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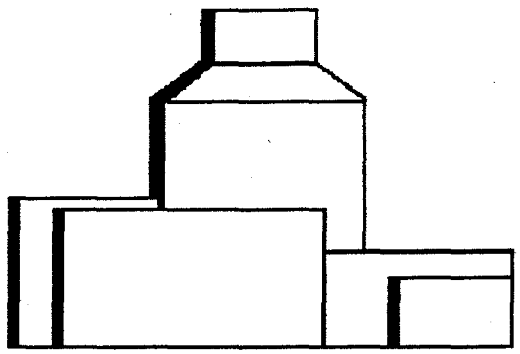
PDR600



Plutonium Disposition Study

Phase 1b Final Report

for
The Department of Energy
San Francisco, CA

September 15, 1993

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PDR600

Phase 1b Status Report

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DE-AC03-93SF19683

1.0 Summary

This report provides the results of the Westinghouse activities performed as part of the Plutonium Disposition Study Phase 1b under DOE contract DE-AC03-93SF19683. These activities, which took place from May 16, 1993 to September 15, 1993, build upon the work completed in Phase 1a, which concluded on May 15, 1993. In Phase 1a, three Plutonium Disposal Reactor (PDR) options were developed for the disposal of excess weapons grade plutonium from returned and dismantled nuclear weapons. In the resulting Phase 1a report, Westinghouse demonstrated the ability of the PDR600, a modified version of the AP600 commercial nuclear plant, to successfully dispose of the weapons grade plutonium according to the DOE requirements.

As a result of the Phase 1a activities, and subsequent review, a workscope for Phase 1b was developed. This workscope included the presentation of the Phase 1a results to the DOE Technical Review Committee at Piney Point, MD, as well as responses generated to questions from DOE and other review bodies. In addition, several technical studies were conducted in anticipation of Phase 2 of the PDR Study.

This report documents the results of several tasks that were performed to further knowledge in specific areas leading up to Phase 2 of the PDR Study. The Westinghouse activities for Phase 1b are summarized as follows:

- Resolved technical issues concerning reactor physics including equilibrium cycle calculations, use of gadolinium, moderator temperature coefficient, and others as documented in Section 2.0.

- Analyzed large Westinghouse commercial plants for plutonium disposal. The large plants were found to be acceptable for disposing of plutonium, and will reduce the number of plants needed.

- Reactor safety issues including the steam line break were resolved, and are included in Section 2.0.

- Several tasks related to the PDR Fuel Cycle were examined. These include fuel fabrication issues related to fuel rod manufacturing, storage, radiation dosage from MOX fuel, as well as post irradiation fuel handling. These studies are presented in Section 3.0.

- Cost and deployment options were examined to determine optimal configuration for both plutonium disposal and tritium production. These studies are presented in Section 4.0.

- Response to questions from DOE and National Academy of Scientists (NAS) reviewers concerning the PDR Phase 1a report are included in Appendix A.

- The presentation of the PDR600 Plutonium Disposition Study to the DOE Technical Review Committee at Piney Point, MD is included in Appendix B.

- A Revised Phase 1a report has been completed and is being issued under separate cover.

2.0 Technical Studies

2.1 Core Physics

The feasibility of operating a 100% MOX core using weapons grade plutonium was demonstrated during the phase I study. The Phase I Extension Report compiles design and analysis performed since the end of Phase I. Much emphasis was placed on modifying core models to ensure the prediction of negative moderator temperature coefficient (MTC) values at all operating conditions. The characteristics of the new designs, called "revised", are described herein. Additional core parameters, such as void coefficient and xenon worth, as well as core characteristics at cold (68 F) conditions have also been studied. Additionally, a study was performed to compare certain characteristic differences between small and large reactors.

Another important continuation of this study was the determination of equilibrium cycle core characteristics. An equilibrium cycle has different burnup and isotopic distribution and generally gives a better and more generic representation of core behavior. Since most emphasis was placed on the spent fuel option, details of revised plutonium destruction option models are not described in this report.

BASIC PHILOSOPHY FOR SPENT FUEL DESIGN

The basic design objective is to meet the program requirement of denaturing 100 MT of weapons grade plutonium in the reactor as economically as possible through the use of a conservative design using proven hardware and software. The basic core characteristics of one-third core MOX fueled PWRs are well studied and understood. Although there have been some studies performed with 100% loadings of <u>reactor discharge grade</u> MOX, no systematic study of a 100% MOX fueled core using <u>weapons grade</u> plutonium had been undertaken before this project. During Phase I a core model was developed which charges 5.5 w/o Pu enrichment fuel and discharges this fuel at 40,000 MWD/MTM exposure. All models show ample conservatism in the design, since more than adequate margin was predicted for peaking factor, shutdown margin and other core performance and safety parameters.

The PDR600 requirements necessitate a heavy loading of glass burnable absorbers and integral burnable absorbers to hold down the excess reactivity at the beginning of cycle. Even though 3500-4000 ppm of soluble boron are permissible from chemical considerations, levels should exceed about 2500 ppm at HFP, (Xe free) conditions in order to predict an MTC more negative than -2.0 pcm/F.

Recent calculations indicate that the allowable enrichment may exceed 6.0 w/o, providing a reduction in the required number of reactors. Another factor which contributes significantly to the number of reactors necessary is the discharge burnup. Since part of the objective is to minimize time requirements, shorter burnups in the range of 30,000 to 35,000 MWD/MTM would yield about 20% Pu-240 content in the spent fuel. The 20% Pu-240 fraction approximates the isotopic fraction in spent commercial reactor fuel and is consistent with the DOE Plutonium Disposition Study Technical Review Committed statement; "the presence of about 20% Pu-240 defines the material as non-weapons grade plutonium," (section SC3-17, 3.1 of the report issued July 2, 1993).

A feed enrichment of 6.5% Pu is about the maximum that can be loaded in PDR600 and remain within the negative MTC limit. A possible avenue to attain higher enrichments is to use gadolinium (Gd) as an integral burnable absorber. Gd has a much higher absorption cross section than ^{10}B and unlike in commercial reactors, there is no penalty for residual end of life poison. Westinghouse has considerable experience using Gd fuel in PWRs. In a uranium fueled core, the enrichment of the Gd loaded fuel rods are generally reduced by about 5% for each percent loading (by weight) of Gd to ensure that those rods are not the power peaks (DNB consequences). Gd loaded MOX fuel is a possibility for PDR600 and will be pursued in Phase II. However, with 20% Pu-240 as the denaturing goal, higher enrichments may not be optimum. This is because the higher the charged plutonium enrichment, the greater the discharge burnup must be to achieve a given Pu-240 content. That being the case, there may be no incentive for using Gd.

It should be pointed out that the combination of stainless steel cladding and IFBA coating, and MOX is new, although there is experience with individual components. A few irradiation demonstration assemblies may be required prior to the use of this fuel in PDR600.

REVISED INITIAL AND EQUILIBRIUM CYCLE MODELS FOR SPENT FUEL OPTION

The revised cycle 1 model has a higher loading of integral burnable absorbers in the core in order to reduce soluble boron and make the MTC more negative. Full length, 1.57 mg/inch IFBA coating is used in all assemblies, which is the standard value used for commercial 0.374 inch uranium fuel rods.

The equilibrium cycle model was developed by using 5.5 w/o enriched once burnt, twice burnt and feed assemblies. All fuel rods are coated with zirconium diboride with a 1.57 mg/inch loading of B10. It may be noted that the standard outing fuel management scheme was used in equilibrium cycle. The following sections describe important core characteristics, and *all future references are directed to the revised model, unless otherwise mentioned.*

a) Critical Boron Level

The boron level in the equilibrium cycle has been reduced from 3100 ppm in the original cycle 1 model to 2100 ppm at BOL (HFP, no xenon). Equilibrium cycle and cycle 1 have almost identical boron levels at BOL, but the equilibrium cycle EOL boron is only 250 ppm as compared to 800 ppm for cycle 1. The boron letdown with depletion behaves similar to a typical uranium fueled core.

b) Peaking Factors

The equilibrium cycle total peaking factor is 1.72 at BOL, while the EOL value is 1.57. Hot channel factors are comparable between cycle 1 and equilibrium cycle. Axial peaking factors are decreased by about 10% in the equilibrium cycle. These PDR600 axial peaking factors are designed to accommodate the peaking factor limits of the AP600.

c) Moderator Temperature Coefficient

The MTC in the equilibrium model is more negative than in cycle 1. The most negative MTC for the equilibrium cycle is -30 pcm/F, similar to uranium fueled cores (HFP, no boron). The predicted MTC for cycle 1 at HZP is -4.3 pcm/F, making the most positive and negative MTC limits within the design basis limit of the AP600.

A very important facet of the PDR600 design is that there is enough flexibility to tailor MTC values to suit reactor operation and safety needs. Calculations indicate that MTC values similar to uranium fueled cores can be easily achieved by balancing burnable absorbers and soluble boron. Consequently, the effect of MTC on cool-down accidents is not expected to be more severe than that for uranium fueled cores.

d) Boron Coefficient

The boron coefficient is much smaller for the PDR600 than for the AP600. It varies slightly from -2.8 to -3.9 pcm/ppm, whereas -9 to -13 pcm/ppm is the range for AP600. The low boron worth is beneficial for boron dilution accidents, and early indications do not reveal any impact of low boron worth on the steamline break accident..

e) Xenon Worth

Like all other reactivity worths, the xenon worth has decreased by about a factor of two in PDR600. The xenon worth at HFP BOL and EOL is 1230 pcm and 1500 pcm, respectively, for the equilibrium cycle. Typically, in uranium fueled cores, the xenon worth is around 2500 to 3000 pcm. The low worth of xenon has an important advantage for stability against xenon induced oscillations, making axial xenon transients less severe in PDR600.

f) Control Rod Worth and Pattern

The control rod worth is, in general, decreased in the equilibrium cycle, which is the normal trend. Compared to the initial cycle, the "Five clusters, D Bank" worth is reduced from 620 pcm to 560 pcm at BOL, HFP and from 690 pcm to 600 pcm at EOL, HFP.

Similarly, the total rod worth is reduced from 9460 pcm to 9300 pcm at BOL, HFP and from 10700 pcm to 10500 pcm at EOL, HFP.

PDR600 is designed to be a base load plant, and therefore the control rod pattern selection process is less severe. In the present design, five rods making a cross pattern are selected as the D Bank. The D Bank selection is crucial, since it is normally the only bank inserted in the reactor at hot full power to control power peaking. The D Bank worth is adequate to control power peaking during reactor operation. "Five clusters, D Bank" could be easily changed to "nine clusters, D Bank" with an associated change in rod insertion limit, and control bank configuration studies will be investigated in Phase II. It can be concluded that the addition of 8 rod clusters at the periphery and the replacement of 16 gray rods by full strength rods provide PDR600 worths approximating the AP600 values.

g) Shutdown Margin

The calculated shutdown margin is decreased from 4.9% in cycle 1 to 4.4% in equilibrium cycle at BOL and from 5.5% to 4.7 % at EOL. This represents a reduction of about 10% to 20% at BOL and EOL, respectively. But, with a predicted shutdown margin of about 3% , and considering an AP600 shutdown margin requirement of 1.6%, substantial margin exists in the design.

h) Core Characteristics at Cold Conditions

The soluble boron concentration for cycle 1 and equilibrium cycle are both 2700 ppm for unrodded BOL conditions. The refueling boron levels at 68 F for the initial and equilibrium cycles are 2100 and 2200 ppm, respectively (all-rods-in and keff = 0.95). Therefore the maximum boron concentration used for all reactor conditions, including refueling, is well below its solubility limits. The minimum shutdown boron levels at cold BOL condition for cycle 1 and equilibrium cycle are 1300 and 1700 ppm, respectively. This cold shutdown condition assumes 1% as the required shutdown margin and includes an additional 100 ppm for conservatism.

i) Ejected Rod Accident

The ejected rod worths and peaking factors are essentially the same for all models and cycles. The limiting ejected rod worth and peaking factors are less than 50% of AP600 values. Evaluating this in terms of safety limits defined in the phase I report, it is seen that a large amount of margin exists.

REACTOR SIZE

A 1933 MWt reactor, designated PDR600 and based on the AP600, was used exclusively for Phase I evaluations. This power level was appropriate because it represents Westinghouse's version of future commercial nuclear units. Of course Westinghouse has extensive experience with larger commercial reactor sizes. During Phase I extension, a comparison of the 1933 MWt size versus larger reactors, producing 3000 MWt or more, was performed. Stockpile transmutation rates were analyzed, as well as some of the nuclear performance associated with "full-size" plants.

Spiking Option:

The PDR600 size is the clear choice here. A mass flow comparison to a 3411 MWt (high power density) core and a 3150 MWt (reduced power density) core indicates that it requires 2.5 years to dispose of the stockpile, regardless of power level. Therefore, since the objective for this option is to spike fuel as rapidly as possible, PDR600 receives the highest rating because it costs the least.

This conclusion was based on equal refueling time per assembly, as well as fuel fabrication capacity for all cases. Refueling time used is 8.5 assemblies per day and the fabrication plant capacity is 1750 assemblies per year based on the Westinghouse Columbia Plant capacity. Since the AP600 has the fewest assemblies, its refueling times and cycle lengths are shortest, but it requires the most number of cycles for stockpile disposition. The larger reactors need longer refueling times and cycle lengths, but require fewer number of cycles. These effects cancel each other, with the result being that the same time frame (2.5 years) is required in all cases.

In this evaluation, no credit was given to larger reactors for the greater number of MW generated during the 2.5 year period.

Spent Fuel Option:

Determination of optimum reactor size for the spent fuel option requires an extensive evaluation involving the interaction of program requirements, nuclear performance and costs. The nuclear performance evaluation involves maximizing Pu loading, determining the denaturing capability and cycle lengths connected with those loadings, then evaluating the ability to safely operate the reactor with those loadings. This section only addresses the nuclear performance side of the story.

During Phase I efforts, a maximum MOX enrichment of 5.5w/o Pu was specified and average discharge burnup was set at 40,000 MWD/MTM . This was with consideration that the stockpile must be completely processed through the reactor in a 25 year operating window, and that the discharged plutonium isotopics must resemble those of spent commercial fuel (at least 20% Pu-240). To satisfy these requirements, 10 PDR600 units are necessary. Phase I extension efforts indicate the potential to increase the enrichment to 6.2% Pu while decreasing the average burnup to 30,000 MWD/MTM, thereby reducing the number of required PDR600 units to 7.

Employment of Westinghouse "full-size", 4-Loop, 3411 MWt plants reduces the absolute number of units required to denature the stockpile. Based on similar enrichments and burnups as mentioned previously for the PDR600, 6 units would be necessary if 5.5% enriched Pu fuel were burned to 40,000 MWD/MTM, whereas only 4 units would be required if 6.2% enriched Pu fuel were burned to 30,000 MWD/MTM.

A cursory examination was performed to assess the control rod shutdown margin for a Standard Westinghouse 4-Loop Plant fully loaded with 5.5% Pu enriched MOX. Control rod patterns chosen were: the standard 53 control rod cluster positions, 73 control rod cluster positions (53 standard plus 12 spare and 8 part-length positions), and 85 positions that complete the "checkerboard pattern" of possible control rod cluster positions. The 53 rod pattern exists in many operating plants, the 73 rod pattern would require some modification of existing plants but does not require additional head penetrations, while the 85 rod pattern requires a new head design. The 53 position pattern is clearly not sufficient. Calculations indicate the 73 position pattern provides a small margin at BOL, but extrapolation of normal trends in worths and requirements indicate no margin will exist at EOL. Estimates for the 85 position pattern suggest that this configuration may be adequate for all times-in-life.

These control rod margin conclusions are based primarily on uranium fuel values for feedbacks and steamline break, and a thorough evaluation with MOX fuel must be performed in Phase II.

Pu Destruction Option:

Little effort was spent on this option. Mass flow evaluations indicate 6 Standard Westinghouse 4-Loop Plants are required to dispose the stockpile, vs. 15 PDR600 units. Since this option has only a maximum enrichment of 4% Pu, the 73 control rod cluster positions may be adequate for all times-in-life, but no calculations were performed to verify this.

VOIDING COEFFICIENT

The moderator voiding coefficient was predicted for the spent fuel option (also applies to the spiking option). Conditions used were beginning-of-cycle with the maximum soluble boron level. Independent models were generated with the TORTISE (2-D spatial) and PHOENIX (2-D L[illegible]e) codes, with both codes predicting negative voiding coefficients. The value for hot-full-power at beginning-of-cycle is approximately -70 pcm/%Void. The minimum moderator density employed in either code package is about 0.3 gm/cc, because they experience difficulty in predicting voiding below this level. In Phase II, the KENO-V.a code will be used to predict values for the very low density range.

CORE PHYSICS SUMMARY AND CONCLUSIONS

The Phase I Extension Program is a more complete and in-depth study of PDR600 core design and analyses, which commenced during Phase I. The original model has been revised to accommodate sufficient negative moderator temperature coefficients at all operating conditions. New parameters, such as void coefficient, xenon worth and core parameters at cold conditions, have been calculated to complete the design envelope. Equilibrium cycle core models have been developed, where nuclear characteristics are of a more generic representation of core behavior. In addition, an evaluation has been made to compare differences between small and larger reactors.

The design flexibility was reconfirmed during this extension period. Core reactivity and peaking factor parameters are under the AP600 limits so that the licensing activity can be minimized. All calculations were performed with the latest Westinghouse code system (Phoenix-P/ANC), which was validated against the operating PWR plant (Beznau 1), providing additional confidence in critical core physics and safety parameters. Considering the larger deviation of the PDR600 neutron spectrum from the uranium fueled core, validation activity of the code system will be continued in Phase II. It must also be emphasized that the spent fuel option of PDR600 has been analyzed using the proven fuel rod, assembly, burnable absorbers, and control rod system designs while maintaining the thermal-hydraulic core conditions similar to AP600.

2.2 Reactor Safety

Steam System Piping Failure

The rupture of a main steam line is an event that has the potential to be adversely affected by some of the anticipated nuclear design characteristics associated with the PDR600 plutonium core. Specifically, a more negative moderator temperature coefficient and reduced boron worth are two aspects of a plutonium based core design that may alter the general passive plant response to the steam line break event. Reference 1 documents the steam line break results for the AP600 plant, which uses a "standard" uranium core design, but has all the same basic passive protection features found in the PDR600. From a safety analysis perspective, with the exception of certain events strongly affected by core design related differences, the PDR600 transient plant response is generally expected to closely reflect that of the AP600. The non-LOCA analysis reported in Reference 2 presents results for three transients thought to be strongly tied to the core design. The analysis that follows, for the steam line break, simply extends the analysis of Reference 2 to an additional accident.

Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system (RCS) causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high-power peaking factors predicted when the most reactive RCCA is assumed to be stuck in its fully withdrawn position. The core is ultimately shut down by boric acid solution delivered by the passive core cooling system.

As discussed in Reference 1, the analysis of the main steam line rupture with respect to core response, is performed to demonstrate that the following Standard Review Plan Section 15.1.5 (Reference 3) evaluation criterion is satisfied:

> Assuming the most reactive stuck RCCA, with or without offsite power, and assuming a single failure in the engineered safety features (ESFs), the core cooling capability is maintained. Radiation doses do not exceed the guidelines of 10 CFR 100.

Although departure from nucleate boiling (DNB) and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the analysis of Reference 1 demonstrates that, for the AP600, the DNB design basis is not exceeded for any such rupture, even assuming the most reactive RCCA stuck in its fully withdrawn position. The analysis reported here for the PDR600 does not include explicit consideration of DNB, but instead presents only the overall transient predicted for the steam line break event.

A major steam line rupture is classified as an American Nuclear Society Condition IV event, a limiting fault. Core response effects of minor secondary system pipe breaks are bounded by the analysis for a main steam line rupture. The major rupture of a steam line is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown, thereby reducing the likelihood that the reactor will return to power. The analysis of this transient, presented here, considers only the most limiting break size, a double-ended rupture. Many of the key assumptions used in this analysis are discussed in Reference 4, which also includes consideration (on a generic basis) of a spectrum of break sizes and power levels.

For the PDR600, the following functions provide the protection for a steam line rupture:

- Core makeup tank actuation from any of the following:

  1- Two out of four low pressurizer pressure signals.

  2- Two out of four high-1 containment pressure signals.

  3- Two out of four low steam line pressure signals in any loop.

- The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safeguards signal.

- Redundant isolation of the main feedwater lines.

  Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves, a safeguards signal will rapidly close all feedwater control valves and back up feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

- Trip of the fast acting main steam line isolation valves (MSIVs) (designed to close in less than 10 seconds) on:

1- Two out of four high-1 containment pressure.

2- Two out of the four low steam line pressure signals in any one loop (above permissive P-11).

3- Two out of four high negative steam pressure rates in any one loop (below permissive P-11).

Two fast-acting main steam isolation valves are provided in each steam line; these valves fully close within 10 seconds of actuation following a large break in the steam line. For breaks downstream of the main steam line isolation valves, closure of at least one valve in each line will completely terminate the blowdown. For any break in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the main steam line isolation valves fails to close.

Flow restrictors are installed in the steam generator outlet nozzle as an integral part of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

ANALYSIS OF EFFECTS AND CONSEQUENCES

Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

- The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code (Reference 5) has been used.

- For the analysis of Reference 1, the detailed thermal and hydraulic behavior of the core following a steam line break is also predicted using the THINC computer code (Reference 5). This code is then used to determine if DNB occurs for the core transient conditions computed by the LOFTRAN code. For the current PDR600 analysis, this explicit DNB calculation has not been performed. Instead, the analysis has been limited to the LOFTRAN predictions noted above. The current evaluation is intended to examine the core average power and gross plant response to a steam line break, assuming nuclear design parameters are assigned values consistent with expected PDR600 performance.

The following conditions are assumed to exist at the time of a main steam line break accident:

- End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted by the insertion limits so that the shutdown margin requirements are satisfied. Consistent with the Reference 1 assumptions for the AP600, a shutdown margin of 1.6 percent is used.

- A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The core power reactivity feedback is modeled as a function of thermal power and core mass flow. The core properties used in LOFTRAN for feedback calculations are generated by combining those in the sector nearest the affected steam generator with those associated with the remaining sector. The resultant properties reflect a combination process that accounts for inlet plenum fluid mixing and a conservative weighting of the fluid properties from the coldest core sector.

  For the current PDR600 analysis, the specific weighting factors and feedback parameters used are those from the AP600 analysis of Reference 1. Application of this input to this PDR600 scoping study is judged reasonable, based on a comparison of the predicted nuclear design parameters for the PDR600 and AP600 cores, as defined in References 2 and 1, respectively.

- A conservative boron worth of -3.0 pcm/ppm, which is much less negative than generally predicted for a Westinghouse pressurized reactor design at end-of-cycle conditions. In fact, this value is only about one-third of that typically seen. Relative to the steam line break event, the concern is that this low boron worth may significantly reduce the effectiveness of the core makeup tanks and the accumulators (if actuated) in mitigating the transient.

- The portions of the passive core cooling system used in mitigating a steam line rupture are the core makeup tanks and the accumulators. There are no single failures which will prevent core makeup tank injection. In modeling the core makeup tanks and accumulators conservative assumptions are used that minimize the capability to add borated water. Specifically, the core makeup tank injection line characteristics modeled reflect the failure of one core makeup tank discharge valve.

- The initial boron concentration in the PDR600 core makeup tanks is assumed to be 5000 ppm. This compares with the 3200 ppm assumed for the AP600 steam line break analysis of Reference 1. The initial boron concentration in the PDR600 accumulators is assumed to be 3500 ppm compared with the 1900 ppm assumed for the AP600 steam line break analysis of Reference 1. The indicated increases in both these parameters are intended to offset the effects of the reduced boron worth and are considered to be acceptable from a plant operations perspective.

- Maximum overall fuel-to-coolant heat transfer coefficient, to maximize the rate of RCS cooldown.

- Since the steam generators are provided with integral flow restrictors having a 1.4-square foot throat area, any rupture with a break area greater than 1.4-square feet, regardless of location, has the same effect on the primary plant as the 1.4-square-foot-double-ended rupture. The limiting case considered for determining the core power and RCS transient is the complete severance of a main steam line with the plant initially at no-load conditions, with full reactor coolant flow, and offsite power available.

  The assumption that offsite power remains available throughout the transient is consistent with the methodology defined in Reference 4. Within the context of the analysis, the continued availability of offsite power translates into the reactor coolant pumps continuing to operate throughout the course of the event. This maximizes the RCS cooldown, thereby increasing the magnitude of the subsequent return to power. However, the passive protection features of the PDR600 act in such way as to minimize the impact of the offsite power assumption on the final transient results. Specifically, the protection system for the PDR600 automatically provides a safety-related signal that initiates the coastdown of the reactor coolant pumps in parallel with core makeup tank actuation. Since this reactor coolant pump function is actuated early in the event, there would be very little difference between the predicted results for cases with and without offsite power.

- As indicated earlier, the current PDR600 steam line break analysis employs substantial nuclear design input from the AP600 analysis of Reference 1. Implicit in these inputs is consideration of power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures as determined for end of core life. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The predicted power peaking factors account for the effect of local voids in the region of the stuck RCCA during the return to power phase following the steam line break. These voids act in conjunction with the large negative moderator temperature coefficient to partially offset the large peaking factors associated with the assumed stuck RCCA. The magnitude of the power peaking factors depends upon the core power, temperature, pressure, and flow, and are different for each case studied.

The analysis for the steam line break assumes initial hot standby conditions at time zero since this represents the most pessimistic initial condition. If the reactor is just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip setpoint. Following a trip at power, the RCS contains more stored energy than at no-load; the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy reduces the cooldown caused by the steam line break before the no-load conditions of reactor coolant system temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same general manner as in the analysis which assumes no-load condition at time zero. However, he resulting transient is less limiting with respect to DNB considerations.

- In computing the steam flow during a steam line break, the Moody Curve (Reference 6) for f(L/D) = 0 is used.
- Perfect moisture separation in the steam generator
- Maximum cold startup feedwater flow plus nominal 100 percent main feedwater flow
- Four reactor coolant pumps are initially operating
- Manual actuation of the passive residual heat removal system at time zero is conservatively assumed to maximize the cooldown

Results

The calculated sequence of events for the reported case is shown on Table 2-1. The results presented are a conservative indication of the events which would occur assuming a steam line rupture, since it is postulated that all of the conditions described above occur simultaneously.

Core Power and Reactor Coolant System Transient

Figures 2-1 through 2-10 show the reactor coolant system transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load conditions.

Offsite power is assumed available so that, initially, full reactor coolant flow exists. During the course of the event, the reactor protection system initiates a coastdown of the reactor coolant pumps in conjunction with actuation of the core makeup tanks. Steam release from more than one steam generator will be prevented by automatic trip of the fast-acting main steam isolation valves in the steam lines on high containment pressure or low steam line pressure signals. Even with the failure of one valve, release is limited to no more than 10 seconds for the other steam generator while the one generator blows down. The main steam isolation valves fully close in less than 10 seconds from receipt of a closure signal.

As shown in Figure 2-2, the core attains criticality with the RCCAs inserted (with the design shutdown margin assuming one stuck RCCA) before significant amounts of boron solution from the core makeup tanks or the accumulators enters the RCS. A peak core power significantly lower than the nominal full-power value is attained.

The calculation of core boron concentration assumes the boric acid is mixed with and diluted by the water flowing in the RCS before entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and from the core makeup tanks or accumulators (or both). The variation of mass flow rate in the RCS due to water density changes is included in the calculation. So is the variation of flow rate from the core makeup tanks or accumulators (or both) due to changes in the RCS temperatures, RCS pressure, and the pressurizer level. The RCS and passive injection flow calculations include explicit modeling of the associated line losses.

At no time during the analyzed steam line break event do the core makeup tank levels even approach the setpoint for the actuation of the automatic depressurization system. The combination of injection flow from the accumulators and recirculation flow from the RCS cold legs into the core makeup tanks maintains a relatively large core makeup tank water inventory throughout the event.

The passive residual heat removal (PRHR) system provides a passive, long-term means for removing the core decay and stored heat by transferring the energy via the PRHR heat exchangers to the in-containment refueling water storage tank (IRWST). Normally, the PRHR is actuated automatically when the steam generator level falls below the low wide range level setpoint. For the main steam line rupture case presented, the PRHR system is conservatively actuated at time zero to maximize the cooldown.

Margin to Critical Heat Flux

A PDR600 specific DNB evaluation analysis has not been performed as part of the current effort. However, such an analysis has been performed for the AP600 as documented in Reference 1. The similarities between the plant responses for the limiting AP600 case presented in Reference 1 and the representative PDR600 case documented here suggest that a PDR600 design can be developed that would meet the DNB limits associated with the steam line break event.

Conclusions

The analysis shows that for a representative case, using approximate nuclear design inputs to model the response of a plutonium based core design, the predicted response of the PDR600 to a limiting steam line break event is consistent with that predicted for the AP600. The analysis of Reference 1 documents that with its uranium based core design, the AP600 design meets all the safety related criteria associated with the steam line break event. The assumptions used in the current PDR600 analysis include the use of higher boron concentrations in the plant passive protection system, but the specific values modeled should not impose any excessive burden on plant operations. Based on this analysis, it is expected that a detailed, PDR600 specific steam line break analysis would produce acceptable results that meet all the applicable licensing requirements.

PDR600

References

1. "Simplified Passive Advanced Light Water Reactor Plant Program, AP600 Standard Safety Analysis Report," DE-AC03-90SF18495, June 26, 1992.

2. "PDR600, Plutonium Disposition Study for the Department of Energy, San Francisco, CA," DOE/SF/19683--1, DE-AC03-93SF19683, May 15, 1993.

3. NUREG-0800, US NRC Standard Review Plan, Section 15.1.5, "Steam System Failures Inside and Outside Containment (PWR)," Rev. 2, July 1981.

4. Wood, D. C., and Hollingsworth, S. D., "Rector Core Response to Excessive Secondary Steam Releases," WCAP-9226 (Proprietary), WCAP 9227 (Nonproprietary), January 1978.

5 Appendix B of Reference 2 (above) .

6. Moody, F. S., "Transactions of the ASME," Journal of Heat Transfer, Figure 3, page 134, February 1965.

Table 2-1

Time Sequence of Events for the Main Steam Line Rupture Accident

Event	Time (seconds)
Steam line ruptures	0
Safeguards actuation signal on low safeguards steam line pressure	0.8
Main steam line isolation	12.8
Main feedwater isolation	12.8
Pressurizer empty	13.6
Core makeup tank actuation	22.8
Boron reaches core	44.8
Criticality obtained	51.2

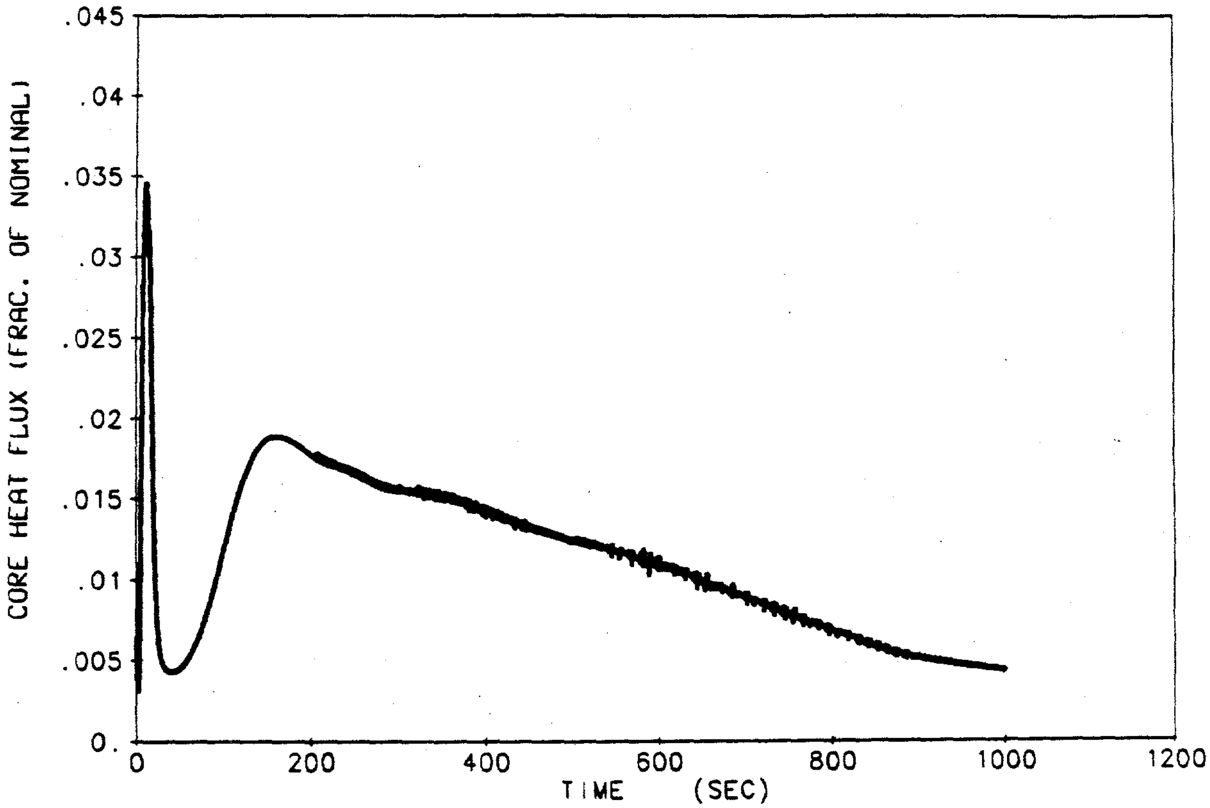


Figure 2-1
Full Double-Ended Steam Line Rupture Accident

Core Heat Flux vs Time

PDR600

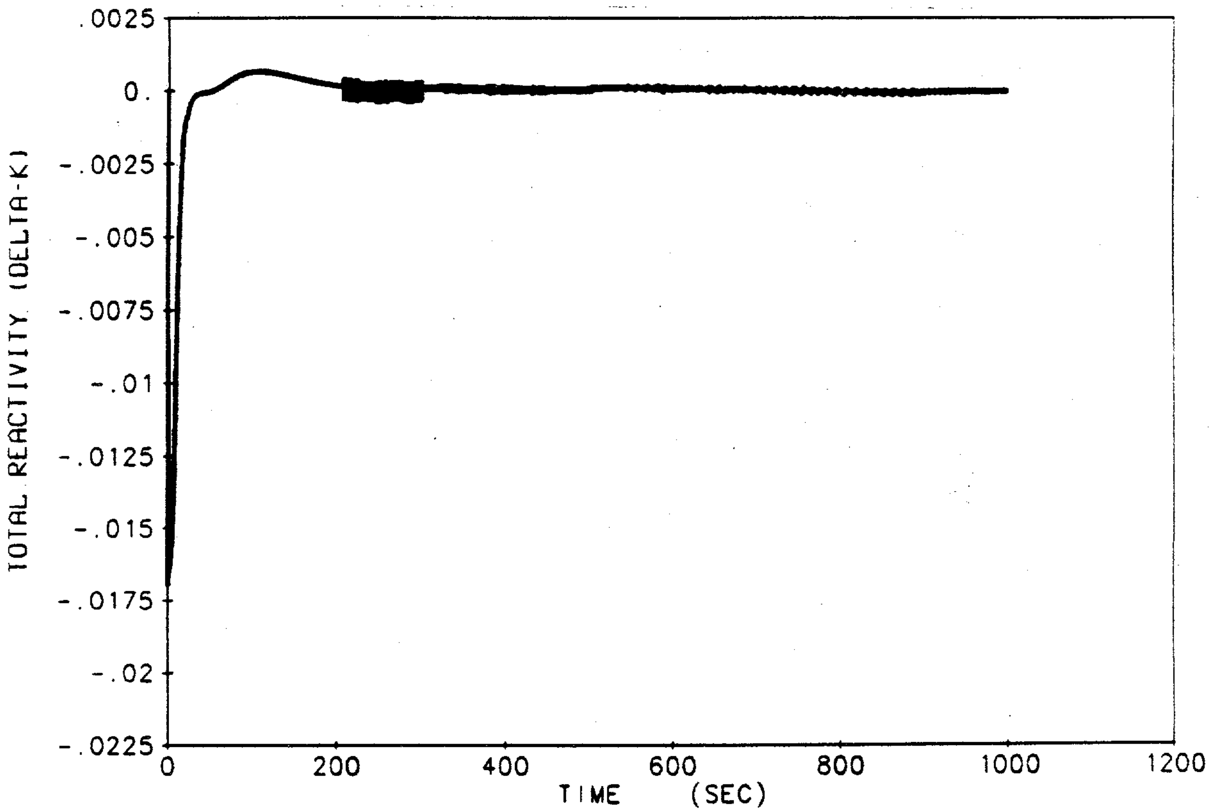


Figure 2-2
Full Double-Ended Steam Line Rupture Accident

Total Reactivity vs Time

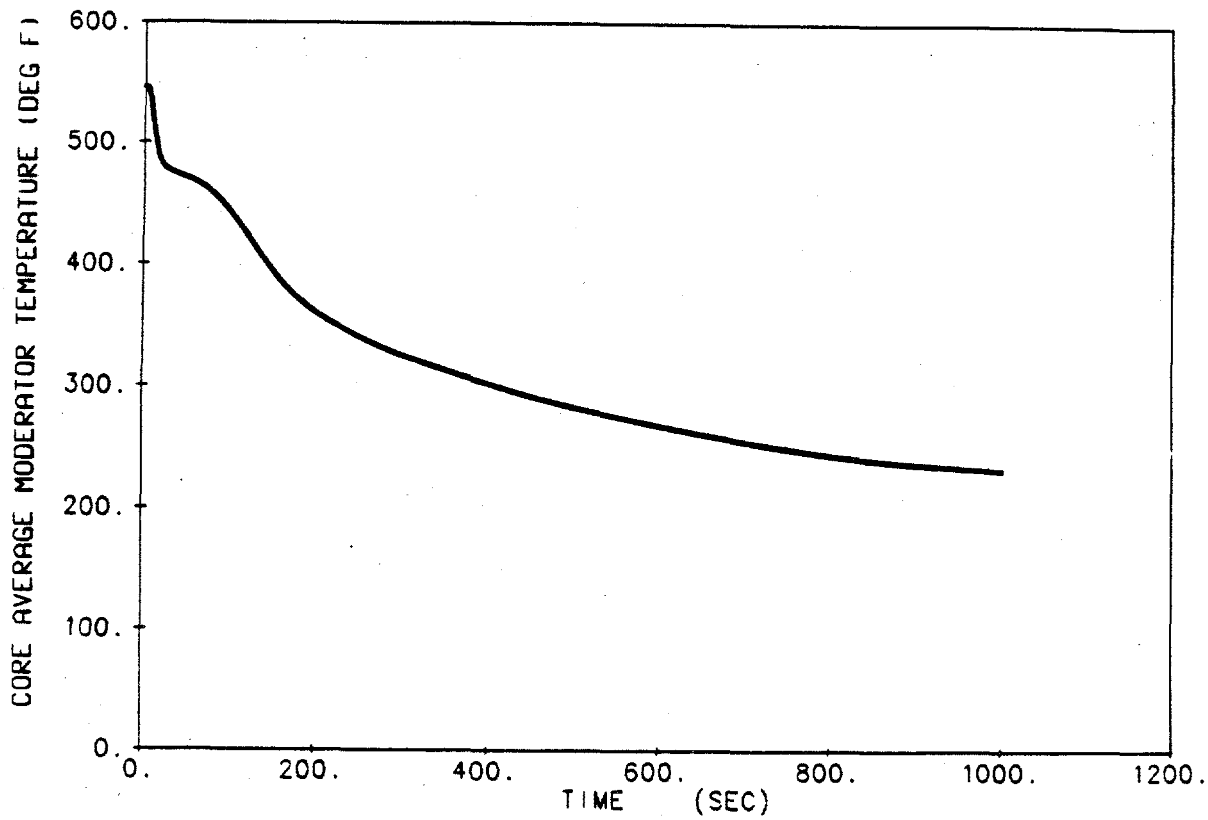


Figure 2-3
Full Double-Ended Steam Line Rupture Accident

Core Average Moderator Temperature vs Time

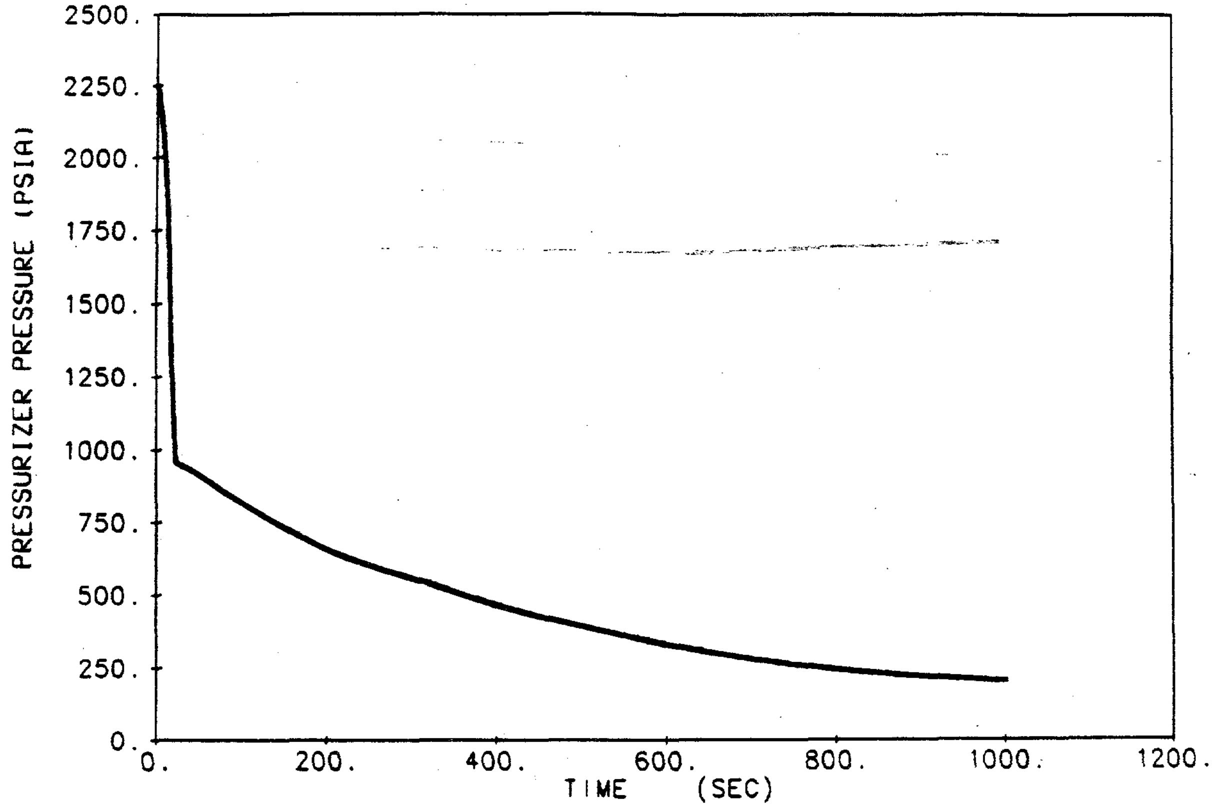


Figure 2-4
Full Double-Ended Steam Line Rupture Accident

Pressurizer Pressure vs Time

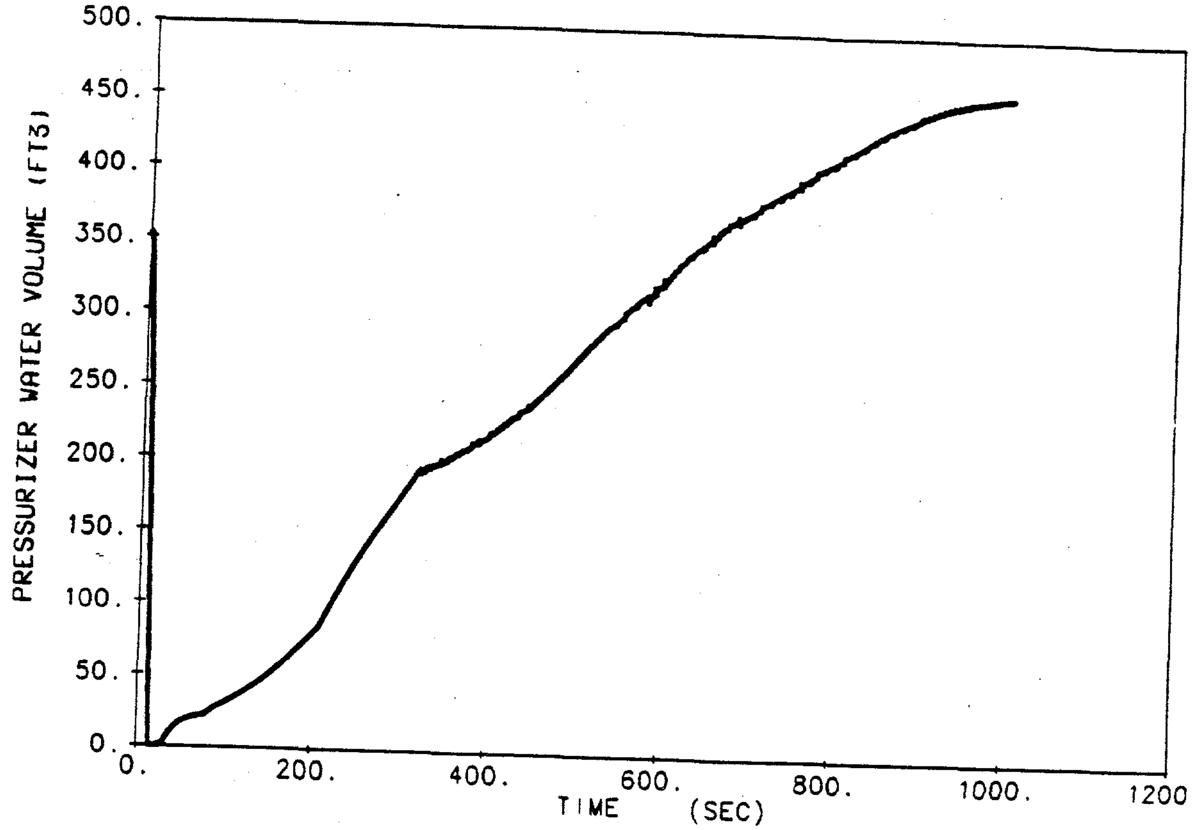


Figure 2-5
Full Double-Ended Steam Line Rupture Accident

Pressurizer Water Volume vs Time

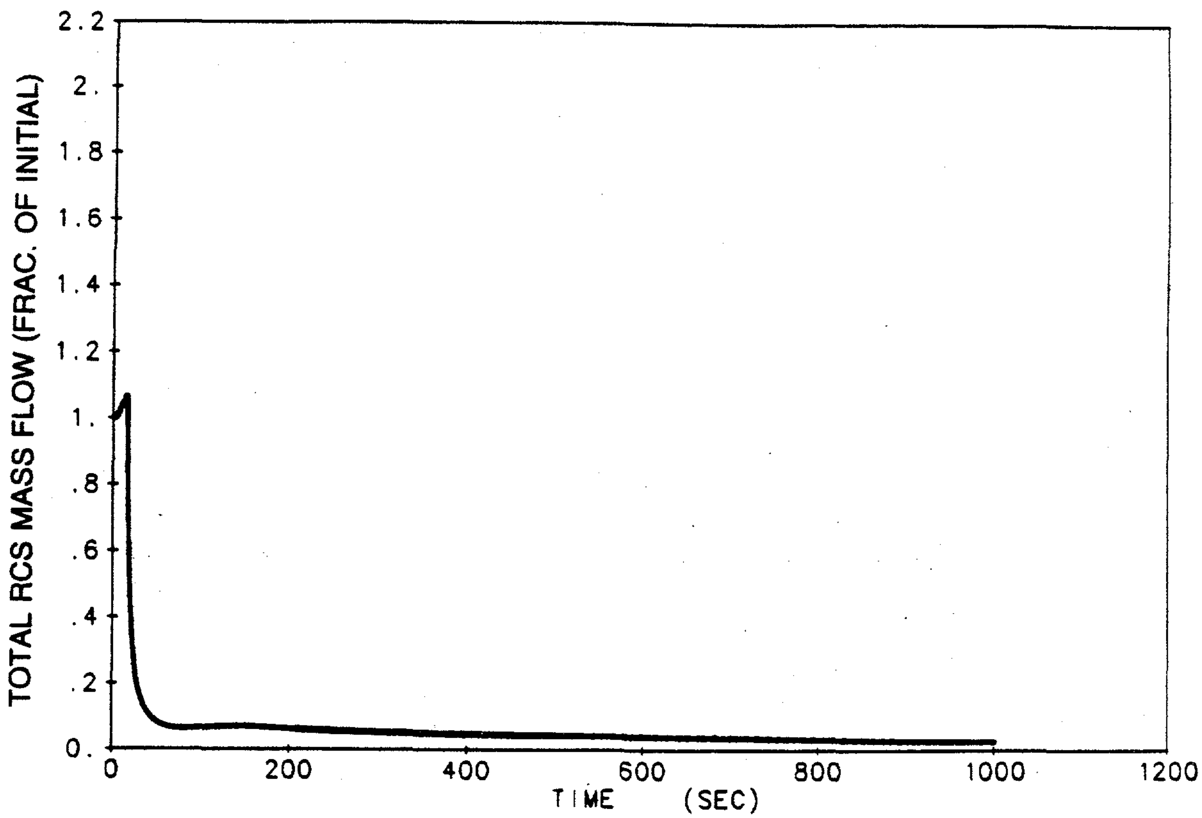


Figure 2-6
Full Double-Ended Steam Line Rupture Accident

Total Reactor Coolant System Mass Flow Rate vs Time

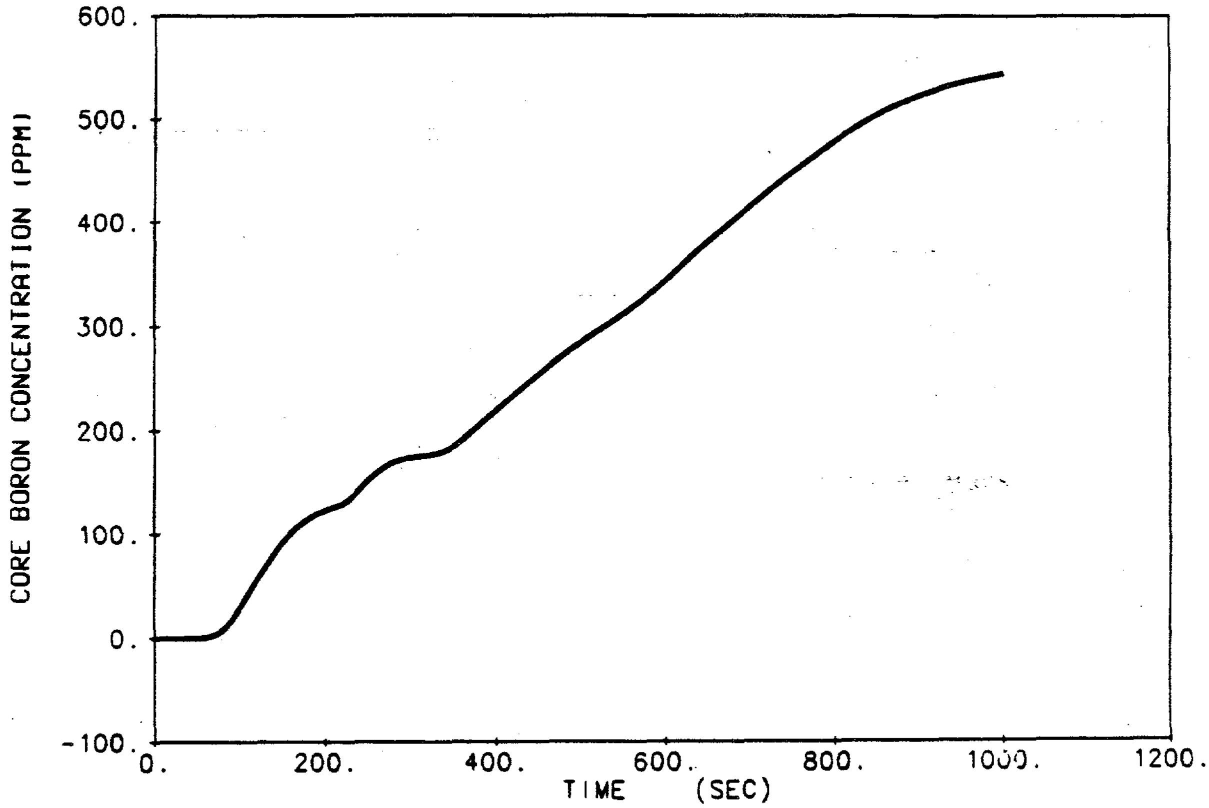


Figure 2-7
Full Double-Ended Steam Line Rupture Accident

Core Boron Concentration vs Time

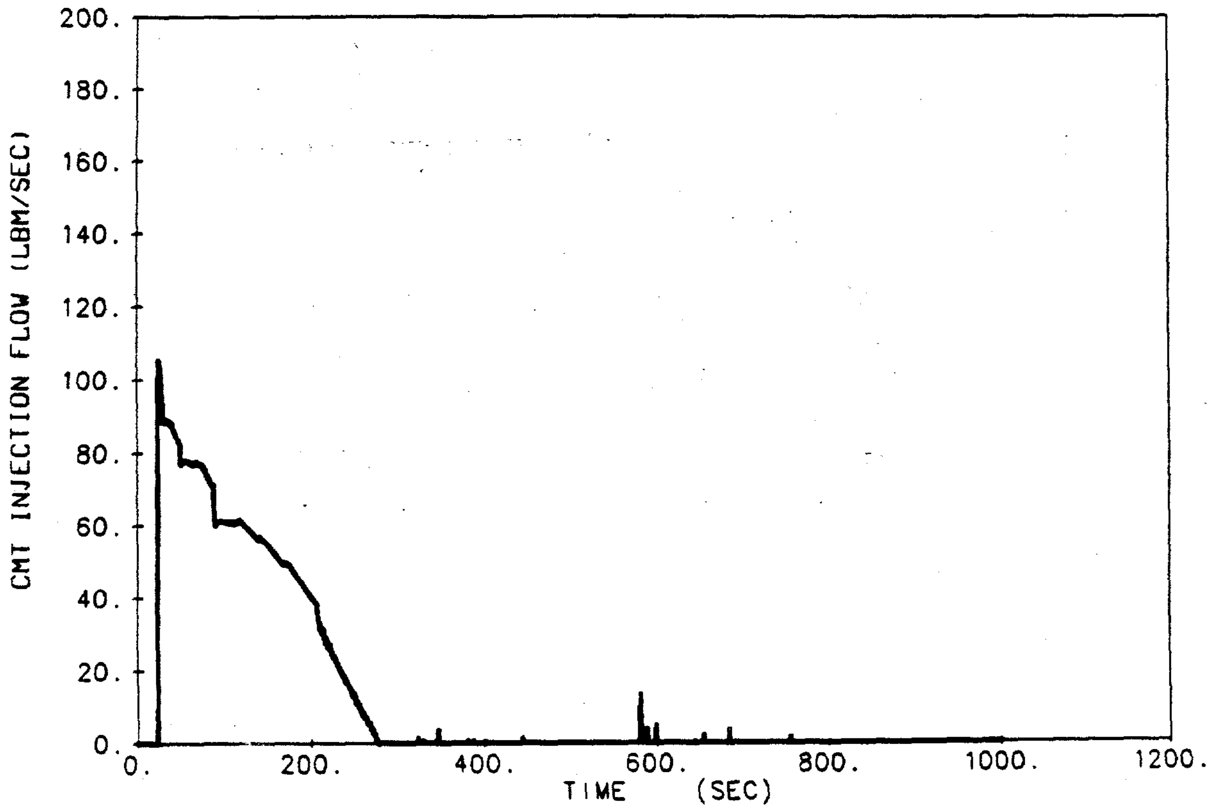


Figure 2-8
Full Double-Ended Steam Line Rupture Accident

Core Makeup Tank Injection Flow vs Time

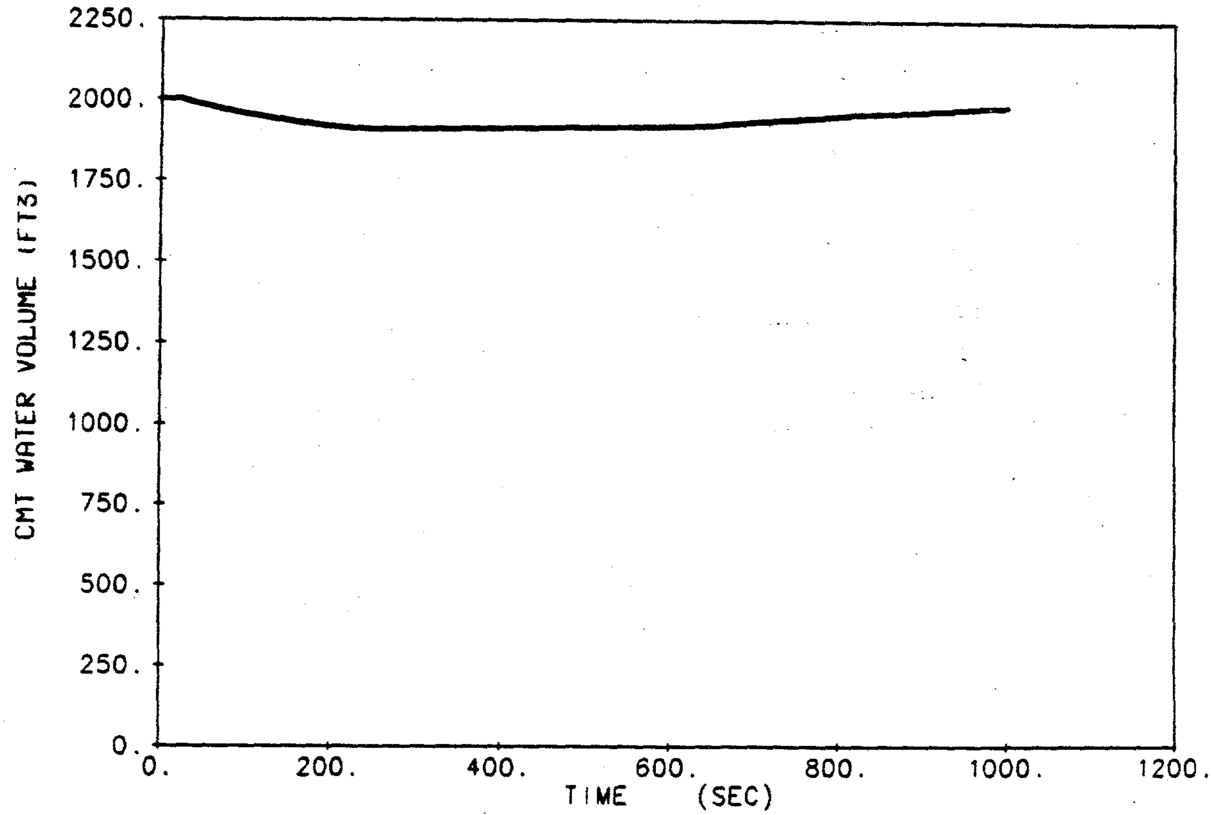


Figure 2-9
Full Double-Ended Steam Line Rupture Accident

Core Makeup Tank Water Volume vs Time

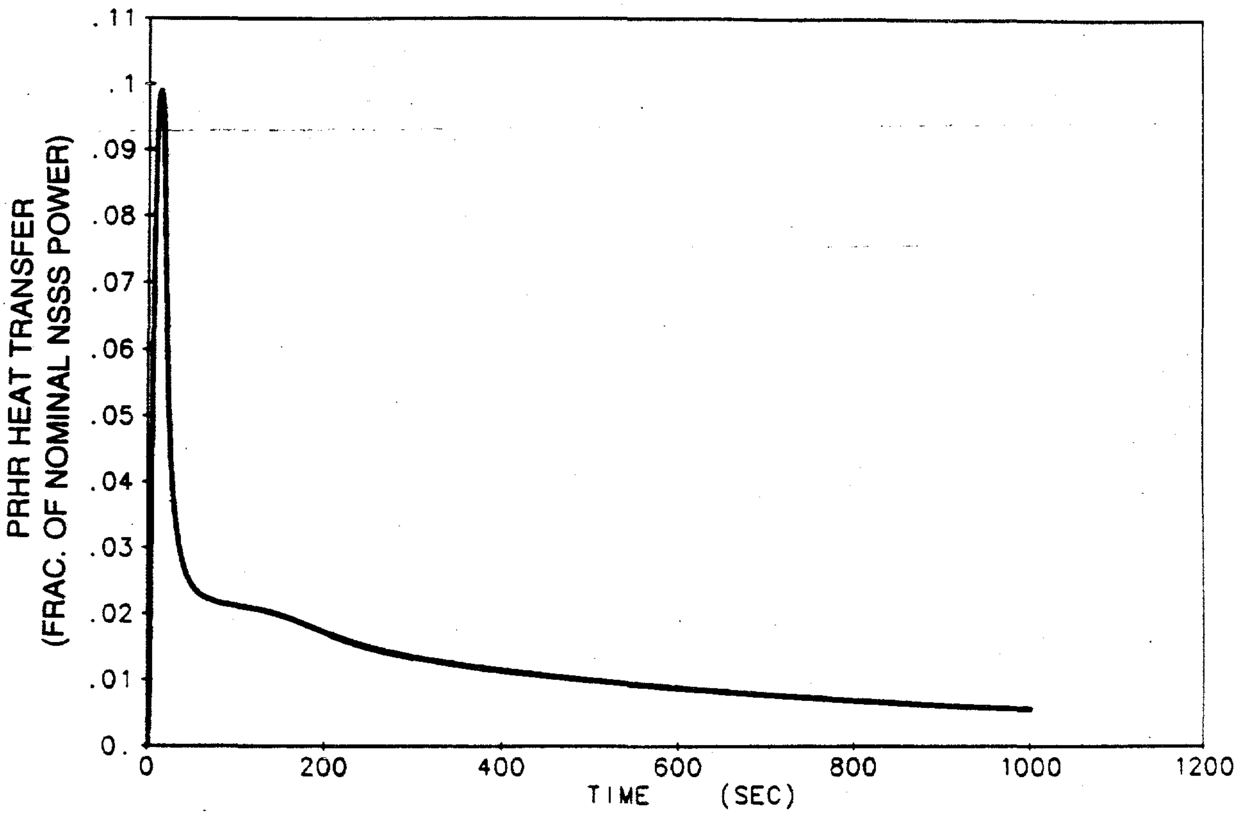


Figure 2-10
Full Double-Ended Steam Line Rupture Accident

Passive Residual Heat Removal Sytem Heat Transfer vs Time

3.0 Fuel Cycle

3.1 Fuel Fabrication

3.1.1 Storage Provisions

A study was done to identify material storage requirements for a MOX fuel fabrication facility. The study is summarized as follows:

DEPLETED URANIUM OXIDE STORAGE

Depleted uranium oxide is received in 55 gallon drum shipping containers which corresponds to full production for one month. The material is transferred from the shipping containers into four storage tanks, two for feed to the dissolver tanks, and two for blending with the MOX master blend co-precipitation product. Each tank has the capacity for two and a half day's supply. In the case of the blending tanks, 13 containers are used to fill each of the 3510 Kg capacity tanks; the dissolver feed tanks require 4 containers for each batch and have a nominal capacity of 990 Kg.

The powder is transferred from the storage tanks to the uranium oxide feed hoppers. The described arrangement is based on the receipt of depleted uranium dioxide and requires that the dissolver feed is first oxidized to uranium trioxide before dissolution.

MOX STORAGE

Nine silos are provided for the storage of prepared mixed oxide powder for each production line. Each silo is sized to furnish 225 Kg per batch of MOX powder feed to the compaction presses.

GREEN PELLET STORAGE

The green pellet storage system is designed to handle 247 MT of heavy metal fuel per year. This corresponds to 123.5 MT for each of the two production lines. 33 Boats of 16 Kg capacity per boat are used for green pellet storage for each line. The physical size of the boats is 12" by 12".

LOW DENSITY PELLET STORAGE

Low density pellets are recycled to the sintering furnace on a boat basis. Provision is made to store 21 boats line of low density pellets for each line. The boats have a capacity of 15 Kg each and are 12" by 12" in size.

SINTERED PELLET STORAGE

Sintered pellets are stored on an accumulation conveyor for six hours until the QC evaluation is completed. 29 boats of 16 Kg capacity are used in each production line for the QC holdup of each batch of sintered pellets.

ACCEPTABLE PELLET STORAGE

Storage trays provide for interim storage of 4600 Kg of acceptable pellets (2300 Kg per line). The pellet storage trays are 15" wide by 25" long and hold 25 rows of pellets, each row is twenty four inches long. The trays are placed in a storage area which is temperature, pressure, and humidity controlled. The discharge section contains provisions for elevated temperature drying.

FUEL ROD STORAGE

Completed, inspected fuel rods are stored in a centralized rod storage area which contains the rods from both assembly lines. The rod storage unit is designed to provide compartmentalized storage for acceptable fuel rods prior to fuel bundle assembly. Fuel rods loaded on storage trays are transferred to this area from the inspection stations. A remotely-controlled stack retriever unit provided for several months supply of $(Pu/U)O_2$ fuel rods.

A separate rod storage unit is provided to store the LEU fuel rods used for the tritium production option.

FUEL ASSEMBLY (BUNDLE) STORAGE

Completed, inspected fuel bundles are transferred into this storage area by crane in a vertical position with crane speeds limited to 60 inches per minute horizontally and 15 inches per minute vertically.

Racks are provided for the vertical storage of 484 fabricated fuel assemblies ready for shipment. The racks are designed to prevent damage to the fuel assemblies during normal and seismic conditions. The racks are similar to those used in the Auxiliary Building for new fuel storage and occupy approximately 530 ft^2 of storage area.

3.1.2 Radiation Dosage From MOX

Estimated doses from mixed oxide unirradiated fuel have been made to indicate the possibility of shielding needs during fabrication of this fuel. Results are summarized in Table 3-1. The estimated doses are based on the latest standard, ANSI/ANS-6.1.1-1991, for neutron and gamma-ray fluence-to-dose factors for exposure to an anthropomorphic phantom. For some cases, the latest values are less conservative than earlier standards. The dose results indicate that both gamma and neutron radiation levels should be considered when procedures are developed for handling quantities of the mixed oxide fuel.

The fuel was assumed to be 95% depleted uranium (0.2% U-235) and 5% weapons grade plutonium (initial composition: 0.05% Pu-238, 93.6% Pu-239, 5.9% Pu-240, 0.4% Pu-241, and 0.05% Pu-242). It was further assumed that the plutonium would have 2 x 10^{-9} parts of Pu-236 (2 x 10^{-7} atom %). The plutonium was assumed to have decayed for 15 years after separation which results in decay products of the Pu-236 and about 0.2% Am-241 from decay of the Pu-241.

The consequences of the 15 year decay and the gamma and neutron source from the mixed oxide was calculated using the ORIGEN2 code. It was found that the most important radiation dose from bare fuel is due to the Am-241 low energy gamma rays. These gamma rays can be easily shielded. Additional gamma rays also come from Pu-239 and from the Pu-236 decay products. These sources become significant when the low energy gamma rays are shielded. In addition, the fuel produces neutrons from a combination of (n,α) reactions on oxygen and from spontaneous fission (primarily from Pu-240). The neutrons constitute only a small fraction of the unshielded dose (about 5%) but can become an appreciable fraction of the dose when the gammas are shielded (about 25% with 1 mm lead shielding).

For the unshielded fuel, a majority of the dose comes from Am-241 low energy gamma rays. These gamma rays are easily shielded and do not contribute significantly to the dose with 1 mm of lead shielding. In the shielded case, a majority of the dose comes from Pu-239 higher energy gamma rays. Shielding the higher energy gamma rays and the neutrons will require additional shielding to minimize exposure.

Values in Table 3-1 are given for a case of a single pellet for both no shielding and for shielding by 1 mm of lead. It is seen that the exposure is small except very near the pellet, and that the lead provides a shielding factor of about 5. Comparison with preliminary dose values reported earlier (Table 1.1.6.4-3 of the Plutonium Disposition Study) indicates that the preliminary values were conservative for the bare case (4.3 mrem/hr vs the latest value of 2.7 at 1 cm). However with 1 mm of lead shielding, the preliminary dose rates underestimated the dose (0.2 mrem/hr vs 0.48 at 1 cm) since the contribution of the higher energy gamma rays and neutrons were not included in the earlier study. Due to the rapid changes in absorption cross section with energy for low energy gamma rays, the error in the results is relatively large. A margin of a factor of 2 should be allowed for the bare dose rates.

Extrapolating fro- - single pellet, the exposures for 1 kg of fuel material are estimated. These values are upper limits since no further self-shielding of the material was included. Estimates were also made of a long column of fuel, as for fabricating a single pin. The doses for a single pellet may be used for contact in these cases also. It is seen that even with these latter quantities of pellets, doses are small except very close to the fuel and the main concern will be exposure to the hands during glove box operations.

Table 3-1

Estimated Dose from Mixed Oxide Fuel

Distance from Pellet	Dose (mrem/hr) No Shield	Dose (mrem/hr) 1 mm Lead Shield
Single Pellet		
Contact	35	----
1 mm	17	2.8
1 cm	2.7	0.48
1 foot	.005	.001
1 m	.0005	8×10^{-5}
1 kg of Material		
1 foot	1.0	0.18
1 m	0.09	0.016
Single Pin		
1 foot	0.35	0.064
1 m	0.10	0.019

3.2 Post Irradiation Fuel Handling

As part of the phase Ib effort, a conceptual study was initiated to determine the optimum plant layout for a spent fuel pool (for the plutonium spiking option) that could accommodate 4350 fuel assemblies. That number of assemblies must be stored following the irradiation of 100 MT of Pu-239 at one reactor (30 cycles at 145 assemblies each cycle). The study was performed when the plutonium spiking option or combination of options was still being considered.

The spent fuel pool layouts were based on fuel storage racks having a center-to-center spacing of 10.9 inches. That is the same rack spacing used for the Spent Fuel Option. In addition, the plant model used for the study was the AP600 plant.

The objective of the study was to minimize the floor space required for the spent fuel pool by integrating the spent fuel assemblies with the remainder of the plant. Two spent fuel pool layouts were developed. One layout uses a 41 foot wide fuel handling machine such as that planned for the AP600 plant. Two spent fuel pools, separated by the fuel canal, are required to accommodate the fuel assemblies. The two pool layout would occupy approximately 5500 more square feet of plant floor area per floor than the spent fuel option.

A second layout uses a wider, 54 foot wide fuel handling machine such as that used by Northeast Utilities. That layout uses one spent fuel pool that is wider than the pool used for the spent fuel option and would require approximately 3600 more square feet of plant floor area per floor.

In conclusion, the preliminary study shows that if a very large storage area for spent fuel assemblies is desired and the AP600 plant is used as the plant basis, a wider fuel handling machine such as that used by Northeast Utilities should be used so that plant floor area can be minimized.

Drawings of this spent fuel storage concept have generated.

4.0 Cost and Deployment Options

4.1 Plant Configuration Options

4.1.1 Plutonium Disposal

Four PDR600 reactors were configured in different Spiking/Spent Fuel operational modes and their costs, revenues and time period to dispose of 100 Mt of Pu were calculated. Two basic assumptions were made about MOX fuel loading and cycle times. A spiking mode is defined as a cycle of a complete core 66.9 MT MOX in 18 months with a burnup of 12,000 MWD/Mt. A spent fuel mode is defined as 3 cycles over 54 months with a total burnup of 36,000 MWD/Mt. Reactors were either run in a spiking mode or spent fuel mode. When all the plutonium had been fabricated and inserted in reactors, the reactors were then run in a spent fuel mode on uranium fuel. The total number of PDR600 reactors was restricted to 4 with from 0 to 4 reactors operating in a spiking mode and 4 to 0 reactors in a spent fuel mode.

A MOX fabrication plant was sized to the throughput required for the particular configuration of spiking and spent fuel reactors. The fabrication plant capacity, capital costs and production costs are given in Table 4-1.

The net present value of revenues less costs is also shown in Table 4-1 for stainless steel as well as zircaloy cores. The calculations were made for 40 year reactor lives with the first reactor coming on line in 2003. Positive net present values result for all configurations with the 4-reactor spent fuel mode of operation approximately 50% higher than the 4-reactor spiking mode of operation. Disposal of 100 Mt of plutonium varies from 12 to 29 years with stainless steel cores and 17 to 40 years with zircaloy cores.

No reinsertion of the MOX fuel is assumed for the spiking option. Considerable burnup is left in two of the three regions of a spiking option core and reinsertion would lower fuel cycle costs by not having to use as much uranium fuel. The net present values shown are a minimum as higher net present values would occur with reinsertion.

Compared to the negative cash flow of the single PDR600 spiking option which irradiated all the fuel in 2.5 years with no reinsertion, the 18 month spiking irradiation of 12,000 MWD/Mt burnup gives positive cash flows and net present values and demonstrates that the plutonium disposal program could be accomplished with four AP600 reactors.

Table 4-1

Net Present Value of Spiking/Spent Fuel
PDR600 Configurations

	Reactor Configuration				
Spiking Reactors (1 cycle), No.	0	1	2	3	4
Spent Fuel Reactors (3 cycles), No.	4	3	2	1	0
Fab Plant Capacity, Mt MOX/yr.	50	100	150	150	200
Fab Plant Capital Cost[1], $ Millions	222	337	430	430	511
Fab Plant Production Cost, $/Kg HM	761	633	595	595	546
Stainless Steel Core					
Net Present Value, $ Millions	1826	1581	1383	1275	1202
Years to Dispose of 100 MtPu[2]	29	20	16	14	12
Zircaloy Core					
Net Present Value, $ Millions	1882	1617	1333	1240	1152
Years to Dispose of 100 MtPu	40	29	23	19	17

(1) Preoperational plus capital cost
(2) Years from start of first reactor operation

Note: 1 cycle equals 18 months with a burnup of 1200 MWD/Mt
3 cycles equals 54 months with a burnup of 36,000 MWD/Mt
No reinsertion of single cycle fuel is assumed although this would be possible.

4.1.2 Tritium Production

An 600 MW and 1000 MW PDR were analyzed to determine which reactor would be most cost effective in producing tritium for DOE and generating power as an independent power producer (IPP) for sale to a utility. The enrichment penalties and costs for producing tritium in the two reactors are given in Table 4-2.

Table 4-2

Fuel Cycle Costs With and Without Tritium

	600 MW	1000 MW
Equilibrium Region Enrichment, %		
With Tritium	5.0	4.5
Without Tritium	3.5	4.0
Burnup, MWD/MTU	38,255	46,576
Fuel Cycle Cost ¢/KWh		
With Tritium	0.695	0.520
Without Tritium	0.508	0.462
Annual Fuel Cycle Cost, $ Millions		
With Tritium	29.2	36.4
Without Tritium	21.3	32.3
Annual Fuel Cost for Tritium Production	7.9	4.1

The enrichment required to meet tritium production goals was calculated to be 5.0% for the AP600 and 4.5% for the APWR 1000, an increase of 1.5% and 0.5% respectively. The enrichment increase results in an annual fuel cycle cost increase of $7.9 million for the 600 MW plant, and $4.1 million for the 1000 plant.

The two reactors were compared with an IPP form of ownership and base case financial assumptions as shown in Table 4-3. The figure of merit used to compare the two reactors was the 30 year levelized electricity price (constant $) with an internal rate of return (IRR) of 17.5% to the equity shareholder. A secondary goal which needed to be met was that the debt coverage ratio (annual operating profit divided by annual debt service) must be greater than 1.4.

An assumption in the analysis is that DOE will pay the independent power producer $200 million per year for producing the tritium. A sensitivity analysis is performed on the DOE payment as well as different financing parameters and capital cost.

Table 4-3

Base Case Financial Assumptions

Overnight Capital Costs Debt/Equity Share of Ownership Interest Rate on Debt Internal Rate of Return on Equity Minimum Debt Coverage Ratio DOE Payment Construction Loan Interest Rate	$2500/KW (both 600 MW and 1000 MW) 80%/20% 8%/yr. 17.5%/yr. 1.4 $200 million/yr. 8%/yr.

The assumption of an overnight capital cost of $2500/KW and an overall permitting, construction and startup schedule of six years results in an as-built cost of $2.92 billion for the 600 MW plant and $4.87 billion for the 1000 MW plant (4%/yr. inflation). Operating costs were assumed to be 1.2 ¢/KWh for both reactors.

The results of the base case comparison are shown in Table 4-4 together with different assumptions on financing and capital cost. The 600 MW plant is seen to have an advantage over the 1000 MW plant, ranging from 10% to 47% depending on the assumption made. The 600 MW plant advantage stems from the DOE payment being distributed over a smaller number of KWh with the 600 MW plant, thus resulting in a lower required electricity price to make the target equity return. The electricity prices required to make the target return at different DOE payments are shown in Table 4-5 and Figure 4-1. The 600 MW plant is shown at overnight costs of $2500/KW and $2200/KW. At $2500/KW and no DOE payment, the 1000 MW plant electricity price is slightly less than the AP600 because of a marginally lower fuel cost.

At equal electricity prices for both plants, eg. 4.4 ¢/KWhr, the 600 MW plant requires a DOE payment of $100 million per year versus $150 million per year for the 1000 MW plant. The additional fuel payment of $3.8 million per year (Table 1) for the 600 MW plant is not that significant for the level of DOE payment being considered.

Table 4-4

Sensitivity Analysis of Financial Parameters

	30-Year Levelized Electricity Price, ¢/KWh[1]		
	600 MW	1000 MW	% Advantage 600 MW
Base Case	3.25	4.12	26.8
100% Debt Financing[1]	4.16	5.02	20.6
Overnight Cost 600 MW @ $2200/KW 1000 MW @ 2500/KW	2.81	4.12	46.6
$100 Million DOE Payment	4.37	4.80	9.8
9% Debt and Construction Loan	3.54	4.41	24.6
70% Debt/30% Equity	3.78	4.65	23.0

(1) The 30-year levelized electricity price (constant 1993$) required for a 17.5% internal rate of return on equity. For 100% debt financing, the electricity price is that required for a minimum debt coverage ratio of 1.40.

Table 4-5			
	30-Year Levelized Electricity Price, ¢/KWh		
DOE Payment, $ Millions	600 MW plant		1000 MW plant
	$2200/KW	$2500/KW	$2500/KW
0	5.08	5.50	5.47
50	4.51	4.93	5.14
100	3.95	4.37	4.80
150	3.40	3.80	4.45
200	2.81	3.25	4.12

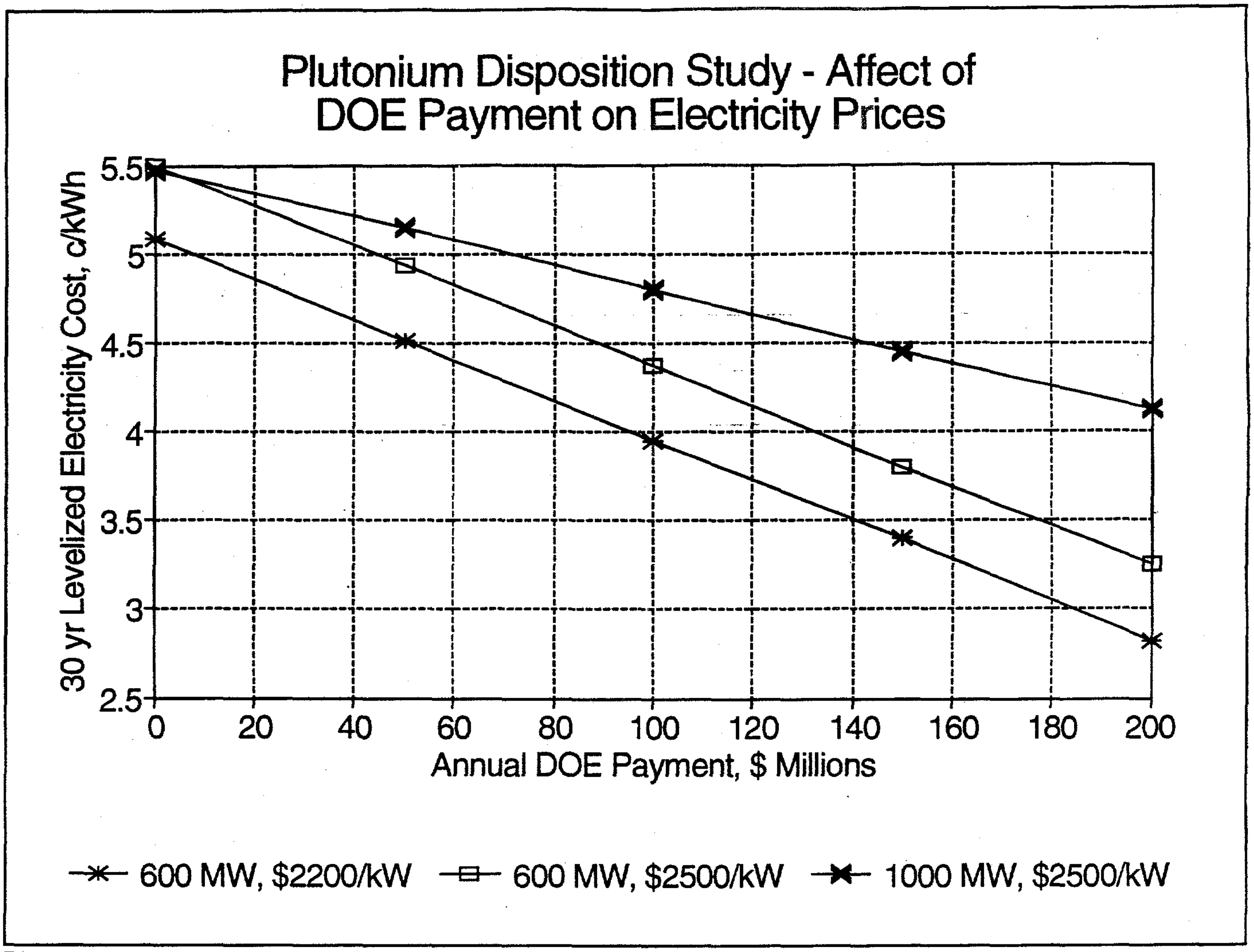

Plutonium Disposition Study - Affect of DOE Payment on Electricity Prices
30 yr Levelized Electricity Cost, c/kWh
5.5
5
4.5
4
3.5
3
2.5
0
20
40
60
80
100
120
140
160
180
200
Annual DOE Payment, $ Millions
600 MW, $2200/kW
600 MW, $2500/kW
1000 MW, $2500/kW

Figure 4-1

4.2 Calculation of Plutonium Value

This study provides details of the plutonium value numbers and calculations presented during the Phase 1a review at Piney Point. The calculations have been refined to be specific for plutonium fissile enrichment and expanded them to show Pu value as a function of:

1) Uranium fuel cycle cost assumptions
2) MOX fuel fabrication costs
3) Stainless steel vs. Zircaloy core
4) Passive, low power density core technology vs. conventional technology

A plutonium indifference value is calculated based on equilibrium regions of Zircaloy-clad uranium fuel at 3.54% enrichment for an AP600 core (passive technology) and 4.2% for a conventional core. The uranium fuel cycle cost is calculated using prices for uranium, conversion, enrichment and fabrication. The disposal charges are assumed to be equal for both uranium and plutonium fuel and therefore do not enter into the calculation. Uranium equilibrium enrichments and fuel cycle costs are shown in Table I for passive and conventional technologies at two different sets of assumptions for fuel cycle costs - the DOE guidelines and current marketplace costs.

Plutonium indifference value, in $/gm-fissile, is calculated by setting a plutonium fuel cycle cost including some value for Pu plus the MOX fabrication cost equal to the uranium fuel cycle cost for the same energy output of each fuel. In our example case, burnups have been set at 40,000 MWD/MT for both uranium and plutonium fuels. At equal burnups:

Value of Pu + MOX fab cost = U_3O_8 + conversion + enrichment + uranium fuel fabrication

In the example case shown in Table 4-5, a plutonium fabrication facility producing 150 MT/yr. of MOX fuel is estimated to have a unit fabrication cost of $850/KgHM for either stainless steel or Zircaloy clad fuel.

Pu enrichments, total and fissile, are given in Table 4-5 for both stainless steel and Zircaloy cores for passive and conventional technologies. At these enrichments, Pu values are calculated for the different fuel cycle cost assumptions as shown in Figure 4-2. The affect of passive AP600 and conventional PWR technologies is shown in Table 4-6.

The differences in uranium fuel cycle cost assumptions result in a factor-of-three difference in Pu value; for the stainless steel core, Pu value increases from about $3/gm at current market costs to $10.8/gm at DOE guideline costs. Pu has a higher value in a Zircaloy core because the Pu enrichments are lower and the fuel is used more efficiently. A 50% increase in Pu value occurs for a Zircaloy core compared to a stainless steel core.

When the PDR600, with its low power density core, is compared to conventional technology at equilibrium uranium enrichments of 4.2%, Pu value is seen to increase by 50% because of better fuel efficiency. Plutonium use in a PDR600 with Zircaloy fuel is seen to have values of $10 to $22/gm, compared to a conventional PWR plant over the fuel cycle cost range shown - ie. if Pu burned in a PDR600 produced power which replaced power from a conventional PWR, the Pu values would be as shown.

Affect of MOX Fabrication Cost on Pu Value

Given the DOE assumption for fuel cycle costs, and a stainless steel or Zircaloy clad core, the variation in Pu value for different MOX fabrication costs can be seen in Table 4-7.

If the MOX fabrication cost equals the uranium fuel cycle cost of $1413/KgHM (DOE guidelines) the Pu value is reduced to zero and one is indifferent, from a cost standpoint, of using uranium or plutonium. At current market conditions, indifferent MOX fabrication cost is $1022/KgHM. With a stainless core, MOX fabrication costs of $850/KgHM and DOE fuel cycle costs, the Pu value is $10.8/gm or $1 billion for 100 metric tonne of plutonium.

Table 4-5

PLUTONIUM INDIFFERENCE VALUES INPUT ASSUMPTIONS

	AP600 Passive Technology Low Power Density Core		Conventional PWR Technology	
	FCC(1)	FCC(2)	FCC(1)	FCC(2)
Uranium Equilibrium Region Enrichment, %	3.54	3.54	4.2	4.2
Uranium Equilibrium Region*	1413	1022	1689	1224
Pu Fabrication cost, $/KgHM	850	850	850	850
Pu Equilibrium Region Enrichment, % (% fissile)				
Stainless Steel Core	5.5 (5.2)	5.5 (5.2)		
Zircaloy Core	4.0 (3.76)	4.0 (3.76)		

*Note: Uranium Equilibrium Fuel Cycle Cost equals the sum of uranium, conversion, enrichment and fabrication costs. Disposal costs are assumed to be equal for the uranium and plutonium fuel cycles.

FCC(1) Fuel Cycle Cost Case 1 (DOE Guideline): Uranium - $25/lb.; Conversion - $10/KgU feed; Enrichment - $125/SWU; Fabrication - $260/KgU.

FCC(2) Fuel Cycle Cost Case 2 (Current Fuel Cycle Prices): Uranium - $15/lb.; Conversion - $7/KgU feed; Enrichment - $100/SWU; Fabrication - $200/KgU.

Table 4-6

Affect of AP600 Passive and Conventional PWR Technologies on Plutonium Value, $/gram (fissile)

	Fuel Cycle Cost DOE Guidelines		Fuel Cycle Cost Current Marketplace	
	PDR600 vs. AP600	PDR600 vs. Conventional PWR	PDR600 vs. AP600	PDR600 vs. Conventional PWR
Stainless Steel Core	10.8	16.1	3.3	7.2
Zircaloy Core	15.0	22.3	4.5	9.9

$$\text{Pu value, \$/gm(f)} = \frac{\text{\$/KgU (uranium fuel cycle) - \$/KgHM (MOX fab cost)}}{\text{(Pu Enrichment (\% fissile)/100) * (1000 gm/Kg)}}$$

Table 4-7

Effect of MOX Fabrication Cost on Pu Value

MOX Fabrication Cost	Pu Value $/gm(f)	
$/KgHM	Stainless	Zircaloy
500	17.6	24.3
750	12.8	17.7
1000	7.9	10.9
1250	3.1	4.3
1500	-1.7	-2.3

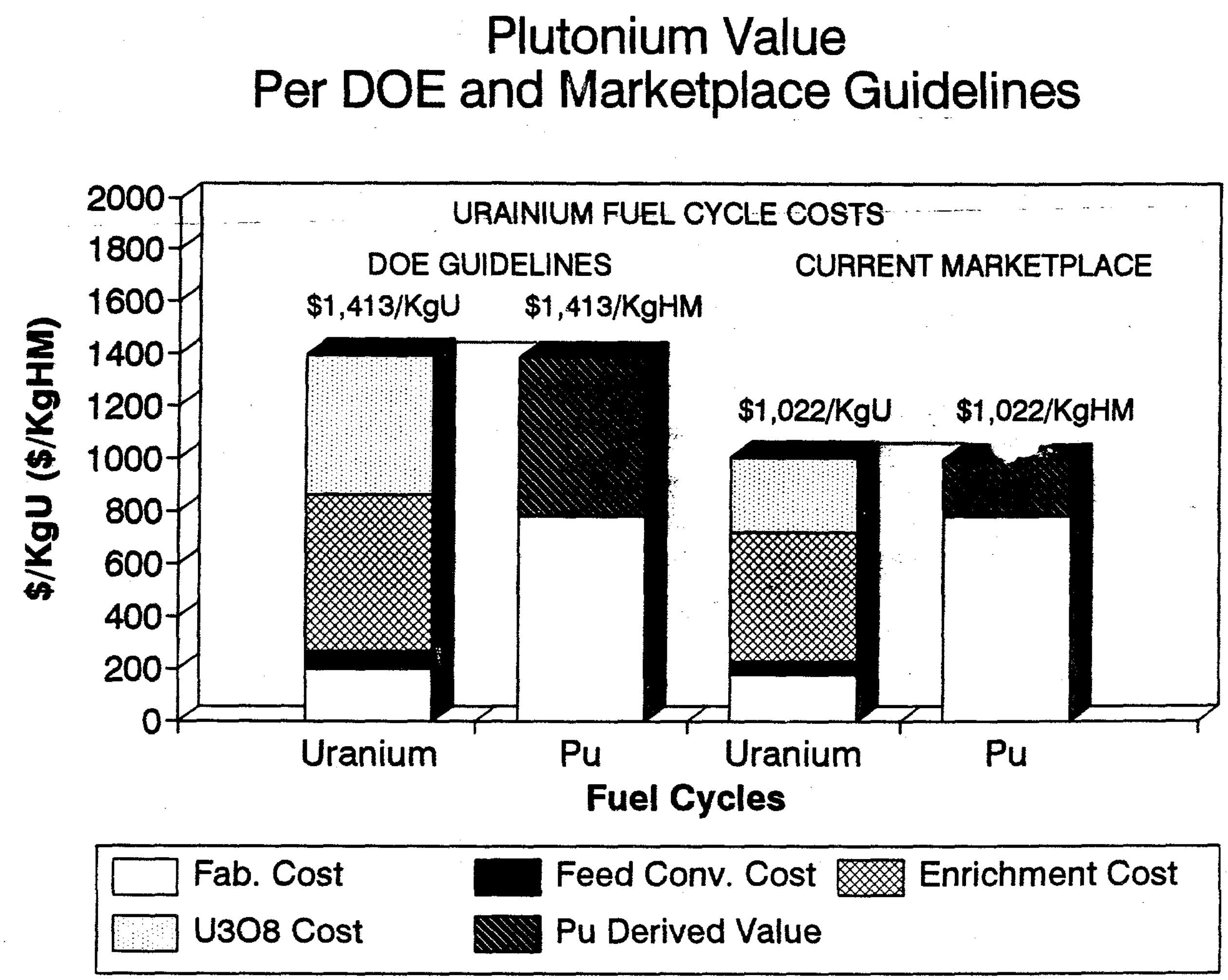

Plutonium Value
Per DOE and Marketplace Guidelines
URAINIUM FUEL CYCLE COSTS
DOE GUIDELINES
CURRENT MARKETPLACE
$1,413/KgU
$1,413/KgHM
$1,022/KgU
$1,022/KgHM
$/KgU ($/KgHM)
2000
1800
1600
1400
1200
1000
800
600
400
200
0
Uranium
Pu
Uranium
Pu
Fuel Cycles
Fab. Cost
Feed Conv. Cost
Enrichment Cost
U3O8 Cost
Pu Derived Value

Figure 4-2

PDR600

APPENDIX A

RESPONSE TO NATIONAL ACADEMY OF SCIENCE QUESTIONS ON PHASE 1A REPORT

DE-AC03-93SF19683

RESPONSES TO DOE QUESTIONS

Question #1:

What is your best judgment of the overnight cost for a greenfield MOX facility in the United States with a capacity of:

a) 50 tHM/yr.
b) 100 tHM/yr.
c) 200 tHM/yr.
d) 400 tHM/yr

Response:

Overnight capital costs for a MOX fuel fabrication facility at different throughputs are given in Table 1. The capital costs of the PDR study (vendor costs) plus the preoperational costs (owner costs) for licensing, permitting G&A, taxes, insurance training and owner project management and engineering integration are included. Total capital costs in 1993 constant dollars are seen to vary from $222 million for a 50 tHM/yr. facility to $774 million for a 400 tHM/yr. facility.

Table 1

MOX Fuel Fabrication Plant
Overnight Capital Cost

	$ Millions, 1992 $		
MOX Fabrication Plant Throughput, tHM/yr.	Vendor(2)	Owner(1)	Total
50	192.9	29.4	222.3
100	292.4	44.6	337.0
200	443.2	67.6	510.8
400	671.8	102.5	774.3

(1) Preoperational costs in PDR Study, include license and permits, G&A, taxes, insurance, training and management and engineering integration.

(2) Estimate includes 15% contingency based on current Westinghouse experience in MOX Fuel Fabrication instead of the 25% contingency used in the DOE evaluation.

Question 2:

What is your best estimate of the annual operating costs of facilities operating at these capacities? (Capacities given in Question 1)

Response:

The annual operating costs for the MOX facility for different throughput rates are given in Table 2. The operating costs include the costs of depleted uranium used to mix with the plutonium. An annual charge for decommissioning and disposing (D&D) of the fabrication facility is also given in Table 2. The capital cost of decommissioning and disposal is assumed to be equal to 10% of the overnight capital cost. Funds for D&D are assumed to accrue in a sinking fund, with an interest rate 2% over the cost of inflation. A D&D charge per KgHM of MOX is calculated for each throughput and added to the annual operating cost to obtain an annual facility operating cost.

Table 2

Annual Operating Costs
MOX Fuel Fabrication Plant

MOX Fabrication Plant Throughput, tHM/yr.	Operating Cost $ Millions	$/KgHM		
		Operating Cost	D&D Cost	Annual Facility Operating Cost
50	36.5	730	31	761
100	60.9	609	24	633
200	105.5	528	18	546
400	189.8	475	14	489

Question 3:

What is your estimate of the cost of MOX fabrication in such a greenfield MOX facility (at each of these capacities), per Kg HM, and which part of that cost is for:

a) Capital cost, including: overnight capital cost (direct plus indirect plus contingency), interest rate, and interest during construction.

b) Operating cost.

Response:

The MOX unit fabrication costs in $/KgHM are given for two types of ownership - DOE at government cost of capital and private industry ownership at a cost of capital reflective of 50% debt/50% equity financing. The unit MOX fabrication costs are given in Table 3 and 4 for the two types of ownership. The financial assumptions used to calculate a levelized fixed charge rate for the capital component are given in Table 5. Constant dollar values were used in this analysis. The construction loan interest rate for private ownership was assumed to be 6.0% (constant $) for calculating the interest during construction; for DOE ownership a 3.5% interest rate was used. The constant dollar interest rate used to calculate a levelized fixed charge rate for determining a unit cost per KgHM was 8.5% for private ownership and 3.5% for DOE ownership.

Table 3

MOX Unit Fabrication Costs
DOE Ownership

	$/KgHM, 1992 $ MOX Fabrication Capacity, tHM/yr.			
Cost Category	50	100	200	400
Direct	296	225	170	129
Indirect[1]	94	71	54	41
Total Base Construction Cost	390	296	224	170
Contingency @ 15%	59	44	34	26
Total Overnight Construction Cost	449	340	258	196
Interest During Construction	59	45	34	26
As-Built Construction Cost, Constant $	508	385	292	222
Operating Cost	761	633	546	489
Total Unit Fabrication Cost	1269	1018	838	711

(1) Indirect costs include preoperational costs of PDR Study.

Table 4

MOX Unit Fabrication Costs
Private Ownership

	$/KgHM, 1992 $ MOX Fabrication Capacity, tHM/yr.			
Cost Category	50	100	200	400
Direct	423	320	243	184
Indirect[1]	134	102	77	59
Total Base Construction Cost	557	422	320	243
Contingency @ 15%	84	63	48	36
Total Overnight Construction Cost	641	485	368	279
Interest During Construction	152	115	87	66
As-Built Construction Cost, Constant $	793	600	455	345
Operating Cost	761	633	546	489
Total Unit Fabrication Cost	1554	1233	1001	834

(1) Indirect costs include preoperational costs of PDR Study.

Table 5

Financial Assumptions for Capital Cost Component of MOX Fabrication Cost

	DOE Ownership		Private Ind. Ownership	
	Nom. $	Const. $	Nom. $	Const. $
Construction Loan, %	7.6	3.5	10.2	6.0
Funding				
Debt Ratio, %	100	100	50	50
Equity Ratio, %	0	0	50	50
Cost of Capital				
Debt, %	7.6	3.5	8.2	4.0
Equity, %	--		17.5	13.0
Blended	7.6	3.5	12.9	8.5
Tax Rate, % (Federal, State, Local)	0	0	36.6	36.6
Levelized Fixed Charge Rate	0.125	0.101	0.192	0.144

For both forms of ownership, economic life of facility equals 15 years, annual capital replacement equals 0.5% of initial capital cost, and property taxes and insurance equal 2% of initial capital cost. These assumptions are factored into the calculation of the levelized fixed charge rate on capital.

Question #4:

What operating experience is available for LWR's with 100 percent MOX cores, and other MOX core fractions above 50 percent?

Response:

In the past, no commercial sized LWR's, specifically PWR's, have been fueled with full core loads of MOX fuel. Reactors were generally charged with 1/3 core loads of MOX fuel of reprocessed plutonium. The Core II of the Saxton reactor has operated with a full MOX fuel load. The reactors were designed to operate in this mode and have sufficient shut down margin to accept core loads with MOX fuel. It is not known that any LWR's were previously charged with partial loads of weapons grade plutonium reprocessed from low burnup fuel.

Question #5:

In your judgment, how feasible would it be (and how long a period of shutdown would be required) to adapt existing U.S. LWR's (either operating or more than half completed) to han[illegible] full MOX core loadings?

Response:

If PWR reactor cores are charged with full core loads of MOX fuel from weapons grade plutonium the control rod worth must be increased. Early PWR's were designed to accept one third of the fuel as a mixture of natural UO_2 and PuO_2 from reprocessed fuel. In this mode of operation, the number of fuel length control rods had to be increased and spare control rod locations were provided. At this time, the condition of these spare location in the heads and upper internals in these existing plants is not known and probably differs from plant to plant. Even with these spare locations in existing plants increased control rod worth and other design changes in operating reactors may not be adaptable for operation with a full MOX fuel core load without major reactor modifications. This needs to be confirmed by analyses which may be unique for each reactor.

In the extreme, modifications of an existing plant to enable operation with full MOX core loads may require a new reactor head and upper internals with additional control rod positions. Exchanging the head and upper internals may also require a penetration in the containment unless a sufficiently large equipment hatch to transfer the components exists. The work to modify the reactor is summarized below:

- MOX core design
- Reactor plant modification

<u>Without head change</u>	<u>With head change</u>
- Control Rod Drives	- Containment penetration
- Control rods	- New reactors head
- Instrumentation	- New upper internal
- Control System	- Instrumentation
	- Control rod drives
	- Control rods
	- Control system

- Reactor operation modifications
- Safety analysis report
- Relicensing

Implementation of these major changes requires approximately four years. The actual reactor shut down time may be limited to 6 to 12 months in a preplanned effort.

APPENDIX B

PINEY POINT PRESENTATION

PDR600

PDR600 PLUTONIUM CONSUMPTION STUDY

PRESENTATION TO
DOE TECHNICAL REVIEW COMMITTEE

May 24, 1993

PINEY POINT, MARYLAND

1:00	DOE OPENING REMARKS	
1:05	INTRODUCTION	TRAVIS
1:10	TECHNOLOGY BASE - REACTOR	
	FUEL CYCLES	GARKISCH
	CORE PHYSICS	RATHBUN
	CORE T&H/TRANSIENT ANALYSES	ROSENTHAL
2:00	TECHNOLOGY BASE - BALANCE OF COMPLEX	
	FEEDSTOCK RECEIPT AND PREPARATION	GARKISCH
	FUEL FABRICATION	GARKISCH
	FUEL DISPOSITION & STORAGE	SCHREIBER
2:30	SAFETY AND ENVIRONMENT	SCHULZ
2:50	SAFEGUARDS & SECURITY	BRINTON
3:05	COST AND SCHEDULE	MOTTLEY
3:25	INFRASTRUCTURE	RITTENBERGER
3:40	SUMMARY OF ALTERNATE STRATEGIES	TRAVIS

0630R/052193-1

PRESENTATION TO
DOE TECHNICAL REVIEW COMMITTEE

(CONTINUED)

3:55	SUMMARY	TRAVIS
4:00	SUBCOMMITTEE BREAKOUT SESSIONS	
	SAFEGUARDS & SECURITY	BRINTON
	SAFETY AND ENVIRONMENT	SCHULZ
	TECHNOLOGY BASE - REACTOR	RATHBUN BISWAS SUDOL TRAVIS SCHREIBER ROSENTHAL BRUCE
	TECHNOLOGY BASE - BALANCE OF COMPLEX	GARKISCH MURRAY BUCKNER
	INFRASTRUCTURE	RITTENBERGER
	COST AND SCHEDULE	MOTTLEY STEVENS
5:15	DINNER	
6:00	ADJOURN	

PDR 600 CORE DESIGN

- USES THE AP600 REACTOR CORE DESIGN AS BASIS
- STANDARD PROVEN CORE AND FUEL ASSEMBLY DESIGN FEATURES
  - 145 assemblies in 12 foot core
  - 69 control rod cluster assemblies
  - Power and flow unchanged from AP600

SPENT FUEL OPTION

- 100 Mt PLUTONIUM IS DISPOSITIONED IN 8 TO 10 REACTORS IN 25 YEARS
- FIRST REACTOR IS DEPLOYED IN YEAR 9
- TWO PER YEAR IN YEARS 10, 11, 12 AND ONE IN YEAR 13
- TARGET BURNUP - 40,000 MWD/T AVERAGE
- 66,000 Kg HM IN CORE AT 5.5% ENRICHMENT
- 0.374 INCH DIAMETER MOX FUEL ROD WITH STAINLESS STEEL CLADDING
- ONCE THROUGH FUEL CYCLE
- 200 MT/YEAR MOX FUEL FABRICATION FACILITY
- DISPOSED AS COMMERCIAL FUEL

ALTERNATE FUEL CYCLE OPTIONS

SPIKING OPTION

- SAME CORE DESIGN AS SPENT FUEL OPTION
- ONE PDR-600 DEPLOYED
- REFUEL MONTHLY
- 13 DAYS REACTOR OPERATION, 17 DAYS REFUELING
- 800 MT/YEAR MOX FUEL FABRICATION FACILITY REQUIRED
- RESTRICTIONS ON DISPOSITION OF FUEL ADDRESSED LATER
- 100 MT PLUTONIUM DISPOSITIONED IN 2.5 YEARS

ALTERNATE FUEL CYCLE OPTIONS (Continued)

PLUTONIUM DESTRUCTION OPTION

- 100 MT PLUTONIUM DISPOSITIONED IN 14 REACTORS IN 25 YEARS
- FIRST REACTOR DEPLOYED IN YEAR 9
- TARGET BURNUP OF 50,000 MWD/T AVERAGE
- 1/3 BATCH FUEL CYCLE OF 18.5 MONTH (AT 75% CAPACITY)
- 49,000 Kg OF HM AT 4% ENRICHMENT
- 0.360 INCH DIAMETER MOX FUEL ROD WITH ZIRLO CLADDING
- ONCE THROUGH FUEL CYCLE IN FIRST 25 YEARS
- 200 MT/YEAR MOX FUEL FABRICATION FACILITY REQUIRED
- REPROCESSED FUEL MAY BE USED IN YEAR 26

FUEL DESIGN AND PERFORMANCE

PDR 600 FUEL DESIGN

- PDR 600 USES THE STANDARD WESTINGHOUSE VANTAGE 5 FUEL ASSEMBLY DESIGN
- THE ASSEMBLY IS CURRENTLY SUPPLIED FOR REFUELING OF PWR'S
- THE PDR 600 FUEL ROD DESIGN IS QUALIFIED FOR HIGH BURNUP (>60,000 MWD/T)

FUEL DESIGN AND PERFORMANCE (Continued)

SPENT FUEL & SPIKING OPTIONS

- ZIRCALOY IS REPLACED WITH STAINLESS STEEL TO INCREASE THE FUEL ENRICHMENT
- STAINLESS STEEL OR INCONEL GRIDS AND STRUCTURES
- MOX FUEL AND STAINLESS STEEL CLADDING ARE QUALIFIED TO BURNUPS EXCEEDING 50,000 MDW/T
- SUFFICIENT DATA ARE AVAILABLE TO SUPPORT DESIGN AND A LICENSING APPLICATION
- FOR THE SPIKING OPTION, FISSION GAS PLENUM VOLUME CAN BE REDUCED
- SIMPLIFICATIONS OF THE FUEL DESIGN FOR THE SPIKING OPTION ARE CONSIDERED TO REDUCE COST

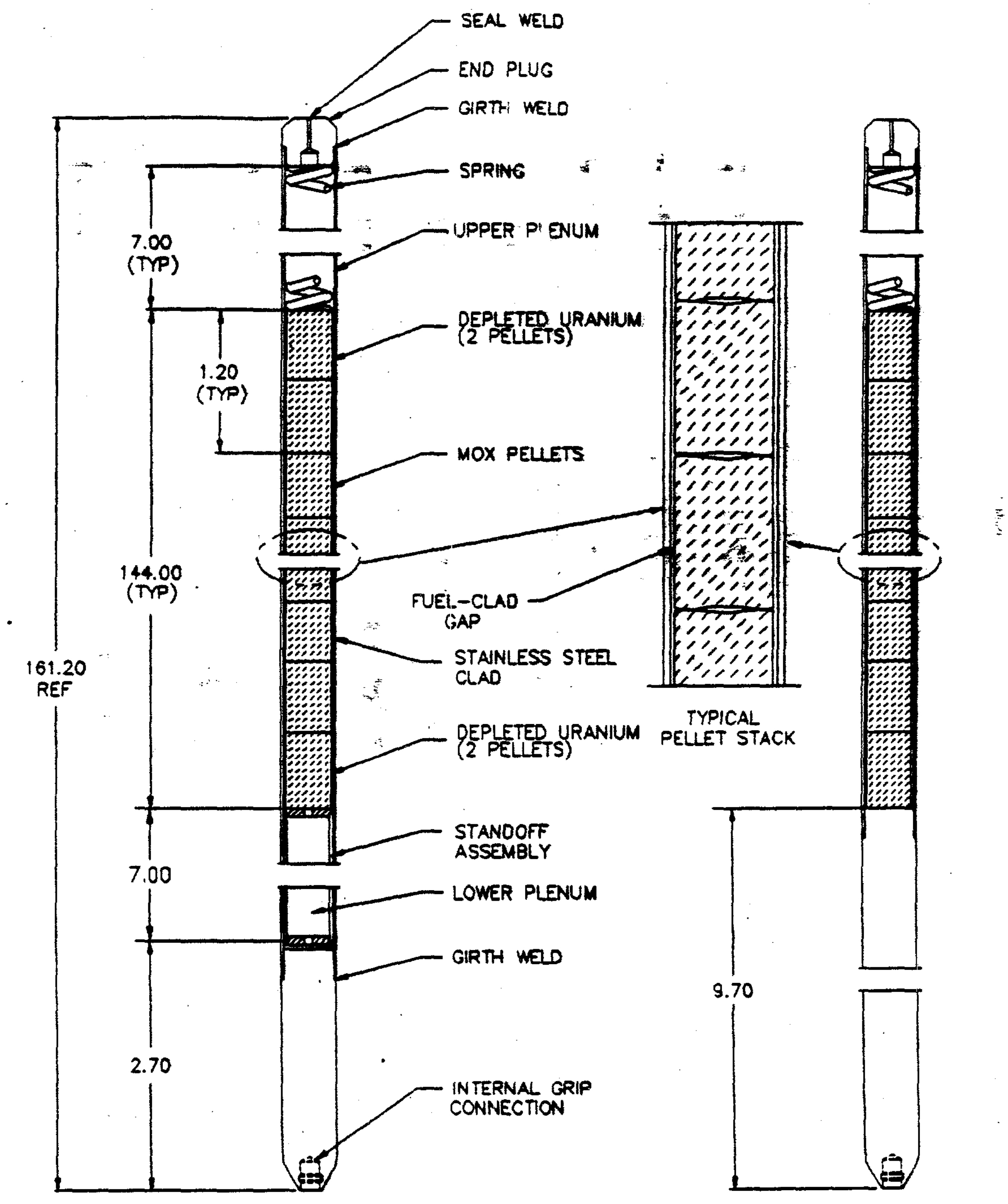
FUEL DESIGN AND PERFORMANCE (Continued)

PLUTONIUM DESTRUCTION OPTION

- STANDARD W - VANTAGE 5 (OFA) FUEL ASSEMBLY DESIGN WITH ZIRLO
- FUEL ROD DIAMETER 0.360 INCH NOT OPTIMIZED FOR Pu DESTRUCTION
- ANNULAR PELLETS WITH A HOLE OF 20% OF PELLET CROSS-SECTION AREA
- ASSEMBLY QUALIFIED TO BURNUPS > 50,000 MWD/T

PDR600

PDR600 FUEL ROD DESIGN



SPENT FUEL OPTION
PU DESTRUCTION OPTION

SPIKING OPTION

PDR600

PLUTONIUM DISPOSTION REACTOR

PDR600 Core Neutronics Performance

Piney Point Evaluation Meeting

May 24, 1993

Debdas Biswas

Mel Buckner

Roy Rathbun

Westinghouse Savannah River Company

PDR600

EVALUATIONS PERFORMED

- Assembly Dosage
- Pu Transformation Rate
- Single Assembly Criticality
- Core Loading Patterns
- Critical Boron Concentrations
- Radial and Axial Power Distributions
- Temperature Feedbacks
- Control Rod Worths and Requirements

PDR600 ASSEMBLY DOSAGE REQUIREMENT MET WITH VERY SHORT EXPOSURE

- Assembly must have Gamma Levels >100 rem/hr at 3 feet after 2 Year Cooldown

- Applies to Spiking and Spent Fuel Options

- Spiking Condition Used
  - 100 Hour Exposure in PDR600
  - Low Power Position
  - 2 Year Cooldown

- Resulting Dose (from assembly center):
  - 2450 rem/hr at 3 feet
  - 570 rem/hr at 6 feet

Plutonium Dispositioned per Reactor

Spent Fuel Option

Reactor Designation	R1	R2	R3	R4	R5	R6	R7	R8
Reactor Operation Years During 25 Year Period	16	15.5	15	14.5	14	13.5	13	12.5
Operating Cycles Initiated in 25 Year Period	10	10	9	9	9	8	8	8
Plutonium Disposition per Reactor (MT)	14	14	13	13	13	12	12	12
Accumulated Disposition (MT)	14	28	41	54	67	79	91	103

Equivalent Parameters for all Reactors:

- Separate Batches per Core = 3
- New Assemblies Loaded per Cycle = 48 (approx.)
- Feed Pu Enrichment (% Heavy Metal) = 5.5
- Period for Successive Cycle Startup (months) = 20.3
- Pu Charged to Initial Core (kg) = 3345
- Pu/Assembly (kg) @ Feed Enrichment = 25.4
- New Pu loaded per Equilibrium Cycle (kg) = 1227
- Average Discharge Burnup (MWD/MTM) = 40,000

PDR600

Plutonium and Uranium Charge and Discharge Inventories

Assembly Average Values for Spent Fuel Option

Category	Isotope	Charged (kg)	Discharged (kg) (40,000 MWD/MTM)	Discharged Pu Isotopic Fractions
Plutonium:				
	Pu238	0.01	0.1	0.006
	Pu239	23.75	10.8	0.60
	Pu240	1.5	4.4	0.24
	Pu241	0.10	2.4	0.13
	Pu242	0.013	0.4	0.02
Uranium:				
	U235	0.8	0.5	
	U236	0	0.1	
	U238	435	424	
Pu Enrichment (%)		5.5	4.1	
Fissile Pu (%)		5.2	3.0	

PDR600

Plutonium Dispositioned by Spiking Option Reactor

(Single Reactor)

Fraction of Core Charged Each Cycle	100%
Pu Charged to Core (kg)	3345
100% Power Operation Period (days)	13
Period for Successive Cycle Startup (months)	1
Assembly Average Discharge Burnup (MWD/MTM)	376
Total Operation Time Required to Disposition 100 MT of WG Plutonium (yr)	2.5

Plutonium Dispositioned per Reactor
Pu Destruction Option

Reactor Designation	R1	R2	R3	R4	R5	R6	R7	R8	R9	R10	R11	R12	R13	R14	R15
Years of Reactor Operation During 25 Year Period	16	15.5	15	14.5	14	13.5	13	12.5	12	11.5	11	10.5	10	9.5	9
Number of Operating Cycles Initiated in 25 Year Period	11	11	10	10	9	9	9	9	8	8	8	7	7	7	6
Plutonium Disposition per Reactor (MT)	8.2	8.2	7.5	7.5	6.9	6.9	6.9	6.9	6.2	6.2	6.2	5.6	5.6	5.6	4.9
Accumulated Disposition (MT)	8.2	16	24	31	38	45	52	59	65	72	78	83	89	95	100

Equivalent Parameters for all Reactors:

- Separate Batches per Core = 3
- New Assemblies Loaded per Cycle = 48 (approx.)
- Feed Pu Enrichment (% Heavy Metal) = 4.0
- Period for Successive Cycle Startup (months) = 18.5
- Pu Charged to Initial Core (kg) = 1666
- Pu/Assembly (kg) @ Feed Enrichment = 13.5
- New Pu loaded per Equilibrium Cycle (kg) = 653
- Assembly Average Discharge Burnup (MWD/MTM) = 50,000

PDR600

Plutonium and Uranium Charge and Discharge Inventories

Assembly Average Values for Pu Destruction Option

Category	Isotope	Charged (kg)	Discharged (kg) (50,000 MWD/MTM)	Discharged Pu Isotopic Fractions
Plutonium:				
	Pu238	0.01	0.06	0.001
	Pu239	12.65	2.39	0.36
	Pu240	0.8	2.35	0.35
	Pu241	0.05	1.19	0.18
	Pu242	0.01	0.71	0.11
Uranium:				
	U235	0.6	0.2	
	U236	0	0.1	
	U238	324	314	
Pu Enrichment (%)		4.00	2.09	
Fissile Pu (%)		3.76	1.11	

PU-239 DESTRUCTION

- Spent Fuel Option
  55 %

- Spiking Option
  Negligible

- Pu Destruction Option
  81%

CORE DESCRIPTION (Spent Fuel)

Parameter	PDR600	AP600
◆ Power, MWth	1933	1933
◆ No. of Assemblies	145	145
◆ Assembly Type	17x17	17x17
◆ Fuel Rod Dia (inch)	0.374	0.374
◆ Fuel Rod Cladding	SS-304	Zirc-4
◆ Enrichment, w/o	4.5/5.0/5.5(Pu)	2.0/2.5/3.0(U)
◆ Burnable Absorber		
– Discrete Pyrex BA	1216	1424 (WABA)
– IFBA, 3 mg/inch of B10	All Rods	None
◆ Control Rods (Ag-In-Cd)	69	61
◆ Cycle length, MWD/MTM	13,300	16,600

PDR600

CORE CHARACTERISTICS (Spent Fuel)

SUMMARY

Parameter	PDR600	AP600
◆ PPM, HFP, BOL/EOL	3056/944	1020/10
◆ MTC, BOL, HFP/HZP, pcm/°F	-10.0/-0.1	-9.3/-2.0
◆ Doppler Power Coef, pcm/%P	-20.0 to -9.0	-10.7 to -7.0
◆ $F_{\Delta H}$, HFP, ARO	1.44 to 1.33	1.39 to 1.22
◆ Boron Worth, pcm/ppm	-2.9 to -3.6	-8.8 to -12.5
◆ Total Rod Worth, %$\Delta\rho$	10.66 to 11.29	10.99 to 10.98
◆ Shutdown Margin, %$\Delta\rho$	4.97 to 5.67	4.16 to 3.59

PDR600

CORE DESCRIPTION (Pu Destruction Option)

Parameter	Spent Fuel	Pu Destruction
◆ Fuel Rod Dia (inch)	0.374	0.360
◆ Pellet Type	Solid	Annular
◆ Fuel Rod Cladding	SS-304	Zirc-4
◆ Enrichment, w/o	4.5/5.0/5.5	2.5/3.0/4.0
◆ Burnable Absorber		
– Discrete Pyrex BA	1216	1216
– IFBA, all rods	3 mg/inch	1.0 mg/inch
◆ Cycle Length, MWD/MTM	13,300	16,600
◆ Discharge Burnup, MWD/MTM	40,000	50,000

CORE CHARACTERISTICS (Pu Destruction)

SUMMARY

Parameter	Spent Fuel	Pu Destruction
◆ PPM, HFP, BOL/EOL	3056/944	2593/249
◆ MTC, BOL, HFP/HZP, pcm/°F	-10.0/-0.1	-3.3/+5.3
◆ Doppler Power Coef, pcm/%P	-20.0 to -9.0	-19.0 to -9.2
◆ $F_{\Delta H}$, HFP, ARO	1.44 to 1.33	1.49 to 1.30
◆ Boron Worth, pcm/ppm	-2.9 to -3.6	-5.4 to -8.6
◆ Total Rod Worth, %$\Delta\rho$	10.66 to 11.29	12.96 to 15.91
◆ Shutdown Margin, %$\Delta\rho$	4.97 to 5.67	6.35 to 8.07

PDR600

Core Neutronics Conclusions

- **Spent Fuel Option**
  - Standard Materials Employed
  - Control Worths, Temperature Coefficients and Peaking Factors are within Realm of Normal Experience
  - Fuel Charged to 8 Reactors in 25 Years Disposes 100 MT
  - End Product Similar to Commercial Spent Fuel Material
- **Spiking Option**
  - 100 Hours of Operation Spikes Fuel to Requirements
  - Identical to Spent Fuel Option at Start of Life
  - Single Reactor Spikes 100 MT in 2.5 Years
- **Pu Destruction Option**
  - Water-to-Metal Ratio Increased to Enhance Destruction
  - Low Enrichments and High Burnups Employed
  - Over 70% of Fissile Charge is Destroyed
  - Fuel Charged to 15 Reactors in 25 Years Disposes 100 MT
  - 90% Fissile Pu Destruction with Ternary Fuel in 12 Reactors

PDR600

CORE THERMAL & HYDRAULIC ANALYSIS

- **Standard Westinghouse methods used:**
  - **Ensure adequate heat transfer from fuel to coolant**
  - **Prevent departure from nucleate boiling (DNB)**

- **DNB design basis:**
  - **At least 95% probability that DNB will not occur on limiting fuel rod for Condition I & II events at a 95% confidence level**
  - **Requirement for Condition I (operational) and Condition II (faults of moderate frequency)**
  - **DNB design basis met: DNB ratio (DNBR) $\geq$ safety analysis limit**

CORE THERMAL & HYDRAULIC ANALYSIS

- **Core limits used to set overtemperature & overpower ΔT reactor trips - defining DNB related criteria:**
  - **Minimum DNBR $\geq$ safety analysis limit value**
  - **Hot channel exit quality $\leq$ upper limit of quality range for applicable DNB correlation**
  - **$T_{HOT} < T_{SAT}$: Measured loop ΔT (THOT - TCOLD) is proportional to power**

- **<u>General Conclusion</u>:**

  PDR600 T & H design parameters match or approximate those for AP600 - DNB design basis will be met for the PDR600 - ensuring acceptable T&H performance

PDR600

LOSS-OF-COOLANT ACCIDENTS

- Complete LOCA break size spectrum in the AP600 SSAR
- For PDR600: Same computer codes & general methods as for AP600 SSAR
- AP600 small break LOCA analyses (6-inch equivalent diameter and under):
  - No core uncovery with conservative evaluation model assumptions
  - Only passive safety systems credited
  - Change from AP600 fuel minimally affect results - PDR600 should be acceptable

LOSS-OF-COOLANT ACCIDENTS

- **Large break LOCAs for AP600:**
  - **Significant margin to the PWR licensing limits**
  - **State-of-the-art, best estimate computer code (Currently being licensed with USNRC as a best estimate LOCA methodology per 1988 Appendix K rule change)**
  - **Acceptability should apply to PDR600**

- **General Conclusion:**

  PDR600 large and small break LOCA analyses should meet applicable acceptance criteria; similarities between PDR600 & AP600 plus AP600 results/margins

NON-LOCA ACCIDENT ANALYSIS

- **Selected non-LOCA events potentially affected - key parameter sensitivities:**
  - **Rod ejection:** Reduced delayed neutron fraction (β_{eff})
More negative Doppler
Power peaking
Ejected rod worth
Clad material
  - **Complete loss-of-flow (CLOFA):** Increased $F_{\Delta H}$
Greater rod drop time
  - **Anticipated transient: without SCRAM** More negative Doppler
Different moderator temperature coeff.

(<u>Note</u>: ATWS not a design transient; DAS provides backup trip/protection)

NON-LOCA ACCIDENT ANALYSIS

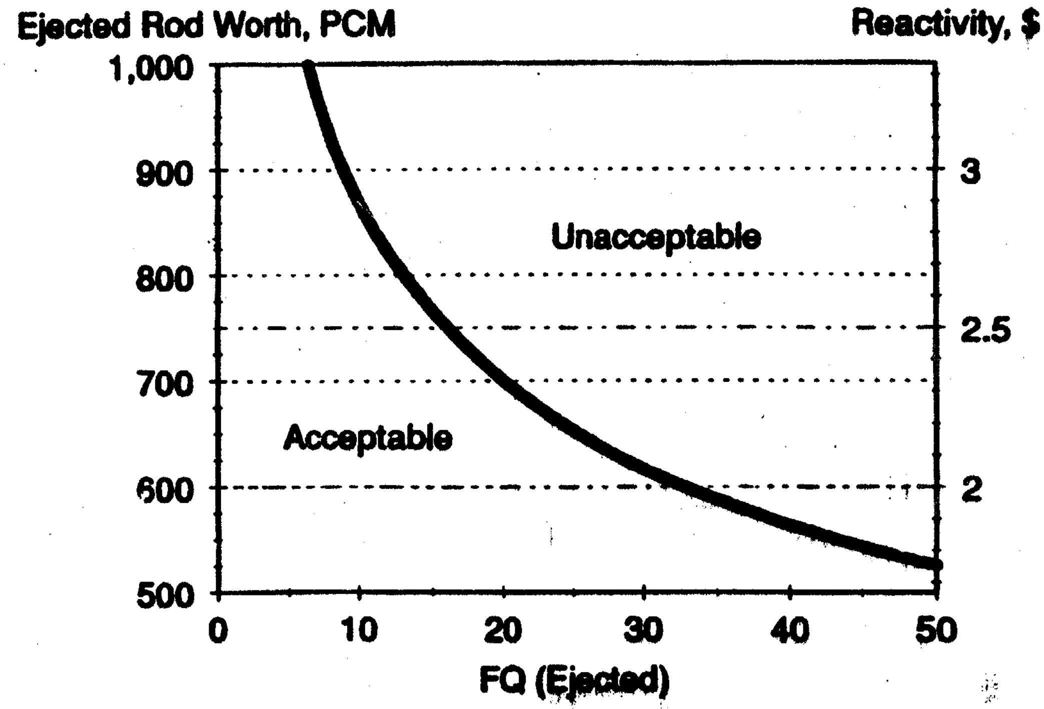
- **Results:**
  - Rod ejection: Low ejected rod worths
    Low peaking factors
    Very favorable results
  - CLOFA: Substantial $F_{\Delta H}$ margin
    Margin for increased rod drop time
  - ATWS: Response typical of Westinghouse PWRs
    Complies with ATWS rule for Westinghouse plants

- **General Conclusion:**

  All three PDR600 core options should yield acceptable results for non-LOCA transients

PDR600

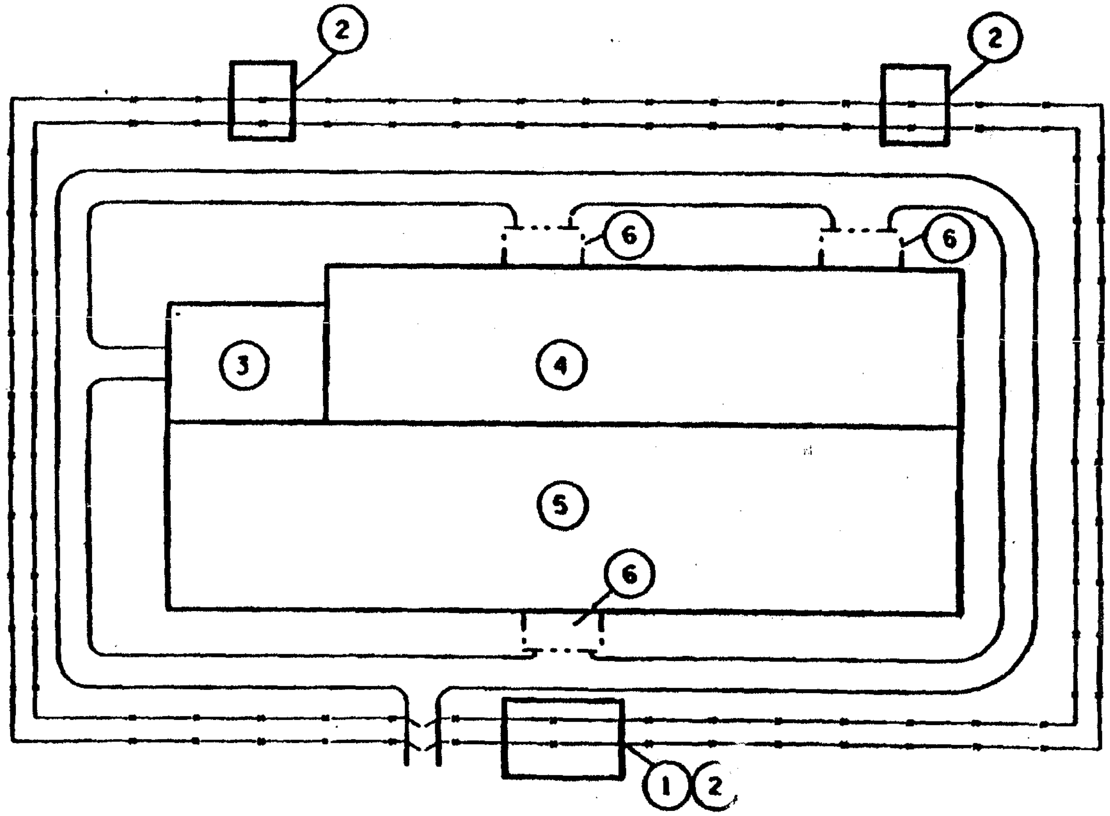
RCCA EJECTION EOL HOT ZERO POWER



Predicted for PDR600
Spent Fuel Case

PDR600

FUEL AND TARGET COMPLEX



LEGEND

1. ENTRY CONTROL FACILITY
2. TRANSHIP WAREHOUSE
3. PLUTONIUM STORAGE FACILITY
4. TRITIUM PROCESS FACILITY
5. FUEL FAB. FACILITY
6. TRUCK PARKING AREA

PLUTONIUM TRANSHIP WAREHOUSE FUNCTIONS

- ACCOUNTABILITY IN:
  - RECEIVING
  - UNPACKAGING
  - ANALYSIS
  - STORAGE
  - TRANSFER TO FUEL FABRICATION

FEEDSTOCK RECEIPT AND PREPARATION

- RECEIVE WEAPONS GRADE PLUTONIUM AS METAL
- DoE FACILITY RECASTS WEAPON PITS INTO LOGS OR GRANULES (POTENTIALLY DILUTED WITH ^{238}U)
- PLUTONIUM METAL DISSOLVED TO PRODUCE MASTER BLEND
- WET PROCESS ALSO NEEDED FOR PROCESSING SCRAP RETURNS
- REMOVE AMERICIUM TO <0.2 a/o

FUEL FABRICATION FACILITY

- SIZED FOR 200 MT/Y CAPACITY
- 100,000 TO 130,000 FT^2
- CAPITAL COST OF $450 MILLION FOR DEVELOPED TECHNOLOGY
- ONLY POWDER/PELLET/ROD FABRICATION
- ASSEMBLY SKELETONS AND COMPONENTS BOUGHT
- MOX FABRICATION COST OF $530/Kg OF HM WITHOUT FACILITY COST
- ASSUMES GREENFIELD FACILITY

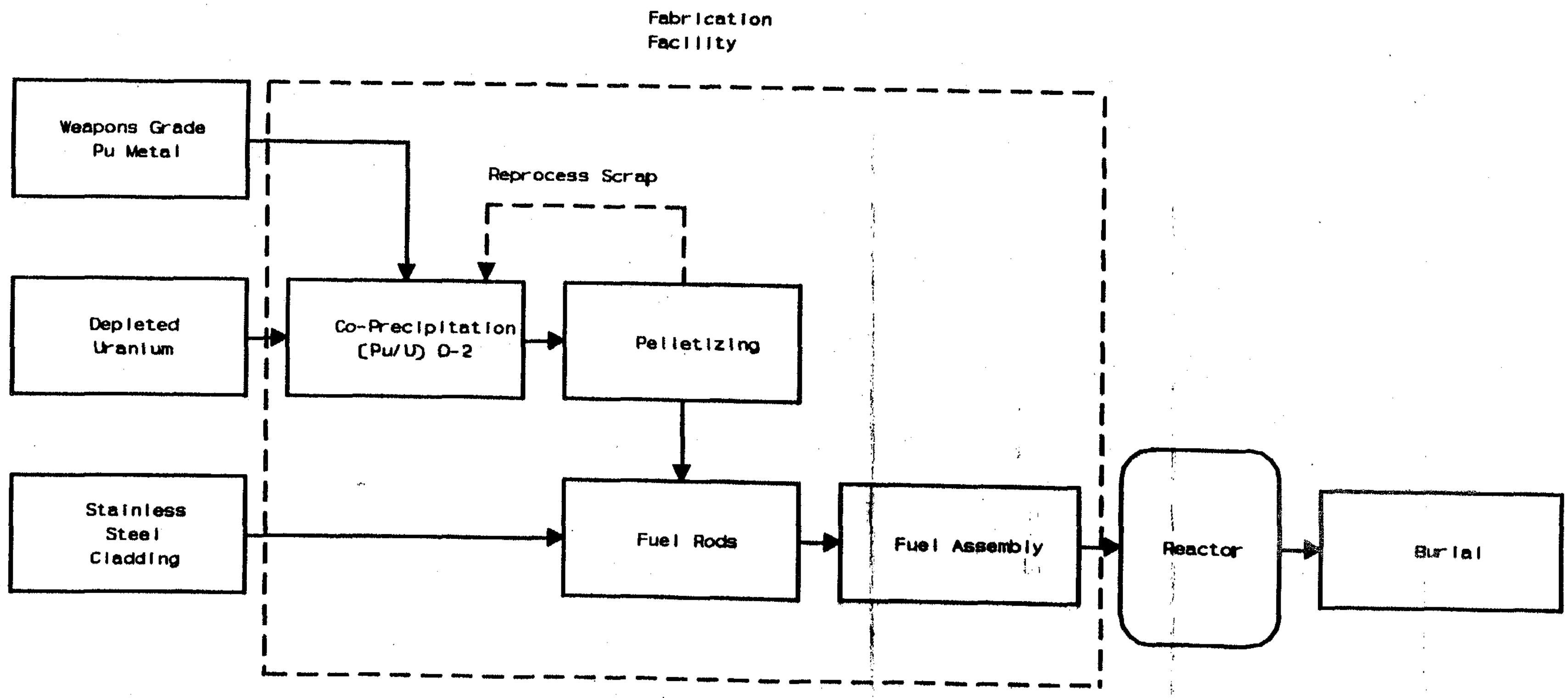
EXISTING SITES EVALUATED:

- SAVANNAH RIVER SITE
- BARNWELL, SC
- FMEF AT HANFORD

FUEL FABRICATION FACILITY MUST BE ENGINEERED

- FUEL FABRICATION FACILITY SIZE AND EQUIPMENT SCOPED
- GENERIC FABRICATION PROCESSES FOR POWDER, PELLET, ROD, ASSEMBLY
- MAJOR DESIGN DECISIONS:
  - Design for weapons grade plutonium or also reprocessed plutonium
  - Facility design life
  - Design responsibility (DoE Nat'l Lab or commercial contractor)
- ISSUES WHERE DESIGN WORK MUST BE FOCUSED:
  - Confinement and shielding
  - Reliable automation
  - Dust control
  - Automated Inspection
  - Integration of safeguards and security

GENERIC MOX FUEL FABRICATION PROCESSES



TECHNOLOGY FOR MOX FUEL FABRICATION DEVELOPED

- W-REPROCESSED FUEL PLANT (RFP) IN ANDERSON, SC
- W-CNFD FUEL FABRICATION PLANS (PROPRIETARY)
- SRP MOX FABRICATION STUDY
- BNFL MOX PROCESS (PROPRIETARY)
- FFTF - CRBRP MOX FUEL SPECIFICATIONS
- SAFF LINE EXPERIENCE
- W-HANFORD COMPANY STUDY ON FABRICATION IN FMEF
- FOREIGN MOX PLANT DESIGNS
- B&R PLANT DESIGN EXPERIENCE

FUEL TRANSPORTATION, STORAGE & DISPOSAL

- NEW FUEL TRANSPORTATION AND STORAGE
- SPENT FUEL WET STORAGE
- DRY STORAGE, TRANSPORTATION AND DISPOSAL

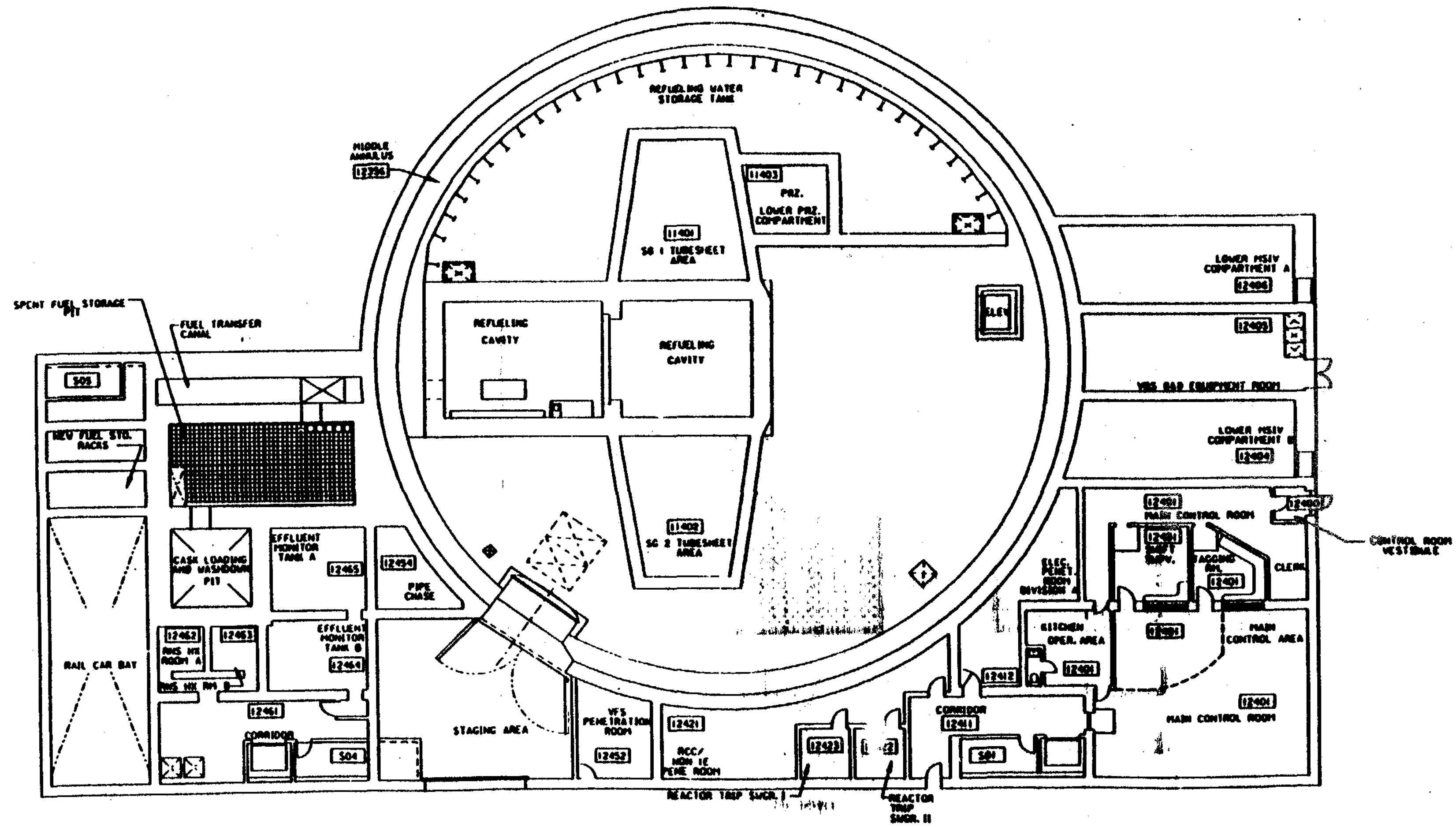
FUEL STORAGE

	AP600	SPENT FUEL	MAX. DESTR.	SPIKING
NEW FUEL STORAGE PROVIDED/REQUIRED	1/3 CORE	1/3 CORE	1/3 CORE	1 CORE
RACK SPACING	10.9	10.9	10.9	10.9
SFP CAPACITY PROVIDED/REQUIRED (# ASSEMBLIES)	619	531	531	1650 TO 4350

PDR600

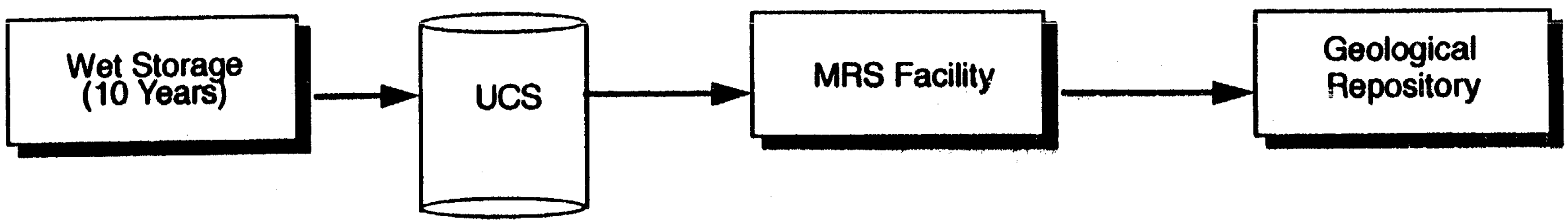
Nuclear Island General Arrangement

Plan at El. 117' - 6"



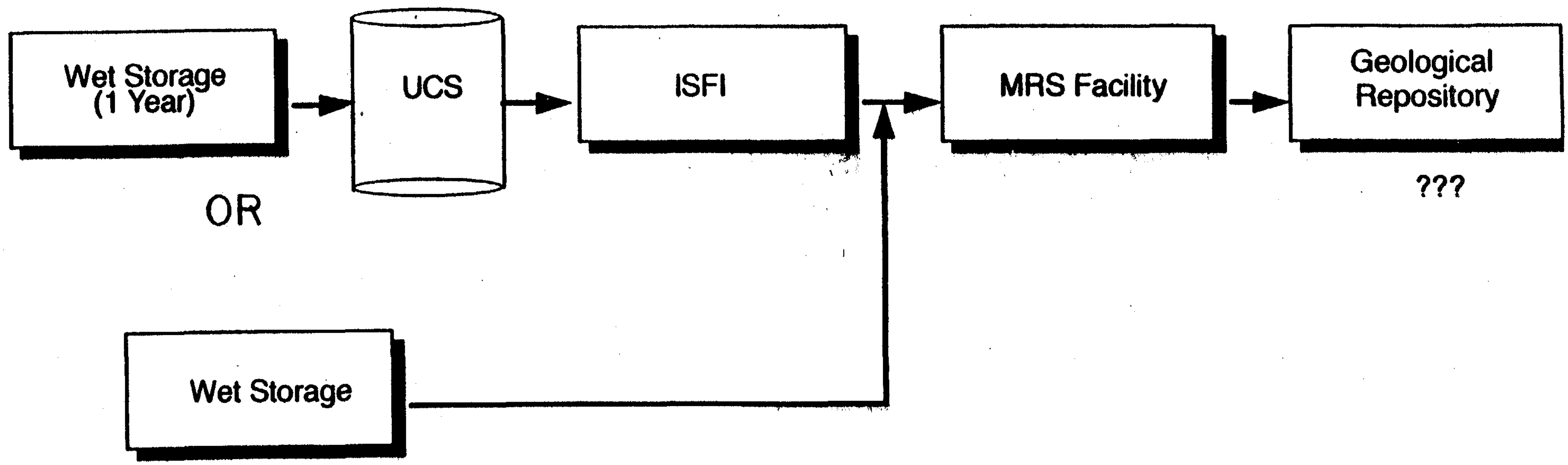
Dry Storage, Transport & Disposal

Spent Fuel & Maximum Destruction Options:



Dry Storage, Transport & Disposal

Spiking Option:



CONCLUSIONS

- NO MAJOR ISSUES EXIST FOR SPENT FUEL OR MAXIMUM DESTRUCTION OPTIONS
- FOR SPIKING OPTION:
  - ADDITIONAL ON-SITE STORAGE REQUIRED
  - POTENTIAL ISSUE WITH ULTIMATE GEOLOGICAL STORAGE

SAFETY AND ENVIRONMENTAL

- PDR CHANGES IN AP600
  - MOX fuel with $\leq$7.0% Enrichment
  - SS fuel cladding
  - More control rods
  - Higher soluble boron concentration
- EVALUATION OF PDR600 SHOW SAFETY/ENVIRONMENTAL PREFERENCE SIMILAR TO AP600
  - Transients
  - LOCA
  - PRA
  - Waste production

PDR600 SAFETY APPROACH

- A STABLE, FORGIVING PDR600 PLANT DESIGN THAT ACCEPTS MISTREATMENT OR ANOMALIES AND REMAINS IN NORMAL OPERATION
- PROTECTION AGAINST POTENTIAL RADIATION RELEASES TO THE PUBLIC THROUGH THE VARIOUS PHYSICAL PLANT BOUNDARIES
- SAFETY-RELATED SYSTEMS FOR MITIGATION FUNCTIONS
- DIVERSE MITIGATION FUNCTIONS WITHIN THE SAFETY-RELATED SYSTEMS
- ADDITIONAL MARGIN PROVIDED BY NON-SAFETY SYSTEMS, AND
- FEATURES TO MITIGATE CORE DAMAGE CONSEQUENCES IN THE UNLIKELY EVENT THAT ALL THE OTHER FEATURES DO NOT WORK

THESE DEFENSE-IN-DEPTH FEATURES PROVIDE SIGNIFICANTLY ENHANCED PROTECTION AGAINST CORE DAMAGE.

PLANT COMPONENT REDUCTION

REDUCTION OF COMPONENTS TO BE MAINTAINED IN PDR600

Plant Features	Conventional 2-Loop Plant	Reference Commercial Plant	% Reduction
Pumps • Active Safety • Non-nuclear Safety	 25 188	 None 131	 Eliminated 30
Valves • > 2 in. • ≤ 2 in.	 2553 8820	 1528 3678	 40 58
Heat Exchangers	99	71	28
Snubbers	> 1000	~ 20	98
Cable (power, I&C, lighting, Communication, and grounding cable included)	3,623,715 lin ft	~ 1,000,000 lin ft	70
Safety-Grade Piping • Large Bore • Small Bore	 44,300 ft 18,614 ft	 7,210 ft 4,110 ft	 84 78
Piping System Welds • NSSS • RCS	 ~ 2,000 32	 ~ 1,000 15	 53 50

TRANSIENT AND ACCIDENT MARGINS

TRANSIENT AND ACCIDENT MARGINS

Event/Analysis	Limiting Parameter	NRC Limit	Reference Commercial Plant	Conventional 2-Loop Plant
Probabilistic Risk Assessment	• Core Damage Freq[1] • Signif. Release Freq	1×10^{-4}/yr 1×10^{-6}/yr	4.2×10^{-7}/yr $< 3 \times 10^{-8}$/yr	$\sim 5 \times 10^{-5}$/yr $\sim 5 \times 10^{-6}$/yr
Large LOCA	• Peak Clad Temp • Peak Containment Pressure • Containment Pressure @ 24 hrs	2200°F ≤90% of Design ≤50% of Design	1800°F 90% of Design 25% of Design	~2200°F 90% of Design 30% of Design
Small LOCA	• Peak Clad Temp • Reactor Vessel Water Level	2200°F	~1000°F No Core Uncovery for Breaks <6" dia.	1300°F Core Uncovery Occurs
Steamline Break	• Peak Containment Pressure	<100% of Design DNBR Margin	92% of Design Margin Provided	99% of Design Margin Provided
Anticipated Transient without Trip	• RCS Peak Pressure • Core Power (steady state)	3200 psig	[illegible]00 psig 0% Power w/o Rods	2900 psig ~10% Power w/o Rods
Complete Loss of RCP Flow	• Departure from Nucleate Boiling Ratio Margin	DNBR ≥1.30	≥1.6	≥1.38

1. Includes internal events at both power and shutdown conditions.

PDR600

EVENT CORE DAMAGE FREQUENCIES FOR PDR600

EVENT CORE DAMAGE FREQUENCIES, (PER YEAR) FOR PDR600: DEFENSE IN DEPTH ENHANCES SAFETY			
EVENTS AT POWER*	REF. COMMERCIAL PLANT	CURRENT PLANTS	REDUCTION
Transients	7.1E-8	1.3E-5	180
Blackout	2.9E-9	6.6E-6	230
SG tube rupture	2.6E-9	1.7E-6	650
LOCA-Small	7.7E-8	8.0E-6	100
LOCA-Medium	8.8E-8	5.0E-6	60
LOCA-Large	1.6E-8	8.0E-7	50
ATWT	4.5E-8	2.2E-6	50
Loss-of-Cooling	1.6E-9	1.1E-5	6800
Interfacing LOCA	<E-9	1.0E-6	1000
Vessel rupture	3.0E-8	3.0E-7	10
Total	3.3E-7/yr	5.0E-5/yr	150
Without non-safety systems(1)	2.6E-6/yr	~1E-3yr	380

(1) Without non-safety pumps, valves and AC power.

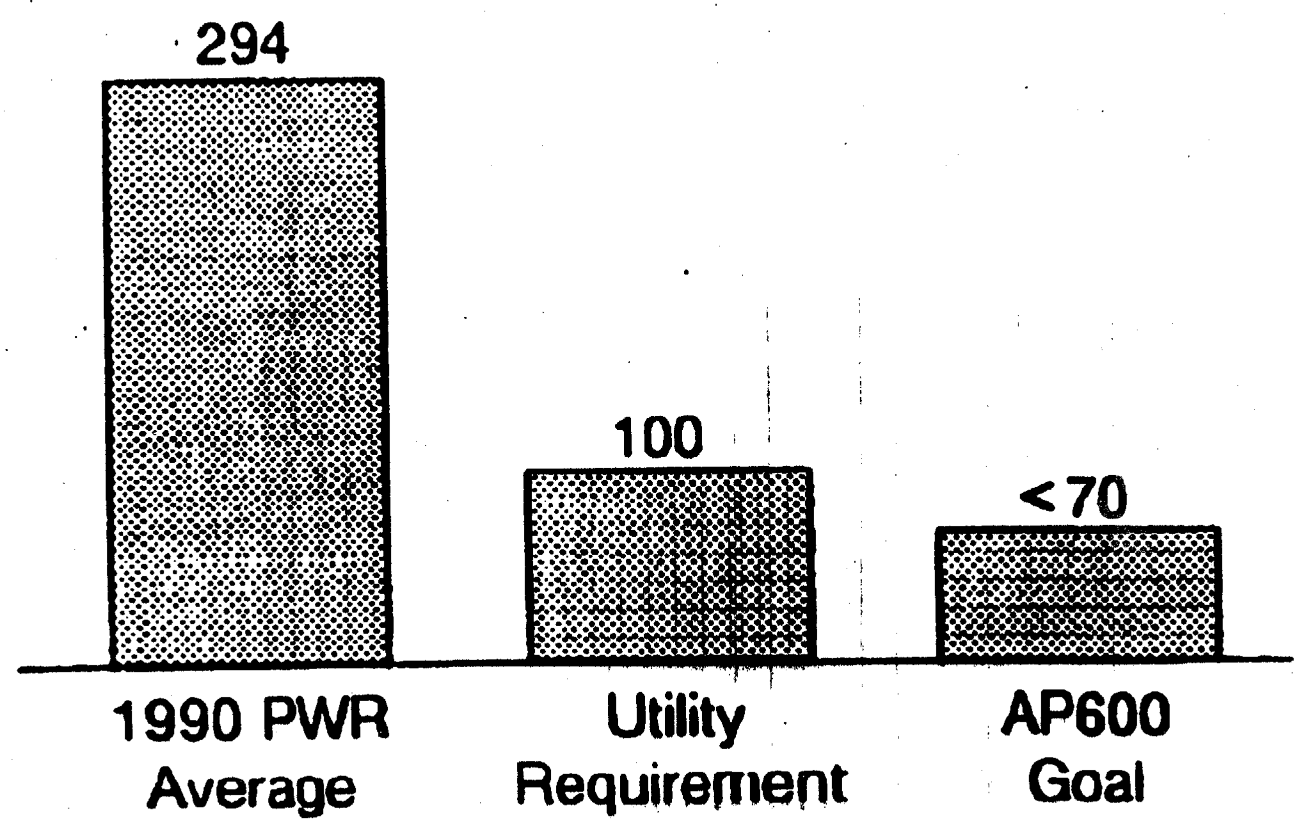
WASTE VOLUMES FOR PDR600

WASTE VOLUMES FOR PDR600		
	ALWR GOAL	REFERENCE COMMERCIAL PLANT ESTIMATE
Gaseous effluents, Ci/yr	200	$\leq$ 200[1]
Liquid effluents, Ci/yr	0.05	$\leq$ 0.05[1]
Solid wastes, Ft^3/yr	1750	1729[2] [3]

1. Total radioactivity, excluding tritium, will be equal to or lower than the 1984-85 10% best plants in the U.S.
2. Low-level dry and wet waste will be equal to or lower than the 10% best plants in the U.S.
3. Current industry average is 7050 ft^3.

PDR600

OCCUPATIONAL RADIATION EXPOSURE ESTIMATE (MAN-REM PER YEAR)



SAFEGUARDS, SECURITY, AND EMERGENCY PREPAREDNESS

PRE-CONCEPTUAL FEATURES

- PERVASIVE MATERIAL CONTROL AND ACCOUNTABILITY (MCA)
  - ITEM CONTROL
  - REAL TIME SURVEILLANCE
- GRADED PHYSICAL SECURITY
  - ZONED ACCESS
  - VEHICLE EXCLUSION
- ECONOMICAL PERSONNEL ACCOUNTABILITY
  - SECURITY ACCESS CONTROL
  - SECURITY ZONES

SAFEGUARDS, SECURITY, AND EMERGENCY PREPAREDNESS (Cont'd)

PERVASIVE MATERIAL CONTROL AND ACCOUNTABILITY

- CONFIRMATION MEASUREMENTS
- ITEM CONTROL
  - COMPUTERIZED INVENTORY
  - LABELLED AND MACHINE READ
  - REAL TIME STORAGE POSITION SENSING
  - TWO PERSON RULE
- ATTRACTIVE PLUTONIUM FORMS RETAINED WITHIN ONE MATERIAL ACCESS AREA
  - RECEIPT POINT
  - STORAGE VAULT
  - FUEL FABRICATION

SAFEGUARDS, SECURITY, AND EMERGENCY PREPAREDNESS (Cont'd)

GRADED PHYSICAL SECURITY

- HIGH, VITAL, AND INDUSTRIAL SECURITY ZONES
  - COST EFFECTIVE
  - FOCUS ON LEVEL OF NEED AND UNITY OF PROTECTION
  - GRADED INTRUSION DETECTION
  - LIMITS INSIDER MOBILITY
- TRANSHIP CONCEPT
  - EXCLUDES VEHICLE CROSSING OF HIGH AND VITAL SECURITY ZONE BOUNDARIES
  - SIGNIFICANT INCREASE IN PERFORMANCE
  - COMBINE WITH WAREHOUSING

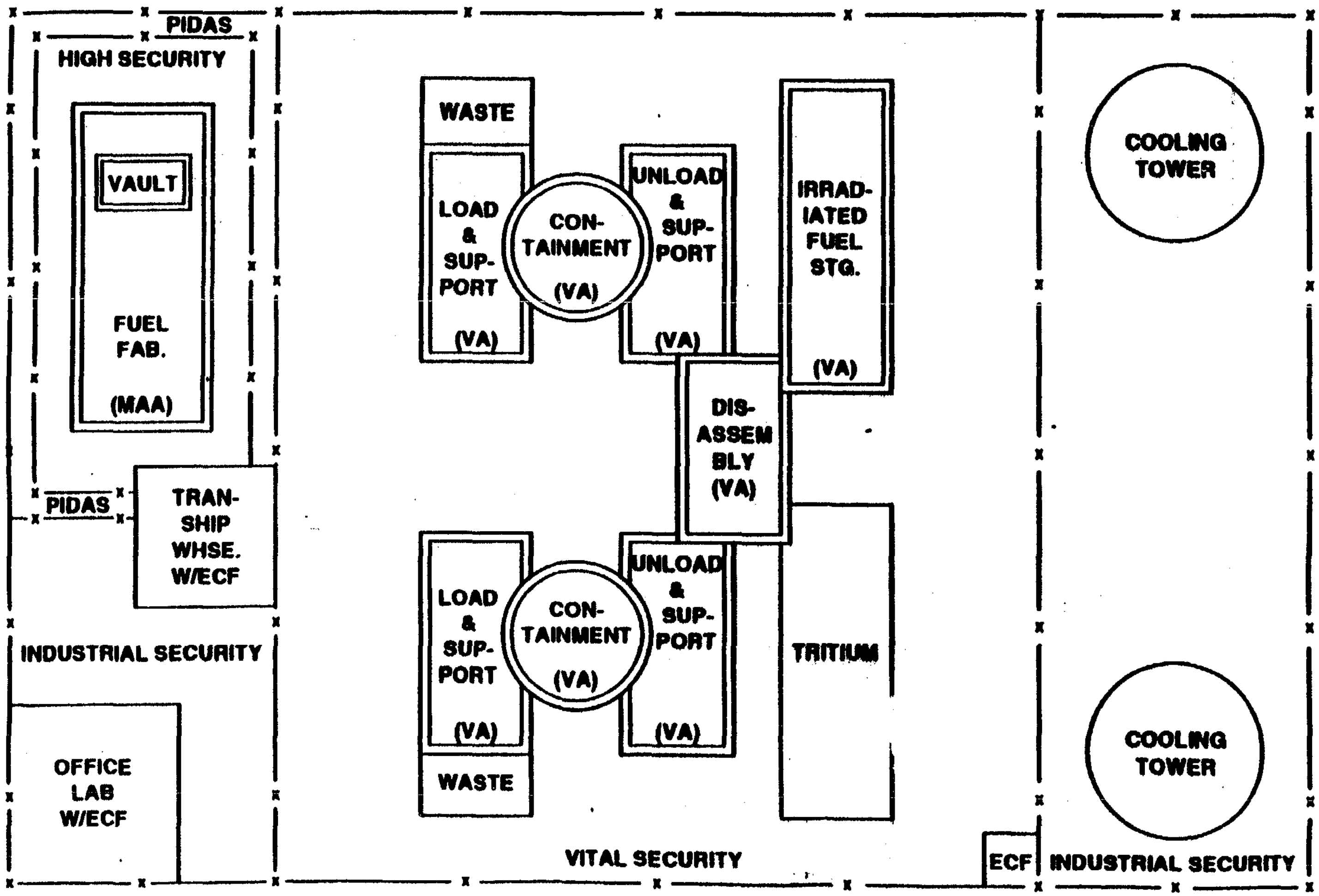
SAFEGUARDS, SECURITY, AND EMERGENCY PREPAREDNESS (Cont'd)

ECONOMICAL PERSONNEL ACCOUNTABILITY

- COMBINE WITH SECURITY
  - COMMON COMPONENTS
  - TRANSPARENT FUNCTION
- CONTROL PERSONNEL
  - ZONAL EVACUATION
  - CONTAINMENT

PDR600

PLUTONIUM DESTRUCTION NOTIONAL SITE LAYOUT



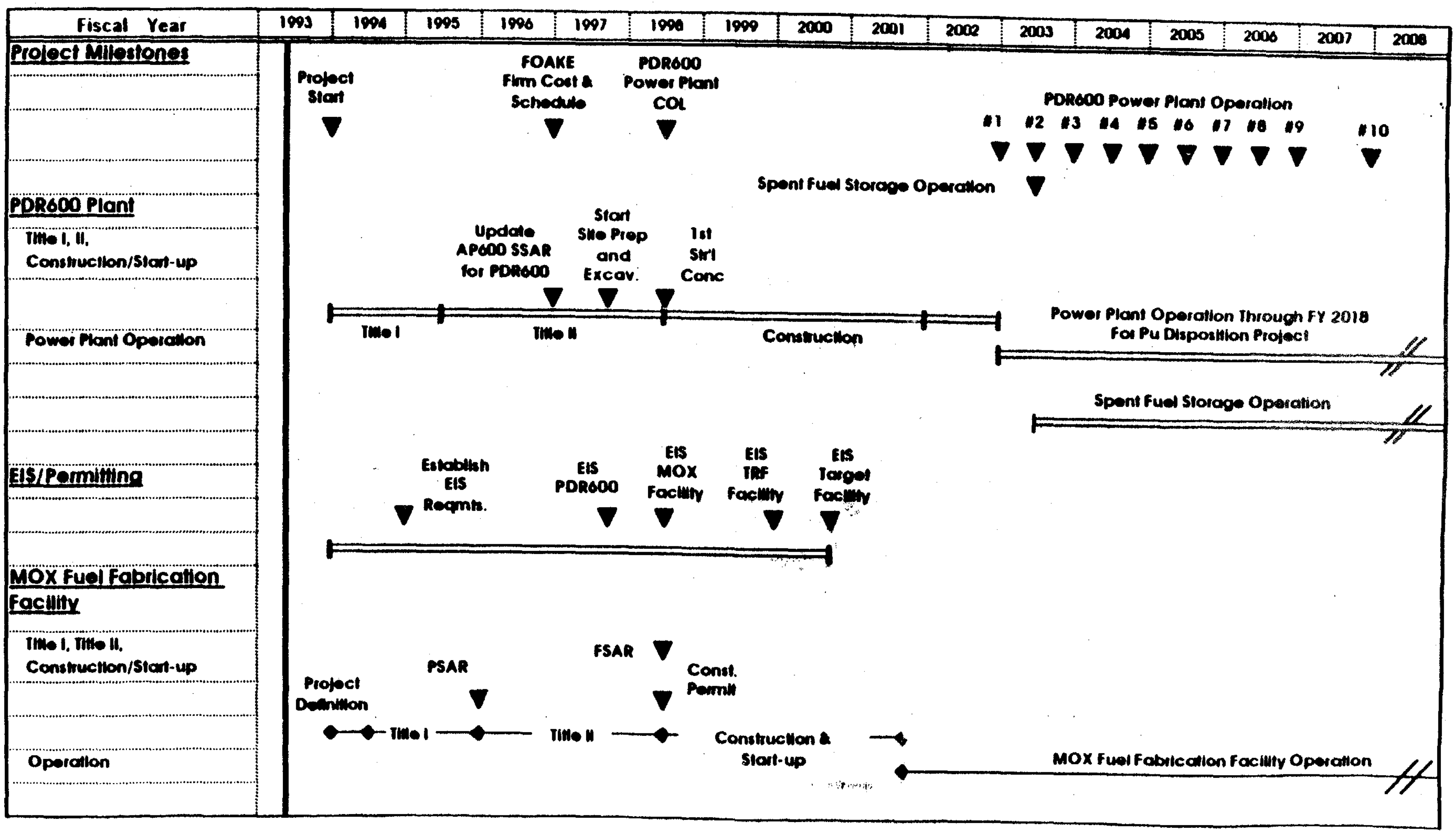
NO SCALE
BUILDING ORIENTATION
DIFFERENT FROM
SITE SCHEMATIC

ECF = ENTRY CONTROL FACILITY
VA = VITAL AREA
MAA = MATERIAL ACCESS AREA
PIDAS = DOUBLE FENCE WITH SENSORS

PDR600

Spent Fuel Option Schedule -

Reactors Begin Operation in 2003



MOX FUEL FABRICATION COSTS
COSTS RANGE FROM $750 TO $1000/KgHM

	Options #2 and #3 200 MT/Year	Option #1 800 MT/Year
Capital Cost, $ Millions	443.2	1010.2
Staffing, No. of People	455	1046
Manufacturing Cost		
• $ Millions	105.5	349.9
• $/KgHM	527.5	437.4
Facility Cost, $/KgHM	243.7	555.5
Total Cost, $/KgHM	771.2	992.9
Total Cost @ 150 MT/Yr. $/KgHM	848.9	N/A

PDR600

PREOPERATIONAL COSTS FOR SPENT FUEL OPTION - PERMITTING AND TRAINING ARE MAJOR EXPENDITURES

	Fuel Fab Facility (200 MT/Yr.)	Reactors (10PDR600s)	Total
R&D	0.0	0.0	0.0
Pre-Title Design & Engineering	0.0	(15.0)[1]	(15.0)
Regulatory, Safety and Environment	13.0	195.1	208.1
Plant Acceptance	Included in Capital Costs		
Reactor Complex Test Operations and Setup	30.1	453.5	483.6
Reactor Complex Procedures	Included in Test Ops./Setup		
Reactor Complex Administration	24.5	171.6	196.1
Total Preoperational Costs	67.6	820.2	887.8
DOE	67.6	709.3	776.9
BOP Owner (ECA)	0.0	110.9	110.9

[1] Design Study was previously funded by DOE/EPRI

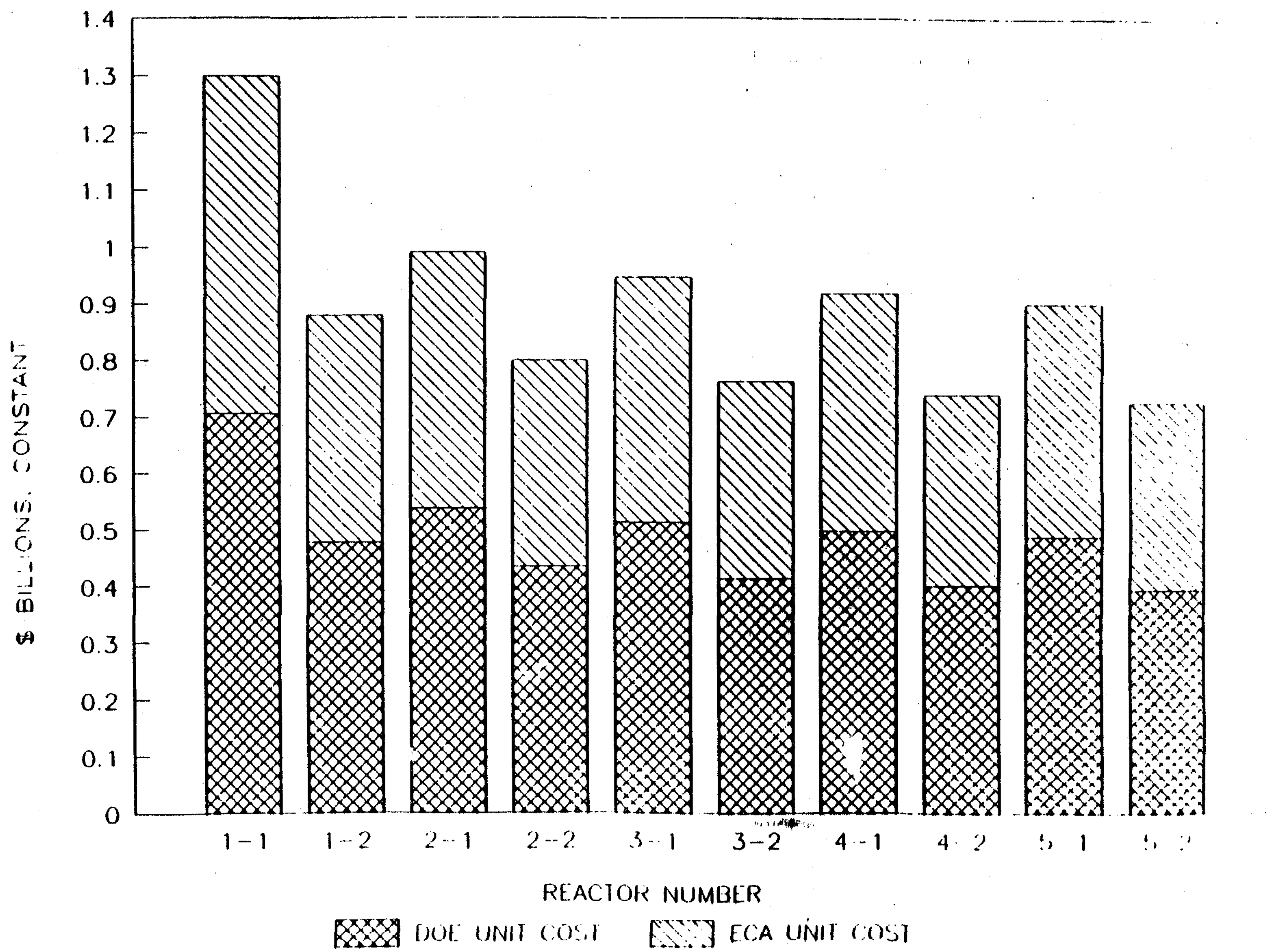
PDR600

REACTOR CAPITAL AND O&M COSTS ARE EVENLY DIVIDED BETWEEN DOE AND ENERGY CONVERSION AREAS (ECA)

	%
CAPITAL	
DOE Scope	
- Yardwork	0.7
- Reactor & Auxiliary Bldg.	12.7
- Access Control Bldg.	0.7
- Annex Bldg.	0.9
- Solid Radwaste Bldg.	0.4
- Fuel Handling Bldg.	1.3
- Miscellaneous Building	1.2
- Reactor Plant Equipment	35.2
- Miscellaneous Equipment	1.2
Total DOE Scope	54.3
BOP Owner Scope (ECA)	
- Turbine Generator Bldg.	2.9
- Turbine Plant Equipment	24.1
- Electrical Plant Equipment	11.3
- Air and Water System	1.9
- Communication System	0.1
- Miscellaneous Equipment	0.2
- Condenser and Heat Reject. System	5.2
Total BOP Owner Scope	45.7
O&M (Excluding Fuel)	
DOE	51.5
BOP Owner (ECA)	48.5

PDR600

SPENT FUEL OPTION UNIT COSTS - 94% LEARNING CURVE USED



PDR600

CAPITAL COST FOR SPENT FUEL OPTION EQUALS $9.2 BILLION

	Total Capital Cost $ Millions
Title 1, 2, 3 Design Engrg. Inspection	
• Design Certification	(120.0)[1]
• FOAKE	(157.2)[1]
• Post-FOAKE	105.0
• Site Engineering	126.0
Facility (Direct Costs)	
• Labor	1,331.8
• Equipment & Materials	4,471.7
Spares	95.0
Initial Loading	Included in fuel fab. costs ($55.2 Million/First Core)
Management & Administration (Indirect Costs)	1,732.7
Contingency	919.4
Total	8,781.6
Total MOX Plant Capital Cost	443.2
Total Program Capital Cost	9,224.8
DOE Share	5,275.8
BOP (ECA) Owner Share	3,949.0

[1] Design Certification and FOAKE are existing funded programs and not included in totals.

PDR600

COST OF CONSTRUCTION AND ENGINEERING FOR THE SPENT FUEL OPTION RANGES BETWEEN $8 - $10 BILLION

	$ Billions		
	DOE	**BOP (ECA)**	**TOTAL**
Option #1 (Pu Spiking)			
• 1 Reactor	2.1	0.5	2.6
Option #2 (Spent Fuel)			
• 8 Reactors	4.8	3.6	8.4
• 10 Reactors	6.0	4.1	10.1
Option #3 (Pu Destruction)			
• 14 Reactors	8.1	5.5	13.6

AVERAGE ANNUAL OPERATIONAL AND LIFE CYCLE COSTS FOR THE SPENT FUEL OPTION - OPERATING MARGIN IS $700 TO $750 MILLION PER YEAR

	AVERAGE OPERATIONAL COSTS & MARGIN MILLION $/YEAR			Life Cycle Cost Million $
	O&M Cost	Revenue	Margin	
Option #1 (Pu Spiking) • 1 Reactor	114.9	136.1	21.2	6,126
Option #2 (Spent Fuel) • 8 Reactors • 10 Reactors	450.8 475.0	1,170.1 1,232.9	719.3 757.9	25,036 27,688
Option #3 (Pu Destruction) • 14 Reactors	611.4	1,648.2	1036.8	37,427

PDR600

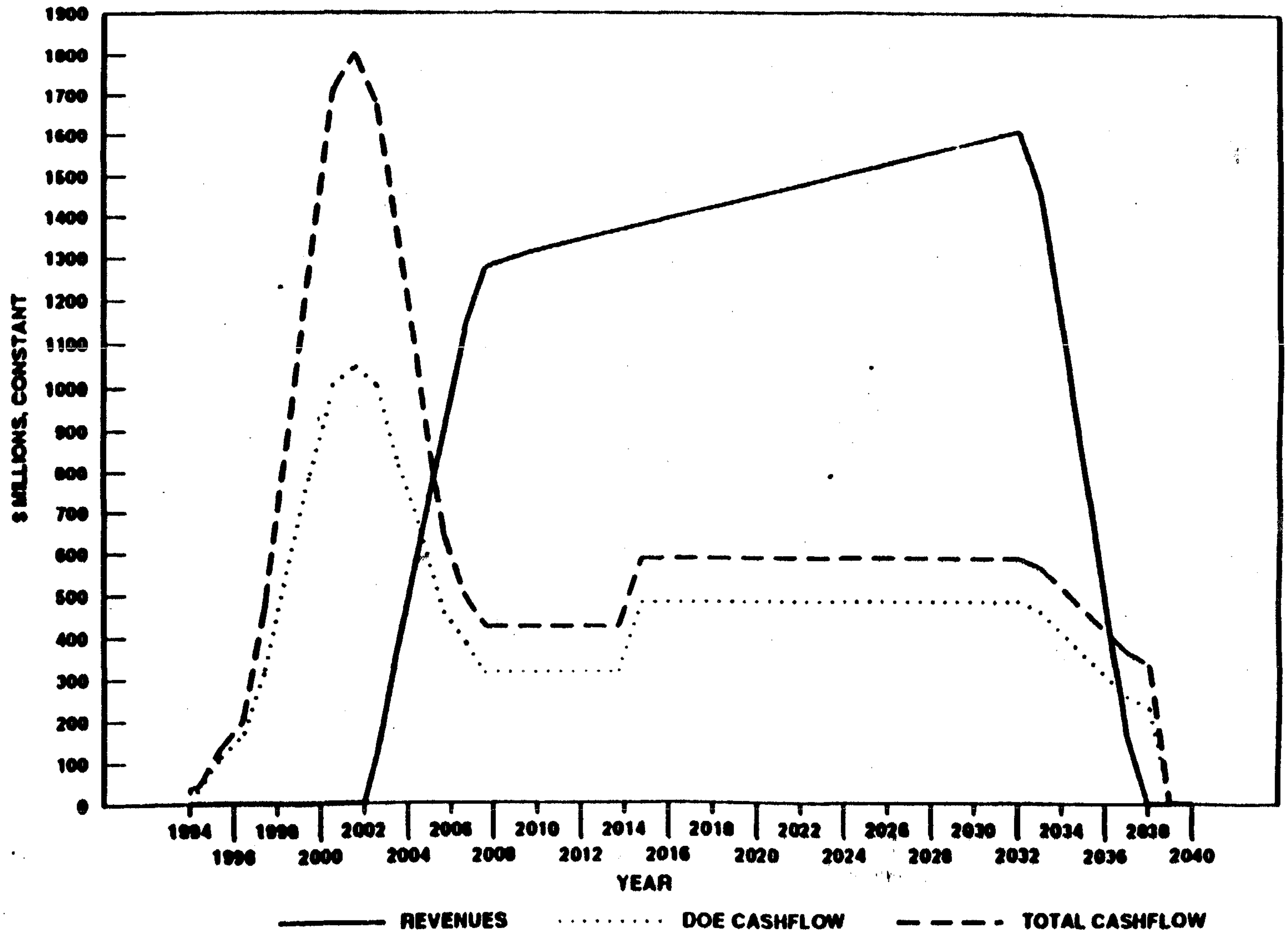
PDR STUDY ANALYSIS ASSUMPTIONS - CUMULATIVE NET CASH FLOW USED FOR BREAK-EVEN ANALYSIS

ANNUAL CASH FLOW	=	Sum of Annual Preoperational, Capital and Operating Costs including MOX or uranium fuel
ANNUAL REVENUES	=	PDR Electric Revenue Rate times kilowatt-hours generated per year
ANNUAL NET CASH FLOW	=	Annual Revenues minus Annual Cash Flow
CUMULATIVE NET CASH FLOW	=	Sum of Annual Net Cash Flow (30 Year Economic Life used for each reactor)

Note: Steam Revenues to DOE were to be calculated by DOE from data provided - C. R. Hudson letter of 4/15/93.

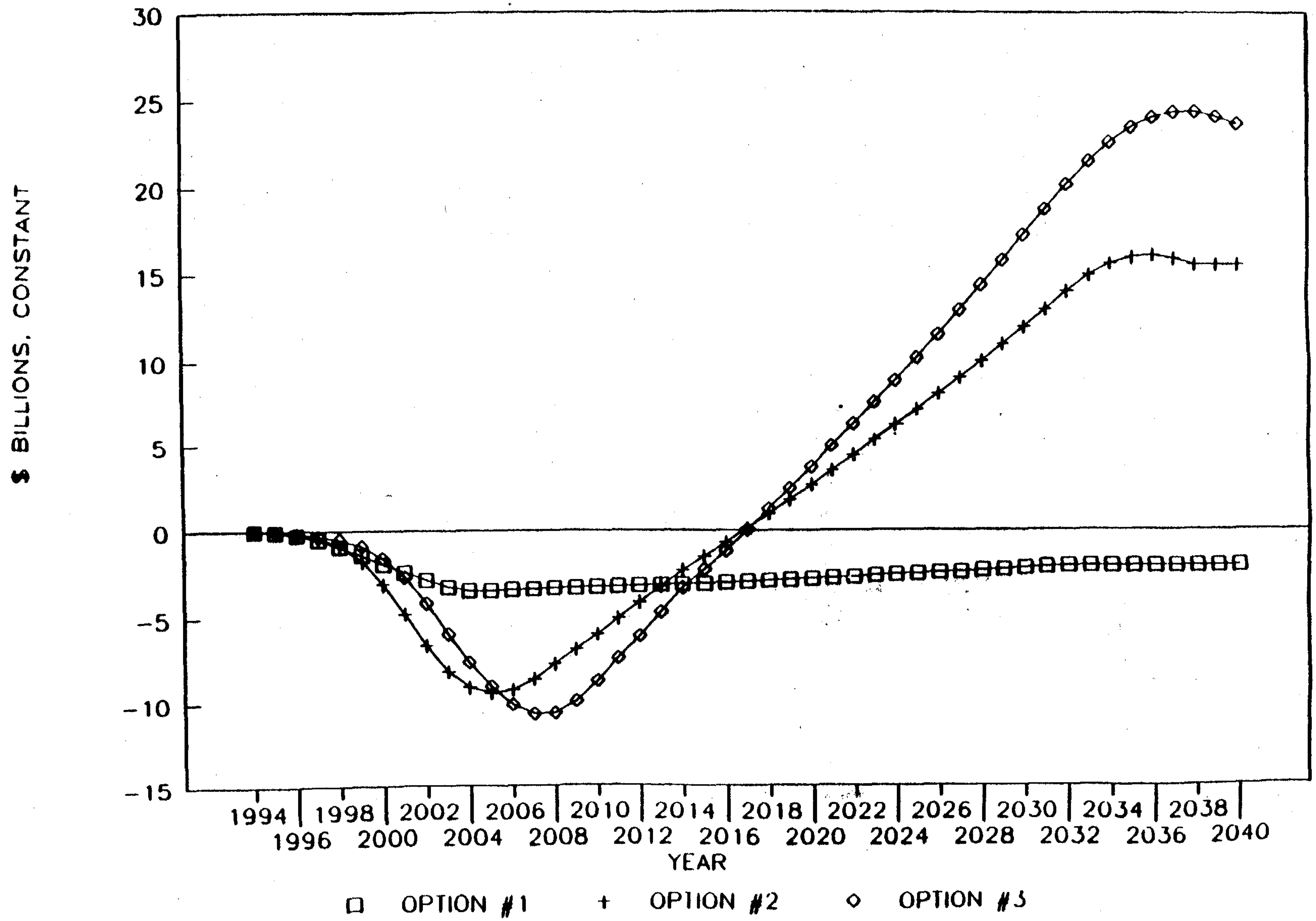
PDR600

ANNUAL REVENUES AND CASH FLOW FOR SPENT FUEL OPTION - PROGRAM EXPENDITURES PEAK AT $1.8 BILLION PER YEAR



PDR600

CUMULATIVE NET CASH FLOW
SPENT FUEL OPTION BREAKS EVEN IN 25 YEAR PROGRAM PERIOD



PDR600

SUMMARY OF COST AND SCHEDULE

- Six year lead time for PDR600 constructions results in first plant being on line in 2003
- 200 MT/Yr. MOX fabrication facility costs $443 million and results in MOX fab costs of $750 to $850/KgHM
- Spent fuel option requires 8 to 10 PDR600 reactors at a construction and engineering cost of $8 - 10 billion
- Pu Spiking and Pu Destruction options have construction and engineering costs of $2.6 and $13.6 billion, respectfully
- Net cash flow of the spent fuel option is breakeven in the 25 year program period and totals $15 billion for 30 year economic life of each reactor
- Net cash flow for the Pu spiking option with one PDR600 does not breakeven in the 25 year program period or a 30 year reactor economic life
- The Pu Destruction option with the 14-15 PDR600s shows breakeven within the 25 year program period and $24-25 billion in net cash flow for a 30 year reactor economic life. (No reprocessing plant)

PDR600

PDR600 PLUTONIUM CONSUMPTION STUDY
PRESENTATION TO TECHNICAL REVIEW COMMITTEE

MAY 24, 1993
PINEY POINT, MARYLAND

INFRASTRUCTURE

PRESENTER
RON RITTENBERGER

DEFINITION OF INFRASTRUCTURE REQUIRED TO SUPPORT THE PDR600

- MAJOR FACILITIES REQUIRED:

  <u>INSIDE REACTOR COMPLEX</u>

  - REACTOR PLANT
  - TURBINE ISLAND
  - PLUTONIUM RECEIVING AND PREPARATION
  - MOX FUEL FABRICATION
  - SPENT FUEL PROCESSING AND/OR STORAGE
  - WASTE PROCESSING

  <u>OUTSIDE REACTOR COMPLEX</u>

  - DESIGN AND DEVELOPMENT
  - MANUFACTURE COMPONENTS AND SYSTEMS
  - OPERATION AND TRAINING
  - EXISTING DOE AND COMMERCIAL FACILITIES
  - WASTE DISPOSAL
  - EXISTING DEVELOPMENT AND TEST
  - NEW DEVELOPMENT AND TEST

MOX FUEL CYCLE TECHNOLOGY HAS BEEN DEMONSTRATED

- MOX FUEL CYCLE TECHNOLOGY IS BEING PURSUED OUTSIDE THE U.S.:
  - FEDERAL REPUBLIC OF GERMANY (ALKEM)
  - UNITED KINGDOM (BNFL)
  - FRANCE (EDF)
  - BELGIUM (BN)
  - JAPAN (TOKAI)
- MOX FUEL HAS BEEN DESIGNED, MANUFACTURED AND TESTED IN THE U.S.
  - SAXTON
  - SAN ONOFRE 1
  - BEZNAU 2 (DESIGNED)

OUTSIDE REACTOR FACILITIES TO IMPLEMENT PDR600 EXIST

<u>DOE FACILITIES:</u>

- SAVANNAH RIVER
- HANFORD
- IDAHO NATIONAL ENGINEERING LABORATORY
- LOS ALAMOS
- ROCKY FLATS
- OAK RIDGE NATIONAL LABORATORY
- NEVADA TEST SITE AND YUCCA MOUNTAIN

<u>COMMERCIAL FACILITIES:</u>

- WESTINGHOUSE ENERGY SYSTEMS
  - NUCLEAR PLANT DESIGN, DEVELOPMENT AND TESTING
  - NUCLEAR FUEL
  - NUCLEAR SERVICE
  - NUCLEAR COMPONENTS
- NUCLEAR PLANT CONSTRUCTORS

EXISTING DEVELOPMENT AND TEST CAPABILITIES TO SUPPORT THE PDR600 ARE EXTENSIVE

- U.S. FACILITIES AVAILABLE FOR ANY DEVELOPMENT AND TESTING REQUIRED
  - DOE FACILITIES: SAVANNAH RIVER, HANFORD, IDAHO NATIONAL ENGINEERING LABORATORY, ARGONNE, OTHERS
  - UNIVERSITIES: COLUMBIA, IOWA, MIT, AND MARYLAND, OTHERS
  - PRIVATE INDUSTRY TESTING FACILITIES: WESTINGHOUSE, BABCOCK AND WILCOX, BATTELLE, OTHERS
- NEW DEVELOPMENT AND TEST FACILITIES ARE NOT NEEDED

INFRASTRUCTURE AND TECHNOLOGY EXIST TO SUPPORT THE IMPLEMENTATION OF PDR600 FOR PLUTONIUM DISPOSITION

- MOX FUEL BURNUP CAPABILITIES HAVE BEEN SUCCESSFULLY DEMONSTRATED IN LWR'S:
  - UNITED STATES: SAXTON AND SAN ONOFRE 1
  - BELGIUM: BR3
  - GERMANY: GKN 1, KKG, KKN
  - SWITZERLAND: BEZNAU 2
- TECHNOLOGY EXISTS COLLECTIVELY, AND HAS BEEN DEMONSTRATED FOR ALL FACILITIES NECESSARY TO SUPPORT THE DISPOSAL OF WEAPONS GRADE PLUTONIUM:
  - REACTOR TECHNOLOGY USING MOX FUEL
  - WEAPONS GRADE PLUTONIUM HANDLING
  - MOX FUEL HANDLING
  - MOX FUEL FABRICATION
  - IRRADIATED FUEL HANDLING
- POTENTIAL SITES TO LOCATE THE COMPLEX AND INDIVIDUAL PLANTS INCLUDE:
  - SAVANNAH RIVER
  - HANFORD
  - IDAHO NATIONAL ENGINEERING LABORATORY
  - OAK RIDGE NATIONAL LABORATORY
  - PREVIOUSLY CONSIDERED COMMERCIAL LOCATIONS
  - GREEN FIELD
  - FORMER SOVIET UNION

SUMMARY OF ALTERNATE STRATEGIES

DEVELOPMENT OF DEPLOYMENT STRATEGIES NEEDS TO CONSIDER:

- SAFEGUARDS TO PREVENT PLUTONIUM DIVERSION
- MINIMIZATION OF COST TO DoE
- DoE SCHEDULE
- PERMANENT DISPOSAL OF IRRADIATED (AND REMAINING) PLUTONIUM

PROGRAM PHASES WHICH NEED TO BE ADDRESSED

- CONVERSION OF WEAPON'S SHAPES TO PLUTONIUM METAL OR OXIDE
- DESIGN/FABRICATION OF A MOX FUEL FABRICATION FACILITY
- LOCATING SITE(S) FOR THE COMPLEX
- DISPOSAL OF SPENT FUEL
- DECOMMISSIONING OF THE REACTOR UPON PROGRAM COMPLETION

PDR600

CONSTRAINTS & OPPORTUNITIES FOR PLUTONIUM DISPOSAL

- MOX FUEL CAN BE LESS EXPENSIVE THAN UO_2 FUEL
  - Free fissile material
  - No capital charges for fuel plant
- USE COMMERCIAL SUPPLIERS FOR FUEL TECHNOLOGY TO ACHIEVE FUEL RELIABILITY
- DoE FINANCING/GUARANTEES REQUIRED
- NEGOTIATE UP-FRONT AGREEMENTS TO SELL REACTORS TO UTILITIES UPON COMPLETION OF THE PROGRAM

ONE POSSIBLE SCENARIO FOR SPENT FUEL OR PU DESTRUCTION OPTION

- BUILD TWO TO FOUR REACTORS AND A FUEL FABRICATION FACILITY AT ONE SITE
- BUILD TWO TO FOUR REACTORS AT ALTERNATE SITES WITHOUT A FUEL FABRICATION FACILITY
- SHIP FRESH FUEL FROM FIRST SITE TO ALTERNATE SITES USING APPROPRIATE SAFEGUARDS
- EVENTUALLY, ALL IRRADIATED FUEL ENDS UP IN A GEOLOGICAL REPOSITORY

PDR600

MATRIX FOR DEPLOYMENT OF REACTOR & FUEL FABRICATION FACILITY OPTIONS

Item	Location #1			Location #2	Location #3
	U.S.			**Russia/C.I.S.**	**Europe/Japan**
• **Fuel Fab. Facility**					
Pu Source	U.S.			Russia/C.I.S.	-------
Pu Conversion	DoE Site			Tomsk, Chelyabinsk	-------
Pu Fabrication	DoE Site Barnwell			Chelyabinsk	Siemens, BNFL, BN, France, Japan
Fabrication Technology	Siemens/BNFL, DoE/Anderson Plant, New Plant			Russian Design, Siemens Plant	EEC/Japan
• **Reactors**					
Types	**PDR**	**Utility Reactor (Operating)**	**Utility Reactor (Partially Finished)**	**VVERs**	**PWRs/BWRs**
	DoE/Gov. Site: Hanford, SRS, ORNL.	No interest at this time.	WNP 1-3, Shorham.	-------	-------

SUMMARY

- PDR600 EFFECTIVE FOR PLUTONIUM DISPOSITION
- PDR600 CAN BE BUILT ALMOST ANYWHERE
- OTHER WESTINGHOUSE REACTOR DESIGNS COULD BE USED
- DISPOSAL OF SPENT FUEL SAME AS COMMERCIAL FUEL
- PDR600 COMPLIES WITH REGULATORY REQUIREMENTS FOR SAFETY AND ENVIRONMENT
- PDR600 CAN PRODUCE TRITIUM IF REQUIRED