Westinghouse
Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

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January 10, 1997

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D. C., 20555

ATTENTION: T. R. QUAY

SUBJECT: AP600 DRAFT SSAR APPENDIX 1B (SAMDA) AND RESPONSE TO REQUESTS FOR ADDITION INFORMATION

Dear Mr. Quay:

Enclosure 1 of this letter provides a draft copy of AP600 SSAR Appendix 1B. Appendix 1B will be included in Revision 11 to the SSAR, which is scheduled for February 28, 1997. No changes are expected between the draft copy enclosed with this letter and the Appendix 1B that will be included in SSAR revision 11. If there are any changes, they will be clearly identified to the staff.

Enclosure 2 provides Westinghouse responses to NRC requests for additional information pertaining to Severe Accident Mitigation Design Alternatives (SAMDA). Specifically, responses are provided for RAIs 100.14 through 100.31. The responses close, from a Westinghouse perspective, the addressed questions. The NRC should review these responses and inform Westinghouse of the status to be designated in the "NRC Status" column of the OITS.

The common theme in both the AP600 SSAR Appendix 1B and the SAMDA RAI responses is that there are no alternate severe accident mitigation design features for AP600 for which the safety benefit outweighs the costs of incorporating the design feature. This conclusion is expected considering that one of the objectives of the Utility Requirements Document (URD) is to address severe accidents. The evolution of the AP600 design has considered severe accidents via the results of the Level 1 and Level 2 Probabilistic Risk Assessment (PRA). Consequently, severe accident mitigation design features not included in other plants, as outline in AP600 SSAR Appendix 1B, are included in the AP600 plant design. Since AP600 meets the URD requirements, and has incorporated the PRA into the design process, the conclusion that no additional SAMDA would provide significant risk benefit should be expected.

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Per the request of the NRC staff, the enclosed information is being provided to the NRC on an expedited schedule with the understanding that the NRC consultants who are to review the enclosed material are only available to review it in the first quarter of 1997. As agreed to during a telecon with Mr. Dino Scaletti, NRC, this information was to be provided by January 10, 1997. Westinghouse expects the NRC will review the enclosures in a timely manner, in accordance with NRC consultants near-term schedule commitments.

Please contact Cynthia L. Haag on (412) 374-4277 if you have any questions concerning this transmittal.

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/jml

Enclosures

cc: D. Scaletti, NRC (enclosures)
J. Sebrosky, NRC (enclosure 1)
J. Kudrick, NRC (w/o enclosures)
N. J. Liparulo, Westinghouse (w/o enclosures)

Enclosure 1 to Westinghouse
Letter NSD-NRC-97-4937

January 10, 1997

APPENDIX 1B

SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

1B.1 Introduction

This report provides an evaluation of Severe Accident Mitigation Design Alternatives (SAMDA) for the Westinghouse AP600 design. This evaluation is performed to evaluate whether or not the safety benefit of the SAMDA outweighs the costs of incorporating the SAMDA in the plant, and is conducted in accordance with applicable regulatory requirements as identified below.

The National Environmental Policy Act (NEPA), Section 102.(C)(iii) requires, in part, that

> ...all agencies of the Federal Government shall ... (C) include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on ... (iii) alternatives to the proposed action.

10 CFR 52.47(a)(ii) requires an applicant for design certification to demonstrate

> ... compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f) ...

A relevant requirement of 10 CFR 50.34(f) contained in subparagraph (1)(i) requires the performance of

> ... a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant ...

In SECY-91-229, the NRC staff recommends that severe accident mitigation design alternatives be addressed for certified designs in a single rulemaking process that would address both the 10 CFR 50.34 (f) and NEPA considerations in the 10 CFR Part 52 design certification rulemaking. SECY-91-229 further recommends that applicants for design certification assess SAMDAs and the applicable decision rationale as to why they will or will not benefit the safety of their designs. The Commission approved the staff recommendations in a memorandum dated October 25, 1991 (Reference 8).

1B.2 Summary

An evaluation of candidate modifications to the AP600 design was conducted to evaluate the potential for such modifications to provide significant and practical improvements in the radiological risk profile of the AP600 design.

The process used for identifying and selecting candidate design alternatives included a review of SAMDAs evaluated for other plant designs. Several SAMDA designs evaluated previously for other plants were excluded from the present evaluation because they have already been incorporated or otherwise addressed in the AP600 design. These include:

- Hydrogen ignition system
- Reactor cavity flooding system
- Reactor coolant pump seal cooling
- Reactor coolant system depressurization
- Reactor vessel exterior cooling.

Additional design alternatives were identified based upon the results of the AP600 probabilistic risk assessment (Reference 1). ~~Fourteen~~ Fifteen candidate design alternatives were selected for further evaluation.

An evaluation of each of these alternatives was performed using a bounding methodology such that the potential benefit of each alternative is conservatively maximized. As part of this process, ~~is~~ it was assumed that each SAMDA performs beyond expectations and completely eliminates the severe accident sequences that the design alternative addresses. In addition, the capital cost estimates for each alternative were intentionally biased on the low side to maximize the risk reduction benefit. This approach maximizes the potential benefits associated with each alternative.

The results show that despite the significant conservatism employed in the evaluation, none of the SAMDAs evaluated provide risk reductions which are cost beneficial. The results also show that even a conceptual "ideal SAMDA", one which reduces the total plant radiological risk to zero, would not be cost effective. This is due primarily to the already low risk profile of the AP600 design, which is approximately two orders of magnitude below existing plants.

1B.3 Selection of SAMDAS

Candidate design alternatives were selected based upon design alternatives evaluated for other plant designs (References 2, 3, and 4) as well as suggestions from AP600 design personnel. Additional candidate design alternatives were selected based upon an assessment of the AP600 probabilistic risk assessment results. ~~Fourteen~~ Fifteen design alternatives were finally selected for further evaluation. These ~~fourteen~~ fifteen SAMDAs are:

- Chemical volume and control system (CV~~CS~~S) upgraded to mitigate small LOCAs
- Filtered containment vent
- Normal residual heat removal system (~~RHR~~ RNS) located inside containment
- Self-actuating containment isolation valves
- Passive containment spray
- Active high pressure safety injection system
- Steam generator shell side passive heat removal system
- Steam generator safety valve flow directed to in-containment refueling water storage tank (IRWST)

- Increase steam generator secondary side pressure capacity
- Secondary containment filtered ventilation
- ~~IRWST discharge valve diversification~~ Diverse IRWST injection valves
- Diverse containment recirculation valves
- Ex-vessel core catcher
- High pressure containment design
- Diverse actuation system (DAS) improved reliability.

A description of each design alternative evaluated for AP600 is presented in Subsection 1B.7.

Several design alternatives addressed in previous SAMDA evaluations for other plants were excluded from further evaluation because the alternatives are already incorporated or otherwise addressed into the AP600 design. These design features include:

- Hydrogen ignition system
- Reactor cavity flooding system
- Reactor coolant pump seal cooling
- Reactor coolant system depressurization
- Reactor vessel exterior cooling.

1B.4 Methodology

The severe accident mitigation design alternatives analysis employs a bounding methodology such that the benefit is conservatively maximized and the capital cost is conservatively minimized for each SAMDA. The risk reduction, capital cost estimates, and cost benefit analysis methods are discussed in this subsection.

1B.4.1 Risk Reduction

Risk for the purpose of this evaluation is the probability of core damage for each accident initiator, multiplied by the consequences of the accident (population dose), expressed in terms of man-rem per year. The total risk is the sum of the risks from all the accidents.

The reduction of risk for each SAMDA is the difference in risk between the AP600 design and an AP600 design with the design alternative incorporated.

It is assumed that each SAMDA works perfectly and completely eliminates the accident sequences that the design alternative addresses. This approach conservatively maximizes the benefits associated with each design alternative, and is not intended to imply that such a perfect design is possible. The SAMDA benefits are the reduction of risk in terms of whole body man-rem per year received by the total population within a 50-mile radius of the AP600 plant site.

Each design alternative is evaluated based on how it affects each of the release categories in the AP600 probabilistic risk assessment.

1B.4.2 Capital Cost Estimates

The capital cost estimates for each SAMDA are intentionally biased on the low side to maximize the risk reduction benefit. All reasonably anticipated one-time capital costs are accounted for in the estimates. Actual plant costs are expected to be higher since the cost estimates do not include the cost of testing and maintenance or the engineering cost to design the alternative to fit into the AP600. The cost estimates are based on ~~1992~~ 1996 U.S. dollars.

1B.4.3 Cost Benefit Analysis

In order to compare the risk reduction, which is reported in man-rem per year, to the capital costs, which are reported in dollars, a common set of units must be established. For this evaluation, the risk reduction is converted to a capital benefit which can then be directly compared with the capital costs.

The benefit of each design alternative is the reduction of risk in terms of whole body man-rem per year received by the total population within a 50-mile radius of the AP600 plant site. Consistent with previous SAMDA evaluations and NRC regulatory analysis guidelines, a value of $1,000 per offsite man-rem averted is used to convert man-rem per year to dollars per year. This value is intended to be the surrogate for all offsite consequences including property damage and is referred to as the annual levelized benefit.

The risk reduction reported as dollars per year is then converted to a maximum capital benefit which can then be compared to the capital costs. The maximum capital benefit is equal to the annual levelized benefit (dollars per year) divided by the annual levelized fixed charge rate.

The annual levelized fixed charge rate is determined from a number of financial factors. These factors are given in Table 1B.4-1 and are taken from the EPRI Technical Assessment Guide (Reference 6). The equations used to determine the annual levelized fixed charge rate are from the Nuclear Energy Cost Data Base (Reference 7). For a nuclear plant economic life of ~~60~~ 30 years and a tax life of 15 years, the annual levelized fixed charge rate is ~~15.4~~ 15.7 percent in current U.S. dollars (with inflation).

1B.5 PRA Release Categories

To assess each design alternative's reduction of risk, the potential for each alternative to reduce the frequency of occurrence or the consequence of each release category is assessed. The steps involved in creating the AP600 release categories are briefly discussed in this subsection.

The AP600 Level 1 plant event trees identify the sequences that lead to core damage. Sequences that have similar characteristics are grouped together into accident subclasses for the containment system analysis. The characteristics considered in the binning of the plant event sequences into the accident classes are as follows:

- The initiating event type -- such as loss of coolant accident (LOCA) or anticipated transient without scram (ATWS), leading to core damage
- The primary system pressure at the time of initial core ~~damage~~ uncovery (high or low)
- Timing of core damage (early or late)
- Containment integrity at the time of core damage (intact, not isolated or bypassed ~~or impaired~~)
- ~~Availability of safety systems after core damage~~
- Disposition of water in the containment at the time of core damage
- ~~Containment pressure and temperature at the time of core damage.~~

Containment event trees for each of the significant accident subclasses are developed and discussed in the AP600 probabilistic risk assessment (Reference 1). Consideration of severe accident phenomena that may challenge containment integrity forms the basis for the nodes on the containment event tree. Operator actions or system top events are generally considered with respect to preventing or mitigating severe phenomena. The containment event tree considers that the following phenomena represent the severe accident issues relevant to the AP600 containment integrity:~~The containment event tree analysis considers both the containment and associated auxiliary systems. In particular, the following items are considered:~~

- In-vessel fuel-coolant interactions
- In-vessel hydrogen generation
- Creep rupture failure of steam generator tubes
- High-pressure melt ejection
- Melt attack on the containment pressure boundary
- Containment overpressurization from decay heat
- Reactor vessel integrity
- Ex-vessel fuel-coolant interactions
- Core-concrete interaction and hydrogen generation
- Hydrogen deflagration and detonation
- Elevated temperatures of the containment shell (diffusion flame heating)
- Elevated gas temperatures (equipment survivability)
- ~~Containment isolation system~~
- ~~Passive containment cooling system~~
- ~~In containment refueling water storage tank injection~~
- ~~Ex-vessel debris cooling.~~

~~The functions accomplished by these systems are:~~

- ~~Maintenance of containment integrity and/or the reduction of containment pressure~~
- ~~Prevention of vessel failure and/or core melt arrest~~
- ~~Cooling of ex-vessel debris.~~

The end-state of each path on the containment event tree describes the effectiveness of the containment to mitigate offsite doses for that accident sequence. The radiological consequences of the core-melt accident are largely determined by three major considerations:

- The mode of the postulated containment failure (bypass, isolation failure, gross failure, or intact containment)
- The time of postulated containment failure relative to the time of major fission-product release from the core or core debris
- Fission-product removal mechanisms in the containment

AP600 PRA does not credit active containment fission-product removal mechanisms such as containment sprays or fan coolers. Therefore, natural deposition processes, gravitational settling, thermophoresis, and diffusiophoresis are relied on to scrub aerosols from the containment atmosphere. The natural processes are time-dependent, thus the mode of containment failure, timing of the containment failure, and magnitude of the offsite release are directly related and treated together for the AP600 containment event tree via release categories. The source term for each release category is calculated with the Modular Accident Analysis Program Version 4.0 (MAAP 4.0) code. ~~The endpoints of the containment event trees paths are grouped into appropriate source term categories based on similar fission product releases. Different endpoints for the AP600 plant are defined, depending on the type of containment failure (bypass, isolation failure, or late overpressure due to core-concrete interaction). If the containment does not fail, the availability of the passive containment cooling system water has a strong influence on the containment pressure, and therefore is used to determine the release category. The source term for a representative sequence in each important accident class is calculated with the Modular Accident Analysis Program Version 4.0 (MAAP 4.0) code.~~

The release categories for the AP600 are defined as follows:

- **IC** -- intact containment;
- **CFE** -- containment failure early, occurring in the time frame between the onset of core damage and the end of core relocation;
- **CFI** -- containment failure intermediate, occurring in the time frame between the end of core relocation and 24 hours after core damage;
- **CFL** -- containment failure late, occurring later than 24 hours after the onset of core damage;
- **CI** -- containment isolation failure, with the failure occurring before the onset of core damage;
- **BP** -- containment bypass, with the bypass occurring before the onset of core damage.

- ~~**OK** -- release associated with the leakage from a containment with passive containment cooling water available,~~
- ~~**OKP** -- release associated with the leakage from a containment with passive containment cooling water not available,~~
- ~~**CC** -- release associated with the leakage from a containment that is pressurized with noncondensible gases generated by core-concrete interaction,~~
- ~~**CI** -- release associated with the leakage from a containment that is bypassed or has not been isolated (impaired).The following subsections present a brief description of the accident sequences from the probabilistic risk assessment which represents each AP600 release category.~~

The following subsections present a brief description of the ~~accident sequences form the probabilistic risk assessment which represents each~~ AP600 release categories ~~category~~.

1B.5.1 Release Category IC - Intact Containment

If the containment integrity is maintained throughout the accident, then the release of radiation from the containment is due to nominal leakage and is expected to be within the design basis of the containment. This is the "no failure" containment failure mode and is termed intact containment. The main location for fission-product leakage from the containment is penetration leakage into the auxiliary building where significant deposition of aerosol fission products may occur.

The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The IC release category frequency is 1.5×10^{-7} per year.

1B.5.2 Release Category CFE - Early Containment Failure

Early containment failure is defined as failure that occurs in the time frame between the onset of core damage and the end of core relocation. During the core melt and relocation process, several dynamic phenomena can be postulated to result in rapid pressurization of the containment to the point of failure. The combustion of hydrogen generated in-vessel, steam explosions, and reactor vessel failure from high pressure are major phenomena postulated to have the potential to fail the containment. If the containment fails during or soon after the time when the fuel is overheating and starting to melt, the potential for attenuation of the fission-product release diminishes because of short fission-product residence time in the containment. The fission products released to the containment prior to the containment failure are discharged at high pressure to the environment as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Containment failures postulated within the time of core relocation are binned into release category CFE.

The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The CFE release category frequency is 6.6×10^{-9} per year.

1B.5.3 Release Category CFI - Intermediate Containment Failure

Intermediate containment failure is defined as failure that occurs in the time frame between the end of core relocation and 24 hours after core damage. After the end of the in-vessel fission-product release, the airborne aerosol fission products in the containment have several hours for deposition to attenuate the source term. The global combustion of hydrogen generated in-vessel from a random ignition prior to 24 hours can be postulated to fail the containment. The fission products in the containment atmosphere are discharged at high pressure to the environment as the containment blows down. Containment failures postulated within 24 hours of the onset of core damage are binned into release category CFI.

The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The CFI release category frequency is 1.3×10^{-11} per year.

1B.5.4 Release Category CFL - Late Containment Failure

Late containment failure is defined as containment failure postulated to occur later than 24 hours after the onset of core damage. Since the PRA assumes the dynamic phenomena, such as hydrogen combustion, to occur before 24 hours, this failure mode occurs only from the loss of containment heat removal via failure of the passive containment cooling system. The fission products that are airborne at the time of containment failure will be discharged at high pressure to the environment, as the containment blows down. Subsequent release of fission products can then pass directly to the environment. Accident sequences with failure of containment heat removal are binned in release category CFL.

The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The CFL release category frequency is 1.5×10^{-11} per year.

1B.5.5 Release Category CI - Containment Isolation Failure

A containment isolation failure occurs because of the postulated failure of the system or valves that close the penetrations between the containment and the environment. Containment isolation failure occurs before the onset of core damage. For such a failure, fission-product releases from the reactor coolant system can leak directly from the containment to the environment with diminished potential for attenuation. Most isolation failures occur at a penetration that connects the containment with the auxiliary building. The auxiliary building may provide additional attenuation of aerosol fission-product releases. However, this decontamination is not credited in the containment isolation failure cases. Accident sequences in which the containment does not isolate prior to core damage are binned into release category CI.

The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The CI release category frequency is 3.6×10^{-10} per year.

1B.5.6 Release Category BP - Containment Bypass

Accident sequences in which fission products are released directly from the reactor coolant system to the environment via the secondary system or other interfacing system bypass the containment. The containment failure occurs before the onset of core damage and is a result of the initiating event or adverse conditions occurring at core uncovery. The fission-product release to the environment begins approximately at the onset of fuel damage, and there is no attenuation of the magnitude of the source term from natural deposition processes beyond that which occurs in the reactor coolant system, in the secondary system, or in the interfacing system. Accident sequences that bypass the containment are binned into release category BP.

The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The BP release category frequency is 1.1×10^{-8} per year.

~~1B.5.1 Release Category OK~~

~~The representative sequence for the OK release category has an initiating event which is a 4-inch diameter loss of coolant accident with a failure of the in-containment refueling water storage tank check valves and normal RHR injection. Core damage begins 2.0 hours into the accident. The in containment refueling water storage tank is not drained into the containment cavity to provide external cooling to the reactor vessel, so the core debris is not maintained in the vessel. The vessel fails at 11.8 hours, and the molten core drains into the containment at low pressure. The debris is quenched and cooled in the reactor cavity, so there is no significant ex-vessel release. The passive containment cooling system and hydrogen igniters are available, and containment pressure remains below design pressure.~~

~~The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The OK release category frequency is 2.5 x 10^{-7} per year.~~

~~1B.5.2 Release Category OKP~~

~~The representative sequence for release category OKP is initiated by a 4 inch diameter loss of coolant accident with failures of the in-containment refueling water storage tank check valves, normal RHR injection, and passive containment cooling system cooling water. Four out of four core makeup tanks and accumulators are available. The in containment refueling water storage tank is not drained into the containment cavity to provide external cooling to the reactor vessel, so the core debris is not maintained in the vessel. Core damage occurs at 2.5 hours and vessel failure occurs at 15.8 hours. The debris is quenched and coolable in the reactor cavity because all of the water holdup volumes are full and the condensation from the passive containment cooling system shell returns water to the containment sump. The containment pressure is elevated over the long term, but it equilibrates at a pressure well below the ultimate capacity of the shell, so containment integrity is maintained. No credit is taken in the analysis for accident management or use of alternative methods of wetting the containment shell.~~

~~Because of the influence of water in the containment, there is essentially no difference in fission product release if the debris remains in the vessel or is released to the containment. The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The OKP release category frequency is 5.6 x 10^{-8} per year.~~

~~1B.5.3 Release Category CC~~

~~The representative sequence for release category CC is initiated by a 4 inch diameter loss of coolant accident with a failure of the in containment refueling water storage tank check valves, normal RHR injection, and the passive containment cooling system water flow. Three out of the four core makeup tanks and accumulators are available. The in-containment refueling water storage tank is not drained into the containment cavity to provide external cooling to the reactor vessel, so the core debris is not maintained in the vessel. The core damage begins at 2.0 hours. The vessel fails at 11.3 hours, and the molten core drains into the cavity at low pressure. The cavity dries out because the water from the available core~~

~~makeup tanks and accumulators is trapped as steam or in water holdup volumes. Passive containment cooling system condensation does not keep up with the rate of boiloff from the debris bed. Core concrete interaction creates noncondensible gases that pressurize the containment. At 24 hours after core damage, the pressure in the containment is essentially equal to design pressure. The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The CC release category frequency is 7.6 x 10^{-10} per year.~~

~~1B.5.4 Release Category CI~~

~~The representative sequence for release category CI is initiated by a loss of feedwater to the steam generators and the failure of the passive residual heat removal and automatic depressurization systems. The containment does not isolate. The containment isolation failure is modeled as the failure of one purge line. The core temperature exceeds 2500°K at 4.2 hours. The operator dumps in-containment refueling water storage tank water into the cavity on a high-high core exit temperature. The water surrounds and cools the reactor vessel, preventing vessel failure. The reactor coolant system hot leg ruptures due to the high temperature and pressure in the reactor coolant system. The remainder of the core melts and falls into the lower head. Fission products released into the containment can be directly transported to the environment. The final release fractions, at 24 hours after core damage, are presented in Table 1B.5-1. The CI release category base frequency is 2.0 x 10^{-8} per year. However, because the frequency of excessive leakage, which exceeds the technical specification leakage, from the other release categories is lumped into the CI release category, the overall release category frequency is taken to be 3.0 x 10^{-8} per year.~~

1B.6 Total Population Dose

To assess the potential benefits associated with a design alternative, estimates are made of the total offsite population dose resulting from each of the release categories (i.e., source terms) identified in Subsection 1B.5. The MELCOR Accident Consequence Code System (MACCS), Version 1.5.11.1 ~~1.5~~ (Reference 5) is utilized for this analysis. The code input is identical to the AP600 probabilistic risk assessment, however the consequence evaluated is the effective whole body equivalent dose (50 year committed), resulting from exposure during the initial 24 hours following the onset of core damage, to the total population within a 50-mile radius of the plant.

Table 1B.6-1 presents the estimated mean 50-mile radius population whole body dose in person-rem (man-rem)~~, and median doses in person-sieverts (1 person-sievert equals 100 man-rems)~~ for each release category~~. Table 1B.6-2 shows~~ and the 50-mile population dose risk (24 hours) for each release category, as well. The ~~as the~~ total risk is ~~of~~ 7.3 x 10^{-3} ~~3.42 x 10^{-3}~~-man-rem per year for the AP600 plant.

1B.7 SAMDA Description and Benefit

This subsection describes each SAMDA and the benefit expected due to the modification. In the evaluation of the risk reduction benefit, each SAMDA is assumed to operate perfectly with 100 percent efficiency, without failure of supporting systems. A perfect SAMDA reduces the frequency of accident sequences which it addresses to zero. This is conservative as it maximizes the benefit of each design alternative. The SAMDA will reduce the risk by lowering the frequency, attenuating the release, or both. The benefit will be described in terms of the accident sequences and dose which are affected by the SAMDAs, as well as the overall risk reduction. Note that for the purposes of these evaluations, increases to release category IC are not factored into the risk benefit calculations. The IC dose is sufficiently small (3E+2) that changes to the IC total frequency do not result in an appreciable change to overall results. This is also a conservative representation since this maximizes the risk reduction.

1B.7.1 Upgrade the CV~~C~~S for Small LOCAs

The chemical, volume, and control system (CV~~C~~S) is currently capable of maintaining the reactor coolant system (RCS) inventory to a level in which the core remains covered in the event of a very small (≤ 3/8" ~~3/4"~~ diameter break) loss of coolant accident (LOCAs). This SAMDA involves providing in-containment refueling water storage tank (IRWST) / containment recirculation connections to the CVS and adding a second line from the CVS makeup pumps to the RCS in order to be able to use the system to keep the core covered during small and intermediate LOCAs. ~~This SAMDA involves upgrading the pumping capacity, and line sizes of the CVCS system in order to be able to use the system to keep the core covered during small (≤ 4" diameter breaks) LOCA accidents, as well.~~

A perfect, upgraded CV~~C~~S system is assumed to prevent core damage in the RCS leak, passive RHR heat exchanger tube ruptures, small LOCA, and intermediate LOCA ~~and medium LOCA~~ of ~~all the very small and small LOCAs in~~ each release category. The CV~~C~~S is assumed to have perfect support systems (power supply, component cooling) and to work in all situations regardless of the common cause failures of other systems. This results in a total averted risk of 5.5×10^{-4} ~~5.80×10^{-5}~~ man-rem per year.

1B.7.2 Filtered Vent

This SAMDA consists of placing a filtered containment vent and all associated piping and penetrations into the AP600 containment design. The filtered vent could be used to vent the containment to prevent catastrophic overpressure failure, and also provides filtering capability for source term release. With respect to the AP600 PRA, the possible scenario in which the filtered vent could result in risk reduction would be late containment overpressure failures (release category CFL). Other containment overpressure failures occur due to dynamic severe accident phenomena, such as hydrogen burn, steam explosion, etc. The late containment failures for AP600 are failures of the passive containment cooling system (PCS). Analyses have indicated that for scenarios with PCS failure, air cooling may limit the containment pressure to less than the ultimate pressure. However, for the purposes of the Level 2 PRA, failure of PCS is assumed to result in containment failure based on an adiabatic heatup. To conservatively consider the risk reduction of a filtered vent, the use of a filtered vent to preclude a late containment failure will be evaluated. A decontamination factor (DF) of 1000 will conservatively be assumed for each PRA Level 1 accident classification, even though it is realized that the dose due to noble gases will not be impacted by the filtered vent since

100% of the noble gas fission products will still be released. Therefore, the risk reduction is equal to the decontamination factor assumed, since the PRA Level 1 accident classification frequencies do not change. The total averted risk for a filtered vent is thus 1.0 x 10^{-3} man-rem/yr. ~~A filtered vent added to the containment would prevent containment failure from slow pressurization events by depressurizing the containment through a filter into the environment. Filtered venting affects the source terms from release categories OKP and CC.~~

~~The sequences in release category OKP have no water cooling of the containment shell and pressurize the containment due to decay heat steaming from debris in the cavity or the RCS. Sequences in release category CC dry out the ex vessel debris bed, and pressurize the containment from non condensible gas generation due to core concrete interaction (CCI) in the reactor cavity. However, neither release category contains sequences in which the containment fails. Release category OKP cases pressurize the containment, but the decay heating and the heat removal from the dry PCS reach equilibrium well before the pressure exceeds ultimate capacity. Release category CC cases pressurize the containment slowly and are not predicted to fail the containment before four days after the initiation of the accident, providing ample time for ad hoc accident management procedures to terminate the CCI and prevent containment failure. In both the OKP and CC release categories, the source term to the environment is not much more than the source term from the OK release category in which the containment remains below the design pressure over the long term.~~

~~Filtered venting of these sequences can be assumed to release 100 percent of the noble gas fission products and approximately 1.0 x 10^{-3} of the aerosol fission products (assuming a decontamination factor of 1000 for the particulates). The source terms of the OKP and CC release categories in which the containment remains intact are significantly smaller than the expected source term from filtered ventilation. Therefore, the filtered vent provides no benefit, and in fact provides a liability to the AP600 design by increasing the residual risk. This design alternative is not analyzed further in terms of cost.~~

1B.7.3 Locate Normal ~~RHR~~ Residual Heat Removal Inside Containment

This SAMDA consists of placing the entire normal residual heat removal ~~(RHR)~~ system (RNS) and piping inside the containment pressure boundary. Locating the RNS ~~normal RHR~~ inside the containment would prevent containment bypass due to interfacing system LOCAs (ISLOCA) of the residual heat removal (RHR) system. In past probabilistic risk assessments of current generation nuclear power plants, the ISLOCA is the leading contributor of plant risk because of large offsite consequences. A failure of the valves which isolate the low pressure RHR system from the high pressure RCS causes the RHR system to overpressurize and fail, releasing RCS coolant outside the containment where it cannot be recovered for recirculation cooling of the core. The result is core damage and the direct release of fission products outside the containment.

In the AP600, the RNS ~~RHR system~~ is designed with a higher design pressure than the ~~RHR~~ systems in current pressurized water reactors, and an additional isolation valve is provided in the design. In the probabilistic risk assessment, no ISLOCAs contribute significantly to the core damage frequency of the AP600 (Reference 1, Chapter 33 ~~Table 7-1~~). Therefore, relocating the RNS ~~normal RHR system~~ of the AP600 inside containment will provide virtually no risk reduction benefit and will not be investigated further in terms of cost.

1B.7.4 Self-Actuating Containment Isolation Valves

This SAMDA consists of improved containment isolation provisions on all normally open containment penetrations. The category of "normally open" is limited to normally open pathways to the environment during power and shutdown conditions, excluding closed systems inside and outside the containment such as RNS ~~normal RHR~~ and component cooling. The design alternative would be to add a self-actuating valve or enhance the existing inside containment isolation valve to provide for self actuation in the event that containment conditions are indicative of a severe accident. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. ~~the frequency of all containment isolation failures are subtracted from the CI release category and are added to the OK release category and the risk is requantified.~~ This does not include induced containment failures which occur at the time of the accident such as in cases of vessel rupture or anticipated transients without scram (ATWS). The benefit results in an averted risk of 7.4×10^{-4} ~~1.13×10^{-3}~~ man-rem per year.

1B.7.5 Passive Containment Sprays

This SAMDA involves adding a passive safety-related ~~grade~~ spray system and all associated piping and support systems to the AP600 containment. A passive containment spray system could result in risk benefits in the following ways:

- scrubbing of fission products, primarily for CI failures,
- assuming appropriate timing, containment spray could be used as an alternate means for flooding the reactor vessel (in-vessel retention) and for debris quenching should vessel failure occur,
- containment spray could also be used to control containment pressure for cases in which PCS has failed.

In order to envelop these potential risk benefits, the risk reduction evaluation will assume that containment sprays are perfectly effective for each of these benefits, with the exception of fission product scrubbing for containment bypass. Thus the risk reduction can be conservatively estimated by assuming all release categories except BP are eliminated. Therefore, passive containment spray results in a total averted risk of 6.9×10^{-3} man-rem per year. ~~A perfect containment spray with perfect support systems is assumed to provide fission product scrubbing and release reduction in the event of a failure of containment isolation. Further, sprays ensure water coverage of any core debris in the containment, preventing core-concrete interaction. To evaluate the benefit of containment sprays, the OKP and CC release category frequencies are added to the frequency of the OK release category, and a dose reduction of 100 is assumed to be applied to the CI release category. This results in a total averted risk of 3.39 x 10⁻³ man-rem per year.~~

Additionally, a nonsafety-related containment spray system is evaluated. The nonsafety-related containment spray system utilizes the fire protection pumps as a motive force for spray, less remotely operated valves but approximately the same amount of piping. It will be conservatively assumed that the risk benefits are the same, and that the risk reduction remains the same for the nonsafety-related containment spray system. Thus, the nonsafety-related containment spray system results in a total averted risk of 6.9×10^{-3} man-rem per year.

1B.7.6 Active High Pressure Safety Injection System

This SAMDA consists of adding a safety-related ~~grade~~ active high pressure safety injection (HPSI) pump and all associated piping and support systems to the AP600 design. A perfect high pressure safety injection system is assumed to prevent core melt for all events but ~~transients and small medium and large LOCAs in each release category. Only~~ excessive LOCA and ATWS ~~are assumed to lead to core damage~~. Therefore, to estimate the risk reduction, only the contributions to each release category of Level 1 accident classes 3C (vessel rupture) and 3A (ATWS) need be considered. The averted risk is 6.1×10^{-3} man-rem per year. ~~the frequency of each release category is reduced by the frequencies of all the LOCAs and transients sequences in the categories, and the risk is requantified. The averted risk is 1.86 x 10⁻³ man-rem per year.~~ This SAMDA would completely change the design approach from a plant with passive safety systems to a plant with passive plus active safety-related systems and is not consistent with design objectives.

1B.7.7 Steam Generator Shell-Side Heat Removal System

This SAMDA consists of providing a passive safety-related ~~grade~~ heat removal system to the secondary side of the steam generators. The system would provide closed loop cooling of the secondary using natural circulation and stored water cooling, thus preventing a loss of primary heat sink in the event of a loss of startup feedwater and passive RHR heat exchanger. A perfect secondary heat removal system would eliminate transients from each of the release categories. In order to evaluate the benefit of this SAMDA, the frequencies of all the transient sequences is subtracted from the overall frequency of each of the release categories and the risk is ~~requantified~~ recalculated. The total risk averted is 5.3×10^{-4} ~~6.7×10^{-4}~~ man-rem per year.

1B.7.8 Direct Steam Generator Relief Flow to the IRWST

This SAMDA consists of providing all the piping and valves required for redirecting the flow from the steam generator safety and relief valves to the in-containment refueling water storage tank (IRWST). An alternate, lower cost option of this SAMDA consists of redirecting only the first stage safety valve to the IRWST. This system would prevent or reduce fission product release from bypassing the containment in the event of a steam generator tube rupture (SGTR) event. In order to evaluate the benefit from this SAMDA (both options), this design change is assumed to eliminate the BP release category. ~~the frequencies of all the SGTR sequences are subtracted from the CI release category frequency and added to the OK release category frequency, and the risk is requantified.~~ The total risk averted is 4.2×10^{-4} ~~6.7×10^{-4}~~ man-rem per year.

1B.7.9 Increased Steam Generator Pressure Capability

This SAMDA consists of increasing the design pressure of the steam generator secondary side and safety valve set point to the degree that a steam generator tube rupture will not cause the secondary system safety valve to open. The design pressure would have to be increased sufficiently such that the combined heat capacity of the secondary system inventory and the PRHR system could reduce the RCS temperature below T_{sat} for the secondary design pressure. Although specific analysis would have to be performed, it is estimated that the design pressure would have to be increased several hundred psi. Like the system described in Subsection 1B.7.8, this design would also prevent the release of fission products which bypasses the containment via the SGTR. Therefore, the risk reduction is also the same as that quantified in Subsection 1B.7.8. The total risk averted is 4.2×10^{-4} ~~6.7×10^{-4}~~ man-rem per year.

1B.7.10 Secondary Containment Filtered Ventilation

This SAMDA consists of providing the middle and lower annulus (below the 135′ 3″ elevation) of the secondary concrete containment with a passive annulus filter system to for filtration of elevated releases. The passive filter system is operated by drawing a partial vacuum on the middle annulus through charcoal and HEPA filters. The partial vacuum is drawn by means of an eductor with motive flow from compressed gas tanks. The secondary containment would then reduce particulate fission product release from any failed containment penetrations (containment isolation failure). ~~the pathways from which the majority of the primary containment leakage is predicted to occur.~~ In order to evaluate the benefit from such a system, this design change is assumed to eliminate the CI release category. ~~the offsite doses from the containment leakage release categories, OK, OKP and CC, and the excessive leakage frequency contribution to the CI release category are assumed to be zero, and the risk is requantified.~~ The total risk averted is 7.4×10^{-4} ~~1.14×10^{-3}~~ man-rem per year.

1B.7.11 Diverse IRWST Injection Valves ~~Diversify the IRWST Discharge Valves~~

This SAMDA consists of changing the in-containment refueling water storage tank (IRWST) injection valve designs so that two of the four lines use diverse valves. Each of the four lines is currently isolated by a squib valve in series with a check valve. In order to provide diversity, the valves in two of the lines will be provided by a different vendor. ~~This SAMDA consists of redesigning the in-containment refueling water storage tank (IRWST) discharge valve configuration from four check valves to two check valves and two air-operated valves.~~ This change will reduce the frequency of core melt by eliminating the common cause failure of the IRWST injection. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of IRWST injection are assumed to be averted. Core damage sequences resulting from a failure of IRWST injection correspond to PRA Level 1 accident classification 3BE; thus, release category 3BE is eliminated. ~~the frequencies of all the release categories is reduced by the contribution of IRWST injection failure sequences, and the risk is requantified.~~ The total risk averted is 5.3×10^{-3} ~~8.33×10^{-3}~~ man-rem per year.

1B.7.12 Diverse Containment Recirculation Valves

This SAMDA consists of changing the containment recirculation valve designs so that two out of the four lines use diverse valves. Each of the four lines currently contains a squib valve; two of the lines contain check valves and the other two contain motor-operated valves. In order to provide diversity, the squib valves in two lines will be made diverse by supplying them from a different vendor. This change will reduce the frequency of core melt by eliminating the common cause failure of the containment recirculation. To estimate the benefit from this SAMDA, all core damage sequences resulting from a failure of containment recirculation are assumed to be averted. Core damage sequences resulting from failure of containment recirculation correspond to PRA Level 1 accident classification 3BL; thus, release category 3BL is eliminated. The total risk averted is 1.5 x 10^{-4} man-rem per year.

1B.7.13~~12~~ Ex-Vessel Core Catcher

This SAMDA consists of designing a structure in the containment cavity or using a special concrete or coating which will inhibit core-concrete interaction (CCI), even if the debris bed dries out. A perfect core catcher would prevent CCI for all cases. However, the AP600 incorporates a wet cavity design in which ex-vessel cooling is used to maintain the core debris in the vessel thus preventing ex-vessel phenomena, such as CCI. Consequently, containment failure due to CCI is not considered in detail for the AP600 Level 2 PRA. For cases in which reactor vessel flooding is failed, it is assumed that containment failure occurs due to ex-vessel steam explosion or CCI. This containment failure is assumed to be an early containment failure, CFE, (due to ex-vessel steam explosion) even though CCI and basemat meltthrough would be a late containment failure. To conservatively estimate the risk reduction of an ex-vessel core catcher, this design change is assumed to eliminate the CFE release category. The total risk averted is 6.1 x 10^{-3} man-rem per year. ~~A perfect core catcher design would prevent CCI entirely, and the benefit from the core catcher would be estimated by assuming that all of the sequences in the CC release category would all result in OK releases. Therefore, the frequency of the CC release category is added to the OK release frequency and the risk is requantified. This SAMDA results in virtually no reduction in risk since the risk from the CI release category, which dominates the plant risk is not reduced in any way by the ex-vessel core catcher. Therefore, this SAMDA is not considered further.~~

1B.7.14~~13~~ High Pressure Containment Design

This SAMDA design consists of using the massive high pressure containment design in which the design pressure of the containment is approximately 300 psi (20 bar) for the AP600 containment. The massive containment design has a passive containment cooling feature much like the AP600 containment. The high design pressure is considered only for prevention of containment failures due to severe accident phenomena such as steam explosions and hydrogen detonation. A perfect high pressure containment design would reduce the probability of containment failures, but would have no reduction of the frequency or magnitude of the release from an unisolated containment (containment isolation failure or containment bypass). To estimate the risk reduction of a high pressure containment design, this design is assumed to eliminate the CFE, CFI and CFL release categories. The total risk averted is 6.1 x 10^{-3} man-rem per year. ~~The AP600 probabilistic risk assessment concluded that the AP600 is not susceptible to containment failure due to severe accident phenomena. Since the AP600 probabilistic risk assessment predicts no overpressure containment failures, the high pressure containment design, at best, provides a risk reduction of virtually zero, and therefore will not be considered further.~~

1B.7.15~~14~~ Increase Reliability of Diverse Actuation System

This SAMDA design consists of improving the reliability of the diverse actuation system (DAS) which actuates engineered safety features and allows the operator to monitor the plant status. A perfectly reliable DAS system would reduce the frequency of the release categories by the cumulative frequencies of all sequences in which DAS failure leads to core damage. In order to evaluate the benefit from the DAS system upgrade, a Level 1 sensitivity analysis assuming perfect reliability of DAS was completed. ~~the frequency of the DAS failure are subtracted from the release category frequencies and the risk is requantified.~~ Using the results of the Level 1 DAS sensitivity, the total risk averted is determined to be 2.2 x 10^{-4} ~~7.18 x 10^{-4}~~ man-rem per year.

1B.8 Results

~~As discussed in Subsection 1B.7, four design alternatives considered for the AP600 provide no benefit for reducing residual offsite risk. These alternatives are:~~

- ~~Filtered vent~~
- ~~Locate the normal residual heat removal system inside containment~~
- ~~Ex-vessel core catcher~~
- ~~High pressure containment design.~~

The ~~remaining~~ design alternatives from Section 1B.7 are evaluated to determine their cost benefit. The results of the ~~remaining~~ severe accident mitigation design alternatives evaluation are summarized in Table 1B.8-1. The first column identifies the design alternative for which a reduction in risk was calculated. The second column is the total man-rem reduction per year for the design alternative. The third column is the capital benefit calculated based on the reduction in risk. This value represents the maximum amount of capital that could be spent in order for the design alternative to be cost beneficial. The next column is the estimated minimum capital costs for the alternative. The final column represents the net capital benefit. The net benefit is calculated by subtracting the capital cost from the capital benefit. A negative benefit is identified by the use of parentheses.

Several ~~Five~~ of the design alternatives evaluated in other SAMDA analyses are included in the current AP600 design. These design features include:

- RCS depressurization system
- Passive residual heat removal system located inside containment
- Cavity flooding system
- Passive containment cooling system

- Hydrogen igniters in a large-dry containment
- Diverse actuation system
- Canned motor RCPs
- Interfacing system with high design pressure

As the AP600 plant core damage frequency is approximately two orders of magnitude lower than for existing plants, the benefits of additional design alternatives are very small. The fifteen SAMDAs analyzed provided little or no benefit to the AP600 design. ~~Four of the SAMDAs analyzed provided no benefit at all and the others analyzed provide negligible benefits.~~

Assuming an additional design alternative was developed which provides a 100 percent reduction in overall plant risk, representing an averted risk of 7.3×10^{-3} ~~3.42×10^{-3}~~ man-rem per year, the capital benefit only amounts to $46.50 ~~$22.20.~~

Because of the small initial risk associated with the AP600, none of the severe accident mitigation design alternatives are cost beneficial.

1B.9 References

1. "AP600 Probabilistic Risk Assessment," Westinghouse Electric Corporation and ENEL, Revision 8, September 1996. ~~June 26, 1992.~~

2. "Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2," Docket Nos. 50-352/353, August 1989.

3. "Supplement to the Final Environmental Statement - Comanche Peak Steam Electric Station, Units 1 and 2," Docket Nos. 50-445/446, October 1989.

4. "System 80+ Design Alternatives Report," Docket No. 52-002, April 1992.

5. Chanin, D.I., Sprung, J.L., Ritchie, L.T., and Jow, H-N, *MELCOR Accident Consequence Code System (MACCS) User's Guide*, NURGE/CR-4691, SAND86-1562, Vol. 1, Sandia National Laboratories, U.S. Nuclear Regulatory Commission, 1990. ~~"MELCOR Accident Consequence Code System (MACCS) Users Guide," NUREG/CR-4691, SAND86-1562, Volume 1, 1990.~~

6. "Technical Assessment Guide," EPRI P-6587-L, Volume 1, Revision 6, September 1989.

7. Nuclear Energy Cost Data Base, DOE/NE-0095, U.S. Department of Energy, September 1988.

8. "SECY-91-229 - Severe Accident Mitigation Design Alternatives for Certified Standard Designs," USNRC Memorandum from Samuel J. Chilk to James M. Taylor, dated October 25, 1991.

9. Chanin, D.I., Rollstin, J., Roster, J., and Miller, L., *MACCS Version 1.5.11.1: A Maintenance Release of the Code*, NUREG/CR-6059, SAND92-2146, Sandia National Laboratories, U.S. Nuclear Regulatory Commission, October 1993.

Table 1B.4-1

SUMMARY OF ANNUAL LEVELIZED FIXED CHARGE RATE ASSUMPTIONS

Financial Factors	**Value**
Discount Rate (before tax)	10.3% (Before Tax), 9.13% (After Tax) ~~11.5%/yr~~
Inflation rate	4.1%/yr ~~5.0%/yr~~
Federal and State Income Tax Rate	40.1% ~~38.0%~~
Investment Tax Credit	0.0%
Property Taxes and Insurance	2.0%
Tax Recovery Period	15 years
Component Book Life	30 years ~~60 years~~
Total Levelized Fixed Charge Rate	**15.7%** ~~15.4 %~~

~~Table 1B.5-1~~

~~SUMMARY OF FISSION PRODUCT RELEASE FRACTIONS~~
~~24 HOURS AFTER CORE DAMAGE~~

	~~OK~~	~~OKP~~	~~CC~~	~~CI~~
~~Xe,Kr~~	~~4.2×10^{-6}~~	~~1.0×10^{-4}~~	~~6.4×10^{-5}~~	~~3.4×10^{-1}~~
~~CsI~~	~~5.6×10^{-2}~~	~~2.0×10^{-6}~~	~~7.9×10^{-2}~~	~~3.7×10^{-2}~~
~~TeO_2~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~SrO~~	~~3.2×10^{-8}~~	~~8.0×10^{-8}~~	~~4.9×10^{-8}~~	~~6.7×10^{-5}~~
~~MoO_2~~	~~5.6×10^{-2}~~	~~9.6×10^{-2}~~	~~6.5×10^{-2}~~	~~1.4×10^{-3}~~
~~CsOH~~	~~5.8×10^{-2}~~	~~2.0×10^{-6}~~	~~9.0×10^{-2}~~	~~3.7×10^{-2}~~
~~BaO~~	~~2.9×10^{-2}~~	~~6.5×10^{-7}~~	~~4.2×10^{-3}~~	~~4.8×10^{-4}~~
~~La_2O_3~~	~~2.0×10^{-8}~~	~~5.5×10^{-8}~~	~~3.1×10^{-8}~~	~~2.0×10^{-6}~~
~~CeO_2~~	~~5.9×10^{-8}~~	~~1.6×10^{-7}~~	~~1.1×10^{-7}~~	~~2.8×10^{-5}~~
~~Sb~~	~~1.0×10^{-6}~~	~~4.8×10^{-6}~~	~~1.1×10^{-6}~~	~~1.1×10^{-3}~~
~~Te_2~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~UO_2~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~Frequency~~	~~2.5×10^{-7}~~	~~5.6×10^{-8}~~	~~7.6×10^{-10}~~	~~3.0×10^{-8}~~

Table 1B.5-1

ENVIRONMENTAL RELEASE FRACTIONS AT 24 HOURS AFTER CORE DAMAGE PER RELEASE CATEGORY

Release Cat.	Environmental Release Fractions at 24 Hours After Core Damage											
	Xe, Kr	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
IC	9.2E-4	4.7E-6	0.0E0	2.9E-7	4.7E-6	4.5E-6	3.1E-6	9.1E-7	1.0E-6	8.2E-6	0.0E0	0.0E0
BP	8.6E-1	3.8E-3	0.0E0	2.4E-5	6.9E-4	3.2E-3	2.4E-4	3.9E-6	9.8E-6	6.1E-2	0.0E0	0.0E0
CI	8.4E-1	3.4E-2	0.0E0	2.1E-3	4.1E-2	3.5E-2	2.2E-2	5.1E-3	6.5E-3	6.1E-2	0.0E0	0.0E0
CFE	7.0E-1	8.3E-2	0.0E0	9.6E-4	2.7E-2	8.1E-2	9.9E-3	6.0E-4	8.8E-4	7.2E-2	0.0E0	0.0E0
CFI	6.2E-1	3.4E-3	0.0E0	5.8E-4	7.8E-3	2.9E-3	5.6E-3	1.1E-3	1.5E-3	1.0E-2	0.0E0	0.0E0
CFL	1.1E-3	1.2E-5	0.0E0	5.9E-7	1.1E-5	1.1E-5	6.1E-6	1.6E-6	1.9E-6	1.7E-5	0.0E0	0.0E0

~~Table 1B.6-1~~

~~AP600 ESTIMATED POPULATION DOSE ESTIMATES~~
~~(Effective Whole Body Equivalent Doses in Person Sieverts)~~

~~Release Catcory~~	~~Distance (Miles)~~	~~Dose (Person Sieverts)~~	
		~~Mean~~	~~Median~~
~~OK~~	~~50~~	~~6.93×10^{-2}~~	~~4.96×10^{-2}~~
~~CI~~	~~50~~	~~$1.14 \times 10^{+3}$~~	~~$7.51 \times 10^{+2}$~~
~~CC~~	~~50~~	~~9.01×10^{-2}~~	~~6.33×10^{-2}~~
~~OKP~~	~~50~~	~~1.34×10^{-1}~~	~~1.02×10^{-1}~~

~~Notes: 1. Doses are based on the 50 year committed dose for exposure during the initial 24 hours following core damage.~~

~~2. One person sievert equals 100 man rem.~~

~~Table 1B.6-2~~

~~**AP600 Base Risk (Whole Body Population Dose to a 50 Mile Radius)**~~

~~Release Category~~	~~Frequency (yr^{-1})~~	~~Mean Consequence (man-rem)~~	~~Risk (man-rem-yr^{-1})~~
~~OK~~	~~2.5×10^{-7}~~	~~6.93~~	~~1.73×10^{-6}~~
~~OKP~~	~~5.6×10^{-8}~~	~~13.4~~	~~7.50×10^{-7}~~
~~CC~~	~~7.6×10^{-10}~~	~~9.01~~	~~6.85×10^{-9}~~
~~CI~~	~~3.0×10^{-8}~~	~~114000~~	~~3.42×10^{-3}~~
		~~Total Risk~~	~~3.42×10^{-3}~~

Table 1B.6-1

AP600 BASE RISK (Whole Body Population Dose to a 50 Mile Radius)

Release Category	**Release Category Frequency (yr^{-1})**	**Mean Population Dose[1] (man-rem)**	**Risk (man-rem-yr^{-1})**
IC	1.5×10^{-7}	$3.12 \times 10^{+2}$	4.7[illegible] $\times 10^{-5}$
CFE	6.6×10^{-9}	$9.25 \times 10^{+5}$	6.13×10^{-3}
CFI	1.3×10^{-11}	$3.35 \times 10^{+5}$	4.39×10^{-6}
CFL	1.5×10^{-11}	$1.05 \times 10^{+3}$	1.59×10^{-8}
CI	3.6×10^{-10}	$2.05 \times 10^{+6}$	7.40×10^{-4}
BP	1.1×10^{-8}	$3.72 \times 10^{+4}$	4.17×10^{-4}
		Total Risk	7.34×10^{-3}

Note:

1. Doses are based on the 50 year committed dose for exposure during the initial 24 hours following core damage.

Table 1B.8-1

AP600 SAMDA RESULTS

Design Alternative	Risk Reduction (manrem/yr)	Capital Benefit ($)	Capital Cost ($)	Net Capital Benefit ($)
Upgrade CVS for Small LOCA	~~5.80×10^{-5}~~ 5.5×10^{-4}	~~<1~~ 4	~~1,460,000~~ 1,500,000	~~(1,460,000)~~ (1,500,000)
Containment Filtered Vent	1.0×10^{-3}	6	5,000,000	(5,000,000)
Self-Actuating Containment Isolation Valves	~~1.13×10^{-3}~~ 7.4×10^{-4}	~~7~~ 5	~~60,000~~ 33,000	~~(60,000)~~ (33,000)
Safety Grade Passive Containment Spray	~~3.39×10^{-3}~~ 6.9×10^{-3}	~~22~~ 44	~~3,500,000~~ 3,900,000	~~(3,500,000)~~ (3,900,000)
Non-Safety Grade Containment Spray	6.9×10^{-3}	44	415,000	(415,000)
Active High Pressure Safety Injection System	~~1.86×10^{-3}~~ 6.1×10^{-3}	~~12~~ 39	20,000,000	(20,000,000)
SG Shell Side Heat Removal	~~6.70×10^{-4}~~ 5.3×10^{-4}	~~4~~ 3	~~1,180,000~~ 1,300,000	~~(1,180,000)~~ (1,300,000)
SG Relief Flow to IRWST	~~6.70×10^{-4}~~ 4.2×10^{-4}	~~4~~ 3	~~560,000~~ 620,000	~~(560,000)~~ (620,000)
Increased SG Pressure Capability	~~6.70×10^{-4}~~ 4.2×10^{-4}	~~4~~ 3	~~2,720,000~~ 8,200,000	~~(2,720,000)~~ (8,200,000)
Secondary Containment Ventilation with Filtration	~~1.14×10^{-3}~~ 7.4×10^{-4}	~~7~~ 5	~~2,000,000~~ 2,200,000	~~(2,000,000)~~ (2,200,000)
~~Diversity IRWST Valves~~ Diverse IRWST Injection Valves	~~8.33×10^{-5}~~ 5.3×10^{-3}	~~<1~~ 34	~~300,000~~ 160,000	~~(300,000)~~ (160,000)
Diverse Containment Recirc Valves	1.5×10^{-4}	<1	150,000	(150,000)
Ex-Vessel Core Catcher	6.1×10^{-3}	39	1,660,000	(1,660,000)
High Pressure Containment Design	6.1×10^{-3}	39	50,000,000	(50,000,000)
More Reliable DAS/~~DIS~~	~~7.18×10^{-4}~~ 2.2×10^{-4}	~~5~~ 2	~~390,000~~ 470,000	~~(390,000)~~ (470,000)

Enclosure 2 to Westinghouse
Letter NSD-NRC-97-4937

January 10, 1997

3045A

Question: 100.14

Describe in detail the process used to identify and evaluate candidate SAMDAs pertinent to the AP600 design. For those SAMDAs which were included in the assessments but which were not described or discussed in Appendix 1B of the AP600 SSAR, entitled "Severe Accident Mitigation Design Alternatives," Revision 1, dated January 13, 1994, provide a description of these candidate SAMDAs, their estimated costs, and the reasons why they were not included in the discussions of Appendix 1B.

Response:

The process used to identify and evaluate candidate SAMDAs pertinent to the AP600 design included a review of SAMDAs evaluated for other plant designs, including the following:

- Limerick (Reference 100.14-1)
- Comanche Peak (Reference 100.14-2)
- System 80+ (Reference 100.14-3)

In addition, the results of the Rev. 0 AP600 PRA were reviewed to assess possible design alternatives. Of the candidate SAMDAs identified from this initial review, the ones which were not included in the SSAR were those which were already included in the AP600 design. These AP600 design features include:

- hydrogen ignition system
- reactor cavity flooding system
- reactor coolant pump seal cooling (AP600 has canned rotor pumps)
- reactor coolant system depressurization
- external reactor vessel cooling.

All other SAMDAs are discussed and evaluated in Appendix 1B of the SSAR.

References:

100.14-1 "Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2," Docket Number 50-352/353, August 1989.

100.14-2 "Supplement to the Final Environmental Statement - Comanche Peak Steam Electric Stations, Units 1 and 2," Docket Numbers 50-445/446, August 1989.

100.14-3 "System 80+ Design Alternatives Report," Docket Number 52-002, April 1992.

SSAR Revision: None.

Question: 100.15

The $20,000,000 projected cost for the addition of an active high pressure safety injection system (HPSI) appears to be excessive. Provide justification for this estimate.

Response:

The estimate for an active high pressure safety injection system (HPSI) was a designer's estimate; no detailed system analysis or cost calculations were performed. Part of the consideration of such a system was the costs associated with the requirements for making such a system safety-related and seismically qualified. Based on the risk reduction, a lower capital cost estimate would not effect the conclusion regarding risk vs cost.

SSAR Revision: None.

Question: 100.16

Provide more complete technical and cost information on the SAMDAs thus far evaluated for the AP600. The additional information should include design descriptions and definition (design feature descriptions, performance requirements, system schematics, etc.) and further details on the estimated costs for each SAMDA.

Response:

The detail requested in this RAI is beyond the scope of evaluating a system which will not be implemented in the AP600 design. Detailed system descriptions, performance requirements, and system schematics were not developed as part of the SAMDA evaluation; rather the AP600 designers were provided a description of the design alternative, and an estimate of the design revisions and cost was completed for use in the SAMDA evaluation. As can be seen in SSAR Appendix 1B, due to the low core damage frequency for the AP600, no design alternative is shown to be cost effective and thus no further effort to define system details is warranted.

SSAR Revision: None.

Question: 100.17

The design alternatives evaluated for the AP600 are stated to have been selected based in part upon design alternatives evaluated for other plant designs. References for Limerick, Comanche Peak, and CE System 80+ are cited as the other designs used for this purpose. However, no mention was made of whether plant improvements considered as part of the NRC Containment Performance Improvement (CPI) program were also included (see NUREG/CR-5567, -5575, -5630, and -5662). Please justify that the set of design alternatives considered for the AP600 include all relevant design improvements considered in these earlier evaluations.

Response:

A review of NUREGs/CR-5567, -5575, -5630, and -5662 indicate that the design alternatives considered as part of the Containment Performance Improvement (CPI) program included design changes to enable:

- RCS depressurization
- Hydrogen control
- Reactor cavity flooding
- Containment venting
- Corrective actions for ISLOCA
- Scrubbing for containment bypasses.

These design alternatives have been considered for AP600. RCS depressurization is accommodated via the automatic depressurization system. Hydrogen control is accomplished in the AP600 large dry containment with hydrogen igniters. Flooding of the reactor cavity is included in the AP600 design. Containment venting via a filtered vent is considered in the SAMDA cost benefit evaluation in the revised Appendix 1B of the AP600 SSAR (revision 11). Addressing ISLOCA via locating normal residual heat removal system inside containment is considered in the SAMDA cost benefit evaluation in the revised Appendix 1B of the AP600 SSAR. Finally, scrubbing releases from containment bypasses is considered an accident management strategy versus a design alternative; such a strategy is included in WCAP-13913, Revision 1, December 1996, Framework for AP600 Severe Accident Management Guidance.

SSAR Revision: None

Question: 100.18

Where available, provide a comparison of the AP600 cost estimates to those for similar design alternatives considered in previous analyses, including the Comanche Peak, Limerick, and Watts Bar SAMDA analyses, the NUREG-1150 studies, and pertinent SAMDA evaluations for the GE ABWR and CE System 80+ designs.

Response:

Comparisons of SAMDAs for other dissimilar plant designs is not necessary for evaluation of the cost benefit of AP600 design alternatives. The cost estimates provided for AP600 design alternatives have been intentionally biased on the low side to maximize the risk reduction benefit. As can be seen in SSAR Appendix 1B, due to the low core damage frequency for AP600, no design alternative is shown to be cost effective, even with the cost estimates minimized.

SSAR Revision: None.

Question: 100.19

Identify and discuss all risk-significant changes made to the design of the AP600 over the past few years which were based on the results of the PRA and/or consideration of SAMDA issues. In addition, specifically identify and discuss risk significant design changes and improvements made since the 1994 SAMDA submittal (Appendix 1B of SSAR). These discussions should note the risk reduction achieved by these changes, as well as their estimated costs.

Response:

No design changes have occurred since 1994 as a result of SAMDA issues.

The changes incorporated into the design since 1994 as a result of PRA are discussed below. These changes were based upon insights from the PRA analyses. The risk reduction, as measured by the reduction in the dose from a severe accident, was not a factor for these changes.

1. The squib valves in the IRWST injection lines are to be diverse from the squib valves in the IRWST recirculation lines. This change improves the ability to flood the reactor cavity should the IRWST injection fail due to a common cause failure of the squib valves. The additional cost to make the valves diverse is estimated at $160,000 per plant.

2. Two service water system (SWS) air-operated valves were changed to motor-operated valves. This was done to improve the reliability of the SWS after the PRA analyses indicated a failure of the air-operated valves could cause a failure of the system. The cost difference between the different kinds of valves is estimated to be $10,000 per plant.

3. The locations of the IRWST vents and stairways for access to lower areas of containment were changed. The hydrogen diffusion flame analysis showed a potential for creep of the containment shell at the previous vent and stairway locations. To eliminate this potential, the locations were changed. There is no significant cost differential for this change.

SSAR Revision: None.

Question: 100.20

Several of the SAMDAs identified in Appendix 1B of the AP600 SSAR, dated January 13, 1994, were discussed in qualitative terms and estimates of costs and risk reduction were not provided. Provide estimates of risk reduction and costs associated with each of these alternatives.

Response:

In the January 13, 1994 SAMDA discussion, the following design alternatives were discussed qualitatively with no risk reduction or cost estimates:

- filtered vent
- locating normal residual heat removal inside containment
- ex-vessel core catcher
- high pressure containment design.

In Revision 11 of Appendix 1B of the AP600 SSAR (February 28, 1997), the filtered vent, ex-vessel core catcher and high pressure containment design are all quantitatively evaluated with risk reduction estimates and capital cost estimates. However, the design alternative for locating normal residual heat removal inside containment continues to be discussed qualitatively since quantitative calculations for cost estimates are not warranted due to virtually no risk reduction benefit.

SSAR Revision: None.

Question: 100.21

The SAMDAs discussed in Appendix 1B were evaluated in terms of only four release categories: OK, CI, CC, and OKP. In Revision 1 of the PRA the release categories were redefined and expanded to nine categories. Update the AP600 SAMDA evaluation relative to the expanded set of release categories.

Response:

In Revision 11 of Appendix 1B of the AP600 SSAR (February 28, 1997), the risk reduction evaluation is updated to reflect the release categories presented in AP600 PRA, revision 8.

SSAR Revision: None.

Question: 100.22

NUREG/CR-5474, entitled "Assessment of Candidate Accident Management Strategies," presented several strategies for preventing core damage and for mitigating the effects of core damage. The SAMDAs identified in Appendix 1B of the AP600 SSAR did not address the area of accident management improvements. Discuss the basis for excluding accident management strategies from the SAMDAs considered for the AP600.

Response:

SAMDA are severe accident mitigation design alternatives. Mitigation of a severe accident involves the application of an accident management strategy. Thus, the SAMDAs evaluated consider accident management strategies via the calculation of risk reduction, and the discussion of how the design alternative would mitigate the severe accident. Detailed accident management strategies were not developed for any design alternatives, or existing design features, and the final detailed AP600 accident management guidance (strategies) have not been developed.

The AP600 design already includes design alternatives which enhance accident management capability, such as:

- hydrogen ignition system
- reactor cavity flooding system
- reactor coolant system depressurization.

Accident management is not part of the AP60 certified design.

SSAR Revision: None.

Question: 100.23

The SAMDA evaluation in Appendix 1B of the AP600 SSAR presented a brief evaluation of the alternative of increasing the reliability of the diverse actuation system (DAS). To what extent is this option of increasing the reliability of the DAS equivalent to making it safety grade? What consideration has been given to making the portion of the diversified actuation system that trips the reactor safety grade, and what would be the improvement in the reliability/availability of the reactor trip portion of the DAS if it were safety grade? What would be the cost of this limited scope upgrade?

Response:

Increasing the reliability of the DAS is not related to, nor is it equivalent to, making the system safety-related. The DAS will be sufficiently reliable to meet the design goals, and it will incorporate industry design and quality standards including validation and verification of the system.

As noted in SSAR Appendix 1B, if it were to be assumed that a DAS improvement "provided a 100 percent reduction in the overall plant risk, representing an averted risk of 7.3×10^{-3} man-rem per year, the capital benefit only amounts to \$46.50." An improvement of the DAS cannot provide a 100 percent reduction in the overall plant risk, so the maximum capital benefit of a revision to the DAS would be less than \$46.50.

If the reactor trip portion of the system were to be made safety-related, it would involve designing and building a safety-related system for the reactor trip function. This would have to be completely separate from the rest of the nonsafety-related DAS. That is, the reactor trip function of the current DAS design is not a separate set of wires and chips. It is one function of many performed by the DAS components. If the reactor trip function were to be safety-related, it would require the design, construction, documentation, and verification of an entirely separate system for that function. This new safety-related diverse reactor trip system would have to include the additional documentation requirements of a safety-related component, as well as the additional redundancy requirements required of such components. Many, if not all, of the components of the safety-related function, would have to be custom designed and constructed instead of using readily available materials as will occur with the current design of the DAS.

The development and construction of this separate, diverse reactor trip function for the DAS is estimated to cost more than the DAS with the current design. This cost is significantly more than the maximum capital benefit of \$46.50. The additional cost to develop the safety-related reactor trip function would result in a very small improvement in the system reliability (due to the added redundancy for the trip function), and the maximum capital benefit could not be realized.

SSAR Revision: None.

Question: 100.24

Based on information in Table 24-1 of the PRA, a significant number of penetrations would be screened out because they are 2-inches in diameter or less and may become plugged due to the particulate source term. However, in most core melt sequences in the AP600 design the core debris would be covered by an overlying water pool, resulting in less airborne fission products and a smaller likelihood of plugging. Without this screening, how much does the probability for failure to isolate containment increase? How would the risk reduction (person-rem per yr) estimates for each SAMDA change if these leak paths were included in the evaluation?

Response:

Once all containment penetrations were identified, screening criteria were used to eliminate containment penetrations which may not be important pathways for releases outside containment. One of the screening criterion, as noted above, is:

- lines penetrating containment which are 2" in diameter or smaller (small lines tend to become plugged due to particulate source term)

This screening criterion would eliminate all containment penetrations of 2" diameter or less. However, there are also other criteria for screening containment penetrations. These include:

- penetration is connected to a closed system inside containment whose integrity is not compromised during an accident,
- line penetrating containment has isolation valves and is part of a "closed system" outside containment, and is capable of withstanding severe accident conditions,
- penetration has at least one blind flange,
- penetration is administratively controlled and normally closed during power operation either by locked closed valves or power removed from valves,
- penetration has valves other than the containment isolation valves inside containment that are normally closed or automatically closed.

For those AP600 containment penetrations less than 2" diameter, all but four of these penetrations would have also been screened by one of the other criteria. For instance, the 2" demin water line is administratively locked closed, and thus may be screened via this criterion. The four penetrations which were screened solely on size are:

- 1" PSS line - Containment Air Sample Return
- 3/8" PSS line - RCS/PXS/CVS samples out
- 1" PCS line - Containment pressure instrument lines
- 1" WLS line - RCDT gas

Thus, the lines screened out by size alone are all 1" in diameter or less. Consequently, quantitatively speaking, consideration of these additional four containment penetrations in the containment isolation failure probability would increase, but less than if all lines 2" or less were included.

For consideration of risk reduction, consider that for release category IC (intact containment), a whole body population dose to a 50 mile radius is 300 man-rem. This dose is based on fission product releases from a leaking containment, with the containment leak rate corresponding to the Tech Spec limit of 0.12% containment volume/day. This limit was modeled in MAAP with a 10E-6 m^2 leak in containment. This corresponds to approximately a 1/16 inch diameter hole.

To conservatively estimate the impact of this on risk reduction, the 300 man-rem predicted by the intact containment cases are ratioed by the respective areas for the 1" bypass and nominal leakage (5.07E-4/2.95E-6 = 172); thus it is assumed that a 1" leak will result in a mean population dose of 5.2E+4 man-rem. Furthermore, assume that all intact containment cases now fall into this new category of 1" containment isolation failures; thus the 1" CI failure release category has a frequency of 1.5E-7 per year. The total risk is thus 5.2E+4 x 1.5E-7 = 7.8E-3 man-rem/year. Previously, the total risk was calculated to be 7.3E-3; with these conservative assumptions, the revised risk is estimated to increase to 1.5E-2. The capital benefit for a "super design alternative" which results in 100 percent reduction in overall plant risk thus increases from $46.50 to $96.00.

The core damage frequency for AP600 is of such small magnitude to render consideration of these four additional 1" diameter containment penetrations insignificant to design alternative evaluations.

SSAR Revision: None.

Question: 100.25

What is the projected reliability of the software used for actuating important systems for core cooling and reactor shutdown? What design criteria and software quality assurance processes, including additional testing, are used for the software to ensure it meets the reliability goal?

Response:

The software for actuating safety systems will conform to the requirements of ANSI/IEEE-ANS 7-4.3.2 (Reference 100.25-1). The quality assurance process for the software is as set forth in the Westinghouse Energy Systems ISO-9001 Quality Assurance program.

The PRA instrumentation and control models include common cause failure of software for actuation logic groups in the protection and safety monitoring system (PMS), common cause failure of software for various output logic groups in the PMS, and common cause failure of software postulated to be common to the PMS and the plant control system (PLS). There is no fixed reliability goal for software. The values included in the PRA models for the postulated common cause failure of the PMS software is 1.1E-05 per demand. The value for the common cause failure of the software common to the PMS and PLS is 1.2E-06 per demand. Both of these values are of sufficient magnitude to ensure that these events appear in the dominant core damage cutsets and risk rankings.

References:

100.25-1 ANSI/IEEE-ANS 7-4.3.2, "Standard Requirements for Digital Computers in Safety Systems of Nuclear Power Generating Systems"

SSAR Revision: None.

Question: 100.26

From the PRA, it is not clear if the shutdown PRA considered situations when the containment is open. Describe how the shutdown PRA addressed the containment being open, and discuss how consistent this model is with expected plant conditions at shutdown, especially during refueling.

Response:

The AP600 Technical Specifications require that containment integrity be maintained during plant operation in Modes 1 through 4. Containment closure capability is required by the Technical Specifications when the plant is in Modes 5 and 6. The modeling in the Shutdown PRA addressed containment status in a manner consistent with expected plant conditions at shutdown, including refueling modes. Further details concerning this topic have been previously provided in the response to RAI 720.306.

SSAR/PRA Revision: None.

Question: 100.27

The AP600 PRA includes importance measures for risk increase and for risk decrease. However, we cannot find Fussell-Vesely measures of importance, or an equivalent measure, that provides a ranking of the importance of events to the core damage frequency (CDF). Although we can infer important event failures that contribute significantly to the CDF from the dominant sequences and their cut sets, we cannot tell the rank-ordered importance of events to CDF. Please provide a rank order of events by contribution to the CDF using the Fussell-Vesely measure or an equivalent measure.

Response:

Chapter 59 of AP600 PRA (revision 8) includes risk increase and risk decrease importance measures for initiating events, for operator actions, and for hardware failures. Both of these importance measures provide rank-ordered indication of the importance of events to CDF. The risk decrease ranking provides information equivalent to that provided by Fussel-Vessely, in that it indicates the relative impact that could be obtained if the basic event in question were guaranteed not to fail.

Table 100.27-1 lists the Fussel-Vessely importance values for basic events in the CDF cutsets for the internal initiating events at power analysis.

SSAR/PRA Revision: None.

Table 100.27-1 Fussel-Vessely Importance Values

RISK IMPORTANCE CALCULATION

Number of Basic Events = 536
Number of Cutsets = 14103

BEV = Basic Event Probability
FV = Fussel Vessely Importance

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
1	ACACV028GO	171	1.750E-03	5.590E-03
2	ACACV029GO	171	1.750E-03	5.590E-03
3	ACAOR001EB	6	7.200E-07	2.277E-06
4	ACAOR001SP	123	7.270E-04	2.325E-03
5	ACATK001AF	8	2.400E-06	7.614E-06
6	ACBCV028GO	164	1.750E-03	4.633E-03
7	ACBCV029GO	164	1.750E-03	4.633E-03
8	ACBOR001EB	3	7.200E-07	1.886E-06
9	ACBOR001SP	116	7.270E-04	1.926E-03
10	ACBTK001AF	5	2.400E-06	6.312E-06
11	ACX-CV-GO	168	5.100E-05	3.192E-02
12	ACX-TK-AF	12	1.200E-07	7.491E-05
13	AD2MOD01	3	5.640E-02	1.646E-06
14	AD2MOD02	3	5.640E-02	1.646E-06
15	AD3MOD03	3	5.640E-02	1.646E-06
16	AD3MOD04	3	5.640E-02	1.646E-06
17	AD4MOD07	12	5.800E-04	4.728E-07
18	AD4MOD08	12	5.800E-04	4.728E-07
19	AD4MOD09	12	5.800E-04	4.728E-07
20	AD4MOD10	12	5.800E-04	4.728E-07
21	ADF-MAN01	57	5.000E-01	6.348E-03
22	ADN-MAN01	507	3.020E-03	1.087E-02
23	ADN-MAN01C	4	5.000E-01	1.133E-02
24	ADX-EV-SA	1985	3.000E-05	3.332E-02
25	ADX-MV-GO	77	1.100E-03	4.400E-04
26	ALL-IND-FAIL	85	1.000E-06	5.211E-05
27	ATW-MAN01	32	3.300E-02	5.969E-04
28	ATW-MAN01C	53	5.170E-01	3.265E-02
29	ATW-MAN03	167	5.200E-02	4.997E-02
30	ATW-MAN04	43	5.200E-02	4.627E-03
31	ATW-MAN04C	50	5.260E-01	4.431E-02
32	ATW-MAN05	5	5.200E-03	4.208E-03
33	ATW-MAN06	1	5.200E-03	6.680E-08
34	ATW-MAN06C	1	5.000E-01	4.109E-03
35	BSIZE	315	5.000E-01	1.269E-01
36	BSIZE-LARGE	304	5.000E-01	1.269E-01
37	CANAV014LA	1	8.760E-03	1.998E-06
38	CANCV015GC	4	2.450E-02	2.092E-06
39	CANTP011RI	121	5.230E-03	8.784E-04

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
40	CASMOD01	3	2.410E-03	3.693E-06
41	CASMOD02	11	2.310E-02	1.453E-05
42	CASMOD03	3	2.310E-02	4.582E-07
43	CAX-CM-ER	9	1.200E-04	1.041E-05
44	CCAMOD03	1	6.140E-04	1.135E-06
45	CCBMOD01	7	4.800E-02	2.461E-05
46	CCX-AV-LA	136	6.100E-05	3.947E-02
47	CCX-BC-SA	9	8.400E-06	1.533E-05
48	CCX-BL-ER	2	1.200E-05	2.618E-05
49	CCX-BY-PN	366	4.700E-05	7.314E-04
50	CCX-BY-PN1	45	5.700E-05	9.909E-06
51	CCX-EAI	1	1.270E-05	2.625E-07
52	CCX-EP-SA	34	8.620E-06	9.330E-05
53	CCX-EP-SAM	273	8.620E-06	1.391E-02
54	CCX-IN-LOGIC-SW	18	1.100E-05	7.032E-03
55	CCX-INPUT-LOGIC	70	1.030E-04	6.617E-02
56	CCX-IV-XR	130	2.400E-05	2.869E-05
57	CCX-IV-XR1	18	2.400E-05	3.541E-06
58	CCX-PL2MOD5	3	6.980E-05	4.628E-06
59	CCX-PL303	15	9.690E-05	2.423E-04
60	CCX-PL3EH0	2	4.030E-06	8.815E-06
61	CCX-PL3MOD1	24	1.410E-04	3.666E-04
62	CCX-PL3MOD1-SW	2	1.100E-05	2.401E-05
63	CCX-PL3MOD5	13	6.980E-05	7.474E-06
64	CCX-PL3MOD5-SW	1	1.100E-05	2.270E-07
65	CCX-PL403	26	9.690E-05	7.360E-05
66	CCX-PL4EH0	20	4.030E-06	2.941E-06
67	CCX-PL4MOD1	27	1.410E-04	1.071E-04
68	CCX-PL4MOD1-SW	23	1.100E-05	8.283E-06
69	CCX-PL903	6	9.690E-05	8.257E-06
70	CCX-PL9MOD1	12	1.410E-04	1.297E-05
71	CCX-PLA03	2	9.690E-05	3.340E-06
72	CCX-PLAMOD1	3	1.410E-04	5.474E-06
73	CCX-PLB03	6	9.690E-05	5.337E-06
74	CCX-PLBMOD1	10	1.410E-04	8.748E-06
75	CCX-PLD03	1	9.690E-05	2.294E-06
76	CCX-PLDMOD1	1	1.410E-04	3.340E-06
77	CCX-PLMMOD4	26	4.980E-05	3.782E-05
78	CCX-PLMMOD4-SW	23	1.100E-05	8.283E-06
79	CCX-PLMOD3	21	1.030E-04	1.968E-05
80	CCX-PLMOD3-SW	1	1.100E-05	2.270E-07
81	CCX-PLSMOD6	32	2.530E-04	5.477E-05
82	CCX-PLSMOD6-SW	1	1.100E-05	2.270E-07
83	CCX-PM-ER	2	1.400E-05	3.058E-05
84	CCX-PMA030	78	9.690E-05	8.503E-05
85	CCX-PMAEH0	20	4.030E-06	2.941E-06
86	CCX-PMAMOD1	86	1.410E-04	1.245E-04
87	CCX-PMAMOD2	4	3.040E-04	1.453E-07
88	CCX-PMAMOD4	26	4.980E-05	3.782E-05
89	CCX-PMB030	89	9.690E-05	1.832E-05
90	CCX-PMBMOD1	110	1.410E-04	2.769E-05

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
91	CCX-PMBMOD2	4	3.040E-04	1.453E-07
92	CCX-PMC030	2	9.690E-05	4.410E-06
93	CCX-PMCMOD1	2	1.410E-04	6.414E-06
94	CCX-PMCMOD2	4	3.040E-04	1.453E-07
95	CCX-PMCMOD4	1	4.980E-05	1.915E-06
96	CCX-PMD030	48	9.690E-05	1.003E-04
97	CCX-PMDEH0	20	4.030E-06	2.941E-06
98	CCX-PMDMOD1	50	1.410E-04	1.460E-04
99	CCX-PMDMOD2	4	3.040E-04	1.453E-07
100	CCX-PMDMOD4	32	4.980E-05	4.784E-05
101	CCX-PMS-HARDWARE	116	7.890E-05	2.853E-02
102	CCX-PMXMOD1-SW	321	1.100E-05	1.778E-02
103	CCX-PMXMOD2-SW	18	1.100E-05	7.032E-03
104	CCX-PMXMOD4-SW	67	1.100E-05	2.671E-04
105	CCX-SFTW	191	1.200E-06	1.485E-02
106	CCX-TRNSM	315	4.780E-04	1.248E-03
107	CCX-TT-UF	115	1.170E-04	1.428E-04
108	CCX-VS-FA	15	3.840E-05	1.461E-04
109	CCX-XMTR	284	4.780E-04	2.532E-02
110	CCX-XMTR1	1	4.780E-04	4.452E-06
111	CCX-XMTR195	103	4.780E-04	2.495E-02
112	CDNTF01BRI	36	5.230E-03	2.306E-04
113	CIAEP014SA	1	1.710E-04	3.902E-08
114	CIB-MAN00	54	1.840E-03	8.251E-03
115	CIB-MAN01	51	1.340E-03	1.554E-03
116	CIX-AV-LA	1	7.700E-04	7.153E-06
117	CMA-CV	10	2.000E-06	1.016E-05
118	CMA-PLUG	97	7.270E-04	4.408E-03
119	CMAAV014LA	10	1.590E-03	1.286E-05
120	CMAAV015LA	10	1.590E-03	1.286E-05
121	CMAOR001EB	10	7.200E-07	3.656E-06
122	CMATK002AF	10	2.400E-06	1.220E-05
123	CMB-CV	2	2.000E-06	9.022E-07
124	CMB-PLUG	23	7.270E-04	3.395E-04
125	CMBAV014LA	2	1.590E-03	1.141E-06
126	CMBAV015LA	2	1.590E-03	1.141E-06
127	CMBOR001EB	2	7.200E-07	3.252E-07
128	CMBTK002AF	2	2.400E-06	1.083E-06
129	CMX-AV-LA	43	9.600E-05	4.545E-04
130	CMX-CV-GO	89	5.100E-05	3.273E-02
131	CMX-TK-AF	7	1.200E-07	7.630E-05
132	CMX-VS-FA	116	3.840E-05	2.540E-02
133	CONDVACUUM	11	1.000E-03	5.664E-05
134	CV3EPCPASA	2	1.710E-04	1.045E-05
135	CVBPM01BTM	32	2.190E-02	3.116E-04
136	CVMOD01	29	2.210E-04	5.829E-04
137	CVMOD02	8	1.410E-03	1.135E-04
138	CVMOD03	16	1.120E-02	1.445E-04
139	CVMOD04	47	7.370E-04	1.989E-03
140	CVMOD05	26	2.880E-02	5.968E-04

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
141	CVMOD07	26	2.710E-02	5.635E-04
142	CVN-MAN00	5	3.100E-03	5.187E-03
143	CVN-MAN02	2	1.580E-03	1.466E-07
144	CVN-MAN03	2	1.070E-03	8.111E-06
145	CVNMV090GC	2	8.760E-02	1.460E-06
146	CVNMV091GC	2	8.760E-02	1.460E-06
147	CVX-PM-ER	3	3.700E-05	8.194E-05
148	DAS	360	1.000E-02	1.924E-02
149	DUMP-MAN01	4	1.320E-03	3.369E-05
150	EC0MOD01	1175	5.080E-03	1.465E-03
151	EC1BS001LF	5	4.800E-06	8.176E-08
152	EC1BS001TM	84[illegible]	2.700E-03	1.646E-03
153	EC1BS011TM	28[illegible]	2.700E-03	6.238E-04
154	EC1BS012TM	33[illegible]	2.700E-03	9.217E-04
155	EC1BS013TM	21[illegible]	2.700E-03	6.626E-04
156	EC1BS111TM	20[illegible]	2.700E-03	7.675E-05
157	EC1BS112TM	39	2.700E-03	4.243E-04
158	EC1BS121TM	201	2.700E-03	5.853E-05
159	EC1BS122TM	6	2.700E-03	3.775E-06
160	EC1BS131TM	39	2.700E-03	4.243E-04
161	EC1CB100VO	92	4.200E-03	1.453E-05
162	EC1MOD11	16	4.800E-05	3.707E-06
163	EC1MOD12	14	4.800E-05	6.824E-06
164	EC1MOD13	32	4.800E-05	4.880E-06
165	EC1REDG1GA	3	4.360E-03	8.218E-07
166	EC2BS002LF	6	4.800E-06	2.207E-07
167	EC2BS002TM	488	2.700E-03	6.185E-04
168	EC2BS021TM	56	2.700E-03	1.976E-04
169	EC2BS022TM	234	2.700E-03	2.[illegible]96E-04
170	EC2BS023TM	211	2.700E-03	3.000E-04
171	EC2BS211TM	35	2.700E-03	9.903E-05
172	EC2BS212TM	13	2.700E-03	9.840E-05
173	EC2BS221TM	208	2.700E-03	1.377E-04
174	EC2BS222TM	9	2.700E-03	4.298E-06
175	EC2BS231TM	13	2.700E-03	9.840E-05
176	EC2CB200VO	103	4.200E-03	4.174E-05
177	EC2MOD21	1	4.800E-05	7.922E-08
178	EC2MOD22	10	4.800E-05	1.472E-06
179	EC2MOD221	1	1.680E-05	4.854E-07
180	EC2MOD23	30	4.800E-05	1.321E-06
181	EC3BS003TM	1	2.700E-03	1.608E-06
182	EC4BS004TM	2	2.700E-03	1.648E-06
183	EC4BS041TM	2	2.700E-03	1.648E-06
184	EC4BS411TM	2	2.700E-03	1.648E-06
185	ECX-CB-GC	71	7.300E-04	6.625E-05
186	ECX-CB-GO	60	4.200E-04	3.784E-05
187	ED1BSDS1LF	5	4.800E-06	8.176E-08
188	ED1BSDS1TM	170	3.000E-04	8.527E-05
189	ED1MOD01	85	5.040E-04	3.560E-05
190	ED1MOD03	190	2.700E-03	3.616E-04
191	ED1MOD06	66	3.480E-04	2.740E-06

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
192	ED1MOD07	107	3.050E-04	2.240E-05
193	ED1MOD11	81	3.170E-04	5.380E-05
194	ED1MOD113	81	3.170E-04	5.380E-05
195	ED1MOD13	115	3.170E-04	2.334E-05
196	ED2BSDS1LF	5	4.800E-06	8.176E-08
197	ED2BSDS1TM	97	3.000E-04	3.245E-05
198	ED2MOD03	69	2.700E-03	2.347E-05
199	ED2MOD11	101	3.170E-04	3.428E-05
200	ED3BSDS1TM	33	3.000E-04	2.426E-05
201	ED3MOD01	83	5.040E-04	6.026E-05
202	ED3MOD03	44	2.700E-03	2.538E-05
203	ED3MOD04	89	2.190E-02	6.932E-05
204	ED3MOD07	400	3.050E-04	6.886E-03
205	ED4BSDS1TM	157	3.000E-04	2.834E-05
206	ED4MOD02	1	1.920E-04	8.218E-08
207	ED4MOD03	30	2.700E-03	5.627E-07
208	ED4MOD11	165	3.170E-04	3.267E-05
209	ED4MOD112	164	3.170E-04	3.237E-05
210	FWBMOD11A	1	3.340E-04	2.069E-06
211	FWDMOD11B	1	3.340E-04	2.069E-06
212	FWMOD010	10	1.410E-02	1.514E-05
213	FWMOD013A	32	1.410E-02	3.268E-05
214	FWMOD013B	57	1.410E-02	3.823E-05
215	FWMOD028	160	1.410E-02	7.617E-04
216	FWMOD03A	36	1.700E-02	3.993E-05
217	FWMOD03B	64	1.700E-02	4.710E-05
218	FWMOD067A	36	1.410E-02	5.969E-04
219	FWMOD067B	10	1.410E-02	1.152E-05
220	FWNCV029CO	8	2.190E-04	2.548E-06
221	FWX-MV2-GO	15	5.500E-04	3.547E-05
222	FWX-PM2-FS	15	5.400E-04	3.480E-05
223	HPM-MAN01	1	5.020E-04	2.578E-07
224	IDABSDD1LF	20	4.800E-06	3.505E-06
225	IDABSDD1TM	67	3.000E-04	2.991E-04
226	IDABSDK1LF	20	4.800E-06	3.505E-06
227	IDABSDK1TM	55	3.000E-04	2.975E-04
228	IDABSDS1LF	20	4.800E-06	3.505E-06
229	IDABSDS1TM	73	3.000E-04	2.994E-04
230	IDAFD003RQ	23	1.200E-05	9.030E-06
231	IDAFD004RQ	23	1.200E-05	9.030E-06
232	IDAMOD04	119	3.170E-04	1.967E-05
233	IDAMOD05	74	5.160E-04	3.666E-06
234	IDAMOD06	9	4.320E-05	7.467E-08
235	IDAMOD07	26	2.190E-02	4.027E-06
236	IDAMOD08	44	3.170E-04	1.061E-06
237	IDBBSDD1LF	5	4.800E-06	1.383E-05
238	IDBBSDD1TM	158	3.000E-04	1.000E-03
239	IDBBSDK1TM	12	3.000E-04	1.676E-06
240	IDBBSDS1LF	5	4.800E-06	1.383E-05
241	IDBBSDS1TM	164	3.000E-04	1.001E-03

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
242	IDBFD013RQ	26	1.200E-05	3.765E-05
243	IDBMOD24	18	3.170E-04	1.674E-05
244	IDBMOD25	6	5.160E-04	5.321E-07
245	IDBMOD27	26	2.190E-02	4.027E-06
246	IDCBSDD1LF	3	4.800E-06	2.252E-08
247	IDCBSDD1TM	203	3.000E-04	1.133E-04
248	IDCBSDS1LF	3	4.800E-06	2.252E-08
249	IDCBSDS1TM	209	3.000E-04	1.136E-04
250	IDCFD007RQ	27	1.200E-05	3.129E-06
251	IDCMOD28	18	3.170E-04	1.674E-05
252	IDCMOD29	6	5.160E-04	5.321E-07
253	IDCMOD31	26	2.190E-02	4.027E-06
254	IDDBSDD1LF	25	4.800E-06	1.734E-05
255	IDDBSDD1TM	169	3.000E-04	1.289E-03
256	IDDBSDK1LF	20	4.800E-06	3.505E-06
257	IDDBSDK1TM	81	3.000E-04	3.175E-04
258	IDDBSDS1LF	25	4.800E-06	1.734E-05
259	IDDBSDS1TM	175	3.000E-04	1.289E-03
260	IDDFD019RQ	33	1.200E-05	4.470E-05
261	IDDFD020RQ	23	1.200E-05	9.030E-06
262	IDDMOD32	119	3.170E-04	1.967E-05
263	IDDMOD33	74	5.160E-04	3.666E-06
264	IDDMOD34	9	4.320E-05	7.467E-08
265	IDDMOD35	26	2.190E-02	4.027E-06
266	IDDMOD38	44	3.170E-04	1.061E-06
267	IEV-ATW-S	91	2.050E-02	2.256E-03
268	IEV-ATW-T	13	1.170E+00	4.210E-03
269	IEV-ATWS	230	4.810E-01	5.309E-02
270	IEV-CMTLB	1404	8.940E-05	2.093E-02
271	IEV-ISLOC	1	5.000E-11	2.956E-04
272	IEV-LCAS	174	3.480E-02	1.024E-03
273	IEV-LCCW	244	1.440E-01	7.252E-04
274	IEV-LCCND	316	1.120E-01	6.112E-03
275	IEV-LLOCA	642	1.050E-04	2.967E-01
276	IEV-LMFW	253	3.350E-01	1.790E-03
277	IEV-LMFW1	150	1.920E-01	1.040E-03
278	IEV-LOSP	694	1.200E-01	5.956E-03
279	IEV-LRCS	34	1.800E-02	7.508E-05
280	IEV-MLOCA	1713	1.620E-04	3.683E-02
281	IEV-NLOCA	3383	7.700E-04	1.863E-01
282	IEV-POWEX	391	4.500E-03	1.084E-02
283	IEV-PRSTR	330	2.500E-04	3.298E-03
284	IEV-RCSLK	802	1.200E-02	1.338E-02
285	IEV-RV-RP	1	1.000E-08	5.912E-02
286	IEV-SGTR	507	5.200E-03	3.597E-02
287	IEV-SI-LB	287	1.040E-04	2.258E-01
288	IEV-SLB-D	26	5.960E-04	5.595E-05
289	IEV-SLB-U	117	3.720E-04	7.275E-04
290	IEV-SLB-V	197	1.210E-03	2.347E-03
291	IEV-SLOCA	1657	1.010E-04	2.396E-02
292	IEV-TRANS	446	1.400E+00	6.736E-03

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
293	IRBEP117BSA	43	1.710E-04	8.858E-07
294	IRBEP123ASA	16	1.710E-04	1.157E-05
295	IRBEP123BSA	16	1.710E-04	1.157E-05
296	IRDEP118BSA	43	1.710E-04	8.858E-07
297	IRWMOD01	103	1.200E-02	1.239E-04
298	IRWMOD03	243	1.200E-02	1.960E-04
299	IRWMOD05	56	1.460E-03	4.241E-03
300	IRWMOD06	54	1.460E-03	4.239E-03
301	IRWMOD07	40	1.460E-03	1.507E-06
302	IRWMOD08	42	1.460E-03	2.861E-06
303	IRWMOD09	33	1.460E-03	1.229E-05
304	IRWMOD10	161	1.460E-03	7.289E-05
305	IRWMOD11	141	1.460E-03	2.123E-05
306	IRWMOD12	289	1.460E-03	9.370E-05
307	IWA-PLUG	229	2.400E-04	1.479E-01
308	IWACV122AO	62	1.750E-03	5.080E-03
309	IWACV124AO	60	1.750E-03	5.079E-03
310	IWARS118BFA	112	8.760E-04	1.171E-05
311	IWARS123BFA	17	8.760E-04	7.133E-07
312	IWB-PLUG	226	2.400E-04	6.706E-05
313	IWBCV122AO	45	1.750E-03	1.839E-06
314	IWBCV124AO	45	1.750E-03	3.448E-06
315	IWBRS118AFA	20	8.760E-04	6.410E-06
316	IWBRS123AFA	31	8.760E-04	2.538E-03
317	IWCRS120BFA	233	8.760E-04	5.313E-05
318	IWCRS125BFA	17	8.760E-04	7.133E-07
319	IWDRS120AFA	122	8.760E-04	4.080E-05
320	IWDRS125AFA	31	8.760E-04	2.538E-03
321	IWNTK001AF	45	2.400E-06	1.275E-05
322	IWX-CV-AO	1958	3.000E-05	5.179E-02
323	IWX-CV1-AO	1	5.400E-07	3.322E-04
324	IWX-EV-SA	1819	2.600E-05	4.481E-02
325	IWX-EV1-SA	1	1.000E-05	6.148E-03
326	IWX-EV3-SA	23	1.000E-05	7.525E-06
327	IWX-EV4-SA	969	2.600E-05	2.206E-01
328	IWX-FL-GP	1258	1.200E-05	1.319E-02
329	IWX-XMTR	454	4.780E-04	4.125E-02
330	LPM-MAN01	132	1.340E-03	1.044E-03
331	LPM-MAN02	312	3.300E-03	8.427E-03
332	MDAS	751	1.000E-02	1.278E-02
333	MSAEPSD1SA	3	1.710E-04	6.958E-06
334	MSAEPSD2SA	3	1.710E-04	6.958E-06
335	MSAEPSD3SA	3	1.710E-04	6.958E-06
336	MSAEPSD4SA	3	1.710E-04	6.958E-06
337	MSAEPSD5SA	3	1.710E-04	6.958E-06
338	MSAEPSD6SA	3	1.710E-04	6.958E-06
339	MSAEPSD7SA	3	1.710E-04	6.958E-06
340	MSAEPSD8SA	3	1.710E-04	6.958E-06
341	MSHTP001RI	30	5.230E-03	7.494E-03
342	MSHTP002RI	30	5.230E-03	7.494E-03

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
343	MSMODV001	2	2.710E-02	3.453E-07
344	MSMODV003	2	2.710E-02	3.453E-07
345	MSMODV005	2	2.710E-02	3.453E-07
346	MSMODV007	2	2.710E-02	3.453E-07
347	MSX-AV-FA	15	1.500E-03	[illegible]
348	OTH-BL	11	1.900E-01	1.440E-04
349	OTH-MGSET	37	1.750E-03	4.599E-03
350	OTH-PO	1	1.200E-04	1.117E-06
351	OTH-PRES	52	2.000E-03	4.927E-04
352	OTH-PRESU	139	3.270E-01	3.331E-02
353	OTH-PRSOV	392	1.000E-02	1.082E-02
354	OTH-R05	694	7.000E-01	5.956E-03
355	OTH-SDMAN	44	7.700E-04	2.038E-03
356	OTH-SGTR	512	1.000E-02	1.468E-02
357	OTH-SGTR1	30	6.700E-03	5.413E-04
358	OTH-SLSOV	167	1.100E-02	1.076E-03
359	OTH-SLSOV1	282	2.100E-02	1.029E-02
360	OTH-SLSOV2	45	1.000E-02	7.906E-04
361	OTH-SLSOV3	109	5.400E-03	5.222E-04
362	PCNHR001ML	45	2.400E-06	1.275E-05
363	PL20301ASA	2	1.160E-03	1.155E-05
364	PL20301BSA	2	1.160E-03	1.155E-05
365	PL2MOD11	4	2.090E-03	2.464E-05
366	PL2MOD52	2	8.740E-04	5.297E-07
367	PL30301ASA	5	1.160E-03	2.428E-05
368	PL30301BSA	4	1.160E-03	2.180E-05
369	PL30302ASA	2	1.160E-03	6.952E-06
370	PL30302BSA	1	1.160E-03	4.469E-06
371	PL3MOD11	11	2.090E-03	5.677E-05
372	PL3MOD12	4	2.090E-03	1.846E-05
373	PL40301ASA	135	1.160E-03	1.019E-05
374	PL40301BSA	112	1.160E-03	9.169E-06
375	PL40302ASA	66	1.160E-03	3.270E-06
376	PL40302BSA	43	1.160E-03	2.252E-06
377	PL4EH0A1SA	32	8.000E-05	5.181E-07
378	PL4EH0A2SA	11	8.000E-05	1.563E-07
379	PL4MOD11	172	2.090E-03	2.025E-05
380	PL4MOD12	102	2.090E-03	7.809E-06
381	PL4XS00ASA	37	8.000E-05	6.017E-07
382	PL5MOD11	2	2.090E-03	2.755E-07
383	PL70302ASA	2	1.160E-03	7.035E-07
384	PL70302BSA	2	1.160E-03	7.035E-07
385	PL7MOD12	2	2.090E-03	1.266E-06
386	PL90302ASA	2	1.160E-03	7.035E-07
387	PL90302BSA	2	1.160E-03	7.035E-07
388	PL9MOD12	3	2.090E-03	1.905E-06
389	PLAMOD12	1	2.090E-03	6.385E-07
390	PLMMOD41	100	6.350E-04	3.733E-05
391	PLMMOD42	23	6.350E-04	4.663E-07
392	PLSMOD61	1	3.460E-03	2.471E-07
393	PLSMOD62	4	3.460E-03	2.293E-06

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
394	PMA0301ASA	135	1.160E-03	1.016E-05
395	PMA0301BSA	112	1.160E-03	9.147E-06
396	PMA0302ASA	56	1.160E-03	3.051E-06
397	PMA0302BSA	33	1.160E-03	2.033E-06
398	PMAEH0A1SA	32	8.000E-05	5.181E-07
399	PMAEH0A2SA	11	8.000E-05	1.563E-07
400	PMAMOD11	175	2.090E-03	2.024E-05
401	PMAMOD12	86	2.090E-03	7.361E-06
402	PMAMOD31	73	5.020E-03	2.740E-04
403	PMAMOD41	86	6.350E-04	4.221E-06
404	PMAMOD42	18	6.350E-04	3.694E-07
405	PMAXS00ASA	37	8.000E-05	6.017E-07
406	PMBMOD11	9	2.090E-03	6.811E-07
407	PMBMOD32	73	5.020E-03	2.740E-04
408	PMCMOD33	60	5.020E-03	2.697E-04
409	PMD0301ASA	135	1.160E-03	1.016E-05
410	PMD0301BSA	112	1.160E-03	9.147E-06
411	PMD0302ASA	56	1.160E-03	3.051E-06
412	PMD0302BSA	33	1.160E-03	2.033E-06
413	PMDEH0A1SA	32	8.000E-05	5.181E-07
414	PMDEH0A2SA	11	8.000E-05	1.563E-07
415	PMDMOD11	186	2.090E-03	2.120E-05
416	PMDMOD12	86	2.090E-03	7.361E-06
417	PMDMOD34	60	5.020E-03	2.697E-04
418	PMDMOD41	86	6.350E-04	4.221E-06
419	PMDMOD42	18	6.350E-04	3.694E-07
420	PMDXS00ASA	37	8.000E-05	6.017E-07
421	PMS-RTSWITCH	1	3.000E-05	1.271E-07
422	PRAAV108LA	21	1.090E-03	5.341E-06
423	PRAAV108TM	14	5.000E-04	2.007E-06
424	PRAMOD10	11	2.110E-03	1.072E-04
425	PRAMOD9	42	1.410E-02	8.055E-04
426	PRBAV108LA	21	1.090E-03	5.341E-06
427	PRBAV108TM	14	5.000E-04	2.007E-06
428	PRBMOD10	11	2.110E-03	1.072E-04
429	PRCEP101SA	2	1.710E-04	7.768E-06
430	PRCEP108SA	2	1.710E-04	7.768E-06
431	PRDEP108SA	2	1.710E-04	7.768E-06
432	PRI-MAN01	2	4.960E-04	2.254E-05
433	PXX-AV-LA	1220	9.600E-05	1.120E-03
434	RC1CB051GO	107	4.200E-03	4.112E-04
435	RC1CB052GO	107	4.200E-03	4.112E-04
436	RC1CB053GO	107	4.200E-03	4.112E-04
437	RC1CB054GO	107	4.200E-03	4.112E-04
438	RC1CB061GO	107	4.200E-03	4.112E-04
439	RC1CB062GO	107	4.200E-03	4.112E-04
440	RC1CB063GO	107	4.200E-03	4.112E-04
441	RC1CB064GO	107	4.200E-03	4.112E-04
442	RCX-RB-FA	161	8.100E-06	5.373E-03
443	REA-PLUG	166	2.400E-04	6.006E-04

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
444	REACV119GO	163	1.750E-03	8.747E-05
445	REAMOV117TM	18	5.000E-04	3.560E-06
446	REB-PLUG	207	2.400E-04	7.895E-04
447	REBCV119GO	296	1.750E-03	1.126E-04
448	REBMOV117TM	96	5.000E-04	6.517E-06
449	REC-MANDAS	288	1.160E-02	1.297E-02
450	REC-MANDASC	429	5.060E-01	3.161E-02
451	REG-MAN00	321	2.040E-01	9.911E-04
452	REN-MAN04	393	1.000E-02	4.044E-02
453	REX-FL-GP	1258	1.200E-05	1.319E-02
454	RHN-MAN01	124	2.900E-03	2.417E-03
455	RHN-MAN01C	11	5.000E-01	9.260E-04
456	RHN-MAN06	246	3.750E-03	5.629E-05
457	RN11MOD3	372	1.410E-02	1.199E-02
458	RN22MOD4	372	1.410E-02	1.199E-02
459	RN23MOD5	372	1.410E-02	1.199E-02
460	RNAEP01ASA	54	1.710E-04	4.[illegible]08E-06
461	RNAEP01BSA	57	1.710E-04	4.929E-06
462	RNAEP022SA	27	1.710E-04	1.298E-04
463	RNAMOD06	418	3.400E-02	1.133E-03
464	RNAMOD09	130	5.070E-02	2.145E-03
465	RNBEP011SA	27	1.710E-04	1.298E-04
466	RNBMOD07	444	3.400E-02	1.236E-03
467	RNBMOD10	130	5.070E-02	2.145E-03
468	RNDEP023SA	27	1.710E-04	1.298E-04
469	RNNCV013GO	101	1.750E-03	1.450E-03
470	RNX-CV-GO	26	5.100E-05	3.876E-05
471	RNX-KV-GO	60	6.100E-04	4.894E-04
472	RNX-KV1-GO	1[illegible]	4.900E-03	4.110E-03
473	RNX-PM-ER	23	1.600E-05	1.205E-05
474	RNX-PM-FS	65	7.700E-04	6.202E-04
475	ROD-CTRL-SYS	24	6.600E-04	5.152E-05
476	RPTMOD01	52	8.760E-04	8.138E-05
477	RPTMOD02	52	8.760E-04	8.138E-05
478	RPTMOD03	52	8.760E-04	8.138E-05
479	RPTMOD04	52	8.760E-04	8.138E-05
480	RPTMOD05	52	8.760E-04	8.138E-05
481	RPTMOD06	52	8.760E-04	8.138E-05
482	RPTMOD07	52	8.760E-04	8.138E-05
483	RPTMOD08	52	8.760E-04	8.138E-05
484	RPX-CB-GO	201	4.200E-04	2.136E-02
485	SFBEP028SA	6	1.710E-04	1.830E-06
486	SFNMV067GC	2	1.100E-02	2.093E-06
487	SG1TF51ARI	6	5.230E-03	1.964E-06
488	SG2TF50ARI	8	5.230E-03	4.[illegible]3E-05
489	SGAAV040LA	2	1.090E-03	1.[illegible]E-06
490	SGAOR--DAS-SP	14	7.220E-03	1.2[illegible]0E-04
491	SGATL--DAS-UF	11	5.230E-03	8.714E-05
492	SGBAV040LA	51	1.090E-03	1.24[illegible]-03
493	SGBAV074LA	11	8.760E-03	8.136E-05
494	SGBAV075LA	11	8.760E-03	8.136E-05

	BASIC EVENT ID	CUTSETS	BEV PROB.	FV
495	SGBOR--DAS-SP	14	7.220E-03	1.220E-04
496	SGBTL--DAS-UF	11	5.230E-03	8.714E-05
497	SGX-MV-GO	14	5.500E-04	3.310E-05
498	SWAMOD03	2	6.340E-04	1.525E-06
499	SWAMOD09T	16	2.520E-04	2.613E-05
500	SWB-001TM	7	3.800E-02	1.953E-05
501	SWBMOD02	3	2.440E-02	7.207E-06
502	SWBMOD11P	1	1.410E-02	1.638E-07
503	SWN-MAN03	20	4.000E-02	4.492E-04
504	TCBMOD01B	1	2.520E-02	1.608E-06
505	VF1AV004	1	8.760E-03	8.572E-08
506	VFIAV010	1	8.760E-03	8.572E-08
507	VFOAV003	1	8.760E-03	8.572E-08
508	VFOAV009	1	8.760E-03	8.572E-08
509	VFSFRAC	2	1.200E-01	1.714E-07
510	VWAMOD01	6	2.520E-04	2.693E-05
511	VWAMOD02	12	6.120E-04	8.366E-05
512	VWAMOD03	6	2.520E-04	2.693E-05
513	VWBMOD04	74	1.830E-02	7.961E-04
514	VWBMOD05	92	2.190E-02	9.794E-04
515	VWBMOD06	16	[illegible]80E-03	1.960E-04
516	VWN-MAN01	16	[illegible]60E-03	1.955E-04
517	VWX-RF-ER	2	1.200E-05	2.618E-05
518	WLIAV004LA	1	8.760E-03	7.153E-07
519	WLIAV055LA	1	8.760E-03	7.153E-07
520	WLOAV006LA	1	8.760E-03	7.153E-07
521	WLOAV057LA	1	8.760E-03	7.153E-07
522	ZANMOD01	45	8.400E-05	3.570E-06
523	ZANTR-2AHF	5	2.880E-05	4.280E-08
524	ZANTR-2BHF	2	2.880E-05	1.230E-08
525	ZO1DG001TM	561	4.600E-02	4.689E-04
526	ZO1MOD01	298	2.020E-02	1.509E-04
527	ZO1MOD04	23	1.250E-03	1.242E-06
528	ZO2DG002TM	512	4.600E-02	6.808E-04
529	ZO2MOD01	284	2.020E-02	2.711E-04
530	ZO2MOD03	2	1.000E-04	2.974E-07
531	ZO2MOD04	30	1.250E-03	8.848E-06
532	ZOX-BL-ES	9	6.000E-05	3.119E-07
533	ZOX-DG-DR	60	4.400E-04	3.966E-05
534	ZOX-DG-DS	42	2.800E-04	[illegible].535E-05
535	ZOX-PD-ER	24	1.300E-04	4.414E-06
536	ZOX-PD-ES	118	2.000E-03	1.932E-04

CALCULATED CDF = 1.69E-07

Question: 100.28

Please provide Chapters 55 (Se[illegible] Evaluations) and 57 (Fire Evaluations) of the AP600 PRA.

Response:

Chapter 57, Internal Fire Analysis, has been provided with AP600 PRA Revision 8 (September 1996).

Chapter 55, Seismic Margins Analysis, was previously provided with the response to RAI 720.158. It is being revised to address other RAIs. The updated version will be provided in early 1997.

SSAR/PRA Revision: None.

Question: 100.29

Identify the most important structures, systems, and components (SSCs) relied upon to prevent and mitigate core damage during and following seismic events. Discuss the basis for establishing the importance of these SSCs.

Response:

The requested information can be found is the response to RAI 720.158. The seismic margin analysis is being revised and will be included as Chapter 55 of the AP600 PRA in early 1997.

SSAR/PRA Revision: None.

Question: 100.30

Provide an assessment of major contributors to risk from external events, and design alternatives considered and/or implemented by Westinghouse to reduce risk from each of these contributors.

Response:

Chapters 57 and 55 of the PRA discuss the internal fire analysis and seismic margins analysis, respectively. Chapter 58 of the PRA discusses other external events and the probability of an accident leading to severe consequences due to an external event.

The AP600 SSAR (Chapter 2) also discusses the ability of the plant to withstand events such as high wind, seismic events and external floods.

No design alternatives were considered or implemented on AP600 to reduce risk from external events as a result of a SAMDA evaluation.

SSAR/PRA Revision: None.

Question: 100.31

The AP600 design suggests several SAMDAs not yet considered that might prove cost effective. Evaluate and discuss the following possible candidate design alternatives.

- Increased regulatory oversight of the most risk-significant non-safety SSCs
- Improving the instrumentation and controls (quality of components, quality/maturity index of software)
- Use of fan coolers (FCs) to remove fission products, and possibly upgrading the FCs and support systems to improve reliability
- Addition of a non-safety grade in-containment spray system
- Increasing the thickness of the reactor cavity concrete to reduce the likelihood of containment failure by cavity melt-through.

Response:

Increased regulatory oversight of the most risk-significant nonsafety SSCs

There is a program in place entitled Regulatory Treatment of Nonsafety-related Systems (RTNSS) which Westinghouse is implementing on AP600 with the NRC staff. This program evaluates the AP600 on a deterministic and probabilistic basis to determine if any further regulatory oversights is needed.

The cost associated with increased regulatory oversight is believed to be significant due to the associated administrative burdens ultimately placed on the nonsafety-related SSCs. As shown by the risk reduction of the previous design alternatives, due to the low core damage frequency of the AP600 design, severe accident mitigation design alternatives have very low risk reductions. Coupled with the significant cost, this option is not a viable SAMDA.

Improving the instrumentation and controls (quality of components, quality/maturity index of software)

The AP600 instrumentation and controls is believed to be of sufficient quality as is evident by the low failure probability of the I&C components. As shown by the risk reduction of previous design alternatives, due to the low core damage frequency of the AP600 design, severe accident mitigation design alternatives have very low risk reductions. Coupled with the increased costs, this option is not a vaiable SAMDA.

Use of fan coolers (FCs) to remove fission products, and possibly upgrading the FCs and support systems to improve reliability

Containment fan coolers are included in the AP600 design. Finalization of the AP600 severe accident management guidance should include the use of the fan coolers in accident management strategies to remove containment energy, control fission products, and control hydrogen. These points are included in WCAP-13913, Revision 1, December 1996, "Framework for AP600 Severe Accident Management Guidance."

As seen with the evaluation of other design alternatives, upgrading the fan coolers design will result in an insignificant risk reduction (versus cost) due to the low core damage frequency of the AP600 design.

Addition of a non-safety grade in-containment spray system

Evaluation of a nonsafety-related in-containment spray system is included in the revised Appendix 1B of the AP600 SSAR (revision 11, February 28 1997). As shown by the risk reduction presented in SSAR Appendix 1B, due to the low core damage frequency of the AP600 design, a severe accident mitigation design alternative such as a nonsafety-related spray system has very low risk reductions. Coupled with the large capital cost, this option is not a viable SAMDA.

Increasing the thickness of the reactor cavity concrete to reduce the likelihood of containment failure by cavity melt-through

Since the reactor cavity flooding system provides a means to preclude CCI by maintaining the core debris in the reactor vessel, there is little risk reduction to be gained for further design changes to address CCI. Additionally, changes to the reactor cavity would result in a large cost. As with other alternatives, this is not a viable SAMDA.

SSAR/PRA Revision: None.