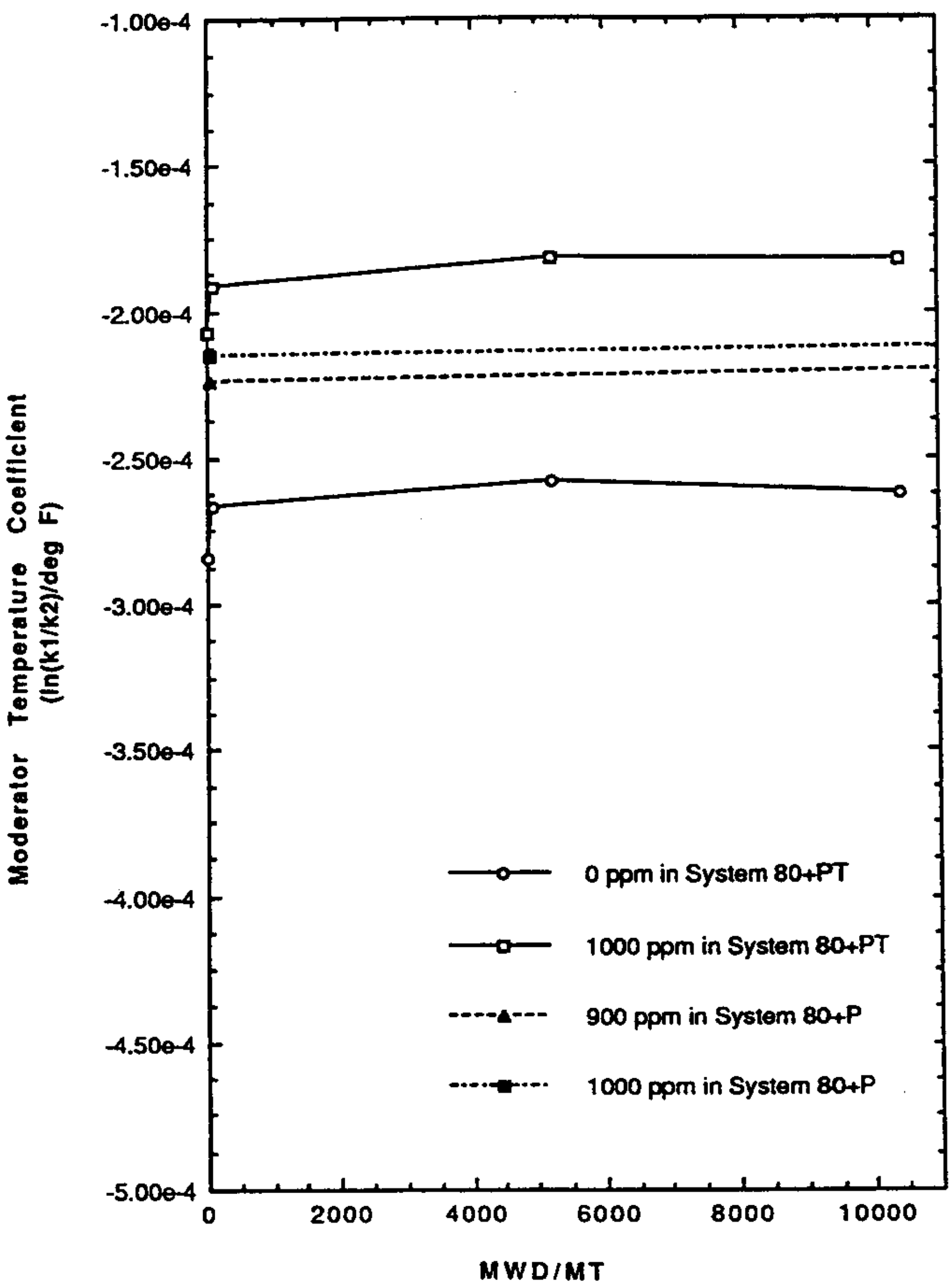


FIGURE III.G-5
DROPPER COEFFICIENT CURVE FOR SYSTEM 80 + TRITIUM PRODUCTION FUEL

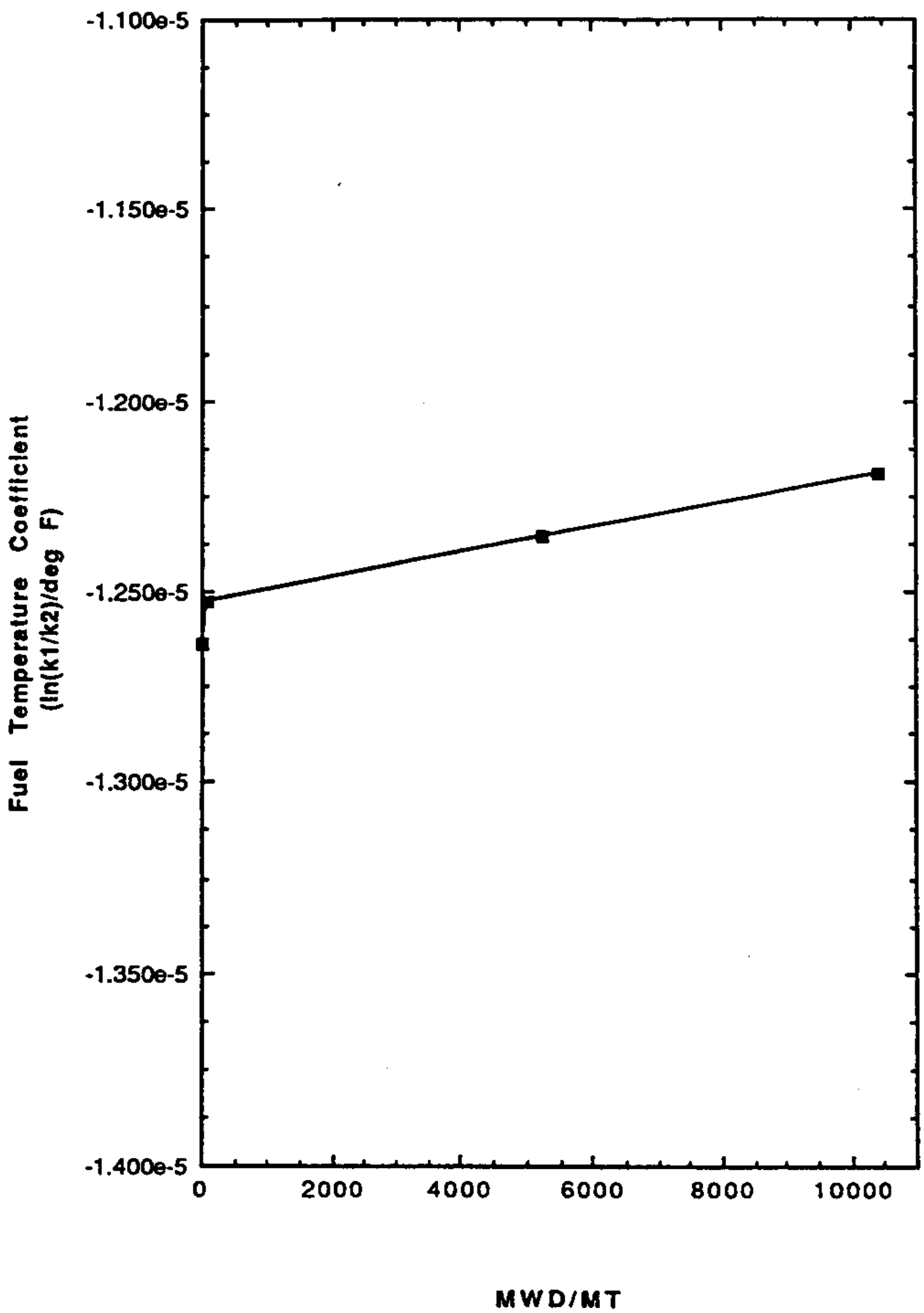


FIGURE III.G-6
PLUTONIUM ISOTOPIC FUNCTIONS OF BURNUP FOR SYSTEM 80 + PLUTINIUM BURNER AND SYSTEM 80 + TRITIUM PRODUCTION FUEL

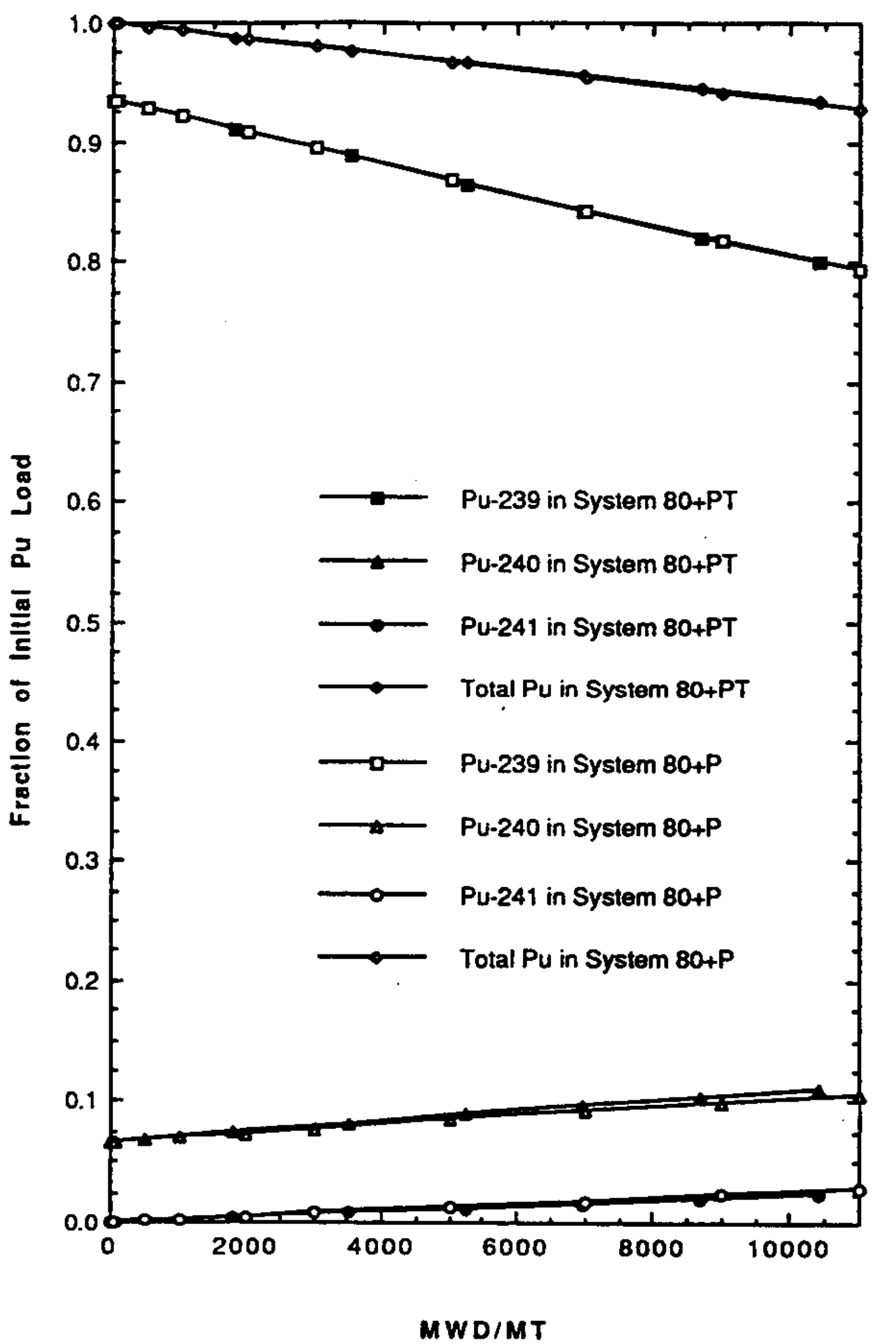
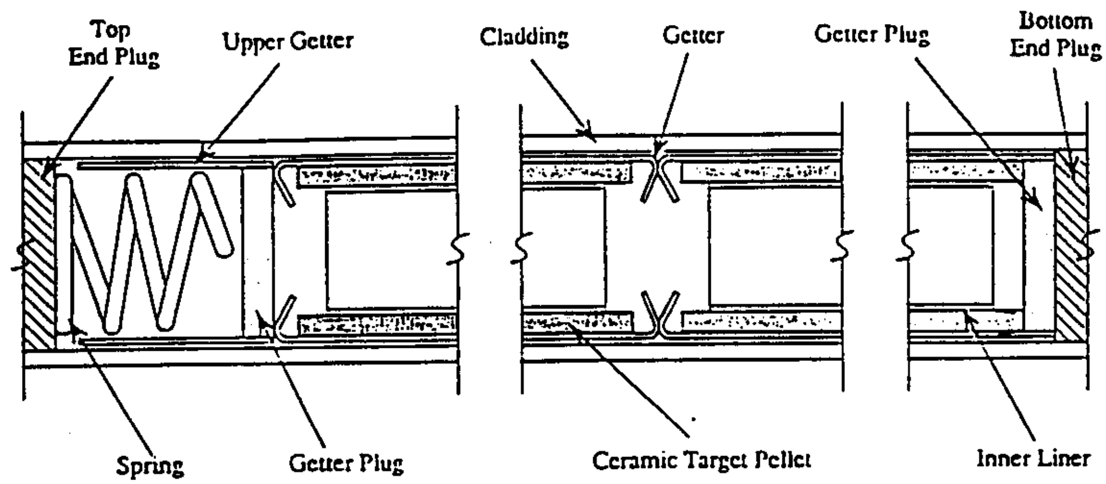
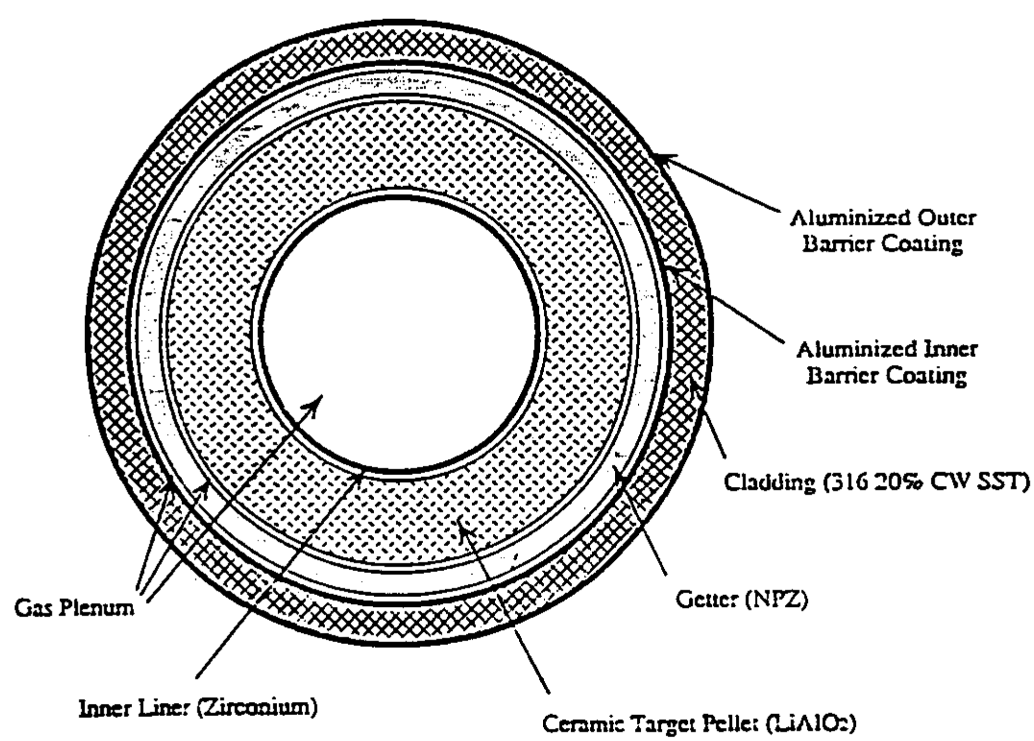


FIGURE III.G-7
SYSTEM 80 + TRITIUM PRODUCTION TARGET CROSS SECTIONAL VIEW





Not to scale

FIGURE III.G-8
TARGET FABRICATION FACILITY PROCESS FLOW DIAGRAM

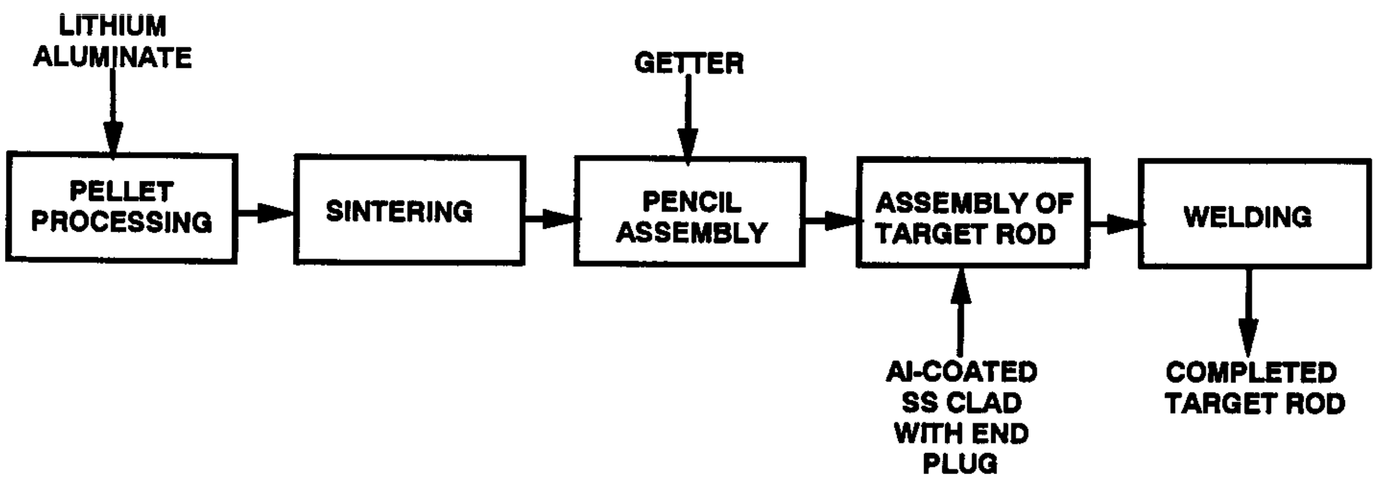
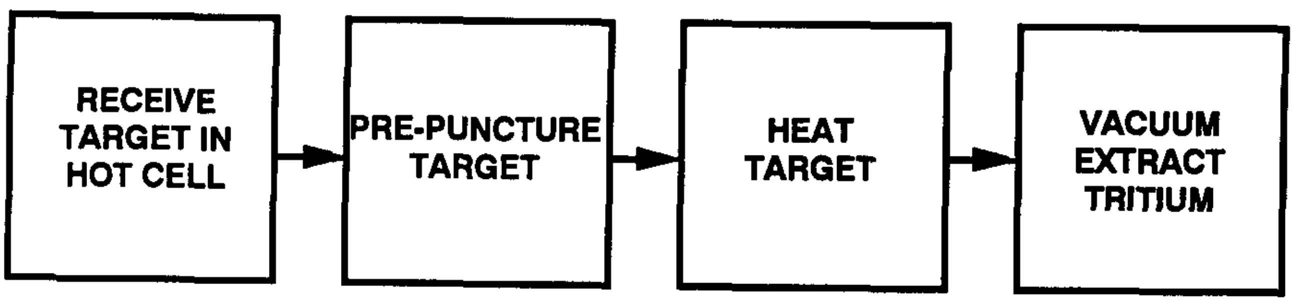


FIGURE III.G-9
TRITIUM RECOVERY FACILITY PROCESS FLOW DIAGRAM



H. FUEL PROCESS FABRICATION FACILITIES

The fabrication process (Figure III.H-1) for System 80+ MOX fuel is similar to the process widely used to fabricate UO_2 fuels for LWRs. The process steps include receiving and storing PuO_2 and depleted UO_2, blending these oxides and forming MOX fuel pellets, encapsulating the pellets into fuel pins, and assembling the pins into fuel bundles. All nuclear materials are initially supplied by DOE as described below. All non-nuclear materials are acquired by the fuel fabricator from commercial sources. The PuO_2 is mixed with depleted UO_2 to a concentration of about seven weight-percent. A burnable poison, Er_2O_3, in the amount of about 2% is added to the mixture. The mixture is pressed into pellets that are sintered in a hydrogen furnace. The sintered pellets are ground to size, inspected, and loaded into pins. The pins are assembled into fuel bundles which are transported to a System 80+ reactor for loading and irradiation.

Nuclear Materials

The process assumes that plutonium will be provided by DOE as PuO_2 to purity and physical properties specifications. The PuO_2 is assumed to be provided in isotopically uniform batches of 100 kg. subdivided into lots of 2 kg. The PuO_2 is further assumed to be provided as needed to meet fabrication schedules such that a maximum 90-day and a minimum 10-day inventory is held in storage at the fabrication plant. The uranium requirements are assumed to be provided by DOE as depleted UF6 in isotopically uniform batches of 10 MT to purity and physical properties specifications.

Receiving

The plutonium will be received and stored in the incoming PuO_2 shipping containers. A system of tags and seals will be used to verify content and composition of the sealed PuO_2 containers. A robotics handling system will receive, verify, identify, weigh, and place the PuO_2 containers in the storage vault. The PuO_2 container gross weight, net weight, serial number, and storage location will be automatically transmitted to the process control computer to maintain material balance.

The uranium will be received as UO_2 from the supplier. The depleted UF6 supplied by DOE will be converted to UO_2, according to specifications, by a commercial fuel supplier. The identity and quantity of UO_2 will be maintained by batch; UO_2 transferred into the fabrication process will be recorded automatically.

Samples will be taken from both the PuO_2 and UO_2 batches at the packaging sites to verify isotopic and chemical compositions and physical properties. Systems of tags and seals will be instituted to maintain lot identity and accountability and to minimize plutonium handling. The characteristics of the UO_2 and PuO_2 will be tightly specified to insure sinterability and high process yields.

The Am-241 content in the surplus plutonium should not be a problem. The shielding and automated handling equipment in the plutonium fabrication line should permit the surplus plutonium to be fabricated as is without requiring the removal of Am-241.

Powder Preparation

The PuO_2 and UO_2 will be withdrawn from storage as needed for processing. The expendable PuO_2 containers will be opened in a glove box. The container identity and tare weight will be recorded in the process control computer. The PuO_2 will be pneumatically transferred to batching hoppers for blending. The UO_2 will be similarly transferred into batching hoppers in the blending glove box. Erbium oxide, Er_2O_3, of the specified purity and physical properties will be transferred into the blending glove through an airlock.

The initial powder operation will prepare a master blend of UO_2 containing approximately 20% PuO_2. The master blend will be thoroughly mixed using blenders and ball mills to insure homogeneity. All powders entering the blending operations, either at this step or subsequent steps, will have been precisely weighed, highly characterized, and controlled by lot.

The master blend will be subsequently diluted with UO_2 to the final composition. Recycled MOX powders from dry scrap recycle operations will be recycled into this blend. Erbium oxide powder will be added also to this blend to meet the final composition specifications. This material will be blended and ball-milled to assure thorough mixing. An organic binder will be blended into the mixed oxides to control density and porosity distribution in the sintered pellets. The blended material will be pressed into large diameter compacts. The compacts will be crushed in a hammer mill and the resulting granules will be sieved to obtain the required feed size for pellet pressing. The oversize and undersize granules will be returned to the compact press feed hopper.

Pellet Pressing

A lubricant will be added to the pellet feed granules to facilitate pellet pressing. The granulated pellet feed will be pressed into pellets. Sample pellets will be taken to verify proper green density. The pellets will then be loaded into sintering boats. Loaded boats will be weighed and automatically transferred to the binder removal furnace.

Binder Removal

The organic binder and lubricant will be removed in a remotely-operated, electrically-heated muffle furnace within the glove-box containment. Boats of pellets will be charged into, and removed from, the furnace through purge chambers to ensure retention of furnace gases and prevent introduction of outside cases into the furnace. Boats of pellets will move through the furnace in a controlled flowing gas atmosphere. Furnace exhaust cases will pass through a gas treatment system to remove vaporized organics and reduce the temperature before the gas is discharged through HEPA filters. Upon exiting the furnace, pellets will be placed in the surge storage area pending transfer to sintering.

Sintering

The pellets will be sintered to about 95% theoretical density in a high-temperature furnace within the containment. Sintering will employ a multizone, electrically-heated furnace containing an argon-hydrogen reducing atmosphere. Boats of pellets will be automatically conveyed into and out of the furnace through purge chambers to prevent introduction of outside eases into the furnace.

Full boats of sintered pellets will be remotely transferred from the sintering furnace to a sampling station. Samples will be taken for chemical and physical analyses. Based on analytical results, the pellets will be rejected, transferred to a vacuum furnace for outgassing, or accepted. Pellets meeting specifications will be unloaded from the sintering boats and stored. Rejected pellets will be crushed, ground, and recycled. The empty sintering boats will be cleaned and reused.

Outgassing

The sintered pellets that require removal of excess cases will be loaded into a batch muffle furnace and treated in a vacuum at temperatures up to 1,000°C. After analytical sampling to verify successful treatment, accepted pellets will be transferred to pellet storage. Rejected pellets will be crushed, ground, and recycled.

Pellet Inspection and Finishing

The sintered pellets will be automatically inspected for surface flaws and gaged for length and diameter. Oversized pellets will be segregated for surface grinding. Undersized and flawed pellets will be crushed and recycled.

Pin Assembly

A mechanized process will load the pin components (fuel pellets and nonfuel components) into cladding tubes and decontaminate the cladding tube ends. A horizontal conveyor will move pellets from station to station. Primary containment will be a sealed housing over the conveyor and over each work station. The pin loading steps are (1) Column Makeup, where pellets are received, stacked into specified columns, and weighed; and (2) Cold Component Makeup, where small nonfuel components are received from stock and manually loaded into the system via an airlock.

Zircaloy cladding tubes will be received from storage with the bottom end caps welded in place. The tubes will be equipped with a loading funnel and identified using a bottom end cap reader. The tubes will be moved on handling trays. Each tube will be inserted through the loading station airlock and the loading funnel will be manually positioned against the loading sleeve.

A loaded pellet magazine will be positioned so that the fuel column is in front of the pin loader. A push rod equipped with force feedback will be used to push the pellets into the cladding. The nonfuel component magazine will then be indexed into place and the nonfuel components will be used to pushed into the cladding. The loading funnel will then be removed from the cladding tube end and replaced with a plug. The pin will be withdrawn onto the transfer conveyor. The pin end will have alpha contamination that will be removed using a dry decontamination system prior to welding to prevent contaminating the weld. At the welding station the end cap will be automatically positioned and welded in place. The loaded fuel pins will be evacuated and back-filled with helium.

After welding pins will be surveyed for alpha contamination and automatically sorted. Pins that are free from contamination will be loaded into a combination helium leak-test chamber and transfer container. The transfer container will be evacuated and the fuel pins will be checked for helium leakage using a mass spectrometer. The fuel pins will then be transferred to the next processing area for etching, autoclaving and inspection. The pins will then be transferred to the fuel assembly room. Contaminated pins will be decontaminated, processed as above, and transferred to the assembly room. Pins that cannot be decontaminated and pins with helium leaks will be disassembled and the pellets will be reloaded into new pins.

Fuel assembly will follow typical System 80+ UO_2 fuel assembly steps. Assembly will be in an enclosed room within the plutonium fabrication plant. After assembly the finished fuel elements will be stored until shipped to the reactor.

Fuel Fabrication Facilities

The MOX fuel fabrication process (Table III.H-1) will require scales; blenders; pulverizing and grinding mills; sieves; pelletizing and compacting presses; furnaces for binder removal, sintering, outgassing, and powder oxidation and reduction; centerless grinders; and inspection, pin loading, and welding equipment. All of this equipment will be contained in glove-box facilities. Other equipment requirements will include hardware preparation, etching, autoclaves, testing and inspection, and fuel assembly; all of which will be contained in rooms within the plutonium fabrication plant. Process space requirements will total about 31,000 sq. ft. for a 50 MT per year MOX plant and 50,000 sq. ft. for a 100 MT/per year plant.

No active Pu fuel fabrication facilities exist in the U.S. However, DOE has a partially completed facility, the Fuel Materials and Examination Facility (FMEF) at Hanford. The FMEF Building meets "Class 1" safety and environmental criteria for storing and processing plutonium. FMEF contains a plutonium fuel fabrication line that has the potential to be expanded to meet the MOX fuel fabrication capacity requirements (estimated at 100 MT MOX per year) for surplus plutonium disposal. The plutonium fabrication line was designed for LMR fuels, but it has most of the needed production capabilities for System 80+ fuels. However, facility modifications and expansion will be needed to meet the required production levels and accommodate the longer System 80+ fuel pins. Additional new glove box facilities will be required to meet the production levels including a separate, duplicate 50 MT/year MOX line covering the process steps from powder blending through pin welding. Other new (non-glovebox) facilities will be needed for autoclaving, cladding preparation, pin inspection, and assembly. Space is available in the FMEF for all of these new facilities and facility modifications.

The reference facility concept is to use the FMEF and plutonium fabrication facilities to develop, demonstrate, and fabricate surplus plutonium into system 80+ MOX fuels. The FMEF is located on the Hanford reservation in an area that provides both security for plutonium processing and storage and ready accessibility to international inspectors and observers that may be required as part of an arms reduction agreement. The FMEF location raises no concerns about intermingling weapons production with arms control activities. The plutonium fabrication line can be activated quickly and economically to serve the development and demonstration needs for lead test assemblies initially and subsequently for fabrication of System 80+ MOX fuel at the required production level. Thus, use of the FMEF and plutonium fabrication facilities can get a surplus plutonium disposal program off to a quick start and reduce overall program costs.

This concept has two elements. First is to provide a safe and secure storage facility for surplus plutonium until the plutonium is fabricated into System 80+ MOX fuel. The FMEF has existing vault storage capacity that can accommodate about 10 MT of plutonium in the form of oxide. In addition, the building has unused space that could be modified for the storage of larger amounts plutonium, if needed. The FMEF building is complete and can probably be activated for Pu storage within a year. Use of the FMEF for storage requires the activation of security facilities and implementation procedures and the training of security personnel. The implementation of security facilities and procedures are expected to be the pacing items.

The second element is the development and fabrication of mixed oxide fuel for System 80+ reactors. The plutonium fabrication line is in place and can readily be adapted to fabricate up to 50 MT per year of mixed oxide fuel assemblies for System 80+ reactors. The plutonium fabrication line can also be adapted to produce plutonium fuels that contain no fertile isotopes, e.g., U-238 or Th-232. The plutonium fabrication line has not been activated and will probably require 1-3 years for modifications and startup. Plutonium test assemblies could be fabricated in the plutonium fabrication line and introduced into System 80+ reactors within three years. Expansion of the plutonium fabrication line to 100 MT MOX per year capacity will require about five years.

If the FMEF and its plutonium fabrication line are not used, then a new MOX fabrication plant will be required. The new plant will use the same processes and have the same equipment and facilities described above and have the a capacity to produce about 8000 target pins per year. All of the pin components except the $LiAlO_2$ pellets will be procured from commercial suppliers. In the target fabrication plant $LiAlO_2$ powders will be pressed into pellets and sintered. The pellets and other components will be assembled into target pins. The pins will be welded shut. Thirty-two target pins will be placed in each bundle. However, siting, design, and construction of the FTFF will probably take 3-6 years in parallel with the reactor facility and cost about $450 million to implement.

Fuel Development and Testing

The plutonium fabrication line has never been activated with plutonium. Thus, the fuel fabrication processes initially can be developed and the process equipment tested using stand-in "cold materials before the facility becomes "hot" with plutonium. Equipment testing and process development will use depleted UO_2 as a stand-in for MOX.

Mixed oxide fuel fabrication will require process development to define the processing steps and parameters in detail. The characteristics of the incoming Er_2O_3, UO_2, and PuO_2 need to be defined and controlled precisely to maximize process yields. The powder preparation and blending processes will require listing to assure complete blending, conformance to purity specifications, and that the MOX powders moil sinterability requirements. Pellet pressing and sintering operations require precise controls to achieve the required density and porosity distribution in the sintered pellets. Process controls will employ feedback mechanisms to maximize process yields and minimize plutonium handling. Process development may lead to variations in the process steps described above to achieve higher yields, reduce plutonium handling, and minimize waste. Potential variations include, among other things, blending initially to the final composition, elimination of binders and lubricants, and alternative powder preparation and activation processes.

Fuel designs for both the disposal of surplus Pu and the production of tritium will employ higher Pu enrichments than have been used in LWRs in the past. This will require a fuel development and irradiation testing program. Test MOX assemblies will be required for irradiation in selected System 80+ reactor(s).

The decay of Pu-241 to Am-241 produces high gamma radiation levels that may require measures to reduce personnel radiation exposures and possibly to limit the content of Am-241 during fuel fabrication. At this time we believe that the shielding and automated handling equipment in the plutonium fabrication line will permit the surplus plutonium to be fabricated without refinement into system 80+ fuel. However, if the Am-241 content is too high, the plutonium can be refined prior to its use. But, compared to reactor grade plutonium, the Am-241 content in the surplus plutonium should be low and not a problem.

Strict Pu accounting, procedures will be implemented to keep track of plutonium inventories in storage and in each process step. Pu inventories will be accurately controlled for safety and safeguards reasons. The isotopic content of each batch of PuO_2 will be known to determine its fissile value and its accountability (total Pu) value.

Nonfertile Fuel Development, Testing and Fabrication

Nonfertile plutonium fuels for System 80 reactors will be comprised of PuO_2 dispersed in an aluminum oxide (Al_2O_3) matrix. The fabrication process will be identical to the MOX process except that Al_2O_3 will be substituted for depleted UO_2. The plutonium fabrication line will be used for blending, pellet preparation, pin loading and welding. Fuel assembly operations will be identical and will use the same FMEF facilities. Al_2O_3 powders will be procured commercially.

Development of the fabrication process will require less than one year. However, fuel testing will require about five years. Irradiation of at least 20 lead test assemblies will be required before full scale implementation of nonfertile fuels is undertaken; most of the test assemblies will be carried to goal exposure. Many of the test assemblies will undergo destructive examination in hot cells following irradiation.

The FMEF and plutonium fabrication line will have sufficient capacity to support full core loadings of the nonfertile fuels.

Waste from MOX Fuel Fabrication

MOX fuel fabrication will generate two types of alpha-contaminated waste: MOX scrap and other (primarily glove box) waste. Glove box waste will include paper, plastics, gloves, metals, HEPA filters, discarded equipment, and other miscellaneous materials. Although MOX fuel not meeting process/specifications will be recycled to the extent practical, an estimated 0.5% of the total MOX fuel fabricated will be disposed of as MOX scrap. The MOX scrap will total about 8000 kg and contain about 500 kg Pu; this scrap will be contaminated with impurities that are too costly to remove. The MOX scrap will have a tap density of about 2 g/cc and occupy about four cubic meters prior to packaging. Glove box and other alpha-contaminated wastes are expected to be about 100 cubic meters per year, or about at 1500 cubic meters in total.

The generation of all alpha-contaminated waste will be minimized to the extent practical. However, because of the low value of the plutonium, scrap recovery operations will be limited to mechanical recycling methods and will not employ liquid and gaseous chemical recovery methods. Volume reduction techniques, such as compaction and incineration, will be used prior to disposal. Surface contamination will be removed where practical. Because of the absence of chemical recovery methods, waste generation in the MOX plant will be higher, about double, that in an equivalent UO_2 fuel plant. The volume of scrap generation in the MOX plant is controllable, but since the waste from MOX fuel fabrication will be comparatively proliferation-resistant the objectives of the surplus plutonium disposal program will be met. Thus, extensive procedures for waste minimization will not be needed.

Non-irradiated plutonium-containing waste from the fabrication plant will be packaged for disposal as transuranic waste in a geologic repository, such as the Waste Isolation Pilot Plant (WIPP) or the commercial spent fuel repository. Acceptance criteria and packaging designs will need to be developed for the various waste forms for both repositories.

The outlook for WIPP is not clear. Test quantities of transuranic defense wastes have been placed recently in the WIPP facility. DOE plans to evaluate transuranic waste disposal in WIPP over a five year period and then decide whether to make

WIPP a permanent repository. EPA has proposed new performance standards for WIPP that cover transuranic waste; these have been published in the Federal Register and are currently in the review and comment phase.

The geologic disposal of MOX fuel fabrication waste will require packaging the material in a suitable waste form for disposal in the repository. If WIPP is not selected as a permanent repository, the commercial spent fuel repository is the alternative. The commercial spent fuel repository is scheduled to begin receiving waste in year 2010. In addition to spent fuel from commercial reactors, the repository will accept defense high level waste and other radioactive wastes that require deep geologic disposal. Presumably this will include waste from surplus weapons plutonium fuel fabrication.

Proliferation Concerns

MOX fuel fabrication for System 80+ reactors offers relatively few opportunities for an outside organization to acquire and divert plutonium. The most vulnerable time is in the initial oxide state prior to blending. The plutonium is pure at this point and requires the least amount of processing facilities to convert it into weapons uses. After mixing, the volume increases substantially and more complex processing is required. MOX fuel fabrication takes place in compact facilities amenable to tight security and close monitoring. The shipment of the fuel assemblies to the reactors provides another point of vulnerability that requires safe and secure procedures.

The storage and fabrication activities for System 80+ fuels provide opportunities to demonstrate that the storage and processing of surplus plutonium within internationally accepted safeguards can be implemented and verified in the US and that these methods are applicable to the storage and processing of surplus plutonium at other locations in the future.

TABLE III.H-1. PROCESS AND SPACE REQUIREMENTS FOR A 50 MT/YEAR MOX FABRICATION PLANT

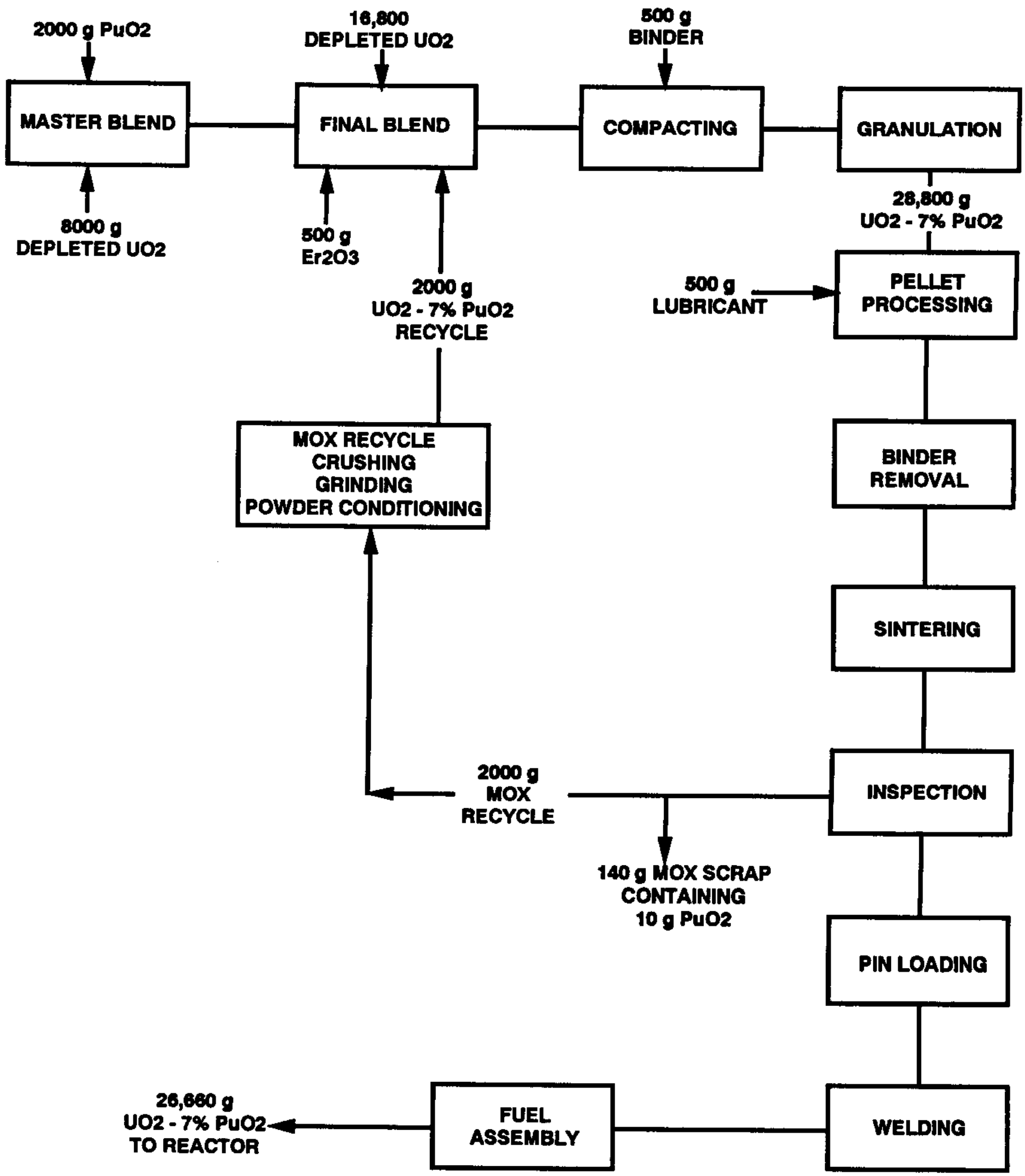
PROCESSOR STEP	PRINCIPLE EQUIPMENT	SPACE SQ.FT	TIME* HOURS	STATIONS
BLENDING				
MASTER BLEND	BLENDER	600	1	2
FINAL BLEND	BLENDER	600	1	2
COMPACTION	PRESS	200	0.5	1
GRANULATION	HAMMER MILL SIEVES BLENDERS	600	0.5	1
TOTAL		**2000**		
PELLET PREPARATION AND RECYCLE				
PELLETIZING	PRESS	1200	1	2
BINDER REMOVAL	FURNACE	600	4	1
SINTERING	FURNACE	1200	16	2
INSPECTION	CENTERLESS GRINDER	400	1	1
	OUTGASSING FURNACE	400	6	1
	INSPECTION	400	2	1
MOX RECYCLE	CRUSHERS	500	2	1
	BALL MILLS	500	4	1
	FURNACES	800	18	1
TOTAL		**6000**		
PIN LOADING	PIN MAKEUP	1000	1	2
	PIN LOADER	1000	1	2
PIN WELDING	WELDER	1000	1	2
PIN LOADING TOTAL		**3000**		
FUEL	LEAK DETECTOR	500	1	1
ASSEMBLY	ETCHING	1000	1	1
	AUTOCLAVING	1000	16	4
	ASSEMBLY	2500	8	1
	STORAGE	2000	4000	400
FUEL ASSEMBLY TOTAL		**7000**		

TABLE III.H-1. PROCESS AND SPACE REQUIREMENTS FOR A 50 MT/YEAR MOX FABRICATION PLANT (CONTINUED)

PROCESSOR STEP	PRINCIPLE EQUIPMENT	SPACE SQ.FT	TIME* HOURS	STATIONS
HARDWARE	INSPECTION	3000	2	1
PREP	WELDING			
SNM STORAGE AND PREP		1000	1	1
PROCESS SPACE TOTAL		22000		
SHIPPING AND RECEIVING		3000		
ANALYTICAL SERVICES		6000		
TOTAL		31000		

* AVERAGE TIME IN PROCESS STEP PER PRODUCTION UNIT, INCLUDING IN-PROGRESS STORAGE

FIGURE III.H-1
REPRESENTATIVE FLOW FOR MOX FABRICATION PROCESS



I. SPENT FUEL PROCESSING DISPOSITION

1. Overview

Following discharge from the reactor the spent MOX assemblies will be disassembled in the reactor storage pool for removal of burnable poison or tritium target pins. The poison pins will be separately packaged in canisters and stored in the pool until shipment to the repository for disposal. The target pins will be separately packaged for transfer to the tritium extraction facility. After removal of the poison and tritium pins, the spent fuel assemblies that will be fabricated. After cooling, the spent fuel assemblies will be packaged into canisters. The canisters will be loaded into casks for shipment to the repository for disposal.

The processes and facilities required for disposal of spent System 80+ MOX fuel assemblies will present no more difficulty than disposal of spent UO_2 assemblies from System 80+ reactors. The fuel form and geometry will be nearly identical. The spent MOX fuel will contain a higher plutonium content, but the isotopic content of the plutonium will be similar to "reactor grade" plutonium contained in spent UO_2 fuels. The fission product concentration and heat generation rate will also be similar.

2. Fuel Handling Operations

After discharge from the reactor, the spent fuel assemblies will be transferred to the reactor storage pool. The spent fuel handling machine will remove the poison and target pins. The poison pins will be packaged in canisters and stored in the storage pool. The target pins will be packaged in canisters that will be loaded into shipping casks for transfer to the tritium extraction facility. The spent fuel assemblies will be transferred to storage locations in the pool. A single control room will remotely control all transfers in the reactor pool; a computer will precisely track and record all transfers.

After cooling, the spent fuel assemblies will be packaged into canisters that will serve as repository disposal packages. The canisters will be loaded into casks for shipment to the repository. All packaging operations will take place under water in the reactor storage pool. The cask will be submerged in the deep pool for loading. Specialized robots will precisely align and load the fuel canisters into compartments within the cask. Before removing the cask from the pool, the cask lid will be positioned, bolted, and sealed in place using specialized robots. Cask shipment to the repository will be by truck, rail, or dedicated train. (The spent fuel operations for the MOX fuels are essentially identical to those that apply to spent UO_2 fuel. There are many types of cask designs and the exact loading procedures will be specific to the casks used. Some cask designs may require dry loading; these facilities would be located adjacent to or above the storage pool.)

The fuel transfer and loading operations must be performed carefully in order to avoid breaking or dropping fuel assemblies that might lead to fission gas release and spent fuel contamination in the pool. Subcritical configurations must be maintained; borated stainless steel baskets and shims will be used as necessary. No consolidation or reracking of the MOX fuel is expected; the pool(s) will have sufficient capacity to store all of the MOX fuel, i.e. about 3500 fuel assemblies. If multiple pools are used, pin disassembly operations, canister loading, and cask loading will all take place in the main pool; the other pools will serve only as storage locations.

The composition and identity of the spent fuel assemblies will be determined and recorded following discharge from the reactor. The identity and location of the fuel assemblies will be verified periodically throughout the period from reactor discharge through emplacement in the repository. After irradiation the MOX fuel will be comparatively invulnerable to diversion because of the high radiation levels, the higher Pu-240 levels, and the need for expensive reprocessing facilities. However, the risks of diversion will not be eliminated and surveillance of spent MOX fuel will be maintained until its irretrievable disposal in a repository.

The disposal of spent MOX fuel assemblies presents no significant increase in safety or environmental risks compared to U02 fuels. The acceptance criteria for spent MOX fuels in the commercial repository are expected to be similar to spent U02 fuels. However, licensing amendments and procedures specific to the MOX fuel are required to cover handling, storage, packaging, shipment, and disposal. The licensing amendments must consider, among other things, the potential impacts of the spent MOX fuels on criticality and operating safety.

J. SECURITY AND SAFEGUARDS

1. Introduction

The Plutonium Burner Reactor Facility (PBRF) is being studied by the Department of Energy (DOE) as a method to dispose of special nuclear materials (SNM) that are being recovered from disassembled weapons. The facility may be built on a "Green Field" DOE site or could adapt existing facilities at either DOE or non-DOE sites. If the PBRF is built at an existing site, the Safeguards and Security System will have to interface with the system that is presently being used at the site. If the PBRF is built as a new facility the Safeguards and Security System will be a completely new design. The PBRF will have to have a security guard force, the size of which will be determined. Presently, the Safeguards and Security Systems at existing sites are undergoing transition and the PBRF will have to meet or exceed the new requirements. Improvements in the Safeguards and Security Systems at the existing sites include more extensive use of computers, communications systems using fiber optic for transmission, improved intrusion detection systems, non-destructive analysis (NDA) systems, software that can be used to analyze the effectiveness of the Safeguards and Security Systems and improved equipment for visual verification and assessment of any suspected intrusion.

The PBRF will either be an NRC licensed facility or will be a DOE facility that is licensable. In either case the facility will meet all the DOE and NRC safeguards and security requirements. The PBRF Safeguards and Security System will be designed to meet the applicable DOE Orders and the applicable portions of Title 10 CFR 70-75. The system will be classed as a graded system and will be designed to meet the requirements as dictated by the category and attractiveness value of the SNM. The SNM is assumed to be Category 1 (Pu greater than 6Kg) and attractiveness "C" for oxide type material. Note: The facility that is used by the DOE to convert the Pu metal to an oxide will have to be designed to handle Category I attractiveness "A" material.

The program for the disposal of excess plutonium includes several options ranging from building a completely new facility, "Green Field", to using existing facility where they fit the program and can meet the new requirements. Each one of the options will require some different aspects of the Safeguards and Security plans. Safeguards and Security Plans would be developed during subsequent phases of this program and will be available to integrate into the facility designs. For the use of existing facilities, the designs would be reviewed to bring them up to meet the present day standards, equipment would be compared with the state-of-the-art, and a vulnerability analysis would be done to ensure the systems can meet the design base threat. The Safeguards and Security Systems become a part of the Master Safeguards and Security Agreement (MSSA) and Site Safeguards and Security Plan (SSSP) which will be in place before facility operation.

The Safeguards and Security System consists of the following three-sub systems:

- The Physical Security Sub-system
- The Materials Control Sub-system

- The Materials Accountability Sub-system

Physical security and material control are closely related and will effectively use much of the same equipment. The equipment used for Intrusion Detection and Assessment will serve both for physical security and material control. Some specialized equipment will be used for only one function, such as explosives detection that will be part of physical security. SNM detection devices and NDA systems will be used for materials control to detect attempts at unauthorized diversion or theft of SNM.

The material accountability sub-system will require special equipment to enable taking physical inventories and provide near-real-time accounting of the SNM from the time of receipt at the PBRF as fuel and target materials until the irradiated material leaves the PBRF site. Additional equipment may be needed to provide for independent inventories and audits of the SNM.

The Safeguards and Security System will be designed to provide defense in-depth that ensures that only authorized personnel, equipment and materials are permitted entry into the PBRF to perform authorized work in order to prevent unauthorized removal and/or diversion of SNM.

To accomplish this primary objective, the Safeguards and Security System that will be designed for the PBRF includes:

- A fenced perimeter with intrusion detection and assessment equipment to prevent unauthorized entry.

- Authorized entry into the area through designated points that contain scanning equipment to prevent the entry or removal of unauthorized materials or equipment.

- An identification system to identify each person entering the facility and determine if the person is authorized to enter.

- Inner barricades or fences to define and prevent unauthorized entry of personnel or materials into special areas that contain vital equipment or SNM. The barricades are also to prevent unauthorized removal of SNM.

- A transship facility where all materials entering or leaving the facility are scanned and/or examined as needed.

Requirements for PBRF

The Safeguards and Security System for PBRF will provide protection against the DOE-defined generic threat for all applicable categories of potential insider adversaries, outside adversaries, or combinations of insiders and outsiders. This includes:

- Physical protection against acts of theft or diversion of SNM; acts that create radiological incidents which might endanger employees or public health and safety; or sabotage for interrupting production activities, which might adversely impact national security.
- Materials/personnel control systems and material accountability systems for protection of SNM from adversarial acts involving the unauthorized movement of SNM within a facility or the unauthorized removal of SNM from a facility.

The Safeguards and Security System will be designed to meet the DOE Orders 5632 Series, 5633 Series and the General Design Criteria 6430.1 the Safeguards and the applicable portions of Title 10 CFR 51-199.

2. Physical Security System

The Physical Security System will be designed to provide securing for the facility, the SNM, and the vital equipment within the PBRF. The SNM at the PBRF is classified as Category I because of the plutonium quantity. The attractiveness is classified as "C" for high grade material in fuel element form or as oxides. If the mission for the PBRF is changed to the tritium producing mode, the Lithium 6 that would be used for targets would be classed as Other Nuclear Materials (ONM) Category IV Attractiveness E. The reportable quantities of Lithium 6 is a kilogram. The tritium that would be produced is classified as Category III when the quantity exceeds 50 grams. The targets will be classified as Category III or IV depending upon the amount of tritium they contain. The reportable quantity for tritium is 1/100 of a gram. The depleted uranium that will be used to dilute the PuO_2 for the mixed oxide type fuel will be classified as Category IV Attractiveness E when the total material transactions for a year exceed 10 metric tons. The reportable quantity for depleted uranium is a kilogram. The security system will be designed as a graded system to meet the requirements for the protection of Category I SNM. The objective of the security system is to protect against:

- Theft of SNM, unauthorized removal of SNM from a Material Access Area (MAA).
- Diversion of SNM, unauthorized placement of SNM within a MAA.
- Radioactive sabotage, any deliberate act against any SNM facility or shipment which could endanger the public health or safety.
- Industrial sabotage; any deliberate act directed against a SNM facility, component or property intended to cause damage, obstruct productivity or interrupt normal functions.

In order to accomplish the objectives, the protection system will meet the following performance requirements:

- Prevent unauthorized access of persons, vehicles and material into protected areas, material access areas (MAA) and vital islands/areas.

- Permit only authorized activities and conditions within protected areas, MAA's and vital islands/areas.

- Permit only authorized placement and movement of SNM within MAA's.

- Permit removal of only authorized and confirmed forms and amounts of SNM from MAA's.

- Provide for authorized access and assure detection of and response to unauthorized penetrations of the protected areas.

- Minimize interference with plant operations and does not create personnel safety problems.

To meet these performance requirements the security systems will have the following design features:

- The security system will use the design philosophy of defense-in-depth. An outer fence around the perimeter of the PBRF will define the limited access area. This fence will prevent inadvertent entry into the limited access area. The first line of defense will be a fence type barrier surrounding the protected areas (PA) of the PBRF. This barrier will be designed to prevent or delay entry into the PA except through the designed entry point. Within the PA there will be other barriers enclosing the MAA's and the vital islands/areas. The inner barriers will also be designed to delay unauthorized entry until the security force can arrive to aid in prevention of intrusion. These inner areas within the PA will also have one controlled entry point to ensure that only authorized personnel enter. The entry points will have equipment to detect unauthorized entry of explosives and weapons.

- A transship facility will be provided. This facility will receive and inspect all materials entering or leaving the protected areas. This will greatly reduce or eliminate vehicular traffic into and out of the protected areas. If trash compaction is used, and with the placement of some fire fighting equipment within the protected areas, there will be virtually no vehicle traffic across the PA fence. This will greatly reduce the potential for introducing contraband into or covert removal of SNM materials from the PBRF. The use of this facility will also greatly reduce the potential to introduce large quantities of explosives. A controlled vehicle entrance will be provided even though the use rate is expected to be very low. A controlled train entrance will be provided if the plan is to eventually remove the spent fuel to permanent repository.

- An intrusion detection system (ID) will be provided to detect any attempt to penetrate a barrier at a location other than the designated entry point. Closed circuit television (CCTV) cameras will be located strategically throughout the PBRF to provide remote visual observation of designated areas. Security forces will also be provided to observe remote TV screens, provide periodic patrols, provide entry control and to respond to intrusion alarms to prevent or delay an unauthorized penetration of the PBRF. The ID and Access Control Systems will be protected by a tamper detection and alarm system.

Security System - General Features

The protection system will be designed with a computer system and multiple connecting loops, such that, no single failure or line break can cause the system to fail. This system will require at least two computers, with one computer designated the backup computer. The backup computer will monitor the condition of the primary computer and will automatically take over control of the system if the primary computer fails. The connections between the computers and control points will be made with fiber optics. Tamper detection and alarm will be provided for the system. The system will be designed so components can be removed from service for test and maintenance without interfering with normal operation. Fiber optics will be used throughout (where applicable) to reduce the potential of EMF interference.

Persons entering the PBRF will use state-of-the-art identification including a biometric system. Identification systems will be located at entrances to vital areas, MAA's and other restrictive zones areas within the protected area. Work in vital areas and MAA's will require the two person rule.

The need for detection and protection against unfriendly aircraft will be investigated later.

Central Alarm Station/Secondary Alarm Station (CAS/SAS)

The central control point for the physical security of the PBRF is the Central Alarm Station (CAS). The CAS will be one terminal point for all information generated by the Safeguards and Security system. The CAS will be located within the PA and will be manned 24 hours a day by members of the security force. The facility will house the safeguards and security computers and will have terminals that will display alarms and outputs from the CCTV system. The Secondary Alarm Station (SAS) required by DOE 5632.2A.12 and 10 CFR 73.55.e will be provided.

Security Force

The security force for the PBRF as required by DOE 5632.7 will be supplied.

Physical Protection of Classified Matter

The protection of classified matter that is generated by or used for the design of the PBRF will be covered by a Security Plan.

Security Power

The normal power for the Safeguards and Security System will be supplied from at least two separate trains with backup from the uninterruptible power supply and/or the emergency diesel generators in order to ensure operation during loss of offsite power or failure of one train. An engine driven generator dedicated to safeguards and security, or an onsite alternate power source will be used during station loss of AC power.

Vital Islands/Areas

Vital Islands or areas where vital equipment such as emergency diesel generators are installed, will be located within the protected area. These islands/areas will have clearly defined perimeters such as building walls or chain link fences. Access to the vital island/areas will be controlled. Access will be restricted to a need-to-be-there basis and the approved access list will be reviewed periodically by the operations and security supervisors and approved by the operations manager.

3. Safeguards

Safeguards is an integrated system of physical protection, material accounting, and material control measures designed to deter, prevent, detect, and respond to unauthorized possession, use or sabotage of SNM. The physical protection aspects were covered to the security section.

Material Control and Accountability (MC&A)

The MC&A system for the PBRF will be designed to comply with the general DOE policy and the specific requirements given in the DOE Orders 5000.3B, 5633 Series, 5632 Series, and 6430.1 Division 13. In addition, the MC&A system will comply with the applicable parts of 10 CFR Parts 70, 73, and 74. This section describes the design requirements that are specific to the PBRF, and will identify the organizational and administrative requirements.

It is assumed for the primary option that the PBRF will be built at an independent site in Mid America. The MC&A system will have to be developed as a stand-alone system for an independent site having a separate reporting identification symbol (RIS). Other options for the PBRF include siting all or part of the facility at existing DOE sites.

Basic Requirements

Responsibility for implementing, managing, and administrating the MC&A program at the PBRF will be divided between operations management and the Site Safeguard and Security Organization. Organizational independence and separations of MC&A functions enable internal controls to be implemented as a defense against the "insider" threat. Custodianship will be assigned to individuals who have no need to have "hands-on" access to nuclear materials on a routine basis. Material handlers will not be allowed to have unrestricted access to the MC&A database.

The MC&A plan will need to identify documented training programs for all personnel who have MC&A functions. The overall responsibility and administration of PBRF personnel training and qualification will be assigned to the MC&A Manager.

Implementation of the MC&A system at the PBRF will comply with the DOE concept of graded safeguards. The most stringent control and accountability measures will be applied to the fully enriched PuO_2 that is received at the facility and is diluted with depleted UO_2 and is fabricated into the unirradiated fuel assemblies. This stringent control will need to be reflected by more frequent physical inventories and stricter control, response, and assessment measures for the unirradiated material than for the irradiated fuel.

Nuclear Material Accounting

A computerized near-real-time accounting system will need to be established to maintain current knowledge of the location and amount of the nuclear materials at the PBRF. The accounting structure will be established around at least six Material Balance Areas (MBA). The Material Receiving and Storage (MRS), Fuel Pin Fabrication (FPF), Fuel Element Assembly (FEA), Fuel Receiving and Storage (FRS), the Reactor Containment Building (RCB), and the Spent Fuel Storage (SFS) each consistute a separate MBA. Accounting will be based on material balances to the point in the process where the fuel pins are assembled into fuel assemblies and a material signature and balance is made. The accounting will then switch to an item control. Each fuel assembly will be uniquely identified item with nuclear materials contents that is based on measurements made at the FEA. Plutonium burnup and isotopic conversion will be determined by reactor production and radioactive decay calculations based on measured reactor operating conditions and updated when actual measured values are obtained by destructive analysis.

The nuclear material tracking system will need to be an integral part of the accounting system at the PBRF and will require maintenance of records of the quantity and location of SNM in the fuel fabrication process and each fuel assembly on a near-real-time basis, making it possible to rapidly assess the status of the nuclear material inventory at all times.

The nuclear material inventory will be verified by performing physical inventories in accordance with DOE orders. Physical inventory of the RCB MBA will be taken during refueling outages at intervals as near 12 months as is practicable. Conducting physical inventories will be facilitated by using automatic bar code or character readers to identify each item in the inventory listing. Advance technologies such as image-based inventory systems will be utilized where practical.

Reconciliation of the physical inventory to the book inventory involves verifying the quantity of each nuclear material and finding that all items are positively identified and in their authorized location. An item inventory difference is not acceptable. All missing items must be investigated and reported in accordance with DOE Order 5000.3B Occurrence Reporting and Processing of Operations Information, and located and returned to an authorized location.

Because nuclear material receipts, shipments, and inventory at the reactor site of the PBRF will ordinarily consist of fuel assemblies that are uniquely identified items whose integrity can be visually verified, and thus are tamper-indicating by themselves, verification measurements will consist of positive identification check and a qualitative measure of the SNM. Confirmatory measurements that may include gross weight determinations and non-destructive assay (NDA) will be performed as part of inventory verification on a statistical sample. A measurement and measurement control program will need to be instituted at the fuel fabrication facility to ensure the measurement quality.

Nuclear Material Control

The nuclear material control program for the PBRF will provide the assurance that the quantity, status and physical location of all nuclear materials in the inventory are known and are protected following the graded safeguards concept. The control program consists of four functional areas: access controls, material surveillance, material containment, and detection/assessment.

- **ACCESS CONTROLS MC&A**

  access controls will include a system for controlling access to nuclear materials, MC&A data, and data acquisition systems which include hardware and software design features that limit access by potential insider adversaries. The nuclear material access controls make use of the physical security system that limits access at several levels, including the limited area perimeter, the protected area (PA), and the material access area (MAA). Administrative controls will limit the number of persons authorized to access the PA and the MAA to those having a need to be there. Access to MC&A data and data handling systems will be controlled by hardware and software designs.

  All nuclear materials at the PBRF will be used and stored within the PA. In addition, all Category I quantities will be used and stored only within the MAA defined by the boundaries in the PBRF. Normal access to the PA and the MAA will be controlled by entry control systems.

- **MATERIAL SURVEILLANCE**

  The nuclear material control program for the PBRF will include measures to detect and deter unauthorized movements or activities involving nuclear materials. The near-real-time material tracking system will provide knowledge of the authorized locations for all nuclear materials in the PBRF. Administrative procedures will require that Category I quantities of nuclear materials in process will be observed by two knowledge persons (the "two-person rule") when not in controlled storage and under electronic surveillance. Procedures will be implemented to ensure only authorized persons have access to the storage locations for SNM and that SNM movements in and out of the location are authorized.

- **MATERIAL CONTAINMENT**

  The nuclear material containment program for the PBRF consists of the establishment of controls on the material access area, storage locations, and in-process areas. These controls will include procedures for defining and limiting authorized movements, processing and storing nuclear materials, assigning custodial responsibilities, implementing the TID program, conducting portal searches for nuclear materials, and conducting other material containment measures to detect and/or prevent unauthorized movements or removals of nuclear materials.

- **DETECTION AND ASSESSMENT**

  The MC&A program will include the requirements for equipment with the capability to detect and assess unauthorized removal of nuclear materials. This part of the MC&A program will consists of:

  - implementation of daily administrative checks in the Category I MBAs to quickly detect and respond to anomalies that could indicate theft or diversion

  - a formal program for using TIDs for detecting unauthorized access to nuclear materials, critical MC&A data bases, or automatic data processing systems

  - use of SNM portal monitors to search and detect unauthorized removals of nuclear materials at the MAA and PA exit points

  - monitoring of all waste streams leaving a material access area to detect nuclear material losses in effluents.

In addition, other detection and assessment measures may be established, as necessary, to protect against FBRF site-specific threatens that are yet to be determined.

K. ENGINEERING AND TECHNICAL DATA

This section presents the various engineering and technical data specifically requested in the attachment to the Plutonium Disposition Study Requirements Document as Figures A1-A10. The data is presented in graphic or tabular form and referenced in the appropriate report sections.

Cost and schedule estimates corresponding to Figures A11-A16 are presented in Section VI; also plant parameters (RD 4.3.1.2) are given in Table II-1 at the end of the Section II previously.

Tables III.K-1 through 10 provide the following data:

K-1: Occupational radiation exposure vs time
K-2: Megawatt thermal installed capacity vs time
K-3: Kg of Pu stockpile reduction vs time
K-4: KW-hr produced vs. time
K-5: Kg of strategic materials required
K-6: Kg feed isotopes to reactor vs. time
K-7: On-site spent fuel storage vs. time
K-8: Overall radwaste generated vs. time
K-9: Kg of each actinide output vs. time
K-10: Kg output of fission products from the complex vs. time

Table III.K-1a

System 80+ Reactor Complex Person-Rem Occupational Radiation Exposure - Annual

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	23	23	23	24	24
2001	200	114	114	97	121
2002	200	205	205	97	194
2003	200	296	296	97	194
2004	100	364	364	97	194
2005		364	364	97	194
2006		364	364	97	194
2007		364	364	97	194
2008		364	364	97	194
2009		364	364	97	194
2010		364	364	97	194
2011		364	364	97	194
2012		364	364	97	194
2013		364	364	97	194
2014		364	364	97	194
2015		341	341	97	194
2016		250	250	97	194
2017		159	159	97	194
2018		68	68	97	194
2019				97	194
2020				97	194
...	...	...	...	...	...
2029				97	194
2030				97	170
2031				97	73
...	...	...	...	...	...
2059				97	
2060				73	

(1) Based on Operations for Disposition of 100 MT Weapons-grade Pu

Table III.K-1b

System 80+ Reactor Complex Person-Rem Occupational Radiation Exposure - Cumulative

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	23	23	23	24	24
2001	223	137	137	121	146
2002	423	341	341	218	340
2003	623	637	637	315	534
2004	723	1,001	1,001	412	728
2005		1,366	1,366	509	922
2006		1,730	1,730	606	1,116
2007		2,094	2,094	703	1,310
2008		2,458	2,458	800	1,504
2009		2,822	2,822	897	1,698
2010		3,186	3,186	994	1,892
2011		3,551	3,551	1,092	2,086
2012		3,915	3,915	1,189	2,280
2013		4,279	4,279	1,286	2,474
2014		4,643	4,643	1,383	2,668
2015		4,985	4,985	1,480	2,862
2016		5,235	5,235	1,577	3,056
2017		5,394	5,394	1,674	3,250
2018		5,463	5,463	1,771	3,444
2019				1,868	3,638
2020				1,965	3,832
...	...	...	...	...	...
2029				2,838	5,579
2030				2,935	5,749
2031				3,032	5,821
...	...	...	...	...	...
2059				5,749	
2060				5,821	

(1) Based on Operations for Disposition of 100 MT Weapons-grade Pu

ABB
ASEA BROWN BOVERI

Table III.K-2a

System 80+ Reactor Complex Installed MWt Thermal Capacity vs. Time – Annual

Year	Pu Spiking (S–0)	Spent Fuel (SF–0)	Pu Destruction (D–0)	Spent Fuel (SF–1)	Spent Fuel (SF–2)
1999					
2000	3,800	3,800	3,800	3,800	3,800
2001		3,800	3,800		3,800
2002		3,800	3,800		
2003		3,800	3,800		
2004					
2005					
2006					
2007					
2008					
2009					
2010					
2011					
2012					
2013					
2014					
2015					
2016					
2017					
2018					
2019					
2020					
...	...	...	...	...	...
2029					
2030					
2031					
...					
2059					
2060					

(1) Core Thermal Capacity Based on All–Plutonium–Reactor Operation

Table III.K-2b

System 80+ Reactor Complex Installed MWt Thermal Capacity vs. Time – Cumulative

Year	Pu Spiking (S–0)	Spent Fuel (SF–0)	Pu Destruction (D–0)	Spent Fuel (SF–1)	Spent Fuel (SF–2)
1999					
2000	3,800	3,800	3,800	3,800	3,800
2001	3,800	7,600	7,600	3,800	7,600
2002	3,800	11,400	11,400	3,800	7,600
2003	3,800	15,200	15,200	3,800	7,600
2004	3,800	15,200	15,200	3,800	7,600
2005	3,800	15,200	15,200	3,800	7,600
2006	3,800	15,200	15,200	3,800	7,600
2007	3,800	15,200	15,200	3,800	7,600
2008	3,800	15,200	15,200	3,800	7,600
2009	3,800	15,200	15,200	3,800	7,600
2010	3,800	15,200	15,200	3,800	7,600
2011	3,800	15,200	15,200	3,800	7,600
2012	3,800	15,200	15,200	3,800	7,600
2013	3,800	15,200	15,200	3,800	7,600
2014	3,800	15,200	15,200	3,800	7,600
2015	3,800	15,200	15,200	3,800	7,600
2016	3,800	15,200	15,200	3,800	7,600
2017	3,800	15,200	15,200	3,800	7,600
2018	3,800	15,200	15,200	3,800	7,600
2019	3,800	15,200	15,200	3,800	7,600
2020	3,800	15,200	15,200	3,800	7,600
...	...	...	...	...	...
2029	3,800	15,200	15,200	3,800	7,600
2030	3,800	15,200	15,200	3,800	7,600
2031	3,800	15,200	15,200	3,800	7,600
...	...	...	...	...	...
2059	3,800	15,200	15,200	3,800	7,600
2060	3,800	15,200	15,200	3,800	7,600

(1) Core Thermal Capacity Based on All–Plutonium–Reactor Operation

Table III.K-3a

System 80+ Reactor Complex Kg of Pu Stockpile Reduction vs. Time - Annual

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999	20,000	6,670	6,670	6,670	6,670
2000	20,000	6,670	6,670	6,670	6,670
2001	20,000	6,670	6,670	6,670	6,670
2002	20,000	6,670	6,670	6,670	6,670
2003	20,000	6,670	6,670	6,670	6,670
2004		6,670	6,670	6,670	6,670
2005		6,670	6,670	6,670	6,670
2006		6,670	6,670	6,670	6,670
2007		6,670	6,670	6,670	6,670
2008		6,670	6,670	6,670	6,670
2009		6,670	6,670	6,670	6,670
2010		6,670	6,670	6,670	6,670
2011		6,670	6,670	6,670	6,670
2012		6,670	6,670	6,670	6,670
2013		6,670	6,670	6,670	6,670
2014					
2015					
2016					
2017					
2018					
2019					
2020					
...	...	...	...	...	...
2029					
2030					
2031					
...	...	...	...	...	...
2059					
2060					

(1) Based on Time of Fuel Fabrication in Reactor Complex

Table III.K-3b

System 80+ Reactor Complex Kg of Pu Stockpile Reduction vs. Time – Cumulative

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999	20,000	6,670	6,670	6,670	6,670
2000	40,000	13,330	13,330	13,330	13,330
2001	60,000	20,000	20,000	20,000	20,000
2002	80,000	26,700	26,700	26,700	26,700
2003	100,000	33,300	33,300	33,300	33,300
2004		40,000	40,000	40,000	40,000
2005		46,700	46,700	46,700	46,700
2006		53,300	53,300	53,300	53,300
2007		60,000	60,000	60,000	60,000
2008		66,700	66,700	66,700	66,700
2009		73,300	73,300	73,300	73,300
2010		80,000	80,000	80,000	80,000
2011		86,700	86,700	86,700	86,700
2012		93,300	93,300	93,300	93,300
2013		100,000	100,000	100,000	100,000
2014					
2015					
2016					
2017					
2018					
2019					
2020					
...	...	...	...	...	...
2029					
2030					
2031					
...	...	...	...	...	...
2059					
2060					

(1) Based on Time of Fuel Fabrication in Reactor Complex

Table III.K–4a

System 80+ Reactor Complex Net KWhr Electric Power Generation – Annual

Year	Pu Spiking (S–0)	Spent Fuel (SF–0)	Pu Destruction (D–0)	Spent Fuel (SF–1)	Spent Fuel (SF–2)
1999					
2000	1.176E+09	2.065E+09	2.065E+09	2.201E+09	2.201E+09
2001	4.702E+09	1.032E+10	1.032E+10	8.802E+09	1.100E+10
2002	4.702E+09	1.858E+10	1.858E+10	8.802E+09	1.760E+10
2003	4.702E+09	2.684E+10	2.684E+10	8.802E+09	1.760E+10
2004	2.351E+09	3.304E+10	3.304E+10	8.802E+09	1.760E+10
2005		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2006		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2007		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2008		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2009		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2010		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2011		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2012		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2013		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2014		3.304E+10	3.304E+10	8.802E+09	1.760E+10
2015		3.097E+10	3.097E+10	8.802E+09	1.760E+10
2016		2.271E+10	2.271E+10	8.802E+09	1.760E+10
2017		1.445E+10	1.445E+10	8.802E+09	1.760E+10
2018		6.195E+09	6.195E+09	8.802E+09	1.760E+10
2019				8.802E+09	1.760E+10
2020				8.802E+09	1.760E+10
...	...	...	...	...	...
2029				8.802E+09	1.760E+10
2030				8.802E+09	1.540E+10
2031				8.802E+09	6.602E+09
...	...	...	...	...	...
2059				8.802E+09	
2060				6.602E+09	

(1) Based on Power Generation for Disposition of 100 MT Weapons–grade Pu, and no Additional Fuel

Table III.K-4b

System 80+ Reactor Complex Net KWhr Electric Power Generation - Cumulative

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	1.176E+09	2.065E+09	2.065E+09	2.201E+09	2.201E+09
2001	5.878E+09	1.239E+10	1.239E+10	1.100E+10	1.320E+10
2002	1.058E+10	3.097E+10	3.097E+10	1.980E+10	3.081E+10
2003	1.528E+10	5.782E+10	5.782E+10	2.861E+10	4.841E+10
2004	1.763E+10	9.085E+10	9.085E+10	3.741E+10	6.602E+10
2005		1.239E+11	1.239E+11	4.621E+10	8.362E+10
2006		1.569E+11	1.569E+11	5.501E+10	1.012E+11
2007		1.900E+11	1.900E+11	6.381E+10	1.188E+11
2008		2.230E+11	2.230E+11	7.262E+10	1.364E+11
2009		2.560E+11	2.560E+11	8.142E+10	1.540E+11
2010		2.891E+11	2.891E+11	9.022E+10	1.716E+11
2011		3.221E+11	3.221E+11	9.902E+10	1.892E+11
2012		3.552E+11	3.552E+11	1.078E+11	2.068E+11
2013		3.882E+11	3.882E+11	1.166E+11	2.245E+11
2014		4.212E+11	4.212E+11	1.254E+11	2.421E+11
2015		4.522E+11	4.522E+11	1.342E+11	2.597E+11
2016		4.749E+11	4.749E+11	1.430E+11	2.773E+11
2017		4.894E+11	4.894E+11	1.518E+11	2.949E+11
2018		4.956E+11	4.956E+11	1.606E+11	3.125E+11
2019				1.694E+11	3.301E+11
2020				1.782E+11	3.477E+11
...	...	...	...	...	...
2029				2.575E+11	5.061E+11
2030				2.663E+11	5.215E+11
2031				2.751E+11	5.281E+11
...	...	...	...	...	...
2059				5.215E+11	
2060				5.281E+11	

(1) Based on Power Generation for Disposition of 100 MT Weapons-grade Pu, and no Additional Fuel

Table III.K-5

System 80+ Reactor Comp[illegible] Kg of Strategic Materials Required per Operating Year

Material	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
Strategic Metals					
Uranium Tails	277,570	92,523		23,131	46,262
Zircaloy	100,704	33,568	33,568	8,392	16,784
Inconel	2,268	856	856	214	428
Erbium	5,520	1,840	4,600	460	920
Other					
Boron-Carbide	1,638	596	756	149	298

(1) Based on Operations for Plutonium Disposition

Table III.K-6a

System 80+ Reactor Complex Core Feed Loading of Pu - Annual

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	6,670	6,670	6,670	6,670	6,670
2001	26,680	6,670	6,670	6,670	6,670
2002	26,680	6,670	6,670	6,670	6,670
2003	26,680	6,670	6,670	6,670	6,670
2004	13,340	6,670	6,670	6,670	6,670
2005		6,670	6,670	6,670	6,670
2006		6,670	6,670	6,670	6,670
2007		6,670	6,670	6,670	6,670
2008		6,670	6,670	6,670	6,670
2009		6,670	6,670	6,670	6,670
2010		6,670	6,670	6,670	6,670
2011		6,670	6,670	6,670	6,670
2012		6,670	6,670	6,670	6,670
2013		6,670	6,670	6,670	6,670
2014		6,670	6,670	6,670	6,670
2015					
2016					
2017					
2018					
2019					
2020					
...	...	...	...	...	...
2029					
2030					
2031					
...	...	...	...	...	...
2059					
2060					

(1) Distributions by Kg of Feed Isotopes are shown in Table III.K-9c through III.K-9e

Table III.K-6b

System 80+ Reactor Complex Core Feed Loading of Pu - Cumulative

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	6,670	6,670	6,670	6,670	6,670
2001	33,330	13,300	13,300	13,300	13,300
2002	60,000	20,000	20,000	20,000	20,000
2003	86,670	26,670	26,670	26,670	26,670
2004	100,000	33,330	33,330	33,330	33,330
2005		40,000	40,000	40,000	40,000
2006		46,670	46,670	46,670	46,670
2007		53,330	53,330	53,330	53,330
2008		60,000	60,000	60,000	60,000
2009		66,670	66,670	66,670	66,670
2010		73,330	73,330	73,330	73,330
2011		80,000	80,000	80,000	80,000
2012		86,670	86,670	86,670	86,670
2013		93,330	93,330	93,330	93,330
2014		100,000	100,000	100,000	100,000
2015					
2016					
2017					
2018					
2019					
2020					
...	...	...	...	...	...
2029					
2030					
2031					
...	...	...	...	...	...
2059					
2060					

(1) Distributions by Kg of Feed Isotopes are shown in Table III.K-9c through III.K-9e

Table III.K-7a

System 80+ Reactor Complex Spent Fuel Assemblies Discharged to On-site Storage - Annual

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	241				
2001	964				
2002	964				
2003	964				
2004	482	241	241		
2005		241	241		
2006		241	241		
2007		241	241		
2008		241	241		
2009		241	241		
2010		241	241		
2011		241	241		
2012		241	241		
2013		241	241		
2014		241	241		
2015		241	241		
2016		241	241		
2017		241	241		241
2018		241	241		241
2019					241
2020					241
...	...	...	...	...	...
2029					241
2030					241
2031					241
...	...	...	...	...	...
2059				241	
2060				241	

(1) One Core contains 241 Fuel Assemblies
(2) Kg of Pu in Discharge Fuel is given in Table III.K-9a

Table III.K-7b

System 80+ Reactor Complex Spent Fuel Assemblies Discharged to On-site Storage - Cumulative

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	241				
2001	1,205				
2002	2,169				
2003	3,133				
2004	3,615	241	241		
2005		482	482		
2006		723	723		
2007		964	964		
2008		1,205	1,205		
2009		1,446	1,446		
2010		1,687	1,687		
2011		1,928	1,928		
2012		2,169	2,169		
2013		2,410	2,410		
2014		2,651	2,651		
2015		2,892	2,892		
2016		3,133	3,133		
2017		3,374	3,374		241
2018		3,615	3,615		482
2019					723
2020					964
...	...	...	...	...	...
2029					3,133
2030					3,374
2031					3,615
...	...	...	...	...	...
2059				3,374	
2060				3,615	

(1) One Core contains 241 Fuel Assemblies
(2) Kg of Pu in Discharge Fuel is given in Table III.K-9b

Table III.K-8a
ANNUAL SOLID WASTE VOLUMES AND ACTIVITIES/REACTOR

WASTE	UNSOLIDIFIED VOLUME		TOTAL ACTIVITY CI (BECQUEREL)
	M^3	FT^3	
Resins	39.22	1386	599. (2.22E+13)
Filters	14.60	516	30.7 (1.14E+12)
Dry Active Waste*			
Non-compactible	1.08	38	1.2E-3 (4.44E+07)
Compactible	220.32	7785	16.2 (5.99E+11)
Boron Evaporator Bottoms	1.14	40	18.7 (6.92E+11)

WASTE	RADIONUCLIDE	% ABUNDANCE
Resins	H-3	0.08
	Co-60	37.5
	Co-58	12.1
	Co-57	0.1
	Mn-54	3.8
	Cs-134	7.5
	Cs-137	9.9
	Sb-122	0.6
	Cr-51	0.08
	Sb-125	0.06
	C-14	0.2
	Fe-55	11.0
	Ni-63	16.7
	Sr-90	0.3
	Cm-242	0.003
	Pu-241	0.07
Filters	C-14	0.2
	Mn-54	2.5
	Co-58	5.6
	Co-60	36.1
	Sr-90	0.002
	Ce-144	0.5
	Nb-95	1.1
	Pu-241	0.05
	Fe-55	48.5
	Ni-63	5.3
	H-3	0.2
	ΣTRU	0.002
	Cm-242	0.002

Table III.K-8a (Continued)
ANNUAL SOLID WASTE VOLUMES AND ACTIVITIES

WASTE	RADIONUCLIDE	% ABUNDANCE
Dry Active Waste (compacted & non-compacted)*	Cr-51	13.15
	Mn-54	5.34
	Co-58	25.05
	Co-60	27.98
	Nb-95	2.23
	Pu-241	0.36
	Fe-55	20.4
	Ni-63	2.94
	Zr-95	1.68
	Cs-137	0.83
	C-14	0.04
Boron Evaporator Bottoms	Co-60	11.1
	Ni-63	15.7
	Cs-134	15.7
	Cs-137	32.5
	Fe-55	25.0

* After treatment in the shredder/dry waste processing unit and cement solidification of the residue, a volume reduction factor of 12.5 would be applied to the initial waste volume.

TABLE III.K-8b

SOURCES, ESTIMATED VOLUMES AND ACTIVITIES OF LIQUID WASTE/REACTOR

Liquid Waste Source	Flow Rate (GPD)	Activity (PCA) [1]	LWMS Collection Tank	Collection Time (Days)	Processing Time (Days)	Discharge Fraction
SHIM BLEED	1830	1.0	Equipment Waste	95 (1)	0.76	0.1
EQUIPMENT DRAINS	250	1.0	(2)	(2)	(2)	(2)
- Reactor Drain Tank						
- Equipment Drain Tank						
CLEAN WASTE	700	0.2	Equipment Waste	30	0.76	0.1
- Reactor Grade Lab Drains						
- Aerated Equipment Drains						
DIRTY WASTE	3200	0.021	Floor Drain Waste	6.7	0.76	1.0
- Containment Sump						
- Plant Floor Drains						
- Fuel Pool Liner Leakage						
- Containment Cooling Condensate						
- Equipment and Area Non-detergent Decon						
STEAM GENERATOR BLOWDOWN	1.0(3)	—	(3)	—	—	0.0
DETERGENT WASTE	(4)	(4)	Laundry and Hot Shower	(4)	(4)	1.0

NOTES:

1. Shim bleed collection time based on 40% of Holdup Tank capacity collection volume.
2. Hydrogenated primary system equipment drain fluids (i.e., Reactor Drain Tank and Equipment Drain Tank inputs) normally recycled directly to the Volume Control Tank.
3. Full blowdown flow processed by Blowdown System and recycled to condensate system demineralizers.
4. Detergent wastes collected and discharged without treatment consistent with NUREG-0017 method.

TABLE III.K-8c

SOURCES, VOLUMES AND FLOW RATES OF STRIPPED GASES FROM THE PRIMARY COOLANT/REACTOR

Waste Gas Source	Flow Rate(a) (SCFM)	Annual Volume(b) (SCF/yr)
PROCESS GAS HEADER (HYDROGENATED)		
CVCS Gas Stripper	.32	145,000
Volume Control Tank	.004	1,624
Equipment Drain Tank		
Reactor Drain Tank (3)	.02	7,759
PROCESS VENT HEADER (AERATED)		
Blowdown Recycle IX (2)	32	112
Purification IX (2)	32	112
Deborating IX	16	56
Lithium Removal IX	16	56
Pre-Holdup IX	16	56
Boric Acid Condensate IX	16	56
Liquid Waste Process IX (6)	96	336
Boric Acid Concentrator	1	2,626
Reactor Makeup Water Tank	22	127,480
Holdup Tank	22	127,480
Boric Acid Tank		
Laundry & Hot Shower Tank (2)	7	17,567
Floor Drain Waste Tank (2)		
Equipment Waste Tank (2)		
Waste Monitor Tank (4)	7	53,325
SG Drain Tank (2)		
Spent Resin Tank (3)	22	1,337
Gas Stripper Vent		
Process Gas Adsorp'n Bed Drain		
Misc. Vents and Drains		

NOTES: (a) Flow rates are estimated maximums, not continuous.

(b) Volumes include anticipated operational occurrences.

Table III.K-9a

System 80+ Reactor Complex Kg of Pu Discharged to On-site Storage - Annual

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	6,399				
2001	25,597				
2002	25,597				
2003	25,597				
2004	12,799	4,859	2,597		
2005		4,859	2,597		
2006		4,859	2,597		
2007		4,859	2,597		
2008		4,859	2,597		
2009		4,859	2,597		
2010		4,859	2,597		
2011		4,859	2,597		
2012		4,859	2,597		
2013		4,859	2,597		
2014		4,859	2,597		
2015		4,859	2,597		
2016		4,859	2,597		
2017		4,859	2,597		4,750
2018		4,859	2,597		4,750
2019					4,750
2020					4,750
...	...	...	...	...	...
2029					4,750
2030					4,750
2031					4,750
...	...	...	...	...	...
2059				4,750	
2060				4,750	

(1) Distribution by Kg of Actinide Isotopes is given in Table III.K-9c through III.K-9e

Table III.K-9b
System 80+ Reactor Complex Kg of Pu Discharged to On-site Storage – Cumulative

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	6,399				
2001	31,997				
2002	57,594				
2003	83,191				
2004	95,990	4,859	2,597		
2005		9,717	5,194		
2006		14,576	7,791		
2007		19,434	10,388		
2008		24,293	12,985		
2009		29,151	15,582		
2010		34,010	18,179		
2011		38,868	20,776		
2012		43,727	23,373		
2013		48,585	25,970		
2014		53,444	28,567		
2015		58,302	31,164		
2016		63,161	33,761		
2017		68,019	36,358		4,750
2018		72,878	38,955		9,500
2019					14,250
2020					19,000
...	...	...	...	...	...
2029					61,750
2030					66,500
2031					71,250
...	...	...	...	...	...
2059				66,500	
2060				71,250	

(1) Distribution by Kg of Actinide Isotopes is given in Table III.K-9c through III.K-9e

Table III.K-9c

Fuel Cycle Actinide Inventory vs. Time (kg per Feed Core)

Plutonium Spiking Alternative

MONTHS	0.0	3.0
EFPD	0.0	39.0
U235	1.8414E+02	1.7633E+02
U236	0.0000E+00	6.4269E-01
U238	9.1879E+04	8.9321E+04
NP237	0.0000E+00	3.3283E-01
PU238	0.0000E+00	5.0580E-03
PU239	6.2367E+03	5.8988E+03
PU240	4.3356E+02	4.9023E+02
PU241	0.0000E+00	1.0242E+01
PU242	0.0000E+00	3.2473E-02
AM241	0.0000E+00	1.9742E-02
AM243	0.0000E+00	1.1688E-04
CM242	0.0000E+00	1.1350E-04
CM244	0.0000E+00	5.0612E-07
TOTAL HM	9.8733E+04	9.5897E+04
TOTAL U	9.2063E+04	8.9498E+04
TOTAL PU	6.6703E+03	6.3993E+03
TOTAL AM+CM	0.0000E+00	1.9973E-02

Table III.K-9d

Fuel Cycle Actinide Inventory vs. Time (Kg per Feed Core)

Spent Fuel Alternative

MONTHS	0.0	12.0	24.0	36.0	48.0
EFPD	0.0	274.0	494.0	822.0	1096.0
U235	1.8414E+02	1.6198E+02	1.4130E+02	1.2165E+02	1.0395E+02
U236	0.0000E+00	5.4947E+00	1.0256E+01	1.4398E+01	1.7788E+01
U238	9.1879E+04	9.1293E+04	9.0701E+04	9.0090E+04	8.9488E+04
NP237	0.0000E+00	1.3388E-01	4.8512E-01	1.0275E+00	1.6726E+00
PU238	0.0000E+00	4.5221E-03	3.6721E-02	1.2340E-01	2.7469E-01
PU239	6.2367E+03	5.3194E+03	4.4842E+03	3.7181E+03	3.0635E+03
PU240	4.3356E+02	6.9670E+02	8.8812E+02	1.0229E+03	1.1021E+03
PU241	0.0000E+00	1.9000E+02	3.6334E+02	5.0725E+02	6.1072E+02
PU242	0.0000E+00	5.5972E+00	2.1992E+01	4.8553E+01	8.1952E+01
AM241	0.0000E+00	3.0387E+00	1.0589E+01	2.0352E+01	2.9674E+01
AM243	0.0000E+00	3.7457E-01	2.8795E+00	9.2802E+00	1.9920E+01
CM242	0.0000E+00	2.6212E-01	1.9930E+00	6.3764E+00	1.3559E+01
CM244	0.0000E+00	2.2039E-02	3.2674E-01	1.5908E+00	4.5152E+00
TOTAL HM	9.8733E+04	9.7676E+04	9.6627E+04	9.5562E+04	9.4538E+04
TOTAL U	9.2063E+04	9.1460E+04	9.0853E+04	9.0226E+04	8.9610E+04
TOTAL PU	6.6703E+03	6.2117E+03	5.7577E+03	5.2969E+03	4.8585E+03
TOTAL AM+CM	0.0000E+00	3.6974E+00	1.5788E+01	3.7599E+01	6.7668E+01

Table III.K-9e

Fuel Cycle Actinide Inventory vs. Time (Kg per Feed Core)

Plutonium Destruction Alternative

MONTHS	0.0	12.0	24.0	36.0	48.0
EFPD	0.0	274.0	494.0	822.0	1096.0
U235	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
U236	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
U238	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
NP237	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
PU238	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
PU239	6.2369E+03	4.7197E+03	3.3308E+03	2.0743E+03	1.0603E+03
PU240	4.3357E+02	6.8860E+02	8.4159E+02	8.9655E+02	8.3968E+02
PU241	0.0000E+00	2.2314E+02	4.1325E+02	5.3665E+02	5.6161E+02
PU242	0.0000E+00	7.5777E+00	3.1340E+01	7.3906E+01	1.3542E+02
AM241	0.0000E+00	3.5292E+00	1.1779E+01	2.0593E+01	2.5187E+01
AM243	0.0000E+00	5.8035E-01	4.6179E+00	1.5516E+01	3.5166E+01
CM242	0.0000E+00	3.4076E-01	2.6743E+00	8.7850E+00	1.9119E+01
CM244	0.0000E+00	3.8910E-02	6.2100E-01	3.1864E+00	9.7576E+00
TOTAL HM	6.6705E+03	5.6435E+03	4.6367E+03	3.6295E+03	2.6862E+03
TOTAL U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL PU	6.6705E+03	5.6390E+03	4.6170E+03	3.5814E+03	2.5970E+03
TOTAL AM+CM	0.0000E+00	4.4892E+00	1.9692E+01	4.8080E+01	8.9230E+01

Table III.K-10a

System 80+ Reactor Complex Kg Output of all Fission Products from the Complex - Annual

Year	Pu Spiking (S-0)	Spent Fuel (SF-0)	Pu Destruction (D-0)	Spent Fuel (SF-1)	Spent Fuel (SF-2)
1999					
2000	661	1,161	1,161	1,238	1,238
2001	2,645	5,807	5,807	4,951	6,189
2002	2,645	10,453	10,453	4,951	9,902
2003	2,645	15,098	15,098	4,951	9,902
2004	1,322	18,582	18,582	4,951	9,902
2005		18,582	18,582	4,951	9,902
2006		18,582	18,582	4,951	9,902
2007		18,582	18,582	4,951	9,902
2008		18,582	18,582	4,951	9,902
2009		18,582	18,582	4,951	9,902
2010		18,582	18,582	4,951	9,902
2011		18,582	18,582	4,951	9,902
2012		18,582	18,582	4,951	9,902
2013		18,582	18,582	4,951	9,902
2014		18,582	18,582	4,951	9,902
2015		17,421	17,421	4,951	9,902
2016		12,775	12,775	4,951	9,902
2017		8,130	8,130	4,951	9,902
2018		3,484	3,484	4,951	9,902
2019				4,951	9,902
2020				4,951	9,902
...	...	...	...	...	...
2029				4,951	9,902
2030				4,951	8,664
2031				4,951	3,713
...	...	...	...	...	...
2059				4,951	
2060				3,713	

(1) Kg Output of High Activity Spent Resins

Table III.K-10b

System 80+ Reactor Complex Kg Output of all Fission Products from the Complex – Cumulative

Year	Pu Spiking (S–0)	Spent Fuel (SF–0)	Pu Destruction (D–0)	Spent Fuel (SF–1)	Spent Fuel (SF–2)
1999					
2000	661	1,161	1,161	1,238	1,238
2001	3,306	6,968	6,968	6,189	7,426
2002	5,951	17,421	17,421	11,139	17,328
2003	8,596	32,519	32,519	16,090	27,229
2004	9,919	51,102	51,102	21,041	37,131
2005		69,684	69,684	25,992	47,033
2006		88,267	88,267	30,943	56,934
2007		106,849	106,849	35,893	66,836
2008		125,432	125,432	40,844	76,737
2009		144,014	144,014	45,795	86,639
2010		162,596	162,596	50,746	96,541
2011		181,179	181,179	55,697	106,442
2012		199,761	199,761	60,647	116,344
2013		218,344	218,344	65,598	126,245
2014		236,926	236,926	70,549	136,147
2015		254,347	254,347	75,500	146,049
2016		267,123	267,123	80,451	155,950
2017		275,253	275,253	85,401	165,852
2018		278,737	278,737	90,352	175,753
2019				95,303	185,655
2020				100,254	195,557
...	...	...	...	...	...
2029				144,813	284,675
2030				149,764	293,339
2031				154,714	297,052
...	...	...	...	...	...
2059				293,339	
2060				297,052	

(1) Kg Output of High Activity Spent Resins

IV. TECHNOLOGY NEEDS

This section describes areas of technology that will need to be further developed based on the study of the application of System 80+ to the Plutonium Burner Reactor Facility. The areas of needed technology involve the Reactor Complex, Fuel Cycle and Tritium Recovery Facility.

A. REACTOR COMPLEX

1. Validation of methods for Nuclear Design and Safety Analysis

Nuclear design and safety analyses for the Pu burning core will need to be performed using NRC approved computer codes. Since the computer codes and methodology have been approved for low enrichment UO_2 fuel cycles, it is necessary to provide verification that the codes are applicable to the design and safety analyses for the plutonium fueled reactor. The code verification would include detailed review of the methodology and data base, testing and benchmark calculations, and demonstration that all design basis and accident analysis models are sufficiently conservative such that uncertainties are bounded by the analysis.

The scope of core design and safety analysis applications subject to verification is summarized below.

Core Design

The System 80+ plutonium burner core design covers the areas of nuclear design, core performance analysis, design basis safety analysis, and beyond design basis events including ATWS and total loss of feedwater. Evaluation is also performed for severe accident phenomena related to the plutonium core design, and assessment of mitigation features for postulated events.

The concept design work in this area will have to develop reference design features including fuel cycle, core physics parameters, and identification of limiting safety related events for reactivity insertion, loss of flow, and other safety-significant occurrences. Detailed design analyses include the following.

Thermal hydraulic analysis

A detailed thermal hydraulic analysis and fuel performance analysis of the reference Pu core design will need to be performed using NRC approved design methodology. The thermal hydraulic performance of the reference Pu burning core requires evaluation for all performance-related and safety-related design bases.

Neutronics/kinetics evaluations

Detailed neutronics evaluation of the reference Pu burning core design includes depletion isotopics, reactivity coefficients, control worths, and power distribution as a function of burnup. Detailed design is based on the NRC approved methods (DIT and ROCS/MC code systems). Analyses are performed for core stability and power distribution control. Reactivity insertion events will be analyzed using NRC approved methodology (including the HERMITE space-time nuclear analysis code) as part of transient analyses.

The nuclear design library would be based on ENDF/B-VI data. Verification of the nuclear design methodology for mixed-oxide applications would be provided based on analyses of critical experiments, mixed-oxide operational cores, and benchmark comparisons to alternate analyses based on Monte Carlo methods (e.g., MCNP code).

Safety analysis and safety margins evaluation

Safety analyses are needed to be performed for limiting occurrences determined for moderate frequency, infrequent, and limiting fault design basis events. The LOCA and non-LOCA safety analyses would be evaluated using NRC approved licensing methodology consistent with the applications for CESSAR-DC which have been reviewed and approved by the NRC staff. The analyses performed would include small-break and large-break LOCA events, steam generator tube rupture, control rod misoperation and inadvertent withdrawal events, control rod ejection events, and steam line break, in order to demonstrate the reactor and safety systems design meets licensing basis safety criteria. The most limiting transients identified from the concept design evaluation would be included. Beyond design basis events including total loss of feedwater and ATWS events would be analyzed. In addition to use of conservative licensing methodology, evaluations based on realistic evaluation models developed for System 80+ would be performed to determine best-estimate performance for safety related occurrences (such as small-break LOCA) in order to demonstrate investment protection.

Severe accident evaluations

Postulated severe accidents will need to be evaluated, including use of deterministic methodologies, and survey of relevant physical and experimental data, in order to assess the significance of the plutonium (e.g., mixed-oxide fuel) core on severe accident phenomenology, and to assess the mitigation features of the System 80+ design for this application. The potential for recriticality following a severe accident and the consequences or mitigation of such recriticality will need to be addressed. If necessary, modifications to the System 80+ features for severe accident mitigation will need to be evaluated. The severe accident evaluation would be used to support containment event tree analysis for level 2 PRA evaluation.

Tritium Production Mission

The neutronic, kinetic, and thermal hydraulic performance of the tritium production reference core design requires verification analogously to the reference Pu burning core. To the extent possible, the performance of the reference tritium production core needs to be evaluated relative to the reference Pu burning core.

2. Probabilistic Risk Analysis (PRA) and Source Term

PRA modeling and evaluation for the System 80+ plutonium burner is a necessary technology area for safety analysis and severe accident evaluations. The PRA development work is necessary for review for applicability and modification of the detailed PRA completed for the reference System 80+ plant design. The reference System 80+ PRA is fully based on the most current PRA methodology and groundrules requirements of the EPRI ALWR Program and has completed detailed review by the NRC.

Level 1 PRA development is required based on the safety analyses and review of effects of the plutonium fuel cycle on the PRA models, fault trees (i.e., system failure modes), operational requirements, and data base. The results of the Level 1 evaluation will identify any potential vulnerabilities and design modifications for the plutonium burner, and would provide a preliminary quantification of the overall core damage frequency and the most significant initiating events.

Further development is required for evaluation of the severe accident vulnerabilities, including the effect on the containment event tree and Level 2 PRA. This work would be directed at identifying any additional vulnerabilities which may result from the plutonium fuel cycle, and the effect on containment performance. Depending on the results of this evaluation, modifications to containment systems may be proposed and evaluated for improved severe accident mitigation.

In order to provide consistency with source term technology for new generation LWRs, it is necessary to evaluate the effect of the plutonium fuel cycle on the source term. The reference System 80+ design (UO_2 fuel cycle) described in CESSAR-DC includes a more realistic source term, which would establish a licensing precedent. This development area will assess the practical implications of a realistic source term for the plutonium fuel cycle and conduct analyses of the effect of the plutonium fuel on the physical source term.

3. Man-machine Interfaces

Due to the unique requirements of the plutonium burning and tritium production missions, development effort is required for man-machine interfaces in the reactor complex. The scope of this effort includes evaluation of man-machine interfaces in the System 80+ reactor plant, electric generation facilities, fuel fabrication facilities, fuel receipt and storage facilities, waste management facilities, and other support and processing facilities consistent with the DOE requirements for the complex. This effort would use as a basis the extensive man-machine interface studies and the methodology for the System 80+ design described in CESSAR-DC, including the results of NRC review, to evaluate the application of man-machine interface design principles within the reactor complex, including the application of the System 80+ Advanced Control Complex (ACC).

4. Plant System Features Specific to Mission

The System 80+ reactor plant was specifically designed to provide a high degree of fuel management flexibility, including the capability to accommodate large loadings of plutonium fuel with relatively minor modifications to the plant systems. The DOE plutonium disposition mission and reactor complex requirements impose a number of requirements which exceed those of anticipated operations for commercial plutonium recycle (i.e., nuclear design and safeguards requirements associated with weapons grade plutonium fuel, tritium production capability, and other facility requirements). These requirements are addressed in the concept evaluation. However, it is evident that plant system optimization is required in areas affected by the DOE requirements for the purpose of assuring that specific systems and processes are designed economically for the plutonium disposition mission and support refining the cost and schedule data for the complex. Part of the scope of this area of technology development would involve systematic evaluation of the mission capability of the complex in order to identify those areas where evaluation would lead to the greatest potential benefit, and to perform selected evaluations on this basis. Areas identified so far include the following:

- Soluble Boron Form - Perform a cost/benefit evaluation for use of enriched vs. natural soluble boron in the reactor coolant.

  Use of enriched soluble boron has been evaluated for commercial operating plants with many potential benefits identified (i.e., significant O&M benefits of reduced waste water generation, improved coolant chemistry control, reduced corrosion potential, elimination of heat tracing, etc.). A major impediment to adopting enriched soluble boron for operating plants has been the added cost and problems associated with flushing operating systems containing natural soluble boron (e.g., tanks, piping systems, spent fuel pool, etc.) with enriched soluble boron while maintaining operation and assuring that natural boron does not "hide-out" in these systems. This problem does not exist for a new plant which would be initially supplied with enriched natural boron. In addition, the plutonium burner design would receive a proportionally greater O&M benefit using enriched soluble boron because of the substantially greater boration requirements for normal operations, shutdown, refueling, and safety systems. Other aspects considered in the cost-benefit evaluation would include capital cost savings, cost of enriched vs. natural soluble boron, and safety systems performance with enriched soluble boron.

- Waste Minimization for Disposal - Develop a plan for ultimate disposal of all identified waste streams, addressing major issues associated with each stream.

  Repository acceptance criteria will need to be considered in developing the reactor complex waste disposal program. Current and anticipated repository constraints would be reviewed and evaluated for impact on the reactor complex activities. Related fuel fabrication development is addressed in Section B below.

- Storage Requirements - Develop and optimize the reactor complex radioactive waste storage requirements based on the identified waste stream quantities and the anticipated capability to ship the waste offsite for ultimate disposal.

- Irradiation Effects of Plutonium Fuel - Comprehensively evaluate neutron fluence and gamma heating rates for the proposed fuel cycles. Evaluate additional materials restrictions or component modifications to extend design life margins and improve performance of reactor components.

5. Safeguards and Security

Safeguards and security is an important development aspect of the plutonium burner reactor facility. To date, plutonium bearing fuels have been manufactured in small quantities and commercial reactors have operated with relatively few mixed oxide fuel assemblies in order to develop operational data. Presently there are no mixed oxide fabrication plants operating in the U.S. In order for the safeguards and security to be factored into the facility concepts there needs to be in place a Physical Security Plan and a Material Control and Accountability Plan. These plans will establish the requirements that must be met by the plant design. These requirements are obtained from the DOE Orders such as the 5632 series and the 5633 series, the General Design Manual 6430.1A and the applicable portions of Title 10 of the Code of Federal Regulations.

By having the safeguards and security documents in place as the concepts for the complex develop, the designers can include these requirements into all facility plans. As the design progresses vulnerability analyses will be run to determine if the facility meets the requirements and can withstand the design bases threat as defined by the DOE. For options that use existing facilities a vulnerability analysis may be performed to determine the areas of the facilities that will need to be upgraded to comply with the DOE requirements.

Recommended development work includes preparation of Preliminary Physical Security and Material Control and Accountability plans, and a vulnerability review for the proposed facilities. It is also recommended that the DOE ongoing program for development of sensors and instrumentation for the Pantex facility be continued and extended to include the needs of the plutonium burner reactor facility.

B. FUEL CYCLE

1. Fuel Demonstration

Principal activities and schedule for a fuel demonstration program for the fuel to be employed in the System 80+ plutonium burning core will need development consistent with the DOE requirements. A mixed-oxide fuel type containing erbium burnable poison (i.e., UO_2-PuO_2-Er_2O_3) is expected based on the conceptual analyses completed. The fuel demonstration program development will identify the extent and nature of the irradiation tests and follow-on post irradiation examinations. Candidate operating reactors for irradiation of the demonstration assemblies will be identified.

A detailed cost and schedule for the recommended fuel demonstration program would also be developed.

If the non-fertile fuel concept is pursued, the fuel development and irradiation demonstration will require a substantially more extensive program than for MOX fuel. Non-fertile fuel is not proven technology for commercial reactors. Therefore, a large fuel development program is required as discussed in the next section.

2. Front-end Process Development

The detailed processing needs for converting the plutonium in each category of material into an acceptable feed material for System 80+ must be developed. The scope of work includes detailed description of the categories of potential surplus plutonium, based on chemical and isotopic content and physical forms, e.g., metal or oxide, of the plutonium, including the estimated quantities of plutonium available in each category. Specifications will be developed for the receipt of plutonium oxide.

The process assumes that plutonium will be provided by DOE as PuO_2 to purity and physical properties specifications. In addition, the uranium requirements are assumed to be provided by DOE as depleted UF_6 in conformance with the purity and physical property specification.

Emphasis will be placed on defining processing and facility requirements, waste generation, and costs. Techniques to minimize processing will be evaluated including chemical and isotopic considerations. The output of this development activity will be a report specifying the plutonium and the quantities needed and a description of the feasible processing requirements for DOE converting the plutonium into acceptable feed material for fabrication into System 80+ fuel.

The activity will also further develop the Fuel Fabrication Facility sizing requirements. An advanced pre-conceptual design for the MOX plant will be developed which will include mass balance, equipment description, equipment sizes, weights, glovebox and process space, other auxiliary facilities, and a budget cost/schedule estimate.

3. MOX Fuel Development and Testing

Mixed oxide fuel fabrication will require process development to define the processing steps and parameters in detail. The characteristics of the incoming Er_2O_3, UO_2, and PuO_2 need to be defined and controlled precisely to maximize process yields. The powder preparation and blending processes will require testing to assure complete blending, conformance to purity specifications, and that the MOX powders meet sinterability requirements. Pellet pressing and sintering operations require precise controls to achieve the required density and porosity distribution in the sintered pellets. Process controls will employ feedback mechanisms to maximize process yields and minimize plutonium handling. Process development may lead to variations in the process steps described above to achieve higher yields, reduce plutonium handling, and minimize waste. Potential variations include, among other things, blending initially to the final composition, elimination of binders and lubricants, and alternative powder preparation and activation processes.

Fuel designs for both the disposal of surplus Pu and the production of tritium will employ higher Pu enrichments than have been used in LWRs in the past. This will require a fuel development and irradiation testing program. Test MOX assemblies will be required for irradiation in selected ABB-CE reactors.

The decay of Pu-241 to Am-241 produces high gamma radiation levels that may require measures to reduce personnel radiation exposures and possible to limit the content of Am-241 during fuel fabrication. At this time we believe that the shielding and special handling equipment in the plutonium fabrication line will permit the surplus plutonium to be fabricated without refinement into System 80+ fuel. However, if the Am-241 content is too high, the plutonium can be refined prior to its use. Compared to reactor grade plutonium, the Am-241 content in the surplus plutonium should be low and relatively inconsequential.

Strict Pu accounting procedures will be developed to keep track of plutonium inventories in storage and in each process step. Pu inventories will be accurately controlled for safety and safeguards reasons. The isotopic content of each batch of PuO_2 will be known to determine its fissile value and its accountability (total Pu) value.

4. Nonfertile Fuel Development, Testing, and Fabrication

Nonfertile plutonium fuels for System 80+ reactors may be comprised of PuO_2 dispersed in an aluminum oxide (Al_2O_3) matrix. The fabrication process will be identical to the MOX process except that Al_2O_3 will be substituted for UO_2 tails. The plutonium fabrication line will be used for blending, pellet preparation, pin loading and welding. Fuel assembly operations will also be identical and will use the same facilities. Al_2O_3 powders will be procured commercially.

Development of the fabrication process will be similar to that for MOX fuel. However, fuel development and testing requirements will be substantially greater. It is estimated that test irradiation of at least 20 lead test assemblies will be required before full scale implementation of non-fertile fuels is undertaken; most of the test assemblies will be carried to full discharge exposure. Many of the test assemblies will undergo destructive examination in hot cells following irradiation.

5. Waste Streams/MOX Fuel Fabrication

Development is required to identify and characterize waste streams generated in the fabrication process in order to describe and evaluate options for minimizing the generation of radioactive and hazardous wastes. MOX fuel fabrication will generate two types of alpha-contaminated waste: MOX scrap and other waste. Other waste will include paper, plastics, gloves, metals, HEPA filters, discarded equipment, and miscellaneous materials.

Options for minimizing the generation of all alpha-contaminated waste will need evaluation. However, because of the low value of the plutonium, scrap recovery options will be limited. Volume reduction techniques, such as compaction and incineration, and surface decontamination would be evaluated. The volume of scrap generation in the MOX plant is controllable, but since the waste the MOX fuel fabrication will be comparatively proliferation-resistance the objectives of the surplus plutonium disposal program may be met without extensive procedures for waste minimization. Some scrap will be contaminated with impurities that are too costly to remove. Waste minimization options will also include maximizing processing yields and maximizing MOX scrap recycle.

Non-irradiated plutonium containing waste from the fabrication plant will require disposal as transuranic waste in a geologic repository, such as the Waste Isolation Pilot Plant (WIPP) or the commercial spent fuel repository. Repository acceptance criteria and packaging requirements will be evaluated as elements of cost-effective waste minimization.

The waste minimization options include minimizing the front end processing of surplus plutonium. For example, the weapons plutonium could be processed to remove the Am-241 and other impurities. However, with shielding and automated handling equipment, the plutonium fabrication line could permit the surplus plutonium to be fabricated without requiring the removal of Am-241. Other options for minimizing the front end processing will be evaluated, such as blending, process modifications, design modifications, and PuO_2 treatment.

6. Spent Fuel Handling, Transportation and Storage Including Criticality/Safety Studies

A combined development activity for the Reactor Complex and Fuel Cycle is required to develop spent fuel transport and storage designs for the preferred deployment options, including detailed description of activities associated with transport of spent/fuel from the reactor vessel to the storage location in the fuel racks.

A related development activity is the design of the fuel storage management, including the auxiliary fuel pools for the single System 80+ reactor plant options, additional design information for the fuel handling activities and the fuel buildings, criticality studies to confirm the fuel storage scheme, and development of fuel building sizing information.

Confirmatory analysis is required to ensure that the overall fuel pool design will provide sufficient shielding to comply with NRC Radiation Protection Guidelines.

C. TRITIUM RECOVERY FACILITY

The technology required to fabricate some of the target components was developed under the Tritium Target Development Program (TTDP), however some development items will still need to be addressed before the targets could go into nuclear service. The program was successful in procuring aluminum barrier coated stainless steel cladding tubes which meet the requirements for the program, however it was not shown that the cladding tubes could be fabricated in large quantities with consistently acceptable coatings. An NDE technique was developed for evaluating the quality of the coating in the tubes. In addition, nickel coated getter tubes were successfully procured from commercial sources. Since the program was terminated before either of these processes could be nuclear qualified, additional development will be required to qualify the vendors.

End cap welding techniques which maintain the integrity of the barrier coating were also developed during the TTDP. A technique was developed for the initial end cap, but development of the final end cap weld was not completed.

Most of the major technical difficulties concerning target components were addressed as part of the TTDP. Some additional development would be required before production targets could be fabricated. In addition to the above mentioned items, other development needs include determining the optimal pellet pressing methodology.

Most of the physical phenomena associated with the extraction processes were characterized under the TTDP, however development was not completed. The method of pre-puncture to deploy was not determined in the TTDP however, and depends partially on the furnace design parameters desired. If the targets are sliced in half for example, the furnaces could be sized to accept half the length of the target instead of having to be large enough to contain the entire thirteen foot length.

Additional development is required on several other recovery mechanisms. These include the desorbtion of tritium from the getter and target pellet. All development work to date has been done using scale size targets and non-radioactive materials (hydrogen and deuterium). While it has been postulated what the differences between the hydrogen and deuterium desorbtion and tritium desorbtion might be, tritium desorbtion from any component has not been demonstrated. Desorbtion from the lithium aluminate has not yet been investigated since this requires radioactive tests.

All desorbtion work under the TTDP was performed using individual components of the target assembly. The simultaneous recovery of tritium from all target components has not been demonstrated. All of the results to date have been extrapolated to obtain expected desorbtion times and efficiencies for tritium, but these extrapolations have significant uncertainties.

Additional development work is required to obtain data for recovery of tritium. The TTDP recommended that pilot scale radioactive tests be completed, and that full scale non-radioactive tests be performed. After this work is complete, the specification of actual extraction plant parameters can be completed.

V. REGULATORY CONSIDERATIONS

A. LICENSING

The policy embodied in the System 80+ plutonium-disposition reactor is that safety and protection of the environment have the highest priority in accomplishing the mission of designing, constructing, and operating the reactor and associated support facilities. The reactor facility will be designed, constructed, and operated in compliance with all applicable Federal, State, and local statues and regulations. Full compliance with DOE Order 5400.1, "General Environmental Protection Program," other relevant DOE Orders, and Executive Orders for the protection of the environment will be invoked for the facility extending from preliminary design through decommissioning. Existing environmental protection regulations will be considered a minimum objective for the System 80+ ALWR plutonium-disposition facility; the detailed design will be planned, designed, and constructed with adequate margins so that anticipated future environmental regulations and standards can be accommodated.

The C-E System 80 standard design was developed during the early 1970's, and received NRC Final Design Approval in December, 1983. Three standard design System 80 plants, constructed at Palo Verde, began commercial operation between January 1986 and January 1988. Four additional System 80 plants are currently under construction at the Yonggwang and Ulchin sites in Korea.

System 80+ is one of two Advanced Light Water Reactor standard design plants leading the review process for design certification by the NRC. Once certified, the System 80+ standard plant can be referenced without concern for further NRC review, or licensing delays initiated through intervenor hearings. One-step licensing by an applicant under 10 CFR 52, and compliance with ITAAC [Inspection, Test, Analysis and Acceptance Criteria] for confirmatory items identified in the CESSAR-DC Final Safety Analysis Report are the only remaining regulatory actions.

The ability to utilize both uranium and plutonium has been a basic design tenant incorporated into the System 80 Standard Plant. That facility has remained inherent in the evolutionary System 80+ advanced light water reactor currently being licensed by ABB-CE. Certification of the System 80+ Standard Plant design is currently a priority activity within ABB-CE, and is supported at the highest levels of the NRC. System 80+ is in the final stages of licensing with the NRC as part of the DOE-sponsored ALWR certification program, with no technical issues remaining to be resolved. A draft Safety Evaluation Report for the System 80+ standard design was issued by the NRC in September 1992; staff issuance of the System 80+ Final Safety Evaluation Report is expected in 1994.

GESMO, the Generic Environmental Statement on Mixed Oxide fuel, was in the final draft stages of review and all technical issues had been adequately addressed, when a government policy directive canceled the option for spent fuel reprocessing, utilization of open-cycle plutonium fuel, or mixed-oxide cores. Based on the review and evaluation of plutonium utilization at that time, the NRC found no objection to the use of mixed-oxide fuel. Therefore, NRC approval of a plutonium-disposition facility at a single site may be facilitated.

Utilizing the System 80+ ALWR to burn mixed oxide fuel provides an optimum solution to the need for transmuting the growing stockpile of weapons-grade plutonium while having the ability to produce electricity for the commercial market. A further benefit of the System 80+ ALWR design is the capability to produce tritium. Transmuting weapons-grade plutonium would involve new issues with the use of mixed-oxide plutonium fuel fabrication, use of mixed-oxide fuel in the reactor, tritium target manufacture and use of tritium targets, tritium target extraction, and plutonium safeguards.

B. UTILIZATION PERMITS

During the initial phase of this project, a plan will be developed to identify and document the commitments, assumptions, and bases on which the permitting schedule is founded. All applicable federal and state permits, and the lead time and required schedule for each permit, to operate the System 80+ ALWR as a plutonium-disposition facility will be identified. Special permit conditions applicable to the plutonium-disposition facility will be documented, on a schedule that will permit application, review and approval by controlling authorities, and implementation of permit conditions consistent with the dates needed to support plant construction and operation. The plan will be developed in cooperation with the DOE to take advantage of review activities performed by the NRC while defining the role of the NRC and DOE in reviewing the plutonium-disposition plant design. The plan will define all permitting, both federal and state, for the System 80+ ALWR-based complex for the missions of plutonium disposition, electric power generation, and tritium production. Compliance with safety and environmental requirements will be demonstrated, as will licensability under NRC regulations.

Issuance of some permits will be subordinate to other permits controlled through a hierarchy of federal, state and local regulations. For example, maintenance of air and water quality will first be under the jurisdiction of National Environmental Policy Act [NEPA] and DOE, followed by state and local requirements. Also, permitting for the Tritium Recovery Facility will be governed by DOE regulations, while licensing the System 80+ ALWR will be regulated to NRC requirements; permits for utilization of mixed-oxide fuel will involve both NRC and DOE review and approval. All activities involving the System 80+ plutonium-disposition facility will meet all applicable NEPA standards as required in DOE Order 5440.1D. Later phases of the program will address the schedule and need for permits involving refueling, retrofit, decommissioning, dismantling, and disposal or storage of plant components.

The plant permitting schedule will give consideration to the need for the complex interaction of archeological preservation, air and water quality, plant construction, road or waterway construction to support heavy equipment transportation, source term radiological impact, and environmental impact. A living data base will be developed to identify all required permits and licenses to ensure that the hierarchy and schedule for such are cross-indexed and current for each phase of the System 80+ plutonium-disposition facility. A preliminary listing of typical licenses and permits required for the facility is given in Table V-1.

C. SITING AND OTHER CONCERNS

Details of the plant permitting and their relation to a designated site will be developed after a specific site has been selected. Site permitting considerations will include impact on air and water quality, water rights, land use, solid and hazardous waste, wildlife impact, timber harvest, road and waterways, archaeological or historic preservation, Federal Aviation Administration, and native American Indian rights. Best available radionuclide control technology will be implemented to assure compliance with the National Emission Standards for Hazardous Air Pollutants and the National Pollutant Discharge Elimination System.

C. ENVIRONMENTAL

Compliance with the National Environmental Policy Act, with DOE and NRC requirements, and with State permitting will be observed for the design, operation, safeguards, waste disposal, and protection of the public and the environment. Air and water quality protection, as required through federal and state acts will be implemented. Protection of plant personnel, and ensuring the health and safety of the public are fundamental to the design, construction, operation, and decommissioning of the System 80+ plutonium-disposition facility.

The National Environmental Policy Act will be the basis for the environmental plan. A plan to produce and review the draft Environmental Impact Statement in accordance with the provisions of the National Environmental Policy Act will be developed during the initial phase of this program. Compliance with Occupational Health and Safety, radiological safety, fire protection, and radiological emergency planning will be considered separately from environmental permitting requirements.

The implementation of the environmental policy will assure that the environment will be adequately protected from actions taken during the design, construction, and operation of the System 80+ plutonium-disposition facility. The plant designer and constructor will work closely with DOE to assure compliance with environmental statutes and regulations, and that environmental requirements contained in DOE Order 4700.1, "Project Management System," are implemented.

Elements to be considered when implementing the strategy for Environmental Policy include:

- Develop the Environmental Compliance Plan,
- Design, construct, test, operate, and decommission the System 80+ plutonium-disposition facility in compliance with environmental standards,
- Provide environmental compliance oversight, and
- Conduct applicable NEPA reviews and preparation of required NEPA documentation.

The requirements embodied in Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," will be utilized when preparing the Environmental Report. Consistent with these requirements, Chapter 12 will list all licenses, permits, and other approvals required by Federal, State, and Local government agencies having jurisdiction for the protection of the environment.

E. SAFEGUARDS AND SECURITY

A safeguards plan will be developed and implemented for the protection of plutonium in the fuel fabrication process, use of transport equipment, along transportation routes [if fabrication is located at a separate facility], and at the utilization facility. The identification of licenses and permits essential for the safeguards and security process, especially long-lead permits, will be developed during the initial phase of this effort.

F. LICENSING CHALLENGES

Challenges to licensing System 80+ as a plutonium-disposition facility include:

- Securing timely approval for construction;
- Licensing MOX fuel fabrication and utilization; and
- Licensing tritium target assemblies and the tritium recovery facility.

These are discussed further in Section VII regarding the various Deployment Strategies.

Table V-1
Typical Regulatory and Environmental Permits and Licenses for Plutonium-Disposition Facility

Permit or License	Requirements
National Emission Standards for Hazardous Air Pollutants (NESHAP)	Radiological source term Calculations. EPA approval prior to NESHAP application. Detailed procurement activities schedule required prior to start of construction. Applicability to specific facilities to be identified. Compliance with State requirements to be determined.
Construction permit/Operating license	NRC approval of the System 80+ standard plant. Approval of mixed-oxide fuel. Approval of the tritium target and tritium recovery facility.
Prevention of Significant Deterioration (PSD) of Air Quality	PSD permit process is independent of NESHAP. Controlled by State of residence for facility. Approval required prior to start of construction for facility that will emit regulated pollutants.
Air Quality	Diesel generators and concrete batch plant will be only source of air pollutants other than radionuclides. Limited diesel operating time per year may exclude need for permit.
Erosion Control Plan	Governs impact on terrain due to timber harvest, altering groundwater flow patterns, and storm water erosion control.
National Pollutant Discharge Elimination System (NPDES)	Governs effluent quality and quantity for all liquid discharges from facility. Storm water and process waste water control. An approved erosion control plan may be required.
Wetlands	Impact on protected wetlands.
Domestic (potable) water	Drilling of wells and water treatment systems.
Sanitary Waste water Treatment	NPDES requires discharge characteristics, anticipated manpower loading (utilization) and schedule. Discharge paths must be identified. Permit required for the construction of the waste water treatment plant.
Transportation	Safeguards for shipment of plutonium, mixed-oxide fuel, tritium.
Solid Waste Disposal	Identify non-hazardous, non-radioactive waste disposal by type and rate.
Federal Aviation Agency	Tall structures or cranes over 200 feet above ground level.
Navigable Waters	Modification to navigable water.
Timber Harvest	Forest management plan, if appropriate, to be developed.
National Historic Preservation Act (NHPA)	Survey of artifacts or discovery of archaeological items in any area of disturbance during facility construction.
American Indian Religious Freedom Act	Disturbance of areas considered "sacred" to Indian cultures.
Fish and Wildlife Coordination Act	Endangered species and migratory bird impact.

VI. COST AND SCHEDULE ESTIMATES

A. PRE-OPERATIONAL COSTS

Pre-operational cost elements are Research & Development, Pre-Title I engineering, regulatory safety and environmental efforts in support of each of the alternatives, plant startup and testing including operations procedures, as well as Reactor Complex administrative costs.

Research & Development activities for this project are mainly directed at fuel and target fabrication issues which include development and irradiation of demonstration assemblies. The other area of significance is the analysis of fuel performance characteristics. Successful completion of these activities will support specification of all fuel fabrication parameters and provide the basis for plant safety analyses.

It should be noted that because the System 80+ plant design includes provisions for accommodating plutonium as a fuel, very minor hardware changes are necessary to meet the requirements of this project. Since these changes are already identified, there are minimal development requirements for the System 80+ portion of the Reactor Complex.

Pre-Title I engineering, regulatory safety and environmental efforts in support of each of the alternatives, plant startup and testing, including operations procedures and Reactor Complex administrative costs are also provided within this scope.

The cost estimates in Section B were prepared in accordance with the guidelines provided by DOE. When it becomes time to actually locate the site for the PDR, it is quite likely that the plant can be located in a region of the United States which enjoys substantially better labor rates, productivity rates, etc. In addition, it is quite likely that a System 80+ design and/or construction program will be going on in Taiwan, Korea, or the U.K. at the same time as the PDR program. This would provide an opportunity to substantially lower the capital cost estimates.

Based upon separate evaluations that have been performed, it is estimated that the capital cost estimates in Section B could be lowered as much as $500 million if built in the Savannah River site area.

B. CAPITAL COSTS

The scope of work included within the Capital Cost Estimate includes all engineering, design, materials, commodities, equipment, installation, erection, testing and facilities necessary for an operating System 80+™ nuclear power plant coupled with a Fuel and Target Fabrication Facility. Costs for a Tritium Recovery Facility are shown separately.

<u>System 80+</u>

The Capital Cost Estimate was developed based upon the System 80+ Standard Nuclear Power Plant design accomplished to date under the USNRC Design Certification Program. Significant estimating activities have taken place as part of the System 80+ development and application process. Development of capital costs for plutonium disposition are based on those other initiatives. The following outlines the basis for the System 80+ Capital Cost Estimate. The estimate was assembled in accordance with the Plutonium Disposition Study (PDS) Requirements Document (RD) the Cost Estimate Guidelines (CEG) for Advanced Nuclear Power Technologies dated March 1993 (ORNL/TM/10071R3), and the Mid-term Review Agreements and Commitments.

A high confidence level capital cost estimate for System 80+ had been previously developed. The methodology for development of that estimate involved the following discrete steps. Since System 80+ is readily adaptable to the plutonium disposition mission, capital cost estimate development for the plutonium disposition options involved revisions for conformance to the Cost Estimate Guidelines and adjustments for minor scope variations.

Quantity Development

Individual components, including material commodities and bulks within the scope of supply, were quantified by computerized material takeoffs of drawings and diagrams, by manual takeoffs, and by adjusting quantities from previous nuclear power plant experience. Computer based systems were used for those areas and commodities which are significant contributors to the total capital cost.

Because the structural quantities in the nuclear island and turbine island represent a significant portion of the capital cost, they were completely modeled in 3-dimensional CAD. Concrete and steel were quantified utilizing the computer resource and used directly in the estimate, providing what we consider to be highly accurate scope definition.

Piping, which is another major contributor to total capital cost was partially quantified by 3D computer, as a function of the existing plant design detail.

Valves, which are also a significant contributor to total plant costs were quantified using computer takeoff, based upon system P&IDs.

Mini- Specifications

In order to facilitate a comprehensive, high confidence survey of component and equipment suppliers in a short time frame, component-specific "mini-specifications" were developed from full scope vendor-specific specifications used in previous procurements and distributed to all potential suppliers.

Equipment costs per these mini-specifications were assembled based on quotations. Manhours estimates for erection contracts are based on quotations. Since the scope of work was accurately expressed in the mini-specifications and supply and erection is based on budgetary quotations, these estimates generally provide a reasonably high confidence level.

Major Equipment

Pricing for the following major equipment was based on vendor quotations.

- NSSS

  The entire NSSS scope of supply, including the reactor vessel, steam generators, pressurizer, reactor coolant pumps and piping, shutdown cooling, CVCS, safety injection, and containment spray systems, along with related instruments and controls for the reactor protection system and the NUPLEX 80+ Control Complex.

- Turbine Generator

  The turbine generator scope of supply including turbines, generator, auxiliary equipment such as reheaters, moisture separator, lube oil equipment, hydrogen cooling, and the condenser; conceptual design including preliminary heat balances has been accomplished in order to properly size the equipment.

- Condenser/Feedwater Heaters

Major Equipment Erection

A mechanical erection contractor estimated the manhours to erect the NSSS. A contractor also estimated the manhours for the turbine generator, condenser and feedwater heater erection. The balance of major equipment erection scope was estimated using previous construction experience.

Bulk Material and Labor

Bulk material quantities were developed based on computer aided design, manual takeoffs and adjustment for comparable nuclear project plant data. Reference was made to historical information where appropriate to test the reasonableness of the information developed.

Bulk materials were priced in accordance with the Cost Estimate Guidelines where applicable. Bulk material installation rates were also based on the Cost Estimate Guidelines.

Table VI B-1 reflects representative quantity information of selected commodities of a single System 80+ complex.

Crew Labor Rates

Labor rates used in the estimate for erection/installation crews are those specified in the Cost Estimate Guidelines.

Indirect Cost

Indirect costs are costs required to support the construction effort but not identifiable to a specific end use account. The indirect costs for this construction effort were developed based on historical information, adjusted, where necessary for the scope and complexity of the project. A non-manual staffing profile was developed consistent with the project manual labor supervisory requirements, the project schedule, and the type of work anticipated. In addition, the associated support staff requirements were evaluated. Experienced construction personnel assessed the temporary facility requirements, as well as the construction equipment needs required for the identified construction methodologies.

Project Management and Engineering

The engineering costs were developed by each engineering discipline based upon the specific engineering products and efforts required. In addition, the engineering resources required to support procurement and construction activities were assessed and included.

Project Management costs have been included. Costs for administration support services within the project organization have been included. These services include cost and scheduling, construction liaison, procurement, and other support services.

Fuel and Target Fabrication Facility

Fuel and Target Fabrication Facility costs were based on previous studies.

Table VI B-2 summarizes the capital costs of the three base case plutonium disposition options and the two additional options, and the Tritium Recovery Facility. Capital cost information at the EEDB account level for the various deployment options is included in Tables VI B-5 to VI B-8. Table VI B-3 and VI B-4 provide operating information about the various options. Table VI B-9 separates capital costs by FTFF and Energy Conversion Area (ECA) by unit number by deployment option.

Figures VI B-1 through B-16 present the cumulative cash flow projections vs. time, capital cost vs. time, R&D cost vs. time and preoperational costs vs. time.

C. OPERATIONAL AND MAINTENANCE COST

Operating and Maintenance Costs (O&M) are defined as the costs to operate and maintain the complex from initial operation through decommissioning. This portion of the overall reactor complex estimate addresses power production from the Plutonium Disposition Reactor (PDR) and the Energy Conversion Area (ECA). The Requirements Document, Section 4.0, Information Requirements, Sections 4.5 and 4.5.1.3 identify the information requested for the Operating and Maintenance Cost portion of the Study.

Staffing and Cost Methodology

The basic methodology utilized in the Plutonium Disposition Study for the five deployment options involved development of a "bottoms up" estimate. A review of industry data was used to facilitate development process by providing a cross check on the reasonability of the estimate. Aspects reviewed in the industry data included costs and staffing levels by the number of units per plant, generating unit size, reactor type, and year of commercial operation. In addition, other industry data was reviewed regarding D&D, spare parts inventories, etc. Particular benefits of this approach includes a better understanding of the O&M cost impacts of advanced design features built into the System 80 + design, and therefore inherently beneficial to the Plutonium Disposition Reactor.

The "bottoms-up" approach is defined as beginning with a "zero" base and developing cost estimates and workforce requirements for each of the deployment options. A recent ABB-CE reactor study provided recent System 80+ staffing estimates as a starting point for the PDR. The estimates followed the guidance and requirements provided in the PDS Requirements Document and the "Cost Estimating Guidelines for Advanced Nuclear Power Technologies," March 1993 (CEG). Verification based on an independent study was also factored into the final analysis. Utilizing these data sources, a "site" workforce complement (O&M on-site staff and Administration & General staff) and an annual cost estimate was developed.

Staffing Requirements

Staffing requirements for the Plutonium Disposition Plant are based on a "bottoms up" approach. The recent System 80 + staffing allocation developed for another installation served as an excellent starting point. Those System 80+ staffing allocations were the result of a comprehensive assessment conducted by members of the System 80+ team with a background in power plant operations and maintenance. The goal was to establish an optimum staff for normal operation, refueling outages and abnormal occurrences, taking into account NRC requirements, INPO recommendations and utility operating experience.

As an evolutionary Advanced Light Water Reactor, numerous design features have been incorporated into System 80+ with the objective of improving operational capabilities. These improvements (e.g., architectural accessibility, simplified systems designs, improved materials, equipment and technology, etc.) have the effect of reducing staffing levels below the requirements of the latest generation of operating reactors on a comparable basis for the System 80+ as modified for the plutonium disposition mission. This required review of previous staffing estimates, with consideration for advanced design features, design and functional changes for the plutonium disposition mission, and adjustment of the staffing estimates, as appropriate.

The basic workforce allocation may be categorized into three major areas: management, staff support, and shift personnel. Staffing was reviewed by functional areas considering specific tasks and duties. Typically, staffing of functional groups are either task oriented (e.g., maintenance, operations, chemistry) or are dependent on the size of other groups (e.g., administration, medical, safety). Although the functions and responsibilities of the groups remain consistent between the System 80+ study and the PDS, the staffing increased for the plutonium disposition mission based on several factors. For example, since the plant was required to be a separate operating company and along with DOE security requirements, the structure of the station organization was adjusted to satisfy the PDS Requirements Document.

The labor costs are segmented between the PDR and the ECA. To segment these costs the labor and costs of the workforce are factored to tasks performed in their respective section of the facility. This percentage may be related to the workforce allocated to the PDR or ECA. As with the initial workforce allocation this study was performed as a "bottoms up" approach.

Staffing Assumptions

The staffing philosophy and associated assumptions include:

All shift sections have a supervisor that rotates with their personnel.

Shift positions are self-relieving, except for the senior on-shift supervisors.

Operations has primary fire brigade responsibilities, with Security providing backup.

A regulatory compliance representative from each group is provided as needed.

Clerical and managerial support is provided in each group.

Shift personnel work a 5 shift, 12 hour workday rotation.

Training is provided within the shift rotation.

When personnel are not actively training, they are used to supplement the day staff.

Maintenance and testing of plant equipment is scheduled on a 24 hour per day basis to maximize the use of shift personnel, facilities and equipment.

The shift crews are "self-sufficient" in that they have a representative from all necessary work groups to perform routine maintenance and surveillance testing.

The support staff is a mixture of specialized and cross-trained disciplines allowing the flexibility to perform outage and operating responsibilities without unreasonable overtime or outside assistance.

Outage staffing was evaluated for the 17 day refueling (only) and the 50 day refueling durations. The personnel allocated for the outage will incur planned overtime. Additional "specialty" personnel are provided to perform functions such as turbine inspection and repair, steam generator inspection, etc. This contract support is supplemental to the normal complement of plant personnel and under the direction of plant management.

Staffing levels for dedicated common facilities or functions are included in the single reactor generating unit staffing. Examples of common facilities and functions are:

Training facilities, emergency response, water/sewage treatment facilities, and environmental programs.

Principal Functional Areas

A brief discussion of the principal functional areas is provided below.

The **Operations** group includes four functional areas. The shift support area includes relief supervisors, training and scheduling personnel and other administrative positions. The operations support area focuses on procedure development. Operations engineering support is responsible for work coordination within operations, including outage items. The shift crew is tasked with the implementation of station operating plans, operations surveillance's and monitoring of plant equipment. The shift crew also satisfies minimum crew requirements of the proposed Technical Specifications included in CESSAR-DC for System 80+.

The **Maintenance** group includes six functional areas. The mechanical group consists of specialized pump crews, diesel generator control, shift crews performing routine maintenance, and supervisory support. A mechanical engineering group provides the system and component expertise for the mechanical group. Instrument Electrical engineering personnel perform the tasks associated with calibration labs, process instrumentation surveillance and repair, valve support, and supervisory support. Shift crews within this group perform routine maintenance and surveillance. An Instrument/Electrical engineering group is also provided for system and component expertise. The maintenance work planning group prepares the routine and outage work orders for the craft groups to implement. QA/Materials is responsible for the procurement of supplies and processing receivable. QA also includes technical support, inspections and supervisory support for the group.

The **Chemistry** group includes three functional areas with a staff and shift crew responsible for each of these. One group provides routine sampling of the primary, secondary, and environmental processes. The chemistry group supports water treatment and production. The environmental group is responsible for compliance with regulatory compliance for non-nuclear wastes.

The **Radiation Protection** (RP) group consists of three functional areas. RP provides routine support for all work groups and overall station operation. In addition, count room personnel support analyses of surveys and samples. The ALARA group supports work control and station personnel to ensure the lowest doses achievable during station operation.

The **Integrated Scheduling** group includes two functional areas for routine scheduling and outage scheduling. The routine scheduling areas is responsible for the day to day execution of surveillance, testing and repair activities. The outage scheduling area assemblies work packages, prepares the outage plan prior to the outage and tracks outage activities during the outage.

The **Performance** group consists of three areas. The reactor engineering group is responsible for core analysis, fuel load patterns and required surveillance of core parameters. The test group has the primary responsibility for accomplishment of surveillance testing to satisfy regulatory and Code requirements. Engineering support provides expertise with test methods, procedure development, and problem resolution.

The **Security** group consists of two staff areas and the shift crews. One functional areas within security is responsible for badging and access authorization of the station personnel while the other group provides compliance and training support. The shift crew performs monitoring and access control.

Safety is responsible for three functional areas within the station. The medical group provides access screening, routine physical and emergency medical services. The fire protection group provides fire prevention and brigade expertise, including training of the fire brigade. The industrial hygiene group performs the duties of ensuring compliance with OSHA and state industrial regulations including training on these subjects.

Training for power plant personnel is provided in four areas by the training department. General employee training includes all that is necessary to satisfy requirements for access to the facility. Operator training is for systems training, initial licensing and operator re qualification training. Simulator training is also included in this area. Technical services training is allocated for the Chemistry, RP and Performance groups. Maintenance training is provided for the mechanical and instrument/electrical craft personnel. Staff functions are consistent with INPO accreditation guidelines.

Administration Services is divided into four areas. The broadest of these is general employee services, which includes human resources and station management. Accounting is responsible for contracts, accounts payable and receivable, and payroll. Regulatory compliance is the focal point for station interface with DOE, NRC and other governing regulatory agencies. Licensee event reporting, routine reporting and expertise in licensing issues are tasked to this group. The Emergency Planning Group is placed under Administration Services to develop the emergency plan and training exercises.

General

The O&M costs for the PDR and ECA were developed. Annual on-site staff salaries which are shown in the CEG Guidelines, Table 4.4 (Page 51) were used to develop annual labor cost. An additional 10 percent was added for overtime, supplemental pay, etc., and an additional 10 percent was added for social security tax and unemployment insurance premiums. The detailed outage cost data was developed for: a) pricing out off-site technical support required for outages, b) validation that the total direct O&M estimate included sufficient dollars for outage cost. Annual nuclear regulatory fees were assumed to be $2.8 million (1992$) per unit.

Administrative and General Costs

The total Administrative and General costs were also developed using the Cost Estimate Guidelines (March 1993). The pension and benefits account which includes workman's compensation insurance was calculated at 25% of the sum of on-site and off-site direct salaries (excluding off-site overhead). Estimates for annual premiums for nuclear plant insurance for advanced nuclear plants were provided in Table 4.5 (CEG page 52). Finally, other administrative and general expenses were calculated at 15 percent of the direct power generation accounts.

Outage Costs

The economics carrying over from the System 80 + design improvements were factored into the analysis (e.g. ability to do comparatively more preventive and corrective maintenance on line than current industry; more effective access to the steam generator for maintenance; etc.) Outage cost was evaluated for the 20 day and 50 day refueling outages, which include refueling. A 17 day outage is possible for refueling only. The basic assumption for all five (5) deployment options as to bring approximately 100 contractors to supplement the existing workforce and cost out a proportional amount for the Plutonium Disposal Reactor and the Energy Conversion Area to accomplish each outage. The total cost of $3.2 million is applicable to all cases and depending on the cycle length (number of outages) for each deployment option, the annual cost estimate includes sufficient dollars.

Decontamination and Decommissioning (D&D)

This study was developed to compare/analyze the different methods for calculating the costs for Decontamination and Decommissioning. The cost for decontamination utilized in this analysis was considered to be included in the decommissioning cost. The funding requirements are based on the Cost Estimating Guidelines (CEG).

The use of the specific DOE Cost Estimating Guidelines is illustrated below.

Decommissioning Cost (Millions $) = 165 + 0.020 (P-1200)

where: P = Unit Thermal power (MWt)

S-O	Spiking	3817 MWt 165 + .020 (3817-1200)	=	$217M
SF-0	Spent Fuel	3817 MWt ($217 M/Unit) x (4 Units)	=	$868M
D-O	Destruction	3817 MWt ($217 M/Unit) x (4 Units)	=	$868M
SF-1	Spent Fuel	3817 MWt ($217 M/Unit) x (1 Unit)	=	$217M
SF-2	Spent Fuel	3817 MWt ($217 M/Unit) x (2 Units)	=	$434M

Component Replacement and Operating Spares

Programmatic Evaluation:

An effective maintenance program requires timely availability of parts, materials and services. The procurement process for each of the deployment options will assure that parts, materials, and services are available when needed. In the development of the dollar values for component replacement and operating spares, the following aspects were considered:

Standardization of equipment to minimize inventory levels
Vendor recommended spare parts list
Spare parts lead time on a "just in time" delivery concept
Spare parts shelf life

Spare parts pricing
Potential for development of cooperative stocking programs with vendors and other users
Utilization of industry standard equipment

Cost Analysis:

The cost analysis was based on a recent ABB-CE study, industry data, and an assessment of the NSSS and turbine generator spare parts. This analysis provided a study basis of $54 million for a single unit, $72 million for a dual unit, and $112.5 million for a four unit site.

The O&M cost estimate detail, including staffing requirments for the three base cases and the two deployment options are presented in Tables VI C-1 through 7.

D. TRITIUM RECOVERY FACILITY

Tritium Recovery Facility costs were based on previous studies.

E. REVENUES FROM SALE OF ELECTRIC POWER

The following method was used in calculating Annual Electric Revenue:

Step 1 Gross electrical output (MWe) minus ECA houseloads equals net MWe delivered.

Step 2 Net MWe delivered multiplied by the capacity factor multiplied by "Annual Maximum Generation Hours" equals MWH

Deployment Option S-0 (Spiking)	43% Capacity Factor
Required deployment options (SF-0, D-0)	75% Capacity Factor
Alternate deployment options (SF-1, SF-2)	80% Capacity Factor

Step 3 MWH multiplied by "electrical price off grid" rate equals Total Annual Electric Revenue.

[Note: ECA house load is approximately 45 MW].

The revenue projections for the three base cases and the two deployment options are presented in Tables VI E-1 through 5.

F. SCHEDULE

The overall summary program for the base case deployment options is presented in a format referred to as the Project Summary Network (PSN). The PSN is a schedule of activity at the total project level that displays the significant stages together with identification of key events and milestones against a calendar time scale.

The PSN divides the total project scope into the main work group elements. Major tasks within each work group are identified together with logic ties that indicate the significant relationships between specific tasks and events. These are: Licensing, Engineering & Procurement, Construction, Commissioning, Fuel and Target Fabrication and Tritium Recovery. The intervals between contract placement and start of site activities are clearly identified. The major constraints to the start of major areas of construction work, including major equipment deliveries, are shown.

Due to the summary level of the PSN it is not specifically site related. Site differences would appear on schedules reflecting a higher level of detail. Such differences would not impact the overall project duration.

For the System 80+ complex the work that is described and scheduled in the Project Summary Network at a relatively high level is defined in greater detail in individual schedules developed for other initiatives. These programs are Level II and Level III Network Schedules. Each sublevel schedule breaks the activities down by primary work groups shown by the higher level schedules and shows the work in greater detail according to lower elements in the Work Breakdown Structure. The activity durations shown on the Level II correlate exactly with the corresponding summary activity durations, logic ties, and major milestones shown on the Project Summary Network. Level III relates work elements to components and resources, etc.

The major engineering is reflected by the Engineering and Procurement Section on the PSN. Engineering will be completed on a schedule to support construction requirements.

Construction has a duration of 54 months from first concrete to fuel loading. Once begun, the critical path is through the nuclear island concrete, containment sphere, primary loop components and the reactor shield building. The Construction Schedule is labor intensive due to the short schedule duration. As usual for construction, civil work is followed by mechanical, electrical, and instrumentation and controls erection and installation. The construction organization will perform component tests such as hydro tests prior to turning systems over to the commissioning organization.

The PSN reflects licensing activities at a summary level. Development of a Licensing Plan and detailed schedule is proposed in Phase II of the PDS.

Commissioning activities will begin approximately 36 months before fuel load. All system tests will be performed. The NSSS testing will include cold hydro and hot functional testing prior to fuel load. A system by system program with further breakdowns by sub-system where necessary would be developed as part of a detailed commissioning schedule and include the operator training programs. All activities are totally integrated both within a system and to other systems, and integrated with other final plant activities.

Figures VI F-1 through VI F-3 reflect the PSNs for the base case plutonium disposition options. The schedules reflect authorization on October 1, 1993 and first unit operation in March of the year 2001 with operation of subsequent units at succeeding six month intervals.

TABLE VI.B-1
MAJOR COMMODITY QUANTITIES
FOR OPTION S-0

COMMODITY	QUANTITY
CONCRETE	382,600 CY
FORMWORK	5,256,000 SF
REINFORCING	51,700 TN
STRUCTURAL STEEL	10,600 TN
EMBEDDED IRON	10,914,000 LB
POWER CABLE	630,000 LF
I&C CABLE	2,990,000 LF
CABLE TRAY	75,0000 LF
CONDUIT	632,000 LF
PIPE "2" AND SMALLER	205,000 LF
PIPE 2.5" AND LARGER	168,000 LF

NOTE

THESE ARE REPRESENTATIVE QUANTITIES AND TAKE INTO ACCOUNT LARGER FUEL POOLS THAN STANDARD DESIGN BECAUSE OF FUEL STORAGE REQUIREMENTS.

NO QUANTITIES ARE INCLUDED FOR THE FUEL AND TARGET FABRICATION FACILITY.

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TABLE VI.B-2
CAPITAL COST SUMMARY
(JAN' 92 $ MILLION)

	BASE OPTIONS			OTHER OPTIONS		
	SPIKING S-O	SPENT FUEL SF-O	DESTRUCTION D-O	SPENT FUEL SF-1	SPENT FUEL SF-2	TRITIUM RECOVERY
PRE-OPERATIONAL						
Fuel R&D	33	33	60	33	33	-
Tritium Recovery	-	-	-	-	-	85
Fuel & Target Fabrication	45	45	45	45	45	-
PDR/ECA Pre-Op	150	225	225	150	175	-
CAPITAL						
PDR/ECA	1,437	4,961	4,961	1,437	2.633	-
Fuel & Target Fab.	700	450	450	450	450	-
Tritium Recovery	-	-	-	-	-	225
Indirect Cost	1,354	3,990	3,990	1,354	2,234	11
TOTAL CAPITAL	3,491	9,401	9,401	3,241	5,317	236
GRAND TOTAL	3,719	9,704	9,731	3,469	5,570	321

PDR/FTFF	3,319	8,343	8,370	3,069	4,835
ECA	400	1,361	1,361	400	735
GRAND TOTAL	3,719	9,734	9,731	3,469	5,570

TABLE VI.B-3
OPERATING DATA

	SPIKING S-O	SPENT FUEL SF-O	DESTRUCTION D-O	SPENT FUEL SF-1	SPENT FUEL SF-2
NUMBER OF REACTORS	1	4	4	1	2
Rx Size Capacity (MWt)/Unit	3,800	3,800	3,800	3,800	3,800
CYCLE EXPOSURE (MWD/MTHM)	1,500	10,550	(1089 EFPD)	10,550	10,550
CYCLE LENGTH (Days)	90	365	365	365	365
CAPACITY FACTOR (%)	43	75	75	80	80
AVERAGE DISCHARGE BURNUP (MWD/MWth)	1,500	42,200	42,200	45,000	45,000
Pu DISPOSITION (MT/YR) Normal Options	25	6 20 (SPIKE)	6 20 (SPIKE)	2 7 (SPIKE)	4 14 (SPIKE)
REQ'D FUEL FABRICATION RATE (MT/YR)	300	100	100	100	100
SPENT FUEL STORAGE CAPACITY REQ'TS. CELLS	5,000	1,250 Normal SFP System 80+	1,250 Normal SFP System 80+	1,250 Normal SFP System 80+	1,250 Normal SFP System 80+
TIMEFRAME TO START OF OPERATIONS (YRS) - FUEL FABRICATION - REACTOR	 6 7	 6 7	 6 7	 6 7	 6 7
Pu DISPOSITION DURATION (YRS) (AFTER START OF OPERATIONS)	4	18	18	60 15 (SPIKE)	30 8 (SPIKE)
Pu DESTRUCTION (%)	1	98	99	98	98
COST OF CONSTRUCTION AND ENGINEERING ($) MILLIONS	3,719	9,704	9,731	3,469	5,570

TABLE VI.B-4
OPERATING DATA

	SPIKING S-O	SPENT FUEL SF-O	DESTRUCTION D-O	SPENT FUEL SF-1	SPENT FUEL SF-2	TRITIUM RECOVERY
Electric Generation (MWe) (Per Unit)	1,350 Gross 1,256 Net	1,350 Gross 1,256 Net	1,350 Gross 1,256 Net	1,350 Gross 1,256 Net	1,350 Gross 1,256 Net	1,200 Gross 1,115 Net
Cycle Length	3 Months 39 EFPD	12 Months 274 EFPD	12 Months 274 EFPD	12 Months 292 EFPD	12 Months 292 EFPD	12 Months 274 EFPD
Refueling & Maintenance Outage (Days) Per Unit	17 Day Refueling, 4 Outages/Yr. Avg. of 20 Days each (80 Days/Yr Total)	17 Day Refueling 50 Days (Total)	17 Day Refueling 50 Days (Total)	17 Day Refueling 50 Days (Total)	17 Day Refueling 50 Days (Total)	N/A
Add'l Commercial Ops Duration (YRS)	56	42	42	0	30	N/A
Cost of Construction & Engineering ($) (Millions)	3,719	9,704	9,731	3,469	5,570	321
Fuel Cost ($/YR)	0	0	0	0	0	N/A
O&M Costs ($/YR) Millions						
PDR	75.4	189.4	189.4	73.2	98.8	N/A
ECA	22.1	53.6	53.6	19.6	27.4	
FTFF	75.0	75.0	75.0	75.0	75.0	N/A
TRF	-	-	-	-	-	7.0
Avg. Cost of Operation ($/YR)	97.5	243	243	92.8	126.2	N/A
ECA Capital Improvements ($/Yr) Millions	2.9	10.7	10.7	2.9	5.7	N/A
Capacity Factors%	43	75	75	80	80	75

TABLE VI. B–5
COST ESTIMATE BY EEDB COST ACCOUNT
SPIKING OPTION (S–0)

THOUSANDS OF JANUARY, 1992 DOLLARS

EEDB ACCT. NUMBER	ACCT. DESCRIPTION	FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL	TOTAL $
211	YARDWORK INCL. LAND COSTS		602,154	$13,428	$8,478	$21,906
212	REACTOR BUILDING	$3,202	2,643,856	$66,045	$45,566	$114,813
213	TURBINE BUILDING	$744	844,240	$21,618	$21,368	$43,730
214	SECURITY BUILDING AND GATE HOUSE	$15	43,452	$1,119	$651	$1,785
215	AUXILIARY BUILDING	$11,346	5,506,578	$135,380	$55,568	$202,294
216	RADWASTE FACILITY	$599	658,591	$16,445	$7,290	$24,334
218B	ADMINISTRATION AND SERVICE BUILDING	$650	75,343	$1,929	$1,886	$4,465
218C	ONSITE STANDBY AC POWER GENERATION	$582	368,329	$8,996	$3,959	$13,537
218D	FIRE PUMPHOUSE	$30	2,737	$67	$39	$136
218K	PIPE & ELECTRIC TUNNELS		386,860	$9,259	$3,838	$13,097
218R	AUXILIARY BOILER BUILDING		937	$25	$72	$97
218T	ULTIMATE HEAT SINK STRUCTURE		788,976	$19,171	$13,184	$32,355
218Z	OTHER MISC. STRUCTURES	$334	123,811	$3,019	$1,671	$5,024
21	STRUCTURES & IMPOVEMENTS	$17,502	12,045,864	$296,501	$163,570	$477,573
222	MAIN HEAT TRANSPORT SYSTEM	$387,552	356,803	$8,892	$14,424	$410,868
223	SAFEGUARDS SYSTEM	$3,117	236,090	$5,894	$3,852	$12,863
224	RADWASTE PROCESSING	$2,364	100,509	$2,543	$1,787	$6,694
225	FUEL HANDLING & STORAGE	$3,143	144,483	$3,606	$3,276	$10,025
226	OTHER REACTOR PLANT EQUIPMENT	$4,712	1,608,258	$40,347	$23,068	$68,127
227	REACTOR PLANT INSTRUMENTATION & CONTROL	$602	21,792	$578	$847	$2,027
228	PLANT SIMULATOR	$5,957	17,000	$452		$6,409
22	REACTOR PLANT EQUIPMENT	$407,447	2,484,935	$62,312	$47,254	$517,013
231	TURBINE GENERATOR	$113,648	344,786	$8,600	$7,205	$129,453
233	CONDENSING SYSTEM	$28,492	133,537	$3,394	$2,275	$34,161
234	FEED HEATING SYSTEM	$20,773	230,522	$5,787	$5,640	$32,200
235	OTHER TURBINE PLANT EQUIPMENT	$1,469	291,432	$7,450	$5,703	$14,622
237	TURBINE PLANT MISC. ITEMS	$217	34,673	$868	$746	$1,831
23	TURBINE PLANT EQUIPMENT	$164,599	1,034,950	$26,099	$21,569	$212,267

TABLE VI. B-6
COST ESTIMATE BY EEDB COST ACCOUNT
SPENT FUEL AND DESTRUCTION OPTIONS (SF-0 & D-0)

THOUSANDS OF JANUARY, 1992 DOLLARS

EEDB ACCT. NUMBER	ACCT. DESCRIPTION	FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL	TOTAL $
211	YARDWORK INCL. LAND COSTS		2,302,711	$51,350	$13,400	$64,750
212	REACTOR BUILDING	$11,702	10,424,670	$260,050	$172,344	$444,096
213	TURBINE BUILDING	$2,717	3,228,509	$82,670	$78,111	$163,498
214	SECURITY BUILDING AND GATE HOUSE	$55	166,261	$4,283	$2,381	$6,719
215	AUXILIARY BUILDING	$38,592	15,464,433	$378,906	$142,910	$560,408
216	RADWASTE FACILITY	$2,189	2,522,242	$62,983	$26,656	$91,828
218B	ADMINISTRATION AND SERVICE BUILDING	$2,374	288,425	$7,385	$6,895	$16,654
218C	ONSITE STANDBY AC POWER GENERATION	$2,126	1,408,356	$34,400	$14,465	$50,991
218D	FIRE PUMPHOUSE	$108	10,472	$255	$143	$506
218K	PIPE & ELECTRIC TUNNELS		1,480,144	$35,428	$14,032	$49,460
218R	AUXILIARY BOILER BUILDING		3,582	$95	$264	$359
218T	ULTIMATE HEAT SINK STRUCTURE		3,018,350	$73,342	$48,186	$121,528
218Z	OTHER MISC. STRUCTURES	$1,222	473,760	$11,551	$6,110	$18,883
21	STRUCTURES & IMPOVEMENTS	$61,085	40,791,915	$1,002,698	$525,897	$1,589,680
222	MAIN HEAT TRANSPORT SYSTEM	$1,283,479	1,364,249	$33,997	$52,707	$1,370,183
223	SAFEGUARDS SYSTEM	$10,840	895,634	$22,364	$13,992	$47,196
224	RADWASTE PROCESSING	$8,632	378,404	$9,577	$6,470	$24,679
225	FUEL HANDLING & STORAGE	$11,134	376,367	$9,370	$8,453	$28,957
226	OTHER REACTOR PLANT EQUIPMENT	$17,127	5,832,452	$146,352	$80,850	$244,329
227	REACTOR PLANT INSTRUMENTATION & CONTROL	$2,201	83,021	$2,203	$2,962	$7,366
228	PLANT SIMULATOR	$13,221	31,450	$837		$14,058
22	REACTOR PLANT EQUIPMENT	$1,346,634	8,961,577	$224,700	$165,434	$1,736,768
231	TURBINE GENERATOR	$415,290	1,318,326	$32,883	$26,329	$474,502
233	CONDENSING SYSTEM	$104,116	510,564	$12,977	$8,312	$125,405
234	FEED HEATING SYSTEM	$75,909	881,456	$22,131	$20,610	$118,650
235	OTHER TURBINE PLANT EQUIPMENT	$5,368	1,110,532	$28,393	$20,803	$54,564
237	TURBINE PLANT MISC. ITEMS	$792	123,634	$3,095	$2,615	$6,502
23	TURBINE PLANT EQUIPMENT	$601,475	3,944,512	$99,479	$78,669	$779,623

TABLE VI. B-6 (cont'd)
COST ESTIMATE BY EEDB COST ACCOUNT
SPENT FUEL AND DESTRUCTION OPTIONS (SF-0 & D-0)

EEDB ACCT. NUMBER	ACCT. DESCRIPTION	THOUSANDS OF JANUARY, 1992 DOLLARS FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL	TOTAL $
241	SWITCHGEAR		73,496	$1,976	$32,983	$34,959
242	STATION SERVICE EQUIPMENT	$24,331	818,629	$21,639	$121,009	$166,979
243	SWITCHBOARDS		89,828	$2,416	$16,382	$18,798
244	PROTECTIVE EQUIPMENT		195,989	$5,270	$3,713	$8,983
245	ELECT. STRUCT. & WIRING CONTNR		4,883,381	$131,315	$13,979	$145,294
246	POWER & CONTROL WIRING		2,262,035	$60,826	$30,361	$91,187
24	ELECTRIC PLANT EQUIPMENT	$24,331	8,323,358	$223,442	$218,427	$466,200
251	TRANSPORTATION & LIFT EQUIPMENT	$30,691	142,343	$3,489		$34,180
252	AIR WATER & STEAM SERVICE SYSTEMS	$32,172	2,309,148	$58,617	$34,983	$125,772
253	COMMUNICATION & SECURITY SYSTEM	$329	907,454	$24,317	$13,074	$37,720
255	WASTEWATER TREATMENT EQUIPMENT		31,059	$777	$590	$1,367
256	MAINTENANCE & TEST EQUIPMENT	$14,026	3,386	$84	$30	$14,140
25	MISCELLANEOUS PLANT EQUIPMENT	$77,218	3,393,390	$87,284	$48,677	$213,179
261	STRUCTURES	$6,508	1,686,998	$41,380	$23,826	$71,714
262	MECHANICAL EQUIPMENT	$58,142	1,618,083	$39,813	$5,576	$103,531
26	MAIN COND. HEAT REJECT SYSTEM	$64,650	3,305,081	$81,193	$29,402	$175,245
31	FUEL & TARGET HANDLING FACILITY	$270,000	3,807,100	$90,000	$90,000	$450,000
	TOTAL DIRECT COSTS	$2,445,393	72,526,933	$1,808,796	$1,156,506	$5,410,695
900	INDIRECT COSTS		22,682,877	$1,690,356	$2,524,892	$4,215,248
	TOTAL COSTS	$2,445,393	95,209,810	$3,499,152	$3,681,398	$9,625,943

TABLE VI. B–7
COST ESTIMATE BY EEDB COST ACCOUNT
SPENT FUEL OPTION (SF–1)

THOUSANDS OF JANUARY, 1992 DOLLARS

EEDB ACCT. NUMBER	ACCT. DESCRIPTION	FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL	TOTAL $
211	YARDWORK INCL. LAND COSTS		602,154	$13,428	$8,478	$21,906
212	REACTOR BUILDING	$3,202	2,643,856	$66,045	$45,566	$114,813
213	TURBINE BUILDING	$744	844,240	$21,618	$21,368	$43,730
214	SECURITY BUILDING AND GATE HOUSE	$15	43,452	$1,119	$651	$1,785
215	AUXILIARY BUILDING	$11,346	5,506,578	$135,380	$55,568	$202,294
216	RADWASTE FACILITY	$599	658,591	$16,445	$7,290	$24,334
218B	ADMINISTRATION AND SERVICE BUILDING	$650	75,343	$1,929	$1,886	$4,465
218C	ONSITE STANDBY AC POWER GENERATION	$582	368,329	$8,996	$3,959	$13,537
218D	FIRE PUMPHOUSE	$30	2,737	$67	$39	$136
218K	PIPE & ELECTRIC TUNNELS		386,860	$9,259	$3,838	$13,097
218R	AUXILIARY BOILER BUILDING		937	$25	$72	$97
218T	ULTIMATE HEAT SINK STRUCTURE		788,976	$19,171	$13,184	$32,355
218Z	OTHER MISC. STRUCTURES	$334	123,811	$3,019	$1,671	$5,024
21	STRUCTURES & IMPOVEMENTS	$17,502	12,045,864	$296,501	$163,570	$477,573
222	MAIN HEAT TRANSPORT SYSTEM	$387,552	356,803	$8,892	$14,424	$410,868
223	SAFEGUARDS SYSTEM	$3,117	236,090	$5,894	$3,852	$12,863
224	RADWASTE PROCESSING	$2,364	100,509	$2,543	$1,787	$6,694
225	FUEL HANDLING & STORAGE	$3,143	144,483	$3,606	$3,276	$10,025
226	OTHER REACTOR PLANT EQUIPMENT	$4,712	1,608,258	$40,347	$23,068	$68,127
227	REACTOR PLANT INSTRUMENTATION & CONTROL	$602	21,792	$578	$847	$2,027
228	PLANT SIMULATOR	$5,957	17,000	$452		$6,409
22	REACTOR PLANT EQUIPMENT	$407,447	2,484,935	$62,312	$47,254	$517,013
231	TURBINE GENERATOR	$113,648	344,786	$8,600	$7,205	$129,453
233	CONDENSING SYSTEM	$28,492	133,537	$3,394	$2,275	$34,161
234	FEED HEATING SYSTEM	$20,773	230,522	$5,787	$5,640	$32,200
235	OTHER TURBINE PLANT EQUIPMENT	$1,469	291,432	$7,450	$5,703	$14,622
237	TURBINE PLANT MISC. ITEMS	$217	34,673	$868	$746	$1,831
23	TURBINE PLANT EQUIPMENT	$164,599	1,034,950	$26,099	$21,569	$212,267

TABLE VI. B-7 (cont'd)
COST ESTIMATE BY EEDB COST ACCOUNT
SPENT FUEL OPTION (SF-1)

THOUSANDS OF JANUARY, 1992 DOLLARS

EEDB ACCT. NUMBER	ACCT. DESCRIPTION	FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL	TOTAL $
241	SWITCHGEAR		18,313	$494	$9,094	$9,588
242	STATION SERVICE EQUIPMENT	$6,659	216,863	$5,714	$33,247	$45,620
243	SWITCHBOARDS		22,589	$609	$4,527	$5,136
244	PROTECTIVE EQUIPMENT		50,552	$1,362	$995	$2,357
245	ELECT. STRUCT. & WIRING CONTNR		1,286,898	$34,680	$3,846	$38,526
246	POWER & CONTROL WIRING		597,525	$16,007	$8,360	$24,367
24	ELECTRIC PLANT EQUIPMENT	$6,659	2,192,740	$58,866	$60,069	$125,594
251	TRANSPORTATION & LIFT EQUIPMENT	$9,061	42,075	$1,031		$10,092
252	AIR WATER & STEAM SERVICE SYSTEMS	$8,829	614,090	$15,584	$9,763	$34,176
253	COMMUNICATION & SECURITY SYSTEM		236,444	$6,339	$3,578	$9,917
255	WASTEWATER TREATMENT EQUIPMENT	$90	9,010	$225	$161	$476
256	MAINTENANCE & TEST EQUIPMENT	$3,839	886	$21	$8	$3,868
25	MISCELLANEOUS PLANT EQUIPMENT	$21,819	902,505	$23,200	$13,510	$58,529
261	STRUCTURES	$1,781	441,021	$10,817	$6,518	$19,116
262	MECHANICAL EQUIPMENT	$15,911	423,189	$10,413	$1,526	$27,850
26	MAIN COND. HEAT REJECT SYSTEM	$17,692	864,210	$21,230	$8,044	$46,966
31	FUEL & TARGET HANDLING FACILITY	$270,000	3,807,100	$90,000	$90,000	$450,000
	TOTAL DIRECT COSTS	$905,718	23,332,304	$578,208	$404,016	$1,887,942
900	INDIRECT COSTS		7,189,887	$492,582	$1,011,785	$1,504,367
	TOTAL COSTS	$905,718	30,522,191	$1,070,790	$1,415,801	$3,392,309

TABLE VI. B–8
COST ESTIMATE BY EEDB COST ACCOUNT
SPENT FUEL OPTION (SF–2)

EEDB ACCT. NUMBER	ACCT. DESCRIPTION	THOUSANDS OF JANUARY, 1992 DOLLARS FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL	TOTAL $
211	YARDWORK INCL. LAND COSTS		1,186,255	$26,454	$10,367	$36,821
212	REACTOR BUILDING	$6,212	5,370,892	$133,978	$91,500	$231,690
213	TURBINE BUILDING	$1,442	1,663,217	$42,589	$41,459	$85,490
214	SECURITY BUILDING AND GATE HOUSE	$29	85,619	$2,206	$1,264	$3,499
215	AUXILIARY BUILDING	$20,489	7,968,337	$195,235	$75,872	$291,596
216	RADWASTE FACILITY	$1,162	1,298,201	$32,415	$14,143	$47,720
218B	ADMINISTRATION AND SERVICE BUILDING	$1,260	148,479	$3,802	$3,659	$8,721
218C	ONSITE STANDBY AC POWER GENERATION	$1,129	725,651	$17,724	$7,679	$26,532
218D	FIRE PUMPHOUSE	$57	5,392	$131	$76	$264
218K	PIPE & ELECTRIC TUNNELS		762,411	$18,248	$7,446	$25,694
218R	AUXILIARY BOILER BUILDING		1,846	$49	$140	$189
218T	ULTIMATE HEAT SINK STRUCTURE		1,554,618	$37,775	$25,578	$63,353
218Z	OTHER MISC. STRUCTURES	$649	243,996	$5,949	$3,243	$9,841
21	STRUCTURES & IMPOVEMENTS	$32,429	21,014,914	$516,555	$282,426	$831,410
222	MAIN HEAT TRANSPORT SYSTEM	$704,852	702,895	$17,515	$27,982	$750,349
223	SAFEGUARDS SYSTEM	$5,755	461,449	$11,523	$7,428	$24,706
224	RADWASTE PROCESSING	$4,583	194,962	$4,934	$3,435	$12,952
225	FUEL HANDLING & STORAGE	$5,911	193,914	$4,828	$4,487	$15,226
226	OTHER REACTOR PLANT EQUIPMENT	$9,093	3,004,978	$75,403	$42,923	$127,419
227	REACTOR PLANT INSTRUMENTATION & CONTROL	$1,168	42,774	$1,135	$1,573	$3,876
228	PLANT SIMULATOR	$5,957	17,000	$452		$6,409
22	REACTOR PLANT EQUIPMENT	$737,319	4,617,972	$115,790	$87,828	$940,937
231	TURBINE GENERATOR	$220,477	679,231	$16,942	$13,978	$251,397
233	CONDENSING SYSTEM	$55,275	263,054	$6,686	$4,413	$66,374
234	FEED HEATING SYSTEM	$40,300	454,136	$11,402	$10,942	$62,644
235	OTHER TURBINE PLANT EQUIPMENT	$2,850	572,184	$14,629	$11,044	$28,523
237	TURBINE PLANT MISC. ITEMS	$420	63,698	$1,595	$1,388	$3,403
23	TURBINE PLANT EQUIPMENT	$319,322	2,032,303	$51,254	$41,765	$412,341

TABLE VI. B-8 (con't)
COST ESTIMATE BY EEDB COST ACCOUNT
SPENT FUEL OPTION (SF-2)

THOUSANDS OF JANUARY, 1992 DOLLARS

EEDB ACCT. NUMBER	ACCT. DESCRIPTION	FACTORY EQUIPMENT	SITE LABOR HOURS	SITE LABOR	SITE MATERIAL	TOTAL $
241	SWITCHGEAR		37,861	$1,018	$17,511	$18,529
242	STATION SERVICE EQUIPMENT	$12,918	421,725	$11,148	$64,247	$88,313
243	SWITCHBOARDS		46,274	$1,244	$8,697	$9,941
244	PROTECTIVE EQUIPMENT		100,963	$2,715	$1,971	$4,686
245	ELECT. STRUCT. & WIRING CONTNR		2,515,650	$67,646	$7,422	$75,068
246	POWER & CONTROL WIRING		1,165,275	$31,334	$16,119	$47,453
24	ELECTRIC PLANT EQUIPMENT	$12,918	4,287,748	$115,105	$115,967	$243,990
251	TRANSPORTATION & LIFT EQUIPMENT	$16,294	73,339	$1,797		$18,091
252	AIR WATER & STEAM SERVICE SYSTEMS	$17,080	1,189,712	$30,200	$18,572	$65,852
253	COMMUNICATION & SECURITY SYSTEM		465,795	$12,488	$6,941	$19,429
255	WASTEWATER TREATMENT EQUIPMENT	$175	17,749	$443	$313	$931
256	MAINTENANCE & TEST EQUIPMENT	$7,447	1,744	$44	$16	$7,507
25	MISCELLANEOUS PLANT EQUIPMENT	$40,996	1,748,339	$44,972	$25,842	$111,810
261	STRUCTURES	$3,455	868,934	$21,313	$12,646	$37,414
262	MECHANICAL EQUIPMENT	$30,867	833,679	$20,512	$2,960	$54,339
26	MAIN COND. HEAT REJECT SYSTEM	$34,322	1,702,613	$41,825	$15,606	$91,753
31	FUEL & TARGET HANDLING FACILITY	$270,000	3,807,100	$90,000	$90,000	$450,000
	TOTAL DIRECT COSTS	$1,447,306	39,210,989	$975,501	$659,434	$3,082,241
900	INDIRECT COSTS		12,288,268	$882,991	$1,527,143	$2,410,134
	TOTAL COSTS	$1,447,306	51,499,257	$1,858,492	$2,186,577	$5,492,375

TABLE VI. B-9
CAPITAL COSTS BY UNIT BY DEPLOYMENT OPTION

	OPTION: S-0			OPTION: SF-0			OPTION: D-0			OPTION: SF-1			OPTION: SF-2		
	PDR/FTFF	ECA	TOTAL	PDR/FTFF	ECA	TOTAL	PDR/FTFF	ECA	TOTAL	PDR/FTFF	ECA	TOTAL	PDR/FTFF	ECA	TOTAL
UNIT NO. 1	$3,319	$400	$3,719	$3,070	$400	$3,470	$3,097	$400	$3,497	$3,069	$400	$3,469	$3,070	$400	$3,470
UNIT NO. 2	N/A	N/A	N/A	$1,765	$335	$2,100	$1,765	$335	$2,100	N/A	N/A	N/A	$1,765	$335	$2,100
UNIT NO. 3	N/A	N/A	N/A	$1,756	$320	$2,076	$1,756	$320	$2,076	N/A	N/A	N/A	N/A	N/A	N/A
UNIT NO. 4	N/A	N/A	N/A	$1,752	$306	$2,058	$1,752	$306	$2,058	N/A	N/A	N/A	N/A	N/A	N/A
TOTALS	$3,319	$400	$3,719	$8,343	$1,361	$9,704	$8,370	$1,361	$9,731	$3,069	$400	$3,469	$4,835	$735	$5,570

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TABLE VI.C-1
BASE CASE DEPLOYMENT OPTIONS ANNUAL O&M COST FORMAT
(MILLIONS)

	SPIKING S-0		SPENT FUEL SF-0		DESTRUCTION D-0		SPENT FUEL SF-1		SPENT FUEL SF-2	
	PDR	ECA	PDR	ECA	PDR	ECA	PDR	ECA	PDR	ECA
DIRECT POWER GEN										
On-Site Staff	37.7	10.2	95.4	27.3	95.4	27.3	37.7	10.2	50	14
Maintenance Mat'l										
Fixed	6.7	1.8	16.2	4.7	16.2	4.7	6.4	1.7	8.3	2.3
Variable	4.1	1.1	9.8	2.8	9.8	2.8	3.8	1	4.9	1.4
Supplies & Expenses										
Fixed	2.1	0.6	4.7	1.4	4.7	1.4	1.8	0.5	2.3	0.7
Variable	1.4	0.4	3	0.9	3	0.9	1.1	0.3	1.4	0.4
Off-Site Tech Supp	1.9	2.3	1.9	2.3	1.9	2.3	0.5	0.6	0.9	1.1
PDR/ECA Total		$70.3		170.4		170.4		65.6		87.7
Nuclear Reg. Fees		2.8		11.2		11.2		2.8		5.6
Total O&M Costs		$73.10		$181.60		$181.60		$68.40		$93.30
ADMIN. & GENERAL										
Pension & Benefits		10		25.6		25.6		10		13.3
Nuclear Insur. Prem.		4.2		9.5		9.5		4.2		5.9
Other A&G Expenses		10.2		26.3		26.3		10.2		13.7
Total A&G Costs		$24.40		$61.40		$61.40		$24.40		$32.90
TOTAL ANNUAL COSTS		$97.50		$243.00		$243.00		$92.80		$126.20
Disposal & Decomm.		868		868		217		217		434
Operating Spares		54		112.5		112.5		54		72

TABLE VI.C-2
BASE CASE DEPLOYMENT OPTIONS
OUTAGE COST

			PDR	ECS	TOTAL
I.	CONTRACT SUPPORT				
	A.	Turbine (35 People - 300 hr @ $45.68)	$0	$479,640	$479,640
	B.	Head (23 People - 180 hr @ $45.68)	$189,115	$0	$189,115
	C.	Valves (23 People - 180 hr @ $36.75)	$106,502	$45,644	$152,145
	D.	Steam Generator (25 People - 120 hr @ $57.75)	$173,250	$0	$173,250
II.	IN-HOUSE SUPPORT (Incremental)				
	A.	Maint (230 People - 200 hr @ $34.00)	$1,094,800	$469,200	$1,564,000
	B.	Operations (35 People - 60 hr @ $42.73)	$62,813	$26,920	$89,733
	C.	Security (50 People - 8 hr @ $21.00)	$7,560	$840	$8,400
	D.	QA/QC (15 People - 200 hr @ $32.20)	$96,600	$0	$96,600
			$1,261,773	$496,960	$1,758,733
III.	MATERIALS / DIRECT PURCHASES		$294,209	$173,781	$467,990
		TOTAL OUTAGE COST	$2,024,848	$1,196,025	$3,220,873

IV. TOTAL SITE OUTAGE SCHEDULE

	OPTION	
1.	S-0 Spiking	Average 4 outages per year (Average 4 outages/Unit)
2.	SF-0 Spent Fuel	4 outages per year (1 outage/Unit)
3.	D-0 Destruction	4 outages per year (1 outage/Unit)
4.	SF-1 Spent Fuel	1 outage per year (1 outage/Unit)
5.	SF-2 Spent Fuel	2 outages per year (1 outage/Unit)

NOTE: Outage Costs remain consistent for all deployment options

TABLE VI.C-3
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS: SPIKING S-0

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Executive Management	9	$1,098,000	$0	$1,098,000
Administrative Division				
Regulatory Compliance/OEA	1	$54,000		$0S54,000
Safety/Health	2	$61,800	$20,600	$82,400
Security	125	$3,273,750	$363,750	$3,637,500
Training	36	$1,593,000	$531,000	$2,124,000
Safety	4	$150,000	$50,000	$200,000
Medical	2	$100,000	$0	$100,000
Fire Protection	2	$75,000	$25,000	$100,000
Emergency Planning	2	$108,000	$0	$108,000
Environmental	3	$121,500	$40,500	$162,000
Administrative	12	$388,800	$0	$388,800
Administrative Division Total	**189**	**$5,925,850**	**$1,030,850**	**$6,956,700**
Operations Division				
~~Operations~~	~~110~~	~~$4,862,550~~	~~$2,083,950~~	~~$6,946,500~~
Maintenance Division				
Maintenance	230	$7,627,053	$3,268,737	$10,895,790
Radwaste/Decon	10	$413,000	$0	$413,000
Facilities	2	$44,250	$14,750	$59,000
System Engineering	12	$482,400	$160,800	$643,200
Outage Mgmt	21	$704,025	$234,675	$938,700
QA/QC	15	$670,500	$0	$670,500
Warehouse	8	$228,000	$76,000	$304,000
Maintenance Division Total	**298**	**$10,169,228**	**$3,754,962**	**$13,924,190**
Technical Division				
Plant Tech Support	12	$418,560	$104,640	$523,200
Chemistry	62	$2,701,650	$900,550	$3,602,200
HP/ALARA/Radiation Protection	73	$3,270,400	$0	$3,270,400
Reactor Engineering	2	$125,200	$0	$125,200
Other Non-Nuclear Site	47	$1,142,100	$380,700	$1,522,800
Engineering/Design	18	$723,600	$241,200	$964,800
Licensing	3	$160,800	$0	$160,800

TABLE VI.C-3 (Continued)
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS:
SPIKING S-0

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Technical Division Total	217	$8,542,310	$1,627,090	$10,169,400
Administrative & General	13	$834,000	$0	$834,000
TOTAL	836	$31,431,938	$8,496,852	$39,928,790
Overtime, Supplemental, Etc.		$3,143,194	$849,685	$3,992,879
Social Security, Unemployment		$3,143,194	$849,685	$3,992,879
GRAND TOTAL		$37,718,326	$10,196,222	$47,914,548

TABLE V1.C-4
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS: SPENT FUEL SF- 0

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Executive Management	9	$1,098,000	$0	$1,098,000
Administrative Division				
Regulatory Compliance/OEA	4	$216,000	$0	$216,000
Safety/Health	8	$247,200	$82,400	$329,600
Security	300	$7,857,000	$873,000	$8,730,000
Training	36	$1,593,000	$531,000	$2,124,000
Safety	16	$600,000	$200,000	$800,000
Medical	8	$400,000	$0	$400,000
Fire Protection	8	$300,000	$100,000	$400,000
Emergency PlanninG	8	$432,000	$0	$432,000
Environmental	12	$486,000	$162,000	$648,000
Administrative	15	$486,000	$0	$486,000
Administrative Division Total	415	$12,617,200	$1,948,400	$14,565,600
Operations Division				
Operations	370	$16,355,850	$7,009,650	$23,365,500
Maintenance Division				
Maintenance	580	$19,233,438	$8,242,902	$27,476,340
Radwaste/Decon	40	$1,652,000	$0	$1,652,000
Facilities	4	$88,500	$29,500	$118,000
System Engineering	34	$1,366,800	$455,600	$1,822,400
Outage Mgmt	58	$1,944,450	$648,150	$2,592,600
QA/QC	42	$ 1,877,400	$0	$ 1,877,400
Warehouse	22	$627,000	$209,000	$836,000
Maintenance Division Total	780	$26,789,588	$9,585.152	$36,374,740
Technical Division				
Plant Tech Support	16	$558,080	$139,520	$697,600
Chemistry	164	$7,146,300	$2,382,100	$9,528,400
HP/ALARA/Radiation Protection	186	$8,332,800	$0	$8,332,800
Reactor Engineering	4	$250,400	$0	$250,400
Other Non-Nuclear Site	132	$3,207,600	$1,069,200	$4,276,800
Engineering/Design	44	$1,768,800	$589,600	$2,358,400
Licensing	5	$268,000	$0	$268,000

TABLE V1.C-4 (Continued)
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS:
SPENT FUEL SF- 0

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Technical Division Total	551	$21,531,980	$4,180,420	$25,712,400
Administrative & General	18	$1,137,000	$0	$1,137,000
TOTAL	2143	$79,529,618	$22,723,622	$102,253,240
Overtime, Supplemental, Etc.		$7,952,962	$2,272,362	$10,225,324
Social Security, Unemployment		$7,952,962	$2,272,362	$10,225,324
GRAND TOTAL		$95,435,542	$27,268,346	$122,703,888

TABLE VI.C-5
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS
DESTRUCTION D-0

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Executive Management	9	$1,098,000	$0	$1,098.000
Administrative Division				
Regulatory Compliance/OEA	4	$216,000	$0	$216,000
Safety/Health	8	$247,200	$82,400	$329,600
Security	300	$7,857,000	$873,000	$8,730,000
Training	36	$1,593,000	$531,000	$2,124,000
Safety	16	$600,000	$200,000	$800,000
Medical	8	$400,000	$0	$400,000
Fire Protection	8	$300,000	$100,000	$400,000
Emergency Planning	8	$432,000	$0	$432,000
Environmental	12	$486,000	$ 162,000	$648,000
Administrative	15	$486,000	$0	$486,000
Administrative Division Total	415	$12,617,200	$1,948,400	$14,565,600
Operations Division				
Operations	370	$16,355,850	$7,009,650	$23,365,500
Maintenance Division				
Maintenance	580	$19,233,438	$8,242,902	$27,476,340
Radwaste/Decon	40	$1,652,000	$0	$1,652,000
Facilities	4	$88,500	$29,500	$118,000
System Engineering	34	$1,366,800	$455,600	$1,822,400
Outage Mgmt	58	$1,944,450	$648,150	$2,592,600
QA/QC	42	$1,877,400	$0	$1,877,400
Warehouse	22	$627,000	$209,000	$836,000
Maintenance Division Total	780	$26,789,588	$9,585,152	$36,374,740
Technical Division				
Plant Tech Support	16	$558,080	$139,520	$697,600
Chemistry	164	$7,146,300	$2,382,100	$9,528,400
HP/ALARA/Radiation Protection	186	$8,332,800	$0	$8,332,800
Reactor Engineering	4	$250,400	$0	$250,400
Other Non-Nuclear Site	132	$3,207,600	$1,069,200	$4,276,800

TABLE VI.C-5 (Continued)
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS
DESTRUCTION D-0

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Engineering/Design	44	$1,768,800	$589,600	$2,358,400
Licensing	5	$268,000	$0	$268,000
Technical Division Total	**551**	**$21,531,980**	**$4,180,420**	**$25,712,400**
Administrative & General	**18**	**$1,137,000**	**$0**	**$1,137,000**
TOTAL	**2143**	**$79,529,618**	**$22,723,622**	**$102,253,240**
Overtime, Supplemental, Etc.		$7,952,962	$2,272,362	$10,225,324
Social Security, Unemployment		$7,952,962	$2,272,362	$10,225,324
GRAND TOTAL		**$95,435,542**	**$27,268,346**	**$122,703,888**

TABLE VI.C-6
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS
SPENT FUEL SF-1

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Executive Management	9	$1,098,000	$0	$1,098,000
Administrative Division				
Regulatory Compliance/OEA	1	$54,000	$0	$54,000
Safety/Health	2	S61,800	S20,600	S82,400
Security	125	$3,273,750	$363,750	$3,637,500
Training	36	$1,593,000	$531,000	$2,124,000
Safety	4	$150,000	$50,000	$200,000
Medical	2	$ 100,000	$0	$100,000
Fire Protection	2	S75,000	S25,000	S100,000
Emergency Planning	2	S108,000	$0	$108,000
Environmental	3	$121,500	$40,500	$162,000
Administrative	12	S388,800	SO	S388,800
Administrative Division Total	189	$5,925,850	$1,030,850	$6,956,700
Operations Division				
Operations	110	$4,862,550	$2,083,950	$6,946,500
Maintenance Division				
Maintenance	230	S7,627,053	S3,268,737	S10,895,790
Radwaste/Decon	10	S413,000	$0	S413,000
Facilities	2	S44,250	S14,750	S59,000
System Engineering	12	S482,400	S160,800	S643,200
Outage Mgmt	21	S704,025	S234,675	S938,700
QA/QC	15	$670,500	$0	$670,500
Warehouse	8	$228,000	$76,000	$304,000
Maintenance Division Total	298	$10,169,228	$3,754,962	$13,924,190
Technical Division				
Plant Tech Support	12	S418,560	S104,640	S523,200
Chemistry	62	S2,701,650	S900,550	S3,602,200
HP/ALARA/Radiation Protection	73	$3,270,400	$0	$3,270,400
Reactor Engineering	2	S125,200	SO	S125,200
Other Non-Nuclear Site	47	$1,142,100	$380,700	$1,522,800
Enpineering/Design	18	$723,600	$241,200	$964,800
Licensing	3	$160,800	$0	$160,800

TABLE VI.C-6 (Continued)
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS
SPENT FUEL SF-1

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Technical Division Total	217	$8,542,310	$1,627,090	$10,169,400
Administrative & General	13	$834,000	$0	$834.000
TOTAL	836	$31,431,938	$8,496,852	$39,928,790
Overtime, Supplemental, Etc.		$3,143,194	$849,685	$3,992,879
Social Security, Unemployment		$3,143,194	$849,685	$3,992,879
GRAND TOTAL		$37,718,326	$10,196,222	$47,914,548

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TABLE VI.C-7
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS
SPENT FUEL SF-2

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Executive Management	9	$1,098,000	$0	$1,098,000
Administrative Division				
Regulatory Compliance/OEA	2	$108,000	$0	$108,000
Safety/Health	4	$123,600	$41,200	$164,800
Security	150	$3,928,500	$436,500	$4,365,000
Training	36	$1,593,000	$531,000	$2,124,000
Safety	8	$300,000	$100,000	$400,000
Medical	4	$200,000	$0	$200,000
Fire Protection	4	$150,000	$50,000	$200,000
Emergency Planning	4	$216,000	$0	$216,000
Environmental	6	$243,000	$81,000	$324,000
Administrative	12	$388,000	$0	$388,000
Administrative Division Total	230	$7,250,100	$1,239,700	$8,489,800
Operations Division				
Operations	185	$8,177,925	$3,504,825	$11,682.750
Maintenance Division				
Maintenance	290	$9,616,719	$4,121,451	$13,738,170
Radwaste/Decon	20	$826,000	$0	$826,000
Facilities	2	$44,250	$14,750	$59,000
System Engineering	17	$683,400	$227,800	$911,200
Outage Mgmt	29	$972,225	$324,075	$1,296,300
QA/QC	21	$938,700	$0	$938,700
Warehouse	11	$313,500	$104,500	$418,000
Maintenance Division Total	390	$13,394,794	$4,792,576	$18,187,370
Technical Division				
Plant Tech Support	12	$418,560	$104,640	$523,200
Chemistry	82	$3,573,150	$1,191,050	$4,764,200
HP/ALARA/Radiation Protection	93	$4,166,400	$0	$4,166,400
Reactor Engineering	2	$125,200	$0	$125,200
Other Non-Nuclear Site	66	$1,603,800	$534,600	$2,138,400
Engineering/Design	22	$884,400	$294,800	$1,179,200
Licensing	3	$160,800	$0	$160,800

TABLE VI.C-7 (Continued)
BASE CASE DEPLOYMENT OPTIONS STAFFING REQUIREMENTS
SPENT FUEL SF-2

CATEGORY	NO OF PERSONS	PDR LABOR COST	ECA LABOR COST	TOTAL LABOR COST
Technical Division Total	**280**	**$10,932,310**	**$2,125,090**	**$13,057,400**
Administrative & General	**13**	**$834,000**	**$0**	**$834,000**
TOTAL	**1107**	**$41,687,129**	**$11,662,191**	**$53,349,320**
Overtime, Supplemental, Etc.		$4,168,713	$1,166,219	$5,334,932
Social Security, Unemployment		$4,168,713	$1,166,219	$5,334,932
GRAND TOTAL		**$50,024,555**	**$13,994,629**	**$64,019,184**

TABLE VI.E-1
BASE CASE DEPLOYMENT OPTIONS - ANNUAL REVENUE REQUIREMENTS
ENERGY CONVERSION AREA: SPIKING S-O

YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)
2000	4,906,634	$27.55	$135.18
2001	4,906,634	$28.01	$137.43
2002	4,906,634	$28.38	$139.25
2003	4,906,634	$28.76	$141.11
2004	4,906,634	$29.14	$142.98
2005	4,906,634	$29.54	$144.94
2006	4,906,634	$29.95	$146.95
2007	4,906,634	$30.37	$149.01
2008	4,906,634	$30.80	$151.12
2009	4,906,634	$31.24	$153.28
2010	4,906,634	$31.69	$155.49
2011	4,906,634	$31.98	$156.91
2012	4,906,634	$32.27	$158.34
2013	4,906,634	$32.57	$159.81
2014	4,906,634	$32.87	$161.28
2015	4,906,634	$33.17	$162.75
2016	4,906,634	$33.48	$164.27
2017	4,906,634	$33.78	$165.75
2018	4,906,634	$34.10	$167.32
2019	4,906,634	$34.42	$168.89
2020	4,906,634	$34.74	$170.46
2021	4,906,634	$35.07	$172.08
2022	4,906,634	$35.40	$173.69
2023	4,906,634	$35.74	$175.36
2024	4,906,634	$36.07	$176.98
2025	4,906,634	$36.42	$178.70
2026	4,906,634	$36.76	$180.37
2027	4,906,634	$37.12	$182.13
2028	4,906,634	$37.47	$183.85
2029	4,906,634	$37.83	$185.62

TABLE VI.E-2
BASE CASE DEPLOYMENT OPTIONS - ANNUAL REVENUE REQUIREMENTS ENERGY CONVERSION AREA: SPENT FUEL SF-0

YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)	YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)
2000	34,232,328	$27.55	$943.10	2030	34,232,328	$38.20	$1,307.67
2001	34,232,328	$28.01	$958.85	2031	34,232,328	$38.57	$1,320.34
2002	34,232,328	$28.38	$971.51	2032	34,232,328	$38.94	$1,333.01
2003	34,232,328	$28.76	$984.52	2033	34,232,328	$39.32	$1,346.02
2004	34,232,328	$29.14	$997.53	2034	34,232,328	$39.70	$1,359.02
2005	34,232,328	$29.54	$1,011.22	2035	34,232,328	$40.00	$1,369.29
2006	34,232,328	$29.95	$1,025.26	2036	34,232,328	$40.48	$1,385.72
2007	34,232,328	$30.37	$1,039.64	2037	34,232,328	$40.88	$1,399.42
2008	34,232,328	$30.80	$1,054.36	2038	34,232,328	$41.28	$1,413.11
2009	34,232,328	$31.24	$1,069.42	2039	34,232,328	$41.69	$1,427.15
2010	34,232,328	$31.69	$1,084.82	2040	34,232,328	$42.10	$1,441.18
2011	34,232,328	$31.98	$1,094.75	2041	34,232,328	$42.52	$1,455.56
2012	34,232,328	$32.27	$1,104.68	2042	34,232,328	$42.94	$1,469.94
2013	34,232,328	$32.57	$1,114.95	2043	34,232,328	$43.37	$1,484.66
2014	34,232,328	$32.87	$1,125.22	2044	34,232,328	$43.81	$1,499.72
2015	34,232,328	$33.17	$1,135.49	2045	34,232,328	$44.24	$1,514.44
2016	34,232,328	$33.48	$1,146.10	2046	34,232,328	$44.69	$1,529.84
2017	34,232,328	$33.78	$1,156.37	2047	34,232,328	$45.14	$1,545.25
2018	34,232,328	$34.10	$1,167.32	2048	34,232,328	$45.59	$1,560.65
2019	34,232,328	$34.42	$1,178.28	2049	34,232,328	$46.06	$1,576.74
2020	34,232,328	$34.74	$1,189.23	2050	34,232,328	$46.52	$1,592.49
2021	34,232,328	$35.07	$1,200.53	2051	34,232,328	$47.00	$1,608.92
2022	34,232,328	$35.40	$1,211.82	2052	34,232,328	$47.47	$1,625.01
2023	34,232,328	$35.74	$1,223.46	2053	34,232,328	$47.98	$1,642.47
2024	34,232,328	$36.07	$1,234.76	2054	34,232,328	$48.46	$1,658.56
2025	34,232,328	$36.42	$1,246.74	2055	34,232,328	$48.95	$1,675.67
2026	34,232,328	$36.76	$1,258.38	2056	34,232,328	$49.45	$1,692.79
2027	34,232,328	$37.12	$1,270.70	2057	34,232,328	$49.96	$1,710.25
2028	34,232,328	$37.47	$1,282.69	2058	34,232,328	$50.48	$1,728.06
2029	34,232,328	$37.83	$1,295.01	2059	34,232,328	$51.00	$1,745.85

TABLE VI.E-3
BASE CASE DEPLOYMENT OPTIONS - ANNUAL REVENUE REQUIREMENTS ENERGY CONVERSION AREA: DESTRUCTION D-O

YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)	YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)
2000	34,232,328	$27.55	$943.10	2030	34,232,328	$38.20	$1,307.67
2001	34,232,328	$28.01	$958.85	2031	34,232,328	$38.57	$1,320.34
2002	34,232,328	$28.38	$971.51	2032	34,232,328	$38.94	$1,333.01
2003	34,232,328	$28.76	$984.52	2033	34,232,328	$39.32	$1,346.02
2004	34,232,328	$29.14	$997.53	2034	34,232,328	$39.70	$1,359.02
2005	34,232,328	$29.54	$1,011.22	2035	34,232,328	$40.00	$1,369.29
2006	34,232,328	$29.95	$1,025.26	2036	34,232,328	$40.48	$1,385.72
2007	34,232,328	$30.37	$1,039.64	2037	34,232,328	$40.88	$1,399.42
2008	34,232,328	$30.80	$1,054.36	2038	34,232,328	$41.28	$1,413.11
2009	34,232,328	$31.24	$1,069.42	2039	34,232,328	$41.69	$1,427.15
2010	34,232,328	$31.69	$1,084.82	2040	34,232,328	$42.10	$1,441.18
2011	34,232,328	$31.98	$1,094.75	2041	34,232,328	$42.52	$1,455.56
2012	34,232,328	$32.27	$1,104.68	2042	34,232,328	$42.94	$1,469.94
2013	34,232,328	$32.57	$1,114.95	2043	34,232,328	$43.37	$1,484.66
2014	34,232,328	$32.87	$1,125.22	2044	34,232,328	$43.81	$1,499.72
2015	34,232,328	$33.17	$1,135.49	2045	34,232,328	$44.24	$1,514.44
2016	34,232,328	$33.48	$1,146.10	2046	34,232,328	$44.69	$1,529.84
2017	34,232,328	$33.78	$1,156.37	2047	34,232,328	$45.14	$1,545.25
2018	34,232,328	$34.10	$1,167.32	2048	34,232,328	$45.59	$1,560.66
2019	34,232,328	$34.42	$1,178.28	2049	34,232,328	$46.06	$1,576.74
2020	34,232,328	$34.74	$1,189.23	2050	34,232,328	$46.52	$1,592.49
2021	34,232,328	$35.07	$1,200.53	2051	34,232,328	$47.00	$1,608.92
2022	34,232,328	$35.40	$1,211.82	2052	34,232,328	$47.47	$1,625.01
2023	34,232,328	$35.74	$1,223.46	2053	34,232,328	$47.98	$1,642.47
2024	34,232,328	$36.07	$1,234.78	2054	34,232,328	$48.45	$1,658.56
2025	34,232,328	$36.42	$1,246.74	2055	34,232,328	$48.95	$1,675.67
2026	34,232,328	$36.76	$1,258.38	2056	34,232,328	$49.45	$1,692.79
2027	34,232,328	$37.12	$1,270.70	2057	34,232,328	$49.96	$1,710.25
2028	34,232,328	$37.47	$1,282.69	2058	34,232,328	$50.48	$1,728.05
2029	34,232,328	$37.83	$1,295.01	2059	34,232,328	$51.00	$1,745.85

TABLE VI.E-4
BASE CASE DEPLOYMENT OPTIONS - ANNUAL REVENUE REQUIREMENTS ENERGY CONVERSION AREA: SPENT FUEL SF-1

YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)	YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)
2000	9,128,621	$27.55	$251.49	2030	9,128,621	$38.20	$348.71
2001	9,128,621	$28.01	$255.69	2031	9,128,621	$38.57	$352.09
2002	9,128,621	$28.38	$259.07	2032	9,128,621	$38.94	$355.47
2003	9,128,621	$28.76	$262.54	2033	9,128,621	$39.32	$358.94
2004	9,128,621	$29.14	$266.01	2034	9,128,621	$39.70	$362.41
2005	9,128,621	$29.54	$269.66	2035	9,128,621	$40.00	$365.14
2006	9,128,621	$29.95	$273.40	2036	9,128,621	$40.48	$369.53
2007	9,128,621	$30.37	$277.24	2037	9,128,621	$40.88	$373.18
2008	9,128,621	$30.80	$281.16	2038	9,128,621	$41.28	$376.83
2009	9,128,621	$31.24	$285.18	2039	9,128,621	$41.69	$380.57
2010	9,128,621	$31.69	$289.29	2040	9,128,621	$42.10	$384.31
2011	9,128,621	$31.98	$291.93	2041	9,128,621	$42.52	$388.15
2012	9,128,621	$32.27	$294.58	2042	9,128,621	$42.94	$391.98
2013	9,128,621	$32.57	$297.32	2043	9,128,621	$43.37	$395.91
2014	9,128,621	$32.87	$300.06	2044	9,128,621	$43.81	$399.92
2015	9,128,621	$33.17	$302.80	2045	9,128,621	$44.24	$403.85
2016	9,128,621	$33.48	$305.63	2046	9,128,621	$44.69	$407.96
2017	9,128,621	$33.78	$308.36	2047	9,128,621	$45.14	$412.07
2018	9,128,621	$34.10	$311.29	2048	9,128,621	$45.59	$416.17
2019	9,128,621	$34.42	$314.21	2049	9,128,621	$46.06	$420.46
2020	9,128,621	$34.74	$317.13	2050	9,128,621	$46.52	$424.66
2021	9,128,621	$35.07	$320.14	2051	9,128,621	$47.00	$429.05
2022	9,128,621	$35.40	$323.15	2052	9,128,621	$47.47	$433.34
2023	9,128,621	$35.74	$326.26	2053	9,128,621	$47.98	$437.99
2024	9,128,621	$36.07	$329.27	2054	9,128,621	$48.45	$442.28
2025	9,128,621	$36.42	$332.46	2055	9,128,621	$48.95	$446.85
2026	9,128,621	$36.76	$335.57	2056	9,128,621	$49.45	$451.41
2027	9,128,621	$37.12	$338.85	2057	9,128,621	$49.96	$456.07
2028	9,128,621	$37.47	$342.05	2058	9,128,621	$50.48	$460.81
2029	9,128,621	$37.83	$345.34	2059	9,128,621	$51.00	$465.56

TABLE VI.E-5
BASE CASE DEPLOYMENT OPTIONS - ANNUAL REVENUE REQUIREMENTS ENERGY CONVERSION AREA: SPENT FUEL SF-2

YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)	YEAR	ANNUAL NET GEN MWe	COST -1992 DOLLARS per MWe	ANNUAL REVENUE (MILLIONS)
2000	18,257,242	$27.55	$502.99	2030	18,257,242	$38.20	$697.43
2001	18,257,242	$28.01	$511.39	2031	18,257,242	$38.57	$704.18
2002	18,257,242	$28.38	$518.14	2032	18,257,242	$38.94	$710.94
2003	18,257,242	$28.76	$525.08	2033	18,257,242	$39.32	$717.87
2004	18,257,242	$29.14	$532.02	2034	18,257,242	$39.70	$724.81
2005	18,257,242	$29.54	$539.32	2035	18,257,242	$40.00	$730.29
2006	18,257,242	$29.95	$546.80	2036	18,257,242	$40.48	$739.05
2007	18,257,242	$30.37	$554.47	2037	18,257,242	$40.88	$746.36
2008	18,257,242	$30.80	$562.32	2038	18,257,242	$41.28	$753.66
2009	18,257,242	$31.24	$570.36	2039	18,257,242	$41.69	$761.14
2010	18,257,242	$31.69	$578.57	2040	18,257,242	$42.10	$768.63
2011	18,257,242	$31.98	$583.87	2041	18,257,242	$42.52	$776.30
2012	18,257,242	$32.27	$589.16	2042	18,257,242	$42.94	$783.97
2013	18,257,242	$32.57	$594.64	2043	18,257,242	$43.37	$791.82
2014	18,257,242	$32.87	$600.12	2044	18,257,242	$43.81	$799.85
2015	18,257,242	$33.17	$605.59	2045	18,257,242	$44.24	$807.70
2016	18,257,242	$33.48	$611.26	2046	18,257,242	$44.69	$815.92
2017	18,257,242	$33.78	$616.73	2047	18,257,242	$45.14	$824.13
2018	18,257,242	$34.10	$622.57	2048	18,257,242	$45.59	$832.35
2019	18,257,242	$34.42	$628.41	2049	18,257,242	$46.06	$840.93
2020	18,257,242	$34.74	$634.26	2050	18,257,242	$46.52	$849.33
2021	18,257,242	$35.07	$640.28	2051	18,257,242	$47.00	$858.09
2022	18,257,242	$35.40	$646.31	2052	18,257,242	$47.47	$866.67
2023	18,257,242	$35.74	$652.51	2053	18,257,242	$47.98	$875.98
2024	18,257,242	$36.07	$658.54	2054	18,257,242	$48.45	$884.56
2025	18,257,242	$36.42	$664.93	2055	18,257,242	$48.95	$893.69
2026	18,257,242	$36.76	$671.14	2056	18,257,242	$49.45	$902.82
2027	18,257,242	$37.12	$677.71	2057	18,257,242	$49.96	$912.13
2028	18,257,242	$37.47	$684.10	2058	18,257,242	$50.48	$921.63
2029	18,257,242	$37.83	$690.67	2059	18,257,242	$51.00	$931.12

FIGURE VI. B-1
CASH FLOW
FOR S-0 DEPLOYMENT OPTION

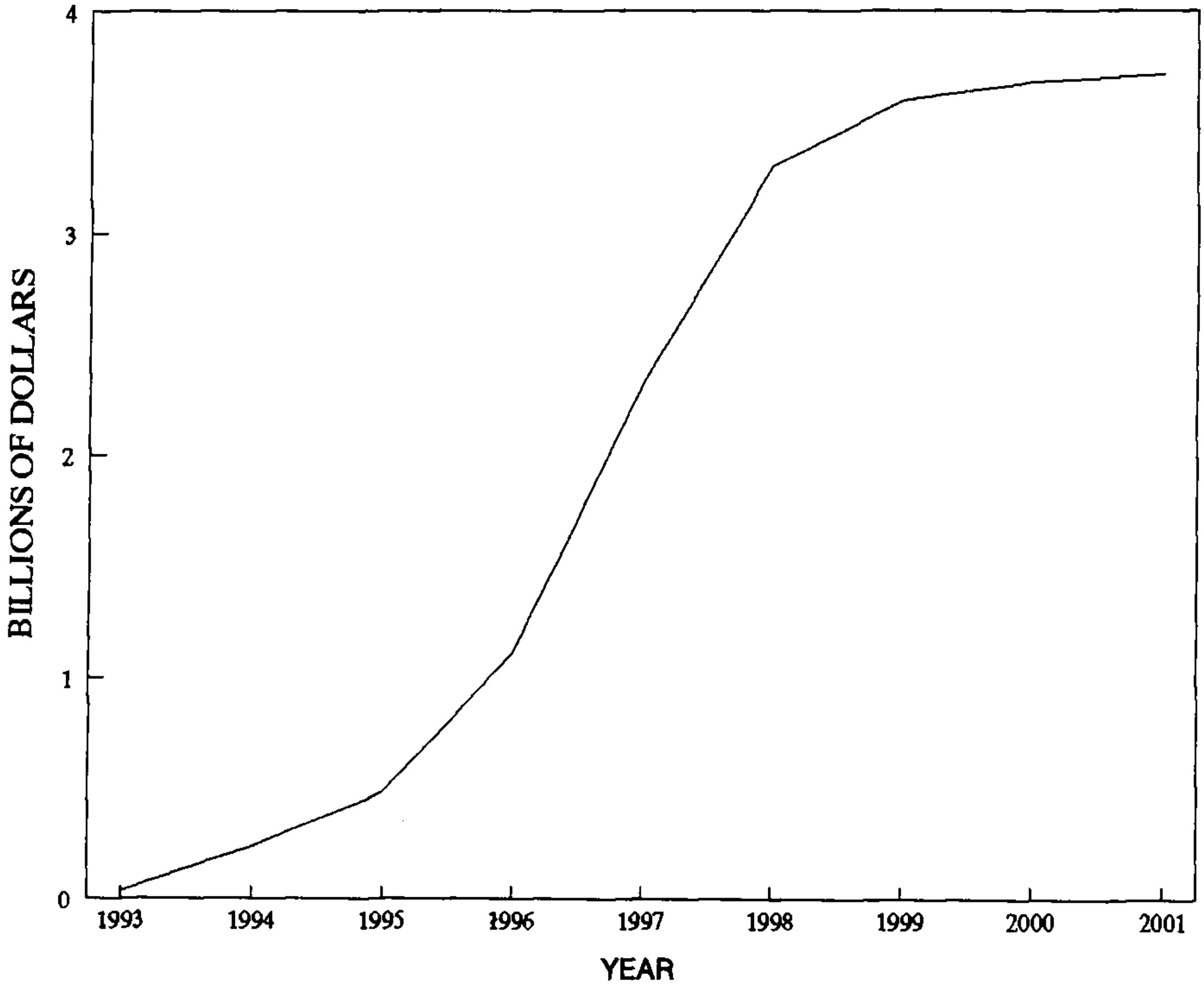


FIGURE VI. B-2
CAPITAL COST
FOR S-0 DEPLOYMENT OPTION

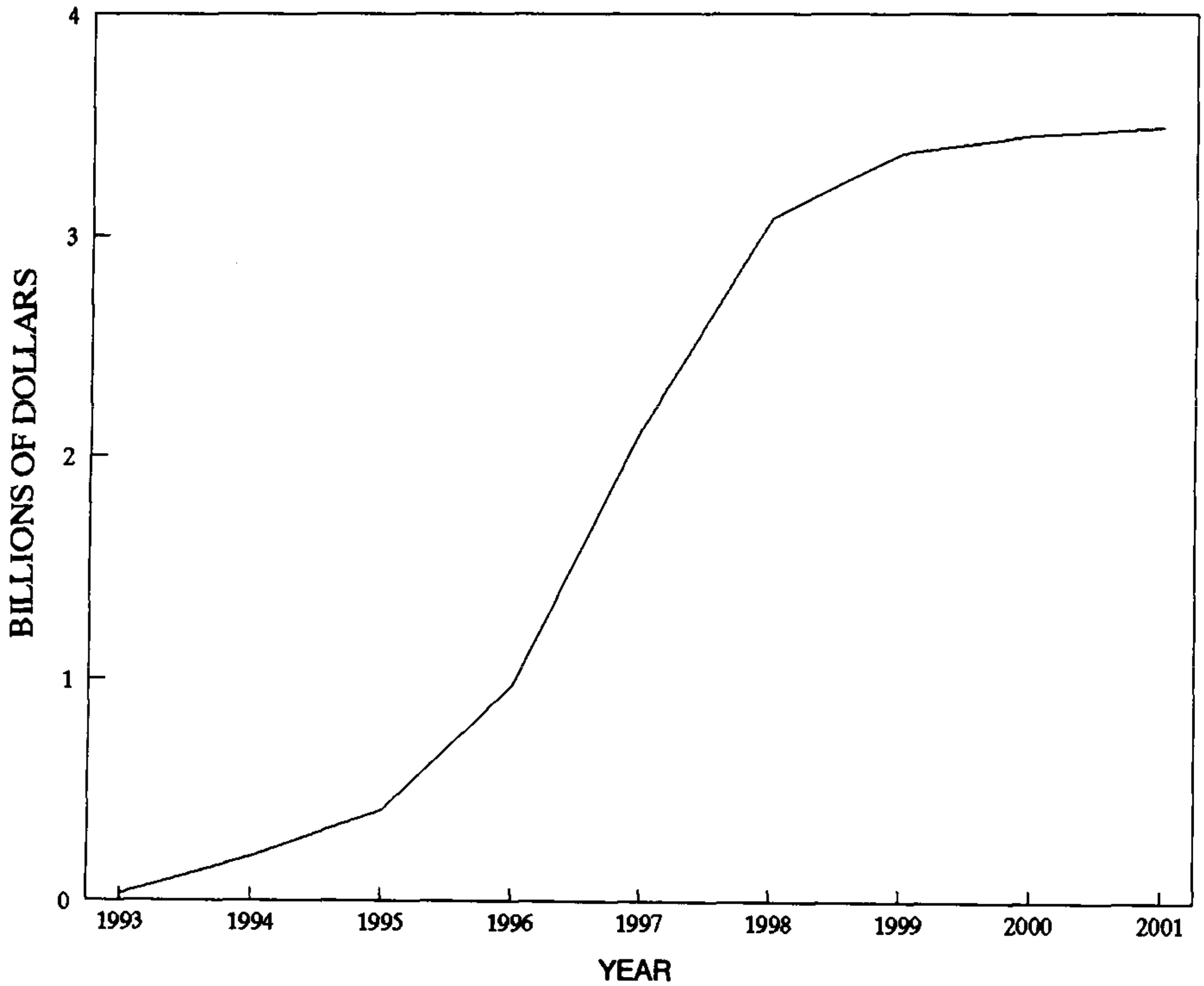


FIGURE VI. B-3
CASH FLOW
FOR SF-1 DEPLOYMENT OPTION

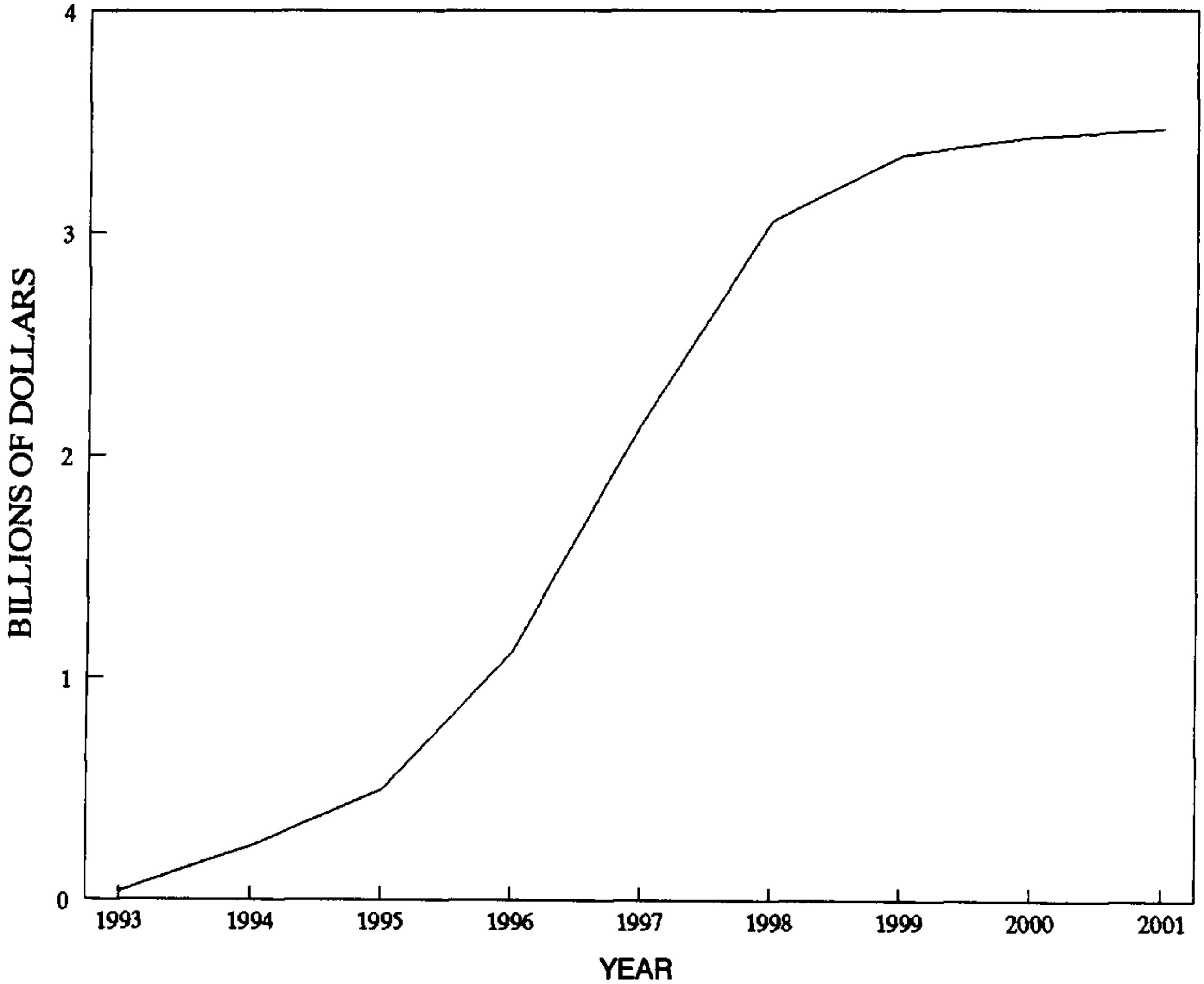


FIGURE VI. B-4
CAPITAL COST
FOR SF-1 DEPLOYMENT OPTION

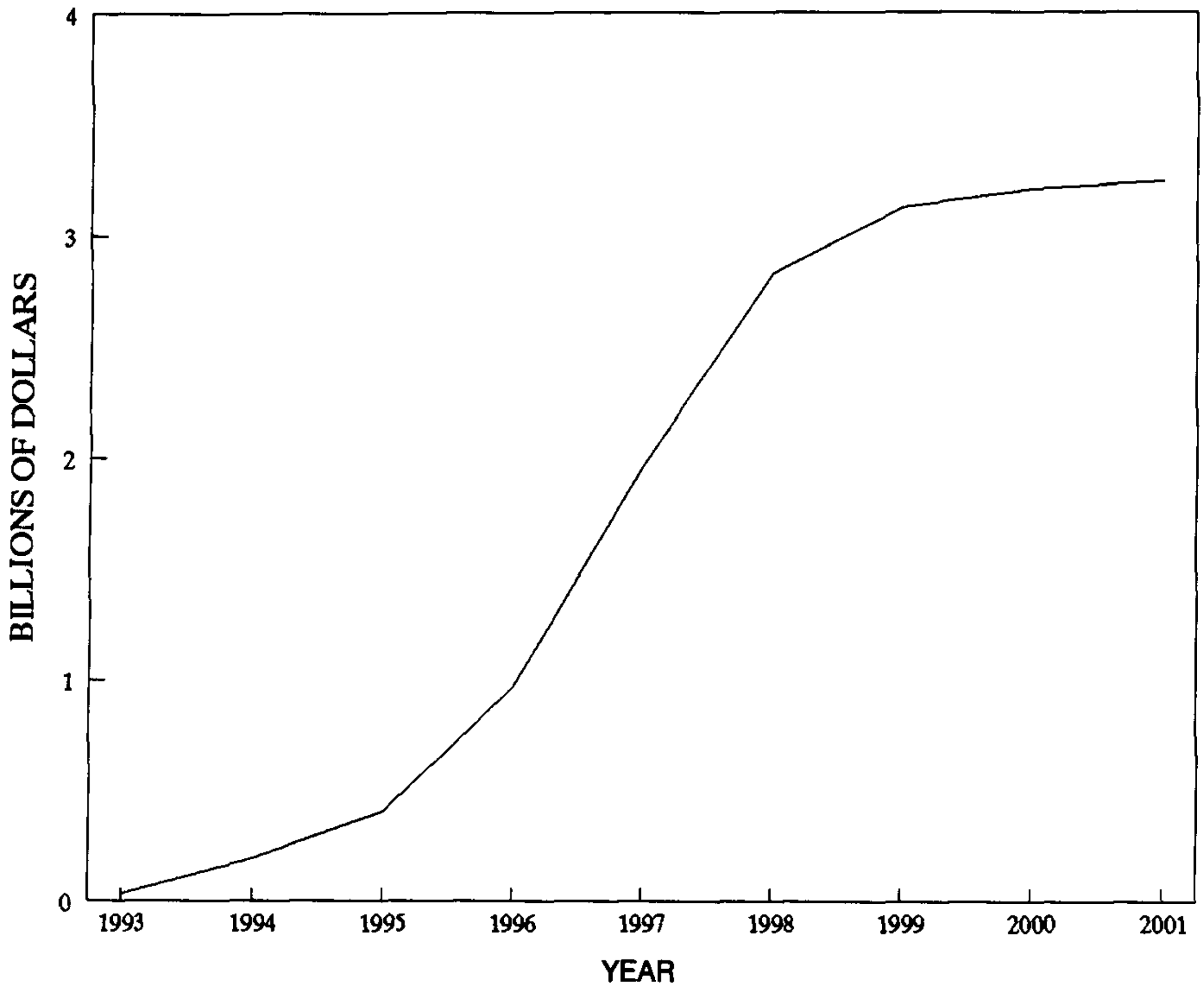


FIGURE VI. B–5
CASH FLOW
FOR SF–2 DEPLOYMENT OPTION

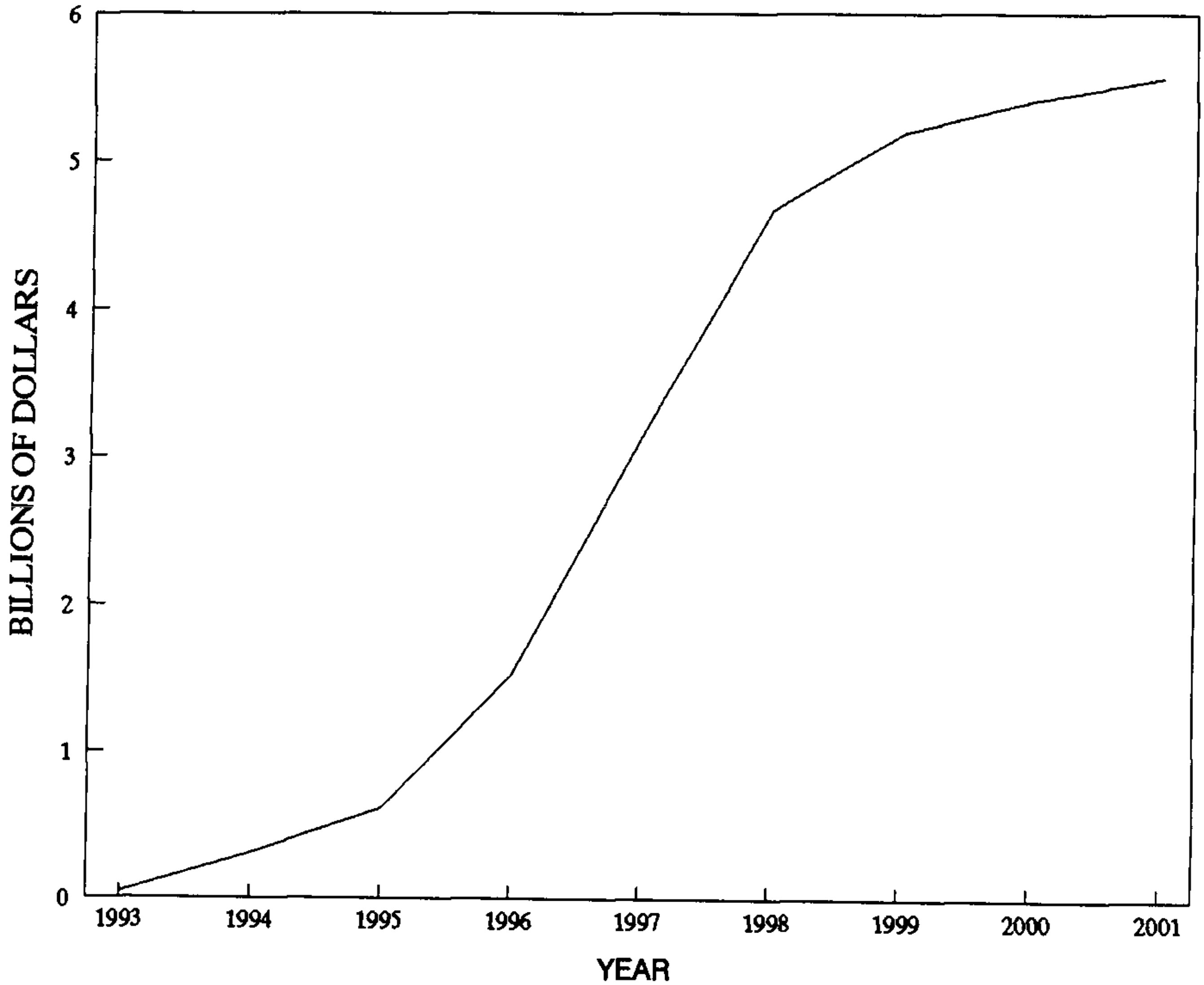


FIGURE VI. B-6
CAPITAL COST
FOR SF-2 DEPLOYMENT OPTION

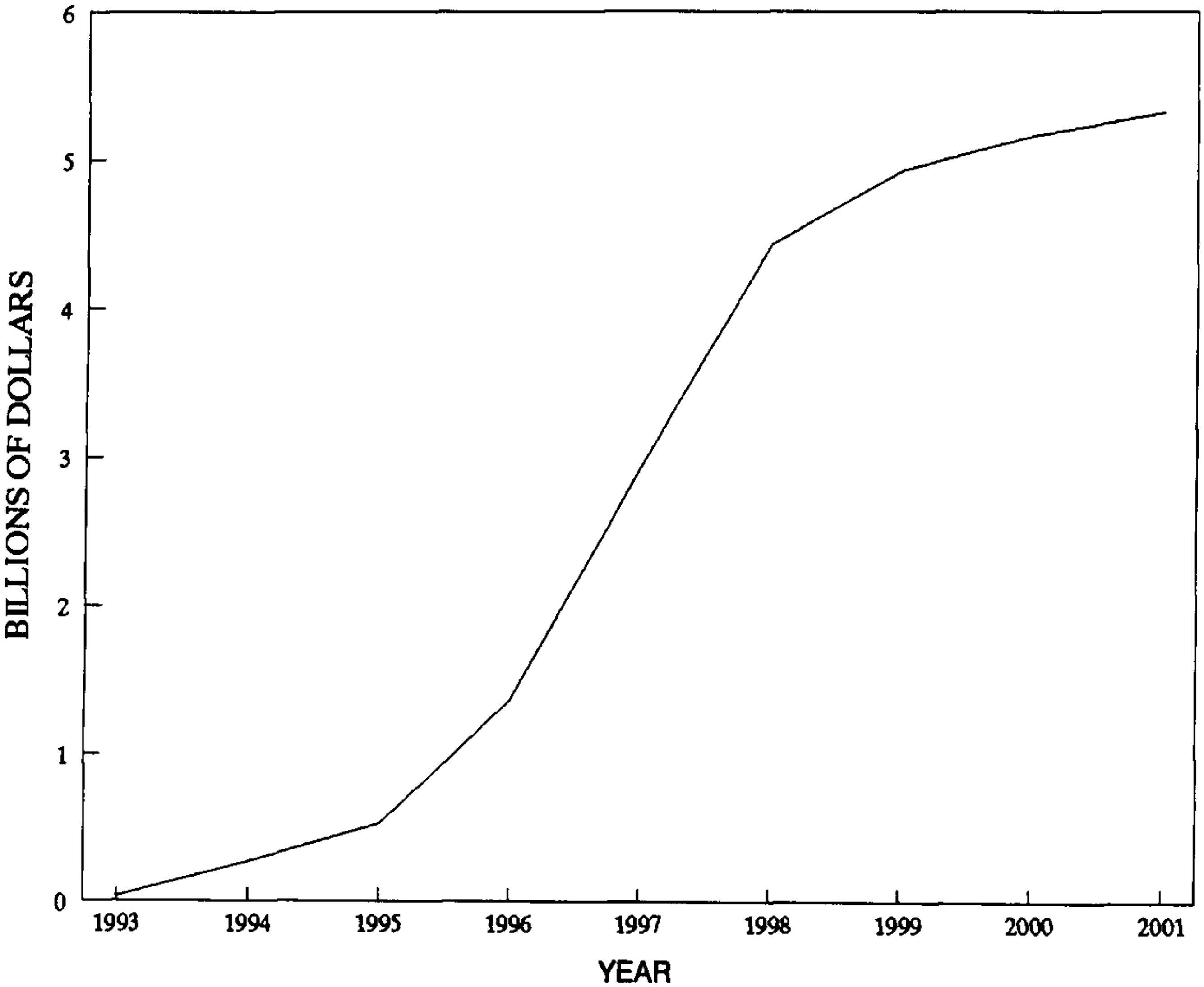


FIGURE VI. B-7
CASH FLOW
FOR SF-0 DEPLOYMENT OPTION

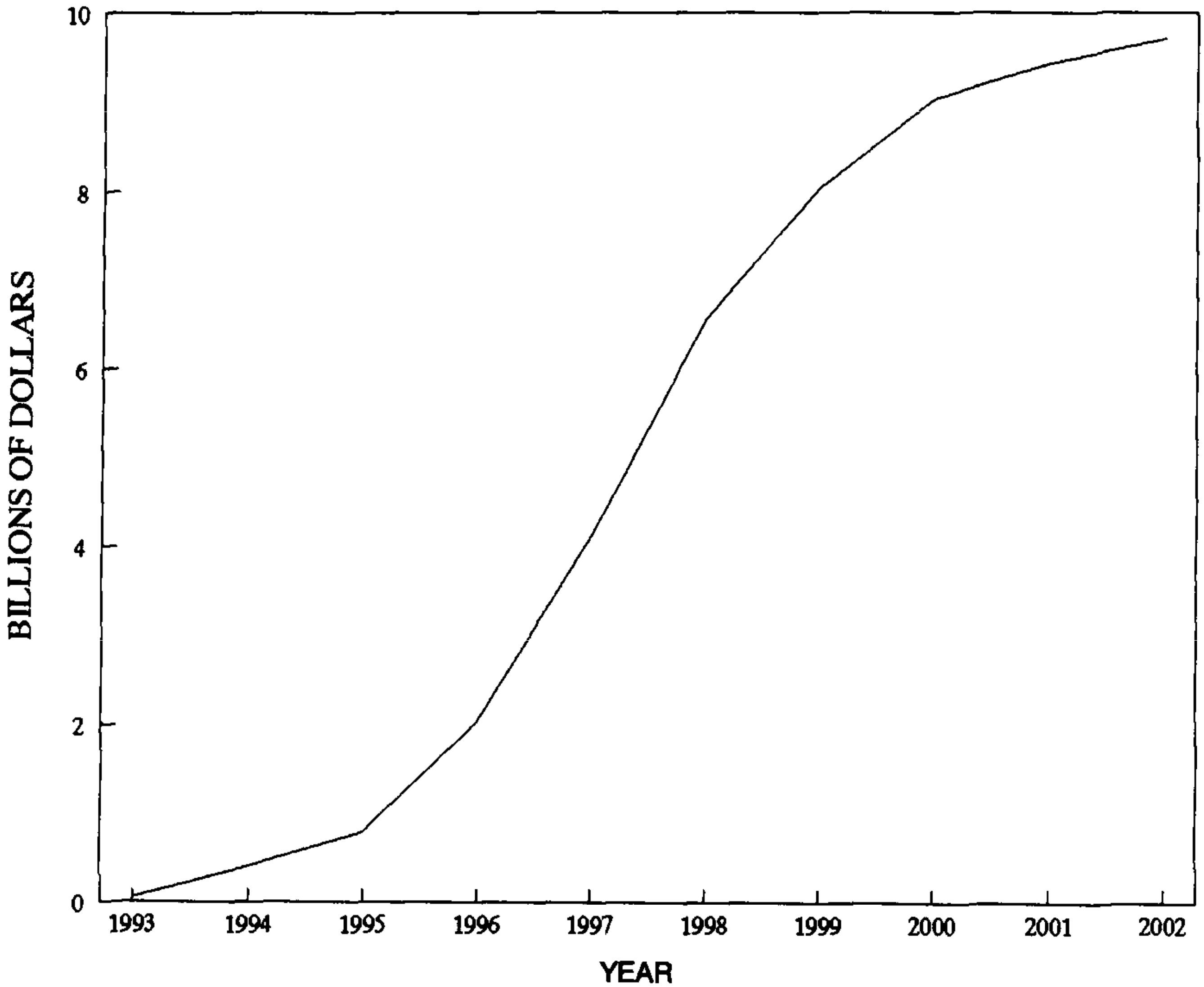


FIGURE VI. B-8
CAPITAL COST
FOR SF-0 DEPLOYMENT OPTION

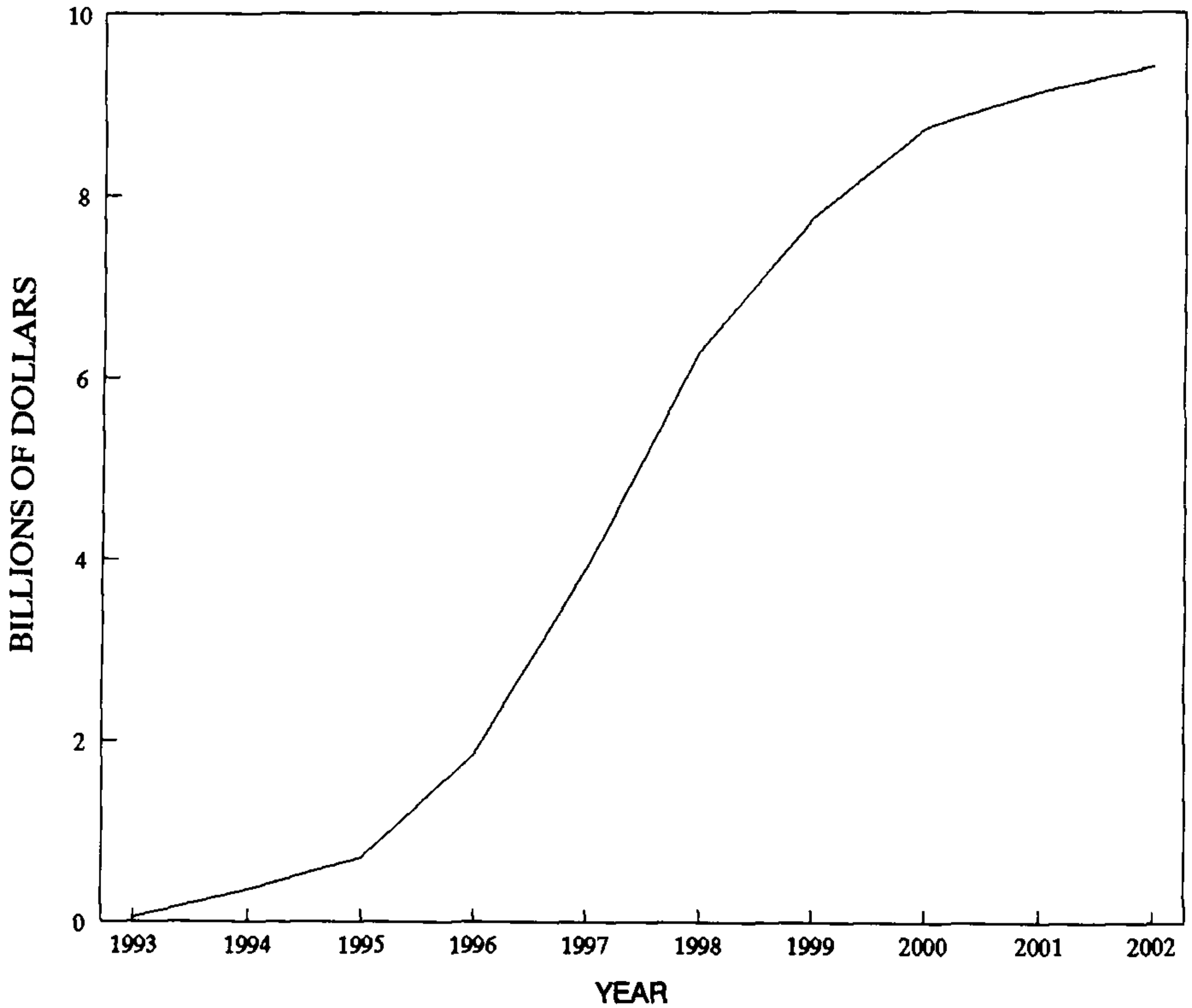


FIGURE VI. B-9
CASH FLOW
FOR D-0 DEPLOYMENT OPTION

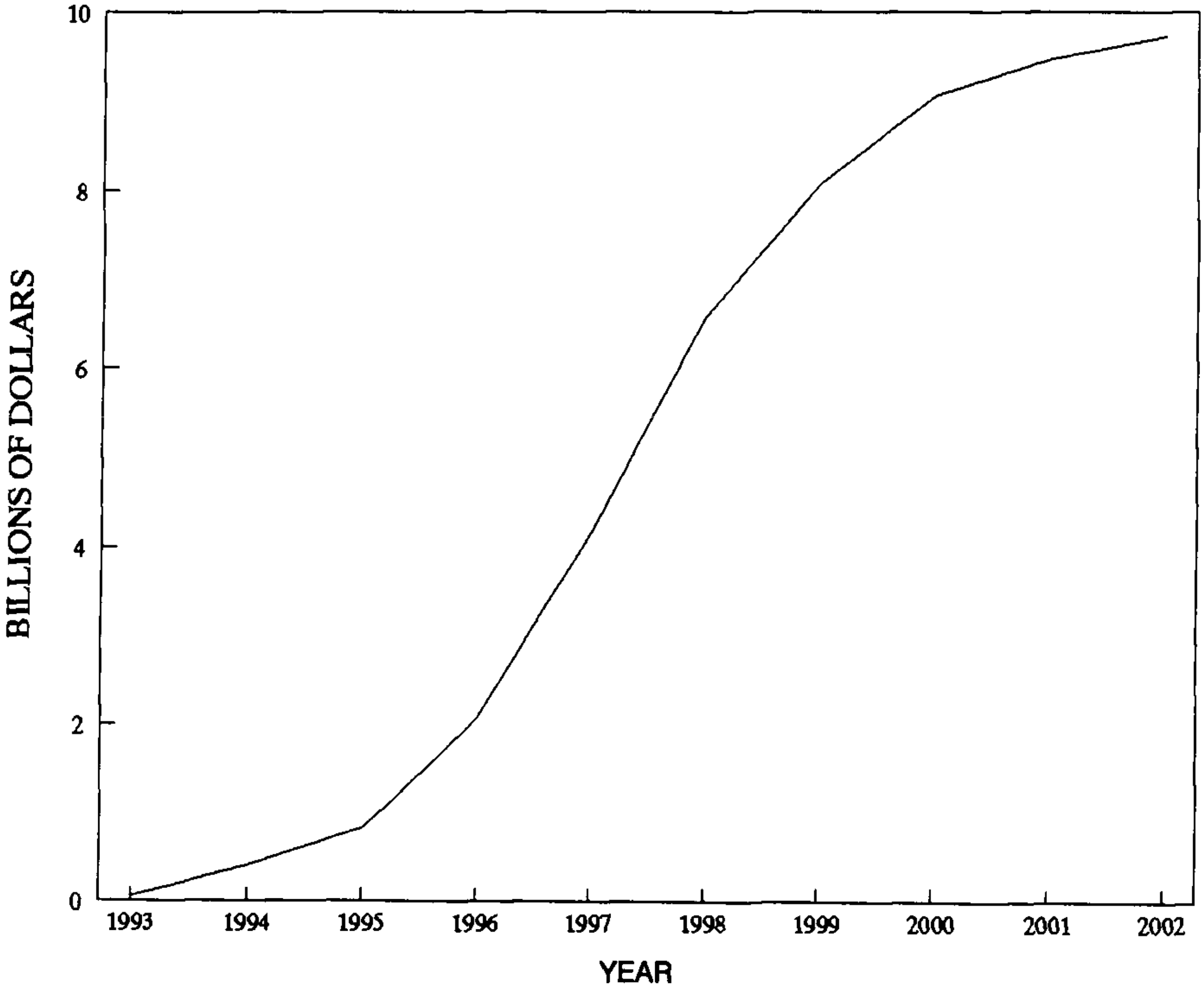


FIGURE VI. B-10
CAPITAL COST
FOR D-0 DEPLOYMENT OPTION

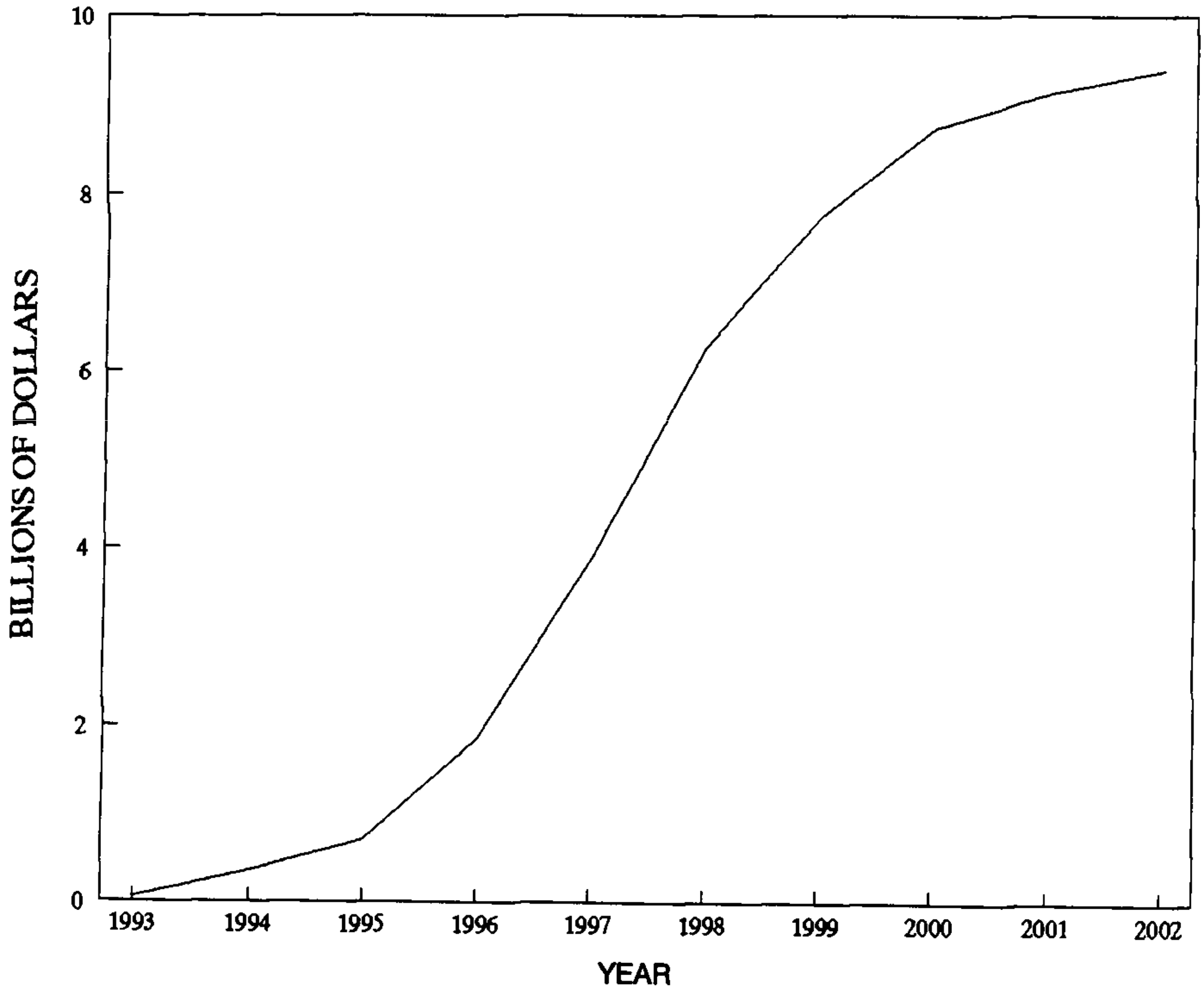


FIGURE VI. B-11
R & D COSTS
FOR S-0, SF-0, SF-1 & SF-2 DEPLOYMENT OPTIONS

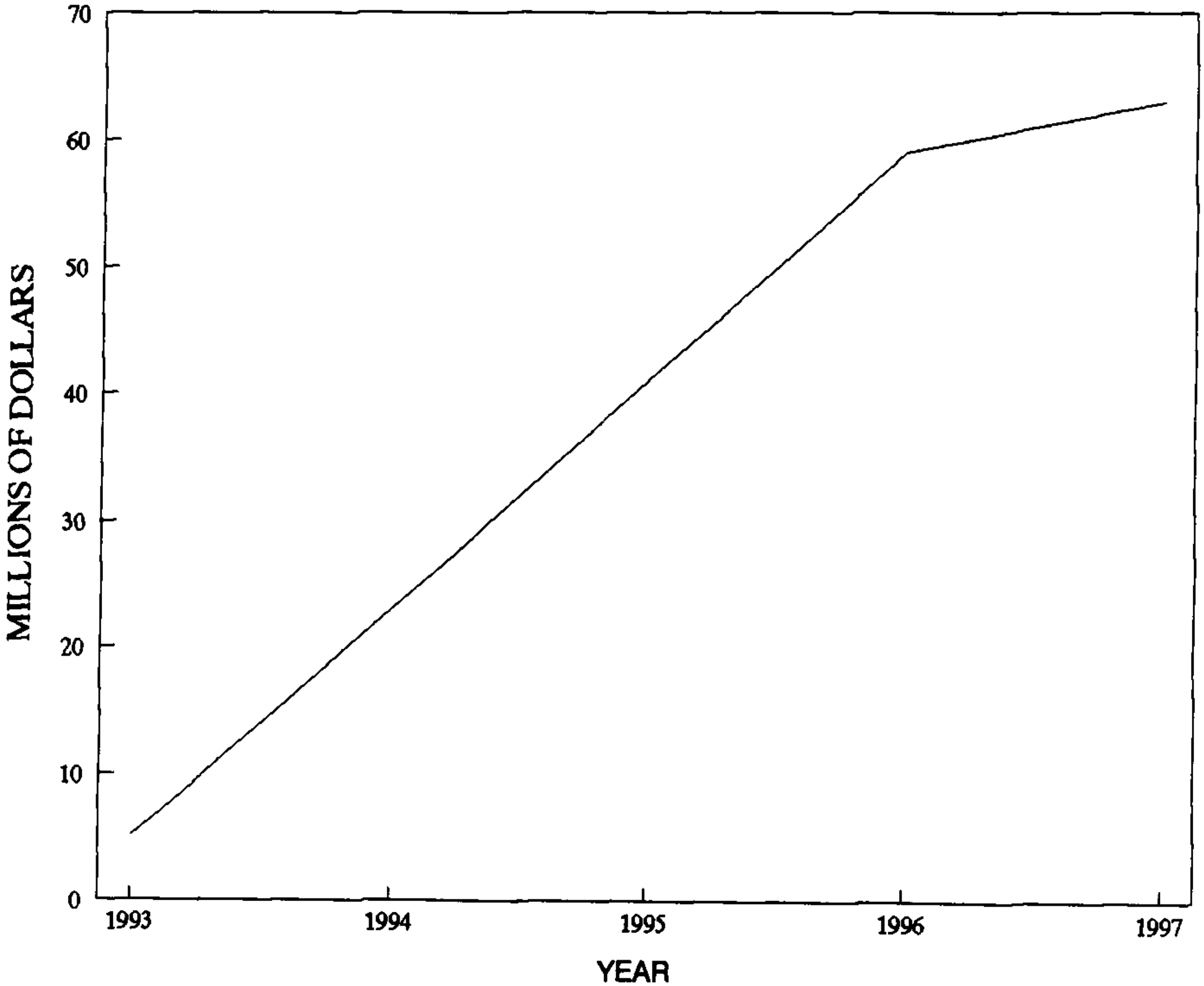


FIGURE VI. B-12
R & D COSTS
FOR D-0 DEPLOYMENT OPTION

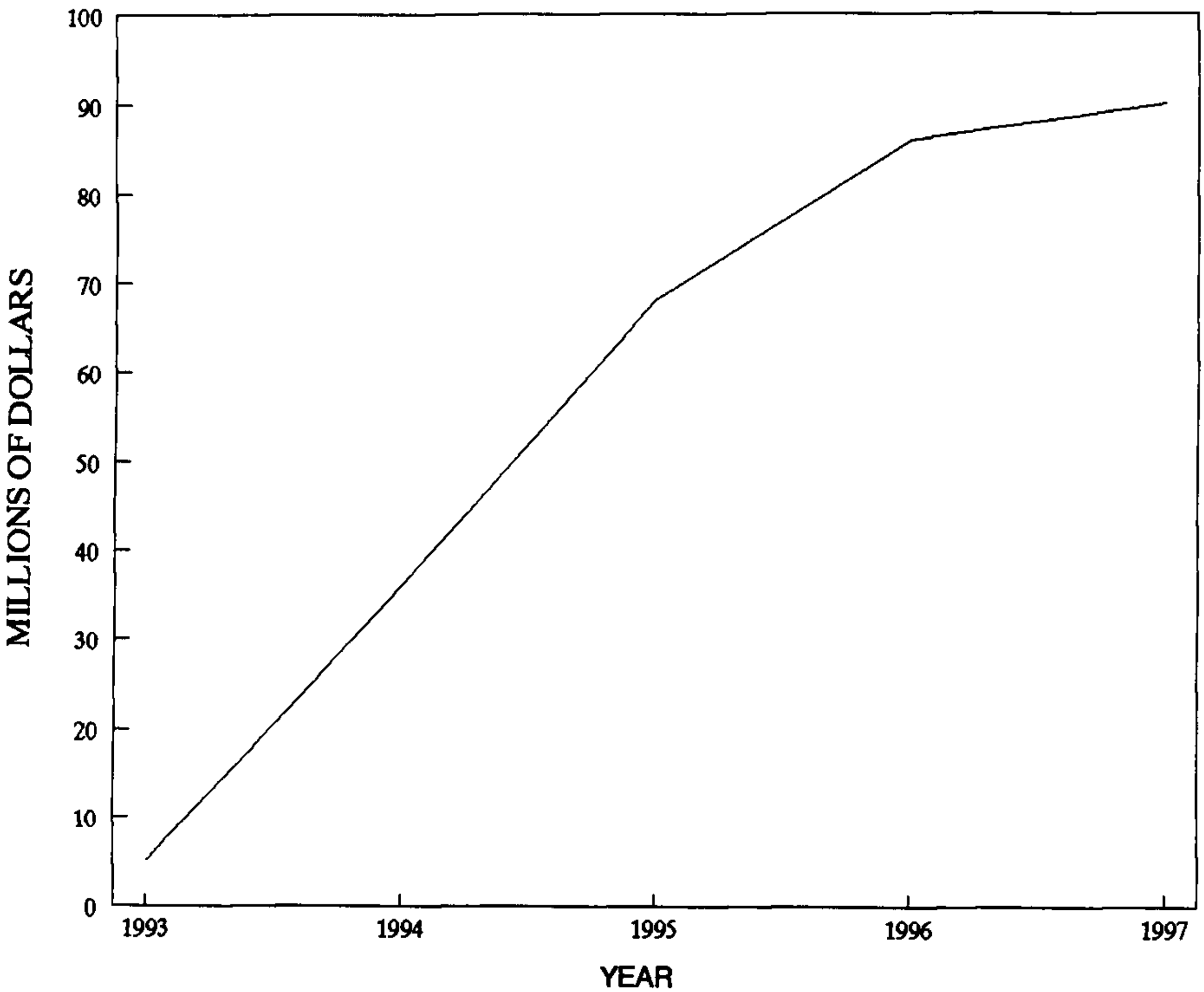


FIGURE VI. B-13
PRE-OPERATIONAL COSTS
FOR S-0 & SF-1 DEPLOYMENT OPTIONS

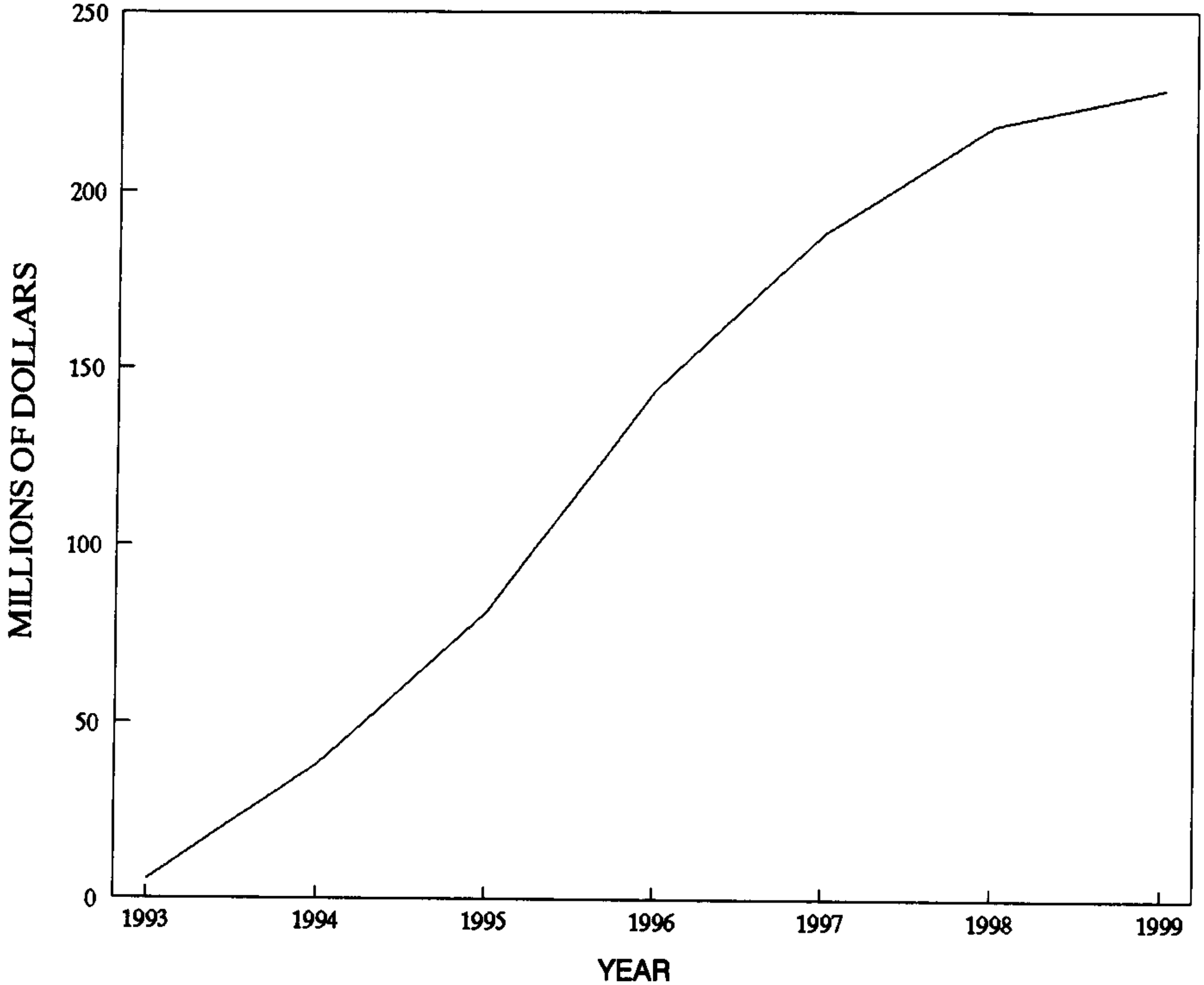


FIGURE VI. B-14
PRE-OPERATIONAL COSTS
FOR SF-0 DEPLOYMENT OPTION

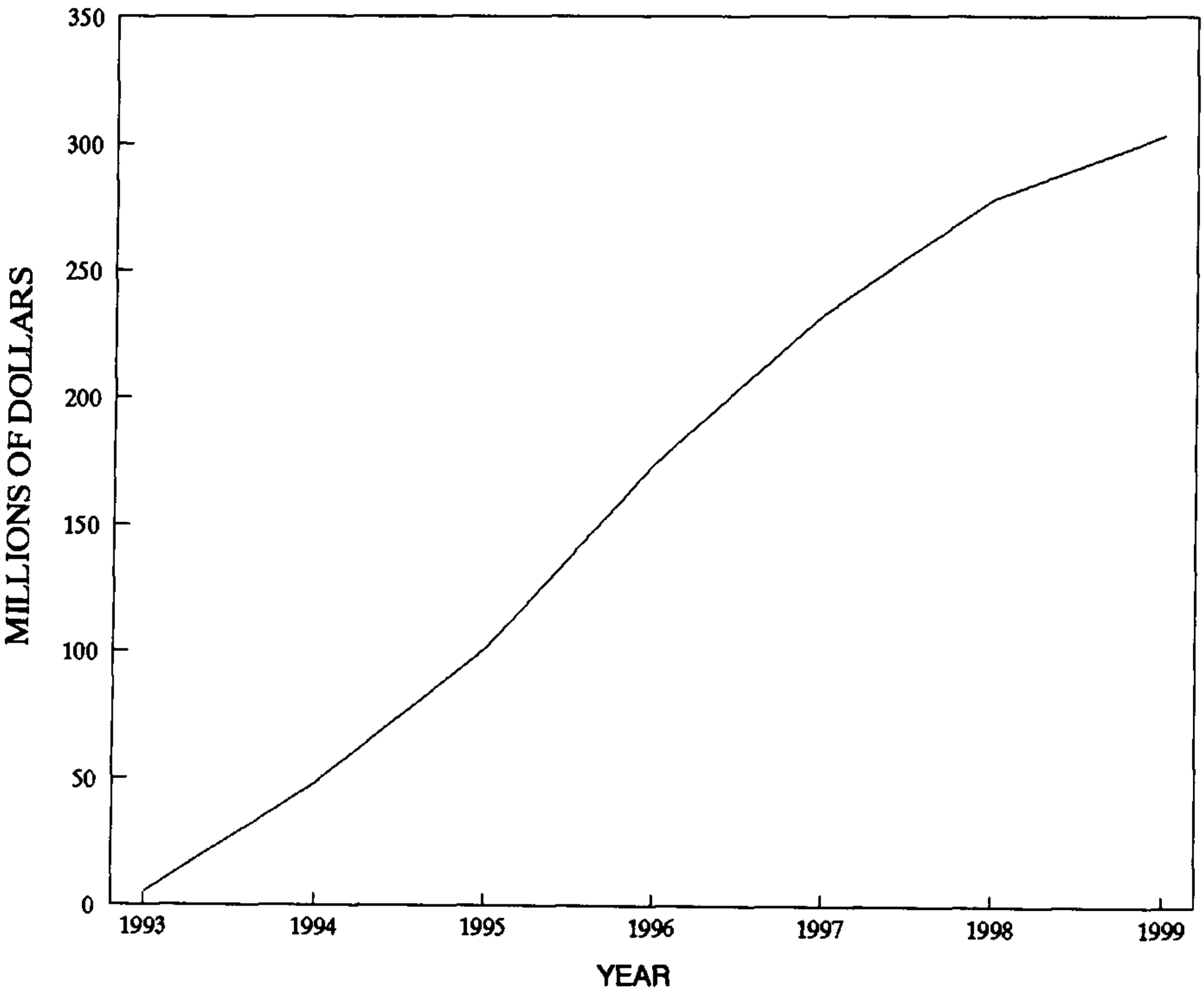


FIGURE VI. B-15
PRE-OPERATIONAL COSTS
FOR D-0 DEPLOYMENT OPTION

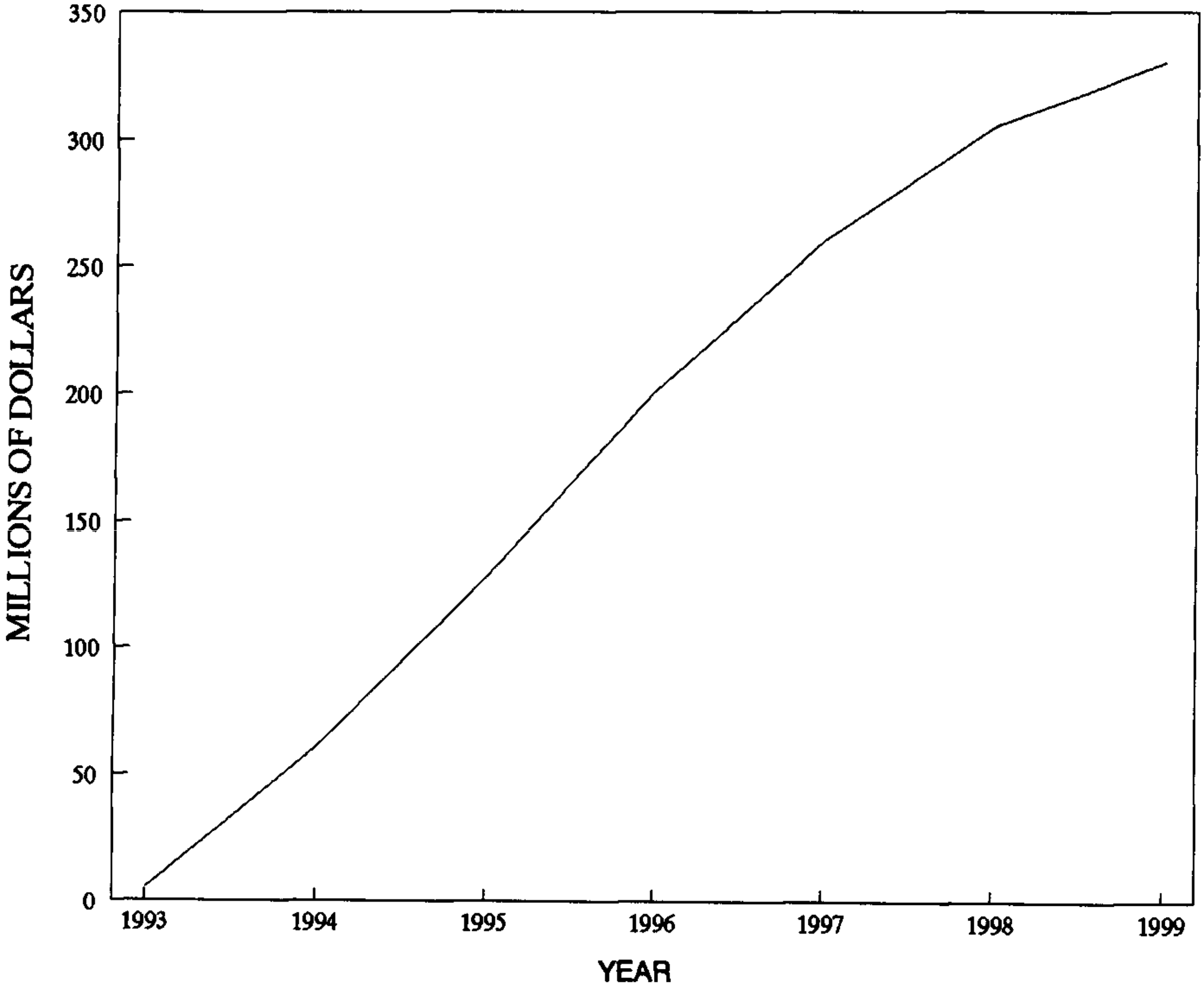
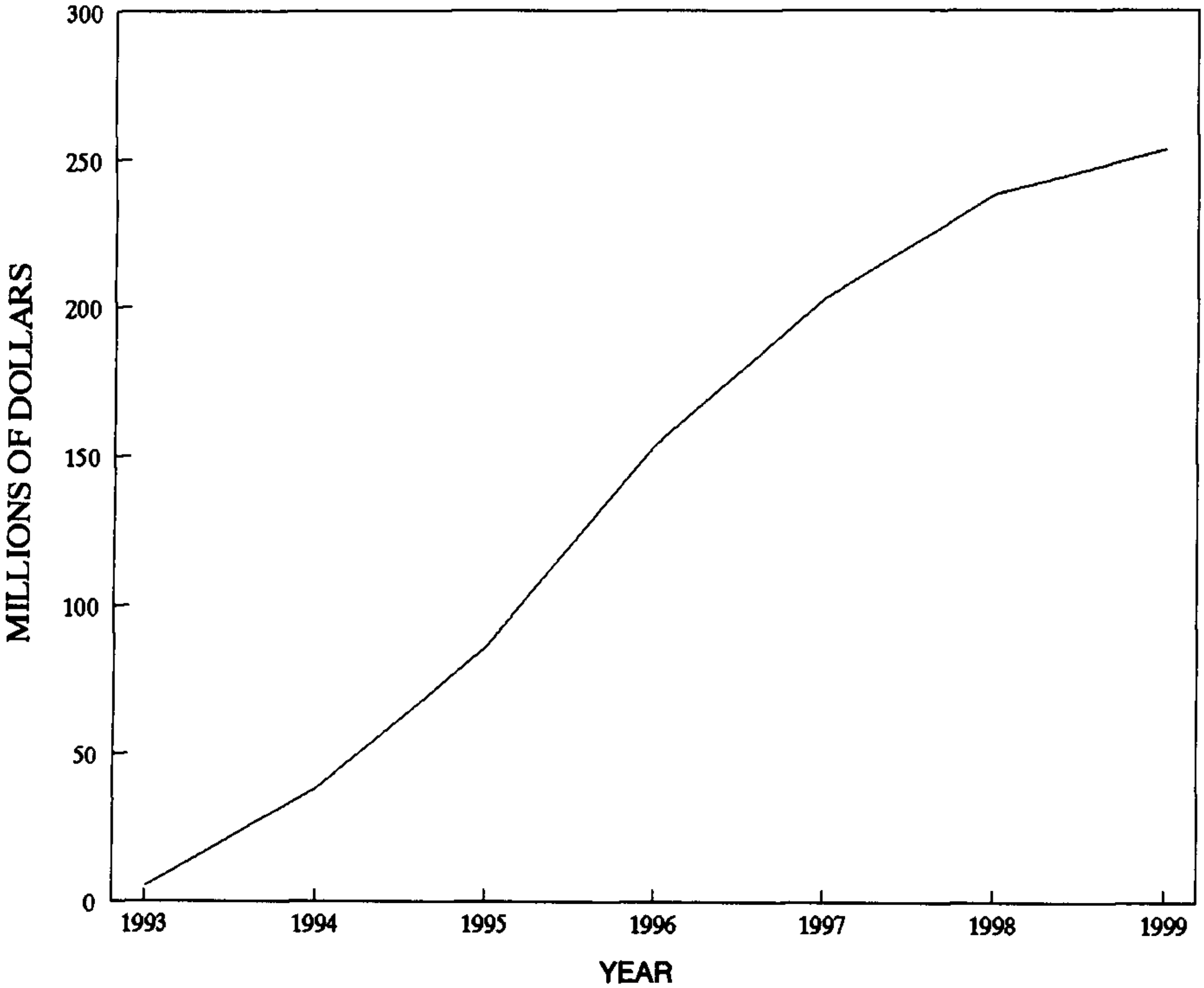


FIGURE VI. B-16
PRE-OPERATIONAL COSTS
FOR SF-2 DEPLOYMENT OPTION



VII. DEPLOYMENT STRATEGY

A. INTRODUCTION AND OVERVIEW

The purpose of this section is to address the deployment options as required by the Plutonium Disposition Study (PDS) Requirements Document (RD). This overview briefly introduces the approach taken in responding to the basic requirements in later subsections.

The RD establishes basic requirements for the time period and amount of plutonium to be disposed and capabilities that must be provided (e.g., capability for tritium production). The RD methodology requires development and characterization of three reactor core design cases (based on the fuel cycle) that are referred to as Spiking, Spent Fuel and Destruction. These required core design cases are constrained in that they must satisfy all RD requirements.

The RD (Section 5.1, Development Strategies) states: "The deployment strategy outlined in the previous sections envisions one or more large reactor complexes, probably located on Federal land and wholly supported by Federal funding." Therefore, the deployment strategies to be addressed include the three required reactor core design cases, located on Federal land and wholly supported by Federal funding. These are referred to herein as the Required Deployment Options and are designated according to the fuel cycle used as S-0 (Spiking), SF-0 (Spent Fuel) and D-0 (Destruction), respectively.

The RD (Section 5.1, Development Strategies) goes on to state: "Designers are requested to discuss and describe any other possible deployment strategies that could be considered within the U.S. or outside the U.S., through international cooperation, etc." This requirement solicits a discussion of other possible reactor deployment options, which may or may not satisfy all RD requirements. This implicitly recognizes that benefits may be obtained by relaxing the constraints imposed by RD requirements. This tradeoff allows less extreme reactor core designs (fuel cycles) to be utilized and is particularly beneficial with regard to establishing a more economic balance between the mission time and number of reactors required.

Consideration of other deployment options also includes alternative strategies regarding location and funding. This extends the range from that stipulated for the required options (i.e., totally Government owned and operated facilities on Federal lands) to encompass possible private investment and other siting alternatives.

Consistent with the intent of the plutonium mission, two additional deployment options have been developed. These may be considered as variations of the required Spent Fuel (SF) option, in terms of the reactor core design (fuel cycle). They are referred to herein as Other Deployment Options and are designated as SF-1 and SF-2, since they include one and two reactors, respectively. The benefits obtained from these options results principally from relaxing the mission time constraint of the RD requirements.

It should also be recognized that the location (reactor complex siting) and funding alternatives applicable to Other Deployment Options are rather generic; they are not exclusively associated with SF-1 or SF-2. For example, the SF-1 Deployment Option is defined in terms of the number of reactors, reactor core and fuel cycle parameters, etc. It is not defined in terms of where it is located or how it is financed. It could equally well be located anywhere in the U.S. or one of the Commonwealth of Independent States (CIS) and financed by a variety of options. Therefore, these and other factors affecting Other Deployment Options are addressed separately in the Discussion of Other Deployment Options section.

In addition, the Other Deployment Options section includes a discussion of alternatives referred to as Special Deployment Options. This includes use of the existing, partially completed WNP-3 reactor, and deployment in Russia or one of the CIS States.

Tables are provided at the end of Section VII. Tables VII.A-1 and VII.A-2 are provided to identify some of the chief characteristics of the deployment options. Table VII.E-1 provides a more comprehensive summary of the main characteristics of the five principal deployment options.

Lastly, the RD (Section 5.2, Challenges) states: "The designer shall identify the 10 most difficult challenges (e.g., development requests, licensing approval, etc.) that would be faced if the alternative were to be pursued." Challenges are summarized in Table VII.E-2, which also includes a relative indication (Low, Medium, High) of the degree of challenge presented to each deployment option. Challenges are also noted in pertinent subsections throughout this report.

It should also be noted that the intent of the RD and the information required and/or desired in this final report was clarified by DOE, principally in the project review meetings held in Windsor, Connecticut on March 31, 1993, and in subsequent guidance. In particular, these interactions affected the interpretation of the core design capability and the presentation of cost information. These points are discussed in the appropriate subsections.

1. Required Cases: Methodology

The RD methodology is apparently intended to explore the extreme limits of the performance envelop applicable to each technology in terms of minimum mission time (Spiking) and elimination of plutonium to the maximum extent (Destruction).

Within the context and constraints of the overall set of RD requirements, the required reactor core design cases (fuel cycles) might be interpreted as the extreme cases for establishing some of the technical limits of a conceptual solution space for the plutonium disposition problem. In this respect, then, Required Deployment Options do not represent optimized choices for a practical deployment option. Perhaps, for the purposes of the PDS, the Required Deployment Options may best be interpreted as an ***indicator*** of the maximum technological capability for plutonium disposition according to reactor type and core design. However, with respect to location on Federal land and Federal funding, the Required Deployment Options are realistic.

In the process of identifying potential deployment options, preliminary reactor core designs were developed that satisfied the RD. Given these preliminary core designs and principal design options, an initial set of potential deployment options was developed for further discussion and potential investigation. Prior to the March 31, 1993 meeting with DOE, reactor core designs were developed that provide the dual capability of *simultaneous* production of tritium and continuing destruction (consumption) of plutonium. Based on DOE guidance at that meeting, the reactor core designs and fuel cycles were revised without this constraint. The Required Deployment Options are now more focused on the parameter of interest for each case (e.g., minimizing mission time for Spiking, etc.).

Similarly, after March 31, cost reporting requirements have been clarified. Although the RD contemplates full Federal funding and location on Federal lands for the Required Deployment Options, emphasis has been placed on accounting for the power plant split into an Energy Conversion Area (ECA) and Plutonium Disposition Reactor (PDR). The ECA, the portion of the plant devoted to conversion of steam production to electric generation, could be operated as a separate entity. This model is similar in concept to one which might be used by an IPP (Independent Power Producer) or EWG (Electric Wholesale Generator) project. Estimates are provided herein for the conceptual split between DOE and ECA ownership and operation. However, for the Required Deployment Options, it is not clear that such estimates are representative of optimized choices as would be the case for SF-1 or SF-2.

2. Other Deployment Options: Methodology

The RD request for a description and discussion of other possible U.S. or international deployment options fosters a more creative, yet practical, consideration of the means for deploying one or more PDRs. This permits deployment options to be identified that provide a different optimization of cost, extent of burnup, and mission time than what is embodied in the RD specified methodology. The objective is simply to achieve mission goals for plutonium disposition in a more practical manner.

That methodology serves the purposes of the PDS and provides a somewhat level playing field for an economic comparison of alternatives, although some of the assumptions, (e.g., stipulation of a 75% capacity factor), may significantly underestimate the capability provided. Still, the optimum solution to the plutonium disposition problem need not necessarily be within the solution space bounded by the Required Deployment Options.

Selection of a deployment option obviously requires a broad group of factors to be addressed in addition to the reactor core design and fuel cycle. A more realistic, pragmatic approach is needed to address important factors not considered by the simplified and constrained methodology of the required cases.

Some of the factors include: the many aspects of location, tradeoffs of characteristics in some areas to obtain benefits in others, financing arrangement to include private and government sources, proliferation risks associated with siting and transportation, numerous foreign deployment considerations, potential use of existing facilities at the Savannah River Site, Hanford, and/or other Federal facilities, use of hybrid combinations of fuel cycles (e.g., spiking followed by spent fuel), U.S. and foreign cooperative deployments, international funding sources, relationships to other international efforts, and so forth.

Given that these factors could not be treated rigorously here, a qualitative approach has been used. Initial discussions focused on identification of a set of potential deployment options for further discussion and investigation. Potential reactor core designs and fuel cycles were developed in an attempt to provide a better optimization of factors. In particular, the difficulty of funding the capital cost for four reactors was recognized. From a practical perspective, one or two reactors are much more attractive.

As these discussions and investigations continued, the set of options was narrowed to focus on those deployment options considered to be most feasible. The following subsections provide a brief description and discussion of two deployment options, SF-1 and SF-2, which are one and two reactor versions of the more economic SF fuel cycle, respectively. These options could be located and funded in a variety of ways and are impacted by several of the aforementioned factors. Therefore, those aspects and the challenges to deployment are discussed in Section D, following a description and discussion of the basic reactor core and fuel cycle characteristics of each option.

3. Summary of Deployment Options

Tables VII.A-1 and VII.A-2 at the end of this section summarize the chief characteristics of the various deployment options. Table VII.A-3 provides a more comprehensive comparison of the main characteristics of the five principal deployment options.

B. REQUIRED DEPLOYMENT OPTIONS

The Required Deployment Options are briefly described and discussed in the following subsections in order to comment on the significance of the option, identify controlling factors, drawbacks and/or weaknesses, etc. As appropriate, comments address the economics of commercial operation, reasonability of assumptions regarding significant parameters (e.g., capacity factor) and so forth. This discussion is intended to put the deployment options into proper perspective.

It should be noted that an alternative core design can be employed for tritium production, in compliance with the RD and DOE guidance. The tritium core produces 3410 MWt and 1115 MWe (net). This is based on meeting the tritium production requirement with a single reactor. This constitutes an alternative core design and is not considered a deployment option, per se. Technically, it is available for all deployment options. Similarly, the Spiking fuel cycle could be used for any deployment option should it prove desirable at some later time.

1. Description of S-0: Required Deployment Option for Spiking

The Spiking option defines the limit of minimum mission time. A single reactor is capable of very rapidly satisfying the Spiking requirement due to the combination of large reactor size and its capability to accommodate full MOX (mixed oxide) cores. The plutonium disposition core produces 3800 MWt and 1256 MWe (net). The fuel cycle required to satisfy the spiking requirements is less than three months. As a result, the plutonium disposition mission is accomplished in only four years and three months after the start of operations.

The advanced design features of System 80+ are used to great advantage in this option. For example, these features provide the ability to accomplish refueling quite rapidly. (In fact, if an outage were required *exclusively* for refueling, it requires only 17 days.) Many surveillance, test and maintenance actions may be performed while at full power during normal operation. This permits required outage work to be distributed between the frequent refueling outages required for the Spiking option each year.

2. Discussion of S-0: Required Deployment Option for Spiking

The principal significance of the Spiking option is simply to define the technical limit of minimum mission time. Only one reactor is required, and only a short operating period is needed to satisfy RD Requirement 3.2.1.3 for material characteristics of greater than 100 rem/hr at three feet after a two year cooldown period. However, it is believed that the rate of fuel fabrication will become the limiting factor for this option. Therefore, the operating cycle was shortened until fuel fabrication and other practical considerations became controlling. The resulting minimum cycle exposure is 39 EFPD. This exposure will result in material characteristics that greatly exceed the RD requirement.

This option results in plutonium destruction of about 4% overall. Only a small fraction of the energy value of the fuel is utilized, which represents a considerable waste of resources and greatly reduces the revenues from electricity production. Even at its best, since Spiking requires numerous outages each year, the capacity factor is low and the option is not economically attractive.

No compelling reasons have been identified to develop additional Spiking cases in the Other Deployment Options section, which focuses on more practical deployment options. However, it should be noted that Spiking, as a fuel cycle option, remains available for any deployment option. It can also be combined with other fuel cycle options in a variety of ways.

3. Description of SF-0: Required Deployment Option for Spent Fuel

The Spent Fuel option is an intermediate option between the extreme limits defined by the Spiking and Destruction options. It defines the most economical disposition option subject to the RD constraints. The RD constraints on mission time of 25 years requires four reactors to satisfy the Spent Fuel requirement. The plutonium disposition core produces 3800 MWt and 1256 MWe (net). An annual fuel cycle is used with a 50 day outage each year. The stipulated capacity factor is 75% and is conservative. As a result, the plutonium disposition mission is accomplished in 18 years after the start of operations.

Positioned as it is between options for minimum mission time and maximum extent of plutonium destruction, the technically distinguishing characteristic of this option is the similarity of the discharged fuel to typical commercial reactor spent fuel. Average discharge burnup is in excess of 42,200 MWD/MTHM. This option transforms the plutonium isotope ratios such that Pu-240 constitutes approximately 23% of the plutonium in the discharge fuel.

4. Discussion of SF-0: Required Case for Spent Fuel

The significance of the Spent Fuel case is that it defines the extent of fuel burnup and plutonium transformation necessary in the fuel cycle. At least for the U.S., there does not appear to be any compelling reason to drive for a lower plutonium content in the PDR spent fuel than that which results from the commercial nuclear industry.

This option is also well proven technology with no need for research and development programs. Thus, there is little technical risk to affect cost and schedule. The total quantity of plutonium, and more generally, the total quantity of spent fuel and other wastes to be disposed of from operation of the PDR, is small with respect to that of the commercial industry.

Considering electric sales revenues, the economics of SF-0 are the best that are possible *subject to the constraints imposed by RD*. However, this option does not represent an optimum cycle from an economic perspective. The controlling factors are the 25 year mission time for 100 MT of plutonium, which necessitates four reactors, and the stipulated 75% capacity factor. This reactor design and fuel cycle is capable of a higher capacity factor (80%). Further, the design life is 60 years, such that a large period of post-mission commercial operation is potentially available. Thus, the development of Other Deployment Options SF-1 and SF-2 are based on this basic reactor core design and fuel cycle, but with the RD schedule constraint relaxed.

5. Description of D-0: Required Deployment Option for Destruction

The Destruction option defines the extreme limit of maximum extent of plutonium destruction. Again, the RD constraints of a 25 year mission time results in four reactors. The plutonium disposition core produces 3800 MWt and 1256 MWe (net). An annual fuel cycle is used with a 50 day outage each year. The stipulated capacity factor is 75% and is conservative. As a result, the plutonium disposition mission is accomplished in 18 years after the start of operations.

The technically distinguishing characteristic of this option is the core design, which uses non-fertile fuel. The core is designed for extended burnup using burnable poisons, such that the end of core life is the result of insufficient reactivity to continue operations. This results in destruction of 83% of the initial Pu-239 and 61% of the core's total initial plutonium inventory, which is greater than SF-0.

6. Discussion of D-0: Required Case for Destruction

The Destruction option is relevant principally from the perspective of defining the technical limit of plutonium destruction. This leads to the selection of non-fertile fuel. The general characteristics of the generating capacity, annual fuel cycle, outages, electric revenues, and so forth appear quite comparable to the SF-0 Spent Fuel option, However, while there is substantial experience with the MOX fuel used for SF-0, there is very little experience with non-fertile fuel cores. This introduces schedule and financial risks that are markedly higher for the Destruction option, but difficult to quantify. There would also be other, second order impacts on the cost, for example, by virtue of the cost of non-fertile fuel, which may result in a less economic option than SF-0.

Also, as noted above, the ability to accomplish any incremental destruction beyond that provided by a Spent Fuel cycle is of questionable value unless the plutonium content of all commercial reactor spent fuel is reduced to comparable levels.

C. OTHER DEPLOYMENT OPTIONS

The Other Deployment Options are briefly described and discussed in the following subsections. As for the analogous section on Required Deployment Options, subsequent subsections are intended to put the deployment options into proper perspective. These subsections provide comment on the significance of the option, identify controlling factors, drawbacks and/or weaknesses, etc. As appropriate, comments address the economics of commercial operation, reasonability of assumptions regarding significant parameters (e.g., capacity factor) and so forth.

A description and discussion of Other Deployment Options SF-1 and SF-2 are provided first. (In the absence of other overriding factors, no optimized versions of Spiking or Destruction core designs and fuel cycles are considered reasonable due to the unattractive characteristics of such cycles.) This is followed by a discussion of two Special Options, the potential conversion and completion of the existing WNP-3 facility and location in Russia or another CIS State.

This section focuses on identification of practical deployment options and principal factors affecting deployment. This requires a broader view of optimization for deployment. The intention is to obtain more practical deployment options by selectively relaxing certain constraints of the RD requirements, principally mission time. The principal objective remains to dispose of the excess plutonium inventory as rapidly as practical with a credible deployment option.

Credible deployment options must propose possible solutions to problems presented by the existence of numerous legal, political, institutional and financial factors. For example, proliferation considerations affect the location, transportation, extent of burnup and other characteristics of any deployment option. A brief discussion of factors affecting realistic deployment scenarios is provided in Section D within the context of a broader perspective on the significant factors affecting and obstacles to successful deployment.

As noted previously, that prior to the March 31, 1993 meeting with DOE, reactor core designs were developed that provide the dual capability of ***simultaneous*** production of tritium and continuing destruction of plutonium. Although this line of development has not been continued, it represents a latent capability that could be developed if it is judged to provide significant value to DOE. This illustrates how the inherent design flexibility of System 80+ provides great advantage in developing a PDR reactor core and fuel cycle design tailored and optimized to the plutonium disposition mission. The options presented in the following sections are a small subset of the available options that were selected as the most promising on the basis of currently available information.

1. Description of SF-1 and SF-2: Optimized Spent Fuel, One and Two Reactor Versions

These options are one and two reactor versions of the same reactor core design as was developed for SF-0. As noted previously, SF-0 defined the most economical disposition option subject to the RD constraints. The most limiting constraint identified for SF-0, the controlling factor for that option, is the RD limit on mission time of 25 years for the 100 MT of plutonium, which required four reactors. The stipulated capacity factor of 75% is also conservatively low, and an evaluation has been performed to verify that a capacity factor of 80% is reasonable.

The SF-1 and SF-2 options simply provide a more practical, economic Spent Fuel option by allowing the mission to extend beyond the 25 year RD requirement. The plutonium disposition core remains the same and produces 3800 MWt and 1256 MWe (net) using an annual fuel cycle with a planned 50 day outage each year. Other aspects are discussed below.

2. Discussion of SF-1: Optimized Spent Fuel, One Reactor

The significance of the SF-1 deployment option is that it satisfies all mission objectives with the exception of the 25 year mission time with a single reactor concept. The principal focus is to use the most practical and economical reactor core and fuel design, an optimized Spent Fuel approach.

The difficulties (e.g., financing, fuel fabrication rate, etc.) associated with deploying a four reactor concept such as SF-0 are significantly less for a single reactor concept. In order to reduce the cost to the Federal government, an option is needed that is capable of attracting private investment.

The SF-1 deployment option is responsive to these needs. It provides the capacity to dispose of the entire 100 MT inventory in an economical fashion over 60 year plant life. The reactor core design is most practical by using the proven base of MOX fuel, rather than non-fertile fuel. The fuel fabrication rate of one full core per year or less is relatively low. Compared to a four reactor concept, the funding requirements are substantially reduced. Thus, the single reactor concept improves the probability of deployment.

Because the System 80+ design is already commercially viable, attracting significant private investment is possible, rather than relying on full Federal funding. Importantly, the quantity of fuel is sufficient for long term electric generation, and MOX fuel is sufficiently proven to reduce overall development risks to a low level. Thus, the prospects for obtaining private investment, for example, by an Electric Wholesale Generator (EWG), is favorable. (Certain arrangements, funding and guarantees may be required of the Federal government in order to secure any private investment, as discussed in a subsequent section.)

The SF-1 (and SF-2) options greatly improve the potential for favorable public relations. It can have the effect of reducing the cost to the public, via private investment, producing needed electric power at a very competitive price, and avoiding adverse environmental impacts of power generation from other sources while eliminating plutonium. This is truly a "swords to plowshares" deployment option.

In the first 15 years of operation on annual fuel cycles, the total inventory of plutonium will be exposed well beyond that required to render the spent fuel self protecting (spiked). Should it be considered necessary to complete the spiking mission more rapidly, it would be possible to consider spiking all the fuel first, as in the S-O option. Fuel could be stored and then returned to the reactor (after all the material has been spiked) for continued operation to more completely dispose of the plutonium, which avoids loss of the energy value of the fuel and reduces the net cost by the electric sales revenues. These alternatives for accelerating disposition by spiking are not necessarily considered desirable, and certainly are not the most economical. However, they do provide a potentially valuable benefit to DOE that is important to recognize: it is possible to accelerate the disposition schedule. The option to accelerate the deployment schedule is not foregone by a decision to adopt the SF-1 deployment option.

Lastly, if it were determined that a foreign location would dispose of, for example, 50 MT of the inventory, SF-1 would be an ideal option for disposing of the other 50 MT. In that case, the single reactor option would complete the disposition mission in 30 years and be available for commercial operation for the remaining 30 years of design life. This variation of the SF-1 option illustrates an important benefit in having the flexibility to provide an economic deployment option over a wide range of quantities of plutonium to be disposed. Other characteristics and alternatives for this situation would be similar to that discussed for SF-2, below.

3. Discussion of SF-2: Optimized Spent Fuel, Two Reactors

The significance of deployment option SF-2 is that it is an economical, two reactor concept. Again, the principal focus is to use of the most practical and economical reactor core and fuel design, an optimized Spent Fuel approach. The discussion for the SF-1 option applies here with the simple change to a two reactor concept, which places a relatively greater emphasis (higher value) on rapid completion of the plutonium disposition mission. This option reduces the plutonium disposition mission time to 30 years following initial operation.

The SF-2 option still requires substantially less funding than four reactor concepts such as SF-0 and provides a correspondingly higher probability of deployment. Thus, continued operation is possible for both reactors for 30 additional years as commercial electric generating facility. The capital cost of the electric generating facilities may be recouped during the first 30 years, after which it would be viable to use purchase commercial UO_2 fuel for continuing operation. The higher availability of electric generation from a two unit commercial generating facility may also provide a more attractive basis for securing private investment than for the single unit option, SF-1.

The economics of Advanced Light Water Reactors were the subject of a recent USCEA (U.S. Council for Energy Awareness) study, "Advanced Design Nuclear Power Plants: Competitive, Economical Electricity" (June 1992), noted the advantageous economics of building and operating dual units on a single site. Of course, realizing the economic advantage of dual reactors on a common site requires that there be a sufficient demand for power to support power purchase contracts for the electric sales. Therefore, the potential for supporting this level of power sales in the southeast U.S. was evaluated. It was confirmed that there is sufficient need for power in regions such as the southeast U.S. to economically justify deployment of the SF-2 option.

The SF-2 option emphasis on completing the mission more rapidly than for the SF-1 option results in the total inventory of plutonium being transformed to the isotopic characteristics of commercial spent fuel within an operating time of 31 years at a 75% capacity factor. At a more realistic 80% capacity factor, the time required would be reduced to approximately 29 years. Fuel exposures are also well beyond that required to render the spent fuel self protecting during the first 8 years of operation on annual fuel cycles.

Should it be considered necessary to complete the spiking mission more rapidly, it would be possible to consider spiking all the fuel first, as in the S-0 option. Fuel could be stored and then returned to the reactor (after all the material has been spiked), which would require a very short period (of less than three years, depending on the timing between startup of the units. Thereafter, any continued operation would more completely dispose of the plutonium, again avoiding the loss of the energy value of the fuel and reducing the net cost by the electric sales revenues.

As another potential option, the first reactor could be initially operated with the Spiking fuel cycle. When the second unit begins operation, it could run a Spent Fuel cycle using the fuel discharged from the first reactor. After all the fuel is spiked, the first reactor could then be operated on a Spent Fuel cycle.

These alternatives for accelerating disposition by spiking are not necessarily considered desirable, and certainly are not the most economical. However, they do provide a potentially valuable benefit to DOE that is important to recognize: the option to accelerate the disposition schedule is not foregone by a decision to adopt the SF-2 deployment option.

<u>Special Deployment Options</u>

Special Deployment Options are a subset of Other Deployment Options that are intended to explore creative means of disposing of the plutonium. Literal application of each of the RD requirements is not always applicable in these cases. Two cases are discussed below: WNP-3 completion and location in Russia or some other CIS State.

4. WNP-3 Completion

Washington Nuclear Project-3 (WNP-3) was evaluated as a potential Deployment Option. WNP-3 is a 75% complete nuclear plant, owned by the Washington Public Power Supply System and located in the western portion of the state, near the Satsop River. The plant has been in a preservation mode for nearly ten years.

WNP-3 was considered because it includes a System 80 nuclear steam supply system. As noted earlier in this report, the System 80 reactor is virtually identical to the new System 80+ reactor and, thus, would be fully capable of utilizing a 100% MOX reactor core.

During a tour of the WNP-3 facility and a meeting with representatives of the Supply System, the following observations were made:

a. There is no technical reason that the WNP-3 facility could not be completed and serve the mission assumed for a single unit facility.

b. It would be desirable, if not necessary, to evaluate the plant design to determine whether any of the System 80+ advanced features could be backfit into the unit, on an economical basis. The resulting calculated safety level would be greater than that of the original System 80 design, but would not be able to approach the safety level of System 80+ without significant plant additions, such as an on-site combustion turbine.

c. The unit would have to include additional structures and equipment for a much larger fuel storage facility and for safeguards facilities.

d. A MOX fuel manufacturing facility and a tritium handling facility would either have to be built onsite or these materials would have to be transported across the state from the Hanford reservation.

e. There would probably be very strong opposition to the plant's completion as a plutonium burner from members of the public in Western portions of Washington state. It is expected that the opposition would be even more vociferous if the tritium mission were implemented.

f. Unless the completed unit is still owned by the Supply System, there are still legal entanglements involved in selling the unit to another party (e.g., DOE or an Independent Power Producer). However, the legal entanglements to selling the unit are less severe than existed several years ago. Furthermore, the Supply System is precluded by state law from participating in an IPP to complete and operate the unit.

g. Although WNP-3 uses a standardized System 80 NSSS, the remainder of the plant is a custom design. Therefore, the cost estimates to complete the licensing and construction of the unit are still substantial.

h. Because the Northwest region enjoys some of the lowest cost electricity in the nation, the revenues that could be obtained in a competitive sale of electricity are probably 30-40% lower than could be generated in the eastern U.S.

Based upon a qualitative evaluation of these observations, it was decided that the WNP-3 completion option would not be considered the first choice. Although the unit could be completed at a lower cost than for construction of a new unit, the potential revenues from electricity sales are correspondingly lower, as well. Public opposition, concerns about transportation of MOX and tritium, and potential legal entanglements all present areas of significant uncertainty that cannot be easily resolved. However, other expedient avenues may develop which may well turn completion of this unit into a viable option.

5. Location in Russia or the Commonwealth of Independent States (CIS)

Since half of the plutonium disposition materials are provided from Russia, it is reasonable to consider a Special Deployment Option for deployment there or in a CIS State. There are several aspects to this Special Deployment Option. However, a few general comments are appropriate.

The former Soviet Union designed and built numerous reactors, principally the RBMK and VVER designs. Since the Chernobyl accident, the international community has seen an unprecedented change in political structure of the former USSR, which has simultaneously heightened concerns about the safety of Soviet-designed reactors, and yet, provided genuine opportunities to become involved and provide much needed assistance. There is such a demand for continuing nuclear generation that the proposed shutdown of RBMK and other older designs does not appear likely.

The current situation is the subject of many reports. With respect to deployment options, a recent USCEA report, "The Safety of Soviet-Design Nuclear Plants: A U.S. Industry Perspective," provides a good general summary of relevant information. It points out that Soviet-designed reactors require, to varying degrees, a substantial upgrading in operational safety, operator training and maintenance. However, there is a great need within the former Soviet Union for hard currency to obtain the equipment and spares required for continuing operation, say nothing of safety improvements.

Within this context, then, two approaches are suggested for consideration. In both, by being near the source, safeguards and transportation concerns may be minimized and overall costs might be improved.

The first approach is Russian (or other CIS State) deployment of a PDR (Plutonium Disposition Reactor) based on System 80+ to accommodate 50 MT of plutonium. This option would be strictly focused on the agreements to dispose of the plutonium excess as a result of weapons dismantlement.

If developed properly, this approach could assist to a degree in establishing a Western reactor safety philosophy. This includes the safety philosophy, design criteria and features, operational methods and maintenance and testing practices. This is synergistic with DOE initiatives for operational safety and other similar activities under the support of WANO (World Association of Nuclear Operators), IAEA (International Atomic Energy Agency), the Common Market and others. A limited technology transfer is also inherent with this deployment option.

The second approach encompasses the benefits of the first approach and is more ambitious. It involves proposing deployment of one or more PDRs in exchange for phaseout of currently operating RBMKs (and possibly some other concessions). This would permit shutdown of reactors considered by many to fall short of minimum safety standards while providing the power demands that apparently necessitate their continuing operation. Moreover, the plutonium disposition mission might constitute a minor part of such an effort if a sufficient number of reactors were planned. This approach could be quite flexible in terms of the number of reactors to be built and the schedule; the plutonium disposition mission would be accomplished first in any scenario.

This option places greater reliance on coordinating overall assistance to former Soviet States in order to achieve potentially greater benefits. The infrastructure to support current operating plants is not fully adequate. There is a great deal of technical talent available in Russia and CIS States that could be placed into productive work. The various Ministries could be involved to foster technical expertise and facilitate appropriate regulatory controls. This would not only develop an appropriate infrastructure to support System 80+ plants designs, but also foster development of an improved infrastructure for older Soviet-designed reactors. It could be coordinated with efforts undertaken in support of, for example, the Lisbon Initiatives to improve plant safety, to the benefit of both programs.

Two principal and interrelated challenges are associated with either of these deployment approaches. One challenge is to meet the need for funding from international sources. Direct funding, guarantees of loans, etc. necessary to enable private participation. It would seem reasonable that safety improvements (including avoidance of potential severe accidents) would have sufficient tangible value to support this program, if the cause is championed within the international community.

The other challenge has to do with the role of the U.S. Government. Although direct Federal government involvement is required for any foreign deployment, such options may best be pursued under the ***leadership*** of the U.S. Government, rather than responding to private initiatives. An aggressive, proactive stance would greatly facilitate this option. It is advantageous to first have the basic agreements established by the U.S. and foreign governments. This permits the U.S. to appropriately influence the structure of needed agreements, including application of IAEA safety standards, safeguards and inspections, and funding arrangements and guarantees. Other issues that should be addressed by direct government involvement include safeguards for international transportation, technology transfer and, especially, nuclear liability. These and other aspects of foreign deployment require negotiation. Then, deployment arrangements for a Plutonium Disposition Reactor can be finalized by private firms within the framework established by the Government.

D. DISCUSSION OF DEPLOYMENT ISSUES

The various deployment options have been described and briefly discussed in previous subsections. In those subsections, the focus was on those characteristics, considerations, benefits, and obstacles that were applicable to the particular deployment option under discussion. An effort was made to place the deployment options into proper perspective by commenting on the significance of the option, identifying controlling factors, drawbacks and/or weaknesses, addressing the economics for commercial operation and reasonability of assumptions (e.g., capacity factor), etc. However, as noted, certain considerations, such as location and funding, are rather generic and are best discussed separately in this section.

Also, especially for the Other Deployment Options, a broad range of factors should be addressed in addition to the reactor core design and fuel cycle. These factors include the many aspects of location, including proliferation risks associated with siting and transportation, tradeoffs of characteristics in some areas to obtain benefits in others, financing arrangement to include private and government sources, numerous foreign deployment considerations, perspectives, potential use of existing facilities (e.g., Savannah River Site, Hanford, and/or other Federal facilities), use of hybrid combinations of fuel cycles (e.g., spiking followed by spent fuel), US & foreign cooperative deployments, international funding sources, relationships to other international efforts, and so forth. Some of these factors have been touched on in prior discussion.

A detailed, comprehensive and in-depth treatment of these factors is beyond the scope of this report, although the most difficult challenges for any of the deployment options result from consideration of issues surrounding these factors. Therefore, this section provides a qualitative discussion of some of the more important factors/issues and identifies general obstacles to deployment.

1. Factors/Issues Affecting Deployment

General

The following subsections address factors and issues which require some clarification or for which discussion separate from a specific deployment option is appropriate. For example, the "licensing and regulatory" issue must be carefully defined in order to highlight the issues of concern as distinct from what might be generally thought of as either a licensing or a regulatory issue.

There are other factors and issues for which no specific discussion is presented. This may be because of the subject was not within the scope and schedule for this report. More importantly, however, there are significant issues that present a challenge to any and all deployment options. For example, public perceptions are quite important in all cases and extremely so for U.S. deployment. However, there is generally not an exclusive relation to a specific deployment option. Thus, discussion has been limited to areas for which the impacts significantly discriminate between deployment options. In this example, public acceptance of tritium production presents a somewhat greater challenge since it is not aligned with a "swords to plowshares" concept.

Licensing and Regulatory

The System 80+ design is in the final stages of Design Certification by the U.S. NRC. A Draft Safety Evaluation Report (DSER) was issued in September 1992. Excellent progress has been made in responding to NRC questions and DSER items. Steady progress is also being made regarding new element of NRC licensing for Design Certification under 10 CFR 52 such as the ITAACs (Inspections, Tests, Analyses and Acceptance Criteria). Based on this progress, the Final Safety Evaluation Report is expected in early 1994.

As an evolutionary Advanced Light Water Reactor (ALWR) in the final stages of Design Certification, System 80+ provides an excellent basis for the plutonium disposition mission. It is a proven design that minimizes the licensing and regulatory obstacles associated with more developmental technology at a less advanced stage of licensing with the NRC. It can be deployed rapidly with limited licensing and schedule risk, which reduces associated costs and schedule risks. Thus, design-related licensing and regulatory risks are principally limited to aspects of the design associated with modifications for the plutonium disposition mission. Several licensing factors involve design for special considerations, such as satisfying both NRC and DOE requirements for safeguards and security. Such adaptations are not generally considered to present significant obstacles.

Licensing by NRC for commercial operation is considered to be essential to attracting any private investment, and particularly for SF-1 and SF-2 deployment options. In combination with appropriate funding arrangements and government guarantees, this makes significant private participation achievable.

A few technical issues in the licensing/regulatory area are more noteworthy. This group includes licensing of: full MOX cores with a high plutonium content, MOX fuel fabrication, tritium targets, and safety analyses for plutonium cores, particularly if non-fertile fuel is used. There are cost and schedule risks associated with licensing risks such as the potential for identification of new safety issues and the schedule of regulatory authorities (DOE and NRC).

There are direct means for potentially resolving the technical aspects of these issues, such as suitable regulations for plutonium cores. For example, information from the discontinued GESMO (Generic Environmental Statement Mixed Oxide) process may be used to good advantage. However, perhaps the greatest licensing and regulatory challenges result from the need for a collaborative regulatory agreements between DOE and NRC. The regulatory split of authorities and roles between DOE and NRC

Location

The RD presumes location on a Federal site. This is considered to be the most practical option, including for the case of private involvement as a commercial electric generating facility. The issues and concerns supporting the use of a government site are well known and include proliferation considerations, ability to provide appropriate safeguards, limiting transportation of plutonium, etc.

During the course of this study, consideration was given to alternate locations and deployment options, such as WNP-3, for which it is not expected that all the facilities required of the reactor complex could be collocated with the reactors. This requires use of existing government facilities and/or separately located new facilities. Thus, the transportation of materials becomes a significant consideration. While such an approach is feasible, it is clearly advantageous to limit transportation and maximize the use of government facilities that already provide (or could provide with minor modification) appropriate safeguards. Collocation of facilities on a single Federal site is most ideal.

In addition, selection of a government site may provide ready access to government facilities that could be used to support the plutonium disposition mission, either directly or with some modification. This would also tend to reduce the costs from that required for a green field site and make available the existing infrastructure. The existence of local human resources with the appropriate qualifications for this mission is of great value to achieving a short deployment schedule.

The area required for siting, for example, the SF-2 deployment option, requires area for two reactors units and the other facilities of the reactor complex. The area required is minimized by the large capacity of the reactor(s), which requires fewer units than would otherwise be the case. Again, considering the other facilities required for the reactor complex and the licensing and permits required, this could be more easily accomplished for a government site than elsewhere.

Foreign Deployment

As discussed in a previous section, consideration has been given to potential deployment in Russia or another CIS State. This deployment option could best be pursued under the leadership of the U.S. Government. This would permit structuring the desired and appropriate agreements, application of IAEA safety standards, safeguards and inspections, and funding arrangements and guarantees. Other issues that should be addressed by direct government involvement include safeguards for international transportation, technology transfer and nuclear liability. These and other aspects of foreign deployment require negotiation. It is desirable to first have the basic agreements established by the U.S. and foreign governments. Then, deployment arrangements for a Plutonium Disposition Reactor can be finalized by private firms.

Schedule

The plutonium disposition mission should be initiated as soon as practical. Given the uncertainties inherent to our changing world environment, it appears most prudent to begin during the present "window of opportunity." The need to act now requires selection of technology that eliminates or minimizes to the maximum extent any development programs. Proven technology is essential.

The period for completion of the mission appears to be less critical and depends, in part, on whether a foreign deployment option selected. For example, if the SF-1 option were deployed in the U.S. and a CIS State, the schedule for the foreign deployment may be more important than for the U.S. deployment option. If only U.S. deployment is contemplated, then the timetable for completion of the mission becomes relatively more important. That would then favor U.S. deployment of the SF-2 option coupled with a more aggressive schedule for obtaining the foreign plutonium.

The deployment options discussed herein provide the essential proven technology and flexibility in satisfying variable mission completion schedules. As discussed in prior sections, each deployment option has inherent capability to alter the completion schedule by changing the core designs and fuel cycles utilized. This ability provides the schedule flexibility which is an advantage in dealing with uncertainties from all sources.

Joint Government & Private Participation

Ownership and funding alternatives that were considered for the Other Deployment Options ranged from a totally Government owned, funded and operated facility to full private ownership and funding with operation for the Government. It was concluded that the practical optimum clearly requires a combination of government ownership and financing within an arrangement that facilitates partial private investment and ownership.

As noted previously, requested reporting requirements emphasize an approach wherein the power plant is split such that the Energy Conversion Area (ECA) is operated as a separate entity. This model is similar in concept to one which might be used by an IPP (Independent Power Producer) or EWG (Electric Wholesale Generator) project. Estimates are provided herein for the conceptual split between DOE and ECA ownership and operation. Such an approach is only a first order approximation to the Other Deployment

Options SF-1 and SF-2. The RD methodology is also insensitive to many important factors that could not be fully addressed within the scope and schedule for this report.

An attractive potential option for private investment as an Exempt Wholesale Generator (EWG) under the Energy Policy Act of 1992. This approach could be used with a concept wherein the entire power production portion of the reactor complex could be developed and operated by an EWG. That is, the entire System 80+ design as modified for the plutonium disposition mission would be included, but other facilities, such as fuel fabrication, would be excluded. This concept would allow the facility to obtain a commercial license from the NRC under conditions favorable to private investment and reduce the net cost to the government over the project life.

To expand briefly, the EWG is exempt from Holding Company Act of 1935 but under the jurisdiction of the Federal Energy Regulatory Commission (FERC) per Federal Power Act. It is not exempted from state utility regulation, although the degree to which state authorities might wish to exert their regulatory authority is uncertain. Joint utility and non-utility ownership is allowed. This approach is consistent with the overall FERC policy supportive of market-based pricing to promote competition. Further, FERC now has expanded authority to require wheeling. This assures access to meet market demands. There are adequate demands to support the SF-2 deployment option in the southeast U.S.

Consideration of the arrangements that would be necessary to attract private investment in such a venture involves several issues. Power sales agreements are required for the EWG to sell the electricity generated. Commercial terms and agreements must be secured, which necessitates some guarantees that the project would be continued and available for commercial power generation and sales. Thus, the government would need to provide appropriate financial and other guarantees to obtain private investor funding during the construction phase and limit the EWG's liabilities should the project be cancelled or operate at low power level due to government exercise of its' options.

Additional clauses would be required to address nuclear and commercial liability arising from the potential for accidents during the plutonium disposition mission. For example, decontamination & decommissioning (D&D) might have increased costs due to contamination as a result of plutonium disposition over that which would result from operation aa a commercial facility. Similar concerns apply for tritium production. Accidents or spills, could exacerbate this financial liability, so the government would have to provide some liability protection to the EWG. Particularly for a Federal site, and considering the relatively low portion of overall costs resulting from D&D, it may be simpler to have D&D be the government's liability. In any case, this illustrates the type of agreements that would be needed.

This concept would be realistic for deploying SF-2 or SF-1 options provided the government provided adequate guarantees for funding the initial construction and limiting the risks to the EWG to an acceptable level. In return, the EWG could expeditiously place the reactor(s) into operation and begin to repay the government for its' investment.

Pending further analyses, it is expected that the government would need to provide to the EWG the MOX fuel free of charge, a government plant site, loan guarantees for private investor funds to construct the plant, provide separately the other reactor complex facilities required for the plutonium disposition mission, fund the detailed engineering and NRC licensing of System 80+.

Some detailed consideration of the application of government rules to this conceptual EWG project is necessary. This would consider, for example, the applicability and impact of DOE Orders versus NRC regulations. However, structuring a government and private cooperative venture has the potential to rapidly dispose of the plutonium more efficiently and at a lower cost to the government over the life of the electric generating plant than any other deployment option.

Summary of Obstacles to Deployment

Table VII.D-1 lists and qualitatively ranks the challenges to the various deployment options. These have been addressed in various sections of this report. Some of these challenges are specific to a given deployment option, some are technical in nature, and some are more global in application. Each challenge is judged to be a high, medium or low obstacle to mission objectives and practical implementation according to option.

From this Table it can be seen once again that the Destruction Option (D-O) poses the highest levels of challenges, and the optimized spent fuel options (SF-1, SF-2) the least.

TABLE VII.A-1
DEPLOYMENT OPTIONS
CLASSIFIED BY FUEL CYCLE

FUEL CYCLE	FUEL DESIGN	REQUIRED DEPLOYMENT OPTIONS	OTHER DEPLOYMENT OPTIONS	
SPIKING	MOX	S-0		
SPENT FUEL	MOX	SF-0	SF-1	SF-2
DESTRUCTION	NON-FERTILE	D-0		

TABLE VII.A-2
LOCATIONS AND FUNDING SOURCES

DEPLOYMENT OPTIONS	LOCATIONS	FUNDING SOURCES
S-0, SF-0, D-0	Required Option, Federal location	Required Option, Wholly Federal Owned and Funded
SF-1, SF-2	U.S. or Foreign	- Federal - Federal and Private - Federal, Private, Foreign

TABLE VII.A-3, COMPARISON OF DEPLOYMENT OPTION CHARACTERISTICS

	REQUIRED DEPLOYMENT OPTIONS			OTHER DEPLOYMENT OPTIONS		REACTOR DESIGN
	S-0: SPIKING	SF-0: SPENT FUEL	D-0: DESTRUCTION	SF-1: OPTIMIZED SPENT FUEL	SF-2: OPTIMIZED SPENT FUEL	TRITIUM CORE Applies to all
Number of Reactors	1	4	4	1	2	
Core Thermal Output (MWt)	3800	3800	3800	3800	3800	3410
Electric Generation (MWe)	1350 Gross 1256 Net	1350 Gross 1256 Net	1350 Gross 1256 Net	1350 Gross 1256 Net	1350 Gross 1256 Net	1200 Gross 1115 Net
Cycle Length / Exposure (Months / EFPD)	3 Months 39 EFPD	12 Months 274 EFPD	12 Months 274 EFPD	12 Months 292 EFPD	12 Months 292 EFPD	12 Months, 274 EFPD
Number of Cycles	1	4	4	4	4	4
Average Discharge Burnup (MWD/MWth)	1500	42,200	42,200	45,000	45,000	
Fuel Fabrication Rate (MT Pu/Yr)	25.67 (4 cores/yr)	6.67 (1 core/yr)	6.67 (1 core/yr)	1.67 to 6.67 (0.25 to 1 cores/yr)	3.33 to 6.67 (0.5 to 1 cores/yr)	
Planned Refueling & Maintenance Outage (Days)	90 Days (Total/yr)	50 Days	50 Days	50 Days	50 Days	
Spent Fuel Storage Req'ts (CELLS)	Single Large SFP 5000	Normal SFP 1250	Normal SFP 1250	Normal or Large SFP 1250 or 5000	Normal or Large SFP 1250 or 2500	
Time (Yrs) To: - Fuel Fabrication - Reactor Operation	6 7	6 7	6 7	6 7	6 7	
Pu Disposition Duration (Yrs After Start of Operations)	4	18	18	60	30	
Additional Commercial Operation Duration (Yrs)	56	42	42	0	30	
Pu Transformation - Pu-239 (%) - Pu-240	92 8	63 23	41 32	63 23	63 23	
Pu Destruction - Pu-239 (%) - Total (%)	5 4	51 27	83 61	51 27	51 27	
Fuel Costs ($/Yr)	-0-	-0-	-0-	-0-	-0-	-0-
Non-Fuel O&M Costs ($M/Yr) [1992 $]	$97.50	$243.00	$243.00	$92.8	$126.2	
Capacity Factor	43%	75%	75%	80%	80%	75%

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TABLE VII.D-1, SUMMARY OF DEPLOYMENT OPTION CHALLENGES

CHALLENGE	REQUIRED DEPLOYMENT OPTIONS			OTHER DEPLOYMENT OPTIONS	
	S-0, SPIKING	SF-0, SPENT FUEL	D-0, DESTRUCTION	SF-1, OPTIMIZED SPENT FUEL	SF-2, OPTIMIZED SPENT FUEL
Non-Fertile Fuel Development	-	-	H	-	-
DOE-NRC Collaboration for Regulation	H	H	H	H	H
Siting & Permitting	H	H	H	H	H
Plant Safety Analyses	L	L	H	L	L
Schedule, Construction	L	L	L	L	L
Schedule, Licensing	L	L	H	L	L
Construction Manpower & Material Resources	L	H	H	L	M
Safeguards & Material Accountability	L	L	H	L	L
Fuel Fabrication Capacity / Schedule	H	M	H	L	L
Industry Base for Deployment	L	L	L	L	L
Tritium Target Development	-	L	-	L	L
Reactor Complex Technology Needs	L	L	M	L	L
APR Fuel Cycle Development	L	L	M	L	L
Difficulty of Attracting Private Investment Capital	H	M	H	L	L
NRC Licensing	L	L	H	L	L
Spent Fuel Storage, Transportation & Disposal	L	L	L	L	L

Key: L = Low, M = Medium, H = High