DOE Plutonium Disposition Study

Analysis Of Existing ABB-CE Light Water Reactors For The Disposition Of Weapons-Grade Plutonium

Contract No. DE-AC03-93 SF19682

**A Final Report
by
Combustion Engineering, Inc.
Windsor, Connecticut**

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PLUTONIUM DISPOSITION IN EXISTING REACTORS

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EXECUTIVE SUMMARY

Under contract from the Department of Energy, ABB-CE has been furthering the study of Light Water Reactors to dispose of mission quantities of excess weapons grade plutonium in existing reactors of ABB-CE designs, irradiating the fuel to produce electric power, and discharging the spent fuel in a condition similar in activity and plutonium isotopic content to uranium-fueled LWR spent fuel. Earlier studies performed by ABB-CE for the disposition of weapons-grade plutonium focused on the System 80+ Evolutionary Reactor design, now in the last stages of licensing for Design Certification by the NRC, and currently being offered commercially in world-wide markets. Following the initial phase of the study, an assessment report by the National Academy of Sciences was issued in February, 1994. This report recommended that, for the long-term disposition of the excess weapons plutonium, the technologies of vitrification with high-level waste, and use of the material as fuel in existing or modified nuclear power plants, were the most promising. In this report, it was specifically mentioned that the C-E System 80 reactors, which were originally designed for 100% plutonium fuel operation, were particularly suitable for the mission. Therefore, ABB-CE and its team members have taken an initial look at the potential for existing ABB-CE reactors in operation in the U.S., of System 80 design and others, to carry out the specified plutonium disposition mission of rendering 50 Metric Tons (MT) of weapons-grade plutonium into the "Spent Fuel" condition, over the remaining licensed lifetime of the reactor. An alternative mission of the disposal of 100 MT of weapons-grade plutonium was also considered. This report gives the findings of this brief study.

Three System 80 reactors were constructed and placed in operation, at the Palo Verde Site in Arizona, outside Phoenix. These reactors, owned by a group of utilities in the area, contain the features specifically designed for the handling of plutonium fuel, including extra control rod worth and systems designed to accommodate the greater decay heat and the higher soluble boron requirements that are attendant to an all-plutonium core. At the time that System 80 was designed, there was national concern that the energy resources of the world were being depleted at too great a rate, and therefore the interest was expressed in reactor designs that could accommodate all-plutonium cores, using the recycled plutonium from discharged LWR fuel, in order to conserve uranium ore. Other System 80 reactors were ordered, but never completed. One, at Satsop, Washington, the WNP-3 plant, was 75% complete when the owners, Washington Public Power Supply System, terminated construction.

Other reactors of earlier designs were also included in the study. Three reactors rated at 3410 MWt each were placed into operation prior to System 80. Two are at San Onofre Nuclear Generating Station, Units 2 and 3, near San Onofre, California, and the other at the Waterford Steam Electric Station, Waterford-3, at Taft, Louisiana. A fourth unit, the Arkansas Nuclear One, Unit 2 (ANO-2), located at Russellville, Arkansas, rated at 2825 MWt, was also considered. All units considered are of recent enough vintage that there would be sufficient operating life available to be a factor in

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the mission. None of these units has any features that make it particularly suited for MOX fuel.

The System 80 units are able to operate with full MOX cores, as expected. Because they were designed with only 97 control rod locations, instead of 101 locations for System 80+, and because two of the control rod locations contain water-level monitoring instrumentation, required as a result of the TMI incident, the available rod worth in System 80 is lower than in System 80+, and the permissible plutonium loading is lower - 4.5% in System 80 versus 6.7% in System 80+. This results in a lower throughput of weapons-grade plutonium, and higher costs, since more MOX fuel assemblies will have to be fabricated, shipped, handled and stored. It would be possible to add extra control rod nozzles to the Reactor Vessel Closure Head, thereby increasing the ability to load more plutonium. The desirability of this would depend on the cost versus the benefit of an accelerated mission schedule.

The non-System 80 units (the 3410 MWt and 2825 MWt units), although not designed for MOX fuel, can also operate with full MOX cores, but at a lower loading:- 3%. This is possible only with weapons-grade plutonium, where the content of the higher plutonium isotopes is limited. Notably, the Pu^{240} content is limited to 6.5%. For reactor-grade plutonium, that would be obtained from the reprocessing and recycle of spent LWR fuel, where the Pu^{240} content is over 20%, these reactors could only accept about 1/3 MOX fuel assemblies in the core. The additional neutron absorption due to the higher isotopes of Pu reduces the available control rod worth to the point where safe shutdown limits cannot be achieved under certain design conditions. The lower plutonium loading makes the mission with the non-System 80 units less desirable from an economic standpoint, as the amount of MOX fuel fabricated, shipped, handled and stored would have to be increased to accommodate the full mission.

Estimates of the safety analysis parameters with the System 80 and non-System 80 designs give confidence that the plants could be operated within existing safety guidelines with the MOX cores.

Fabrication of the MOX fuel would essentially be the same as reported earlier in the Evolutionary Reactor Report. The fuel constituents would be the same, except for the plutonium loading and the burnable poison loading. There would be no large change in fuel throughput due to the use of existing reactors; therefore the fuel facility was assumed to be the same. The fuel design for the plants considered is essentially identical to the System 80+ design. The fuel qualification program would be the same. In the case of the existing reactors, however, an alternative to irradiation in the test reactor ATR would be the irradiation of Lead Test Assemblies in the selected reactor.

Fuel management for the System 80 reactors was selected as a two-batch, 18-month cycle, resulting in a three year fuel residence time. The Palo Verde reactors are all operating currently on 18-month cycles, for reasons of improved capacity factor. With

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selected. This also produces the desired Pu level in the discharged fuel. Neither of these fuel management schemes has been studied enough to the point where it is optimized, from a cost and schedule to complete the mission viewpoint.

There are no technology needs for the use of weapons-grade plutonium as fuel in the existing reactors. The fuel technology needs are the same as for the Evolutionary Reactor.

Utility contacts have been sought with Arizona Power, the operator and one of the owners of the Palo Verde units, and with Washington Public Power Supply System, owner of the incomplete WNP-3 unit. At the time of this writing, Arizona Power has not given their agreement to participate in the study. This has made the study incomplete in several areas, where specific site and owner information is required. Washington Public Power, however, has given their agreement to consider the mission. However, there is apparent political opposition in the region of the WNP-3 reactor to the completion of the plant, as well as accepting plutonium as fuel. The owners of the non-System 80 units have not been contacted to determine their interest at this point.

The interfaces between the DOE, the utility and the vendors are likely to be very complex. If the DOE requires control of the operation of the reactor, in order to assure the mission and to comply with international safeguards requirements, the takeover of an existing, operating reactor that is in the utility rate base, and has power purchase agreements with large customers, and supply agreements with vendors for fuel and other services, will be very complicated and may become the driver for the DOE as far as cost and schedule are concerned. The complications may be mitigated if the DOE merely sets up an agreement with the owner for supply of MOX fuel, with control of the operations residing with the owner. However, even in this case, there are several uncertainties such that the owners will want indemnification by the DOE of any risk.

Because of the large uncertainties in DOE-utility interfaces, cost estimates are not possible at this point. Only when dialog with the owners is conducted would it be possible to estimate the costs. Estimates have been made of licensing, assuming no protracted intervention or any new regulations, and of some modifications in the event that mission performance is desired. Other changes, such as safeguards and security requirements, have not been estimated, since they would have to be done with cooperation from the present owners.

Safeguards and security issues for the case of existing reactors are of more concern for the case of existing reactors than was the case for the System 80+ plant. In part, these concerns are due to the assumed location of the fuel fabrication facility away from the reactor site, so that the fresh fuel containing weapons-grade plutonium has to be transported over public rights of way. Existing Safe Secure Transport vehicles, if available, might be used for this purpose. Secondly, the reactor site, which has

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some safeguards systems, will need to be reviewed and the safeguards upgraded where necessary to comply with existing regulations.

Public and political acceptance is also a matter of concern. The Governor of Washington has stated his opposition to completion of WNP-3, and to the use of that reactor for plutonium disposition. There will undoubtedly be other opponents. At issue is the use of a private site for this purpose, when the more suitable infrastructure available at a Government site, such as Hanford or Savannah River, is present and in operation. *The influence of local and regional politicians will likely determine the national political response, through pressure and lobbying efforts on elected officials in Washington.*

The most challenging aspects of the use of existing reactors for plutonium disposition are: the issue of political acceptance; the utility willingness to participate; and the cost and risk to the Government of acquisition of a reactor or reactors for the mission. The advantages of the use of existing reactors will depend on the cost comparisons with construction of an evolutionary reactor at a Government site. The schedule in either case is determined by the fuel fabrication facility schedule to produce the fuel assemblies.

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1.0 INTRODUCTION

1.1 Technical Synopsis of Study from Inception to the Present

1.1.1 Program background

Phase I of the study, completed in May, 1993, focused on the three requested alternatives for disposition of the excess weapons-grade plutonium: spiking, spent fuel and destruction. The spiking alternative required that the fuel be irradiated as quickly as possible to create a radiation barrier to diversion. The spent fuel alternative, through longer irradiation, would convert the plutonium into a spent fuel form similar to the commercial spent fuel already being stored in the U.S. and the Newly Independent States of the former Soviet Union. The destruction alternative would accomplish maximum destruction of the total amount of plutonium, including all isotopes, within the allotted 25 year mission time frame. A "triple mission" reactor, dispositioning plutonium, producing saleable electricity and irradiating targets for tritium production, was conceived.

Additional studies were authorized by the DOE by modifications to the original contract. The objective of these modifications was to continue the work initiated under the original scope of work, to pursue some of the identified uncertainties in more detail, concentrating on the Spent Fuel alternative, and with less emphasis on the tritium aspects. The mission requirements were revised, to include a base case of 50 MT Pu to be put into the spent fuel mode in 25 years from the start of the program (including design and construction), with alternatives of 50 MT Pu over the life of the plant, 100 MT Pu in 25 years, and 100 MT Pu over the life of the plant. The program evaluation requirements, including cost, licensing status and safeguards evaluation, were also put on a more systematic basis. This work was covered in an earlier Final Report, submitted on April 30th, 1994, herein referred to as the Evolutionary Reactor Report.

An additional modification has been added to the workscope, to study existing ABB-CE reactors for their capability to undertake the Pu disposition mission. This work is covered in this final report. Two cases, to disposition 50 MT and 100 MT excess weapons-grade plutonium, are to be investigated, using existing U.S. reactors, or those under construction. The time to complete the mission was not specified, but would be related to the remaining lifetime of the reactor(s) chosen. The number of existing reactors to be used for the mission would depend on the capability of each to dispose of plutonium, and their available lifetime.

1.1.2 Task Description and Summary

Task 1 Identify U.S. ABB-CE designed LWRs which can be converted to MOX fuel use. Required Information:

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- MOX Core Loading (one-third of full MOX core load)
- Plutonium Enrichment in MOX fuel which the reactor can utilize
- Conversion Requirements to include technical and physical changes to current design, testing and any additional facilities needed to store, handle and transfer MOX fuel and MOX spent fuel (assume the MOX fuel facility is owned by the Government)
- Remaining Lifetime of the reactor (do not assume any extensions to current NRC regulations)
- Weapons grade plutonium throughput

Summary of Work Performed:

Core reactivity and basic fuel management calculations were conducted on the selected reactors (with emphasis on the System 80 units as being the most desirable choice). Methods used were identical to those reported in the Evolutionary Reactor Report. From these calculations, the basic mission capability was assessed. The selected reactors were studied for modification, such as the addition of control rod nozzles to increase rod worth, and internals and control system modifications that might also be needed. Other system modifications studied included the use of enriched boric acid as soluble poison, and examination of the fuel pool capacities.

Sections Where Reported:

Sections 1.2, 2.1, 2.2, 2.3, 2.5, 2.6, 3.1, 3.3. Section 10 also describes information on the alternative mission.

Task 2: For each reactor, determine the required MOX fuel design and provide any research, testing, or operational data that exist for the determined design. Provide design and development needs to test and license the fuel design (include NRC and Environmental Protection Agency (EPA) Requirements).

Summary of Work Performed:

The basic geometry and mechanical characteristics, materials and fabrication techniques of the fuel assemblies for the selected existing reactors are the same as for System 80+. There will be some differences in plutonium loading, according to the ability of the reactors to load MOX fuel. These differences are not expected to affect licensability or EPA requirements. Therefore, the fuel technology and fuel qualification sections provided in the Evolutionary Reactor Report apply to the existing reactors. An additional factor, in that the existing reactor availability presupposes the use of that reactor for the irradiation of Lead Test Assemblies, is discussed.

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Sections Where Reported:

Section 2.1, 2.6, 3.2, 3.3, 4.4, 5.3.

In the Evolutionary Reactor Report: Section 3.2, Appendix D.

Task 3: Determine the institutional and regulatory requirements and necessary institutional, Government and commercial interfaces required to burn plutonium in existing reactors.

Summary of Work Performed:

The reactor operating and facility licenses for the operating plants were reviewed. Licensing strategies for each selected reactor were identified. Government, institutional and commercial interfaces were identified with various scenarios.

Sections Where Reported:

Section 1.3, 4.1, 5.2, 5.4.

Task 4: Determine any additional requirements to process and store the waste resulting from burning MOX fuels.

Summary of Work Performed:

The spent fuel pool for the selected reactors (Palo Verde) was reviewed for capacity and upgrade requirements. Reactor waste streams were identified and assessed in comparison to uranium fuel operations.

Sections Where Reported:

Section 2.2, 2.5, 8.3

Task 5: Cost and schedule. Provide detailed cost assessments and schedules for each identified LWR. Integrate licensing requirements (NRC,EPA, etc.) into cost and schedule estimates.

Summary of Work Performed:

Cost assessments and schedules for converting to plutonium disposition were estimated for some of the major modification items. Economic factors (incremental costs associated with using weapons plutonium) were listed and where possible under the scope of work, estimates were made.

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Sections Where Reported:

Section 6.2, 6.6.

Task 6: Determine the possible incentives to offer commercial utilities to burn plutonium in existing reactors.

Summary of Work Performed:

Discussions have been initiated with utilities, but which are largely incomplete.

Sections Where Reported:

shipping, transfer and storage of plutonium if utilizing existing commercial facilities and identify methods to abate the risks.

Summary of Work Performed:

The Evolutionary Reactor Report was reviewed for application to existing reactors, and modified where appropriate to the level of detail of the investigation.

Sections Where Reported:

Section 7.2, 7.3.

Task 8: Identify the environmental issues and methods to address environmental concerns, including public involvement in licensing approvals.

Summary of Work Performed:

The environmental section of the Evolutionary Reactor Report was reviewed for its application to the situation of existing reactors.

Sections Where Reported:

Section 8.1, 8.3, 8.4.

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1.2 Definition of Commercial Power System Basis for PDS

1.2.1 Introduction

The selection of candidate nuclear power plants, with Nuclear Steam Supply Systems by C-E, that would be suitable for the plutonium disposition mission, was made on the basis of:

- Ability to load a maximum amount of plutonium - up to full MOX fuel loading;
- Specific design features geared towards plutonium fuel loading;
- Sufficient remaining lifetime - at least 20 years remaining, counting a conversion time of 5 years;
- Location - preferably a remote site with sufficient capacity at one site to carry out the mission.

Table 1.2-1 lists the plants considered, with major parameters of interest for this study. The following paragraphs discuss the merits of each plant with respect to the plutonium disposition mission ranked in order of suitability.

1.2.2 Palo Verde Nuclear Generating Station (PVNGS)

This is the most suitable plant for the mission of positioning the excess weapons plutonium. The site contains three ABB-CE System 80 reactor power plants, each rated at 3800 MWt core heat, and 1256 MWe net electric power. The design features of System 80 were established specifically to enable the reactor to be fueled with All Plutonium Reactor (APR), or 100% MOX fuel, from recycled plutonium. These features are summarized in Table 1.2-2. Each unit, in the time remaining in the operating license, could disposition 40 to 42 MT weapons grade plutonium, allowing for a 5 year startup phase from today. Because of the three units on one site, and the large reactor size, and the special features to accommodate MOX fuel, this site by itself can accommodate the reference or alternate mission quantities of excess weapons grade plutonium within the lifetime remaining on the units. Furthermore, the site location, west of Phoenix, AZ, in the desert, is well isolated and should be amenable to the additional safeguards attendant to the use of weapons grade plutonium.

The site is located approximately 34 miles west of the nearest boundary of the city of Phoenix in Maricopa County, Arizona. Buckeye Salome Road is north of the site and runs in a northwest to southeast direction. A paved county road, Wintersburg Road, runs north to south along the west edge of the site, and Ward Road (paved), sometimes called Elliot Road, runs east to west at the southern end of the site. A

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Southern Pacific railroad line runs southwest to northeast 5 miles south of the power plant complex.

1.2.3 Washington Nuclear Power Supply - Unit 3 (WNP-3)

As a System 80 plant, this shares the suitability for the mission that the PVNGS units display, in terms of design features and MOX loading capability. It has the same power level as a single PVNGS unit. The plant was only 75% complete, as measured by the constructor's schedule, when construction was halted by the plant owner. Completion of this plant gives the advantage, however, that it could be completely devoted to the plutonium mission. By itself, it could disposition 60 MT weapons grade plutonium, thereby accomplishing the reference mission in its normal licensed lifetime. The major drawback is that the plant would have to be completed, involving considerable expense and time, and due to significant amounts of rework and relicensing.

The project site is located in the southeastern portion of Grays Harbor County. The site is approximately 16 miles east of Aberdeen, the nearest sizable population center, and approximately three miles south of the community of Satsop. The central site area lies in Section 17 of the Township 17 North, Range 6 West. The centerline coordinates for the reactor are N46° 57' 33" Latitude and W 123°, 27' 58" Longitude (WNP-3).

1.2.4 3410 MWt Units

Three reactors are in operation featuring this design. They share the same fuel assembly design as the System 80, but in a smaller core configuration (217 assemblies compared to 241 assemblies for System 80). The thermal power of these reactors is 3410 MWt, and electrical output of 1100 to 1104 MWe. The features specific to plutonium recycle, as listed in Table 1.2-2, are not included in the design. The inherent capability of the each core in these reactors, when considering the use of MOX fuel containing weapons-grade plutonium, and allowing for a 5 year startup, is 32 to 35 MT, in the licensed operating time remaining. Therefore, two of these units could accomplish the reference mission, and all three may be able to accomplish the

The San Onofre site is located on the coast of southern California in San Diego County, approximately 62 miles southeast of Los Angeles and 51 miles northwest of San Diego. The approximate coordinates are latitude 33° 22' 10" N and longitude 117° 33' 30" W. The site is located entirely within the boundaries of the United States Marine Corps Base, Camp Pendleton, California, near the northwest end of the 18-mile shoreline. The site is approximately 4500 feet long and 800 feet wide, comprising 84 acres including 27.7 acres for the power block and site switchyard and 25.1 acres for

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parking, access area and miscellaneous structures. Units 2 and 3 are located southeast of and immediately adjacent to Unit 1.

The property upon which the station is built is being leased from the United States Government until May 12, 2023. The nearest privately owned land is approximately 2.5 miles from the site.

The Waterford-3 site is located on the west (right descending) bank of the Mississippi River near Taft, Louisiana in the northwest portion of St. Charles Parish. About three miles westward is the eastern boundary of St. John the Baptist Parish. The coordinates for the reactor are 29° 59' 42" north latitude and 90° 28' 16" west longitude. The UTM coordinates are 3320744 meters north, and 743963 meters east.

Kenner, the nearest population center is 13 miles east of the site. Approximately 25 miles east-southeast of the site is the city of New Orleans, and approximately 50 miles north-northwest is the city of Baton Rouge. The exclusion radius is taken as 915 meters, and the low population zone is a two mile radius.

1.2.5 2815 MWt Unit

A single plant is in operation featuring this design, and also having the same fuel assembly design as the 3410 MWt plants, but only having 177 fuel assemblies. The thermal power is correspondingly lower, at 2815 MWt, and the electrical power lower, at 858 MWe. This plant also does not have any of the features specific to plutonium recycle, as listed in Table 1.2-2. The capability of this reactor, considering the use of MOX fuel containing weapons-grade plutonium, and allowing for a five-year startup, is 23 MT in the licensed operating time remaining. In conjunction with other units referred to above, it could be used to complete the plutonium disposition mission. This plant is located at Russellville, AR (Arkansas Nuclear One - Unit 2).

Arkansas Nuclear One - Unit 2 is located adjacent to Arkansas Nuclear One - Unit 1 on a peninsula in Dardanelle Reservoir on the Arkansas River in Pope County, Arkansas. The plant is about six miles west-northwest of Russellville, Arkansas, and about two miles southeast of the village of London, Arkansas.

The 1100-acre site allows a minimum exclusion area radius of 0.65 miles. Easements have been obtained from the U.S. Army Corps of Engineers for exclusion rights during periods of emergency to portions of the bed and banks of the Dardanelle Reservoir that are within the 0.65-mile radius.

The site is located in a gently rolling valley south of the Boston Mountains and was previously used mostly for pasture land.

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The plant grade elevation (near the center of the site) is 353 feet above mean sea level (msl) while the Dardanelle Reservoir fluctuates between 336 feet (normal pool) and 338 feet (maximum flood control pool) above msl.

The site is characterized by remoteness from flooding; sound, hard rock for structure foundations; a reliable network for emergency power; and favorable conditions of hydrology, geology, seismology and meteorology.

1.3 Licensing Status

Note: Licensing status reviews for San Onofre Nuclear Generating Station (SONGS 2 and 3) and for Arkansas Nuclear One - Unit 2 (ANO-2) have not been completed.

1.3.1 Palo Verde 1, 2, and 3

The NRC issued operating licenses on December 31, 1984 for Unit 1, December 9, 1985 for Unit 2, and March 25, 1987 for Unit 3. These operating licenses are valid for 40 years. Each of the Palo Verde units is based on the System 80 Standard Design which was designed to accommodate plutonium loadings up to and including that of all-mixed-oxide-fueled core (APR). However, certain sections of the Final Safety Analysis Report (FSAR) need to be modified and associated NRC review and approval are required to address unit operation with mixed-oxide (MOX) fuel. The NRC review and approval are expected to take not more than one year assuming no intervenor. Accordingly, a licensing status of 4 (per Section 3.5 of UCRL-ID-113055) is assigned to Palo Verde Units 1, 2, and 3.

1.3.2 Waterford 3

The NRC issued an operating license for the unit on December 18, 1984. The license is valid for 40 years. Since the unit was not specifically designed to accommodate MOX fuel operation, facility modifications for MOX operation as well as associated NRC review and approval are expected to be required. The NRC review and approval are expected to take not more than one year assuming no intervenor. Accordingly, a licensing status of 3 (per Section 3.5 of UCRL-ID-113055) is assigned to Waterford Unit 3.

1.3.3 WNP-3

The WNP-3 FSAR was docketed by the NRC in August 1982 and seven amendments have been submitted. The last amendment was submitted in February 1988 before the completion of the unit was deferred indefinitely. The FSAR references Combustion Engineering Standard Safety Analysis Report (CESSAR) as WNP-3 utilizes a System 80 Standard Design. Final Design Approval for the System 80 Standard Design was issued on December 21, 1983. The balance of the FSAR has been reviewed in large part by the NRC staff and this review has been documented in a draft Safety

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Evaluation Report (SER). The Operating License Environmental Report was docketed in August 1982. The NRC issued a Final Environmental Impact Statement in May 1985. The Construction Permit is valid through July 1, 1999. The National Pollutant Discharge Elimination System (NPDES) Permit has been reissued four times, and is valid through 1996. All needed permits and licensing, including environmental permits, have been maintained. The unit is based on the System 80 Standard Design which was developed to accommodate plutonium loadings up to and including that of an APR. However, certain sections of the FSAR and Environmental Report need to be modified to address unit operation with MOX fuel and the unit, which, according to the status of the construction schedule when the completion of the unit was deferred, is approximately 75 percent complete, requires an operating license review when the unit is reactivated. The NRC review and hearing associated with the operating license are expected to take about three years. Accordingly, a licensing status of 3 (per Section 3.5 of UCRL-ID-113055) is assigned to WNP-3.

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TABLE 1.2-1

WEAPONS Pu DISPOSITION CAPABILITY OF EXISTING C-E REACTORS

PLANT (note 1)	LOCATION	POWER LEVEL MWe	COMMERCIAL OPERATION DATE	REMAINING LIFETIME YEARS (note 2)	ESTIMATED Pu ANNUAL CAPACITY MT	Pu DISPOSITION CAPABILITY MT (note 3)
ANO-1 Unit	Russellville, AR	858	Mar-80	26	1.10	23.1
San Onofre 2	San Clemente, CA	1100	Aug-83	29	1.35	32.4
San Onofre 3	San Clemente, CA	1100	Apr-84	30	1.35	33.8
Waterford 3	Taft, LA	1104	Sep-85	31	1.35	35.1
Palo Verde 1	Wintersburg, AZ	1270	Feb-86	32	1.50	40.5
Palo Verde 2	Wintersburg, AZ	1270	Sep-86	32	1.50	40.5
Palo Verde 3	Wintersburg, AZ	1270	Oct-87	33	1.50	42.0
WNP-3	Satsop, WA	1240	Not Complete	40	1.50	60.0

Notes: 1 includes only plants less than 20 years old
2 Lifetime from today, assuming a 40 year operating life.
3 Allowing 5 years from today for conversion to Pu burning

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TABLE 1.2-2

PLUTONIUM IMPACT ON PWR SYSTEM REQUIREMENTS

<u>System or Component</u>	<u>Changes in Requirements</u>
Plant Cooling System	1) Increased core decay heat removal capacity for plant cooldown and safety. 2) Accommodation of increased long-term decay for spent MOX 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 Refueling Water Storage Tank (RWST) 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 MOX 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 MOX fuel in conjunction with UO ₂ fuel.
Radwaste System	Addition of tritium removal system to accommodate higher tritium production in primary coolant

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1.4 Reference Case and Alternative

The cases evaluated are discussed in Table 1.4-1.

The base case of dispositioning 50 MT of weapons plutonium and the alternative of dispositioning 100 MT can be accomplished in a number of ways with combinations of existing ABB-CE reactors. Two combinations are shown for the base case and alternate, that minimize the total number of sites involved, and make use of System 80 and Non-System 80 plants. The rate of consumption of plutonium may be increased by conducting reactor modifications, as discussed in Section 2, or by decreasing the discharge burnup of the MOX fuel assemblies. In these cases, a trade-off should exist between shortening the time of completion of the mission and possibly involving fewer plants, and cost of modifications, fuel cost and isotopic content of the discharged plutonium (the lower burnup leads to lower Pu^{240} content, and lower activation of the fuel assembly at discharge, which still may be acceptable from a diversion point of view). This trade study was considered to be beyond the scope of this report, but would be important to the final decision-making process.

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TABLE 1.4-1
REFERENCE CASE AND ALTERNATIVES

	Reference		Alternative	
	Case B-1	Case B-2	Case A-1	Case A-2
Mission - MT Pu	50	50	100	100
Plants Selected	Palo Verde- 1 and 2	SONGS-2 and -3	Palo Verde- 1, -2 and -3	SONGS-2 and 3, Waterford - 3
Years to Complete Burnup	18	19	23	26
Fuel Req'ts MT MOX/yr	67	90	100	135
MOX Fuel Fab'd. MT	1150	1710	2250	3420
Full Power MWe	2540	2200	3810	3304
Estimated Cap. Factor	82%	75%	82%	75%
Elec. Energy MW-Days	1.31×10^7	1.14×10^7	2.57×10^7	2.29×10^7

DOE Plutonium Disposition Study

Analysis Of Existing ABB-CE Light Water Reactors For The Disposition Of Weapons-Grade Plutonium

Contract No. DE-AC03-93 SF19682

**A Final Report
by
Combustion Engineering, Inc.
Windsor, Connecticut**

June 1, 1994

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

2.0 TECHNICAL DESCRIPTIONS

2.1 Reactor and Reactor Cooling System

2.1.1 Mechanical and Neutronic Design and Performance

The following description of the Reactor Cooling System (RCS) applies to all of the plants discussed in this report unless otherwise noted. A comparison of the specific reactor characteristics for a System 80 (3817 MWt) plant and a non-System 80 (3410 MWt or 2825 MWt) plant is shown in Table 2.1.1-1. Most of the description for the 3410 MWt plant is also applicable to the 2825 MWt plant, except for core size, power output and related parameters.

The plants are Pressurized Water Reactor (PWR) systems with two coolant loops. The RCS circulates water in a closed cycle, removing heat from the reactor core and internals and transferring it to a secondary (steam generating) system. The steam generators provide the interface between the reactor coolant (primary) system and the main steam (secondary) system. The steam generators are vertical U-tube heat exchangers in which heat is transferred from the reactor coolant to the main steam system. The System 80 plant steam generator has an integral economizer while the 3410 MWt plant does not have an integral economizer. The System 80 integral economizer enhances the steam generator thermal effectiveness by allowing it to operate at a higher steam pressure without an increase in heating surface. Reactor coolant is prevented from mixing with the secondary steam by the steam generator tubes and by the steam generator tube sheet, making the RCS a closed system. Thus, a barrier is formed to prevent the release of radioactive materials from the core of the reactor to the containment building.

The arrangement of the RCS, applicable to System 80 or non-System 80 plants, is shown in Figures 2.1.1-1 and 2.1.1-2. The major components of the system are the reactor vessel; two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps; a pressurizer connected to one of the reactor vessel outlet pipes; and associated piping. All components are located inside the containment building.

System pressure is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer or is condensed by the pressurizer spray to limit pressure variations caused by contraction or expansion of the reactor coolant. The average temperature of the reactor coolant varies with power level and the fluid expands or contracts changing the pressurizer water level.

The charging pumps and letdown control valves in the Chemical and Volume Control System (CVCS) are used to maintain a programmed pressurizer water level. A continuous but variable letdown purification flow is maintained to keep the RCS

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chemistry within prescribed limits. A charging nozzle and a letdown nozzle are provided on the reactor coolant piping for this operation. The charging flow is also used to alter the boron concentration or correct the chemical content of the reactor coolant.

Other reactor coolant loop penetrations are the pressurizer surge line in one reactor vessel outlet pipe; the four safety injection inlet nozzles, one in each reactor vessel inlet pipe; two outlet nozzles to the shutdown cooling system, one in each reactor vessel outlet pipe; two pressurizer spray nozzles; vent and drain connections; and sample and instrument connections.

Overpressure protection for the reactor coolant pressure boundary is provided by four spring-loaded American Society of Mechanical Engineers (ASME) Code safety valves connected to the top of the pressurizer. These valves discharge to the reactor drain tank where the steam is released under water to be condensed and cooled. If the steam discharge exceeds the capacity of the reactor drain tank, it is relieved to the containment atmosphere via a rupture disc installed in the tank.

Overpressure protection for the secondary side of the steam generators is provided by spring-loaded ASME Code safety valves located in the main steam system upstream of the steam line isolation valves.

Components and piping in the RCS are insulated with a material compatible with the temperatures involved to reduce heat losses and protect personnel from high temperatures.

The RCS shielding permits limited personnel access to the containment building during power operation. The reactor vessel sits in a primary shield well. This and other shielding reduces the dose rate within the containment and outside the shield wall to acceptable levels during full power operation.

The reactor coolant is circulated by four electric-motor-driven single-stage centrifugal pumps. The pump shafts are sealed by mechanical seals. The seal performance is monitored by pressure and temperature sensing devices in the seal system.

Reactor coolant enters the reactor vessel through inlet nozzles, flows downward between the reactor vessel shell and core barrel into the lower plenum where flow distribution is equalized, and then upward through the core removing heat generated by the nuclear reactor core. Coolant leaves the reactor vessel shell through outlet nozzles and enters the tube side of the steam generators, where heat is transferred to the secondary system. Steam generated in the shell side of the steam generator passes through moisture separators and dryers to ensure that moisture content at the steam generator outlet is minimized. After leaving the steam generators, reactor coolant is returned to the reactor vessel by limited leakage mechanical seal reactor coolant pumps.

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The supports for each of the individual components of the RCS are designed to form an integrated support system which mitigates the effects of earthquakes, branch line breaks, and steam line breaks. This capability has been achieved without sacrificing the ability of the RCS to expand thermally with minimum restraint.

The design pressure and temperature of the RCS are 2500 psia and 650°F. The system will operate at a pressure of 2250 psia and has a design lifetime of 40 years.

2.1.1.1 Reactor Vessels

The pressurized water reactor vessels for the ABB-CE plants of interest for plutonium disposition were previously manufactured by ABB-CE at its Chattanooga facility. During their manufacture all pertinent requirements of the applicable ASME Boiler and Pressure Vessel Code were meticulously followed. Figure 2.1.1-3 shows a typical reactor vessel assembly.

Figure 2.1.1-4 describes the coolant flow through a typical reactor vessel. Reactor coolant enters the reactor vessel through inlet nozzles, flows downward in the annulus between the reactor vessel shell and core support barrel into the lower plenum where flow distribution is equalized by the flow baffle, and then upward through the core fuel assembly removing heat generated by the nuclear reactor core. Coolant leaves the reactor vessel shell through outlet nozzles destined for the inlet plenum of a steam generator.

System 80 Reactor Vessel

The in-service System 80 reactor vessels, shown in Figure 2.1.1-5, are viable candidates for the plutonium disposition program. The vessels were fabricated and designed to ASME Code addenda later than the Summer of 1972; therefore, nonductile (fracture) evaluations were performed on each vessel. Other than their size, the System 80 reactor vessels are distinguishable from 3410 MWt plant vessels by their column supports and bottom head location of their ICI nozzles. A comparison of the System 80 characteristics to those of the 3410 MWt plant reactor vessel is presented in Table 2.1.1-2.

The System 80 reactor vessel CEDM nozzles penetrate the closure head, while the ICI nozzles are located in the bottom head. The existing CEDM nozzle penetration pattern for the System 80 vessel closure head is shown in Figure 2.1.1-6. With the exception of the vent pipe penetration, all of the closure head penetrations are receptive to the installation of 4.752 in. diameter CEDM nozzles. At least sixteen (16) pattern locations are available for additional CEDM nozzles (See Figure 2.1.1-6). The CEDM nozzles are installed in the closure head using partial penetration welds on the inside surface; therefore, per the ASME Code, the nozzle welds shall be subjected to substantially no mechanical loads during normal operation. This is assured by an interference fit between the CEDM nozzles and the closure head penetrations.

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

3410 MWt Plant Reactor Vessel

The in-service 3410 MWt plant reactor vessels, Figure 2.1.1-7, can be used in the plutonium disposition program. The vessels were fabricated and designed to ASME Code addenda earlier than the Summer of 1972; therefore, nonductile (fracture) evaluations were not made on the vessel shells. A comparison of the 3410 MWt plant reactor vessel characteristics to those of the System 80 is presented in Table 2.1.1-2.

In addition to the single vent pipe penetration, the closure head of the 3410 MWt reactor vessel contains penetrations for CEDM and instrument nozzles (See Figure 2.1.1-8). The 91 CEDM penetrations are designed for 4.050 in. diameter nozzles and the ten (10) instrument penetrations for 5.563 in. diameter nozzles. Nine (9) nozzles can be added within the limiting surface of the CEDM nozzle pattern. The

operations. This is assured by an interference fit between the nozzles and the closure head penetrations.

2.1.1.2 Reactor Vessel Internals

The Reactor Vessel Internals provide the structural support for the core and fuel located within the reactor vessel. The internals also provide for the alignment of the core with the CEA's, CEDMs, and instrumentation, and serve as a shield to protect the reactor vessel from radiation damage. In addition, they also accommodate the expansion and contraction of the core.

The Reactor Vessel Internals are composed of core support structures and internal structures. Core support structures are those that restrain the core. All other structures within the reactor vessel, exclusive of the fuel assemblies, in-core instrumentation, and control element assemblies, are considered internal structures. Core support structures are designed, fabricated and stamped in accordance with the ASME boiler and pressure vessel code Section III, Sub-section NG.

The Reactor Vessel Internals and their relationship to the rest of the reactor are shown in Figure 2.1.1-9. The Reactor Vessel Internals guide the primary coolant through the core. The core support barrel assembly, in conjunction with the water in the downcomer annulus, provides a shield for the reactor vessel against radiation damage by the neutron flux. The core support structures also provides alignment of the core with the control drives, instrumentation, and the reactor vessel.

The reactor vessel internals are comprised of two basic components: (1) a core support barrel assembly and, (2) an upper guide structure assembly. Both assemblies are

supporting the upper end tube sheet structure that provides a guide path for individual control rods. The lower end of the tube sheet provides support and alignment of the upper end of the Fuel Assemblies. A shroud assembly within the cylindrical shell provides a guide path for the control element assemblies (CEAs). The arrangement of the upper guide structure is shown on Figure 2.1.1-10.

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The unique System 80 tubesheet upper guide structure allows control assemblies to serve more than one fuel element and permits greater flexibility in selecting rod strength in relation to the functions performed. Two types of control assemblies are provided, utilizing four or twelve absorber rods. Four-rod assemblies are used for load maneuvering and shaping of radial power distribution. This design feature allows use of the same proven control element drive mechanism for all types of control assemblies.

Twelve rods or fingers are used for the reactivity shutdown control assemblies which bridges five fuel assemblies. This expanded spread of control fingers, permitted by the design of the tubesheet upper guide structure, provides more uniform and effective reactivity control. This improved method of control rod function is one of the key reasons for the plutonium capability of System reactors.

2.1.1.2.3 Flow Skirt

The flow skirt is a cylindrical structure having a large number of holes and located between the core support barrel and lower head of the reactor vessel. The flow skirt provides coolant flow distribution reducing the pressure drop in the lower plenum region. The flow skirt is designed to withstand the static, cyclic, and shock loads resulting from hydraulic pressure drop, vibration, and earthquake accelerations.

2.1.1.3 Control Element Drive Mechanisms and Control Elements

Control Element Assemblies (CEAs) for System 80 can be either four or twelve fingered as shown in Figure 2.1.1-11 and Figure 2.1.1-12. The CEA fingers are fixed to a spider which serves as the central support structure for the 4- and 12-fingered CEA assemblies. The hub of the spider also couples the CEA to the drive mechanism through the Control Element Drive Mechanism (CEDM) extension shaft assembly.

Each control element in the CEA is guided and shrouded from the main coolant flow above the core by the individual control finger shroud tubes and by the CEA shroud. In-core guidance is provided by the guide tube, which is an integral part of the fuel assembly. The guide tubes are open near the bottom to allow entry of coolant. Inlet orifices and control element fingers limit bypass flow through the tubes.

The CEAs for the non-System 80 plants are similar to the four fingered CEA shown on Figure 2.1.1-11. Some have a center finger as well. The locations of the CEAs for the non-System 80 reactor core is given in Figure 2.1.1-14. All CEAs used in the core design contain neutron absorbing (poison) materials which extend over the active length when fully inserted. The locations of the CEAs for the System 80 reactor core is shown in Figure 2.1.1-13.

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The magnetic jack CEDM is a completely sealed, magnetically operated linear actuator. External coils produce a magnetic field, which operates driving and holding latches. Proper sequencing of these coils produces the linear motion of the driveshaft. Using external driving coils, the mechanism transfers loads with no sliding wear on latching members.

The general arrangement of the magnetic jack CEDM is indicated in Figure 2.1.1-15, showing the location of the external coils and latches. The motor housing is of Type 403 heat treated stainless steel and the upper pressure housing is austenitic stainless steel. The pressure retaining components are designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code for 2500 psi and 650°F.

Position indication is achieved by means of redundant reed switch position transmitter assemblies (two per CEDM) attached to the upper pressure housing, and also by counting power pulses which actuate the CEDM driving CEAs up or down. Each reed switch assembly is operated by a magnet in the upper end of the driveshaft. Travel is limited by reed switches which actuate upper and lower electrical limit relays. The travel is also limited by the length of the toothed section of the driveshaft which is designed to prevent the extension shaft assembly from being driven into the pressure housing.

All electrical connections are made at the top of the mechanism with pin type stainless steel electrical connectors. A shroud surrounds the upper pressure housing enclosing the reed switch position transmitter assemblies, providing additional structural support for the CEDM, and serving as a lifting device for removal of the coil stack assembly for inspection of the pressure housing.

The CEDM motor housing is completely seal welded to prevent leakage during operation. However, the design of the CEDM facilitates service of its internals. The top seal on the motor housing is designed to permit removal of the CEDM internals by remotely cutting (and rewelding) the pressure housing. The lower seal and attachment to the vessel nozzle will not be disturbed for any normal mechanism maintenance or inspection but is designed for housing replacement, should a housing become damaged.

2.1.1.4 Reactor Core and Fuel

The System 80 Reactor is designed to generate the required thermal output of 3817 MWt. (core power is 3817 MWt; total power is 3817 MWt, including 17 MWt from the reactor coolant pumps). For the 3410 The core provides a flowpath for the forced circulation of coolant to remove heat generated by the core under all power operating conditions, and for the removal of decay heat by natural or forced circulation under shutdown conditions. The fuel contains the fuel rods which are designed to transfer the heat generated by fission reactions in the fuel pellets to the reactor coolant. The fuel also contains poison rods which provide negative reactivity in the fuel throughout

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core life in conjunction with soluble boron distributed throughout the reactor coolant. The heat transferred from the core and fuel is circulated by the reactor coolant to the steam generator where the heat is used to generate steam.

The fuel rod design provides two barriers to the escape of fission products. The ceramic pellets operate well below the melting point, and as a result, retain most of the fission products within the structure of the fuel. The small fraction of released fission products is contained within the hermetically sealed Zircaloy-4 cladding tube. In conjunction with the reactor coolant system envelope, the containment, and reactor building, these barriers provide five barriers against the release of fission products.

The combination of cladding thickness and fuel rod pressurization provides additional margin to the design criteria regarding internal pressure buildup within the fuel rod and the structural integrity of the fuel cladding. The positive spring loading feature of the leaf spring spacer design restricts lateral fuel rod motion and thus prevents fretting, yet allows free axial expansion of each element. These provisions, combined with the specified burnup and precise quality control during fabrication, reduce the probability of fuel rod failure.

The cylindrical control elements are provided with full length guidance in the core and in the area of radial flow above the core, and move downward under the influence of gravity when released. The unique design of the upper guide structure and control element assemblies provides exceptional mechanical simplicity, and ruggedness for withstanding and protecting all control element assembly fingers from the combined effects of seismic and blowdown loads resulting from a LOCA. This further adds to the reliability of operation of the reactor protection systems. The control element arrangement provides ample shutdown margin under all conditions and without restriction of the fuel management scheme, even with the control element of highest worth stuck out of the core.

2.1.1.4.1 Configuration

The reactor core and fuel for the design is made up of the fuel assemblies including MOX fuel and poison rods, control element assemblies (CEA) and drives (CEDMS).

The System 80 reactor core consists of an array of 241 mechanically identical fuel assemblies in an arrangement that approximates a right circular cylinder. The 3410 MWt design has 217 fuel assemblies in a similar array. The core also includes control element assemblies (CEAs), in-core instrumentation assemblies, and neutron sources inserted into the fuel assemblies, together with the reactor vessel internals which support and position the core and guide the coolant flow. There are no internal shrouds, channels or poison curtains in the core. This simplifies the reactor design and permits flexibility in fuel arrangement. All of the structural materials in the active core zone, including the CEA guide tubes and the spacer grids, are Zircaloy, which eliminates concern over the formation of lower melting point eutectic during a loss of

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coolant accident (LOCA) as a result of using dissimilar metals and which provides excellent neutron economy.

Figure 2.1.1-16 shows the System 80 cross-sectional arrangement of the fuel assemblies as positioned in the reactor core, and the arrangement of the core within the core support structure. The individual components of the reactor core are described in the following paragraphs.

2.1.1.4.2 Fuel Rod

Each fuel rod for the System 80 or non-System 80 plants consists of dish-ended and chamfered MOX pellets, upper and lower spacer discs, a cladding tube, upper and lower end caps, and a helical spring. The fuel is manufactured in the form of pellets having a material microstructure and configuration which minimizes the effects of fuel densification during irradiation. The pellet column rests on an aluminum oxide spacer disc at the bottom of the cladding tube, and is held in place by a stainless steel compression spring acting on a similar spacer disc at the top of the column. The space at the top of the column allows for fuel and gas expansion. A cold diametrical gap of sufficient size is allowed to provide for differential expansion between the cladding and fuel, which limits clad strain. Net unrecoverable circumferential clad strain will not exceed one (1) percent, as predicted by analysis considering clad creep and fuel-clad interaction effects.

The cladding tube is slightly cold-worked Zircaloy-4. The ratio of clad thickness to outer diameter is large enough to provide adequate margin in limiting the maximum short-term cladding stresses to less than the yield strength. The fuel rods are internally pressurized with helium to improve both their reliability and performance characteristics.

Pressurization results in reduced clad stresses and strains because of a smaller

LOCA fuel rod. The cladding tube is designed not only with respect to clad integrity at steady state conditions, but also in the case of transients where peak clad temperature is important.

The end caps are welded to the cladding tubes by an exclusive ABB-CE process involving magnetic force welding. This process has the advantage of producing a high degree of uniformity and consistency of weldment, thus providing a very high degree of leak tightness. There are no gas bubbles, as may be found in more conventional methods of welding, that could lead to weld porosity and leakage. All end cap welds are 100 percent helium leak tested.

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2.1.1.4.3 Fuel Assembly

The fuel rods for the System 80 and non-System 80 plants are arranged in a 16 x 16 square array to form a fuel assembly, as shown in Figure 2.1.1-17. The lower end cap of each fuel rod engages the lower end fittings of the fuel assembly, which provides lateral support of the rods. Lateral support and positioning is maintained throughout the length of the rods by spacer grids. The upper end of the fuel rod is free to expand in the axial direction.

Five (5) Zircaloy guide tubes are welded to the spacer grids and mechanically attached to the upper and lower end fittings. These guide tubes are a key feature of the fuel assembly design. They provide channels which guide the CEAs over their entire length of travel within the core, as well as space for the in-core instrumentation and neutron sources. These guide tubes form the longitudinal structure of the assembly and offer the advantage of simple, rugged construction.

The guide tubes form a closely spaced repetitive array of water channels throughout the core, which permits the use of rugged, yet adequately flexible control elements. The use of rugged, large-size control elements simplifies the design and increases the reliability of associated reactor hardware.

The fuel spacer grids maintain spacing between the fuel rods and restrict their movement. These grids consist of a series of preformed strips joined in an eggcrate fashion and welded together. Each fuel rod is laterally positioned by two leaf springs which are integral with two of the adjacent strips. The springs press the fuel rod against two stiff arches, which are integral with the other two adjacent strips. The grid perimeter strips have punched-out sections adjacent to the fuel rods to provide local cooling. Although the spacer grids position the fuel rods laterally, the rods are still free to expand axially.

The fuel assembly lower end fittings contain coolant flow holes. Alignment of the lower end and support of the assembly is provided by the core support structure. Lateral location of the upper end of each fuel assembly is provided by the lower ends of the control element shroud tubes which extend downward from the alignment plate which is part of the guide structure above the core. Fuel assembly positioning is accomplished by the control element shroud tube extensions into which the fuel assembly extension is guided. The shroud tube extensions prevent the assemblies from moving so far upward as to disengage from the lower core support grid in the event of an accidental flow surge. Holddown springs on the upper end fitting react against the shroud tube extensions preventing fuel assembly uplift due to hydraulic forces and allowing differential expansion between the fuel assembly and the core internals. The upper end fitting also serves as the lifting fixture for the fuel assemblies.

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2.1.2 Modifications to Support Plutonium Disposition Mission

Preliminary calculations indicate that, with the fuel consisting of weapons-grade plutonium in tails uranium, with erbia as burnable poison, both the System 80 and non-System 80 reactor cores can be loaded with 100% MOX fuel assemblies, and meet essential safety and shutdown parameters. Because the weapons-grade plutonium is so low in Pu^{240} and other higher Pu isotopes, it can be substituted, almost at an atom-for-atom rate, for U^{235} in the initial fuel mixture, without seriously changing the performance characteristics. This is not the case for reactor-grade plutonium, where the Pu^{240} content may run to 24% or more, and the higher loading of plutonium required to compensate for the additional neutron absorption, combined with the more negative temperature coefficient of reactivity, seriously reduce the control rod worth available for shutdown. The System 80 reactor was designed specifically to provide a higher rod worth in order to accommodate a full core of reactor-grade plutonium.

As a general rule, the higher the loading of weapons-grade plutonium, the greater is the need for additional rod worth. Thus the System 80 reactors are able to perform better (dispose of more plutonium in an economical way) than the non-System 80 reactors. In order to accelerate the mission, it may be desirable to increase the rod worth in the non-System 80 reactors, and even in the System 80 reactors, so as to increase the plutonium loading per core. Therefore the possibility of adding more control rods was investigated. This would involve modifying the vessel closure head to accept additional control rod drive nozzles, adding additional control and instrumentation, modifying the internals, as well as procuring additional rod drive mechanisms and control rods. For the non-System 80 reactors, the effects on primary systems of the additional decay heat load, the higher soluble boron requirements, and higher gamma heating loads on the internals, need to be addressed, and modifications made to the systems affected as needed. Of all the modifications that may be necessary, those to add more control rods appear to be the most difficult to accomplish and are potentially the most costly.

System 80 Plant Modifications

If higher plutonium loadings are contemplated in order to accelerate the mission, additional control rods and control rod drives may be needed for a MOX fueled core. The System 80 design was specifically developed to accommodate MOX fuel loadings up to and including the 100% MOX fueled core with reactor grade plutonium. Eight spare (unused) control element drive mechanism (CEDM) nozzles are provided in the RV top head. These nozzles will accommodate eight additional control element assemblies (CEA's). The locations of these spare CEDM nozzles are indicated in Figure 2.1.1-18. It should be noted that the reduced worth of individual control rods also has beneficial effects, as it results in a decrease in the worst stuck CEA worth, dropped CEA worth, and ejected CEA worth. This serves to increase safety margins in a number of design basis transients and accidents.

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The Palo Verde System 80 plants have used two of the eight spare CEDM nozzles to install heated juncture thermocouple (HJTC) instrumentation. If instrumentation is installed in a spare CEDM nozzle, that nozzle cannot be used for a CEA unless the instrumentation is relocated elsewhere in the RV. One solution is to install new instrument nozzles at the locations indicated in Figure 2.1.1-18. The instruments now located in CEDM nozzles can then be relocated to the new instrument nozzles making all eight spare CEDM nozzles available for CEA's.

Mid-Loop Water Level Monitoring System

Since the System 80 plants were designed, in reactor instrumentation requirements have been changed. Generic Letter 88-17 requires that water level instrumentation be incorporated into reactors so that the reactor vessel water level during mid-loop operations be properly accounted for and if the water level falls below acceptable levels actions necessary to preserve core integrity be available. The Palo Verde Nuclear Generating Station (PVNGS) currently off-loads its fuel during reduced inventory operations to meet Generic Letter 88-17 requirements. For the plutonium disposition mission, however, it may be advantageous, and could even be demanded by licensing review, to adopt a mid-loop water level monitoring system. ABB-CE has developed an HJTC based system for water level monitoring during these reduced inventory operation that occur during refueling. This system is called the Mid Loop Water Level Monitoring (MLWLM) System. A system is currently in use and approved by the Nuclear Regulatory (NRC) at San Onofre Nuclear Generating Station (SONGS).

The Palo Verde reactor currently has two of the eight spare CEDM locations occupied by Heated Junction Thermocouple Assemblies (HJTC). In order to incorporate the eight CEDMs into the spare locations, the HJTC probes and Mid Loop Water Level Monitoring instruments must be located at different locations. These locations would be below nozzle number 96 and above nozzle number 92 for the HJTCs and below nozzle number 91 and above nozzle number 95 for the MLWLMs. These new locations will require additional nozzles in the reactor vessel and additional guidance hardware in the reactor internals. In order to install the CEDMs in the locations currently occupied by the HJTCs, the hardware located on the CEDM pressure housing and in the reactor internals must be removed.

Removal of Existing HJTC Transition Hub Assemblies

The existing HJTC transition hub assemblies are installed on the CEDM nozzle locations 92 and 96. The HJTC transition hub assemblies must be removed to permit installation the CEDMs and CEAs in all of the spare locations. The transition hub assembly attaches to the CEDM nozzle by means of a threaded connection and an omega seal weld. The omega seal weld must be manually cut or an automatic cutter developed similar to that used for the upper pressure housing omega seal. After the omega seal weld is cut, the transition hub assembly may be unscrewed from the CEDM nozzles.

approximately one foot below the tip of the bullet nose weldment and the bullet nose weldment. Connect an appropriate lifting device to the tip of the bullet nose assembly, raise the refueling water to normal refueling level and slowly withdraw the HJTC probe. Withdraw the HJTC probe until the tip clears the UGS lift rig safety rail and immediately immerse back in the refueling water. Transport and tie off the HJTC probe in the UGS laydown area. The UGS can then be moved to its storage area. The guide system support structure (GSSS) must then be removed. The GSSS is bolted in eight (8) locations to the UGS support barrel. Crimp sleeves are installed on the these bolts

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to prevent them from loosening. These sleeves must be ground off prior to removing the bolts. The bolts must then be detensioned and captured to prevent them from falling into the refueling pool. The GSSS can then be lifted off the UGS and stored. It is anticipated that the GSSS will be radioactively hot and its storage location will be dependent on its radiation level. Storage in the core support barrel side of the refueling pool may be necessary. Work will have to be performed on the GSSS. Holes for the HJTC and MLWLM support hardware are required. In the event that this drilling must be performed underwater, EDM drilling would most likely be used. The HJTC reactor internals supporting structure is located by inserting its lower pins into the lower plate of the UGS support barrel. It is secured into the reactor internals by a weld to the CEA guide shroud in the UGS. This weld must be ground off in order to remove this supporting structure. Once the weld is cut, this support structure may be listed out and disposed. It will not be used as part of the new HJTC guidance structure. Except for the GSSS portion of the reactor internals, the UGS is now prepared to accept the CEAs.

Installation of new HJTC and MLWLM Reactor Internals Support Hardware

New HJTC and MLWLM supporting hardware will resemble that used on the YGN reactor and proposed for the System 80+ reactor. It consists of a tube that will be installed in the GSSS, and go through the lower plate of the UGS support barrel and the fuel alignment plate. Holes must be drilled into these three structures. The holes in the UGS support barrel lower plate will be drilled using EDM techniques under water due to the radiation level of these components. EDM machining also leaves no metal chips to be concerned with after machining. It is also anticipated that holes to install lateral supports for the HJTC and MLWLM guide tube will also be required. It is anticipated that the Guide tube will be installed after the GSSS is reinstalled and secured using a mechanical connection. The reactor internals components will most likely be too radioactively hot to permit welding. As the detailed design progresses, it may be possible to first install the guide tube into the GSSS and weld it in place. The installation into the UGS lower plate and the fuel alignment plate will most likely be a reasonably close fit between the guide tube and the hole to minimize flow vibration effects.

Installation of the HJTC Probes and MLWLM Instrument Assemblies

The HJTC probes and MLWLM instrument assemblies are inserted into the guide paths after installation of the reactor vessel head onto the reactor vessel. The seal rings are then installed onto the transition hub assembly. Install the blind hub over the HJTC probe until it rests on the seal ring. Install and tighten the clamps set. Install the HJTC probe seal and secure it with the drive nut that goes over the HJTC probe.

Revisions to Incorporate CEA Guidance Geometry in the Reactor Internals Guide Structure Support System

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In order to provide the same features of decoupling the extension shafts that the remainder of the GSSS locations provide, an insert must be installed in the locations where the HJTC were previously located. This insert must match the geometry of the other CEA locations in the GSSS.

Non-System 80 Plant Modifications

These plants were not specifically equipped for the plutonium burning mission, but

CEA's; while the System 80 plant has eight spare CEDM nozzles because it was designed for plutonium burning. The modifications for the 3410 MWt plant are discussed below.

As with the System 80 design, additional control rods and control rod drives may be needed for a MOX fueled core in a 3410 MWt plant. Figure 2.1.1-8 shows the layout of the CEDM nozzles in the RV top head for a 3410 MWt plant. There are 91 existing CEDM nozzles, all of which contain CEA's. There are 10 instrumentation nozzles around the top head outer edge. There are nine vacant locations in the RV top head penetration pattern which could accommodate nine additional CEDM nozzles. The vacant location which is 8.180" from the 90°-axis and 8.180" from the 0°-axis is not useful for a CEA because it is not symmetrical to the rest of the CEA pattern. The vent pipe penetration occupies the location symmetrical to this possible CEA location. Therefore, only the other eight vacant locations are useful for CEDM nozzles and CEA's. The installation of these eight new CEDM nozzles is discussed briefly in the next section.

CEDM Nozzle and Instrumentation Nozzle Installation Procedures

The procedure for adding nozzles is very similar to the procedure used in ABB's manufacturing facility at Chattanooga to install the CEDM nozzles in the newly fabricated closure heads. In general, the CEDM nozzles are shrink fitted into holes which are close tolerance in diameter, true position, and perpendicularity to the RV head mating surface. The nozzles are then J-groove welded to the RV head. This type of installation satisfies the ASME Code, Section III requirement of essentially no load on the J-weld and also assures that the CEA's will line up with the core. To accomplish the modification in the field, it is assumed that the RV head has been removed and installed in a fixture which allows full access to the outside and inside surfaces of the head. Local shielding will be required to allow personnel access to accomplish the modifications.

Installation of the CEDMs

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After cleaning the CEDM nozzles, the CEDM motor housing is screwed onto the CEDM nozzle assembly and tightened to the proper torque. The omega seal gap must be verified and the omega seal weld made. The upper pressure housing must be similar to installed onto the motor housing assembly. Hydrostatic testing if required may be performed after installation of these CEDM housings and the HJTC and MLWLM instrument flange assemblies.

The cooling shroud and the upper shroud are then installed on the coil stack assemblies. The conduit assembly is then installed on the upper shroud. The coil stack, upper shroud, and cooling shroud are then lowered over the upper pressure housing and motor assembly until they seat on the coil stack spacer. The RSPTs are then tested, installed in the upper shroud and their position adjusted until proper indication of the limits is achieved. During startup of the plant the CEDM upper pressure housings must be vented

2.1.3 Instrumentation & Control (I&C) Systems

With the addition of control element drive mechanisms comes the need to power and control additional Control Element Assemblies (CEA's). Further changes are or may be required in the plant control, protection, monitoring and sensor systems. The development of the required changes can be accomplished by straightforward analysis, design and qualification activities.

The scope of I&C system rework to be considered is dependent on the status of the individual plant(s) under consideration. For an operating plant, economic considerations dictate that I&C rework have minimal impact on power production. For plants such as WNP-3 which remain to be completed, conversion to a plutonium fuel cycle may provide an opportunity to update obsolescent I&C systems. The design of individual plant I&C systems is of small significance in the weighing of factors used for consideration of plutonium fuel cycle selection candidates. This report will not attempt a comprehensive discussion of differences between individual plant I&C systems.

2.1.3.1 Control Systems

2.1.3.1.1 Control Element Drive Mechanism Control System (CEDMCS)

If additional CEA's are required to obtain greater shutdown margin, for System 80 + this requires converting 13 installed part length CEA's to full length, full strength CEA's. In addition, CEA's and CEDM's must be installed at 8 spare penetration locations. Currently 2 of the 8 spare CEDM locations are occupied by instruments designed to detect the onset of Inadequate Core Cooling. This instrumentation was required as a result of the Three Mile Island "lessons learned" and the unused spare penetrations provided a convenient way to insert instrumentation to detect loss of

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saturation margin and loss of reactor vessel water inventory. The system uses a series of heated junction thermocouples to measure reactor vessel level above the fuel alignment plate. It will be necessary to relocate this instrumentation by drilling additional penetrations in the reactor vessel head and perhaps modifying the reactor internals. Table 2.1.3-1 summarizes the number of CEDM penetrations for the System 80+, System 80, and 3410 Mwt Class plants. Section 2.1.2 discusses the reactor vessel head modification process.

Some plants have a plan to use an additional 2 spare penetrations for midloop monitoring instrumentation. This instrumentation is used when the reactor vessel water level is lowered to allow installation of steam generator nozzle dams so that the steam generator may be drained for inspection. During this operation the water level in the region of the hot leg must be carefully controlled to avoid loss of NPSH at the shutdown cooling pumps. Midloop level monitoring instrumentation design is currently based on heated junction thermocouple probes inserted through 2 spare CEA nozzles. An alternate design will be needed if midloop monitoring is necessary for operational considerations.

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**TABLE 2.1.3-1
REACTOR VESSEL HEAD CEA PENETRATIONS**

	Number of Penetrations		
	System 80 +	System 80 (Palo Verde)	3410 Mwt Class (Waterford)
Installed Full Length CEAs	93	76	83
Installed Part Length CEAs	0	13	8
Spare Penetrations (Unoccupied)	4	6	--
Spare Penetrations (Instrumented)	4	2	--
Total Number of CEDM Nozzles	101	97	91*

*Nine (9) reactor vessel head locations available for additional penetrations.

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

2.1.3.1.2 System 80 Requirements

Currently, Palo Verde does not have midloop monitoring instrumentation. The Palo Verde units unload the core prior to reducing reactor coolant inventory. This lengthens refueling time and reduces capacity factor.

The CEDMCS for System 80 is sized to control 97 CEA's but has installed modules for only the 89 CEA's that are actually used. Additional modules and wiring changes will be necessary to drive 97 CEA's. Problems may be encountered with parts availability due to obsolescence. Other minor required CEDMCS modifications are:

- Interlock changes to convert from part length CEA's to full length CEA's
- CEDMCS Operator's Module reconfiguration
- Core Mimic reconfiguration

2.1.3.1.3 CEDM Motor-Generator Sets

These sets provide uninterrupted power to the CEDMCS which in turn controls power to individual drive coils. Maximum power demand is largely a function of the number of CEA's needing to be moved simultaneously. Although the number of CEA's being controlled has increased, the number of CEA's needing to be moved simultaneously is not expected to increase. Thus, the capacity of the MG sets is expected to be adequate. MG set output breakers actuated by the Supplemental Protection System are sized to accommodate an increased number of CEA's.

2.1.3.1.4 Reactor Regulating System

Retuning this system may be needed based on changed reactor response and revised T_{ave} vs Power program. No hardware changes should be necessary.

2.1.3.1.5 Reactor Power Cutback System

The Reactor Power Cutback System acts to help prevent a plant trip during fast load reductions by dropping selected CEA's. This system will need to be reanalyzed and configured for additional groups of CEA's.

2.1.3.1.6 Other Control Systems

It is not expected that other control systems (with the possible exception of provision for soluble boron control) will need to be modified based on plant operating characteristics. Security considerations may require addition or deletion of remote

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controls. The need for this has not been evaluated. Extensive changes to plant security systems will be needed. These have not been evaluated.

2.1.3.2 Protection System

The Protection System consists of the Reactor Protection System which trips the CEA's and the Engineered Safety Features Actuation System. Setpoint changes are expected based on a revised safety analysis. These can be readily adjusted. Additional trips (up to 12 bistable locations per channel) can be added if analysis determines this to be necessary. CEA position is used as an input to the Core Protection Calculators (CPC's). The CEA Position Isolation Assembly is designed to accommodate 97 CEA's on System 80 plants. Algorithm changes within the CPC's will be necessary to consider the full complement of CEA's and changes in core characteristics.

For System 80, the Reactor Trip Switchgear is sized to interrupt the current to 97 CEA's.

2.1.3.3 Monitoring Systems

2.1.3.3.1 Boronometer

The boronometer provides an online indication of the boron concentration in a sample of the reactor coolant.

The plutonium fuel cycle requires a significantly larger (factors of 2 or 3) amount of soluble B^{10} in the core for initial operation than does the normal low enrichment UO_2 . The increased B^{10} can either be achieved by increasing the concentration of the soluble boron or by utilizing boron enriched in the B^{10} isotope.

Use of non-enriched boron could require the use of extensive heat tracing and significant modification of the boron handling equipment.

Rather than accepting these operational problems for the MOX fuel burner, it has been proposed that boron enriched in the B^{10} isotope be used instead of natural boron. By utilizing soluble boron of the appropriate enrichment, the soluble boron concentration can be quite similar to that used for UO_2 cores, thus minimizing any changes to equipment or procedures.

Use of the enriched boron will place a significantly higher importance on the boronometer as the only quick way that the relative B^{10} content can be determined. Further, since enriched B^{10} is more costly than natural boron, considerable emphasis will be placed on re-cycling the boron for subsequent re-introduction to the reactor.

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Re-cycling of boron from cycle to cycle is done in the United States but usually there is a significant loss of boron during each cycle since the cost of natural boron is not that high as to require careful preservation of inventories. In Europe, boron re-cycling has been more assiduously pursued. This can result in the situation where neutron depletion of the B^{10} must be considered and may require deliberate replacement of the enriched boron on a pre-determined schedule.

The fact that the effectiveness of the soluble boron can no longer be determined by means of standard chemical techniques may require that redundant boronometers be installed and that a procedure for assuring their calibration be incorporated. Portable or remote boronometers may also be required to allow testing of boron prior to its introduction to the primary fluid system. Whatever boron control approach is used, the range and response time of the existing boronometer will need to be reviewed.

2.1.3.3.2 Radiation Monitoring System

Increased tritium concentration in the reactor coolant and auxiliary systems may require additional monitors to assure that release limits are not exceeded.

2.1.3.3.3 Plant Computer

The Core Operating Limit Supervisory System program assists the operator in maintaining the plant within the core related Limiting Conditions for Operation. Changes to software will be needed to accommodate the following factors:

- Spectral induced shift in in-core and ex-core detector response
- Changes in rod shadowing
- Increased number of CEA's
- Changes in core characteristics
- Limit changes

Additional changes will be needed for the Reactor Power Cutback CEA selection algorithm. Displays associated with CEA positions and the calculation of CEA related interlocks will also need to be modified.

A separate secure computer system will be needed for Material Control and Accountability (MC&A). A link from the plant computer to the MC&A computer will be needed to track core burnup.

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2.1.3.3.4 Sensors

Sensor qualification will need to be reviewed to assure operation in a normal and post accident environment is within the sensor qualification envelope. The higher gamma flux at the ex-core startup detector location during normal startups has the potential to reduce the neutron sensitivity of these detectors. Post Accident Monitoring Sensor qualification needs to be compared with increased in containment radiation levels under accident conditions.

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TABLE 2.1.1-1
COMPARISON OF REACTOR CHARACTERISTICS

	3410 Mwt	SYSTEM 80 3817 Mwt
<u>Nuclear Design Data</u>		
Structural Characteristics		
Core Diameter, in. (Equivalent)	136.0	143.6
Core Height, in. (Active Fuel)	150.0	150.0
H/U, Unlimited Assembly (Hot)	4.34	4.23
Number of Fuel Assemblies	217	241
UO ₂ Fuel Rod Locations Per Assembly	236 (*)	236 (*)
Performance Characteristics		
Loading Technique	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone
Fuel Discharge Burnup Mwd/MTU		
Average First Cycle	12,373	13,740
First Core Average	21,700	24,144
Fuel Enrichments W/O U-235		
Region 1	1.27	1.83
Region 2	2.38	2.49
Region 3	2.33	2.95
Region 4		
Control Characteristics		
Critical Boron Concentrations, PPM (beginning of life, Rods out)		
Cold, Zero Power, Clean	899	902
Hot, Zero Power, Clean	832	882
Hot, Equilibrium Xe, Full Power	719	516
Hot, Full Power, Clean	952	764
Control Element Assemblies		
Material (Full/Part Length)	B ₄ C/Ag-In-Cd	B ₄ C/B ₄ C-Inconel
Number of Control Assemblies (Full/Part Length/Part Strength)	83/8/0	76/13/0 (*)

* In the first core, some UO₂ rods may be replaced by burnable absorber rods.

Note: Temperatures are rounded to the nearest degree throughout table.

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

TABLE 2.1.1-1

COMPARISON OF REACTOR CHARACTERISTICS

	3410 Mwt	SYSTEM 80 3817 Mwt
<u>Nuclear Design Data</u> (Cont'd)		
Number of Absorber Rods Per CEA (or RCC) Assembly	4,5/5	4 or 12
Control Element Assemblies		
Material (Full/Part Length)	B ₄ C/Ag-In-Cd	B ₄ C/B ₄ C-Inconel
Number of Control Assemblies (Full/Part Length/Part Strength)	83/8/0	76/13/0 (*)
Number of Absorber Rods Per CEA (or RCC) Assembly	4,5/5	4 or 12
Total Rod Worth (Hot) %	11.35	16.76
Kinetic Characteristics Range Over First Cycle		
Moderator Temperature Coefficient $\Delta\rho/^\circ\text{F}$ (Hot, Full Power) BOL/EOL	-0.5 x 10 ⁻⁴ / -2.3 x 10 ⁻⁴	-0.7 x 10 ⁻⁴ / -2.5 x 10 ⁻⁴
Moderator Pressure Coefficient (BOL) $\Delta\rho/\text{psi}$ (Hot, Operating)	+0.9 x 10 ⁻⁶	+0.69 x 10 ⁻⁵
Moderator Void Coefficient (BOL)	-0.36 x 10 ⁻³	-0.24 x 10 ⁻³
Doppler Coefficient $\Delta\rho/^\circ\text{F}$ (Hot, Operating Range)	-1.13 x 10 ⁻⁵ to -1.87 x 10 ⁻⁵	-1.18 x 10 ⁻⁵ to -1.66 x 10 ⁻⁵
Hydraulic and Thermal Design Parameters		
Total Core Heat Output, Mwt	3390	3800
Total Core Heat Output, Btu/hr.	1157 x 10 ⁷	1297 x 10 ⁷
Heat Generated in Fuel, %	97.4	97.4

* Locations are provided for 8 additional CEAs.

** Inconel part strength CEAs in System 80 +

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TABLE 2.1.1-1
COMPARISON OF REACTOR CHARACTERISTICS

	<u>3410 Mwt</u>	<u>SYSTEM 80 3817 Mwt</u>
<u>Nuclear Design Data</u> (Cont'd)		
System Pressure, Nominal, psia	2250	2250
System Pressure, Minimum Steady State, psia	2200	2200
Hot Channel Factors, Overall		
Heat Flux, F_q	2.34	2.34
Enthalpy Rise, F_H	1.55	1.55
DNB Ratio at Nominal Conditions	2.07 (CE-1) *	1.98 (CE-1) *
Coolant Flow		
Total Flow Rate, lb/hr.	148.0×10^6	164.0×10^6
Effective Flow Rate for Heat Transfer, lb/hr.	144.2×10^6	159.1×10^6
Effective Flow Area for Heat Transfer, ft ²	54.7	60.8
Average Velocity Along Fuel Rods, ft/sec.	16.5	16.8
Average Mass Velocity, lb/hr-ft ²	2.64×10^6	2.62×10^6
Reactor Temperatures, °F		
Nominal Inlet	553	565
Average Rise in Vessel	58	56
Average Rise in Core, °F	59	58
Average Temperature in Core, °F	582.5	594
Average Temperature in Vessel, °F	582	593
Hot Channel Outlet, °F	642	646
Average Film Coefficient, Btu/hr-ft ² -°F	6270	6290
Heat Transfer at 100% Power		
Active Heat Transfer Area, ft ²	61,860	68,320
Average Heat Flux, Btu/hr-ft ²	182,100	184,800
Maximum Heat Flux, Btu/hr-ft ²	426,300	432,700

* Minimum DNBR at nominal conditions.

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TABLE 2.1.1-1
COMPARISON OF REACTOR CHARACTERISTICS

	<u>3410 Mwt</u>	<u>SYSTEM 80 3817 Mwt</u>
<u>Nuclear Design Data (Cont'd)</u>		
Average Thermal Output, kW/ft	5.33	5.41
Maximum Thermal Output, kW/ft	12.5	12.7
Maximum Clad Surface Temperature at Nominal Pressure, °F	657	657
Fuel Centerline Temperature, °F Maximum at 100% Power	3,180	3,205
Engineering Heat Flux Factor	1.03	1.03
<u>Core Mechanical Design Parameters</u>		
Fuel Assemblies		
Rod Bundle Arrangement	16 x 16	16 x 16
Design	CEA	CEA
Rod Pitch, in.	0.506	0.506
Cross Section Dimensions, in.	7.972 x 7.972	7.972 x 7.972
Fuel Weight (as UO ₂), lb.	223,900	257,245
Number of Grids Per Assembly	12	11
Fuel Rods		
Number of Locations	49,500*	56,876*
Outside Diameter, in.	0.322	0.382
Diametral Gap, in.	0.007	0.007
Clad Thickness, in.	0.025	0.025
Clad Material	Zircaloy-4	Zircaloy-4
Fuel Pellets		
Material	UO ₂ Sintered	UO ₂ Sintered
Diameter, in.	0.325	0.325
Length, in.	0.390	0.390
Control Assemblies		
Cladding Material	NiCrFe Alloy	NiCrFe Alloy
Clad Thickness, in.	0.035	0.035
Core Structure		
Core Barrel ID/OD, in.	148/153	157/162

* Some of the rod locations are occupied by burnable absorber rods.

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

TABLE 2.1.1-1
COMPARISON OF REACTOR CHARACTERISTICS

	<u>3410 Mwt</u>	<u>SYSTEM 80 3817 Mwt</u>
<u>Reactor Coolant System Code Requirements</u>		
Component		
Reactor Vessel	ASME III, Class 1	ASME III Class 1
Steam Generator		
Tube Side	ASME III, Class 1	ASME III, Class 1
Shell Side	ASME III, Class 2	ASME III, Class 2
Pressurizer	ASME III, Class 1	ASME III, Class 1
Pressurizer Safety Valves	ASME III, Class 1	ASME III, Class 1
Reactor Coolant Piping	ASME III, Class 1	ASME III, Class 1
<u>Principal Design Parameters of the Reactor Vessel</u>		
Material	Low alloy steel clad with austen- itic SS.	Low alloy steel clad with austen- itic SS.
Design Pressure, psia	2500	2500
Design Temperature, °F	650	650
Operating Pressure, psia	2250	2250
Inside Diameter at Shell, in.	172	182-1/4
Outside Diameter Across Nozzles, in.	253	267-1/4
Overall Height of Vessel and Enclosure Head, ft-in. to top	43' 6-1/2"	50' 1-5/8" (including bottom instrumentation nozzles)
Minimum Clad Thickness, in.	1/8	1/8

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

TABLE 2.1.1-1
COMPARISON OF REACTOR CHARACTERISTICS

	<u>3410 Mwt</u>	<u>SYSTEM 80 3817 Mwt</u>
<u>Principal Design Parameters of the Reactor Coolant Piping</u>		
Material	Carbon steel internally clad with stainless steel	Carbon steel internally clad with stainless steel
Hot Leg - I.D., in.	42	42
Cold Leg - I.D., in.	30	30
Between Pump and Steam Generator - I.D., in.	30	30
Design Pressure, psia	2500	2500
<u>Principal Design Parameters of the Reactor Coolant System</u>		
Operating Pressure, psia	2250	2250
Reactor Inlet Temperature, °F	253	565
Reactor Outlet Temperature, °F	611	621
Number of Loops	2	2
Design Pressure, psia	2500	2500
Design Temperature, °F	650	650
Hydrostatic Test Pressure (cold), psia	3125	3125
Total Coolant Volume, ft ³	10,300	13,125
Total Reactor Flow, gal/min. *	396,000	445,600
<u>Principal Design Parameters of the Reactor Coolant Pumps</u>		
Number of Units	4	4
Type	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge
Design Pressure, psia	2500	2500

* Design minimum.

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

TABLE 2.1.1-1
COMPARISON OF REACTOR CHARACTERISTICS

	<u>3410 Mwt</u>	<u>SYSTEM 80 3817 Mwt</u>
<u>Principal Design Parameters of the Reactor Coolant Pumps (Cont'd)</u>		
Design Temperature, °F	650	650
Operating Pressure, nominal psia	2250	2250
Suction Temperature, °F	553	565
Design Capacity, gal/min.	99,000	111,400
Design Head, ft	310	365
Hydrostatic Test Pressure, (cold) psia	3125	3125
Motor Type	AC Induction Single Speed	AC Induction Single Speed
Motor Rating, hp. (cold)	9,700	12,000
<u>Principal Design Parameters of the Steam Generators</u>		
Number of Units	2	2
Type	Vertical U-tube with integral moisture sep- arator	Vertical U-tube with integral moisture sep- arator and economizer
Tube Material	SB-163 NiCrFe alloy Primary side - low alloy steel clad with aus- tenitic stain- less steel Secondary side - carbon steel	SB-163 NiCrFe alloy Primary side - low alloy steel clad with aus- tenitic stain- less steel Secondary side - carbon steel
Tube Side Design Pressure, psia	2500	2500
Tube Side Design Temperature, °F	650	650
Tube Side Design Flow, lb/hr per steam generator	74 x 10 ⁶	82 x 10 ⁶
Shell Side Design Pressure, psia	1100	1270
Shell Side Design Temperature, °F	560	575
Operating Pressure, Tube Side, Nominal, psia	2250	2250

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

**TABLE 2.1.1-1
COMPARISON OF REACTOR CHARACTERISTICS**

	<u>3410 Mwt</u>	<u>SYSTEM 80 3817 Mwt</u>
<u>Principal Design Parameters of the Steam Generators</u> (Cont'd)		
Operating Pressure, Shell Side, Maximum, psia	1000	1170
Maximum Moisture at Outlet at Full Load, %	0.20	0.25
Hydrostatic Test Pressure, Tube Side (cold), psia	3125	3125
Steam Pressure, psia, at Full Power	900	1070
Steam Temperature, °F at Full Power	532	553
Steam Flow, at Full Power, lb/hr per steam generator	7.565×10^6	8.590×10^6
<u>Containment System Parameters</u>		
Type	Steel-lined prestressed post tensional concrete cylinder, curved dome roof.	(*)
Design Parameters - Containment		
Inside Diameter, ft	150	(*)
Height, ft	240	
Free Volume, ft ³	2.335×10^6	
Reference Incident		
Pressure, psia	75	
Steel Thickness, in. (approx.)		
Vertical Wall	1/4	
Hemispherical Head	1/4	
Concrete Thickness, ft		
Vertical Wall	4-1/3	(*)
Dome	3-3/4	

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

TABLE 2.1.1-1
COMPARISON OF REACTOR CHARACTERISTICS

	<u>3410 Mwt</u>	<u>SYSTEM 80 3817 Mwt</u>
<u>Containment System Parameters (Cont'd)</u>		
Containment Leak Prevention and Mitigation Systems	Leak tight penetration and continuous steel liner. Automatic isolation where required. Continuous steel liner exhaust from penetration rooms directed to vent stack.	(*)
<u>Engineered Safety Features</u>		
Safety Injection System		
No. of High Head Pumps	3	2
No. of Low Head Pumps	2	2
Safety Injection Tanks, No.	4	4
Emergency Power		
Standby Generator Units	4 total for 2 units	(*)
<u>Instrumentation and Control Systems</u>		
Reactor Protective System	Sec. 7.2	Sec. 7.2
Initiating Reactor Trip		
Number of Manual Switches	2 Sets of 2 each	2 Sets of 2 each
Automatic Initiation	2 of 4 Logic	4 channels provided,
Parameter Channels/Logic	for each trip	coincidence of 2 required for trip

* See Site-Specific SAR.

** Nominal

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

TABLE 2.1.1-1

COMPARISON OF REACTOR CHARACTERISTICS

	<u>3410 Mwt</u>	<u>SYSTEM 80 3817 Mwt</u>
ESFAS		
Initiating ESFAS		
Number of Manual Switches	2 Sets of 2 each	2 Sets of 2 each

* See Site-Specific SAR.

*** 2 Class 1E and 1 Non-Class 1E

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

**TABLE 2.1.1-2
COMPARISON OF REACTOR VESSEL CHARACTERISTICS**

SUBJECT	3410 MWt PLANT	SYSTEM 80
Applicable ASME Code	Section III, Class 1	Section III, Class 1
ASME Code Addenda Earlier or Later than the Summer of 1972 ?	Earlier	Later
Nonductile (Fracture) Design Evaluation Performed ?	No	Yes
Design Pressure, psig	2485	2485
Normal Operating Pressure, psig	2235	2235
Design Temperature, °F	650	650
Normal Operating Outlet Temperature, °F	611	621.2
Normal Operating Inlet Temperature, °F	553	564.5
Normal Reactor Coolant Flow Rate, lb/hr	148×10^6	164×10^6
Design Life, years	40	40
Type of Vessel Support	Sliding Base Plates	Columns
Inside Diameter at Shell Beltline, in.	172	182.25
Outside Diameter Across Nozzles, in.	252.53	271
Height of Reactor Vessel Assembly, ft.	43.41	48.65
Minimum Clad Thickness, in.	.125	.125

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

2.2 Infrastructure

2.2 Balance of Plant and Infrastructure

Details of the changes to the existing plant infrastructure in order to carry out the plutonium disposition mission in an existing plant would depend on the plant chosen and the mission involved. In general, the considerations could concern the following aspects:

- **Power Level.** For any of the reactors selected in this study, preliminary calculations of the safety analysis parameters for the initial fuel management plans have indicated that the plant power level need not be derated in order to accommodate the MOX fuel. When detailed calculations are performed, and, if required, new licensing regulations applied, the ability to stay at full licensed power will be reexamined in each case.
- **Fuel Management.** The initial fuel management plans for each existing reactor are discussed in Section 2.6 of this report. The cycle length has been maintained for each reactor, according to the present practice, but the batch size has been increased to allow a higher throughput of plutonium per year. Whereas the fuel management for the existing reactors with uranium fuel is based on economics and the seasonal fluctuations in demand, for the MOX fuel, there are other factors relative to the mission which have to be considered, and changes to the fuel management are to be anticipated.
- **Fuel Storage Pool.** The existing fuel storage facilities and the possible scenarios with the introduction of spent MOX fuel are discussed in Section 2.5. The fuel storage pools all contain spent uranium fuel belonging to the utility operator. Their capacity to hold MOX fuel, which because of its residual reactivity upon discharge may require checkerboard loading or not possible. Such an alternative is an additional spent fuel storage pool facility to be built on site.
- **Safeguards and Security.** The requirements for additional safeguards and security are addressed in Section 7 of this report. The additional safeguards are mostly applied to the transportation of the fresh MOX fuel to the site. After receipt, the fresh fuel is stored under water, where it would be difficult to retrieve and divert to other uses. Additional requirements for the facility may include additional guards, and a Perimeter Intrusion Detection Alarm System (PIDAS) fence, if the equivalent is not available already. Much of the need for modification of the safeguards and security system can only be

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

ascertained after a detailed study of the existing facility and the latest requirements.

- Operations. The mode of operation of the reactor with MOX fuel need not change. Operator training and retraining will be required to acquaint the operating staff with the additional safeguards and security requirements, and the ES&H requirements (which should be minor) due to the MOX fuel. Additionally, there will likely be some changes in Technical Specifications which will require extra training.

2.3 Reactor Safety Systems

This section describes the System 80 principal safety systems:

- Shutdown Cooling System
- Auxiliary (Emergency) Feedwater System
- Safety Injection System
- Containment Spray System

The System 80 CESSAR FSAR describes the shutdown cooling system, the safety injection system, and the containment spray system. The description of the auxiliary feedwater system is left the safety analysis report of each specific plant. The auxiliary feedwater system of the Palo Verde Nuclear Generating Station (PVNGS), an operating version of a System 80, is described as representative of System 80.

Shutdown Cooling System (SCS)

The SCS is used in conjunction with the Main Steam and Main or Auxiliary Feedwater Systems to reduce the temperature of the Reactor Coolant System (RCS) in post shutdown periods from normal operating temperature to the refueling temperature. The initial phase of the cooldown is accomplished by heat rejection from the steam generators (SG) to the condenser or atmosphere. After the reactor coolant temperature and pressure have been reduced to approximately 350°F and 400 psia, the SCS is put into operation to reduce the reactor coolant temperature to the refueling temperature and to maintain this temperature during refueling.

The Shutdown Cooling Heat Exchangers (SDCHX) are also used during the recirculation mode following a Loss-Of-Coolant Accident (LOCA) or Main Steam Line Break (MSLB) for containment spray purposes.

The SCS is used in addition to the SG atmospheric steam release capability and the emergency (auxiliary) feedwater system to cooldown the RCS following a small break LOCA. The SCS would also be used subsequent to steam and feedline breaks and steam generator tube ruptures; and it is used to maintain flow through the core during plant startup.

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

The low pressure safety injection (LPSI) pumps are used as part of the SCS. During shutdown cooling, these pumps take suction from the reactor hot leg pipes and discharge through the shutdown cooling heat exchangers. The flow is then returned to the RCS through the LPSI header to the four cold legs. One LPSI pump is aligned to each shutdown cooling heat exchanger. At the start of shutdown cooling, both of the LPSI pumps are in service. When the RCS temperature is below 200°F, the containment spray pumps may be realigned and started to provide additional flow through the heat exchangers. The flow paths are illustrated in Figures 2.3-1 and 2.3-2.

The shutdown cooling heat exchangers are used to remove decay, sensible and safeguards pump heat during cooldown, and decay and pump heat during cold shutdown. The units are sized to maintain a refueling water temperature of 135°F with the design component cooling water temperature 105°F at 27½ hours after shutdown following an assumed reactor core average burnup of two years. A conservative fouling resistance is assumed, resulting in an additional area margin for the heat exchangers.

Prior to plant heatup above 200°F, the SCS heat exchangers are bypassed to maintain flow through the core without the heat removal effects of the heat exchangers. Flow can be initiated to the heat exchangers, if necessary, to control the heat-up rate. When the reactor coolant pump can be run and prior to reaching 350°F or 400 psia, the LPSI pumps are stopped, and the shutdown cooling heat exchangers are aligned for their containment spray function.

The shutdown cooling heat exchangers may be used to supplement the spent fuel pool cooling heat exchangers when more than one-third of a spent core is stored in the spent fuel pool. Normally, this would be done during refueling when both shutdown cooling heat exchangers are no longer needed to maintain reactor coolant at the refueling temperature. The SCS would be aligned with one heat exchanger train lined up to the spent fuel pool cooling system and the other shutdown cooling heat exchanger train lined up for shutdown cooling of the RCS.

Details of the SCS design and performance are provided in the System 80 CESSAR FSAR, Section 5.4.7.

Auxiliary Feedwater System (AFS)

The AFS is designed to provide steam generator feedwater during startup, hot standby, normal shutdown, and emergency conditions. It provides feedwater for the removal of decay heat from the RCS following reactor shutdown from any power level until such time as cooling by the shutdown cooling system is initiated. The AFS is designed to maintain water level in the steam generators under the following operating modes and accident conditions:

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1. Reactor coolant system cooldown at a maximum rate of 75°F per hour from hot standby to a temperature of 350°F with a loss of offsite power and normal onsite power.
2. Hot standby for 8 hours with a loss of offsite power and normal onsite power.
3. Reactor coolant system cooldown using the intact steam generator following a main steam line break or main feedwater line break inside the containment with a loss of offsite power and normal onsite power.

The AFS consists of one Seismic Category I, motor-driven AFS pump; one Seismic Category I, steam turbine-driven AFS pump; and one non-Seismic Category I, motor-driven AFS pump, associated piping, controls, and instrumentation.

The primary source of auxiliary feedwater is the condensate storage tank. A minimum capacity of 195,000 gallons provides an orderly RCS cooldown to the shutdown cooling initiation conditions. An additional 105,000 gallons of condensate storage capacity provides sufficient feedwater to maintain the plant at hot standby for 8 hours.

For emergency operation, normal flow is from the condensate tank to either the Seismic Category I, motor-driven AFS pump or to the steam turbine-driven, Seismic Category I AFS pump which are located in the main steam support structure. An alternate supply of water is provided by cross-connections to the reactor makeup tank.

A minimum flow recirculation system is provided on each pump discharge with recirculation to the condensate tank. Each pump can supply either steam generator with feedwater.

Condensate recirculation lines are provided downstream of the control stations to allow for system testing up to the feedwater isolation valves. One auxiliary feedwater path to the steam generators is provided for the non-Seismic Category I, motor-driven auxiliary feedwater pump through the feedwater header, with manual operation of feedwater valves possible during emergency operation.

The two Seismic Category I auxiliary feedwater pumps only provide flow to the downcomer feedwater nozzles on each steam generator. Either Seismic Category I auxiliary feedwater pump can supply the necessary feedwater for reactor decay heat removal and reactor cooldown to 350°F, at which point the shutdown cooling system is placed in operation.

Safety Injection System (SIS)

The SIS is the emergency core cooling system (ECCS) and is designed to provide core cooling in the unlikely event of a Loss-of-Coolant Accident (LOCA). The SIS prevents

high-pressure safety injection pumps ensure that the injected flow is sufficient to meet the SIS design criteria. The high-pressure safety injection pumps are also used during the recirculation mode to maintain a borated water cover over the core for extended

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periods of time following a Loss Of Coolant Accident. For long-term core cooling, the HPSI pumps are manually realigned for simultaneous hot and cold leg injection. This insures flushing and ultimate subcooling of the core independent of break location. For small breaks, the HPSI pumps continue injecting into the RCS to provide makeup for spillage out the break while a normal cooldown is implemented.

During normal operation the high pressure safety injection pumps are isolated from the RCS by motor operated valves. During safety injection the HPSI pumps deliver water from the refueling water storage tank to the RCS via the cold leg safety injection nozzles whenever RCS pressure falls below pump shutoff head. During the recirculation mode of operation, the pumps take suction from the containment sump.

The high-pressure safety injection pumps are sized such that one HPSI pump (after consideration of spillage directly out the break) will supply adequate water to the core to match decay heat boiloff rates soon enough to minimize core uncover and allow small break LOCA's to meet the performance criteria of 10CFR50.46. The effectiveness of the pump during a steam line break is also analyzed to assure that the pumps are adequately sized.

The pumps are provided with minimum flow protection to prevent damage resulting from operation against a close discharge. Also, individual HPSI pump ultrasonic flow meters provide low flow alarming.

The design temperature is based on the saturation temperature of the reactor coolant at the containment design pressure plus a design tolerance. The design pressure for the high pressure pumps is based on the shutoff head plus maximum containment pressure plus a design tolerance.

The four safety injection tanks (SIT) discharge their contents to the Reactor Coolant System following depressurization as a result of a Loss-of-Coolant Accident. Each tank is piped into a cold leg of the Reactor Coolant System (RCS) via a safety injection nozzle located on the RCS piping near the reactor vessel inlet. During normal plant operation each safety injection tank is isolated from the Reactor Coolant System by two check valves in series. The safety injection tanks automatically discharge into the RCS if RCS pressure decreases below safety injection tank pressure during reactor operation.

The tank gas/water fractions, gas pressure, and outlet pipe size are selected to allow three of the four tanks to re-cover the core before significant clad melting or zirconium-water reaction can occur following a LOCA. The volume of water in the tanks is conservatively calculated assuming that all water injection prior to the end of the RCS blowdown is lost.

The tanks contain borated water at the minimum required boron concentration for refueling and are pressurized with nitrogen at a nominal pressure of 610 psig.

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Details of the SIS design and performance are provided in System 80 CESSAR FSAR Section 6.3.

Containment Spray System (CSS)

The Containment Spray System uses the refueling water tank of the Chemical and Volume Control System, the containment sump, two containment spray pumps, two shutdown cooling heat exchangers, two independent spray headers, and associated valves, piping, and instrumentation. The system is shown on Figure 2.3-4. The CSS provides cooling sprays of borated water from the upper regions of the containment to reduce containment pressure and temperature during a major breach of the pressure boundary of either the Reactor Coolant or Main Steam and Feedwater Systems inside containment. The spray flow is provided by the containment spray pumps, which take suction from the refueling water tank or the containment sump. The pumps discharge through the shutdown cooling heat exchangers and the spray header isolation valves to their respective spray nozzle headers, then into the containment atmosphere. The spray headers are located in the upper part of the containment building to allow the falling spray droplets time to approach thermal equilibrium with the steam-air atmosphere. Condensation of the steam by the falling spray results in reduction in containment pressure and temperature.

The two containment spray pumps are vertical, single stage, centrifugal pumps driven by induction motors. The pump motors have the capability of starting and accelerating the driven equipment, under load, to design running speed within 5 seconds.

Minimum flow orifices are installed in lines running from the pump discharge to the refueling water tank. These paths ensure that the pumps are not deadheaded if they are inadvertently run against a closed system.

The pump shaft seal materials are designed to operate at a temperature of 400°F, although the maximum temperature to which they will be exposed following an accident will not exceed 350°F. Administrative procedures assure that the containment spray pump will not be exposed to temperatures in excess of 200°F, or pressures in excess of 350 psia during shutdown cooling operation.

A flow restrictor is provided to minimize the loss of fluid in the event of a gross seal failure. Vent and drain connections are provided. An internal drain to the pump section is provided within the pumps casing. Complete draining and flushing of the pump can be achieved prior to maintenance, thus reducing radiation levels.

The containment spray nozzles are attached to and become part of the spray headers located in the upper portion of the containment building. The nozzles disperse the spray solution throughout the containment in the form of droplets to increase the heat transfer surface. The spray nozzles and headers are oriented to ensure maximum

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

effective coverage of the containment volume. The spray nozzles are of the nonclogging type, designed to pass 5/6 inch particles.

The CSS pumps can be realigned to provide supplementary reactor shutdown cooling flow. This application is illustrated on Figure 2.3-2 and briefly discussed with the shutdown cooling system. The CSS is described in the System 80 CESSAR FSAR Appendix to Section 6.

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2.4 MOX Fuel Facility (MF²)

The technical description of the MOX Fuel Facility for the case where existing reactors are employed is similar to that reported in the Evolutionary Reactor Report, Section 2.4.

The following description applies to the receipt and storage of new UO_2 fuel assemblies. It is given for background and to compare with the provisions for MOX fuel receipt and storage, which is given later in this section.

New fuel is stored in the fuel building in rack assemblies in the new fuel storage area. The rack assemblies are made up of individual racks. These racks maintain minimum edge spacing between fuel assemblies. The racks are located in two concrete storage cavities that are utilized for new fuel storage. Each cavity is approximately 8 feet by 23 feet and contains 45 fuel assemblies. Three racks are installed in each cavity forming a 3 by 15 foot array of fuel assemblies. Criticality safety margins are maintained by limiting the size of the array to 90 fuel assemblies, providing adequate separation of the fuel assemblies in the array and defining the array configuration. The current rack array is designed for enrichments of up to 4 wt% U^{235} . If enrichments higher than 4 wt% are planned for fuel loadings, new criticality safety analyses and/or specific storage arrangements will be provided as needed. Storage is provided for at least one-third of a core of new fuel.

2.5.1.2 Spent Fuel

Spent fuel is stored in the fuel building in rack assemblies in the spent fuel storage pool. The spent fuel pool racks are made up of individual modules. A module is an array of storage cells. The storage racks are comprised of 17 modules: twelve 8 by 9, four 8 by 12 arrays and one 9 by 9 array. These racks have the capability to store fuel in three modes a checkerboard pattern, a borated fuel storage mode and a mixed fuel storage mode.

2.5.1.2.1 Checkerboard Fuel Storage Mode

A minimum edge-to-edge spacing between fuel assemblies is maintained by storing the fuel in a checkerboard pattern while using inserts to properly locate a fuel assembly within a storage cavity. In addition, fuel blocking devices are used in rack cells that are not within the acceptable checkerboard storage pattern. Storage for up to 665 new or spent fuel assemblies can be provided in this mode with the pool filled with water. Up to 242 new fuel assemblies can be stored dry in this mode.

2.5.1.2.2 Borated Fuel Storage Mode

A minimum edge-to-edge spacing between fuel assemblies is maintained by storing fuel in rack locations that have been lined with a neutron poison. The fuel assembly

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spacings are the minimum values after allowances are made for fabrication tolerances and predicted deflections resulting from a Safe Shutdown Earthquake (SSE). Storage for up to 1239 new or spent fuel assemblies can be provided in this mode.

2.5.1.2.3 Mixed Mode Fuel Storage

This mode uses variable number of L-inserts and cell blocking devices in conjunction with corresponding numbers of neutron poison inserts provide storage for an appropriate number of fuel assemblies. In this mode, individual regions of the racks are either in the checkerboard mode or the borated mode. All cavities in a checkerboard region immediately adjacent to a borated region must be empty and have cell blocking devices in place. This requirement assures that the criticality criterion have been met.

2.5.1.3 Description of the Containment Fuel storage Racks

The four-cavity containment fuel storage rack is designed as an intermediate storage location for fuel bundles during a refueling. New fuel may be stored before being moved into the core, partially spent fuel may be moved out of the core and stored temporarily to provide spaces for fuel shuffling, or spent fuel may be stored before being sent to the spent fuel pool.

2.5.1.4 Current Palo Verde Fuel Receipt Process

The following is the normal receipt process at Palo Verde. Current Palo Verde fuel arrives in individual canisters containing two fuel assemblies. The canisters are transported to the plant by a truck and off-loaded onto the fuel building floor after they have been cleaned, surveyed and inspected. The canister lid is then removed to gain access to the strongback used for uprighting the fuel assemblies. The shipping brackets are removed and the container strongback slinged and uprighted using the overhead crane. The strongback is then locked in the vertical position. Using the new fuel handling tool, the fuel bundle will be supported by the fuel handling crane. The clamping device securing the fuel bundle to the strongback is then removed. The fuel assembly is lifted out of the canister and placed in the new fuel inspection area. After the fuel is inspected dimensionally and visually, it will be transported to the new fuel storage area. The new fuel storage rack capacity is only sufficient to hold one-third of a core of new fuel. As a minimum two-thirds of a core of new fuel will require storage in the spent fuel storage racks. During refueling, fuel is normally shuffled only from the spent fuel pool to the reactor for speed and efficiency.

Fuel is transported to the new fuel storage area using one of the following methods: 1) fuel is transported using the fuel handling crane and the new fuel handling tool; or 2) fuel is transported using the spent fuel handling machine and the intermediate new fuel handling tool. After the fuel assembly has been transferred, the entire process is repeated for the next fuel assembly, then the canister or cask is reassembled and placed in a location to be transported back to the fuel manufacturing facility. The next

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canister or cask is brought into the new fuel inspection area and the entire process repeated until all of the new fuel has been received. The fuel remains in its storage location until fuel load.

2.5.1.5 Plutonium Fuel Receipt

The following revised fuel receipt sequence and equipment will be required to handle plutonium because it is radioactive when received. The estimated receipt radiation level is 100 - 200 mr/hr.

The Plutonium fuel will be received at the Palo Verde site in their shipping canisters or casks as required. Upon arrival at the site accelerometers, purge gas and radiation levels are checked. Fuel storage canisters or casks are washed, removed from the truck and placed in the fuel building. Specific canister or cask stacking requirements must be adhered to. Normally one truckload of fuel at a time is received, off-loaded and uncrated, inspected and placed in the new or spent fuel storage area. Containers are placed back on the truck and returned to the manufacturer's facility as a new truckload is received.

As plutonium fuel is radioactive, either shielding or special personnel precautions will be required. Three options for fuel transfer from the shipping canister or cask follow:

- 1) The expected radiation level of new plutonium fuel (100 - 200 mr/hr) while not high, is high enough to prevent extended personnel exposure at close distances. Maintaining a moderate distance, e.g. 3 feet or greater will significantly reduce radiation exposure. Shielding of each fuel assembly as it is withdrawn from the shipping canister or cask and using remotely actuated tools will also significantly reduce personnel radiation exposure. This method uses the least amount of shielding, but due to crane heights, the shielded assembly will require its own lifting mechanism with interlocks, etc.
- 2) Removing the fuel assembly into a shielded area for transfer will also significantly reduce personnel radiation exposure. This method will require a wall of shielding (possibly temporary) along the transfer path. This method requires more shielding, but allows the use of existing lifting equipment. It is expected that moderately thick plexiglass (1" or greater) will provide acceptable shielding and still permit visual observation.
- 3) Fuel inspection should be performed in either a shielded fuel assembly arrangement or in a shielded area. Remote visual inspection should not pose any problems. Remote actuation of the current dimensional inspection equipment or development of new remotely operated dimensional inspection equipment may be required. Once inspected, the fuel assembly will be transferred to the spent fuel through the new fuel elevator. Shielding

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requirements will also apply here. Transfer to the new fuel storage area may also be possible providing the new fuel storage area has a shielded cover.

2.5.2 Scenarios for Storage of New Fuel, Existing Spent Fuel and Unloaded Fuel

2.5.2.1 New Fuel

Due to the radioactivity of the new plutonium fuel, it must either be stored under water or in a shielded arrangement. As described above, the new fuel storage area can accommodate 90 fuel assemblies. Although designed for uranium fuel which as new fuel is not radioactive, the storage rack could accommodate plutonium fuel providing a shield was designed to cover the areas not specifically being used, and adequate personnel distance from the fuel is maintained. Normally, for simple and efficient handling, new fuel is shuffled into the reactor from the spent fuel pool using the spent fuel handling machine. Prior to a refueling outage new fuel will be shuffled from the new fuel storage area to the spent fuel pool. For our evaluation of the storage requirements for new and spent fuel with respect to the capacity of the storage area, the new fuel storage area will be considered as a temporary storage area. It will be considered to hold only fuel received after a refueling outage that cannot be transferred into the spent fuel pool until spent fuel assemblies are removed from the plant. Thus its function will be as an overflow area. Its location and lack of safeguards also make it a less secure location than the spent fuel pool with its many interlocks.

2.5.2.2 Existing Fuel

Currently Palo Verde has been operating for about 7 years. At each refueling outage about 1/3 of the fuel assemblies are removed. For 18 month refueling, that would mean that about 1.5 cores have been used. Since the core currently residing in the reactor would have to be removed and stored, space for about 2.5 cores would be required at Palo Verde Unit 1, or about 603 fuel assemblies. In the checkerboard fuel storage mode there is storage for up to 665 fuel assemblies. Thus most of the fuel storage in the checkerboard manner would be used up and any further storage would require storage in the borated or mixed fuel storage mode. Storage for up to 1239 new or spent fuel assemblies can be provided in this mode. To store a new initial plutonium core would as a minimum require a mixed fuel storage mode.

2.5.2.3 Plutonium Fuel

Assuming that the 603 Palo Verde uranium fuel assemblies will remain in storage at Palo Verde, this will leave 636 fuel locations available for storage which is about 2.6 cores. Based on an 18 month 1/2 plutonium reload schedule, 10 years of operation would require 3 cores or 723 fuel assemblies. Additionally a new core for initial installation would require storage be provided for 241 fuel assemblies. A total of 964 plutonium fuel assemblies would require storage in the spent fuel pool to properly support refueling outages. The total of the existing Palo Verde fuel assemblies and the

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plutonium fuel assemblies is 1567 fuel assemblies which exceeds the storage capacity of one unit of the Palo Verde plant by 328 fuel assemblies. This means that approximately 1.36 cores of existing UO_2 or plutonium fuel assemblies will need to be removed prior to the end of the proposed ten year storage period. Depending on the number of Palo Verde units employed in the mission, it may be possible to transfer fuel assemblies between units, rather than build a new facility for storage of spent fuel. These options need to be explored further.

2.5.3 Fuel Loading Process

The description of the fuel loading process will start with fuel being loaded into the fuel transfer system for installation in or removal from the core. Detailed preparations for these operations are not covered. The spent fuel handling machine loads a new fuel assembly into a fuel cavity in the upender portion of the fuel transfer system. The fuel transfer system contains two fuel cavities. The upender will rotate to the horizontal orientation. If the core is unloaded, then no parallel operation will occur over the reactor. If there is a core in the reactor, the refueling machine will remove a fuel assembly from the reactor. The upender with the new fuel assembly will move from the fuel building to the reactor building. The refueling machine will place the fuel assembly in empty fuel cavity in the upender portion in the reactor building side. The refueling machine will then take the new fuel assembly from the other fuel cavity. It will place the new fuel assembly in the prescribed location in the core. The upenders in each side will be rotated to the horizontal orientation and transferred to the fuel building. In parallel with this operation, the spent fuel handling machine is retrieving another fuel assembly from the spent fuel pool to place in the fuel transfer system. Fuel is shuffled in this manner until all scheduled fuel assembly transfers are accomplished.

As the above procedure of using the spent fuel handling machine to remove new fuel assemblies from the spent fuel pool is the most efficient manner of refueling, the handling of MOX fuel should not change the shuffling procedure.

2.5.4 Modifications

For fuel transfer from the new fuel canister or cask to the new fuel storage area, provisions for shielding of the plutonium fuel assemblies will be required. In order to provide visibility for the transfer operations this shielding should be made of plexiglass, either by enclosing only the fuel assembly, or enclosing a prescribed transfer path.

For Receipt inspection of fuel remotely actuated dimensional inspection equipment may be required.

For storage of fuel assemblies, poison inserts if not already available will be required for all locations.

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2.6 Fuel Management

The several approaches to fuel management for existing reactors are possible. In this study, 2 batch approaches for twelve and eighteen month cycles are presented for the 3410 and the System 80 classes of reactors, respectively. Other approaches should be explored to determine the effect these variations will have on cost, mission time, and other energy production parameters.

An important parameter to evaluate the mission completion is the Pu-240 fraction. There is a strong dependence on the time to reach the desired Pu-240 fraction (greater than 20%) and the initial Pu w/o. This is illustrated in Figure 2.6-1. Three state points are highlighted. For the System 80+ (Advanced Light Water Reactor) NSSS system described earlier (Reference 2.6.1), 6.7 w/o Pu was loaded and ~1200 EFPD were needed to reach the greater than 20% Pu-240 target. For the System 80 plant in this study, the Pu-240 target was reached with 4.5 w/o Pu loading and ~900 EFPD. The results for the 3410 class were 3.0 w/o Pu loading and ~550 EFPD.

The fuel is the standard System 80 design with MOX pellets in a 16x16 lattice assembly. The characteristics for the MOX fuel design are discussed in Section 2.1.1. The summary of the fuel management parameters is given in Table 2.6-1.

Fuel Management - Case B1 - System 80, 50 MT

The System 80 reactor system as used for Plutonium disposition is designed for this base case (B1- 50 MT in System 80) to operate for two eighteen month cycles, reloading half the core during each refueling outage. Consistent with the eighteen month cycles currently being utilized by System 80 units, this fuel management is designed for a capacity factor of 82% and will have an average batch discharge of 32,500 MWD/T. Case B1 has every feed assembly loaded with 4.5 w/o PuO₂ mixed with tails UO₂. One third of the feed assemblies are standard MOX; the other two thirds has erbium admixed with the fuel at a 1.6 w/o Er₂O₃ level. Operating on an 18 month cycle, the core average discharge burnup is 32,500 MWD/T.

A typical equilibrium cycle fuel management scheme for System 80 is given in Figure 2.6-2 with a typical Z = 2 pattern. There are two batches (fresh and once burned) with a 40/80 split in each batch representative of the no Erbium and erbium admixed with the fuel pins as discussed above.

Fuel Management - Case B2 - 3410 Class, 50 MT

For the other base case mission (B2- 50 MT in 3410), every feed assembly has 3.0 w/o PuO₂ mixed with tails UO₂. Forty-eight percent of the feed assemblies are standard MOX; the other fifty-two percent has erbium admixed with the fuel at a 1.6 w/o Er₂O₃ level. For the 3410 class plant, a typical equilibrium cycle fuel management scheme is given in Figure 2.6-3 with a typical Z = 2 pattern. There are two batches

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(fresh and once burned) with a 52/57 split in each batch representative of the no Erbium and erbium admixed with the fuel pins as discussed above. The core average discharge burnup is 20,000 MWD/T, operating on a 12 month cycle.

These fuel management schemes meet the mission requirements of Pu-240 fraction and desired Pu through put while still meeting the design objectives of power peaking and safety parameter ranges while retaining ability to shutdown the reactor safely. The design objectives of Pu disposition and Pu-240 isotopic fraction typical of spent fuel from conventional LWRs is accomplished easily at the stated 82% capacity factor. The fuel management for this study was performed using the new ENDF/B-VI library. The verification of this library was discussed in Section 3.2.1 and Appendix B of the Evolutionary Reactor Report.

The scoping study of the physics related safety parameters are summarized in Table 2.6-2 and compared with values calculated earlier for commercial UO_2 and all-plutonium reactor (APR) studies for the System 80 plant in the 1970s. The results for the existing reactors (System 80 and 3410 class plants) Plutonium Disposition operating in the spent fuel mode using weapons-grade plutonium fuel show that they are within the range of those for commercial APR.

Comparison of Physics Parameters for MOX Concept

Table 2.6-2 gives a comparison of physics parameters for the MOX fueled existing reactors 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) and Inverse Boron Worth (IBW)

Calculated values of CBC and IBW at full power are shown in Table 2.6-2 for the MOX cycles in existing reactors using weapons-grade plutonium.

For the MOX cycles, a relatively high CBC exists at BOC, consistent with the reduced soluble boron worth shown by the IBW. The CBC decreases over the cycle, and indicates that a non-trivial amount of excess reactivity (1% to 2% Δk for the System 80 and 3410 class plants) remains at EOC. The CBC and IBW at BOC are higher than those reported for the commercial UO_2 core, but are generally less than that required by the commercial APR with recycled plutonium. In addition, the CBC and IBW levels are lower than those reported for the System 80+.

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b. Control Rod Worth

Calculated values of EOC core reactivity worth of the control element assemblies (CEAs) are shown for hot-full-power conditions in Table 2.6-2 for the existing reactors MOX cycles. The CEA worth values are based on the extended CEA patterns for APR operation as discussed in Section 2.1. On the basis of the calculated results, the available CEA shutdown worth is sufficient for normal operations and safety-related requirements of the MOX cycles. Explicit calculations performed at EOC, 300 °F, in the N-1 configuration show that there is no return to power. The behavior of the CEA worth over the cycle shows continuous increase in worth from BOC to EOC for the MOX cases, consistent with the expected trend based on the depletion of the plutonium and the decrease in the IBW.

c. Moderator Temperature Coefficient (MTC)

Calculated values of MTC at BOC and MTC are given in Table 2.6-2 for the MOX cycles. The high fissile plutonium content (supplemented by the effect of erbium) provides a more negative MTC at BOC than in UO₂ cycles. While these EOC MTC values are more negative than those reported for the commercial cycles, they are deemed to be acceptable. The Erbium loading levels are responsible for the more negative MTC values (both BOC and EOC). While these values are close to the present tech spec limits, it is important to note that even with these negative MTC values at EOC, there is no return to power on a Steam Line Break (SLB) accident. Further studies of burnable absorbers (both levels and types of materials) should allow finer tuning of this MTC range. If the existing reactors were to lower the outlet temperature to meet the EPRI ALWR guidelines (less than 600 °F), the requirements for SLB reactivity insertion would be significantly reduced.

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d. Delayed Neutron Fraction

Comparisons of delayed neutron fraction (beta-effective) and prompt neutron lifetime (l-star) are given for BOC and EOC in Table 2.6-2. For the Mox cycles studied, the values of beta effective are in the range of 0.0033 to 0.0037. These values are higher than those reported earlier in the Evolutionary Reactor Report for the System 80+ and are lower than those for the commercial APR. The amounts of Pu-239 concentration (in combination with the U-238) falls between the two reference cases. Based on evaluations for the commercial APR cycle, the lower beta effective for these existing reactors MOX concepts is expected to be acceptable for safety related performance (e.g., CEA ejection accident).

e. Fuel Temperature Coefficient (FTC)

The FTCs were not calculated for the existing reactor configuration. Since the FTC is proportional to the amount of the U-238 and Pu-240 isotopes present and the flux spectrum, the values will be bounded by those calculated for the System 80+. The FTC should be no more negative than -1.6×10^{-4} d-rho/°F and show little variation with burnup.

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TABLE 2.6-1
EXISTING REACTORS MOX CORE
EQUILIBRIUM CYCLE
FUEL BATCH CHARACTERISTICS

Parameter	System 80	3410 Class
Number of Batches	2	2
Cycle Length (Months)	18	12
Capacity Factor (%)	82	75
Fuel Sub-batch Type	Q0	R0
Number of Assemblies	40	52
PuO ₂ w/o	4.5	3.0
Er ₂ O ₃ w/o	0.0	0.0
Fuel Sub-batch Type	Q1	R1
Number of Assemblies	80	57
PuO ₂ w/o	4.5	3.0
Er ₂ O ₃ w/o	1.6	1.6
Discharge Burnup (MWD/T)	32,500	20,000

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TABLE 2.6-2
SUMMARY OF SAFETY PARAMETERS
COMMERCIAL REACTOR (1970) VERSUS EXISTING REACTORS WITH WEAPONS-GRADE PLUTONIUM

Parameter	Units	Commercial UO ₂ , Eq. Cy.	Commercial APR, Eq. Cy.	System 80 Eq. Cy.
Moderator Temperature Coef	10 ⁻⁴ Δρ/°F			
Full Power BOC		-0.59	-1.00	-2.50
Full Power EOC		-3.24	-3.10	-4.30
Fuel Temperature Coef	10 ⁻⁵ Δρ/°F	-1.25	-1.09	-1.57 ¹¹
Full Power EOC				
BOC Critical Boron Conc.	ppm			
Hot Standby		1589	3189	2783
Full Power, No Xenon		1499	2450	2238
Full Power, Eq. Xenon		1170	2100	1773
EOC Critical Boron Conc.				200
Inverse Boron Worth	ppm/%Δρ			
Full Power, BOC		116.0	383.0	277.0
Full Power, EOC		101.0	331.0	220.0
Number of CEA Drives		87	97	97
CEA Worth, Full Power EOC	%Δρ			
Total		13.80	12.60	14.20
Net		10.20	9.80	11.71
Stuck		3.80	2.80	2.49
Eff. Delayed Neutron Fraction				
Full Power, BOC		0.00625	0.00442	0.00339
Full Power, EOC		0.00546	0.00447	0.00366
Prompt Neutron Lifetime	10 ⁻⁶ Secs			
Full Power, BOC		21.3	6.8	8.92
Full Power, EOC		24.8	7.9	11.38
Reactivity: EOC Cold 300°F	%Δρ			-1.4
Discharge Burnup	MWD/T			32,500

Note 1: Value bounded by that of Reference 2.6.1 for System 80 + ALWR calculation

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2.6.1 Reactor Startup Plutonium Inventory and Discharge Rate

As described in Section 2.6, the existing reactors (System 80 and 3410 classes) Plutonium disposition reactor systems are designed to operate for cycles that are consistent with meeting the disposition mission requirements and the safety parameter window. Therefore two batch approaches for twelve and eighteen month cycles are presented for the 3410 and the System 80 classes of reactors, respectively.

The characteristics for the MOX fuel design are discussed in Section 2.1.1 and are summarized in Table 2.6-1. Typical equilibrium cycle fuel management approaches are presented in Figures 2.6-2 and 2.6-3 for the System 80 and 3410 class plants, respectively.. A typical first cycle for System 80 APR with weapons grade plutonium is presented in Figure 2.6.1-1.

Thus for the System 80 reactor, the fuel management would start with an approach similar to Figure 2.6.1-1 and quickly transition to an equilibrium cycle like Figure 2.6-2. A similar approach could be taken for the 3410 class plant to arrive at an equilibrium cycle like Figure 2.6-3.

The initial (equilibrium) recharge rates are given in Table 2.6.1-1 (2.6.1-2). The batch sizes for the System 80 (3410) class plant changes from 241 (217) to 120 (108) for the equilibrium cycle.

Table 2.6.1-3 illustrates the case B1 schedule for the startup and operation of a two unit System 80 site to dispose of 50 MT of Plutonium over an 23 year period from the start of the contract to convert these plants and to build and operate the fuel fabrication facility. It is assumed that it would take 5 years to prepare for fuel load of the MOX fuel after the award of the contract. Each of these existing System 80 reactor systems has a design life of 40 years. After completing the plutonium disposition mission, these plants can return to burn commercially available fuel for the remaining 10 years of plant life.

Table 2.6.1-4 illustrates the case B2 schedule for the startup and operation of a two unit 3410 class site to dispose of 50 MT of Plutonium over an 26 year period from the start of the contract to convert these plants and to build and operate the fuel fabrication facility. It is assumed that it would take 5 years to prepare for fuel load of the MOX fuel after the award of the contract. Each of these existing 3410 class reactor systems has a design life of 40 years. After completing the plutonium disposition mission, these plants can return to burn commercially available fuel for the remaining 6 years of plant life.

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**TABLE 2.6.1-1
RECHARGE FUEL LOADINGS
ISOTOPIC COMPOSITION
FIRST CYCLE
BATCH TONNES**

		One Batch System 80	One Batch 3410 class
U235		.2051	.1866
U236		0	0
U237		0	0
U238		102.30	93.10
U237		0	0
PU238		0	0
PU239		2.967	1.781
PU240		.2063	.1238
PU241		0	0
PU242		0	0
AM241		0	0
AM243		0	0
CM242		0	0
CM244		0	0
TOTAL-HM		105.70	95.19
TOTAL U		102.50	93.29
TOTAL PU		3.173	1.905
TOTAL OTHER		0	0
Pu Isotope Fraction			

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		One Batch System 80	One Batch 3410 class
Pu-238/Pu		0	0
Pu-239/Pu		.935	.935
Pu-240/Pu		.065	.065
Pu-241/Pu		0	0
Pu-242/Pu		0	0

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**TABLE 2.6.1-2
RECHARGE FUEL LOADINGS
ISOTOPIC COMPOSITION
EQUILIBRIUM CYCLE
BATCH TONNES**

		Case B1 System 80 50 Mt	Case B2 3410 class 50 Mt
U235		.1010	0.09233
U236		0	0
U237		0	0
U238		50.38	46.07
NP237		0	0
PU238		0	0
PU239		2.226	1.336
PU240		.1542	.09288
PU241		0	0
PU242		0	0
AM241		0	0
AM243		0	0
CM242		0	0
CM244		0	0
TOTAL-HM		52.86	47.60
TOTAL U		50.48	46.17
TOTAL PU		2.38	1.429
TOTAL OTHER		0	0

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		Case B1 System 80 50 Mt	Case B2 3410 class 50 Mt
Pu Isotope Fraction			
Pu-238/Pu		0	0
Pu-239/Pu		.935	0.935
Pu-240/Pu		.065	0.065
Pu-241/Pu		0	0
Pu-242/Pu		0	0

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**TABLE 2.6.1-3
DEPLOYMENT STRATEGY SCHEDULE
2 - SYSTEM 80'S
BASE CASE B1 - 50 MT IN 18 YEARS**

Number of Reactor Units 2 Core Power Rating 3800 MWth

Length for Each Cycle 18 Months - Shipping Begins 138 Months after each BOC

Cycle Length	Months	EFPD	CF	Z
	18	450	.82	2

No. of MT PU Burned Per Mission 2.25
 First Core Startup Test Period: 6 months
 Number of Feed Batches for Mission: 23 For Total of: 51.75 MT Pu

Cycle	Months	EFPD	CF	Z
1	60	78	198	216
2	84	96	222	234
3	102	114	240	252
4	120	132	258	270
5	138	150	276	288
6	156	168	294	306
7	174	186	312	324
8	192	204	330	342
9	210	222	348	360
10	228	240	366	378
11	246	258	384	396
12	264		402	0

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**TABLE 2.6.1-4
DEPLOYMENT STRATEGY SCHEDULE
2 - 3410 CLASS PLANTS
BASE CASE B2 - 50 MT IN 19 YEARS**

Number of Reactor Units 2 Core Power Rating 3410 MWth
Length for Each Cycle 12 Months - Shipping Begins 138 Months after each BOC

Cycle Length	Months	EFPD	CF	Z
	12	274	.75	2

No. of MT PU Burned Per Mission: 1.32
First Core Startup Test Period: 6 months
Number of Feed Batches for Mission: 38 For Total of: 50.16 MT Pu

Operating Cycles			Shipping	
Cycle	Unit 1	Unit 2	Unit 1	Unit 2
1	60	72	192	204
2	78	84	210	216
3	90	96	222	228
4	102	108	234	240
5	114	120	246	252
6	126	132	258	264
7	138	144	270	276
8	150	156	282	288
9	162	168	294	300
10	174	180	306	312
11	186	192	318	324
12	198	204	330	336
13	210	216	342	348
14	222	228	354	360
15	234	240	366	372
16	246	252	378	384

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Operating Cycles			Shipping	
Cycle	Unit 1	Unit 2	Unit 1	Unit 2
17	258	264	390	396
18	270	276	402	408
19	282	288	414	420

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2.6.2 Fuel Cycle Characteristics

As described in Section 2.6, the existing reactors operating on the Plutonium disposition mission is designed to operate with fuel management that is similar to that used for the UO_2 fuel cycle. As discussed earlier Table 2.6-1, the System 80 has an 18 month cycle and an 82 percent capacity factor. The 3410 class plants have a 12 month cycle and an 75 percent capacity factor. Typical equilibrium fuel management scheme are given in Figures 2.6-2 and 2.6-3. While there is an economic penalty (relative to the equilibrium cycles), it is possible to design a first cycle that has weapons grade plutonium in every assembly.

Base case B1 is run with 18 month cycles and an 82% capacity factor. The startup of the first reactor unit in this complex has an additional 6 months during the first cycle to complete the additional startup and validation testing for this MOX fuelled system. This is probably an over estimation, since the Plutonium disposition mission in existing plants is expected to have operational characteristics very similar to those experienced in the UO_2 fuel cycle. As shown in Table 2.6.1-3, the other unit on this site will commence operation 18 months after unit 1 and all cycles on this site (with the exception of Unit 1, cycle 1) will operate with 18 month cycles.

Deployment of the mission in the 3410 class plants (Base case B2) is run with annual cycles and a 75% capacity factor. Again the estimation of an additional six months of startup and validation testing time is allowed but is felt to be generous. As shown in Table 2.6.1-4, the other unit on this site will commence operation one year after unit 1 and all cycles on this site (with the exception of Unit 1, cycle 1) will operate with annual cycles.

In this base case B1 (B2) study, the plant will operate for two 18 month (12 month) cycles of 450 (274) EFPD each. The B1 (B2) case operates at a capacity factor of 82 (75)percent. Both of these cases allow 90 days per cycle for refuelling, maintenance, and/or other operations that may be necessary for other plant missions. The increase from 75 to 82 percent is purely a function of the cycle length. Based on the experience of the System 80+ plant design, it is possible that some System 80+ design features might be incorporated that could routinely reduce the outage time to less than 3 months and result in higher capacity factors that could accomplish the plutonium disposition mission sooner and generate more energy and revenue for the project. These System 80+ design features that consider operations and maintenance functions are discussed in Section 4.4 of the Evolutionary Reactor Report.

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The recharging strategy is a simple two batch equilibrium cycle strategy. The fuel management for B1 (B2) follows a strategy similar to that outlined in Figure 2.6-2 (2.6-3). Every cycle, a split feed batch will replace half the fuel assemblies in the core. Each refueling uses the same pattern as was used in the previous cycle. Experience has shown that it is relatively easy to design a transition cycle once the specific operating cycle parameters have been developed.

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2.6.3 As-charged fuel actinide isotopic composition

As described in Section 2.6 and 2.6.2, the existing reactors operating on the Plutonium disposition mission is designed to operate with fuel management that is similar to that used for the UO_2 fuel cycle. As discussed earlier Table 2.6-1, the System 80 has an 18 month cycle and an 82 percent capacity factor. The 3410 class plants have a 12 month cycle and an 75 percent capacity factor. Typical equilibrium fuel management scheme are given in Figures 2.6-2 and 2.6-3.

The recharging strategy is quite simple. The fuel management for B1 (B2) follows a strategy similar to that outlined in Figure 2.6-2 (2.6-3). Every cycle, a split feed batch will replace half the fuel assemblies in the core. Each refueling uses the same pattern as was used in the previous cycle. The as charged fuel actinide isotopic composition for the equilibrium cycle is given in Table 2.6.1-2. The actinides are represented both as tonnes of heavy metal and as isotopic fractions.

Each feed batch is composed of 120.5 (108.5) assemblies loaded every 18 months (12 months). The heavy metal is composed of diluent as tails Uranium (0.2 w/o of U-235) with weapons grade Plutonium (93.5 w/o Pu-239, 6.5 w/o Pu-240). As noted in Table 2.6-1, the B1 (B2) case has a PuO_2 enrichment of 4.5 w/o (3.0 w/o). Er_2O_3 is added to a fraction of the feed batch assemblies to control the reactivity and safety parameters associated with the weapons grade plutonium.

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2.6.4 Discharged fuel actinide isotopic composition

As described in Section 2.6, the existing reactors Plutonium disposition case B1 (B2) is designed to operate with a two batch fuel management on an 18 (12) month cycle for three (two) years with discharge exposure of 32,500 (20,000)MWD/T.

As described above, half the core will be reloaded at each refueling outage. The discharged fuel actinide isotopic composition is given in Table 2.6.4-1. The actinides are represented both as tonnes of heavy metal and as isotopic fractions.

Each discharge batch is composed of 120 (108) assemblies off-loaded at the end of every cycle. For the B1 (B2) case, there are two cycles of 450 (274) EFPD with a capacity factor (CF) of 82 (75) percent.

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**TABLE 2.6.4-1
DISCHARGED FUEL ACTINIDES
ISOTOPIC COMPOSITION**

Batch Tonnes

		Case B1 System 80 50 Mt	Case B2 3410 class 50 Mt
Discharge	MWD/T	32,500	20,000
U235		.05725	0.05753
U236		.008893	0.006768
U237		.000017	0.000014
U238		48.39	44.59
NP237		.000708	0.000379
PU238		.000131	0.000052
PU239		1.125	.7995
PU240		.4322	.2629
PU241		.2524	.1445
PU242		.03846	0.01964
AM241		.001101	0.004103
AM243		.007685	0.00298
CM242		.001958	0.000702
CM244		.00208	0.000600
TOTAL-HM		50.33	45.90
TOTAL U		48.46	44.66
TOTAL PU		1.848	1.227
TOTAL OTH		.0238	.00886

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
TECHNICAL DESCRIPTIONS**

Pu Isotope Fraction			
Pu-238/Pu		.0001	0.0000
Pu-239/Pu		.6086	0.6518
Pu-240/Pu		.2339	0.2143
Pu-241/Pu		.1366	0.1178
Pu-242/Pu		.0208	0.0160

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

2.7 Nuclear Material Transportation

2.7.1 Plutonium Oxide and Uranium Oxide Powder to Fabrication Plant.

The System 80+ plutonium mission plant layout in the Evolutionary Reactor Report considered the MOX fuel fabrication plant to be collocated with the reactor at the same site. This would avoid the shipping of the fresh MOX fuel assemblies from the MOX fabrication plant to the reactor. In the present case of loading existing reactors with MOX fuel assemblies, the MOX fuel fabrication

1. The PuO_2 powder will be shipped from PANTEX facility to MOX fabrication facility in Kenosha, Wi. This is an approximate travel distance of about 1700 miles by road.

2. The UO_2 powder from the DOE enrichment facility will be shipped to Kenosha, Wi. This will require a travel distance of about 1200 miles by road from Oak Ridge, Tenn.

The shipments will be conducted to meet DOE, DOT requirements and NRC regulations on shipping of nuclear materials. The manner of shipment will be similar to the shipments undertaken by DOE in the past of plutonium material from the Rocky Flats, Co facility to Pantex facility in Texas.

The MOX facility is designed for a throughput of 50 Metric tons of MOX fuel per year. To account for losses and maintaining an inventory of raw fuel materials, the transportation is based on 70 Metric tons. The MOX fuel for the reference case will require a transport of about 4.5 tons of PuO_2 powder from Pantex to Kenosha in a year. This amounts to about 400 Kg per month. The incoming plutonium will be transported in a double container package with a maximum amount in each package not exceeding two kilograms. With 2 Kg per container, a total of 200 containers containing PuO_2 powder will be shipped from Pantex facility to MOX fuel fabrication plant at Kenosha, Wi every month.

This shipment can be accomplished by using the Safe Secured Trailers (SST) presently utilized by DOE. The shipment is assumed to travel a total of 300 miles in a day. Thus, a total of six days of travel time will be required to ship PuO_2 powder from Pantex to MOX facility at Kenosha, Wi. These shipments will be done with proper security measures both during the transportation phase and on its arrival at the MOX facility. A proper communication link with the Pantex facility and the MOX fabrication operational center will be maintained during the entire phase of the shipment. Two drivers will accompany each shipment and the number of trucks to be used will depend on the number of containers accommodated in each truck.

reactor at the same site. This would avoid the shipping of the fresh MOX fuel assemblies from the MOX fabrication plant to the reactor. In the present case of loading existing reactors with MOX fuel assemblies, the MOX fuel fabrication plant is assumed for the purposes of this study to be located away from the reactor site at Kenosha, Wisconsin.

1. The PuO_2 powder will be shipped from PANTEX facility to MOX fabrication facility in Kenosha, Wi. This is an approximate travel distance of about 1700 miles by road.

2. The UO_2 powder from the DOE enrichment facility will be shipped to Kenosha, Wi. This will require a travel distance of about 1200 miles by road from Oak Ridge, Tenn.

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PLUTONIUM DISPOSITION IN EXISTING REACTORS

TECHNICAL DESCRIPTIONS

The UO₂ powder will be shipped in 55 gallon drums. The total quantity of UO₂ powder for the reference case to be shipped is about 90 tons per year. This amounts to about 8 tons per month. Each 55 gallon drum will be loaded to contain only about 1 ton of Uranium oxide powder to assist in the handling both at the loading area and the unloading operations at the MOX facility. The Uranium powder can be transported in a manner similar to the one utilized by the commercial nuclear industry. The distance to be travelled is about 1200 miles or a four day trip.

In addition, the fuel tubes and fuel cladding material will be shipped from the Hematite plant in Missouri to the fuel fabrication plant in Kenosha.

2.7.2 Fresh Fuel from Fuel Fabrication Plant

The MOX fuel assemblies fabricated at the Kenosha plant will be shipped for use at one or more of the reactors. The reactors considered for the present study are:

Palo Verde Units 1, 2 or 3;
Washington Nuclear Plant Unit 3
San Onofre Units 2 or 3
Arkansas Unit 2
Waterford Unit 3

The presently certified PWR shipping cases can be utilized for these shipments. Each PWR case can contain two fuel assemblies. It should be noted that the MOX fuel will have inherent radioactive level due to decay of the americium isotopes. Thus, some extra temporary shielding may be needed for the cases to ensure the external dose levels during shipment meet the DOT requirements. The cases are designed to protect the fuel from damage that might occur during a truck wreck and the resulting fire including a plunge into a body of water. The transport vehicles utilized by the commercial nuclear industry for LWR new fuel assemblies can be utilized. The number of new fuel assemblies to be shipped per year depends on the reactor utilized for loading the MOX fuel. For this study, it is assumed that the reactor is System 80, two units, the core contains 241 fuel assemblies with a fuel cycle length of 18 months when one half of the core is replaced. This yields a levelized schedule for shipping of 82 fuel assemblies per year. This is equivalent to 82 cask shipments. Assuming the truck carries 4 casks at each shipment, a total of 21 shipments will be required during the year or approximately two a month. The distance from the Kenosha plant to the reactor site varies from 900 miles for Arkansas Unit 2 to a distance of 2130 miles for the San Onofre plant. This will require 3 to 8 days of travel based on a 300 miles of travel per day.

fuel could then follow in the alternative quarters.

Once the ten year cooling period is over, there will be 241 assemblies available for shipment every year from the System 80 class plant. A typical shipping plan for the eighteen month cycles of base case B1 is shown in Figure 2.7.4-1. After the initial core startup, the batches become available for shipping on a staggered basis - either six or twelve months apart. Depending on the scenario, there could be shipments of zero, 120, or 241 assemblies per year. The

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNICAL DESCRIPTIONS

shipping schedule that corresponds to Figure 2.7.4-1 is discussed in Section 2.6.1-3. With the eighteen month cycles, the shipping scheme becomes much smoother with a three unit site or plan (See alternative A1 in Section 10.1). The shipping of the spent fuel could then follow in the alternative quarters.

The 3410 class plant will provide 217 assemblies for shipment every year after the ten year cooling period is over. A typical shipping plan for the annual cycles of base case B2 is shown in Figure 2.7.4-2. With the exception of a gap between the initial cores on units 1 and 2, the batches become available for shipping every six months. The shipping schedule that corresponds to Figure 2.7.4-2 is discussed in Section 2.6.1-4. The two unit site with annual refueling is a very efficient operation from the standpoint of scheduling the outage related work.

2.7.5 Waste from Reactor

The transportation of waste from the reactor for the case where existing reactors are employed is similar as that reported in the Evolutionary Reactor Report.

Figure 2.1.1-1 REACTOR COOLING SYSTEM ARRANGEMENT-ELEVATION VIEW

Figure 2.1.1-2 REACTOR COOLING SYSTEM ARRANGEMENT - PLAN VIEW

Figure 2.1.1-3 TYPICAL REACTOR VESSEL ASSEMBLY

Figure 2.1.1-4 REACTOR VESSEL COOLANT FLOW

NO.	COMPONENT	QUANTITY
1	VENT PIPE	1
2	CEDM NOZZLES	97
3	CLOSURE HEAD LIFTING LUG	12
4	STUDS, NUTS & WASHERS	54
5	CLOSURE HEAD FLANGE	1
6	REACTOR VESSEL FLANGE	1
7	OUTLET NOZZLE	2
8	ICI NOZZLE	63
9	EXTERNAL SHEAR LUG	4
10	VESSEL SUPPORT PAD	4
11	INLET NOZZLE	4
12	SURVEILLANCE MATERIAL HOLDER	6
13	CORE STABILIZING LUG	6
14	CORE STOP LUG	6
15	FLOW BAFFLE	1

*

INLET NOZZLE ID: 30" (762mm)

OUTLET NOZZLE ID: 42" (1067mm)

Figure 2.1.1-5 SYSTEM 80 (3817 MWt PLANT) REACTOR VESSEL

PLAN VIEW OF CLOSURE HEAD

- 97 - EXISTING CEDM NOZZLE PENETRATIONS
- 1 - EXISTING VENT PIPE PENETRATION
- 16 - ADDITIONAL PENETRATION LOCATIONS AVAILABLE

Figure 2.1.1-6 SYSTEM 80 REACTOR VESSEL CLOSURE HEAD

NO.	COMPONENT	QUANTITY
1	VENT PIPE	1
2	CEM NOZZLE	91
3	INSTRUMENT NOZZLE	10
4	CLOSURE HEAD LIFTING LOG	12
5	STUDS, NUTS & WASHERS	84
6	CLOSURE HEAD FLANGE	1
7	REACTOR VESSEL FLANGE	1
8	OUTLET NOZZLE	2
9	VESSEL SUPPORT PAD	4
10	INLET NOZZLE	4
11	SURVEILLANCE MATERIAL HOLDER	6
12	CORE STABILIZING LOG	6
13	CORE STOP LOG	9
14	FLOW DAFFLE	1

* INLET NOZZLE ID: 30" (762mm)
 OUTLET NOZZLE ID: 42" (1067mm)

Figure 2.1.1-7 3410 MWt PLANT REACTOR VESSEL

- 91 - EXISTING CEDM NOZZLE PENETRATIONS
- ◐ 10 - EXISTING INSTRUMENT NOZZLE PENETRATIONS
- 1 - EXISTING VENT PIPE PENETRATION
- 9 - ADDITIONAL PENETRATION LOCATIONS AVAILABLE

Figure 2.1.1-8 3410 MWt PLANT REACTOR VESSEL CLOSURE HEAD

Figure 2.1.1-9 SYSTEM 80 REACTOR ARRANGEMENT

Figure 2.1.1-10 SYSTEM 80 REACTOR UPPER GUIDE STRUCTURE ASSEMBLY

4 Element CEA

12 Element CEA

Figure 2.1.1-11 FOUR AND TWELVE ELEMENT CONTROL ELEMENT ASSEMBLIES

Figure 2.1.1-12 TWELVE ELEMENT CEA AND FUEL

**System 80 Plant
3817 MWth
Control Rod Pattern**

12 ELEMENT FULL LENGTH CEA's 48

4 ELEMENT FULL LENGTH CEA's 28

4 ELEMENT PART LENGTH CEA's 13

TOTAL 89 CEA's

S DENOTES SPARE CEA LOCATIONS 8

Figure 2.1.1-13

Figure 2.1.1-14

Figure 2.1.1-15 SYSTEM 80 CONTROL ELEMENT DRIVE MECHANISM

Figure 2.1.1-16 SYSTEM 80 REACTOR CORE CROSS SECTION

Figure 2.1.1-17 SYSTEM 80 FUEL ASSEMBLY

Plan View of RV Closure Head

- 89 CEDM Nozzles with CEA's
- ⊙ 8 Spare (no CEA's) CEDM Nozzles
- 4 New Instrumentation Nozzles

Figure 2.1.1-18 SYSTEM 80 RV CLOSURE HEAD PENETRATIONS

FIGURE 2.3-1
SHUTDOWN COOLING SYSTEM
SHUTDOWN COOLING WITH LPSI PUMPS ONLY

FIGURE 2.3-2
SHUTDOWN COOLING SYSTEM
SHUTDOWN COOLING WITH CS & LPSI PUMPS

**FIGURE 2.3-
SAFETY INJECTION SYSTEM**

FIGURE 2.3-4
CSS FLOW DIAGRAM FOR TRAIN 1 OF 2 IDENTICAL TRAINS*

Time to Reach Target Pu-240 Fraction As a Function of Initial Pu Enrichment

Pu-240 vs Exposure Various Initial Pu Loadings

Figure 2.6-1

LEGEND:

N = Quarter Core Box Number
 I = Batch Identifier
 Q0, Q1 = Fresh Burned Fuel
 P0, P1 = Once Burned Fuel

					1 Q0	2 P1	3 Q0	4 P1
			5 P1	6 Q0	7 P0	8 Q0	9 P1	10 Q1
		11 P1	12 Q0	13 P1	14 Q1	15 P0	16 Q1	17 P1
	18 P1	19 Q0	20 P1	21 Q1	22 P0	23 Q1	24 P0	25 Q1
	26 Q0	27 P1	28 Q1	29 P1	30 Q1	31 P1	32 Q1	33 P1
34 Q0	35 P0	36 Q1	37 P0	38 Q1	39 P1	40 Q1	41 P1	42 Q1
43 P1	44 Q0	45 P0	46 Q1	47 P1	48 Q1	49 P0	50 Q1	51 P1
52 Q0	53 P1	54 Q1	55 P0	56 Q1	57 P1	58 Q1	59 P0	60 Q1
61 P1	62 Q1	63 P1	64 Q1	65 P1	66 Q1	67 P1	68 Q1	69 P0

Q0 = 4.5 w/o PuO₂, 0.0 w/o Er₂O₃
 Q1 = 4.5 w/o PuO₂, 2.0 w/o Er₂O₃
 Q2 = 4.5 w/o PuO₂, 2.0 w/o Er₂O₃

R0 = 4.5 w/o PuO₂, 0.0 w/o Er₂O₃
 R1 = 4.5 w/o PuO₂, 1.6 w/o Er₂O₃
 R2 = 4.5 w/o PuO₂, 2.0 w/o Er₂O₃

P0 = 4.5 w/o PuO₂, 0.0 w/o Er₂O₃
 P1 = 4.5 w/o PuO₂, 1.6 w/o Er₂O₃
 P2 = 4.5 w/o PuO₂, 2.0 w/o Er₂O₃

Figure 2.6-2 System 80 Equilibrium Cycle

Q0 3.0 w/o PuO₂, 1.6 w/o Er₂O₃
 Q1 3.0 w/o PuO₂, 1.0 w/o Er₂O₃

R0 3.0 w/o PuO₂, 1.6 w/o Er₂O₃
 R1 3.0 w/o PuO₂, 1.0 w/o Er₂O₃

3410 MW (th) Equilibrium Cycle

Figure 2.6-3

LEGEND:

N = Quarter Core Box Number

P = Previous Cycle Location of Assembly

B = Batch Identifier:

P0, P1 = Fresh

					1	2	3	4
					P0	P0	P0	P0
		5	6	7	8	9	10	
		P0	P0	P0	P1	P0	P1	
	11	12	13	14	15	16	17	
	P0	P0	P0	P1	P0	P1	P0	
18	19	20	21	22	23	24	25	
P0	P0	P1	P1	P0	P1	P0	P1	
26	27	28	29	30	31	32	33	
P0	P0	P1	P0	P1	P1	P1	P0	
34	35	36	37	38	39	40	41	42
P0	P0	P1	P0	P0	P0	P1	P0	P1
43	44	45	46	47	48	49	50	51
P0	P1	P0	P1	P0	P1	P0	P1	P0
52	53	54	55	56	57	58	59	60
P0	P0	P1	P0	P1	P0	P1	P0	P1
61	62	63	64	65	66	67	68	69
P0	P1	P0	P1	P0	P1	P0	P1	P0

P0: 4.5 w/o PuO₂, 0.0 w/o Er₂O₃

P1: 4.5 w/o PuO₂, 1.6 w/o Er₂O₃

Figure 2.6.1-1 System 80 First Cycle

Figure 2.7.4-1

**Shipping Plan
Base Case B2**

Figure 2.7.4-2

PLUTONIUM DISPOSITION IN EXISTING REACTORS TECHNOLOGY NEEDS

3.0 TECHNOLOGY NEEDS

3.1 Reactor

There are no technology needs for the existing reactors for conversion to the plutonium disposition mission. Initial analysis indicates that full MOX fuel cores can be loaded into the existing reactors identified, and the mission carried out within the existing licensed safety parameters. Some Technical Specification changes may have to be made, but these and the analysis to demonstrate the safety and performance with MOX fuel can all be done with existing methods and databases. Any reactor or system modification needs discussed in Section 2, if desired, can be accomplished with existing technology.

3.2 MOX Fuel

The technology needs for MOX fuel that were discussed in the Evolutionary Reactor Report, Section 3.2 and Appendix D, are applicable to the case of existing reactors. The fuel design for the selected existing reactors is very similar to that for the System 80+ for the plutonium disposition mission, that is, the fuel pellet is a mixture of plutonium, uranium and erbium oxides, the cladding and fuel structures are similar, and the dimensions and flow paths similar.

With the existing reactors, it is possible that the DOE takeover of one or more units for the plutonium mission will make available a resource for the irradiation of test fuel, for the fuel qualification program, in the form of a lead test assembly (LTA) to be loaded in one of the selected reactors. This presumes that the arrangements are made beforehand to permit this to happen. The LTA would consist of a regular fuel assembly, containing approximately 10% MOX fuel pins, and the remainder UO₂ fuel pins and poison rods. The MOX fuel pins would have a weapons-grade Pu enrichment selected to be compatible with the remainder of the fuel assembly and the core, equal to or possibly less than the enrichment selected for the design, and would reside in the reactor for one or two years, before removal for examination. This provides an alternative to irradiation of a test fuel rod bundle in the ATR at INEL.

3.3 New Development and Test Facilities

The Evolutionary Reactor Report, Section 3.3, is applicable to the case of existing reactors.

PLUTONIUM DISPOSITION IN EXISTING REACTORS OPERATIONS

4.0 OPERATIONS

4.1 Government, Utility and Private Industry Interfaces

In the Evolutionary Reactor Report, Section 4.1, the interfaces between the Government, the utility and private industry were outlined for the case of a new facility to be built on Government land, and owned and operated, together with a collocated fuel facility, by a private consortium. The transactions between the industry, the reactor facility and the fuel facility remain the same: fuel is supplied to the reactor as required; private industry provides modification services as needed, and may provide the operating staff; the reactor produces electricity which is marketed, but at rates that take into account the changes in the rate structure due to non-utility ownership and mixing of missions; fuel is discharged when spent; and is stored in the fuel pool until it is eventually transported to an off-site repository. In the case of an existing reactor, with the Government ownership, the interfaces involved are substantially different. The following paragraphs outline the considerations that need to be made.

- **Government Owns the Reactor Facility**

The process of "taking over" an existing plant would involve some formidable steps. First, the utility itself, and its shareholders, would have to approve the transaction, which means that it must make sense from their points of view. A prominent consideration would be that there must be no political or business exposure for either the utility or its shareholders. If this step has been successfully taken, the next consideration would involve the state PUC, which would require the removal of the plant asset from the utility's rate base. The utility, which has spent a great deal of effort and resources in getting the facility into the rate base, in a situation wherein it could earn a return on its investment, is likely not willing to have this facility removed and suffer the loss of revenue involved. Typically, nuclear plants make up a disproportionate share of the total fixed assets of the owning utility, and the removal of the reactor from the rate base would have to be accomplished in a manner that would not bankrupt one or all participants. The ensuing public hearing at the state PUC would open the transaction up to public comment, and opponents of the move could make it politically impossible for the PUC to authorize the transaction. Participating munis or co-ops may find that the transfer of the asset could not be done under current tax laws without impairing their ability to issue tax-free bonds, unless the transaction is done for "value", which might be prohibitively expensive for the Government. Contractors for wholesale power may have provisions giving them the right of consent or to withdraw, which would lead to a reconfiguration of the electric supply scenarios for the utility in question, leading to additional hearings. Therefore, the cost to the Government of indemnifying all parties involved, the political problems in

PLUTONIUM DISPOSITION IN EXISTING REACTORS OPERATIONS

dealing with hearings and the many uncertainties all need to be taken into consideration in the approach and the infrastructure.

- **Government Leases the Reactor Facility**

The Government may choose to lease the reactor facility for a public purpose, in an attempt to mitigate the risks mentioned in the above paragraph. Nevertheless, even leasing the plant would likely set off the same hearing rights as previously described, and the same liabilities would have to be dealt with, even though they may not fall directly on the Government. The cost of indemnifying the utility owners would remain as a significant consideration in taking this step.

- **Government Owns the Fuel Facility**

The fuel facility can be treated as in the Evolutionary Reactor Report. Regardless of who owns and operates it, the job of producing the required amount of fresh fuel on schedule to meet the refueling requirements remains, and it is not subject to the complications of taking over ownership, approval by many affected parties including state regulators, etc., as it is being constructed as a new facility.

- **Regulatory Interfaces**

In the case of an existing reactor put to the mission of dispositioning weapons-grade plutonium, an NRC license change would be required, and the transaction would trigger an adjudicatory hearing at the NRC. This hearing would be contested by intervenors, and maybe by those concerned with the government's role in taking over a reactor under the terms of the antitrust provisions in the license. An environmental assessment, and maybe a full Environmental Impact Statement, would be needed. Since the reactor and its license already exist, and the reactor is located not on Government property (in the case of ABB-CE reactors of interest), the regulatory interfaces are likely to be of great concern, in that they would open the door for other upgrades and requirements not originally imposed on the plant.

4.2 Checkout and Startup Tests

Reference is made to the Evolutionary Reactor Report. In Section 4.2 of that report, the testing changes necessary for the initial testing requirements for MOX fuel are detailed. Those items identified as having impact due to the MOX fuel would be applicable to the initial startup of MOX cores with an existing reactor. These items discussed in the reference include: fuel storage and handling; Boron management; core neutronics; MOX fuel facility.

PLUTONIUM DISPOSITION IN EXISTING REACTORS OPERATIONS

In addition, normal startup testing would be required for any equipment that is added or changed before starting the plutonium disposition mission. These activities would be defined by the provider of the new equipment or processes. These would be no different than the testing required, for example, of a new tank or of a replacement steam generator.

4.3 Startup and Initial Operation

Reference is made to the Evolutionary Reactor Report. In Section 4.3 of that report, the testing changes necessary for the startup and initial operation for MOX fuel are detailed. Those items identified as having impact due to the MOX fuel would be applicable to the startup and initial operation of MOX cores with an existing reactor. These items discussed in the reference include: fuel storage and handling; boron management; core physics; MOX fuel facility.

In addition, care should be given to the modification of any operating procedures that would be required for any equipment that is added or changed before starting the plutonium disposition mission. These activities would be defined by the provider of the new equipment or processes. These would be no different than the modification of any operating procedures, for example, of a new tank or of a replacement steam generator.

4.4 Related Operational Experience on Availability and Capacity Factor

Existing reactors for this study have been selected based upon two criteria - more than 20 years left on their operating license and the high power density. Units not meeting these criteria would process small quantities of weapons grade plutonium in the remaining time available. The multi-unit site or utility was seen as an advantage by limiting the number of sites/utilities to perform the mission.

A recent Nuclear News article, (E. Michael Blake, "U. S. Capacity factors: Improvement even at the bottom", pp 36-44, May 1994) gives information on three year average capacity factors for US utilities. The three year average tends to smooth out the variations caused by various cycle lengths and refueling strategies. The standard suggested by Blake is that those units that have a Capacity Factor (CF) greater than 70% are considered "good" and those with a CF of less than 50% are considered "poor".

for all ABB CE units for this time frame is 78.1%.)

Table 4.4-2 shows the change in CF from the previous reporting period (88-90) to the present (91-93). The relatively young Palo verde units (less than 10, years of operation) show dramatic improvement.

PLUTONIUM DISPOSITION IN EXISTING REACTORS OPERATIONS

When looking at multi-unit (reactors at more than one site) experience in Table 4.4-3 (4.4-4), it is there is no obvious difficulty at getting "good" performance from the utilities that presently are operating the System 80 and 3410 class plants.

Based on the improvement trends in capacity factors, we assume that the utilities who choose to participate in the plutonium disposition mission will be able to achieve and 82% CF. Entergy is quoted by Nuclear News as having a "goal ..by next year all of its power plants will rank in the top quartile in capacity factor, regulatory performance, and cost performance."

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
OPERATIONS**

TABLE 4.4-1

1990-1993 SINGLE UNIT RANKINGS
NET DER* CAPACITY FACTOR

Source: Nuclear News May 1994

Rank	Unit	CF	MWe
20	Waterford 3	82.77	1104
33	San Onofre 3	79.69	1080
37	ANO 2	79.06	912
38	San Onofre 2	78.88	1070
49	Palo Verde 3	75.73	1270
60	Palo Verde 1	71.68	1270
67	Palo Verde 2	70.33	1270
Average		76.88	

* Design Electrical Rating

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
OPERATIONS**

TABLE 4.4-2
CAPACITY FACTOR CHANGE
1991-93 VS 1988-90

Source: Nuclear News May 1994

Rank	Unit	CF 88-90	CF 91-93	Change
7	Palo Verde 1	39.47	71.68	32.21
10 *	Palo Verde 3	48.18	75.73	27.55
19	Palo Verde 2	52.95	70.33	17.38
35	ANO-2	73.14	79.06	5.92
39	Waterford 3	78.34	82.77	4.43
41	San Onofre 3	75.88	79.69	3.81
67	San Onofre 2	80.06	78.88	-1.18

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
OPERATIONS**

TABLE 4.4-3

MULTI-UNIT SITE RANKINGS
1991-93 VS 1988-90

Source: Nuclear News May 1994

Rank	Plant (Units)	CF 91-93	CF 88-90	Change
6	St. Lucie (2)	82.3	82.1	0.2
9	ANO (2)	80.8	62.6	18.2
11	San Onofre (2)	79.3	77.9	1.4
19	Palo Verde (3)	72.6	46.2	26.4
20	Calvert Cliffs (2)	72.2	35.8	36.4
31	Millstone (3)	55.9	76.7	-20.8

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
OPERATIONS**

TABLE 4.4-4

**UTILITIES WITH REACTORS
AT MORE THAN ONE SITE**

Source: Nuclear News May 1994

Rank	Utility (#Units,#sites)	CF	CF	
		91-93	88-90 Change	
3	Entergy Operations (4,3)	80.1	70.7	9.4
9	Florida Power & Light(4,2)	70.5	69.9	-0.6
11	Northeast Utilities (5,3)	64.2	71.7	-7.5

PLUTONIUM DISPOSITION IN EXISTING REACTORS REACTOR COMPLEX SAFETY AND LICENSING

5.0 REACTOR COMPLEX SAFETY AND LICENSING

5.1 Reactor Transient and Accident Analysis

5.1.1 Bases for Analyses

5.1.1.1 Event Categorization

The bases for the transient and accident evaluations for System 80 with the plutonium core are the analyses for the Design Basis Events (DBEs) as reported in Chapters 6 and 15 of the Updated Palo Verde FSAR. These same analytical bases, methods and acceptance criteria are applied to the plutonium core.

The Design Basis Events are grouped into categories and the worst events from each category are reported in the FSAR. Since the response of the Palo Verde plant with a plutonium core is similar to the response with a conventional enriched uranium core, the same categories and limiting events are evaluated here. The categories are:

- Design Basis Events
 - Increase in secondary heat removal
 - Decrease in secondary heat removal
 - Decrease in reactor coolant flow
 - Reactivity and power distribution anomalies
 - Increase in RCS inventory
 - Decrease in RCS inventory
 - Radioactive release from a subsystem
- Severe Accident Consideration

Within each event category, the severity of an event is determined in part by the assumptions postulated for the event scenario. The likelihood of the postulated assumptions occurring determines the frequency group for an event scenario. The frequency groups and the general assumptions applied to events in all the event categories are as follows:

- a. Moderate frequency events are expected to occur one or more times in a reactor year. Analytical assumptions for evaluating moderate frequency events include:
 - adverse initial conditions
 - worst stuck CEA
 - only safety grade systems are credited
 - no operator action for 30 minutes
 - loss of off-site power

- b. Infrequent events are expected to occur one or more times in the lifetime of the plant. Analytical assumptions include those for moderate frequency events and in addition include:
 - worst single failure of a safety system (or control system if worse)
- c. Accident events are not expected to occur but are postulated in order to judge the adequacy of protection systems. Analytical assumptions for accidents include those for both moderate and infrequent events.

This grouping of event severity and frequency is directly applicable to Palo Verde with a plutonium core. The general intent of the assumptions to assure conservatively biased analytical results is satisfied, although the magnitude of the plant response to a given assumption may differ. Detailed differences that may be expected are evaluated in the following sections for specific events.

5.1.1.2 Acceptance Criteria

Acceptance criteria for plant response to design basis events differ among the event categories depending on the nature of the event and differ within a category depending on the event frequency group. The acceptance criteria are specified in the Standard Review Plan, NUREG -0800, for each category or event. They include limits in three areas:

1. Fuel integrity criteria prevent release of fission products from the fuel or limit the extent of fuel damage during accidents.
2. Pressure criteria prevent over-pressure in the primary system, secondary system and containment.
3. Dose criteria limit the exposure of the public.

The applicability of these criteria to the Palo Verde plant with a plutonium core is discussed in the following.

5.1.1.2.1 Fuel Acceptance Criteria

Fuel integrity acceptance criteria refer to the integrity of the fuel cladding to contain fission products. It is quantified for most events by the value of the DNBR which is a function of the geometric configuration of the fuel rod and fuel assembly, the heat flux distribution and the coolant temperature, pressure and mass velocity. For the plutonium core, all these characteristics are essentially unchanged or, in the case of heat flux or power distribution, can be explicitly specified. In application of the DNBR limiting value to the Palo Verde safety analyses reported in the updated FSAR, the uncertainties in fuel parameters that influence the DNBR are statistically combined.

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These uncertainties include experimental statistics related to the CE-1 critical heat flux empirical correlation and also statistics related to the physical characteristics of the fuel rod and fuel assembly. Fuel statistical parameters include fuel pellet diameter, density and enrichment (Pu content), clad diameter, and rod spacing and bow within the fuel assembly. For a developed fuel manufacturing process, historical inspection data provide the variations of these parameters and analytical advantage may be taken when variations are reliably smaller than manufacturing drawing tolerances would otherwise allow. Statistical fuel evaluations result in a value of 1.24 for the DNBR Specified Acceptable Fuel Design Limit (SAFDL) including the effect of statistical variations. A new manufacturing process and/or facility for plutonium fuel fabrication may require reevaluation of the fuel statistics and an adjustment of the minimum DNBR criterion. The possibility of a change in DNBR margin available during transient events is acknowledged for new plutonium fuel, but is probably small and is typically overcome with small changes in the Limiting Conditions for Operation (LCO) maintained using the Core Operating Limit Supervisory System (COLSS). Therefore, the value of the DNBR SAFDL used in the Palo Verde updated FSAR is assumed in these preliminary plutonium core safety evaluations.

Another SAFDL applies to the maximum fuel rod linear heat rating that would assure no centerline melt. The value in the updated FSAR is 21 Kw/ft. It is a function mostly of the pellet thermal conductivity which is dependent on burnup, but is also influenced by the fuel rod gap conductance and statistical factors mentioned above. For preliminary evaluations of the plutonium core, the FSAR value is assumed.

5.1.1.2.2 Plant Acceptance Criteria

Pressure acceptance criteria for the primary and secondary systems and the containment are specified in the Standard Review Plan and the ASME B&PV code. The limit values are unchanged for a plant with a plutonium core.

5.1.1.2.3 Dose Acceptance Criteria

Dose acceptance criteria are established by 10 CFR 100 and by the Standard Review Plan for each event category and/or event frequency. The limiting values of dose in the acceptance criteria are unchanged for evaluations of the plutonium core.

Dose values that are compared to the acceptance criteria are calculated for a given event and are a function of the characteristics of the dose source term and of the release pathway. The release pathways (e.g., steam safety valves, atmospheric dump valves, containment leakage, etc.) are determined by the overall plant and component design configuration and are unchanged with the plutonium core, although the magnitude of fluid release will change with changes in transient response.

The dose source term for non-LOCA events is dependent on the core characteristics that determine the isotopic spectrum of fission products that may be released from the fuel following postulated failure of fuel cladding or is established by Technical

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Specifications on allowable concentrations in the primary and secondary coolant during normal operation. The specifications that influence the dose calculated following events where there is no calculated fuel failure are the maximum iodine concentration (typically specified as micro-curies per gram equivalent I^{131}) and the maximum total concentration of other radioactive isotopes in the coolant. These dose limit criteria are the same for uranium or plutonium cores since they are quantitative in radioactivity, independent of the type of core.

When fuel cladding is calculated to fail by DNB, the radioactivity in the gas gap is assumed released into the RCS coolant. The release is typically specified as a fraction of the fuel rod inventory of iodine and noble gases. Release fractions used in the safety analyses are specified by the Standard Review Plan and depend on the diffusion characteristics of the ceramic fuel pellet. Fission product inventory is calculated for each fissile loading and fuel cycle and differs for plutonium and uranium. The combination of different isotopic spectrum and possibly different release fractions may change the dose for events where fuel failure is calculated.

LOCA dose is also determined for the release pathways (that are unchanged from the FSAR) and the assumed releases from the core. For the LOCA analyses, the radioactive release from the core is uncoupled from the event analysis. The U.S. NRC specifies the release fractions and the physical form for selected isotopes. The applicability of the NRC LOCA source term for the plutonium core has to be confirmed and the doses adjusted for the plutonium fission product inventory.

5.1.1.3 Methods and Computer Codes for Safety Analysis

The methods and computer codes employed in the safety analyses may be coarsely divided into three areas - nuclear core design, fuel rod performance and thermal hydraulic fluid systems performance. The FSAR provides descriptions of the various codes employed in these areas and the codes are summarized elsewhere in this report.

Generally, the physical phenomena modeled by these analytical tools are the same for the uranium core evaluated in the FSAR and for the plutonium core. In application, input parameters to the codes will accommodate differences for a plutonium core, although some confirmation of the analytical models for use with plutonium may be required. For example, fuel pellet thermal expansion, swelling, cracking and fission gas release models should be confirmed for the plutonium loading. A significant data base on MOX fuel is available, and the carrying out of the MOX Fuel Qualification Program should provide confirming information. These fuel characteristics may affect the DNBR and fluid conditions during events. On the other hand, the thermal hydraulic systems performance codes should be directly applicable for plutonium cores. Potential sensitivity of safety analysis results to code models for plutonium are identified in the following event evaluations where significant.

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5.1.1.4 Initial Conditions for Safety Analyses

Events evaluated here, and in Chapter 15 of the updated FSAR are analyzed over a range of initial values of the principal process variables. Analysis over a range of initial conditions is consistent with the monitoring function performed by the Core Operating Limit Supervisory System (COLSS) and the flexibility of plant operation which COLSS allows. This flexibility is produced by allowing parameter trade-offs by monitoring the principal process variables, synthesizing the margin to fuel thermal design limits and displaying to the reactor operator the core power operating limit. The range of values of each of the principal process variables considered in the event analyses is listed in Table 5.1-1. Flexibility afforded by the COLSS and margin inherent in the Palo Verde design may allow changes to the initial conditions to prevent event results from violating acceptance criteria.

5.1.1.5 Input Parameter Values for Safety Analyses

The parameters described here are consistent with the initial conditions for the process variables in Table 5.1-1 and represent the plutonium core. Values of these parameters are particularly sensitive to the core composition. Comparisons are given in the following with the parameter values calculated in the same manner for an all uranium core. The safety analyses typically employ bounding values of parameters, so that even if the trend and values provided in Table 5.1-2 differ for the uranium and plutonium cores, analysis results obtained with the bounding valves may not change. Section 2.6 provides discussions of MOX core safety parameters of interest.

5.1.1.5.1 Fuel Temperature Coefficient

Depending on the particular event, the most limiting value of the fuel temperature coefficient could be either the most negative or the least negative limit over a range of time in life, core conditions, conservative bias and calculational uncertainty. Events with a calculated power increase are analyzed with the least negative fuel temperature coefficient to conservatively maximize the power peak, while events with a calculated power decrease are analyzed with the most negative coefficient. Ranges calculated over core lifetime are given in Table 5.1-2. Values of fuel temperature coefficient for the plutonium core are slightly less negative than equivalent values for an all uranium core.

5.1.1.5.2 Moderator Temperature Coefficient

The moderator temperature coefficient is the net effect from the combination of the effect of temperature on the moderator density and the moderator nuclear temperature coefficient. Although the moderator density contribution is always positive in the operating range, the magnitude decreases as soluble boron level is increased, so the density contribution to moderator temperature coefficient might be expected to be less at the higher boron concentrations in the plutonium core.

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The moderator nuclear temperature coefficient is the change in reactivity per unit change in core average temperature at constant moderator density. This nuclear reactivity effect is dependent on the spectral effects associated with the change in thermal scattering properties of water with temperature. The nuclear coefficient is also influenced by the changing neutron spectrum with fuel isotopic changes from beginning to end of cycle. For each event analysis, suitably conservative values are used representing the most limiting combinations of core conditions. Table 5.1-2 gives comparable values of the combined "Moderator Temperature Coefficient" for the plutonium core and an all uranium core.

5.1.1.5.3 Reactivity Control

Means for reactivity control include fixed burnable poison (erbia in fuel rods), soluble poison (boron) and control rods (Control Element Assemblies, CEAs). The reactivity effect on the plant response to design basis events will differ with the plutonium core. The increased inverse boron worth with plutonium that is indicated by Table 5.1-2 can be partially offset in its influence on safety response by employing enriched boron, so that a given volume of coolant carries equivalent reactivity worth.

Increasing the number of CEAs maintains rod worth in the plutonium core for events requiring reactor trip, while the individual rod worth is less, benefiting the rod transient events in both reduced total reactivity increment and in reduced local peaking.

The dynamic effects on reactivity are aggravated in the plutonium core by the smaller values for the effective delayed neutron fraction and prompt neutron lifetime. While the decreases would appear to have an adverse effect upon short period transients such as a rod ejection, the overall consequence is mitigated by the lowered reactivity worth of the ejected rod and the reduced sensitivity of the core power distribution to the local reactivity perturbations. These mitigating effects on the plant response to transients are a consequence of the strong thermal absorption properties which reduce the thermal diffusion length of the plutonium fuel lattice.

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TABLE 5.1-1

RANGES OF INITIAL CONDITIONS FOR SAFETY ANALYSES

<u>Parameter</u>	<u>Units</u>	<u>Range</u>
Core Power	% of 3800 Mwt	0 - 102
Axial Shape Index 0.3 ⁽¹⁾	--	$-0.3 \leq ASI \leq + 0.3$
Reactor Vessel Inlet Coolant Flow Rate	% of 445600 gpm	95 - 116
Pressurizer Water Level	% distance between upper tap and lower tap above lower tap	26 to 60
Core Inlet Coolant Temperature		
< 90% Power	°F	550 - 572
90%-100% Power	°F	560 - 570
Pressurizer Pressure	psia	2000 - 2325
Steam Generator Water Level	% distance between upper tap and lower tap above lower tap	40 to 88
Maximum Initial Linear Heat Rate	Kw/ft	13.5

(1) ASI =
$$\frac{\text{area under axial shape in lower half of core} - \text{area under axial shape in upper half of core}}{\text{total area under axial shape}}$$

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TABLE 5.1-2

**PARAMETERS FOR PALO VERDE SAFETY ANALYSES
URANIUM VERSUS PLUTONIUM CORES**

Parameter Units	UO ₂ Core Eq Cycle	Weapons Grade Pu Core	
		Eq Cycle	1st Cycle
Moderator Temperature Coef (10 ⁻⁴ Δρ/°C)			
Full Power BOL	-0.59	-2.50	-1.72
Full Power EOL	-3.24	-4.30	-4.33
Fuel Temperature Coef (10 ⁻⁵ Δρ/°C)			
Full Power BOL	-1.24	-1.61	-1.61
Full Power EOL	-1.25	-1.40	-0.94
Critical Boron Conc (BOC, unrodded ppm)			
Hot Standby	1589	2783	3073
Full Pwr, No Xe	1400	2238	2631
Full Pwr, Eq Xe	1170	1773	2170
Inverse Boron Worth (ppm/%Δρ)			
Full Power BOL	116.0	277	238
Full Power EOL	101.0	220	169
CEA Worth Full Power EOL (%Δρ)			
Total	13.80	14.2	15.29
Net (with WSR)	10.20	11.71	11.85
Worst Stuck Rod	3.80	2.49	3.44
Number of CEA Drives	89	97	97
Eff Delayed Neut Frac			
Full Power BOL	0.00625	0.00339	--
Full Power EOL	0.00546	0.00366	--
Prompt Neutron Lifetime (10 ⁻⁶ seconds)			
Full Power BOL	21.3	8.92	10.40
Full Power EOL	24.8	11.38	14.80

TABLE 5.1-3
DESIGN BASIS EVENTS FOR SAFETY ANALYSES

Event Frequency of Occur.	Inc. Sec. Sys. Heat Removal	Dec. Sec. Sys. Heat Removal	Dec. RCS Flow	Reactivity & Power Dist. Anomalies	Inc. RCS Inventory	Dec. RCS Inventory	Rad. Release from Subsys or Comp.
Moderate Frequency Events	<ul style="list-style-type: none"> · Inc. Feed. Flow · Dec. Feed. Temp. · Inc. Main Steam Flow · Inadvert. ADV Opening 	<ul style="list-style-type: none"> · Loss of External Load · Turbine Trip · Loss of Cond. Vac. · MSIV Closure · Loss of Normal Feed. Flow · Loss of Non-Emerg AC Power to Station Auxliaries 	<ul style="list-style-type: none"> · Total Loss of RCS Flow 	<ul style="list-style-type: none"> · Uncontrolle d CEA Withdrawal from Subcrit or low power · Uncontrolle d CEA Withdrawal at power · Single CEA drop · Inadvertant Deboration · Startup Inactive RCP 	<ul style="list-style-type: none"> · Inadvertant Oper. of ECCS · CVCS Malfunction-PLCS w/LOOP 	None	None

Infrequent Events	Same as Mod. Freq. but with a single failure	Same as Mod. Freq. but with a single failure	Same as Mod. Freq. but with a single failure	<ul style="list-style-type: none"> Inadvertant Loading of Fuel Ass. into Improper position Inadvertant Deboration with a Single Failure Startup of Inactive RCP with a Single Failure 	Same as Mod. Freq. but with a single failure	None	<ul style="list-style-type: none"> Fuel Handling Accident Spent Fuel Cask Drop Accidents
Accidents	<ul style="list-style-type: none"> Steam Sys. Piping failures Inside & Outside Containment 	<ul style="list-style-type: none"> Feed. Sys. Pipe Breaks 	<ul style="list-style-type: none"> Single RCP Rotor Seizure w/LOOP (RS) RCP Shaft Break w/LOOP (SB) 	<ul style="list-style-type: none"> CEA Ejection 	None	<ul style="list-style-type: none"> LOCAs DE Break of Letdown Line Outside Cont. SGTR SGTR w/LOOP SGTR w/LOOP & stuck open ADV 	<ul style="list-style-type: none"> Postulated Rad. Releases due to liquid tank failures

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TABLE 5.1-4
SINGLE FAILURES ASSUMED FOR SAFETY ANALYSES

A. STEAM BYPASS CONTROL SYSTEM

1. Failure to Modulate Open
2. Failure to Quick Open
3. One Bypass Valve Fails to Quick Close
4. Excessive Steam Bypass Flow
5. Failure to Generate Automatic Withdrawal Prohibit Signal During Steam Bypass Operation
6. Failure to Generate the Reactor Power Cutback Signal

B. REACTIVITY CONTROL SYSTEMS

7. Regulating Group(s) Fail(s) to Insert or Withdraw
8. A Single CEA Stuck*
9. A CEA Subgroup Stuck*
10. Failure to Initiate or Execute the Reactor Power Cutback
11. CEAs Withdraw upon Automatic Withdrawal Prohibit and/or CEA Withdrawal Prohibit

C. FEEDWATER CONTROL SYSTEM

12. Failure of Reactor Trip Override
13. Failure of High Level Override

D. TURBINE-GENERATOR CONTROL SYSTEM

-
-
15. Failure to Modulate the Turbine Control Valves
 16. Failure to Setback Given a Cutback
(100% $\geq$ Initial Power $\geq$ 75%)
 17. Failure to Setback
(75% $\geq$ Initial Power $\geq$ 60%)
 18. Failure to Runback
(60% $>$ Initial Power)
 19. Failure to Trip the Turbine

* Control Element Drive Mechanism does not respond to control signal.
Release of CEA(s) on trip is not inhibited.

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TABLES 5.1-4 (Cont'd)
SINGLE FAILURES ASSUMED FOR SAFETY ANALYSES

- E. PRESSURIZER PRESSURE CONTROL SYSTEM (PPCS)
 - 20. Failure of Spray Control Valves to Open
 - 21. Failure of Spray Control Valves to Close
 - 22. Failure of Backup Heaters to Turn On
 - 23. Failure of Backup Heaters to Turn Off
- F. PRESSURIZER LEVEL CONTROL SYSTEM
 - 24. Backup charging Pump Fails To Turn On
 - 25. Backup charging Pump Fails To Turn Off
 - 26. Letdown Flow Control Valve Fails to Close
 - 27. Letdown Flow Control Valve Fails to Open
- G. MAIN FEEDWATER SYSTEM
 - 28. One MFIV Fails to Close
 - 29. One Back-flow Check Valve Fails to Close
- H. MAIN STEAM SYSTEM
 - 30. One MSIV Fails to Close
 - 31. One Atmospheric Dump Valve Fails to Open
 - 32. One Atmospheric Dump Valve Fails to Reclose
- I. AUXILIARY FEEDWATER SYSTEM
 - 33. Failure of Any One Auxiliary Feed Pump to Start
- J. EMERGENCY CORE COOLING SYSTEM
 - 34. Failure of One HPSI or LPSI Pump
- K. ELECTRICAL POWER SOURCES
 - 35. Loss of Offsite Power After Turbine Trip
 - 36. Failure of One Emergency Generator to Start, Run, or Load
 - 37. Failure of One Breaker to Achieve Fast Transfer to Backup Power Supply

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5.1.2 Event Analyses

5.1.2.1 The Fault Schedule

Categories of events that are analyzed in the plant safety analyses are identified in Section 5.1.1 and a summary listing of the categories, events and event frequency are given in Table 5.1-3. In this section, the event from each category and frequency that most closely challenges the applicable acceptance criteria is evaluated. Analytical results reported in Chapters 6 and 15 are the reference point for these evaluations. The objective is to estimate if, when the initial conditions and plant parameters identified in Section 5.1.1 for the plutonium core are substituted for the Palo Verde parameters for a conventional uranium core, the consequences of the postulated events satisfy the acceptance criteria.

These evaluations employ the same perspective as generally used to establish bounding conditions and methods that assure the physical consequences of the postulated event are no worse than the analytical results. In other words, the direction of trends for without performing analyses on the plutonium core at this time. Similarly, the direction of trends can identify the need for specific event analyses with the plutonium core.

To assure completeness of the evaluation process, a comprehensive listing is made of the event categories, the event frequency group and the individual events considered in each. Table 5.1-3 lists the events that were considered in the Palo Verde design evaluations reported in the updated FSAR, Chapters 6 and 15. Analyses are reported for the most limiting event from each category and frequency group. These same events are evaluated in the following sections.

The evaluations consider the effect of plutonium on the plant transients and the response of available protection, detection and mitigation systems. Plant response is considered according to the Safety Functions that are exercised during each event and the systems and actuation parameters available to initiate and provide each safety function. Only safety grade systems are assumed to mitigate the events for design basis events, and the worst single failure of those systems (or a non-safety grade system if it yields worse results) may be assumed. Table 5.1-4 lists the single failures from which the worst is selected for each event.

Detection of abnormal events and actuation of systems is dependent on the instrumentation response. Uncertainties are assumed in the response of protection and safety systems. Uncertainties account for normal instrument behavior, response in a harsh environment and the effect on response of the particular event, for example, a particularly rapid transient effect on the indicated value. There are no identified effects of the plutonium core relative to an all uranium core for the instrument responses that are required to indicate or mitigate the Design Basis Events.

fault category. The steam piping ruptures are classified as limiting faults.

Detection of events within this group is by RCS and Steam Generator low pressure alarms, steam generator water level alarms, and the high power alarm. Trips include the core protection calculator trips on low DNBR or high local power density, low steam generator pressure, variable overpower, steam generator water level, manual, low RCS pressure, and high containment pressure.

The events within this group are characterized by decreasing primary and secondary pressure and in the case of steam line break, increasing containment pressure. Subsequent to reactor trip there is the potential for actuating the main steam safety valves (unaffected steam generator), main steam and feedwater isolation systems, auxiliary feedwater system, safety injection system, containment isolation system, containment spray system and the atmospheric dump valve system. Eventual operator assisted cooldown following the initiating fault will lead to operator initiation of shutdown cooling. Once in shutdown cooling, the plant can be maintained in a stable condition indefinitely.

The limiting moderate frequency and infrequent events for this category are the Inadvertent Opening of a Steam generator Atmospheric Dump Valve (IOSGADV) and the same event combined with a Single Failure (SF). The major parameter of concern is the minimum hot channel DNBR. This parameter establishes whether a fuel design limit has been violated and thus whether fuel cladding degradation might be anticipated. Those factors which cause a decrease in local DNBR are:

- (a) Increasing coolant temperature
- (b) Decreasing coolant pressure
- (c) Increasing local heat flux (including radial and axial power distribution effects)
- (d) Decreasing coolant flow

necessary, without violating the minimum DNBR criterion for the LOSSADV event without a single failure. When the LOP on turbine trip is included as the worst single failure, the DNBR decreases after reactor trip, resulting in fuel failure in Palo Verde of less than 1.7%. The value of DNBR at trip is determined by the CPC trip setpoint, while the DNBR after trip is mostly determined by the rate of flow coastdown and by the rate of reactivity insertion by the CEAs. Higher CEA worth with plutonium may benefit this event, while the more negative MTC will aggravate results. Narrowing the allowable range of initial conditions or using the margin between the calculated radiological dose and the acceptance criteria and tolerating the potential increase in fuel failures in the plutonium core are ways to accommodate the potential margin decrease.

In other words, the consequences of these moderate frequency and infrequent events can be made acceptable by adjusting the operating parameter space if it becomes necessary.

The limiting accident in the category of increased heat removal events is the Steam Line Break (SLB). Typically, a series of postulated conditions are analyzed in order to determine the worst initial conditions. One set of initial conditions is selected to maximize the post-trip return to power. In the Palo Verde FSAR analyses, post-trip return to power occurs but no fuel failure occurs. The second set of initial conditions is selected to maximize pre-trip potential for fuel failure by DNB. The Palo Verde analyses reported fuel failure for these conditions, amounting to 0.7 percent failed fuel and offsite radiological dose within the 10 CFR 100 acceptance criterion.

The important plant characteristics for the SLB analyses on the plutonium core are the most negative values for moderator and fuel temperature coefficients, the net control rod worth with one stuck rod, the consequent post-trip three-dimensional core peaking factor and the negative reactivity worth of the boric acid safety injection flow. Counteracting core reactivity effects and interacting plant fluid dynamics make a conclusive statement on the direction of the net result of the SLB analyses on a plutonium core difficult without specific analyses.

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In conclusion, the predicted trend for the consequences from a SLB accident with the plutonium core relative to the consequences reported in the Palo Verde FSAR is uncertain at this time.

5.1.2.1.2 Decrease in Heat Removal by the Secondary System Events

The events occurring in this category are characterized by increasing RCS temperature and pressure. The increasing pressure during these events results in a DNBR which also tends to increase. Feedwater line breaks are included in this group. These faults result in consequences which approach the safety limit on RCS pressure of 186.6 bar (2750 psia).

The causes of events within this category include malfunctions in the turbine/generator system, condenser, main steam isolation system, offsite power system, and feedwater system. The feedwater system malfunctions include pipe breaks. Condenser malfunctions occur due to failure of the circulating water system to supply cooling water, failure of the main condenser evacuation system to remove non-condensable gases, or excessive in-leakage of air to the condenser. Malfunctions of the main steam isolation system result from a spurious closure signal. The malfunctions in the offsite power system result from grid instabilities or electrical faults resulting in grid collapse. The feedwater system malfunctions result from loss of the main feedwater pumps or a spurious signal generated by the feedwater control system which results in closure of the feedwater control valves.

Events occurring within this category are classified as either moderate frequency, infrequent, or limiting faults. The turbine/generator, condenser, and main steam isolation system malfunctions are moderate frequency faults. However, when coupled with an additional failure, these faults fall into the infrequent fault category. Feedwater system malfunctions also fall into the above two categories, although pipe breaks are classified as limiting faults.

Detection of events within this group is by the RCS pressure high alarm. Reactor trip will occur on high pressurizer pressure, CPC low DNBR, steam generator water level low and/or high containment pressure.

The events within this group are characterized by increasing primary and secondary pressure and containment pressure for the case of pipe breaks inside containment. Subsequent to reactor trip, there is the potential for lifting the pressurizer and main steam safety valves (unaffected steam generator) and actuating the main steam and feedwater isolation systems, auxiliary feedwater system, containment isolation system, containment spray system and the atmospheric dump valve system. Eventual operator assisted cooldown following the initiating fault will lead to operator initiation of shutdown cooling. Once in shutdown cooling, the plant can be maintained in a stable condition indefinitely.

sequence of events culminates with a reactor trip on high pressurizer pressure, low steam generator water level or high containment pressure. RCS heatup can continue after trip due to a total loss of heat transfer in the ruptured steam generator as it empties. Eventually the decreasing core power following reactor trip reduces the core heat rate to the heat removal capacity of the intact steam generator.

Because of the various interacting physical phenomena, a number of limit conditions are analyzed for this event. The significant assumptions are break size, loss of offsite power or not, and conditions to minimize DNBR. In these analyses, the core characteristics for fuel and moderator temperature coefficients and fuel rod heat transfer are appropriately biased to maximize RCS pressure, secondary pressure or radiological dose. For the maximum pressure analyses there is no significant core power transient, so the fluid system responses are the dominant features that establish acceptable consequences. Decay heat influences the longer time results for the

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radiological dose from secondary fluid releases. The dose is calculated for the steam releases in the maximum pressure analyses, for which no failed fuel is calculated. The 2 hour thyroid dose at the EAB is 9.5 Rem, which indicates margin to the acceptance criterion.

In conclusion, the plutonium core characteristics might influence the DNBR results for the feedwater line break event, but might not increase its dose above the dose with the high pressure analyses. Specific analyses with plutonium core characteristics will confirm the worst situation.

5.1.2.1.3 Decrease in Reactor Coolant Flow Rate Events

The events occurring in this event category are initiated by a decrease in reactor coolant flow. The flow decrease results in increasing RCS temperature and pressure with subsequent degradation in DNBR. Upon reactor and turbine trip the secondary pressure increases. The events within this group result from a loss of offsite power or a mechanical defect in an individual RCP resulting in seizure or a break in the RCP shaft.

Events within this group are classified as either moderate frequency, infrequent or limiting faults. Faults initiated by a loss of offsite power are classified as a moderate frequency fault. When a single failure is included, the fault is classified as an infrequent fault. Faults initiated by RCP mechanical failures are classified as limiting faults.

Detection of events within this category is by alarms on high pressurizer pressure and low thermal margin. Within seconds of event initiation, reactor trips occur on CPC low DNBR or on low reactor coolant flow rate.

These events are characterized by decreasing reactor coolant flow, increasing RCS temperature and pressure, and increasing steam generator pressure in at least one steam generator. Subsequent to reactor trip, there is a potential for lifting the pressurizer and main steam safety valves and actuating the auxiliary feedwater system and the atmospheric dump valve system. Eventual operator assisted cooldown following the initiating fault will lead to operator initiation of shutdown cooling. Once in shutdown cooling, the plant can be maintained in a stable condition indefinitely.

A complete loss of forced reactor coolant flow will result from the simultaneous loss of electrical power to all reactor coolant pumps. The only credible failure which can result in a simultaneous loss of power is a complete loss of offsite power which is assumed to result in a turbine trip rendering the steam dump and bypass system function unavailable. For this event, the minimum DNBR is the major concern and it occurs within the first few seconds. There are no single failure assumptions that can significantly effect the DNBR on such a short time scale. Hence, this event scenario is the limiting moderate frequency and infrequent event.

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The consequences, i.e., minimum DNBR, are determined by the competition between the decreasing DNBR caused by decreasing flow and the increasing DNBR caused by decreasing heat flux after reactor trip. Decreasing flow is almost entirely a fluid system process related to primary system flow resistance and stored energy in the pump flywheel. These characteristics are unchanged with the plutonium core. Increasing DNBR is determined by the interrelationships between the local and core wide power decrease as the control rods drop into the core and the thermal response of the fuel rod and resultant surface heat flux. All of these characteristics may vary with the plutonium core.

Typically, bounding values of the reactivity temperature coefficients are employed in the analyses, i.e., the least negative moderator coefficient and Doppler coefficient.

pellet, gap and cladding thermal/mechanical behavior could affect the decreasing transient heat flux during this event, but they are believed small. The dominant mitigating effect is the negative reactivity transient by the CEAs dropping into the core. There are no differences in the rod motion transient and slightly greater net rod worth (see Table 5.1-2). In conclusion, there may be small differences in the results of the loss of flow transient with the plutonium core that should be overcome with specific analyses and/or by adjusting the initial DNBR margin that is monitored by the COLSS.

The most limiting accident in the flow decrease event category is the single reactor coolant pump rotor seizure combined with loss of offsite power. A single reactor coolant pump rotor seizure can be caused by seizure of the upper or lower thrust-journal bearings. Loss of offsite power subsequent to turbine generator trip may be caused by a complete loss of the external electrical grid triggered by the turbine generator trip. Loss of offsite power is delayed three seconds after turbine trip. The onsite loads will subsequently lose power and the plant will experience a simultaneous loss of feedwater flow, condenser inoperability, and a coastdown of all reactor coolant pumps, but this will not occur early enough on Palo Verde to influence the minimum DNBR.

The increasing temperature of the secondary system leads to a reduction of the primary to secondary heat transfer. Concurrently, the failed reactor coolant pump and the three reactor coolant pumps coasting down result in a lower RCS flow which further reduces the heat transfer capability of the RCS. This decrease in heat removal from the RCS leads to an increase in the core coolant temperatures, a corresponding reduction in the margin to DNB, and an increase in the primary system pressure, caused by the thermal expansion of the RCS fluid. The core coolant temperature peaks shortly after reactor trip. The increase in RCS temperature leads to an increase in RCS pressure, which reaches a maximum within 5 seconds. After this time, the RCS pressure decreases rapidly due to the declining core heat flux, in combination with the

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opening of the main steam safety valves. At 30 minutes, the operator is assumed to use the atmospheric dump valves to begin cooldown.

Analyses for the rotor seizure event determine the maximum offsite radiological dose resulting from fuel failure by DNB, primary to secondary leakage and eventually steam releases via the secondary safety and atmospheric dump valves. Analyses also determine the peak primary pressure. Initial conditions for the radiological dose analyses are selected to maximize core power, minimize DNBR and maximize secondary pressure. The fluid system parameters and plant response characteristics are unchanged for the plutonium core. The *radial peaking, reactivity coefficients and net rod worth* will vary with the plutonium core and will affect the minimum DNBR and amount of fuel failure predicted and may affect the power/pressure transient.

The differences related to the core reactivity and peaking transients are similar to those discussed above for the complete loss of flow events. If the net effect is unfavorable, restrictions on normal Limiting conditions for Operation (LCOs) that increase the initial core thermal margin may accommodate them. These differences affect the radiological dose predicted from fuel failures resulting from DNB. The Palo Verde FSAR reports 4.5% of the fuel in DNB for the rotor seizure event and a dose of 24.0 Rem with an acceptance criterion of 300 Rem.

Radiological dose from the activity initially in the primary coolant, from iodine spiking during the event and from activity initially in the secondary resulting from postulated steam generator tube leakage might not increase because of the conservatism in the analyses. The conservative analyses assume a stuck open atmospheric dump valve after operator action at 1800 seconds which causes a cooldown rate greater than the allowable rate and causes attainment of shutdown entry conditions sooner than if the operator limited the cooldown rate to the maximum allowable value. Entry into shutdown cooling removes the steam generator release pathway and stops offsite releases. To avoid this limit on release, the analyses conservatively assume that the radiological release is prolonged while the operator properly limits the cooldown rate to the allowable limit by manually cycling the atmospheric dump valves. Even if the primary and secondary pressure transients driving the steam release were aggravated by the plutonium core characteristics, the system related releases may not become worse than the releases calculated with the conservative analytical assumptions employed in the FSAR analyses. In conclusion, a specific analysis for the plutonium core would demonstrate any differences.

5.1.2.1.4 Reactivity and Power Distribution Anomalies Events

The events occurring in this fault group are characterized by either an increase in core power or a resultant perturbation in core power distribution. The events resulting in an increase in core power subsequently result in increases in primary temperatures and pressures and degradation in DNBR. The CEA ejection accident included within this

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group has the tendency to result in a core damage sequence due to the occurrence of DNB.

CEA withdrawal events are caused by a single failure in either the control element drive mechanism, control element drive mechanism control system, or reactor regulating system or by operator error. The drop of a single CEA is caused by holding coil failure or loss of power to a coil. Inadvertent deboration is caused by improper operator actions or a failure in the boric acid makeup flow path. The ejection of a CEA is caused by the rupture of a control element drive mechanism housing.

Events occurring within this group are classified as either moderate frequency, *infrequent or limiting events*. The faults within this group which are classified as infrequent faults are the inadvertent loading of a fuel assembly into the improper position and the inadvertent deboration and startup of an inactive RCP coupled with a single failure. The only fault within this group classified as a limiting fault is the CEA ejection.

Detection of events within this group is by a high average coolant temperature alarm or a high neutron flux alarm (Modes 3-6 only). Trips also provide a means for detection. Trips occurring during faults of this group are high logarithmic power level variable overpower, and CPC low DNBR.

For events within this group characterized by either an increase in core power or resultant perturbation in core power distribution, the resulting increase in pressure is accommodated by lifting the primary and secondary safety valves. For certain events initiated from shutdown modes when low temperature overpressure protection is in service, the shutdown cooling relief valves provide overpressure protection. The resulting responses may actuate the auxiliary feedwater system and require cooldown utilizing the atmospheric dump valve system. Eventual operator assisted cooldown following the initiating fault will lead to operator initiation of shutdown cooling. Once in shutdown cooling, the plant can be maintained in a stable condition indefinitely.

The sequential CEA withdrawal events from subcritical or low power and from full power are similar. They cause a positive reactivity transient that increases core power and that challenges the fuel limits for minimum DNBR and maximum fuel rod linear heat generation rate. The significant core parameter is the rate of reactivity addition as the regulating rod banks are withdrawn. Individual rod worth is less for the plutonium core, which will reduce the reactivity insertion rate, but the total sequential bank reactivity insertion rates are not yet known in detail for the plutonium core. The significant mitigating response is the reactor trip from the ex-core neutron detectors of the reactor protection system. The instrument response should be the same for the plutonium core.

Within the core, the event is influenced by the local power distribution and by the reactivity feedback from moderator and fuel temperature. Conservatively large peaking

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factors are typically assumed in analyses of this event, for event initiation from low power or from full power. Neutron slowing down and diffusion characteristics of the plutonium fuel lattice suggest that there may be less effect on the local power distribution from perturbations of the control rod patterns so transient peaking factors may not be adversely different from the values in the Palo Verde updated FSAR. Fuel and moderator reactivity temperature coefficients are taken at BOL when they are less negative and provide less favorable feedback to counteract the increasing core power. The Palo Verde updated FSAR analyses assumed $+0.5 \times 10^{-4} \Delta\rho/^\circ\text{F}$ for the moderator temperature coefficient. However, in the plutonium core, control rod worth, and presumably reactivity insertion rates, may be greater at EOL so it is not apparent which time in life will yield the more limiting consequences for this event. The FSAR analyses show no violation of the limiting values of DNBR and rod linear heat rate, but there is no margin available to offset potential unfavorable trends of the rod withdrawal event consequences in the plutonium core.

The single CEA drop event may not cause a reactor trip but will cause a decrease in DNBR. Negative reactivity inserted by the dropped CEA causes an initial power decrease that is overcome by negative moderator and fuel temperature coefficients. The power returns to its initial value while the core radial peak increases as a result of the perturbation from the dropped rod, causing an increase in local heat flux and a decrease in the DNBR.

The reactivity coefficients for fuel and moderator are both more negative at EOL in the uranium core. A bounding value of $-3.5 \times 10^{-4} \Delta\rho/^\circ\text{F}$ was assumed for the moderator temperature coefficient in the FSAR. Because of the neutron characteristics of the plutonium core as discussed above, the worth of the dropped rod and the perturbation in radial peak from the inserted rod are probably smaller in the plutonium core. Hence, consequences of the control rod drop event are probably not significantly different or worse in the plutonium core.

The startup of an inactive reactor coolant pump event is necessarily initiated from a shutdown mode, since all pumps are required for critical operation. The principal effect is a cold water surge that results in a positive reactivity insertion via the negative temperature coefficients. Values of the coefficients in the plutonium core may be somewhat worse for this event. The event pressure rise is accommodated by the safety valves and the minimum DNBR is not challenged. Hence, there is no significant difference in the consequences.

The inadvertent deboration event may be caused by equipment failure or by improper operator action that results in charging flow with boron concentration below the concentration in the reactor coolant. The limiting situation occurs during cold shutdown when the reactor coolant volume has been drained down. The design objective is to provide reliable automatic indication to the operator in time for action before the boron dilution leads to criticality.

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The indication in the shutdown modes is from the neutron flux alarm on the startup flux channel of the reactor protection system. The alarm occurs on the ratio of the signal value at any time to the reference value initially present. Hence, the response and sensitivity of the instrument is relative, so potential changes in instrument sensitivity with neutron spectra and/or normal boron concentrations in the plutonium core should not invalidate the effectiveness of the boron dilution alarm.

The rate of reactivity insertion by dilution determines the time available for operator action. Thirty minutes is the minimum allowable dilution interval following the alarm. The inverse boron worth is greater for the plutonium core, but the critical boron concentration is also greater. The appropriate comparison between cores is made for the reactivity change caused by a given fractional change in the RCS boron concentration. Current estimates (see Table 5.1-2) show that the reactivity change per fractional change in concentration as unborated water is added to the RCS is smaller for the plutonium core. Therefore, for a given charging flow rate of unborated water, there is a longer time interval available before reaching criticality. *This favorable conclusion must be verified when more nearly complete boron worth calculations are available.*

The CEA ejection event is the limiting accident in the reactivity anomaly event category. It results from a circumferential rupture of the control element drive mechanism housing of the CEDM nozzle. Ejection of the CEA causes the core power to increase rapidly due to the almost instantaneous addition of positive reactivity. Smaller rod worths in the plutonium core make the reactivity addition smaller. The rapid power increase is terminated by a combination of Doppler and delayed neutron effects. The Doppler coefficient can be either more or less negative for the plutonium core, depending on cycle and time in life, and both the delayed neutron fraction and the neutron lifetime are smaller, tending to increase the peak transient power and the rate of power rise. A reduced sensitivity of the core power distribution to local removal of the control rod reduces the increase in local peaking.

There are three acceptance criteria for the CEA ejection event. The first is the maximum primary pressure. It is calculated conservatively assuming there is no break in the primary pressure boundary. The peak pressure occurs within 5 seconds and is strongly dependent on the magnitude of the power peak. For the CEA ejection analysis reported in the FSAR, the calculated peak pressure is about 110% of design pressure. The allowable pressure in the SRP is higher, so there is margin to the specified limits for this event.

The second acceptance criterion is that the maximum radial average fuel enthalpy shall be less than 280 cal/gm to avoid fuel dispersion in the coolant. The calculated value is directly dependent on the magnitude of the power spike and the increase in local peaking. The net effect on these results is uncertain until specific analyses with specific values of core parameters are available. The limit value of 280 cal/gm for fuel pellets with all uranium is assumed applicable for pellets with plutonium and uranium.

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The third acceptance criterion is for radiological dose, which is the sum of two contributions. One contribution is the radioactivity in the coolant initially and that is released via secondary safety valves. This should not change significantly for plutonium since the initial concentration is assumed the same, at the Technical Specification operational limit. The other dose contributor is from activity in the fuel rod gas gap that is assumed to be released instantaneously when the Specified Acceptable Fuel Design Limit on DNBR is violated. The analyses in the Palo Verde updated FSAR Chapter 15 estimate 9.8% of the fuel rods experience DNB. Again, specific analyses with the plutonium core parameter values are needed to determine the net effect on local rod power and DNBR. Dose results reported have significant margin to the acceptable limits of 10CFR100.

In conclusion, specific analyses are required for the CEA ejection event. While the net plutonium effect could yield results that are worse than reported in the FSAR, there is margin available to the acceptance criteria.

5.1.2.1.5 Increase in RCS Inventory Events

The events occurring in this fault group are characterized by increasing RCS inventory with subsequent RCS pressurization. Due to the pressurization the DNBR increases. Causes include operator error, spurious signals causing inadvertent actuation of the safety injection system, and a failure in the pressurizer level control system resulting in excess charging flow to the RCS. These faults are categorized as moderate frequency or, when combined with an additional failure, as infrequent. Detection is by the pressurizer high level alarm or high pressurizer pressure alarms. The only trip required is the high pressurizer pressure trip.

The faults within this group are characterized by increasing RCS inventory and pressure and increasing steam generator pressure following trip. The pressurization of the RCS is accommodated by the *primary safety valves or the shutdown cooling relief valves* if low temperature overpressure protection is in service. The secondary pressurization is accommodated by the main steam safety valves. For certain events, actuation of the auxiliary feedwater and atmospheric dump valve system may be required.

Eventual operator assisted cooldown following the initiating fault will lead to operator initiation of shutdown cooling. Once in shutdown cooling, the plant can be maintained in a stable condition indefinitely.

The inadvertent operation of the ECCS event does not impact normal operation at normal full pressure because the shutoff head of the HPSI pumps is below RCS operating pressure. Inadvertent SIS operation at lower pressure and temperature could result in low temperature pressurization up to the HPSI pump shutoff pressure. While *this scenario does not violate RCS pressure or brittle fracture limits in Palo Verde with a uranium core*, the event should be evaluated if core loadings with plutonium decrease the margins to reactor vessel embrittlement at low temperature.

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A malfunction in the CVCS may cause an event where the charging pumps operate, causing the reactor pressure to increase. When the initiating event is combined with a loss of power and consequent loss of flow, the reduction in primary-to-secondary heat transfer increases the primary pressure rise. These effects are plant system effects which do not change with plutonium. The plutonium core affects the positive reactivity response caused by the lowered temperature of the excess charging flow. A more negative moderator temperature coefficient would be worse, but the temperature effect on the power and pressure is relatively small compared to the effect from increasing inventory. In conclusion, the plutonium core should have only a small effect on the peak pressure consequences of this event and no adverse effect on minimum DNBR.

5.1.2.1.6 Decrease in Reactor Coolant System Inventory Events

The events occurring in this fault group are characterized by decreasing RCS pressure, reduced RCS inventory and an approach to fuel design limits. Reactor coolant pipe breaks are included in this fault group. For in containment breaks a resultant pressurization of the containment occurs.

The causes of the events within this group are in most cases due to a pipe rupture. However, the inadvertent opening of a pressurizer safety/relief valve is a result of mechanical valve failure or operator error. Events within this group are classified as limiting faults.

The inside containment breaks will be detected by means of the low pressurizer pressure and level alarms. For outside containment breaks, in addition to the pressurizer alarms, the operator will be alerted by the regenerative heat exchanger high exit temperature alarm, the letdown line low pressure alarm, the auxiliary building high radiation, temperature, humidity, and sump level alarms, and the volume control tank low level alarm. Reactor trips required are low pressurizer pressure, manual, high steam generator water level, and margin to subcooling.

The faults within this group are characterized by decreasing primary pressure and reduced RCS inventory. Upon reactor trip, steam generator pressure will increase. For in containment pipe breaks, containment pressure limits may be challenged. Secondary pressurization is accommodated by the main steam safety valves. For certain cases auxiliary feedwater actuation and use of the atmospheric dump valve system is required. In order to mitigate events resulting in a loss of RCS inventory, safety injection pump flow is required and in some cases safety injection tank flow is required. The reactor coolant gas vent system is required to mitigate the consequences of the steam generator tube rupture accident. In order to mitigate the containment pressure transient and offsite dose release for inside containment loss of coolant accidents the containment spray and isolation systems are required. In addition, in order to mitigate the offsite dose due to loss of coolant accidents, the auxiliary building and control room ventilation systems are required. Eventual operator assisted cooldown following

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the initiating fault may lead to operator initiation of shutdown cooling. Once in shutdown cooling, the plant can be maintained in a stable condition indefinitely.

The letdown line break event, outside containment, releases primary coolant to the atmosphere. Analyses of this event assume no operator action or automatic mitigating function for 10 minutes. Since there is no credited reactor trip and since there is no significant variation in core power or coolant temperature, there are no reactivity effects. Hence, the analytical response is based upon normal plant functions that would conservatively maximize the break flow rate and these functions do not vary with the plutonium core.

It is assumed that the design ranges for CVCS charging and letdown flow rates are

Design requirements are established, and safety related design basis events that are influenced by charging and letdown flow capacities should be evaluated. For the letdown line break event, the primary parameters are the maximum break flow rate determined by the letdown line configuration, and the maximum break flow temperature, determined by the combination of flow rate and heat transfer capacities of the letdown and regenerative heat exchangers. The break flow temperature influences radiological dose because it determines the fraction of break flow that evaporates into the atmosphere. There is margin between the most limiting dose reported in the updated FSAR for this event and the dose acceptance criteria in the SRP.

The steam generator tube rupture event is characterized by the radiological release of primary coolant via the pathway through the ruptured tube to secondary steam, thence, to the atmosphere. Prior to reactor trip, the RCS temperature does not vary significantly so there are no reactivity feedback effects that would make the core response different with a plutonium core. Other influences on the radiological release are essentially dependent upon the plant responses which are unchanged when a plutonium core is substituted. Decay heat is a significant factor, but the decay heat for the first two to eight hours is no higher with the plutonium core so the magnitude of steam release is no larger. In conclusion, the plutonium core should have no significant effect on the Palo Verde plant response to a SGTR event.

The LOCA event is unique in that the U.S. NRC prescribes one set of assumptions and analytical models for evaluations to show that large scale core damage is avoided and another set for radiological dose evaluations that essentially assumes extensive core damage. Evaluations for these two conditions are analytically decoupled, so each is discussed separately here.

In the core damage evaluation, acceptance criteria are imposed for peak clad temperature, clad oxidation and long term cooling. Physical changes for the plutonium core that may influence the calculated results are outlined. Changes to the fuel pellet

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conductivity and gap conductance will affect the adiabatic heatup portion of the Large Break LOCA event.

Thermal radiation from the hot rod to the neighboring rod array beneficially reduces the hot rod temperature, so a flatter local rod power distribution yields worse results. Shorter neutron slowing down and diffusion lengths in the plutonium core will influence the local rod power distribution, but the direction of the result for the hottest rod array is not known. Thermal radiation is evaluated at two times during the LBLOCA. During refill, before the coolant level rises to the bottom of the fuel, cooling is by thermal radiation, but the duration is short and the temperatures are too low for appreciable radiation. During reflood, after the reflood rate falls below one inch per second and the hot rod has ruptured, the analytical model differentiates between cooling below the ballooned rupture elevation and above. Changes to the assumed steam-plus-thermal radiation cooling above the ruptured elevation could switch the hottest location to above the ruptured elevation, with associated incremental changes in peak clad temperature. Another effect of the local power distribution involves the net amount of gamma and beta radiation interchange between the hot rod and surrounding rods. In a flatter distribution, there is less net power transferred out of the hot rod by nuclear radiation. The effects of local power distribution changes require detailed evaluations for the plutonium core.

Reactivity effects are generally small for LOCA evaluations. The LBLOCA depressurizes the RCS so rapidly that the voiding effect overwhelms temperature effects, and rod insertion is not credited. The Small Break LOCA evaluation models the control rods and a small power rise occurs before the reactor trips on low pressure, but the void reactivity negates it, even before the rod insertion has significant effect. In conclusion, differences in reactivity effects should not be significant in LOCA evaluations of the plutonium core.

Decay heat influences the LOCA event throughout, but estimates of decay heat with the plutonium core shows no increase above that for the uranium core until after the typical period of concern about core temperature. Initial boron concentration and worth has some adverse reactivity effect early in the SBLOCA depressurization before reactor trip, but it is small. Injected boron concentration is significant in the LBLOCA during the long term cooling period. Then, the combination of decay heat, which determines the boiloff rate, and boron concentration determines the length of time available before the reactor operator must divert some of the SI coolant from the cold leg injection nozzle to the hot leg. The time when the boron concentration in the core approaches precipitation is shorter for higher decay heat and for higher injected boron concentrations. One evaluation with a 20% increase in decay heat and with 6200 ppm boron, relative to an evaluation with 4400 ppm, decreased the time available before required operator action to avoid boron precipitation in the core from about 3 hours to about 1-1/2 hours. The shorter time is still acceptable. Long term cooling requirements are probably satisfied with the plutonium core in Palo Verde, which

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reported 2 to 3 hours available time in the updated FSAR. Even so, the use of enriched boric acid in the coolant will mitigate this situation.

Actuation of the engineered safeguards is essential to satisfy the ECCS acceptance criteria, and there are no changes to that part of the Palo Verde Plant with a plutonium core. In conclusion, the LOCA evaluation with a plutonium core should yield results that are close to those reported in the Palo Verde FSAR, but several differences in core details will cause differences in peak clad temperature and clad oxidation and the direction of the trend is uncertain at this time.

The second area of LOCA evaluation, after core damage evaluation, is radiological dose. Calculations of the containment pressure transient yield some values of time

and as prescribed in Regulatory Guide 1.4. The isotopic quantities will change when plutonium is the fissile material but this is accounted for by detailed depletion calculations for specific plutonium configurations and burnup cycles.

In conclusion, differences in the radiological dose from a postulated LOCA in a plant with a plutonium core are dependent on details of the plutonium core and power distributions.

5.1.2.1.7 Radioactive Material Release from a Subsystem or Component Events

The events occurring in this fault group result in ex-core release sequences. The types of faults involve radwaste system failure and the drop of a fuel assembly and spent fuel cask. The cause of the radwaste system failure is the rupture of a tank. The cause of the dropping of a fuel assembly or a spent fuel cask is mechanical failure of the lifting equipment. The tank rupture is classified as a limiting fault. The dropping of a fuel assembly or the spent fuel cask is classified as an infrequent fault. There are no reactor trips associated with these faults. The only means of detecting the tank failure and the spent fuel cask drop is by visual inspection. The occurrence of a fuel assembly drop will result in a high radiation alarms in the containment or the fuel building wherever the fault occurs. Upon dropping a fuel assembly in the containment, the resultant discharge of radionuclides will result in a containment isolation signal, thus isolating the containment. Prior to this, isolation radioisotopes will be filtered by the refueling purge system. Upon dropping a fuel assembly in the fuel building, the fuel building ventilation system will reposition its dampers to ensure filtration of the activity released prior to the releases reaching the dampers. The dropped cask event is eliminated by procedures and design.

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The release of radioactive liquid from the Refueling water Tank (RWT) was the most limiting radioactive tank failure reported in the Palo Verde updated FSAR. The event is characterized by a rapid release of the contents to the environment. In order to dilute the radioactive concentration to allowable levels, the minimum dilution that the site conditions must provide is calculated.

Dilution required is dependent upon tank volume and radioactive isotope concentrations. These are dependent upon the plutonium core isotopic inventory, the operating boron concentration and the CVCS design capacities. Results show more than a factor of ten margin to 10CFR20 limit concentrations for Palo Verde, providing margin for variations with the boron storage tank requirements with a plutonium core.

The fuel handling event consequences are directly dependent upon the quantity of fission product iodine and fission gases in the fuel gap that are assumed released when a fuel assembly is damaged in handling. The NRC specifies assumptions in Regulatory Guide 1.25. These assumptions are assumed valid for plutonium fuel assemblies. The most significant are the fractions of core inventory of iodine and noble gases that are in the fuel rod gap and are released upon fuel clad damage. Specific evaluations of the plutonium fuel assembly inventory are needed. Variations in the inventory, and therefore dose, from that reported in the Palo Verde updated FSAR may be accommodated, if needed, by the margin to the applicable dose acceptance criteria.

5.1.2.2 Severe Accident Considerations

For the existing reactors, severe accident considerations have not been required, and therefore the analysis and any system modifications have not been performed. Section 5.1.2.3 of the Evolutionary Reactor Report covers the analysis and design considerations for System 80+. Similar considerations for existing reactors maybe required if licensing is reopened to allow for MOX fuel.

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5.2 Licensing the Reactor Facility

5.2.1 Palo Verde

The policy associated with converting any of the Palo Verde units into a plutonium-disposition reactor is that safety and protection of the environment will not be adversely impacted by the conversion. The reactor facility will be modified, if needed, and operated in compliance with all applicable regulations. In this connection, the Final Safety Analysis Report (FSAR) will be revised to address unit operation with MOX fuel. The FSAR changes will be submitted to the NRC for review and approval.

Most PWR systems require design modifications to accommodate large loadings of mixed-oxide (MOX) fuel. These modifications may include additional control rods to maintain 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.

The System 80 Standard Design, upon which Palo Verde units are based, however, was specifically developed to accommodate MOX fuel loadings up to and including an All Plutonium Reactor (APR). Consequently, design requirements for APR operation are incorporated in the basic systems of the System 80 design, or design provision is made which facilitate modifications for APR operation. For this reason, the primary licensing impact of APR operation on Palo Verde is limited to modifying the analyses and revising applicable sections of FSAR to address APR operation. These revisions will have to be reviewed and approved by the NRC based on the fact that these revisions will include, among other things, certain changes to the Technical Specifications. However, the technical review by the NRC on these revisions to the FSAR should be straight forward in that they should not involve any unreviewed safety questions since the revisions *would not depart substantially from the current Palo Verde design* which has been reviewed and approved by the NRC.

The FSAR sections dealing with the following systems and subject matters will have to be reviewed for potential revisions to address plutonium fuel loadings:

- * Shutdown Cooling System

The Palo Verde Shutdown Cooling System is designed to handle higher long-term decay heat generation rates associated with MOX fuel without any hardware changes. Therefore, only a confirmatory analysis is required to verify that the higher decay heat load can be accommodated without significantly reducing the existing margin of safety.

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*** Spent Fuel Pool Cooling System**

The Palo Verde Spent Fuel Pool Cooling System is designed to handle increased long-term decay of spent MOX fuel without any hardware changes to the system. Therefore, only a confirmatory analysis is required to verify that the increased long-term decay of spent MOX fuel can be accommodated without significantly reducing the existing margin of safety.

*** Component Cooling Water System**

The Palo Verde Component Cooling Water System is designed to handle higher long-term decay heat generation rates associated with MOX fuel without any hardware changes. Therefore, only a confirmatory analysis is required to verify that the higher decay heat load can be accommodated without significantly reducing the margin of safety.

*** Safety Injection System**

The increased maximum soluble boron concentration required for APR operation in the Safety Injection Tanks needs to be analyzed to determine the new concentration required to maintain adequate shutdown margin and to assess the impact due to a malfunction of equipment important to safety compared to that previously evaluated with a lower soluble boron concentration for the current Palo Verde design. Alternatively, the use of enriched soluble boron would need to be reviewed.

*** Chemical and Volume Control System**

The Palo Verde Chemical and Volume Control System is designed to handle higher soluble boron concentration in the Reactor Coolant System associated with APR operation. Therefore, only a confirmatory analysis is required to verify that the system can in fact handle the higher soluble boron concentration associated with APR operation.

*** Refueling Water Storage Tank**

The increased maximum soluble boron concentration required for APR operation in the Refueling Water Storage Tank will be analyzed to determine the new concentration required to maintain adequate shutdown margin and to assess the impact due to a malfunction of equipment important to safety compared to that previously evaluated with a lower soluble boron concentration for the current Palo Verde design. Alternatively, the use of enriched soluble boron would need to be reviewed.

*** Radioactive Waste Management System**

The liquid and gaseous radwaste systems will be reviewed to determine whether the systems can handle the higher tritium levels.

*** Reactor Internals**

The Palo Verde reactor vessel and reactor internals are designed to tolerate the higher neutron fluence levels and the higher heating rates associated with APR operation. Therefore, only a confirmatory analysis is required to verify that the higher neutron

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fluence levels and the higher heating rates can be accommodated without significantly reducing the existing margin of safety.

*** Core Design**

A detailed thermal hydraulic analysis and fuel performance analysis of the reference Pu core design will 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 basis. Additionally, a detailed neutronics evaluation of the reference Pu burning core design including depletion isotopics, reactivity coefficients, control worths, and power distributions as a function of burnup. Detailed design is based on the NRC approved methods. Analyses are performed for core stability and power distribution control.

*** Accident Analyses**

The LOCA and non-LOCA safety analyses will be evaluated using NRC approved licensing methodology consistent with the applications for Palo Verde FSAR which have been reviewed and approved by the NRC staff. The analyses performed will 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 with MOX fuel operation still meets licensing basis safety criteria.

*** Radiation Protection**

Radioactive decay of plutonium isotopes in fresh MOX fuel requires provision of shielding in the fuel receipt, handling and inspection area. Accordingly, a minor design change in the area of shielding design as well as changes to the FSAR may be required.

*** Technical Specifications**

The technical specifications will be revised to reflect plant operations with the plutonium burning core. The changes may include certain core physics parameters.

With respect to environmental impacts associated with the conversion, a study of the NRC's review history on GESMO, the Generic Environmental Statement on Mixed Oxide Fuel, indicates that NRC review will not involve major contentious issues. The NRC was in the final draft stages of review on GESMO 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 the time, the NRC found no objection to the use of mixed-oxide fuel. Therefore, a finding of no significant impact appears to be warranted for the conversion and NRC approval of the conversion may be facilitated.

During the initial phase of this project, a plan will be developed to identify and document all required Federal and State permits that need to be amended, and the lead time and schedule required for each, to operate the Palo Verde units as a plutonium-

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disposition facility. 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 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 Palo Verde units for the missions of plutonium disposition and electric power generation. Compliance with safety and environmental requirements will be demonstrated, as will licensability under NRC regulations.

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TABLE 5.2-1
TYPICAL REGULATORY AND ENVIRONMENTAL PERMITS AND LICENSES FOR
PLUTONIUM-DISPOSITION FACILITY

<u>Permit or License</u>	<u>Requirements</u>
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.

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.

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TABLE 5.2-1 (CONT'D)
TYPICAL REGULATORY AND ENVIRONMENTAL PERMITS AND LICENSES FOR
PLUTONIUM-DISPOSITION FACILITY

<u>Permit or License</u>	<u>Requirements</u>
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 Religious Freedom Act	Disturbance of areas considered "sacred" to Indian cultures.
Fish and Wildlife Coordination Act	Endangered species and migratory bird impact.

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5.2.2 3410 MWt and 2825 MWt Designs

The policy associated with converting non-System 80 units into a plutonium-disposition reactor is that safety and protection of the environment will not be adversely impacted by the conversion. The reactor facility will be modified as necessary and operated in compliance with all applicable regulations. In this connection, any design changes will be submitted to the NRC for review and approval. In addition, the FSAR will be revised to address unit operation with MOX fuel.

Although the reactor can accept full MOX cores of weapons-grade plutonium, the 3410 MWt (Waterford 3 and SONGS 2 and 3), and the 2825 MWt (ANO-2) units would require design modifications and changes to operational requirements to accommodate large loadings of mixed-oxide (MOX) fuel. These potential modifications and changes include equipment modifications to accommodate higher soluble boron concentrations, core and spent fuel cooling equipment modifications to accommodate the higher decay heat loads associated with irradiated MOX fuel, operational changes associated with 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 changes to fuel storage and fuel handling facilities to safely accommodate MOX fuel. Additionally, changes to the control rod complement (exchanging part-strength rods for full strength rods, and possibly adding additional nozzles for more control rods) may be desired to enhance the plutonium disposition capability. These modifications and changes will result in modifying the analyses and revising applicable sections of the FSAR to address APR operation. These revisions will have to be reviewed and approved by the NRC.

The FSAR sections dealing with the following systems and subject matters will have to be reviewed for potential revisions to address plutonium fuel loadings:

- * Shutdown Cooling System**

The existing Shutdown Cooling System design will be reviewed to determine whether it can handle higher long-term decay heat generation rates associated with APR operation MOX fuel. The system will be modified if necessary.

- * Spent Fuel Pool Cooling System**

The existing Spent Fuel Pool Cooling System design will be reviewed to determine whether it can handle increased long-term decay of spent MOX fuel without any hardware changes to the system. The system will be modified if necessary.

- * Component Cooling Water System**

The existing Component Cooling Water System design will be reviewed to determine whether it can handle higher long-term decay heat generation rates associated with MOX fuel. The system will be modified if necessary.

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*** Safety Injection System**

The increased maximum soluble boron concentration required for APR operation in the Safety Injection Tanks needs to be analyzed to determine the new concentration required to maintain adequate shutdown margin and to assess the impact due to a malfunction of equipment important to safety compared to that previously evaluated with a lower soluble boron concentration for the existing system design. Alternatively, the use of enriched soluble boron would need to be reviewed. The system will be modified if necessary.

*** Chemical and Volume Control System**

The existing Chemical and Volume Control System design will be reviewed to determine whether it can handle higher soluble boron concentration in the Reactor Coolant System associated with APR operation. Alternatively, the use of enriched soluble boron would need to be reviewed. The system will be modified if necessary.

*** Refueling Water Storage Tank**

The increased maximum soluble boron concentration required for APR operation in the Refueling Water Storage Tank will be analyzed to determine the new concentration required to maintain adequate shutdown margin and to assess the impact due to a malfunction of equipment important to safety compared to that previously evaluated with a lower soluble boron concentration for the existing system design. Alternatively, the use of enriched soluble boron would need to be reviewed.

*** Radioactive Waste Management System**

The liquid and gaseous radwaste system designs will be reviewed to determine whether the systems can handle the higher tritium levels. The system will be modified if necessary.

*** Reactor Internals**

The reactor vessel and reactor internals design will be reviewed to determine whether they can tolerate the higher neutron fluence levels and the higher heating rates associated with APR operation. The system design and/or associated operational requirements will be modified as necessary.

*** Core Design**

A detailed thermal hydraulic analysis and fuel performance analysis of the reference Pu core design will 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 basis. Additionally, a detailed neutronics evaluation of the reference Pu burning core design including depletion isotopics, reactivity coefficients, control worths, and power distributions as a function of burnup. Detailed design is based on the NRC approved methods. Analyses are performed for core stability and power distribution control.

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*** Accident Analyses**

The LOCA and non-LOCA safety analyses will be evaluated using NRC approved licensing methodology consistent with the applications for the FSAR which have been reviewed and approved by the NRC staff. The analyses performed will 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 with MOX fuel operation still meets licensing basis safety criteria.

*** Radiation Protection**

Radioactive decay of plutonium isotopes in fresh MOX fuel requires provision of shielding in the fuel receipt, handling and inspection area. Accordingly, a minor design change in the area of shielding design as well as changes to the FSAR may be required.

*** Technical Specifications**

The technical specifications will be revised to reflect plant operations with the plutonium burning core. The changes may include certain core physics parameters.

With respect to environmental impacts associated with the conversion, a study of the NRC's review history on GESMO, the Generic Environmental Statement on Mixed Oxide Fuel, indicates that NRC review will not involve major contentious issues. The NRC was in the final draft stages of review on GESMO 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 the time, the NRC found no objection to the use of mixed-oxide fuel. Therefore, a finding of no significant impact appears to be warranted for the conversion and NRC approval of the conversion may be facilitated.

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 unit for the missions of plutonium disposition and electric power generation. Compliance with safety and environmental requirements will be demonstrated, as will licensability under NRC regulations.

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5.2.3 WNP-3

The policy associated with completing and converting WNP-3 into a 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 statutes and regulations. Among all required permits and licenses, two of them are more prominent than the others. One is obtaining an NRC operating license for WNP-3 for the missions of plutonium disposition. The other is getting the environment report relating to the construction, operation, and decommissioning of WNP-3 plutonium-disposition facility approved in accordance with the *National Environmental Policy Act*. A preliminary listing of typical licenses and permits required for the facility is given in Table 5.2-1.

Most PWR systems require design modifications to accommodate large loadings of mixed-oxide (MOX) fuel. These modifications include additional control rods to maintain 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.

The System 80 Standard Design, upon which WNP-3 is based, however, was specifically developed to accommodate MOX fuel loadings up to and including an All Plutonium Reactor (APR). Consequently, design requirements for APR operation are incorporated in the basic systems of WNP-3 design, or design provision is made which facilitate modifications for APR operation. For this reason, the primary licensing impact of APR operation on WNP-3 design is limited to modifying the analyses and revising applicable sections of the FSAR to address APR operation. These revisions will have to be reviewed and approved by the NRC based on the fact that these revisions will include, among other things, certain changes to the Technical Specifications.

The revisions would not depart substantially from the reference System 80 Standard Design which has been reviewed and approved by the NRC.

The FSAR sections dealing with the following systems and subject matters will have to be reviewed for potential revisions to address plutonium fuel loadings:

- * Shutdown Cooling System

The System 80 Shutdown Cooling System is designed to handle higher long-term decay heat generation rates associated with MOX fuel. Therefore, only a confirmatory

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analysis is required to verify that the higher decay heat load can be accommodated without significantly reducing the margin of safety.

* Spent Fuel Pool Cooling System

The System 80 Spent Fuel Pool Cooling System is designed to handle increased long-term decay of spent MOX fuel without any hardware changes to the system. Therefore, only a confirmatory analysis is required to verify that the increased long-term decay of spent MOX fuel can be accommodated without significantly reducing the margin of safety.

* Component Cooling Water System

The System 80 Component Cooling Water System is designed to handle higher long-term decay heat generation rates associated with MOX fuel. Therefore, only a confirmatory analysis is required to verify that the higher decay heat load can be accommodated without significantly reducing the margin of safety.

* Safety Injection System

The increased maximum soluble boron concentration required for APR operation in the Safety Injection Tanks needs to be analyzed to determine the new concentration required to maintain adequate shutdown margin and to assess the impact due to a malfunction of equipment important to safety compared to that previously evaluated with a lower soluble boron concentration for the reference System 80 design. Alternatively, the use of enriched soluble boron would need to be reviewed.

* Chemical and Volume Control System

The System 80 Chemical and Volume Control System is designed to handle higher soluble boron concentration in the Reactor Coolant System associated with APR operation. Therefore, only a confirmatory analysis is required to verify that the system can in fact handle the higher soluble boron concentration associated with APR operation. Alternatively, the use of enriched soluble boron would need to be reviewed.

* Refueling Water Storage Tank

The increased maximum soluble boron concentration required for APR operation in the Refueling Water Storage Tank will be analyzed to determine the new concentration required to maintain adequate shutdown margin and to assess the impact due to a malfunction of equipment important to safety compared to that previously evaluated with a lower soluble boron concentration for the reference System 80 design. Alternatively, the use of enriched soluble boron would need to be reviewed.

* Radioactive Waste Management System

The liquid and gaseous radwaste systems will be reviewed to determine whether the systems can handle the higher tritium levels.

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* Reactor Internals

The System 80 reactor vessel and reactor internals are designed to tolerate the higher neutron fluence levels and the higher heating rates associated with APR operation. Therefore, only a confirmatory analysis is required to verify that the higher neutron fluence levels and the higher heating rates can be accommodated without significantly reducing the margin of safety.

* Core Design

hydraulic performance of the reference Pu burning core requires evaluation for an performance-related and safety-related design basis. Additionally, a detailed neutronics evaluation of the reference Pu burning core design including depletion isotopics, reactivity coefficients, control worths, and power distributions as a function of burnup. Detailed design is based on the NRC approved methods. Analyses are performed for core stability and power distribution control.

* Accident Analyses

The LOCA and non-LOCA safety analyses will be evaluated using NRC approved licensing methodology consistent with the applications for the FSAR which have been reviewed and approved by the NRC staff. The analyses performed will 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 with MOX fuel operation still meets licensing basis safety criteria.

* Radiation Protection

Radioactive decay of plutonium isotopes in fresh MOX fuel requires provision of shielding in the fuel receipt, handling and inspection area. Accordingly, a minor design change in the area of shielding design as well as changes to the FSAR may be required.

* Technical Specifications

The technical specifications will be revised to reflect plant operations with the plutonium burning core. The changes may include certain core physics parameters.

With respect to the NRC review of the remaining sections of the FSAR, it should be noted that WNP-3 falls within the definition of "Deferred Plants" in the NRC's Policy Statement on deferred plants. This Policy Statement is intended, among other things, to define the applicability of new regulatory requirements by treating deferred plants the same as those under construction. The practical effect is that for WNP-3, new requirements are limited under NRC policy by applying the backfit rule (i.e., 10 CFR 50.109) to all new rules and regulatory positions issued before docketing of the FSAR application in August 7, 1982. Accordingly, the NRC review should be straight forward in that the review of the NSSS scope was completed with issuance of Final

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Design Approval for the System 80 Standard Design and the balance of the FSAR was reviewed in large part by the NRC staff and documented in a draft SER.

With respect to the environmental report associated with a plutonium-disposition facility using WNP-3, a study of the NRC's review history on GESMO, the Generic Environmental Statement on Mixed Oxide Fuel, indicates that NRC review will not involve major contentious issues. The NRC was in the final draft stages of review on GESMO 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 the time, the NRC found no objection to the use of mixed-oxide fuel. Therefore, NRC approval of WNP-3 as a plutonium-disposition facility may be facilitated.

During the initial phase of this project, a plan will be developed to identify and document all applicable Federal and State permits, and the lead time and schedule required for each permit, to operate WNP-3 as a plutonium-disposition facility. 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 WNP-3 for the missions of plutonium disposition and electric power generation. Compliance with safety and environmental requirements will be demonstrated, as will licensability under NRC regulations.

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5.3.1 Safety Analysis and Impact on the Environment

The success of the MOX fuel cycle depends on its performance, safety record and public acceptance. Public acceptance can only be achieved by demonstrating safety in all phases of the MOX fuel cycle, reliability in its performance and safe disposal of generated wastes. To provide reasonable assurance that the MOX fuel fabrication plant can be operated without undue risk to the health and safety of employees and the public, and to the environment, such plant must be located, designed, constructed, tested and operated according to well defined criteria of safety. Also, physical security and SNM accountability and protection must be assured. The main safety problems associated with the mixed oxide fuel fabrication industry is the occupational exposure and the release of Pu bearing material into the environment. The safe operation of the facility and its impact on the environment have to be evaluated for normal operation and for accidental events.

5.3.2 Normal Operation

During plant normal operation, the occupational exposure and the radioactive releases are kept as low as possible by using the highest design standards and then performing and implementing periodic safety, inspection and maintenance programs. Occupational exposure and the Pu releases during plant normal operation are low enough to avoid

and the Pu solid wastes are in accordance with and meet all the local, state and Federal laws and regulations.

5.3.3 Accidental Events

The accidental radiation exposure and release are normally produced by a sequence of events which ultimately result in an exposure to or a release of radioactive material. Considering the risks associated with postulated accidents, the probability of occurrence of the accidents and the severity of their consequences must be taken into account. The plant design provides high reliability or redundant systems in order to assure low and acceptable risks.

The plant events which have to be considered for the design to assure low risks for the environment are: fire, explosion, criticality and loss of confinement. The MOX fuel fabrication plant is designed and operated to assure very low risks for such accidents. Also the plant design will minimize the effect of natural phenomena (such as earthquake, flood, tornado, airplane crashes).

The risks of such events to the environment is evaluated, taking into account the plant site characteristics and the associated plant design. Plant design criteria are based on the premise that the plant location represents a reasonable balance between the advantages and limitations inherent in the size of the exclusion area (distance to

The present knowledge and the expected results of future technological developments concerning commercial plant design and operation are such that the risks associated with such events could be predicted and could be reduced to acceptable risks.

5.3.4 Licensing, DOE Orders, Design Codes and Procedures

5.3.4.1 Licensing

Applicable NRC Regulatory Guide Divisions include:

- Reg. Guide 3 - Fuels and Materials Facilities
- Reg. Guide 4 - Environmental and Siting
- Reg. Guide 5 - Materials and Plant Protection
- Reg. Guide 7 - Transportation
- Reg. Guide 8 - Occupational Health

Plutonium recycle in light water reactors (LWRs) is defined as the use of plutonium-uranium mixed oxide fuels in which plutonium obtained from weapons replaces some portion of the U^{235} normally used for fueling the LWRs. The US NRC has in the past determined that the applicant planning to construct a MOX fuel fabrication facility must submit an application for a Special Nuclear Material facility under 10 CFR 70.

Mixed oxide fuel must be fabricated in specially designed fuel fabrication plant. The general rules utilized to license a LWR fuel fabrication plant may be applicable, however, with the presence of plutonium NRC will certainly take a more conservative approach and impose certain additional requirements to a MOX fuel fabrication facility.

There are three key aspects in the licensing process for the mixed oxide fuel fabrication facility which have an impact on the schedule. These are: 1) Design of facility 2) Environmental assessment, and 3) Construction. The three aspects are discussed in some detail to better understand the MOX reference schedule and justify the time associated with the activities in the schedule spread sheet (see Section 6.0).

The granting of a license by NRC will require a complete assessment of the safety of the facility. This would involve addressing the requirements specified in 10 CFR 70 for special nuclear materials and submitting an application. The application can be prepared with guidance from the regulatory guides. Some of these specifically applicable for a fuel fabrication plant are identified below:

Reg. Guide 3.3- Quality Assurance Program requirements for FTFG

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Reg. Guide 3.4-	Nuclear Criticality and Safety considerations in operation with fissionable materials.
Reg. Guide 3.7 -	Monitoring of combustible gases and vapors in plutonium processing and fuel fabrication plants.
Reg. Guide 3.12 -	General design guide for Ventilation systems of plutonium processing and fuel fabrication plants
Reg. Guide 3.14 -	Seismic Design Classification for plutonium processing and fuel fabrication plants
Reg. Guide 3.16 -	General Fire protection guide for plutonium processing and fuel fabrication plants
Reg. Guide 3.35 -	Assumptions used for evaluating the potential radiological consequences of accidental nuclear criticality in a plutonium processing and fuel fabrication plants.
Reg. Guide 3.39 -	Standard format and content of license applications for plutonium Processing and fuel fabrication plants.
Reg. Guide 3.40 -	Design basis floods for fuel reprocessing plants and for plutonium and fuel fabrication plant.
Reg. Guide 3.42 -	Emergency planning for fuel cycle facilities and plants licensed under 10CFR Parts 50 and 70.
Reg. Guide 5.30 -	Materials protection contingency measures for uranium and plutonium fuel manufacturing plants.
Reg. Guide 5.52 -	Standard format and content of a license physical protection plan for strategic special nuclear material at fixed sites.

5.3.4.2 Environmental Assessment

In the past (November 1975), NRC required the preparation of a Generic Environmental Statement on the use of Recycle Plutonium in Mixed Oxide fuel and it is certain that a similar requirement will be imposed now. Thus an environmental assessment will have to be completed in parallel with the plant SAR prior to the start of construction of the facility. The environmental study and assessment will have to satisfy the requirements specified in 10 CFR 51. This assessment will culminate in the submission of an Environmental Report (ER) to NRC by the applicant to obtain a permit/license for the plant. The ER shall contain a description of the proposed action, namely building, a statement of its purpose and the description of environment effected. The report shall include an analysis which considers and balances the environmental effects of the proposed action (production of radioactive waste, use of water for the facility, release of radioactivity into the air, etc) and the alternatives available for reducing or avoiding adverse environmental impacts. The analysis shall be able to quantify the various factors considered. This involves collection of data on water, air, amount of material handled and obtain permits for water, air, etc which become part of the ER. A number of regulatory guides under Division 4 - Environmental and Siting - should be considered while preparing the ER.

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5.3.4.3 Construction

The location of the Fuel Fabrication and Target Facility has to satisfy the requirements of 10 CFR 100. Although this regulation is specifically applicable to a nuclear power plant, certain activities at the Fuel Facility are similar and one can assume NRC will apply the same criteria. The construction of the facility can start only after completion of the following efforts: site approval, obtain air permits, obtain water permits, submission of the Application, submission of ER, and obtaining a Safety Evaluation Report (SER) and a Final Environmental Impact Statement (FEIS) from the NRC.

5.3.4.4 Uncertainties

This is the first MOX facility to be built in USA and therefore one has to consider a number of unknowns which could impact the schedule in a significant manner.

- The facility is being built for the DOE with the facility licensed by NRC. It is assumed that the NRC would take the lead on licensing. In addition, a draft bill,⁴ presently undergoing hearings, would require all new DOE nuclear facilities to be licensed by the NRC. This includes any radioactive waste management facilities as well as production or research reactors. It appears that DOE and NRC officials are amenable to the draft bill, although NRC Chairman Selin has some reservations with the bill's broad language and lack of additional NRC funding to cover the additional oversight responsibilities.^{5,6,7,8}

⁴ H.R. 3920, sponsored by House Natural Resources Committee Chairman George Miller (D-California), House Natural Resources Subcommittee on Energy and Mineral Resources Chairman Richard Lehman (D-California), House Energy and Commerce Subcommittee on Energy and Power Retiring Chairman Philip Sharp (D-Indiana) and Representative Peter DeFazio (D-Oregon).

⁵ Nucleonics Week article, "Miller Plans to Regulate New, Existing DOE Facilities," Vol. 34, No. 51, December 23, 1993.

⁶ Nucleonics Week article, "Bill to Regulate New, Existing DOE Facilities to be Introduced," Vol. 35, No. 7, February 17, 1994.

⁷ Inside NRC article, "Bill Would Give NRC Oversight Over Any New DOE Nuclear Facility," Vol. 16, No. 5, March 7, 1994.

⁸ The Energy Daily article, "Selin Says No Thanks To Broader Regulatory Role," Vol. 22, No. 45, March 9, 1994.

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- It is assumed that MOX fuel assemblies will be licensed soon and be available for conducting fuel tests. Fuel containing plutonium have not been licensed by NRC to date and it is not known that process will be used to license one.
- One needs to know if the SAR on the MF² should address the issue of severe accidents, for example, MOX fuel melting during its fabrication.

5.3.4.5 DOE Orders

1. DOE Order 5480.1A "Environmental protection, safety, and health protection program for DOE operation"
 - a. Chapter V "Safety of Nuclear Facilities"
 - b. Chapter VII "Industrial Fire Protection"
 - c. Chapter X "Industrial Hygiene Program"
 - d. Chapter XI "Standards of Radiation Protection"
2. DOE Order 5480.4 "Environmental Protection, Safety, and Health Protection Standards"
3. DOE Order 6430.1 "General Design Criteria Manual"
 - a. Chapter I "Criteria Purpose and Application, the Planning and Design Process, and Fundamental Design Requirements"
 - b. Chapter IV "Architectural and Structural"
 - c. Chapter V "Mechanical Systems"
 - d. Chapter VI "Interior Electrical Systems"
 - e. Chapter VII "Interior Telecommunications and Alarm Systems"
 - f. Chapter X "Fire Protection"
 - g. Chapter XII "Water Pollution Control"
 - h. Chapter XXI "Plutonium Facilities"
 - i. Chapter XXIII "Un-irradiated Enriched Uranium Storage Facilities"
4. DOE Order 5630.2 "Control and Accountability of Nuclear Materials, Basic Principles"
5. DOE Order 5630.3 "Documentation of Transactions"
6. DOE Order 5630.6 "Nuclear Materials Management and Safeguards System Data Processing Procedures"
7. DOE Order 5632.1 "Physical Protection of Classified Matter"
8. DOE Order 5632.2 "Physical Protection of Special Nuclear Materials"

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9. DOE Order 5700.6A "Quality Assurance"

5.3.4.6 Design Codes

1. UBC: Uniform Building Code
2. AISC: American Institute of Steel Construction
3. NFPA: National Fire Protection Association
4. ASME: American Society of Mechanical Engineers, Boiler and Pressure Vessel Code Section VIII, "Pressure Vessels"
5. OSHA: Occupational Safety and Health Administration Bulletins
6. Code of Federal Requirements, Title 10 100, "Reactor Site Criteria", USNRC

5.3.4.7 PROCEDURES

The ABB-CE Team will implement the internal design procedures which include:

1. Management Procedures
2. Radiation Protection Procedures
3. Waste Management and Recycle Procedures
4. Quality Assurance Manual
5. Crane and Hoist Safety Manual
6. New Facility Shield Design Criteria
7. Seismic qualification of Category I Structures, Systems and Components
8. Criteria for Criticality Safety limits and controls

5.4 Environmental Impact Review Process

With respect to the environmental report associated with a plutonium-disposition facility using the System 80+ Standard Design, a study of the NRC's review history on GESMO, the Generic Environmental Statement on Mixed Oxide Fuel, indicates that NRC review will not involve major contentious issues. The NRC was in the final draft stages of review on GESMO 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 the 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.

Compliance with the National Environmental Policy Act is the cornerstone for developing the environmental report for the facility. An Environmental Report will be developed and submitted to the NRC in accordance with 10 CFR Part 51 "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions." The Environmental Report will include detailed information concerning the following:

- Purpose of the Proposed Facility

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- The Site and Environmental Interfaces with the Surrounding Region, including:
 - Geography and Demographics
 - Ecology
 - Meteorology
 - Hydrology
 - Geology
 - Regional Historical, Scenic, Cultural and Natural Features
- Description of the Plant, including source terms and effluent treatment systems
- Environmental Effects of Site Preparation, Plant Construction, and Transmission Facilities Construction
- Environmental Effects of Plant Operations
- Effluent and Environmental Measurements and Monitoring Programs
- Environmental Effects of Accidents
- Economic and Social Effects of Plant Construction and Operation
- Alternative Energy Sources and Sites
- Plant Design Alternatives
- Summary Cost-Benefit Analysis
- Environmental Approvals and Consultation

The content and format of the Environmental Report will be consistent with Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations."

This information will allow the Nuclear Regulatory Commission to develop and prepare an Environmental Impact Statement ("EIS"). The EIS will demonstrate that the facility will not have significant negative environmental impact on the region surrounding the facility.

PLUTONIUM DISPOSITION IN EXISTING REACTORS ESTIMATED COST AND SCHEDULE

6.0 ESTIMATED COST AND SCHEDULE

6.1 Summary Life Cycle Cost

The calculation of life cycle cost in the case of a new facility is relatively straightforward, and has been reported in the Evolutionary Reactor Report, Section 6. For an existing reactor facility, to be acquired or otherwise appropriated by the DOE for a mission of public importance, the cost elements contain a great deal of uncertainty, as such a step has never been taken before under current peacetime conditions. Moreover, as pointed out in Section 4.1, the DOE would be taking over an existing reactor that is producing power, and is included in the present owner's rate base, and for which the utility likely has several long term contracts for the supply of electricity, as well as fuel supply and maintenance supply contracts with others. There are many unprecedented steps involved, for which cost estimates are not available.

The major differences in the elements that make up the life cycle cost for an existing reactor, compared to a new reactor, used for plutonium disposition, include the following:

- Cost of acquisition of the facility, including indemnification of all concerned parties
- Cost of licensing for MOX fuel receipt, storage and operation
- Cost of any modifications to the facility, including reactor modifications and other modifications to provide safeguards and security to the MOX fuel and the facilities.
- Cost of operation and maintenance of the reactor during the mission, including the cost of the spent fuel facility for the final ten years after discharge from the reactor
- Cost of returning the reactor facility to its original owner

The first two of these elements, the cost of acquisition and of licensing, are in all probability the largest by far, and also are the most difficult to estimate. No estimates have been made for the acquisition by the Government of the Palo Verde or the 3410 MWt class reactors. Therefore, estimates of a life cycle cost for the acquisition and operation of any of these units for the plutonium disposition mission have not been prepared. A rough estimate of the licensing costs (not including the potential cost due to public reaction) has been prepared.

Modifications to the facilities to accept the MOX fuel and the plutonium disposition mission have been estimated where data is available. In general, the selected reactors can accept full MOX cores at a modest enrichment without major modifications, i.e.,

PLUTONIUM DISPOSITION IN EXISTING REACTORS ESTIMATED COST AND SCHEDULE

without additional control rods and drives. The possibility does exist, however, to improve the mission capability by adding more control rods, as has been discussed in Section 2.1. Other modifications, such as to plant auxiliary systems to take care of the additional decay heat load, for example, will depend upon a detailed evaluation of the existing systems; this has not been prepared. Similarly, additional safeguards and security cost estimates will depend on the existing security situation, requiring detailed surveys of the existing plants and the status of Vulnerability Analysis. This has not been performed.

Cost estimates for a Greenfield MOX Fuel Fabrication facility are similar to the estimate reported in the Evolutionary Reactor Report, Section 6.2, with the exception of the Indirect costs (engineering and allocation), which will be higher than for the collocated facility.

Operating and Maintenance costs have been reviewed for the case of existing reactors. The most significant factor involved, however, would have to do with the change of ownership, resulting in an opportunity to reduce O&M costs by improved management and contracting, rather than costs associated with the plutonium disposition mission. Estimates have been made of the O&M costs, on the basis that the plant would be upgraded in the change of ownership, and cost increments relative to the Plutonium Disposition mission incorporated.

Once the plutonium disposition mission is over, the plant would be returned to its original owner for the remainder of the lifetime in the license. It is assumed that this eventuality would be taken care of in the negotiations for takeover at the onset of the

would be enough time remaining, after the mission has been completed, to make it worth while for the utility to spend the effort to return the plant to the rate base, are considerable. More than likely, the utility would have written the plant off completely, rather than deal with such factors.

6.2 Capital Cost Detail

6.2.1 Cost of Modifications

The existing reactors are capable of operation with full MOX cores without major modification. The System 80 cores can be loaded with about 4.5 % weapons grade plutonium, while the non-System 80 cores can operate with about 3 % weapons grade plutonium. The limitation is in the available control rod worth, which decreases with increasing plutonium loading in the core. Possibilities were therefore reviewed that would provide increased control rod worth by adding extra control rod drive nozzles, making changes to the internals, and adding more control rods and drives, and the associated electronics. These were discussed in Section 2.1.

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For information purposes, costs of making the modifications to the vessel to incorporate eight additional control rods, assuming field operations, have been estimated at about \$2.3 million for materials. Labor costs would vary, depending upon the radioactivity of the vessel and the available space for working. It is assumed that the modification could be performed in a suitable lay-down area with local shielding to protect the workers. Even then, the number of workers employed would be substantial because of radiation dose limitations.

One other possible nuclear system modification to the plant for the purpose of running with MOX fuel is the use of enriched soluble boron in place of the natural boron in the boric acid in the primary coolant. The need to use enriched boron would be determined by a detailed review of the requirements for boron concentration in the coolant for reactivity control. If enriched boric acid is needed, the cost could run as high as \$10 million for the initial charge, depending on the amount of water involved, and the enrichment requirements. The annual makeup is expected to be insignificant.

6.2.2 Cost of Licensing

A preliminary cost estimate has been developed for licensing activities associated with the conversion of a current reactor to MOX fuel. The estimate does not include the physical modifications or the procedural changes necessary to use MOX fuel. This estimate only includes those activities associated with securing permission from the NRC to use MOX fuel. The costs listed here are the costs incurred by the licensee, whether the licensee is the current owner/operator or a new government agency owner/operator. This cost estimate does not include the possibility of protracted intervention or the promulgation of new requirements by the NRC.

The review costs associated with the review of any technical specification amendments are included in the annual fees payable by any licensee in accordance with 10 CFR § 171 (for a commercial nuclear power plant this is approximately \$3.1 million per unit per year). However, the full costs associated with the review of transfer of the license or review of any topical reports submitted to the NRC in support of the use of mixed oxide fuel would be billed to the licensee by the NRC.

<u>Activity</u>	<u>Estimate</u>
Estimated NRC billings for review of topical reports and transfer of the license (this does not include the annual 10 CFR Part 171 fee)	\$ 2,000,000
Final Safety Analysis Report (FSAR) Review and Changes	\$ 1,000,000

The cost is very dependent upon the amount of computer codes and runs needed to model the response to accidents by the NSSS to allow updating of FSAR Chapter 15.

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Technical Specifications review and revisions preparation \$ 200,000

Work performed to model the response of the NSSS to accidents should allow the confirmation of old or calculation of new Technical Specification setpoints.

Physical Security Plan review and revisions preparation (this does not include the cost of any physical modifications associated with the switch to MOX fuel) 250,000

Emergency Planning review and revisions preparation \$ 250,000

Safeguards Procedures review and revisions preparation \$ 500,000

Decontamination & Decommissioning (D&D) Plan review and revisions preparation \$ 250,000

Environmental Impact review and preparation \$ 500,000

Other Federal, State, and Local Permits/Authorizations review, permit fees, and revisions preparation \$ 250,000

General coordination, review, and submittal preparation; this would include legal support and review of the proposed license amendments \$ 800,000

Total Cost \$ 6,000,000

6.2.3 Completion of Washington Nuclear Project-3 (WNP-3)

The following was reported in VII C 4 Pg. VII-10 and 11 of the Phase 1 report dated May 15, 1993, and is repeated here for convenience.

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.

PLUTONIUM DISPOSITION IN EXISTING REACTORS ESTIMATED COST AND SCHEDULE

- 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.

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The WNP-3 evaluation reported above is still valid and was used as a starting point for our review. The capability of WNP-3 which is a System 80 plant is reported in Sections 1 and 2. Since it has not been completed its remaining operating lifetime is forty years. The observation last year that strong opposition from the members of the public could be expected has occurred and various groups in response to the Isaah proposal have expressed opposition to completion as a Plutonium consumer and the Governor has expressed his opposition as well as several local communities. However, there has been no outreach program in the area to explain the mission and its impact on the community. The Washington Public Power Supply Board of Directors recently voted to terminate WNP-3 Project and maintain the preservation mode until the end of December, 1994.

The status of WNP-3 of being 75% complete, having a valid construction permit and the existing estimates to complete result in the following financial assessment based on input from Washington Public Power Supply:

Completion Costs (94 dollars)	\$1,650M
Construction Schedule	5 Years
	<u>1994 in Millions</u>
Total Annual Operating Costs	
WNP-3 OEM	156.6
WNP-3 Capitol	35.0
WNP-3 Debt Service	136.4
Plutonium Disposal Fee	12.5
	<u>340.5</u>
Annual Payments by DOE & Bonneville Power Authority	\$ Millions
BPA	233.2
DOE	<u>107.3</u>
	<u>340.5</u>
	Total
	<u>340.5</u>
Cost of Power to BPA	Mills/KWH
1994 Constat \$ Average Annual	<u>28.40</u>

The above assumes that funds expended to reach 75% complete were sunk costs and are not included in the above.

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Washington Public Power supply has repeatedly stated they are willing to participate in the Plutonium mission. The design of the plant as reported in the NAS study makes it a leading candidate for this mission. The DOE would have to take the initiative to maintain this option as once the preservation status is removed the plant no longer can be considered a candidate.

6.3 Operations and Maintenance Costs

The Evolutionary Reactor Report addressed the O&M costs for a Plutonium Disposition Reactor (PDR) developed from the commercial System 80+ design but modified as necessary to satisfy the plutonium disposition mission. In essence, the commercial System 80+ design provided the baseline from which the PDR costs were estimated. Thus, a relatively detailed basis was available for development of the O&M cost estimates.

The available information and basis for estimating O&M costs is significantly different within the scope and timeframe for this report on existing reactors. As such, it is necessary to discuss the basis for estimating O&M costs. There are generic, design-specific and plant/site-specific aspects to be addressed. It is also essential to *appreciate several aspects of the trends in O&M costs for the U.S. commercial nuclear industry* in order to estimate the impact of using an existing reactor for the plutonium disposition mission. Indeed, the insights and perspective gained from an appreciation of the commercial nuclear industry's response to O&M costs and related requirements for ALWRs were indispensable for generating the O&M cost estimates for the Evolutionary Reactor Phase.

6.3.1 Background Information on O&M Costs of Existing Reactors

In brief, it is well known that the U.S. commercial nuclear industry is in the midst of extraordinary change as a result of the increasingly cost-competitive and market-driven environment. The objectives for cost-competitiveness have become sufficiently compelling as to be addressed on an industry-wide basis by the *Strategic Plan for Improved Economic Performance*, October 1993. That plan, which "is modeled after the Nuclear Power Oversight Committee (NPOC) Strategic Plan for Building New Nuclear Power Plants," addresses economic competitiveness issues as a series of building blocks. The Action Plan for Building Block 1E, Economic Performance Measures, specifically identifies production cost (including O&M) and capacity factor among the primary economic indicators for nuclear plant performance. It also notes both 1) the significant growth over the last decade of O&M and staffing levels and 2) the disparity among similar plants.

O&M is one of the few remaining controllable costs for existing plants, since capital costs are imbedded in the financial (debt) structure. In order to be more economically competitive, utilities are attempting to reduce O&M costs relative to their generation. The companion objective, to increase capacity factors, is probably of greater direct

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financial importance than O&M cost reduction, per se. (Some studies have indicated that the best operated plants have the joint characteristic of lower O&M and higher availability/capacity factor.) Various utilities have stated goals for reducing O&M costs while attaining capacity factors in the 82% to 85% range. The degree to which these goals are achievable and, particularly, sustainable in the long-term, is uncertain. Many approaches are being implemented by these various utilities in an attempt to reduce and control O&M costs. For example, many utilities have downsized, established cost (or profit) centers for nuclear plant sites, reduced or even eliminated full-time on-site contractors, and so forth.

Nuclear plant staffing is a major component of non-fuel O&M costs, often accounting for 65 to 70% of non-fuel O&M costs. Thus, staffing reductions are being considered carefully. Staffing reductions are possible within the limits of existing technology, provided that continued safe and reliable plant operations are not jeopardized. Productivity enhancements for the reduced staff are also being pursued via a combination of managerial improvements and technological advances. Each utility and plant has unique design features, operating philosophies, and approaches to meet these objectives.

A fundamental change in plant operating philosophy is gradually emerging as utilities have focused on reducing O&M costs. Overall, the nuclear industry is making progress

costs. The best quartile of performers in this regard may already be approaching the lower limit of O&M costs that is possible without adversely affecting safety, although capacity factor improvements are expected to continue. Naturally, since the remainder of the industry lags the best performers, the industry average data is expected to show continuing improvement while the best plants settle in a lower than historical O&M cost and the others move in that direction. The rate at which O&M costs and capacity factor can be improved is dependent on institutional factors; the limit of the improvements are set by the plant design and associated technology.

6.3.2 Factors Affecting the Existing Reactor O&M Estimates

The O&M costs of an existing reactor converted to the plutonium disposition mission would not be expected to be simply an additional increment above historical O&M costs for that plant (if currently operating). Further, the differences in O&M costs for conversion of an existing reactor is not simply an incremental increase resulting from the technical differences of using plutonium (MOX) fuel rather than commercial uranium fuel, (given that the plant would have to be modified for the mission). That may be only the minor portion of the O&M cost differences. In fact, conversion of an existing reactor to a PDR would involve both increases and decreases in O&M costs relative to current performance.

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A potentially significant decrease in O&M could result from the change in ownership and management of the plant. The opportunity exists to make a step change (reduction) at the time ownership changes. Various constraints force current plant owners to achieve their O&M cost reduction goals over longer periods of time. These constraints include a host of labor relations, contractual obligations, regulatory concerns and so forth. However, for example, new ownership / management could immediately reduce the initial PDR staff considerably below what the existing reactor staffing may have been at the end of its operation. In general, the degree to which O&M might be reduced depends on the extent of progress already made by the current owner. If the plant exhibits industry leading (low) O&M, the potential for O&M cost reductions may be limited. Conversely, if the plant exhibits average O&M, it may have significant potential for cost reductions if converted to the plutonium disposition mission.

The degree to which management practices can provide an improved (lower) O&M is dependent on ownership, operational control, regulatory authority, the need for international safeguards inspections, and so forth. The operational role of the government versus the private industry is of particular note, since it could be argued that a government operated facility cannot achieve as low an O&M for the PDR as for private enterprise. Thus, on a generic basis for any government operated PDR, some cost penalty above that applicable to private/commercial operations should be applied.

O&M costs may also be reduced as a result of capital improvements made when the existing reactor is converted. The O&M cost reduction features that are already included in the System 80+ design may to some limited extent be provided as backfits to an existing reactor. This presents an important tradeoff of initial (conversion) capital costs versus potential O&M cost savings over future years of plant operation.

6.3.3 Baseline for Estimating Existing Reactor O&M Costs

It is a given that the next generation of commercial nuclear plants must improve safety and reliability over the current generation designs, while at the same time incorporating the *requisite features and staff organization necessary to further reduce O&M costs*. ABB-Combustion Engineering designed the System 80+ Standard Plant in accordance with the EPRI ALWR Utility Requirements Document. System 80+ is an *evolutionary* ALWR based on proven concepts of current nuclear power plants with additional design features and capabilities. Moreover, its design reflects insights gained from the experience of the nuclear industry in the U.S. and internationally. This design was the basis for the Evolutionary Reactor Phase PDR.

Similarly, as the System 80+ design is *evolutionary*, the organizational structure and staffing requirements were an *evolutionary* development. Such developments were reflected, for example, in the assumptions for System 80+ Conduct of Operations. Utilizing those assumptions, all of the major work functions in the plant were identified and an organizational structure was developed to support these functions.

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Thus, System 80+ has the design features, organizational structure, staffing requirements, and associated assumptions that contribute to reduced O&M costs. Clearly, the minimum O&M costs for System 80+ should be less than any current plants will ultimately achieve. Some O&M cost reductions may be possible relative to historical costs, just as the commercial nuclear industry is currently tending toward lower O&M costs. However, existing plants simply do not have the needed features to minimize O&M - either the features that directly contribute or those needed to fully optimize staffing and plant operational improvements.

The degree to which an existing plant's O&M may exceed that of the Evolutionary Reactor Report estimate depends upon the degree to which it differs from System 80+. Thus, System 80 plants (i.e., Palo Verde) would be expected to show the least increase, while older designs would exhibit larger additional costs relative to the Evolutionary Reactor Report estimate.

On this basis, it is recommended that the O&M costs of a PDR converted from an existing System 80 reactor be estimated based on deviations from the Evolutionary Reactor Report estimate rather than on the basis of current operating experience. A detailed assessment for the purpose of this estimate has not been possible within the scope and timeframe of this report. Moreover, detailed plant design and operational information is needed to properly assess the O&M costs. Without the active participation of the current plant ownership, this imposes a large uncertainty in any estimates. The additional O&M costs are made still less certain since it is not known to what extent System 80+ features would be backfit to the existing reactor. These factors result in the need to provide a judgmental estimate based largely on qualitative factors and known differences in design features. Additional costs might be assumed for the plutonium disposition mission given that some of the System 80+ design features cannot be included and others may not be economically justified as a backfit.

Similarly, it is recommended that for earlier designs (i.e., prior to System 80) that the O&M cost estimate be based on the upper quartile (best) O&M cost performance of similar commercial plants plus an adjustment for additional costs specific to the plutonium disposition mission.

6.3.4 Existing Reactor Design Differences from the Evolutionary Reactor Report

During the Evolutionary Reactor Phase of the PDS, the System 80+ Cost Savings Features Database was used as supporting tool in developing the staffing estimate. This database (still under development) was useful in identifying numerous unique System 80+ design features that help reduce O&M costs. Reductions may result from an increase in plant reliability (or availability/capacity factor), a reduction in outage time, or a reduction in staffing. The design features were evaluated for the impact on the work functions and consequential impact on staffing. Many of these features are not present in either System 80 or prior designs. This includes:

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- An improved architectural design that incorporates permanent scaffolds and strategically placed support facilities, and utilizes structural boundaries to eliminate HVAC ductwork where possible.
- Plant safety systems (Safety Injection System, Emergency Feedwater System, etc.) designed to eliminate common headers and use direct vessel injection flowpaths.
- A full featured Information Management System (IMS) developed as an integral part of the design process.
- Increased containment space, especially increased floor space at the operating deck level, allows outage work to proceed with minimal interferences.

Absence of these features are expected to be the principle basis for additional O&M costs for the existing reactors.

Existing Reactor O&M Estimate Recommendations

As a first approximation, it may be assumed that all reasonable backfits are provided at the time of conversion and that no penalty is attached to government ownership or influence on operations. Similarly, management practices are assumed to be the best possible options given industry experience, facility design features and plutonium disposition mission. Based on the preceding discussion and these assumptions, a judgement as to the reasonable O&M estimate may be made.

Industry trends are illustrated in the accompanying figures. Figure 6-1 shows the annual non-fuel O&M costs for one unit plants, two unit plants and the average for 1987 to 1993. The differences in best quartile performers and the averages is not apparent in such aggregate measures. Figure 6-2 shows the trend for a select group of plants (ABB-CE plants plus WNP-2) for 1988 to 1993. Similarly, the considerable variation in annual costs resulting from differences in number of units and total generation tends to mask other factors. As noted previously, considerable study and insight is needed to assess O&M cost data.

It is interesting to note in Figure 6-3 that the annual non-fuel O&M costs for the select group average has stabilized. Another perspective on this group of plants is provided by Figure 6-4, which presents the non-fuel O&M costs on a per KWe basis. The costs seem to have stabilized on this basis, although this measure is less useful than production costs (\$/MWh).

Figures 6-5 and 6-6 illustrate the select group individual and average production costs (dollars per kwh). Production costs are a better measure than costs per installed KWe since additional O&M is desirable if it leads to a greater than proportional improvement

For designs prior to System 80, the O&M cost might be judged to be in the range of perhaps 30% above the best quartile of similar plants. On the basis of production costs, this would imply that annual O&M is less than \$14.43 per MWh.

Figures 6-7 and 6-8 illustrate the annual non-fuel O&M costs and non-fuel production costs for Palo Verde (all units). Again, the trend is toward lower O&M costs over time and relatively stable production costs. Production costs for Palo Verde appear to be about 6% lower than the select group average.

Table 6.3.2-1 of the Evolutionary Reactor Report estimated a single unit annual non-fuel O&M of \$77 million and \$125 million for two units. For System 80 designs, the annual O&M costs may be in the range of 25% additional over the Evolutionary Reactor Report estimates. This corresponds to about \$96 million for a single unit, remarkably close to the average plant annual O&M expenditures of approximately \$100 million per unit in 1993.

6.4 Other Life Cycle Cost Detail

The ORNL letter memo, *Weapons Plutonium Disposition Study: Economic Factors for Disposal in Existing Reactors*, (April 26, 1994), lists factors to be considered in the economic evaluation. Those factors are discussed below. In addition, the related topics of generation capacity, capacity factor and D&D costs are addressed.

- **Capital Costs for Core Modifications and Changes to the Reactor/BOP to allow for Pu Use**

Secure MOX storage building (unirradiated fuel). The present plan is to store the unirradiated MOX fuel in the spent fuel storage pool, as discussed in Section 2.5. The costs of a secure MOX storage building (4000 ft², no licensing or safeguards costs) is about \$2.3M.

Additional Spent Fuel Storage Space. As discussed in Section 2.5, this space may be required if there are no off-site repositories available. The cost of removal of the current owner's fuel is one of the unknowns in the acquisition cost of the plant by the Government. Poison racks for increasing capacity of spent fuel storage would cost about \$7.6M for 2,650 cells. A new 3000 ft² storage building would be about \$5M.

Additional office/changehouse space for security guards and staff. A security guard increase of 15 guards for a single unit plant has been estimated. Office and change space should run about \$400K for 2000 ft² building.

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PIDAS fence around reactor site. This item, if not already installed, could be a significant cost adder. Estimates for a single System 80 unit (8200 lineal ft), are about \$4.5M.

Redesign of Core Structures, Fuel Handling Equipment, etc. for plutonium cores. No major costs are involved here, unless a modification to increase the plutonium throughput is required, in order to decrease the mission time. Modifications in that case are discussed in Section 2.1 and costs in Section 6.2.

- **Projected Change in Annual Operating Costs as a Result of Pu Use.**

Additional Fuel Inspection Costs. Receiving, handling, and inspecting new (i.e., fresh) MOX fuel would differ from that for new commercial (UO_2) fuel assuming that there is a significant radiation field for the new fuel. Procedures related to new MOX fuel could be similar to existing procedures for spent commercial fuel. For example, all movement and inspection would be carried out under water using remote equipment and cameras. This would cause only a small increase in the O&M costs compared to an existing plant's O&M. This is accounted for in the Evolutionary Reactor Report estimate. Procedures for handling and inspecting spent MOX fuel would be nearly identical to procedures dealing with spent UO_2 fuel. Therefore, no additional costs would be incurred for handling spent MOX fuel.

Safeguards & Security. Additional security measures would need to be in place for receiving MOX new fuel and for securing the new fuel storage areas. This may include additional hardware and additional security personnel. It is estimated that 15 additional security guards for a one-unit plant and 28 additional security guards for a two unit plant would be required to supplement the security staff of an existing plant. This would result in an approximate increase in annual O&M costs of \$750,000 for a one-unit plant and \$1.4 million for a two-unit plant. This is accounted for in the Evolutionary Reactor Report estimate.

NRC Inspection Costs. It is estimated that no additional costs would be necessary associated with additional NRC inspections.

Staffing and ES&H Requirements During Refueling. It is estimated that only minor staff increases would be required during refueling operations. This is accounted for in the Evolutionary Reactor Report estimate.

Training for Reactor Operators. Additional training would be required to ensure operators are familiar with the differences in operating characteristics between UO_2 and MOX fuel. This would include changes to nuclear theory fundamental training, procedure training and simulator scenarios. It is estimated that only minor increases in training duration would be required. No additional instructors or additional lengthy classes would be required. This is accounted for in the Evolutionary Reactor Report estimate.

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Procedures for Fuel Acceptance and Spent Fuel Turnover. New procedures and/or rewrite of existing procedures would be required for handling MOX fuel. However, after the change-over is complete, no additional O&M costs would be incurred from procedural changes. The actual processes used for fuel receipt, identification, storage and so forth would be different if the new fuel radiation levels are significant. This would result in a small increase in O&M costs. However, the spent fuel operations are not expected to result in any significantly different processes than are already in place for commercial fuel. Therefore, no increment in O&M costs are expected.

Insurance costs. It is difficult to estimate how the insurance companies would react to such a change in the operating characteristics of the plant and thus the risk of the plant. Technically, the basis for insurance should be little affected. Thus, the Evolutionary Reactor Report estimate is considered a reasonable prediction of total insurance costs.

Annual Costs for Control Materials. The only major material that would significantly change is boron. A MOX fuel burning plant would require higher boron concentrations. This may favor the use of enriched boron which, when initially purchased, would add to the conversion cost. This is covered in Section 6.2. However, during subsequent normal plant operation most of this boron would be recycled and used again. The only additional costs incurred would involve replacing lost or neutron-saturated boron. This cost would be very minor.

- **Plant Life Extension Costs**

Plant life extension is not necessary for the existing ABB-CE plants under consideration.

- **Projected Change in Capacity Factor**

The trend in the industry is in the direction of increasing capacity factors, through improved scheduling of downtime, and maintenance schedules. This is discussed in Section 4.4. No significant increases in downtime have been identified because of the use of MOX fuel. Inspections by the IAEA would likely focus on the paperwork, for ensuring that no fuel has been diverted, and would be concentrated in the fuel storage building. There would be no reason for them to slow down the actual transfer and loading process. In addition, the actual refuelling activities are normally off the critical path of a planned outage.

- **Reactor Downtime for Converting to Pu Use**

In general, a detailed schedule for conversion to Pu use has not been established. The cost of the downtime would depend on whether the unit in question had been taken off the utility rate base at the time that conversions were made. If ownership of the reactor were transferred to the Government prior to the conversion process, which

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might well be a utility requirement, there would be no "replacement power" cost to consider.

- **Schedule and Costs for EIS, Relicensing, Permitting, etc.**

The cost of relicensing, assuming a best case, no intervention scenario, is given in Section 6.2. Other activities, such as a revision of the EIS, are estimated at \$750K. The schedule is estimated at 12 months for relicensing, except for WNP-3, which would require operating permit licensing, and a schedule of three years is estimated..

- **Cost of Modification of Spent Fuel Storage Pools**

The major modification involved would be if a new facility were required. The existing fuel storage pool is adequate to contain the spent MOX fuel, which should have characteristics very much like spent UO₂ fuel. Section 2.5 discusses the storage of spent fuel in the existing Palo Verde fuel storage pool. Relicensing costs would be about \$300K.

- **Additional Change in Reactor D&D Costs**

D&D costs for commercial nuclear plants are uncertain, but constitute a relatively small portion of the life cycle cost. As such, life costs are insensitive to relatively large changes in D&D costs. The methodology used in the Evolutionary Reactor Report (based on ORNL guidance) provides an acceptable estimate. D&D costs are not expected to be any different for the PDR than for an existing reactor.

- **Electrical Generation Capacity**

No reactor downrating due to the use of MOX fuel is anticipated. Therefore, electrical generating capacity should not be affected.

- **Subsidies to the Utility from the DOE**

The question of the cost and arrangements for the acquisition of an existing plant are the most formidable and represent the greatest uncertainty. Some of the major cost items have already been touched upon in other sections, notably Section 4.1. As a general principle, it would seem that the DOE should count on absorbing all the costs on the part of the utility in terms of legal expenses, removal of the reactor from the *utility rate base, termination of existing contracts for services and supplies, including fuel supply, uranium, separative work, etc.*, and costs of any change in the power purchase agreements brought about by a change in ownership. All licensing costs and additional operating costs, including security costs, would be to the Government's account. Any cost involved that would allow the PUC to view the program in a favorable light would also be borne by the Government.

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If the Government were to negotiate an arrangement whereby the utility would keep ownership of the plant, control its operations but just contract with the Government to accept MOX fuel, there might be fewer problems involved. Assuming that the fuel is supplied to the utility free, there would be an adjustment in the PUC negotiated rate due to fuel use charge, that would be passed through to the consumer. This may be an advantage to the utility in the overall lowering of production costs, and the ability to avoid having its large customers obtaining power from other sources. This may be sufficient incentive for a utility to go along with the arrangement, provided the Government indemnified the utility owners and others in the event of unforeseen problems, as well as termination costs in existing fuel supply arrangements. Even then, situations such as negative shareholder reaction and loss of image could be damaging to the utility, and additional compensation may be demanded. Absorption by the Government of other costs, such as a cost of a major outage, including major equipment upgrades (steam generator replacement, for one) might be sufficient enticement to a utility facing such a situation.

- **New Transportation Requirements**

The major cost uncertainty in transportation is the cost of new fuel transportation to the reactor site, from the DOE-owned fabrication facility. The use of SST trucks is probably feasible to transport new fuel assemblies of the design required by the existing plants under consideration for the plutonium disposition mission. Fuel assemblies are normally transported in casks that provide mechanical support to the assembly, and which contain accelerometers to track the shock forces experienced by the assemblies en route. The new fuel casks will require some modification to incorporate shielding to account for the radioactivity of the MOX fuel. This should not be a major expense, however. The need to transport up to 220 fuel assemblies per year, using SST equipment, should be reviewed by others that have knowledge of the availability of such equipment.

Spent MOX fuel transportation presents no differences in comparison with spent UO₂ fuel transportation. Casks and cask designs are available that are suitable. The questions of the numbers of casks available, the cost of transportation to a repository, and the availability of repository space have yet to be answered.

- **Development Cost**

The development cost of fuel will be the same as that reported in the Evolutionary Reactor Report. There is a possibility that a demonstration assembly would be required for insertion into the existing reactor selected for the mission. This could reduce the cost of the program somewhat, if by doing so there would be no need for irradiation in a test reactor (ATR) as well.

There are no other development costs identified for the use of MOX fuel in existing ABB-CE reactors.

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- **MOX Fabrication Facility**

The MOX fabrication facility for the existing reactor case is assumed to be the same as that in the Evolutionary Reactor Report. Costs for a stand-alone facility at a Greenfield site are somewhat greater, because of the unabsorption of indirect costs, that were absorbed in the case of the collocated facilities. These are estimated to be about \$21 Million.

The schedule for the construction of the MOX fuel facility for the Greenfield site would be longer by about 12 months, and the size is assumed to be the same as in the Evolutionary Reactor Report. The fuel design is essentially identical in the selected existing reactors as in System 80+. Differences in plutonium enrichment per assembly (System 80+ can load more weapons plutonium per assembly than can be loaded in existing reactors, without modification) will increase the plant throughput, in terms of MT heavy metal per year, but the facility design is such that these increases can be handled by multi-shift operation.

6.5 Anticipated Cost Risks (Qualitative)

Apart from the cost risks mentioned in Section 6.4 having to do with the acquisition by the Government of the reactor, the indemnification of current interested parties, and the licensing risks, the other cost risks reported in the Evolutionary Reactor Report apply to the case of existing reactors. These include MOX fabrication costs and D&D costs. Other cost risks involve the modifications to the plant, if such are required in order to improve the mission schedule. These modifications were outlined in Section 2.1.

The primary area of cost and schedule risk for the addition of CEDM nozzles or instrumentation nozzles in the reactor vessel top head is the field installation procedure. The risk consists of two elements. First, and causing the most impact, is the down-time for an operating plant. Loss of revenue during a shutdown can approach a million dollars per day. The CEDM nozzle/instrumentation nozzle installation may or may not be critical path. If it is, the duration for installation is difficult to predict. The duration estimate would be based on the number of nozzles to be installed and the practice installations performed on a mock-up. But, as with all field installations, there is a good possibility that durations of installation will be longer than predicted. Durations and costs of installation can be increased by hole machining errors, by previously undetected flaws in the head, high radiation fields which limit personnel exposure times and requires additional personnel due to rapid burn-out rate, and possible conflict with other work going on during the outage.

The second element of the cost and schedule installation risk is manpower needs escalation. The cost per man in the field per day is significant especially when you consider the training on a mock-up and other preparations necessary before ever going to the site. The manpower costs can escalate and the schedule can be delayed due

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to delays caused by other outage work, higher radiation fields than predicted, unexpected problems in installation, mistakes made during installation, unexpected early burn-out of personnel, and availability of site personnel. The cost of the CEDM nozzles and instrumentation nozzles themselves, that is, the part cost, is not high compared to the installation cost and can be predicted fairly accurately. The part cost is not a risk.

6.6 Base Schedule

No base schedule has been developed for the case of existing reactors, since the driving forces will undoubtedly be the unknown negotiations involving the utility, the PUC, interested parties, the NRC and the EPA. From a technical point of view, the schedule driver is the fuel fabrication facility. In this case, the schedule prepared for the Evolutionary Reactor Report, with first MOX operation at Month 80, applies.

6.7 Anticipated Schedule Risks

The anticipated schedule risks reported in the Evolutionary Reactor Report, Section 6.7, apply in the case of existing reactors.

Figure 6-1
INDUSTRY TRENDS
ANNUAL NON-FUEL O&M COSTS

Figure 6-3
CE GROUP AVERAGE
ANNUAL NON-FUEL O&M COSTS

Figure 6-4
CE GROUP AVERAGE
NON-FUEL O&M COSTS PER INSTALLED CAPACITY (KWe NET)

Figure 6-5
CE GROUP
NON-FUEL PRODUCTION COSTS

Figure 6-6
CE GROUP AVERAGE
NON-FUEL PRODUCTION COSTS

Figure 6-7
PALO VERDE NUCLEAR GENERATING STATION
ANNUAL NON-FUEL O&M COSTS

Figure 6-8
PALO VERDE NUCLEAR GENERATING STATION
NON-FUEL O&M PRODUCTION COSTS

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7.0 SAFEGUARDS AND SECURITY

7.1 Fuel and Fabrication facility

The safeguards and security requirements for the fuel fabrication being considered under the Plutonium Disposition Study using existing facilities will not differ significantly from the requirements that were established for a new facility as described in the Evolutionary Reactor Final Report. The use of existing reactors to burn plutonium will increase the distance that the new MOX fuel will have to be transported. If the fuel is manufactured in the FMEF facility at Hanford, and transported for use in the Palo Verde reactors in Arizona, for example, the fuel will have to be transported about 1200 miles over existing highways. The Safe, Secure Trailer (SST) system that is being used to transport missiles and plutonium pits around the country could be used to haul the new MOX fuel to the reactor site. Using the SST system to transport the fuel will be more expensive than the transportation of normal low enrichment uranium fuels but should not be prohibitive. Facilities that are used to fabricate plutonium fuel may be DOE facilities and, therefore, will also have the meet the requirements of DOE orders 5633 and the 5632 series. Where these two sets of requirements conflict, the most restrictive requirement will be used. This will ensure that all minimum safeguards and security requirements are met. Physical protection for the Mixed Oxide Fuel Facility will have the same requirements as identified in the Evolutionary Reactor Final Report.

The MC&A system, for the MF², that was described in the Evolutionary Reactor Final Report will be the same system used if the fuel is used in existing reactors. The transportation of the fuel from MF² to the reactor site will be more involved. Also, the MC&A plan will require some modifications to the existing reactors in order to meet storage restrictions.

7.2 Reactor

The use of weapons grade plutonium fuel in an existing reactor will require some upgrading of the physical security system and some additions to the material control and accountability system. The use of plutonium fuel in an existing light water reactor will require a license modification by the Nuclear Regulatory Commission (NRC) and the facility will have to meet the applicable safeguards and security portions of 10 CFR 70, 72, 73, 74 and possibly 75 and 95.

7.2.1 Security

The existing nuclear reactor facility, designed about 20 years ago, and may or may not have been through a recent upgrade with replacement of outdated equipment. The systems at the existing reactor will be evaluated in relation to the present design threat in order to identify any weaknesses. This evaluation will include systems analysis, vulnerability analyses and performance checks where needed. Both insider, outsider

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and combination insider-outsider threats will be considered. Systems will be required to provide protection from sabotage (both industrial and radiological), espionage, theft or diversion of SNM, loss or theft of classified material or government property, and other hostile acts. The physical security system that is required at the reactor facility includes technological detection, control, and containment measures as well as protection methodologies and procedures, that apply to the protective forces and security programs.

The physical protection system consists of a series of sub-systems (i.e., access control, barrier denial, isolation zone, intrusion detection, assessment, communication, SNM detection, x-ray, security posts and response force). The existing reactors have some form of these sub-systems except the SNM detection system. The existing sub-systems will be evaluated to ensure the concept of "defense-in-depth" is maintained and provides concentric lines of defense. New SNM detection equipment will need to be strategically placed to reduce the potential for theft.

The main and auxiliary power sources for the security system will be located in secure buildings which preclude easy access. Computer facilities will be protected through a series of security systems and procedures.

7.2.1.1 Delay and Denial Systems

Delay and denial of access to key security interests, by unauthorized personnel, at the existing reactors is accomplished through a system of barriers and access control points to *deter and neutralize sabotage attempts and to contain attempts of theft*. The existing facilities use a variety of passive and active barriers that include fences, buildings with locked doors, vaults, secure repositories, and expanded metal cages to delay or deny entry by unauthorized personnel. Vulnerability analyses and/or performance checks, against the Generic Threat Policy, will be used to evaluate the effectiveness of the barriers to meet the requirements of a reactor site using plutonium fuel. These analyses and performance checks will identify any weak points in denying or delaying an adversarial attempt to reach their target. Depending on the extent of the weak points that are identified, additional or improved barriers may be installed, the guard force may be increased or an exception may be requested.

The selected reactors have multiple systems that are capable of shutting down the reactor, maintaining the water to cover the reactor core and removing heat to prevent melting the fuel. These systems or "trains" are separated by barriers to prevent a fire in one area from affecting more than one train. This "train" separation and fire barriers help reduce the vulnerability to sabotage, but some additional barriers may be needed. Any additions of doors and/or barriers will require a review to ensure that the plant operations and safety are not adversely affected.

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7.2.1.2 Entry Control Systems

The system to control the entry into the plant and to some specific locations within the plant will not change significantly from existing systems. System upgrading, similar to what some reactor sites have done, could meet the security requirements with a much smaller guard force. Exit portals will require SNM detectors to help prevent insider theft of SNM.

7.2.1.3 Protective Force Support

The protective force for the reactor site will be need to be evaluated through a performance test to ensure that they are capable of meeting the present day requirements.

7.2.1.4 Intrusion Detection System (IDS)

Existing nuclear reactors have intrusion detection systems that are designed to detect any person crossing the restricted zone on the perimeter. Performance tests will have to be run to determine the system detection limits. The system will have to be extended to include vital areas and the vault-like locations that will be used to store the new plutonium bearing fuel until it is placed in the reactor. The intrusion detection system will be operated and maintained so the false alarm rate does not result in a degraded response.

7.2.1.5 Vital Areas

Present reactor sites have identified some areas as vital (e.g. the emergency diesel generator buildings). A vulnerability analysis will be used to determine if any additional areas need to be added to the list of vital areas. All vital areas will be contained within the protected areas and will have clearly defined perimeters. Access to a vital area will be controlled so entry is limited to appropriately cleared or escorted individuals who require admission to perform their official duties. An approved access list will be kept current. Records will need to be maintained of all persons admitted to vital area who are not employed directly in operations that require access to the vital area. Escorts, when required, will be cleared and authorized persons knowledgeable of facility operations and safeguards and security procedures.

7.2.1.6 Vehicle Entrance and Trans-ship Facility

Vehicles entering a protected area of a reactor site present a difficult inspection problem for the guard force, and could be used to smuggle contraband into the facility and SNM out of the facility. The construction and use of a tranship facility can nearly eliminate the need for any vehicles to enter the protected area with exception of emergency vehicles. These vehicles will be followed into the protected area by the Site Guard Force (SGF) as a way of preventing contraband from being brought into the

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protected area during an emergency. Materials and supplies that are needed for operation of the facility will be brought to the outer side of the transship facility and unloaded. The material will then be inspected and passed through the barrier into the protected area, where the material is either stored until needed or picked up and delivered to the job site.

7.2.1.7 Material Access Areas (MAA)

The portion of the Spent Fuel Storage Facility (SFSF) and the RBC that can receive or store unirradiated MOX fuel will be designated a MAA. Each MAA will be defined by physical barriers and will be contained within the PA. Since the unirradiated fuel will be on site for about 3 months out of every 18 month period, a temporary MAA may be established while the unirradiated fuel is out of the reactor. When the fuel is in the reactor and the head is in place and secure, the MAA would be removed. While the MAA is in place, security will comply with DOE Order 5632.2A requirements. MAA zoned areas will have access and exit control devices.

MAA access control systems will control entry to appropriately cleared or escorted people who require admittance to perform their official duties. Access to MAAs will be through doors or hatches of substantial construction and installation that will be lockable or otherwise secured to offer penetration resistance and impede both surreptitious and forced entry.

SNM entering the MAA at the reactor site will be in the form of fuel elements and will be under Item Control Accounting. An SNM monitor will be located at exit areas, CCTV cameras will provide visual observation within the MAA. A metal and SNM detector will be located at the MAA egress and ingress area. Emergency lighting will provide illumination during power interruptions.

7.2.1.8 Material Balance Area (MBA)

MBAs will be located in the SFSF and the RBC in appropriate locations to aid in the control and accounting of SNM.

7.2.2 Material Control And Accountability

7.2.2.1 Purpose

The MC&A system, in combination with the physical protection system, has three major purposes: 1) to provide assurance that nuclear materials are present in correct amounts and location in the facility, 2) to detect and deter unauthorized movements of nuclear material within the reactor facility or unauthorized removal of nuclear materials from security areas of the facility, and 3) to deter radiological sabotage resulting from adversarial acts involving the unirradiated and irradiated fuel materials.

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7.2.2.2 Scope

The MC&A system will be an element of the integrated safeguards and security system at the reactor site. Its role is to control and account for the nuclear fuel at the facility. The safeguards and security system will provide protection against the Design Bases

combinations of insiders and outsiders. It will be a requirement of the operator of the reactor to protect and safeguard, pursuant to the Atomic Energy Act of 1954, as amended, the nuclear materials that will be stored and used at the reactor site.

An MC&A plan will be prepared describing the program that will be used at the existing reactor site to comply with the requirements in DOE Orders and applicable portions of the code of federal regulations. The plan will reflect the requirements specific to the reactor site including threat considerations, performance criteria, reporting requirements, the accounting system, physical inventories, detection and assessment capabilities, access control, material containment, material surveillance, and MC&A capabilities during emergencies.

7.2.2.3 General

Nuclear material accounting at the reactor site will need to be updated to conform to the latest requirements. The system will provide for tracking nuclear material inventories; documenting nuclear material transactions; issuing periodic reports; and verifying, detecting, and evaluating loss detection elements.

The nuclear material content in the reactor and the SFSF will be based on accountability measurements made at the fuel fabrication facility. Since the receiver at the reactor facility must sign for the amount of SNM that is received, he will have to be assured of the measurement accuracy at the MF² and may station an employee at the fabrication plant. Reactor burnup of Pu²³⁹, will be calculated using operating data.

Because the fuel elements and fuel assemblies have the characteristics of sealed sources, attempts to remove SNM would leave obvious evidence. The material tracking and inventory systems as well as the accounting function at the reactor site will rely on item control accounting strategies and methods and may use a nondestructive analysis technique to confirm the content of the fuel assemblies. Very reliable and durable processes will be required for uniquely identifying the fuel assemblies. The identifying markings will have to remain legible after intense irradiation in the reactor vessel and long-term storage in the SFSF. The legibility will also have to survive corrosion and scale buildup resulting from long-term contact with water.

Special nuclear materials (SNM), MOX, will be located in two areas at the reactor site: the Reactor Building Complex (RBC), and the Spent Fuel Storage Facility (SFSF). Category I quantities of SNM will be stored and used only within the Material Access Areas (MAA) to be defined within the reactor site (e.g. RBC, and the SFSF). All areas will be located within the Protected Area.

7.2.2.5 Accounting System

Nuclear material accounting at the reactor site will be accomplished with a near-real-time computer-based system that will track all fuel assembly transfers into the reactor site, and maintain records of nuclear material inventories including location changes.

To meet the near-real-time requirement, the accounting data base will be updated to reflect fuel transfers when the transfers are made or as soon after the transfer as is practical but no later than the end of the current work shift. This will enable the system to generate a book inventory listing within 3 hours, which is required for all SNM. Data entry will be performed at remote work stations by authorized personnel.

Access limitations to the MC&A computer data system will be implemented to ensure that only authorized persons make adjustments to the accountability data base. Adjustments to the accounts will be reviewed and approved by an authorized person in the MC&A organization. Where operations personnel enter MC&A-related data at remote work stations, overchecks will be implemented to detect any attempt at data falsification.

7.2.2.6 Material Access Area (MAA)

The existing reactors have the RBC located within a fenced area that should meet the requirements of a protected area. An MAA will have to be established in the RBC that will enclose the area in which Category I quantities of SNM are located. Category I quantities of plutonium would be present when new fuel arrives on site. The MAA will include the new fuel receiving and storage area, the spent fuel and refueling portion of the containment building when fuel is being transferred into containment and installed in the reactor vessel. The MAA will need to be established from the time the new fuel arrives until the fuel is placed in the reactor and the vessel head is reinstalled and torqued in place. Removing the MAA status when it is not needed makes the movement of personnel and non-SNM movement is much easier with the reduced access controls and portal monitors.

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The boundaries of the MAA will be defined by a physical structure. The personnel ingress/egress points will be limited to the minimum number required for efficient operations and personnel safety. Access will be limited to properly cleared, authorized persons having work activities in the MAA and to escorted persons needed to perform official functions.

7.2.2.6.1 Access Controls

The purpose of personnel access controls is to ensure that only authorized persons have access to the MAA. This controlled access reduces the potential for theft and diversion of nuclear materials. The physical security system will provide the personnel access control. In addition, access controls within the MAA will be placed on the plutonium storage location.

All category I quantities of SNM are used and stored only within the MAA.

7.2.2.6.2 Material Surveillance

location of plutonium bearing fuel from the time the fuel assemblies arrives at the RBC until they are shipped to the permanent storage sites.

7.2.2.6.3 Daily Administrative Checks

When the MBAs at the reactor site contains Category I quantities of nuclear materials a program of daily administrative checks will be implemented. This program is intended to detect obvious abnormalities or missing items and evidence of tampering, and to provide assurance that theft or diversion of Category I nuclear material is promptly detected.

If an anomaly is discovered during the daily administrative check, attempts will be made to resolve it. If not resolved within a designated time from the discovery, the MC&A Manager and cognizant reactor management will be notified and an investigation will be initiated.

7.2.2.7 Material Balance Areas (MBA)

The reactor site will need to establish two MBAs to provide physical and administrative control of the MOX fuel. One of the MBAs will be the SFSF, which includes the new fuel receiving and storage area and the spent fuel pool, and the other will be the reactor containment building.

Internal transfers of fuel assemblies will require prior authorization. The amount of nuclear materials moved in or out of the MBA will be based on item identity, item count, and verification of the physical integrity of the individual items in the fuel

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assembly for both reactor containment and the SFSF. (The quantity of nuclear material in each item will be based on previously measured values.)

The plutonium in the fuel assemblies will be reduced during the exposure in the reactor and the isotopic mixture will change. The amount of plutonium and the isotopic content will be calculated for each fuel element and entered into MC&A database. This information will be used to identify the amount of plutonium that was "used up" in the reactor and the amount of plutonium that will be stored in the spent fuel pool and eventually in the permanent storage. The amount of plutonium in the spent fuel is also needed for criticality calculations for the spent fuel pool and the permanent storage.

7.2.2.8 Nuclear Material Control

The overall purpose of the nuclear material control program for the existing reactors is to ensure that the status and physical location of all nuclear materials in the inventory are known and are protected consistent with the graded safeguards concept. The program has elements in common with the physical security system. In general, the hardware design and operation aspects of the program (not including MC&A computers and instrumentation) are the responsibility of the physical security system, whereas procedural aspects are the responsibility of the MC&A system.

The existing reactors have a material control program in place. This program will require revision to include the controls for the MOX fuel. System upgrading is expected to include: the new fuel storage will need to be brought up to the standards of a "vault type" room, and SNM detectors capable of detecting plutonium will be needed.

7.2.2.9 Material in Storage

The storage locations in the SFSF will be treated as special control areas. Administrative controls will be established to detect unauthorized activities within the storage locations. The two-person rule will be enforced during all activities, including transfers in or out, physical inventory, and maintenance and repair of equipment.

The new fuel assembly storage area will need to be upgraded to a vault-type room that meets the Category II requirements. In addition, procedures will be implemented to control and validate material movements into and out of the storage locations.

Irradiated fuel assemblies will be stored in underwater storage areas in the SFSF. Because of the intense radiation associated with these materials, the nuclear materials are considered to be self-protecting from the standpoint of theft or diversion. However, due to the potential for radiological sabotage, access controls will be established at the entry points to the facility. The building will be equipped with an intrusion alarm system. When the building is unoccupied, it will be locked and the

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intrusion alarm system will be in operation and monitored at the central alarm station. The existing reactors have some form of access controls on their spent fuel storage facility. These existing controls will be reviewed and upgraded if necessary.

7.2.3 Reactor Building Complex (RBC)

The existing reactor sites have a security system in place including armed security guards. The reactor buildings include the Reactor Containment Building, the Auxiliary Building (with the control room) and the Turbine Building. These buildings are all located inside the fenced protected area that is equipped with an intrusion detection system. The primary access and exit to this complex is one controlled area, and a tamper detection system is used to provide intrusion detection for access locations that are not routinely use. The addition of radiation detection scanning equipment, and x-ray equipment will be needed to aid in detecting unauthorized removal of SNM from the RBC.

Fuel that has been irradiated in the core of a reactor is highly radioactive and cannot be handled except under many feet of water or behind thick shielding walls. Therefore, irradiated fuel is not a likely candidate for theft or diversion. The unirradiated new fuel emits only low levels of radiation and can be handled with minimum exposure to the individual. A new fuel assembly arrives in the reactor containment building from the SFSF through a water canal, where it is picked by the refueling machine, remaining under water at all times, and is placed in prescribed location in the reactor core. During the refueling there are several operations type people involved and the new fuel is not left unattended. There is a low probability for covert theft during the refueling cycle.

The primary security concern for the reactor complex will be sabotage, either industrial or radiological. Design features, that will reduce the affects of any sabotage, include: 1) compartmentalization of the auxiliary building (some of this is in place but may need to be expanded) to reduce the potential for losing more than one safety system and thereby ensuring the capability for the safe shutdown and cooldown of the reactor, 2) limited access to the control room, (this is standard for nuclear reactors) 3) an auxiliary shutdown panel where the reactor can be shutdown if the control room is lost (nuclear reactors are required to have this feature, and 4) defendable station for control of access to the containment building (this feature may need to be added).

The security systems were originally designed many years ago and may or may not have been updated with new equipment. These systems will need to be reviewed and performance tested to ensure they meet the requirements. The low level of enrichment of uranium fuels that the existing reactors now use makes it unattractive for producing a bomb. Therefore the existing reactors would not be expected to have a significant material control and accountability program in place. An MC&A program will need to be developed before mixed oxide fuel can be brought into an existing reactor site.

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7.3 Spent Fuel Storage Facility (SFSF)

The spent fuel storage facility includes an area where new fuel is received and inspected, a storage area where the new fuel is stored until refueling time, a large-deep water pool to store irradiated fuel, and a water canal that connects the storage pool with the refueling canal in the containment building. This water canal permits the irradiated fuel to be moved from the reactor and into the storage pool without being exposed to the air. New fuel is moved from the water storage pool into containment by the reverse process. The SFSF is located in the PA, and the entrance and exit controlled by badge readers.

Theft or diversion cannot be ruled out, but a single fuel element weights 1000 pounds and is about 14 feet long. Unirradiated fuel is stored out of the reactor for only a short period just before the refueling cycle starts. For the Palo Verde plants, an 18 month fuel cycle is used, with about 1/2 of the core replaced during each refueling cycle. Unirradiated fuel is available onsite about 3 months out of 18 month cycle.

Remote control Closed Circuit Television (CCTV) cameras, which may not be part of the existing security system, will be use to observe the new fuel receiving area, the fuel assembly storage area and the refueling canal area where the new fuel is laid down in preparation for moving into the containment building during refueling. Emergency lighting will ensure illumination in case of a power failure.

Access and exit will be controlled and restricted to personnel for authorized work only. Intrusion detection and alarm devices will be used to detect unauthorized access. Security personnel will monitor the new fuel receiving area, the new fuel storage area and the spent fuel storage area with remote controlled CCTV equipment. Emergency lighting will be provided. Radiation detection equipment and x-ray equipment will scan personnel and containers to detect unauthorized removal of SNM from the area. Alarmed sensors will be used to detect the removal of SNM or the inadvertent spreading of contamination.

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8.0 ENVIRONMENT, SAFETY AND HEALTH

8.1 Nuclear Material Transportation

The environment, safety and health aspects of nuclear material transportation for the case where existing reactors are employed are similar to those reported in the Evolutionary Reactor Report, Section 8.1. Additional detail relative to actual transportation routes between the site of the MF² and the reactor(s) will need to be worked out.

8.2 Fuel Activities

The environment, safety and health aspects of fuel activities for the case where existing reactors are employed are similar to those reported in the Evolutionary Reactor Report, Section 8.2.

8.3 Waste Streams

The environment, safety and health aspects of the waste streams for the fuel cycle and fuel fabrication, for the case of existing reactors, are similar to those reported in the Evolutionary Reactor Report, Sections 8.3.1.1 and 8.3.1.2. Section 8.3.1.3 has been prepared based on the Palo Verde plants only.

8.3.1.3 Reactor Waste Streams

Waste minimization has received tremendous attention by the commercial nuclear industry since the time of the WASH-1248 report. This has been the result of numerous factors, including ALARA programs, improved technical capability for waste management, elimination of the sources of waste streams, etc. For example, the experience of the industry reflects a small failed fuel fraction (less than 0.015%). Thus, the source of waste streams originating from reactor operations is small. Similar results are expected for MOX fuel.

Low-level wastes resulting from reactor operations may include sludge, ion exchange resins, evaporator bottoms, aqueous filters, dry filters, contaminated rags, cleaning materials, disposable protective clothing, plastic containers and other materials that require disposal, not classified as either spent fuel or as by-product thereof.

The design emphasizes waste segregation. Waste segregation and hard piping of drains enables effective processing. Segregation by chemical and radiological characteristics decreases the quantity of water that must be treated due to unknown or suspected characteristics. Dedicated drains minimizes the use of floor drains and further reduces entrainment of dirt and debris that gives rise to filter problems. It also provides fringe benefits in reducing exposures, and reduces labor costs.

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Ion exchangers are used to the maximum extent, avoiding the relatively higher maintenance of evaporators and minimizing wastes, including solids from evaporators, liquids from flush lines and off gas stripping columns.

Other solid wastes are eliminated by the operating philosophy and design features in support thereof. For example, hot tool rooms permit storage of tools without the need for extensive decontamination (they don't leave the radiation control zone).

The liquid, gaseous, and solid waste management systems are designed to protect plant personnel, the general public, and the environment by providing means to collect, generate, store, process, sample and monitor radioactive wastes. These systems are designed to meet the limiting requirements on releases in 10 CFR 20, Appendix B; and 10 CFR 50, Appendix I (the ALARA Objectives). Waste management systems are not described in the generic System 80 CESSAR FSAR. Estimates of the maximum and expected activities in the primary coolant including tritium production are presented. Expected volumes of waste inputs to the liquid, gaseous, and solid radwaste managements systems are provided.

Palo Verde Nuclear Generating Station (PVNGS) Unit 1, 2, and 3 are operating versions of the standard System 80. Section 11 of the PVNGS Updated FSAR provides descriptions of the radwaste management systems. These are described below as representative System 80 handling of radwaste.

Liquid Waste Management Systems (LWMS)

The function of the liquid waste management systems is to collect and process radioactive or potentially radioactive liquid wastes generated during plant operation. The LWMS consists of the Secondary Chemistry Control System, (SCCS), Fuel Pool Cooling and Cleanup System (FPCCS), and the Liquid Radwaste System (LRS).

During processing by the liquid waste management systems, radioactivity is removed so that the bulk of the liquid is restored to clean water which is recycled for plant use. The radioactivity removed from the liquids is concentrated in filters, ion exchange resins, and evaporator bottoms. The concentrated wastes are sent to the solid radwaste system for packaging and eventual shipment to an approved offsite disposal location. There are no provisions or significant pathways for the release of radioactive liquids to the environment from PVNGS. All liquid waste is either processed and recycled for plant use or prepared for shipment in accordance with NRC and DOT regulations.

Input waste streams are segregated to facilitate treatment. Wastes containing a high degree of total dissolved solids (TDS), including wastes from the chemical waste neutralizer tanks, chemical drain tanks, and the auxiliary, containment, radwaste, and fuel building sumps, are collected for processing in the high TDS holdup tank. Wastes containing a low degree of TDS, including radioactive wastes from the turbine building,

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auxiliary steam condensate receiver tank, LRS adsorption bed, and recycle monitor tanks, are collected for processing in the low TDS holdup tank. Samples from the high and low TDS tanks are available by placing the tanks in a recirculation lineup and drawing a sample of the pump discharge. Turbine building drains are normally nonradioactive, activity being present only when there are primary-to-secondary leaks. To avoid processing more waste through the LRS than necessary, nonradioactive turbine building drains are processed by the chemical waste system. Besides the low TDS tank, an additional holdup tank is provided to accommodate overflow from either the low TDS or the high TDS holdup tank and is normally isolated from the supply headers. If necessary, this tank can be used to collect either low TDS or high TDS liquid waste. An internal mixing header uniformly mixes the contents of each holdup tank prior to and during processing. Acidic or caustic agents may be added for pH control, and antifoaming agents may be added if surfactants exist in the tank contents. Decontamination facility wastes from Unit 1 only and radio-chemistry laboratory wastes are collected in the chemical drain tanks prior to processing.

High TDS wastes are pumped directly from the holdup tank to the LRS evaporator for processing. The evaporator concentrates the waste up to 50 wt% total dissolved solids excluding neutralized boric acid which is concentrated up to 25 wt%. The evaporator concentrate is pumped to the concentrate monitor tanks and ultimately to the solid radwaste system. The concentrate monitor tanks are kept in a continuous recirculation mode while they contain radioactive concentrate. Samples are taken from an analysis point off the pump discharge. The distillate from the evaporator is passed through the LRS adsorption bed and mixed bed ion exchangers arranged in series and then is sent to the recycle monitor tanks. The monitor tanks' liquid content is then stored for eventual use as makeup for the primary coolant system, the secondary system, or the spent fuel pool. The monitor tanks' contents may also be sent back to the low TDS holdup tank should further processing be desired.

Low TDS wastes are pumped from the holdup tank through the LRS ion exchanger prefilter for removal of larger particles, through the adsorption bed for removal of organics, and through the two mixed bed ion exchangers arranged in series for removal of trace radioisotopes, to the recycle monitor tanks.

Wastes collected in the chemical drain tanks are normally pumped to the high TDS holdup tank for processing, but, if desired, may be sent directly to the solidification system for disposal.

Boric acid from the chemical volume and control system (CVCS), although normally processed through the boric acid concentrator and sent to the refueling water tank, can also be processed through the LRS evaporator should the boric acid concentrator become inoperable. When processing boric acid, the LRS evaporator receives CVCS flow from the CVCS holdup tank pumps. Concentrated boric acid is sent to the concentrate monitor tanks and then to the solidification system. Distillate is sent to

activity waste gas systems which filter waste gas prior to release to the atmosphere. The low activity waste gas systems are the building ventilation exhaust systems, the condenser air removal system, and the turbine gland sealing system.

The GRS collects and processes radioactive or potentially radioactive waste gas. This gas, containing primarily hydrogen, is collected and stored in an oxygen-free system to guard against a hydrogen explosion and to allow for radioactive decay. The GRS has been sized to provide the capability of holding radioactive gas for a 45-day decay period. After holdup, as appropriate, the stored gases are sampled and then discharged at a controlled rate through a filter and a radiation monitor to the radwaste building exhaust. Low activity waste gases are routed to building vents through particulate filters. Personnel exposures and activity releases from these wastes in restricted and unrestricted areas are within the ALARA guidelines set forth in 10CFR50, Appendix I, when combined with the releases from the GRS. The GRS has a collection header, a waste gas surge tank, two waste gas compressors, and three waste gas decay tanks. One compressor is normally used, while the other is on standby. Liquid seals are not incorporated in the system design.

Sources for the GRS include the gases from:

- Reactor drain tank
- Volume control tank
- Refueling failed fuel detectors
- Gas stripper

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- Reactor vessel vent

The high activity gases accumulate in the waste gas surge tank and are compressed and stored in the waste gas decay tanks. When the surge tank pressure reaches 3 psig, the compressor selected for operation starts automatically and starts charging the online decay tank. If the surge tank pressure reaches 3.5 psig, the standby compressor will automatically start. Operation of either compressor automatically stops when the surge tank pressure decreases to 1.5 psig, corresponding to a compressor suction pressure of 0.5 psig. When decay tank pressure reaches 350 psig, an alarm is actuated and compressor operation is terminated manually. Identical compressors are provided to minimize system downtime.

Each decay tank is sampled prior to discharge. No special mixing is considered necessary for the gas. Each sample is analyzed for radioactivity and the concentration, volume, and total radioactivity are recorded. Isotopic content of the waste gases is determined and recorded as specified in the station manual procedures.

The maximum rates and quantities of radionuclides released from the gaseous waste decay tanks will be in accordance with the limits imposed by the plant technical specifications. The rate of release from the decay tanks into the ventilation exhaust is limited so as not to exceed the release limits of 10CFR20. Releases are conducted to meet the ALARA objectives of 10CFR50, Appendix I.

Each tank is capable of being isolated. Only one tank is valved into operation at any one time to diminish the amount of radioactive gas that can be released due to a postulated rupture of a decay tank or connected piping.

Expected annual releases from the GRS are estimated for normal operations using failed fuel corresponding ANSI N237. The holdup time is considered to be 45 days. Estimated releases from the condenser air removal system, turbine gland sealing system, containment building, auxiliary building, fuel building, and turbine building for component activities are also consistent with ANSI N237. Expected effluent site boundary concentrations, based on the worst sector annual average atmospheric dispersion factor, are estimated. Source terms and assumptions based upon Regulatory Guide 1.112, Revision 0, are used where appropriate. Details of the annual releases by isotope and release point are provided in Section 11.3 of the PVNGS updated FSAR.

Solid Waste Management System (SWMS)

Solid waste management is provided by the solid radwaste system (SRS) which is designed to provide holdup, solidification, and packaging of radioactive wastes generated by plant operation, and to store these wastes until they are shipped offsite for burial. The system is located in the radwaste building, which is designed to withstand an operating basis earthquake. In addition, a dry active waste processing

The DAWPS provides the means of compacting and packaging radioactive materials, such as paper, rags, contaminated clothing, gloves, and shoe coverings, and a means for packaging contaminated metallic materials and incompressible solid objects, such as small tools and equipment parts. It also provides a method of solidifying and packaging blowdown demineralizer resin and condensate polishing resin in the event that they become contaminated.

The DAWPS facility provides a centralized handling, processing, packaging, and storage of radioactively contaminated trash generated during the operation and maintenance activities. The contaminated trash, dry active waste (DAW), is composed of rags, paper, plastic, rubber, wood, glass, concrete, and metal.

The waste solidification system operates on a batch basis to solidify chemical drain tank waste, evaporator concentrates, and spent resins. It is also used to encase spent radioactive cartridge filters and other miscellaneous contaminated objects. The system is designed to solidify spent blowdown demineralizer resin and spent condensate polishing resin if required. Sufficient capacity is provided to solidify radioactive wastes resulting from normal plant operations and anticipated operational occurrences.

Liquid inputs from the concentrate monitor tanks, the chemical drain tanks, and the spent resin tanks are fed into the waste feed tank for hold up and chemical treatment prior to solidification. The waste feed pump transfers waste from the waste feed tank to the radwaste/cement mixer and is mixed with cement and dry additive. The waste/cement/dry additive mixture gravity flows from the mixer, through an air-operated bladder valve, filling assembly, and into either a drum or large container. Mixing ratios of the waste/cement mixture are controlled by setting the speeds of the waste feed pump, the cement feeder, and additive feeder.

After filling one large container or up to six drums, a flush of the system is automatically initiated. The flush sends reactor makeup water to the suction side of the waste feed pump and the balance of the system is operated to solidify the flush water as it flushes the waste feed pump, mixer, and associated piping. A secondary flush is also initiated which sends a fixed amount of flush water to the mixer and the mixer discharge plenum. All of the flush water is solidified and packaged.

mixture. Stabilization is achieved by placing the filter cartridges inside high integrity containers.

To ensure that shipping dose rates for containers are not exceeded, two radiation instruments monitor the disposable liner during filling. The surface radiation monitor is calibrated to measure the contact dose rate and is set to alarm if the dose rate approaches the contact limit set by 49CFR173.

The container is moved to the swipe area. If surface contamination is detected, the container is decontaminated as necessary. The containers are then moved to the high activity storage area for decay while awaiting shipment to a burial facility.

Complete waste solidification and absence of free liquid prior to shipment is assured by the implementation of a process control program consistent with the recommendations of Branch Technical Position ETSB 11-3.

Dry waste, components, and equipment that have been contaminated during operation or maintenance are handled, processed, and packaged by qualified plant personnel or by outside contractors specializing in such activities.

The filters are separated into two classifications:

- Cartridge type filters with disposable elements
- Unshielded low activity filters with disposable elements

The cartridge filters are located in shielded compartments in the auxiliary building which are accessed from above by removing a concrete shield plug and replacing it with a working lead shield plug. Using long-handled tools attached to the working plug, the filter vessel pressure seal is opened and the filter cartridge is lifted through a small hole in the working shield into the filter transfer cask. The transfer cask is then transferred via monorail to the radwaste building laydown area where the bottom head is unfastened and the filter lifting device is unlocked. The transfer cask is placed over a 55-gallon drum which has a special filter basket inside, and the filter is lowered into the filter basket. A cement/liquid waste mixture from the waste/cement mixer is fed into and around the filter basket. The container is capped, decontaminated as necessary, and is transferred by overhead bridge crane to the high activity storage area to await shipment to the burial site.

The following are unshielded low-activity filters:

- Two waste gas compressor refilters
- One gaseous radwaste system discharge filter

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- One blowdown demineralizer filter (subsection 10.4.6)

The gaseous radwaste system (GRS) filters are located in the radwaste building. To change a GRS filter, it is isolated, drained, purged with nitrogen, and vented. The vessel flanged cover is opened and the filter element is manually transferred to the solid radwaste area for processing. The blowdown demineralizer filter is located outside the turbine building. To change it, the filter is drained, vented, manually changed, and transferred to the waste solidification system. Unshielded low-activity filters are normally changed on high-differential pressure; however, radiation surveys performed allow changing frequencies to be adjusted to minimize man-rem exposure.

The dry active waste processing and storage (DAWPS) building is divided in three basic compartments:

- 1) storage area, 2) processing area, and 3) offices and change area.

The waste storage area provides storage for dry active waste (DAW) material. The DAW materials are packaged in containers that meet NRC and DOT requirements for shipping and disposal. All storage drums and boxes are decontaminated before being placed into the waste storage area. It is anticipated that approximately 90% of packages stored have a contact dose rate of 15 mR/hr or less. The remaining packages have contact dose rates of 16-100 mR/hr and are placed in the central part of the storage area so that they are appropriately shielded to ensure that general storage and outside yard area dose rates are maintained at 15 mR/hr or less. Shielding arrangements are also such that dose rates in the offices and change area are maintained at less than 0.5 mR/hr.

The processing area contains the sorting, processing, and compaction equipment, a receiving area for incoming material to be processed, an emergency eye-wash station,

unmonitored or unfiltered air release from the processing and sorting areas.

Detailed annual activities by isotope and waste type is presented in Section 11.4 of the PVNGS updated FSAR.

The System 80 waste streams described above reflect the UO_2 core fuel cycle. The uses of a MOX core results in a somewhat different distribution of fission products. Consequently, the details of radionuclide distribution will be different. However, system modifications are not expected to be needed in order to keep within the regulatory requirements. In addition, site-specific characteristics will be reflected in the design to meet the ALARA objective of 10 CFR 50, Appendix I. One specific modification identified in the need for a tritium removal system in order to accommodate the higher tritium buildup in the primary coolant than the UO_2 equilibrium core.

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8.3.2 Plutonium Carryover

The environment, safety and health aspects of Plutonium carryover for the case where existing reactors are employed is similar to those reported in the Evolutionary Reactor Report, Section 8.3.2.

8.4 Protection Against Criticality

The environment, safety and health aspects of protection against criticality for the case where existing reactors are employed is similar to that reported in the Evolutionary Reactor Report, Section 8.4.

PLUTONIUM DISPOSITION IN EXISTING REACTORS NON-PLUTONIUM DISPOSITIONING OPERATIONS

9.0 NON-PLUTONIUM DISPOSITIONING OPTIONS

9.1 Commercial Fuel Operation

Following the plutonium disposition mission, the existing reactors will be able to return to the normal mode of operation with UO_2 fuel for the remainder of the plant licensed lifetime. Provided there is provision for further spent fuel storage, there are no technical indications that the plutonium disposition mission would in any way limit this capability.

The time remaining for the reactors to return to UO_2 fueled operation is, however, short, and the programmatic effort to do so, which may include returning the unit to the utility rate-base, renewing or setting up power purchase agreements, and so on, may be prohibitive. It should be recognized that the plutonium disposition mission will take at least 18 years (see Table 1.4-1), with at least 6-7 years of initial effort, including fuel fabrication. Therefore the minimum program time would be 24-25 years. Since the Palo Verde units today have 32-33 years lifetime remaining (Table 1.2-1), the maximum expected lifetime of the unit after the plutonium disposition mission is 7-9 years, before the licensed lifetime of 40 years is exceeded. Furthermore, the economic viability of the plant compared to newer plants on the system may be marginal, as advances in power generation technology over the intervening 25 years should result in ever-improving efficiency and reliability, making the existing plants outdated. All of these considerations would have to be taken into account in the initial negotiations.

9.2 Spent Fuel Storage

With a ten-year cooling period following discharge from the core, the final spent MOX fuel will be residing in the reactor spent fuel storage pool after the date that the licensed operation of the reactor expires. The requirements for fuel pool cooling, purification, spent fuel handling and monitoring, as well as safeguards (material accountability) will therefore have to continue until after the remainder of the reactor plant has been shut down and decommissioned. The licensing of this particular area of the plant will have to be extended in order to complete the cooling period. This will also involve demonstration of the continued capability of the systems involved to continue to perform their functions. This situation is, however, no different than for a reactor operating on UO_2 fuel, where the spent fuel is kept in the spent fuel pool until it can be removed to a repository off-site.

9.3 Other Applications of This Technology

The dispositioning of weapons-grade plutonium in existing reactors is recognized as a single application program. As U.S. policy is to avoid reprocessing of spent fuel so as to recycle the fissionable materials, the plutonium operation would not lead to any advances in this area that would be of benefit to the rest of the industry in the U.S. The technology of MOX fuel fabrication would not be applicable to other uses. The licensing, permitting and EIS technology would also be only a one-time application. Therefore, the side benefits from this program should be considered to be non-existent.

PLUTONIUM DISPOSITION IN EXISTING REACTORS ALTERNATIVE CASES

10.0 ALTERNATIVE CASES

This section discusses alternatives to the base case scenarios (B1 and B2) presented in previous sections 2.0 through 9.0. In addition to presenting ways to disposition 100 MT of weapons grade plutonium, an alternative is suggested that uses two units within the same utility, but on different sites for the mission to dispose of 50 MT of weapons grade plutonium. All alternatives discussed in this section use the same assumptions on cycle length, capacity factor, and charging isotopics and discharge characteristics as were used in the base cases - the only difference being whether the plant is a System 80 or 3410 class.

Table 10-1 Summarizes the cases to be discussed. To meet the 100 MT mission would require three units (A1) for the System 80 plants. By adding the third unit, the 100 MT mission could be completed in 23 years compared to the 18 years to complete the 50 MT mission B. This is discussed in Section 10.1.

As has been shown in other sections, it is more expensive to complete the plutonium disposition mission in the 3410 class plants as compared to the System 80 plants. During the DOE review meeting with ABB CE in Windsor in April 1994, members of the DOE team reminded ABB that the objective of this report was to look at options that are technically feasible, but might not be the most economical. In addition to the base case B that would use two 3410 plants at the same site, it is possible to go to multiple sites and develop two additional alternatives. Section 10.2 discusses ways to accomplish the 100 MT mission (A2) and suggests another way to accomplish the 50 MT mission (A4).

One other alternative (A3) is the completion of WNP-3. As shown in Table 10-1, finishing the construction of WNP-3 would allow the completion of the 50 MT mission with a single System 80 plant. Further details on this option are presented in Section 10.3.

10.1 Alternative A1 - 100 MT Mission in System 80

The base case (B) for disposing of 50 MT of weapons grade plutonium at a System 80 site is discussed in other sections (2.0 through 9.0). With no other changes, other than adding the third System 80 unit, the 100 MT mission (relative to the 50 MT mission) can be accomplished in 23 (18) years using 45 (23) unit cycles. A deployment plan for alternative A1 is given in Table 10.1-1. The disposition mission (fuel has same isotopics as spent fuel) is therefore complete in 35 years after the contract is signed. The shipping plan for alternative A1 is given in Figure 10.1-1. After the required 10 years of cooling on site, the last fuel assembly is available for transport to a government facility within 45 years of contract initiation.

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With the fuel loading in unit 3 occurring 12 months after unit 2, the three unit site with 18 month fuel cycles quickly arrives at a pattern where there will be a refueling outage every 6 months. This allows the utility to be able to provide power during peak loads and smooth the work loads for the outage management teams. From a scheduling viewpoint, this 100 MT mission will require the fabrication of 241 assemblies every year for 23 years. The first 120 fuel assemblies will be available for shipment in year 16; annual shipments of 241 assemblies will start in year 18.

10.2 Options in 3410 class plants

It is more expensive to complete the plutonium disposition mission in the 3410 class plants as compared to the System 80 plants. At the direction of the DOE team during the mid-contract review, ABB has investigated options that are technically feasible, but might not be the most economical. Base case B uses two 3410 plants at the same site to perform the 50 MT mission. Alternative A discusses the 100 MT mission using three 3410 plants at two different sites in section 10.2.1. It is also possible to use multiple sites to accomplish the 50 MT mission (A4). This is discussed in section 10.2.2.

10.2.1 Alternative A - 100 MT Mission in 3410 Class Plants

To perform the 100 MT mission (A) in 3410 Class Plant, it is necessary to use three units on two different sites. As shown for alternative A in Table 10.2-1, the 3410 class plants take 76 feed batches to complete the 100 MT mission. Since there are no more than two 3410 plants on the same site, it is assumed that the third unit is on the second site and an additional 3 months are allowed for plant modifications, staff training and startup testing. If this places too much pressure on the MOX fabrication facility, the third unit could be delayed another nine months and have no impact on the mission completion date - which now depends on the last batch from Unit 1 being loaded in month 366.

With units 1 and 2 on the same site, the fuel load and fuel shipment schedules would require 217 fresh assemblies to be shipped each year and (after the 10 year decay period) make available 217 assemblies available for shipment to the government repository each year. The third unit on the second site would require half as many shipments on an annual basis.

With the annual fuel cycles, the two unit site can arrange the refueling outages every six months. This gives them the same advantages as for the three unit site and eighteen month cycles discussed for option A1, above. The shipping plan for alternative A is given in Figure 10.2-1.

PLUTONIUM DISPOSITION IN EXISTING REACTORS ALTERNATIVE CASES

10.2.2 Alternative A4 - 3410 Class 50 MT Mission at two sites

If there are problems getting participation of the utility with two 3410 class plants on the same site, another option is to consider a utility that has two plants with the same fuel assembly design and power density on two separate sites. As shown in Table 10-1, the Entergy Corporation (A4) has two units that meet these requirements. While there would be a lower throughput with ANO-2 than with Waterford-3, it is estimated that the 50 MT mission could be completed with 46 unit cycles - compared with the 38 unit cycles for case B. The time to complete the 50 MT mission would increase from 19 years (B2) to 23 years (A4).

While there may be some disadvantages of performing the plutonium disposition mission at two different sites, there doesn't seem to be any technical reason at this point not to consider the two Entergy units.

10.3 Alternative A3 - 50 MT Mission in one System 80

additional costs to bring the unit up to today's licensing standards and using today's material and labor costs, this single unit (with a 40 year plant lifetime) can perform the 50 MT mission in a single plant.

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
ALTERNATIVE CASES**

**TABLE 10-1
CASE DESCRIPTIONS
EXISTING REACTORS
PLUTONIUM DISPOSITION STUDY**

	Base Case B	Base Case B	Alt 1	Alt 2	Alt 3	Alt 4
MT	50	50	100	100	50	50
Time(yrs)	18	19	23	26	35	22
Units	2	2	3	3	1	2
	Palo Verde 1 and 2	SONGS 2 and 3	Palo Verde 1, 2 and 3	SONGS 2 and 3, Waterford 3	WNP-3	Waterford 3 and ANO-2
cycle (months)	18	12	18	12	18	12
Number of unit- cycles	23	38	45	76	23	46
cycle(EFPD)	450	274	450	274	450	274
CF	82	75	82	75	82	75
discharge (MWD/T)	32,500	20,000	32,500	20,000	32,500	20,000
outage (days)	90	90	90	90	90	90

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
ALTERNATIVE CASES**

**TABLE 10.1-1
DEPLOYMENT STRATEGY SCHEDULE
3 SYSTEM 80'S
ALTERNATE CASE A1 - 100 MT IN 28 YEARS**

Number of Reactor Units	3
Core Power Rating	3800 MWth
Length for Each Cycle	18 Months
Shipping Begins	138 Months after each BOC

No. of MT PU Burned Per Mission: 2.25
 First Core Startup Test Period: 6 months
 Number of Feed Batches for Mission: 45
 For Total of: 101.25 MT Pu

	Operating Cycles			Shipping		
Cycle	Unit 1	Unit 2	Unit 3	Unit 1	Unit 2	Unit 3
1	60	78	90	198	216	228
2	84	96	108	222	234	246
3	102	114	126	240	252	264
4	120	132	144	258	270	282
5	138	150	162	276	288	300
6	156	168	180	294	306	318
7	174	186	198	312	324	336
8	192	204	216	330	342	354
9	210	222	234	348	360	372
10	228	240	252	366	378	390
11	246	258	270	384	396	408
12	264	276	288	402	414	426
13	282	294	306	420	432	444
14	300	312	324	438	450	462
15	318	330		456	468	
16	336			474		

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
ALTERNATIVE CASES**

**TABLE 10.1-2
DEPLOYMENT STRATEGY SCHEDULE
3 SYSTEM 3410 CLASS PLANTS
ALTERNATE CASE A - 100 MT IN 31 YEARS**

Number of Reactor Units 3 Core Power Rating 3410 MWth
Length for Each Cycle 12 Months - Shipping Begins 138 Months after each BOC

No. of MT PU Burned Per Mission: 1.32
First Core Startup Test Period: 6 months
Number of Feed Batches for Mission: 76 For Total of:100.32 MT Pu

	Operating Cycles			Shipping		
Cycle	Unit 1	Unit 2	Unit 3	Unit 1	Unit 2	Unit 3
1	60	72	63	192	204	195
2	78	84	81	210	216	213
3	90	96	93	222	228	225
4	102	108	105	234	240	237
5	114	120	117	246	252	249
6	126	132	129	258	264	261
7	138	144	141	270	276	273
8	150	156	153	282	288	285
9	162	168	165	294	300	297
10	174	180	177	306	312	309
11	186	192	189	318	324	321
12	198	204	201	330	336	333
13	210	216	213	342	348	345
14	222	228	225	354	360	357
15	234	240	237	366	372	369
16	246	252	249	378	384	381
17	258	264	261	390	396	393
18	270	276	273	402	408	405
19	282	288	285	414	420	417

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
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	Operating Cycles			Shipping		
Cycle	Unit 1	Unit 2	Unit 3	Unit 1	Unit 2	Unit 3
20	294	300	297	426	432	429
21	306	312	309	438	444	441
22	318	324	321	450	456	453
23	330	336	333	462	468	465
24	342	348	345	474	480	477
25	354	360	357	486	492	489
26	366			498

**Shipping Plan
Alternate Case A-1**

Figure 10.1-1

**Shipping Plan
Alternate Case A-1**

Figure 10.2-1

PLUTONIUM DISPOSITION IN EXISTING REACTORS PUBLIC AND POLITICAL ACCEPTANCE

11.0 PUBLIC AND POLITICAL ACCEPTANCE

11.1 Approach for Achieving Public Acceptance

The approach to achieving public acceptance that was outlined in the Evolutionary Reactor Report, Section 12, is applicable to the case of existing reactors. This approach is based on early and continuous outreach into the community, emphasizing communication in both directions and public involvement. In some ways, the situation is simpler, since the reactors exist, are in operation, and there are public relations formats and contacts available and working. The program would need to be adapted to fit the format and activities that are already in existence. The program would also need to stress the advantages to the community of the plutonium disposition mission. The negotiation with the utility should produce an end result that means lower or more stable electricity costs to the public customers, and more reliable earnings to the stock and bondholders, which should be major selling points. Additional, less materialistic arguments, such as the maturity of nuclear power and the role the community is having in reducing the proliferation of nuclear weapons, may be effective.

The danger that must be guarded against is that the introduction of plutonium in the fuel would trigger a renewed burst of anti-nuclear sentiment among activist groups, intent on shutting down the plant. The public hearings for licensing and the PUC hearings will provide the opportunities for such activities.

11.2 Likelihood of National Political Acceptance

There does not appear to be any reason why the use of existing reactors would have any different national political acceptance than the use of a new evolutionary reactor for the purpose of dispositioning weapons-grade plutonium. In the end, national political acceptance would depend on local pressure, and secondarily on the Government funding outlays and the risks involved. For the new evolutionary reactor, the funding and risk to the Government's account may be less than for existing reactors.

11.3 Likelihood of Local or Regional Acceptance

In April of this year, Governor Mike Lowry of Washington stated that he opposes the plan to use the unfinished WNP-3 reactor, at Satsop, to burn weapons-grade plutonium, converting the warheads to electricity. He stated that if plutonium is to be burned at all, Hanford is the place to do it. At Satsop, he argues, there is only an uncompleted commercial plant, not the infrastructure and facilities needed to handle plutonium such as exist at Hanford.

Such a reaction came during a wide-ranging discussion on the Isaiah Project, that would use the two unfinished reactors, at Satsop and at Hanford (WNP-3 and WNP-1), to carry out the plutonium disposition mission. The opposition expressed by this

PLUTONIUM DISPOSITION IN EXISTING REACTORS PUBLIC AND POLITICAL ACCEPTANCE

regional politician to a plan that otherwise would seem to benefit his state with job creation is a hint that local or regional acceptance, particularly by political figures, might be difficult to obtain, for any site that is not on a Government reservation.

PLUTONIUM DISPOSITION IN EXISTING REACTORS CONCLUSIONS AND RECOMMENDATIONS

12.0 CONCLUSIONS AND RECOMMENDATIONS

12.1 Five Key Challenges to Implementing Proposed Operation

The five key challenges to implementing a Plutonium Disposition mission in existing ABB-CE designed power plants are believed to be :

- The design, construction and operation of a Mixed Oxide Fuel Fabrication Facility.
- The licensing of MOX Fuel Fabrication Facility and the licensing of the MOX Fuel in the ABB-CE designed reactor.
- Public acceptance of the mission and the facility at the selected locations.
- Willingness of the plant's owner to participate. The path that DOE is following would expose owner to criticism from public long before DOE reaches a decision on technology to follow.
- Moving ahead on the DOE process so that existing reactors do not lose capabilities for the mission, i.e. for every year that a decision is not made, Palo Verde (3 units) loses >4.5 metric ton capability.

12.2 Five Key Advantages to Implementing Proposed Operation

The five key advantages to implementing a Plutonium Disposition Mission in existing ABB-CE designed power plants are believed to be:

- The least technical risk methodology in disposition of plutonium is the mission of plutonium in reactors such as Palo Verde or the 3410 MWt plants.
- The least cost option to the U.S. Government are existing reactors such as Palo Verde which were designed for 100% MOX assuming the challenges can be overcome.
- Truly achieving the spent fuel standard when one considers retrievability and presence of weapons grade plutonium is readily achievable in a System 80 and questionable in the vitrification option.
- System 80 was designed for 100% MOX operation with no compromise in safety. The ability to cover the core with control rod access to each fuel assembly is a design basis for System 80 and the reason why for twenty years industry experts have recognized System 80 as the only existing reactors in the world truly designed for 100% MOX operation.

PLUTONIUM DISPOSITION IN EXISTING REACTORS CONCLUSIONS AND RECOMMENDATIONS

- The use of Palo Verde reactors provides capability for up to 150 metric tons at a single site should the quantity to be processed increase in the future. This capability can be used in various ways including lower Pu loading, higher throughputs, later start dates, etc., when the DOE process clarifies the desirable capabilities of a reactor.

12.3 Recommendation

ABB-CE recommends that DOE continue to fund a study of Palo Verde assuming a commitment to participate in the study from Arizona Public Service. A detailed study to document plant specific design modifications, cost and schedule details over a 12-18 month time frame to support the Record of Decision is recommended. Details of this program should be developed with DOE, ABB-CE and Arizona Public Service.

PLUTONIUM DISPOSITION IN EXISTING REACTORS UNIFORM DESCRIPTIVE TECHNICAL DATA

APPENDIX A. UNIFORM DESCRIPTIVE TECHNICAL DATA

Appendix A contains the required uniform descriptive technical data. As described in Section 1.4, the required cases (B1 and B2) and alternatives (A1 and A2) are given in attached Table in the specified format. Since there are two different fuel managements presented for the System 80 and the 3410 class plants, there are differences in the columns that discuss the fuel characteristics - Sections II through VIII. Some of the items in Sections IX and XII (Fuel Fabrication and Environmental, Safety, and Health) are dependent on the number of units as well as the class of plant. The remaining sections are either "N/A" or independent of the number of units required for the option.

There are four notes attached to the Table that explain entries (or lack there of) when the values may not have been clear. The section V has been copied from the "Evolutionary Reactor Report" and represent values for the discharge characteristics for the Uranium cycle for the System 80 + class of plants. It is expected that these will be limiting values compared to the various operating cycles now being investigated for the existing reactors.

2	2	3	3	Number of Plants
N/A	N/A	N/A	N/A	I. Plant construction and operation schedule (relative to project start)
60	60	60	60	1) time construction permit will be granted (mo)
78	72	78	72	2) time operations start, for first reactor (mo)
N/A	N/A	96	63	3) time operations start, for second reactor if multiple reactors (mo)
N/A		N/A	N/A	4) time operations start, for third reactor if multiple reactors (mo)
N/A		N/A	N/A	5) time operations start, for fourth reactor if multiple reactors (mo)
N/A		N/A	N/A	6) time operations start, for fifth reactor if multiple reactors (mo)
N/A		N/A	N/A	7) time operations start, for sixth reactor if multiple reactors (mo)
N/A		N/A	N/A	8) time operations start, for seventh reactor if multiple reactors (mo)
N/A		N/A	N/A	9) time operations start, for eighth reactor if multiple reactors (mo)
3817	3410	3817	3410	10) thermal power per reactor (MWth)
1256	1153	1256	1153	11) net electrical output per reactor (MWe)
35.53	32.86	69.52	65.72	12) net electrical energy output for all reactors during disposition operations (GWy)
0.82	0.75	0.82	0.75	13) capacity factor
18.5	10	11	0	14) REMAINING USEFUL plant life (y)
5	5	5	5	15) licensing status (use levels described in UCRL-113055, Section 3.5)
716.625	739.2	1719	1796.2	16) integral of excess plutonium inventory over time (MT-y)
186	186	222	225	17) time from project start to process half of excess plutonium inventory (y)
284	300	354	388	18) time from project start to process all of excess plutonium inventory (y)
				II. Fresh fuel characteristics
3173	1905	3173	1905	1) reactor Pu inventory - first core (kg Pu)
13.17	8.78	13.17	8.78	2) Pu content (kg Pu/fuel assembly)
0	0	0	0	3) Pu isotopics (kg Pu-238/kg Pu)
0.935	0.935	0.935	0.935	4) Pu isotopics (kg Pu-239/kg Pu)
0.065	0.065	0.065	0.065	5) Pu isotopics (kg Pu-240/kg Pu)
0	0	0	0	6) Pu isotopics (kg Pu-241/kg Pu)
0	0	0	0	7) Pu isotopics (kg Pu-242/kg Pu)
438.59	438.66	438.59	438.66	8) fuel assembly mass (kg)
3.81	3.81	3.81	3.81	9) fuel assembly length (m)
0.2027	0.2027	0.2027	0.2027	10) fuel assembly width (m)
425.31	429.91	425.31	429.91	11) U mass (kg U/fuel assembly)
0.002	0.002	0.002	0.002	12) U-235 enrichment (kg U-235/kg U)
				III. Equilibrium cycle recharge
2380	1429	2380	1429	1) fuel Pu content (kg Pu)
0	0	0	0	2) fuel Pu isotopics (kg Pu-238/kg Pu)
0.935	0.935	0.935	0.935	3) fuel Pu isotopics (kg Pu-239/kg Pu)
0.065	0.065	0.065	0.065	4) fuel Pu isotopics (kg Pu-240/kg Pu)
0	0	0	0	5) fuel Pu isotopics (kg Pu-241/kg Pu)
0	0	0	0	6) fuel Pu isotopics (kg Pu-242/kg Pu)
120.5	108.5	120.5	108.5	7) number of fuel assemblies
18	12	18	12	8) cycle length (mo)
				IV. Fuel characteristics at discharge

8.492E+05	8.358E+05	8.492E+05	8.358E+05	7) heat generation in one fuel assembly (W)
6.039E+10	5.990E+10	6.039E+10	5.990E+10	8) gamma radiation field (rem/h at surface, midplane of fuel assembly)
1.3198E+03	5.0025E+02	1.3198E+03	5.0025E+02	9) neutron radiation field (rem/h at surface, midplane of fuel assembly)
900	548	900	548	10) effective full power days (d)
32500	20000	32500	20000	11) burnup (MWd/kgHM)
0.0367	0.0267	0.0367	0.0267	12) fraction of Pu in heavy metal (kg Pu/kgHM)
V. Fuel characteristics at discharge for commercial basis reactor NOTE (4) *****				
389.8	389.8	389.8	389.8	1) Pu mass discharged per cycle (kg Pu)
0.018	0.018	0.018	0.018	2) Pu-238 fraction (kg Pu-238/kg Pu)
0.527	0.527	0.527	0.527	3) Pu-239 fraction (kg Pu-239/kg Pu)
0.232	0.232	0.232	0.232	4) Pu-240 fraction (kg Pu-240/kg Pu)
0.154	0.154	0.154	0.154	5) Pu-241 fraction (kg Pu-241/kg Pu)
0.07	0.07	0.07	0.07	6) Pu-242 fraction (kg Pu-242/kg Pu)
9.112E+05	9.112E+05	9.112E+05	9.112E+05	7) heat generation in one fuel assembly (W)
6.485E+10	6.485E+10	6.485E+10	6.485E+10	8) gamma radiation field (rem/h at surface, midplane of fuel assembly)
8.084E+02	8.084E+02	8.084E+02	8.084E+02	9) neutron radiation field (rem/h at surface, midplane of fuel assembly)
432	432	432	432	10) effective full power days (d)
47800	47800	47800	47800	11) burnup (MWd/kgHM)
1.016E-02	1.016E-02	1.016E-02	1.016E-02	12) fraction of Pu in heavy metal (kg Pu/kgHM)
9.218E-03	9.218E-03	9.218E-03	9.218E-03	13) fraction of U-235 in heavy metal (kg U-235/kgHM)
VI. Fuel characteristics 1 year after discharge				
0.0009	0.0005	0.0009	0.0005	1) Pu-238 fraction (kg Pu-238/kg Pu)
0.6120	0.6551	0.6120	0.6551	2) Pu-239 fraction (kg Pu-239/kg Pu)
0.2335	0.2154	0.2335	0.2154	3) Pu-240 fraction (kg Pu-240/kg Pu)
0.1310	0.1128	0.1310	0.1128	4) Pu-241 fraction (kg Pu-241/kg Pu)
0.0209	0.0161	0.0209	0.0161	5) Pu-242 fraction (kg Pu-242/kg Pu)
5.245E+03	3.826E+03	5.245E+03	3.826E+03	6) heat generation in one fuel assembly (W)
4.144E+08	2.986E+08	4.144E+08	2.986E+08	7) gamma radiation field (rem/h at surface, midplane of fuel assembly)
5.681E+02	2.024E+02	5.681E+02	2.024E+02	8) neutron radiation field (rem/h at surface, midplane of fuel assembly)
VII. Fuel characteristics 10 years after discharge				
0.0011	0.0006	0.0011	0.0006	1) Pu-238 fraction (kg Pu-238/kg Pu)
0.6413	0.6821	0.6413	0.6821	2) Pu-239 fraction (kg Pu-239/kg Pu)
0.2467	0.2243	0.2467	0.2243	3) Pu-240 fraction (kg Pu-240/kg Pu)
0.0890	0.0762	0.0890	0.0762	4) Pu-241 fraction (kg Pu-241/kg Pu)
0.0219	0.0168	0.0219	0.0168	5) Pu-242 fraction (kg Pu-242/kg Pu)
5.085E+02	3.039E+02	5.085E+02	3.039E+02	6) heat generation in one fuel assembly (W)

0.0006	0.0003	0.0006	0.0003	1) Pu-238 fraction (kg Pu-238/kg Pu)
0.7042	0.7388	0.7042	0.7388	2) Pu-239 fraction (kg Pu-239/kg Pu)
0.2698	0.2416	0.2698	0.2416	3) Pu-240 fraction (kg Pu-240/kg Pu)
0.0013	0.0011	0.0013	0.0011	4) Pu-241 fraction (kg Pu-241/kg Pu)
0.0242	0.0182	0.0242	0.0182	5) Pu-242 fraction (kg Pu-242/kg Pu)
2.993E+02	1.898E+02	2.993E+02	1.898E+02	6) heat generation in one fuel assembly (W)
4.952E+06	3.045E+06	4.952E+06	3.045E+06	7) gamma radiation field (rem/h at surface, midplane of fuel assembly)
45.3	26.17	45.3	26.17	8) neutron radiation field (rem/h at surface, midplane of fuel assembly)
				IX. LWRs: Pu inventory and handling at fuel fabrication facility
				(ref. Management and Disposition of Excess Weapons Plutonium,
				National Academy of Sciences, 1994, Table 3.1)
HIGH	HIGH	HIGH	HIGH	1) DOE attractiveness level of PuO ₂ feed material
HIGH	HIGH	HIGH	HIGH	2) DOE attractiveness level of MOX fuel pellets
800	700	1200	1050	3) average inventory of PuO ₂ feed material (kg Pu) NOTE(1) *****
3173	2858	4760	4287	4) throughput of PuO ₂ feed material (kg Pu/y)
1.5	1	1.5	1	5) PuO ₂ feed material residence time or duration (y)
0	0	0	0	6) number of manual movements of material (e.g. between gloveboxes)
0	0	0	0	7) average inventory of MOX fuel pellets (kg Pu)
3173	2858	4760	4287	8) throughput MOX fuel pellets (kg Pu/y)
1.5	1	1.5	1	9) MOX fuel pellet residence time or duration (y)
0	0	0	0	10) number of manual movements of MOX fuel pellets (e.g. between gloveboxes)
2380	1429	2380	1429	11) average inventory of MOX fuel assemblies (kg Pu)
3173	2858	4760	4287	12) throughput MOX fuel assemblies (kg Pu/y)
1.5	1	1.5	1	13) MOX fuel assembly residence time or duration prior to fuel load(y)
2	2	2	2	14) number of movements of MOX fuel assemblies
485	625	625	900	15) staffing level of MOX fuel fabrication facility (number of persons)
950600	1225000	1225000	1764000	16) effort at MOX fuel fabrication facility (person-h/y)
				X. ALMR: Pu inventory and handling at fuel fabrication facility
				(ref. Management and Disposition of Excess Weapons Plutonium,
				National Academy of Sciences, 1994, Table 3.1)
N/A	N/A	N/A	N/A	1) DOE attractiveness level of Pu feed material
N/A	N/A	N/A	N/A	2) DOE attractiveness level of metal alloy fuel
N/A	N/A	N/A	N/A	3) average inventory of Pu feed material (kg Pu)
N/A	N/A	N/A	N/A	4) throughput of Pu feed material (kg Pu/y)
N/A	N/A	N/A	N/A	5) Pu feed material residence time or duration (y)
N/A	N/A	N/A	N/A	6) number of manual movements of material (e.g. between gloveboxes)
N/A	N/A	N/A	N/A	7) average inventory of metal alloy fuel (kg Pu)
N/A	N/A	N/A	N/A	8) throughput metal alloy fuel (kg Pu/y)
N/A	N/A	N/A	N/A	9) metal alloy fuel residence time or duration (y)

N/A	N/A	N/A	N/A	For Office of Material and Fuel Fabrication Facility (person-h/y)
N/A	N/A	N/A	N/A	XI. MHTGR: Pu inventory and handling at fuel fabrication facility
N/A	N/A	N/A	N/A	(ref. Management and Disposition of Excess Weapons Plutonium,
N/A	N/A	N/A	N/A	National Academy of Sciences, 1994, Table 3.1)
N/A	N/A	N/A	N/A	1) DOE attractiveness level of Pu or PuO2 feed material
N/A	N/A	N/A	N/A	2) DOE attractiveness level of particle fuel
N/A	N/A	N/A	N/A	3) average inventory of Pu or PuO2 feed material (kg Pu)
N/A	N/A	N/A	N/A	4) throughput of Pu or PuO2 feed material (kg Pu/y)
N/A	N/A	N/A	N/A	5) Pu or PuO2 feed material residence time or duration (y)
N/A	N/A	N/A	N/A	6) number of manual movements of material (e.g. between gloveboxes)
N/A	N/A	N/A	N/A	7) average inventory of particle fuel (kg Pu)
N/A	N/A	N/A	N/A	8) throughput particle fuel (kg Pu/y)
N/A	N/A	N/A	N/A	9) particle fuel residence time or duration (y)
N/A	N/A	N/A	N/A	10) number of manual movements of particle fuel (e.g. between gloveboxes)
N/A	N/A	N/A	N/A	11) average inventory of particle fuel assemblies (kg Pu)
N/A	N/A	N/A	N/A	12) throughput particle fuel assemblies (kg Pu/y)
N/A	N/A	N/A	N/A	13) particle fuel assembly residence time or duration prior to fuel load(y)
N/A	N/A	N/A	N/A	14) number of movements of particle fuel assemblies
N/A	N/A	N/A	N/A	15) staffing level of particle fuel fabrication facility (number of persons)
N/A	N/A	N/A	N/A	16) effort at particle fuel fabrication facility (person-h/y)
				XII. Environmental, Safety, and Health
NO	NO	NO	NO	1) requires Am removal (respond with "Y" or "n")
1670	2100	2100	2900	2) dose for life of plant fuel fab operations and intrasite transport (person-rem) NOTE(2) *****
7793	9800	9800	13533	3) dose for life of plant reactor operations person-rem)
3170	3960	5940	7930	4) low level waste generated for life of plant (m^3)
2772	4123	5423	8246	5) spent fuel generated for life of plant (no. of fuel assemblies)
263340	391685	515185	783370	6) spent fuel generated for life of plant (kg)
485.1	721.525	949.025	1443.05	7) spent fuel generated for life of plant (m^3)
*	*	*	*	8) TRU waste generated for life of plant, excluding packaging (m^3) NOTE (3) *****
*	*	*	*	9) TRU waste generated for life of plant, excluding packaging (kg) NOTE (3) *****
5544	8246	10846	16492	10) number of spent fuel movements to end of disposition operations (including last 10 years)
7	7	7	7	11) reactor technology readiness (use levels described in UCRL-113055)
6	6	6	6	12) fuel fabrication technology readiness (use levels described in UCRL-113055, Section 3.4)
7	7	7	7	13) spent fuel storage technology readiness (use levels described in UCRL-113055, Section 3.4)
6	6	6	6	14) spent fuel transport technology readiness (use levels described in UCRL-113055, Section 3.4)
				XIII. Other measures

**Appendix A - File 1
Uniform Descriptive Technical Data
Existing Reactors Report**

Reference Cases		Alternatives	
1	2	1	2
50 MT Pu in PV-1,2,3	50 MT Pu in 3410class	100 MT Pu in PV-1,2,3	100 MT Pu in 3410class

2	2	3	3	Number of Plants
Y	Y	Y	Y	1) physical protection system meets standards in DOE Orders (respond with "Y" or "n")
Y	Y	Y	Y	2) material control and accountability system meets standards in DOE Orders ("Y" or "n")
Y	Y	Y	Y	3) is IAEA monitoring practical (i.e. nothing to prohibit monitoring) ("Y" or "n")
0	0	0	0	4) number of reactors of similar design under construction in the former republics of the USSR
0	0	0	0	5) number of reactors of similar design in operation in the former republics of the USSR

NOTES

NOTE(1) Item IX #3 Based upon shipments every three months to fabrication plant. Shipments can be optimized with respect to size and frequency to best satisfy safeguards requirements.
NOTE(2) Item XII #2. Based upon BNFL experience of 0.5 Rem/person/year.
NOTE (3) Item XII # 8 and 9. Solid waste activities are based on several actual operating nuclear power units. This data is provided in Appendix C, Table 9.4.5-2. Data is provided in terms of specific activity (micro curies/gm) rather than a volume or mass basis. Volume reductions may be applied, as appropriate, based upon economic considerations of waste disposal, personnel exposure, etc. in the year of disposal.
NOTE(4) Items for Block V (Commercial reactor) are from the "Evolutionary Reactor Report", April 1994. The values are expected to bound the values for a particular operating cycle for an existing reactor plant.

**PLUTONIUM DISPOSITION IN EXISTING REACTORS
SUMMARY OF NUCLEAR ENGINEERING CODES**

APPENDIX B. SUMMARY OF NUCLEAR ENGINEERING CODES

There are no changes to the engineering codes discussed in the "Evolutionary Reactor Report", April 1994. Refer to Appendix B of that report for further detail.