APPENDIX 1B

SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

1B.1 AP1000 SAMDA Evaluation

1B.1.1 Introduction

This response provides an evaluation of Severe Accident Mitigation Design Alternatives (SAMDA) for the Westinghouse AP1000 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.

The 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 U.S. Nuclear Regulatory Commission (NRC) staff recommends that SAMDAs 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 1).

1B.1.2 Summary

Note that the AP1000 is similar to the AP600, which has received Design Certification. The evaluation for AP1000 uses the conclusions of the AP600 SAMDA investigation as described below. 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. Since the AP1000 is so similar to the AP600, the list of candidate modifications is the same.

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 and AP1000 designs. These include the following:

- 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 3). The AP1000 probabilistic risk results are similar to those developed for the AP600. Fifteen candidate design alternatives were selected for further evaluation.

An evaluation 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, 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, for the AP600 and AP1000, that despite the significant conservatism used in the evaluation, none of the SAMDAs evaluated provide risk reductions that 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 and AP1000 designs.

1B.1.3 Selection and Description of SAMDAs

Candidate design alternatives were selected based upon design alternatives evaluated for other plant designs (References 4, 5, and 6) as well as suggestions from AP600 and AP1000 design personnel. Additional candidate design alternatives were selected based upon an assessment of the AP600 and AP1000 probabilistic risk assessment results. SAMDA design alternatives were finally selected for further evaluation. These SAMDAs are as follows:

- Chemical, volume, and control system (CVS) upgraded to mitigate small loss-of-coolant accidents (LOCAs)

- Filtered containment vent

- Normal residual heat removal system (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 of steam generator secondary side pressure capacity
- Secondary containment filtered ventilation
- Diverse IRWST injection valves
- Diverse containment recirculation valves
- Ex-vessel core catcher
- High-pressure containment design
- Diverse actuation system improved reliability.

Each SAMDA and the benefit expected due to the modification is described below. 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. For these evaluations, increases to release category IC are not factored into the risk benefit calculations. The IC dose is sufficiently small 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.

The cost benefit methodology of NUREG/BR-0184 (1997) is used to calculate the maximum attainable benefit. This includes replacement power costs. For expected benefit, the change in the CDF frequency (delta-F) is assumed to be equal to the sum of CDF frequencies from internal, external, and shutdown events that are already evaluated. This is bounding, used to calculate the maximum attainable benefit. In practice, there is no design alternative, or SAMDA strategy, whose implementation would reduce the plant CDF to zero (or to an infinitesimally small frequency).

Upgrade Chemical, Volume, and Control System for Small LOCAs

The chemical, volume, and control system is currently capable of maintaining the reactor coolant system inventory to a level in which the core remains covered in the event of a very small (< 3/8-inch diameter break) LOCA. This SAMDA involves providing IRWST containment recirculation connections to the chemical, volume, and control system and adding a second line from the chemical, volume, and control system makeup pumps to the reactor coolant system to be able to use the system to keep the core covered during small and intermediate LOCAs.

A perfect, upgraded chemical, volume, and control system is assumed to prevent core damage in the reactor coolant system leak, passive residual heat removal heat exchanger tube ruptures, small LOCA, and intermediate LOCA release categories. The chemical, volume, and control system is assumed to have perfect support systems (power supply and component cooling) and to work in all situations regardless of the common cause failures of other systems.

Filtered Vent

This SAMDA consists of placing a filtered containment vent and all associated piping and penetrations into the AP1000 containment design. The filtered vent could be used to vent the containment to prevent catastrophic overpressure failure, and it also provides filtering capability for source term release. With respect to the AP1000 Probabilistic Risk Assessment, 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 and steam explosion. The late containment failures for AP1000 are failures of the passive containment cooling system. Analyses have indicated that for scenarios with passive containment cooling system failure, air cooling may limit the containment pressure to less than the ultimate pressure. However, for the Level 2 probabilistic risk assessment, failure of the passive containment cooling system 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 probabilistic risk assessment 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 percent of the noble gas fission products will still be released. Therefore, the risk reduction is equal to the decontamination factor assumed since the probabilistic risk assessment Level 1 accident classification frequencies do not change.

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 normal residual heat removal system 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. Conceptually, the design would be either an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve. This provides the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures that occur at the time of the accident, such as in cases of vessel rupture or anticipated transients without scram.

Passive Containment Sprays

This SAMDA involves adding a passive safety-related spray system and all associated piping and support systems to the AP1000 containment. A passive containment spray system could result in risk benefits in the following ways:

- Scrubbing of fission products could be done 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 passive containment cooling system 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.

Active High-Pressure Safety Injection System

This SAMDA consists of adding a safety-related active high-pressure safety injection pump and all associated piping and support systems to the AP1000 design. A perfect high-pressure safety injection system is assumed to prevent core melt for all events but excessive LOCA and anticipated transients without scram. Therefore, to estimate the risk reduction, only the contributions to each release category of Level 1 accident classes 3C (vessel rupture) and 3A (anticipated transients without scram) need to be considered. 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 it is not consistent with design objectives.

Steam Generator Shell-Side Heat Removal System

This SAMDA consists of providing a passive safety-related 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. This prevents a loss of primary heat sink in the event of a loss of startup feedwater and passive residual heat removal 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 are subtracted from the overall frequency of each of the release categories and the risk is recalculated.

Direct Steam Generator Relief Flow to the In-containment Refueling Water Storage Tank

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 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 event. In order to evaluate the benefit from this SAMDA (both options), this design change is assumed to eliminate the BP release category.

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 passive residual heat removal system could reduce the reactor coolant system 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. This design would also prevent the release of fission products that bypass the containment via the steam generator tube rupture.

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 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). In order to evaluate the benefit from such a system, this design change is assumed to eliminate the CI release category.

Diverse In-containment Refueling Water Storage Tank Injection Valves

This SAMDA consists of changing the 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. For the check valves, alternate vendors are available. However, it is questionable if check valves of different vendors would be sufficiently different to be considered diverse unless the type of check valve was changed from the current swing disk check to another type. The swing disk type is the preferred type for this application and other types are considered to be less reliable. Squib valves are specialized valve designs for which there are few vendors. A vendor may not be willing to design, qualify, and build a reasonable squib valve design for this AP1000 application considering that they would only supply two valves per plant. As a result, this SAMDA is not really practicable because of the uncertainty in availability of a second squib valve design/vendor and because of the uncertainty in the reliability of another check valve type. However, the cost estimate for this SAMDA assumes that a second squib valve vendor exists and that the vendor provides only the two diverse IRWST squib valves. The cost impact does not include the additional first time engineering and qualification testing that will be incurred by the second vendor. Those costs are expected to be more than a million dollars.

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 probabilistic risk assessment Level 1 accident classification 3BE; thus, release category 3BE is eliminated.

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. 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 probabilistic risk assessment Level 1 accident classification 3BL; thus, release category 3BL is eliminated.

In the AP1000 design for recirculation, valve diversity has been introduced to reduce some of the dominant failure modes that were discovered for the AP600.

The four AP600 recirculation squib valves were of the “low-pressure” type and were a part of a single common cause group. In the AP1000, two of these valves that are in series with check valves are designated to be of “high-pressure” type, which are in a common cause group with the same design of valves on the IRWST injection lines. Thus, the common cause failure mode that fails all four recirculation lines in the AP600 is eliminated, and it is replaced with the product of two common cause failure modes, one applicable to the group of six high-pressure squib valves and the other to the two low-pressure squib valves. This design change helps in reduction of recirculation failures.

Ex-Vessel Core Catcher

This SAMDA consists of designing a structure in the containment cavity or using a special concrete or coating that 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 AP1000 incorporates a wet cavity design in which ex-vessel cooling is used to maintain the core debris in the vessel to prevent ex-vessel phenomena, such as CCI. Consequently, containment failure due to CCI is not considered in detail for the AP1000 Level 2 probabilistic risk assessment. 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 melt-through 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.

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 AP1000 containment. The massive containment design has a passive containment cooling feature much like the AP1000 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.

Increase Reliability of Diverse Actuation System

This SAMDA design consists of improving the reliability of the diverse actuation system, which actuates engineered safety features and allows the operator to monitor the plant status. The design change would add a third instrumentation and control cabinet and a third set of diverse actuation system instruments to allow the use of two-out-of-three logic instead of two-out-of-two logic. Other changes, such as adding another set of batteries, have not been included in the cost estimates. A perfectly reliable diverse actuation system would reduce the frequency of the release categories by the cumulative frequencies of all sequences in which diverse actuation system failure leads to core damage. In order to evaluate the benefit from the diverse actuation system upgrade, a Level 1 sensitivity analysis assuming perfect reliability of diverse actuation system was completed.

Locate Normal Residual Heat Removal Inside Containment

This SAMDA consists of placing the entire normal residual heat removal system and piping inside the containment pressure boundary. Locating the normal residual heat removal system inside the containment would prevent containment bypass due to interfacing system LOCAs (ISLOCA) of the residual heat removal 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 residual heat removal system from the high pressure reactor coolant system causes the residual heat removal system to overpressurize and fail, releasing reactor coolant system 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 AP1000, the normal residual heat removal system is designed with a higher design pressure than the 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 (CDF) of the AP1000 (Reference 2, Chapter 33). Therefore, relocating the normal residual heat removal system of the AP1000 inside containment will provide virtually no risk reduction benefit and will not be investigated further in terms of cost.

1B.1.4 Methodology

The severe accident mitigation design alternatives analysis uses a bounding methodology such that the benefit is conservatively maximized and the capital cost is conservatively minimized for each SAMDA.

1B.1.4.1 Total Population Dose

To assess the potential benefits associated with a design alternative, estimates are made of the offsite population doses resulting from each of the release categories (that is, source terms). MACCS2 version 1.12 (Reference 9) is used for the analysis. The NRC sponsored the development of this code. The code performs probabilistic estimates of offsite consequences from potential accidental releases in conformance with Chapter 9 of the probabilistic risk assessment guidelines described in NUREG/CR-2300 (Reference 10).

Doses are determined for the early exposure effects resulting from the initial 24 hours following the core damage initiation. The dose evaluation provides the conditional probability distributions for the consequence measures, which includes the whole-body dose for this analysis. These consequence probability distributions are based on the assumption that the accident that produced the source term has occurred. Therefore, the consequence probability distributions presented result from the variation in dose levels due to the various meteorological conditions. Hence, the actual probability of the identified dose levels would be the probability of the release category that produced the source term occurring multiplied by the probability of the dose level.

The dose risks are quantified by multiplying the calculated fission product release category frequency vector by the release category mean dose vectors. The frequencies for each of the six release categories are quantified in Chapter 45 of the AP1000 Probabilistic Risk Assessment (Reference 2), while the mean doses for each release category are identified in Chapter 49. Table 1B-1 presents the results of the dose risk calculations at the site boundary at 24 hours. The table presents the release category identifier, the release frequency (per reactor-year), the mean dose (in rem), and the resulting risk (in rem per reactor-year). In addition, each table presents the total dose risk and the percent that each release category contributes to the total risk. The information from Table 1B-1 was extracted from Chapter 49 of the AP1000 Probabilistic Risk Assessment.

It is shown that release category CFE presents the largest risk to the site safety.

The release categories for the AP1000 are defined as follows:

- IC – intact containment. Containment integrity is maintained throughout the accident, and the release of radiation to the environment is due to nominal leakage.

- CFE – containment failure early. Fission-product release through a containment failure caused by severe accident phenomenon occurring after the onset of core damage but prior to core relocation.

- CFI – containment failure intermediate. Fission-product release through a containment failure caused by severe accident phenomenon occurring after core relocation but before 24 hours.

- CFL – containment failure late. Fission-product release through a containment failure caused by severe accident phenomenon occurring after 24 hours.

- CI – containment isolation failure. Fission-product release through a failure of the system or valves that close the penetrations between the containment and the environment. Containment failure occurs prior to onset of core damage.

- BP – containment bypass. Fission products are released directly from the Reactor Coolant System to the environment via the secondary system or other interfacing system bypass. Containment failure occurs prior to onset of core damage.

The following subsections present a brief description of the AP1000 release categories.

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.

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.

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.

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 probabilistic risk assessment 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.

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.

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.

1B.1.4.2 AP1000 Risk (CDF, LRF, and POPULATION Dose)

Table 1B-2 presents a summary of the CDF and large release frequency (LRF) risks for the AP1000.

Level 3 analysis is performed only for internal events at power. The ensuing population dose was very low, and it was not pursued for other events. The population dose for internal events is given in Table 1B-3. The information from Table 1B-3 was extracted from Chapter 49 of the AP1000 Probabilistic Risk Assessment.

1B.1.5 Summary of Risk Significant Enhancements

This section summarizes the design enhancements already incorporated into the AP1000 plant due to probabilistic risk assessment insights and results.

- Changed normal position of the two containment motor-operated recirculation valves (in series with squib valves) from closed to open

  The normal position of the two motor-operated valve lines in the two sump recirculation lines has been changed from NORMALLY CLOSED to NORMALLY OPEN to improve the reliability of opening these paths. These two paths support containment recirculation for core cooling and IRWST draining for IVR. This change reduced the CDF and LRF contribution from the failure modes to open the motor-operated valves.

- Changed IRWST drain procedure so it occurs earlier for IVR support

  Credit is taken for operator action to drain the IRWST into the sump to preserve reactor vessel integrity following core melt. The procedure for this severe accident response has been modified so that the operator action associated with IRWST draining is moved to the beginning of the procedure to allow more time for operator success and also to fill the cavity as soon as possible. This improves the probability of success of the operator action.

- Improved IVR heat transfer

  In going from the AP600 to the AP1000, the heat loads during IVR are increased due to the larger core power level, which reduced the margins in the heat removal capability through the reactor vessel head during IVR. To compensate for the increase in core power, the critical heat flux limit on the outside of the reactor vessel has been increased by changes made to the flow path between the outside of the reactor vessel and the reactor vessel insulation. Testing has confirmed the robustness of the IVR heat transfer.

- Improved IRWST vents

  The larger core in the AP1000 can generate more hydrogen in a severe accident. In the AP1000 hydrogen analysis for Level II, it was observed that the standing hydrogen diffusion flames at the IRWST vents resulted in a larger thermal loads to the containment steel shell, potentially leading to containment wall failure. The design of the vents was changed so that the IRWST vents located well away from the containment would open and the IRWST vents located next to the containment would not open during a severe accident to eliminate or minimize this potential concern.

- Incorporated low boron core (anticipated transients without scram)

  In the AP600, anticipated transients without scram (ATWS) contribution to LRF was noticed to be high relative to other initiating events. A low boron core was incorporated into the design to reduce the potential contribution of ATWS to plant risk.

- Added 3rd passive containment cooling drain valve (motor-operator valve diverse to air-operated valve)

  Due to reduced containment surface area per MW of core power, natural air circulation without passive containment cooling system water drain may not always be sufficient for long-term (greater than 1 day) containment heat removal in the AP1000. For the AP600, it was always sufficient for an indefinite time. To reduce the uncertainty in whether air cooling is sufficient to provide adequate long-term containment heat removal, a third path was added to the passive containment cooling system drain lines to increase passive containment cooling system reliability. The isolation valve used in the third path is a motor-operated valve, which is diverse from the air-operated valves used in the other two lines. This provides considerable improvement in the passive containment cooling system water drain reliability.

- Reduced potential recirculation-line squib valve failures

  An examination of AP1000 plant CDF cutsets revealed that the common cause failure of 4/4 recirculation line squib valves is a dominant contributor to CDF and LRF. This failure mode can be reduced by re-aligning the diverse squib valves already used in the AP1000 (and AP600) IRWST injection paths (high-pressure valves) and the containment recirculation paths (low-pressure valves). By making the recirculation squib valves two sets of two low-pressure and high-pressure squib valves, which are different and belong to different common cause failure groups. This design change reduces the common cause failure contribution of the recirculation squib valves. The increase in the group size of the high-pressure squib valves from four to six (including the four from the IRWST injection lines) does not add an appreciable contribution to the plant CDF.

1B.1.6 Specific Site Characteristics

AP1000 Probabilistic Risk Assessment Chapter 49, "Offsite Dose Risk Quantification," is based on an Electric Power Research Institute (EPRI) report (Reference 11) to establish the specific site characteristics for AP1000. Reference 11 Annex B, "ALWR Reference Site," establishes a conservative reference site to represent the consequences of most potential sites with respect to exposure at the site boundary. This reference site was based on the characteristics of 91 U.S. reactor sites that are tabulated in the NRC document, "Technical Guidance for Siting Criteria Development," (NUREG CR-2239) (Reference 12). Annex B provides a summary of the meteorological data to be used in calculating offsite dose.

1B.1.7 Value of Eliminating Risk

The cost benefit methodology of NUREG/BR-0184 (1997) is used to calculate the maximum attainable benefit. This includes replacement power costs. The maximum improvement change in the CDF frequency (delta-F) is assumed to be equal to the sum of CDF frequencies from internal, external, and shutdown events that are already evaluated:

delta F = 5 E-07/year

This is bounding and is used to calculate the maximum attainable benefit. In practice, there is no design alternative, or SAMDA strategy, whose implementation would reduce the plant CDF to zero (or to an infinitesimally small frequency).

PRA Table 49-10, Revision 4, is used to calculate the expected value of the person-rem exposure:

Dose = 179,000 person-rem (0.0432 / 2.41E-07, from Table 49-10)

It is assumed that this dose is applicable to all events (internal, external, at-power, and shutdown). Thus, the consequences (dose and other) from all events are included in the calculations. Uncertainty in this dose is analyzed in sensitivity case 2 given below.

The following cost categories are investigated (NUREG/BR-0184 notation is used):

C1	Public Health (Accident)		5.7.1	5.7.1.3	W(pha)
C2	Public Health (Routine)		5.7.2	5.7.2	V(phr)
C3	Occupational Health (Accident)	Sum of C4 and C5	5.7.3	5.7.3	V(oha)
C4		Accident Related Exposure - ID		5.7.3.3	W(io)
C5		LT Doses		5.7.3.3	W(lto)
C6	Occupational Health (Routine)		5.7.4	5.7.4	V(ohr)
C7	Offsite Property		5.7.5	5.7.5	V(fp)
C8	Onsite Property	Sum of C9, C10, and C11	5.7.6	5.7.6	V(op)
C9		Cleanup and Decon		5.7.6.1	U(cd)
C10		LT Replacement Power		5.7.6.2	U(rp)
C11		Repair and Refurbishment		5.7.6.3	

The present-dollar value equivalent for severe accidents at one unit of the AP1000 is the sum of the offsite exposure costs, offsite economic costs, onsite exposure costs, and onsite economic costs. The present-day value (at 7-percent discount rate) of eliminating all plant CDF (maximum attainable benefit) is calculated to be $21,000, which is a very small dollar value. Thus, any mitigating system or a SAMDA strategy/alternative that reduces the plant risk by a fraction of the total plant CDF must cost less than $21,000 to be cost-effective.

Another calculation of the maximum attainable benefit is made with the discount rate of 3 percent (Table 7-2). The resulting value is $43,000, which is still very small to justify any appreciable investment.

Even if a very conservative multiplicative error factor of 10 were used, the maximum attainable benefit would be limited to a cost below $207,000.

Table 1B-4 summarizes the results of the base case and the sensitivity cases.

In all cases, the values are strongly affected (increased) because of the replacement power cost. This is an inappropriate bias for public decision making, since it does not relate to public safety and it is not a direct cost to the public since the costs are to the utility, and their impact on the electricity rates for the public is unpredictable.

The first sensitivity case is already discussed above. In the second sensitivity case, the dose values are increased (10 times for external, NUREG high-estimates for occupational health). The third sensitivity analysis acknowledges that the delta-F realistically cannot be equal to the total plant CDF; a factor of 0.5 is introduced.

Sensitivity case 4 examines the case where the CDF value (thus the delta-F) is increased by a factor of 2. Finally, sensitivity case 5 looks at what happens if a multiplicative error factor of 10 is applied to the base case. In all cases, the benefits range from very small to modest.

1B.1.8 Evaluation of Potential Improvements

The value of eliminating AP1000 total risk is $21,000, as discussed in Section 1B.1.7. This value is an upper bound for any single engineered design alternative, which would actually reduce CDF and/or LRF a fraction of the values assumed in the base case for calculating the $21,000 value.

For the AP1000, SAMDA design alternatives discussed in this section are found to be not cost effective. One of these alternatives is actually implemented in the AP1000 design (diverse containment recirculation squib valves) to help improve the success likelihood of cavity reflooding operator action in severe accidents. The costs associated with the remaining SAMDA design alternatives are provided in Table 1B-5. Only one design alternative, 3 – namely, self-actuating containment isolation valves – has a cost near $30,000; the remaining alternatives are at least an order of magnitude more costly than $30,000. Thus, only design alternative 3 needs to be further discussed.

1B.1.8.1 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 normal residual heat removal system 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. Conceptually, the design would either be an independent valve or an appendage to an existing fail-closed valve that would respond to post-accident containment conditions within containment. For example, a fusible link would melt in response to elevated ambient temperatures resulting in venting the air operator of a fail-closed valve. This provides the self-actuating function. To evaluate the benefit of this SAMDA, this design change is assumed to eliminate the CI release category. This does not include induced containment failures, which occur at the time of the accident such as in cases of vessel rupture or ATWS. This design alternative provides almost no benefit in reducing plant CDF.

Generously assuming that this design alternative will eliminate CI release totally and that Delta CDF is zero, the benefit of this design alternative is calculated to be at the order of a few thousand dollars. Thus, even the cheapest design alternative does not meet the benefit/cost ratio of 1.

1B.1.8.2 Other New Design Changes

Other design changes, as discussed in Section 1B.1.5, are already incorporated into the AP1000. There is no cost/benefit analysis available for those changes already incorporated.

Two additional design changes not incorporated in the AP1000 were assessed as follows:

Larger Accumulators

Increasing the size of the accumulators would result in a significant increase in cost that would be greater than the cost threshold established by the perfect SAMDA evaluation. In order to have any benefit in the probabilistic risk assessment, the accumulators would have to be increased in size sufficiently to change the large LOCA success criteria from two of two accumulators to one of two accumulators. Westinghouse estimates that the accumulator tanks would have to be increased in size from 2000 ft^3 to 4000 ft^3, and the hardware costs associated with this change would be significant. Such a size increase would also likely result in a change to the design of the DVI piping subsystem. The design of this piping system was established in the AP600 design certification, and the design does not change significantly for AP1000. Recently, Westinghouse completed the leak-before break analysis of the DVI piping, and any change in the DVI piping would result in significant piping reanalysis of the DVI piping. Westinghouse estimates the redesign costs associated with the changes in hardware and piping re-design to be significantly greater than the cost threshold established for the perfect SAMDA discussed above. Therefore this design change was not incorporated.

Larger Fourth-Stage ADS Valves

Increasing the fourth-stage ADS valves in size would result in a significant increase in cost associated with redesigning the AP1000 loop piping and fourth-stage piping configuration. The AP1000 ADS valves were already increased in size compared to the AP600 valves more than the ratio of the power uprate of the AP1000. In order to have any benefit in the probabilistic risk assessment, the 4th stage ADS valves would have to be increased in size sufficiently to change the LOCA success criteria from three of four valves to two of four valves. To accommodate such a change, Westinghouse estimates that the fourth-stage ADS valves would have to increase in size from 14-inch to 18-inch valves and associated piping. In addition, the common fourth-stage inlet piping that connects to the hot leg would have to increase in size from 18-inch to at least 20-inch. This would require a significant redesign of the squib valve and would also result in redesign of the ADS-4 piping which in turn would impact the design of the reactor coolant loop piping.

Finally, such a redesign would require Westinghouse to perform additional confirmatory testing of the passive core cooling system to verify that the behavior of the passive safety systems was not adversely impacted. Westinghouse estimates the cost of this change to be significantly larger than the cost threshold of the perfect SAMDA discussed above. Therefore, this design change was not incorporated.

1B.1.9 Results

Due to the existing low risk of the AP1000 plant, none of the design alternatives described in Section 1B.1.3 meets an acceptable benefit to cost ratio of 1 or greater.

Several of the design alternatives evaluated in other SAMDA analyses are included in the current AP1000 design. These design features include the following:

- Reactor coolant system 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
- Sealless motor reactor coolant pumps
- Interfacing system with high design pressure

As the AP1000 plant CDF is lower than for existing plants, the benefits of additional design alternatives are small. The SAMDAs analyzed provided little or no benefit to the AP1000 design.

Assuming a hypothetical design alternative was developed which provides a 100-percent reduction in overall plant risk, representing an average averted risk of 4.32×10^{-2} man-rem per year, the capital benefit amounts to only $21,000.

1B.2 References

1. "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.

2. "AP1000 Probabilistic Risk Assessment," APP-GW-GL-022, Revision 5, Westinghouse Electric Company, December 2003.

3. "AP600 Probabilistic Risk Assessment," Westinghouse Electric Corporation and ENEL, Revision 8, September 1996.

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

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

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

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

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

9. Chanin, D., Young, M. L., "Code Manual for MACCS2, User's Guide," NUREG/CR-6613, SAND97-0594, Vol. 1, Sandia National Laboratories, U.S. Nuclear Regulatory Commission.

10. "PRA Procedures Guide," NUREG/CR-2300, U.S. Nuclear Regulatory Commission, Vol. 2, Chapter 9, Washington, D.C.

11. EPRI Advanced Light Water Reactor Utility Requirements Document Volume III Annex B "ALWR Reference Site," Revisions 5 & 6, December 1993.

12. NRC NUREG/CR-2239 "Technical Guidance for Siting Criteria Development," prepared by Sandia National Laboratories, D.C. Aldrich, et al., December 1982.

Table 1B-1

POPULATION WHOLE BODY EDE DOSE RISK – 24 HOURS

Release Category	Release Frequency (per reactor year)	Mean Dose (person-sieverts)	Dose (person-REM)	Risk (person-REM per reactor year)	Percentage Contribution to Total Risk
CFI	1.89E-10	7.03E+03	7.03E+05	1.33E-04	0.3
CFE	7.47E-09	8.51E+03	8.51E+05	6.36E-03	14.7
IC	2.21E-07	7.19E+00	7.19E+02	1.59E-04	0.4
BP	1.05E-08	3.23E+04	3.23E+06	3.39E-02	78.4
CI	1.33E-09	2.01E+04	2.01E+06	2.67E-03	6.2
CFL	3.45E-13	7.37E+01	7.37E+03	2.54E-09	0.0
			Total Risk =	4.32E-02	100.0

Table 1B-2

SUMMARY OF AP1000 PRA RESULTS (CDF AND LRF)

Events	Core Damage Frequency (per year)		Large Release Frequency (per year)	
	At-Power	**Shutdown**	**At-Power**	**Shutdown**
Internal Events	2.41E-07	1.23E-07	1.95E-08	2.05E-08
Internal Flood	8.82E-10	3.22E-09	7.14E-11	5.37E-10
Internal Fire	5.61E-08	8.5E-08	4.54E-09	1.43E-08
Sum =	2.97E-07	2.11E-07	2.41E-08	3.53E-08

Note:
For seismic risk, the seismic margins method is used. CDF and LRF are not quantified.

Table 1B-3

POPULATION WHOLE BODY DOSE (EFFECTIVE DOSE EQUIVALENT [EDE]), 0-80.5 KM PERSON-SIEVERTS

24-Hour Case Source Term	Quantiles						Peak Consequence
	Mean	50th	90th	95th	99th	99.5th	
CFI	7.03E+03	5.33E+03	1.31E+04	1.82E+04	3.11E+04	3.59E+04	5.07E+04
CFE	8.51E+03	6.25E+03	1.62E+04	2.31E+04	4.13E+04	5.06E+04	6.40E+04
DIRECT	2.16E+01	1.20E+01	4.78E+01	8.13E+01	1.14E+02	1.23E+02	1.68E+02
IC	7.19E+00	4.21E+00	1.71E+01	2.95E+01	3.56E+01	3.84E+01	5.60E+01
BP	3.23E+04	2.10E+04	6.40E+04	1.03E+05	1.54E+05	1.82E+05	2.64E+05
CI	2.01E+04	1.13E+04	4.71E+04	6.60E+04	1.23E+05	1.48E+05	1.61E+05
CFL	7.37E+01	1.00E+01	1.62E+02	5.91E+02	9.76E+02	1.11E+03	2.56E+03
72-Hour Case Source Term	**Quantiles**						**Peak Consequence**
	Mean	**50th**	**90th**	**95th**	**99th**	**99.5th**	
CFI	1.13E+04	9.02E+03	2.12E+04	2.63E+04	4.09E+04	4.89E+04	6.18E+04
CFE	9.36E+03	6.89E+03	1.898E+04	2.54E+04	4.25E+04	5.12E+04	6.77E+04
DIRECT	2.36E+01	1.35E+01	5.28E+01	8.32E+01	1.15E+02	1.25E+02	1.75E+02
IC	7.87E+00	4.75E+00	1.85E+01	3.00E+01	3.79E+01	4.20E+01	5.83E+01
BP	4.17E+04	2.94E+04	7.99E+04	1.16E+05	2.20E+05	2.61E+05	2.87E+05
CI	2.14E+04	1.25E+04	4.90E+04	7.40E+04	1.27E+05	1.53E+05	1.67E+05
CFL	4.79E+04	3.11E+04	9.57E+04	1.57E+05	2.62E+05	3.01E+05	4.14E+05

Table 1B-4

COST BENEFIT CALCULATION RESULTS FOR DIFFERENT ASSUMPTIONS

	Case Studied	Benefit of Case
Base Case	7% Discount rate	21,000
SC-1	3% Discount rate	43,000
SC-2	High dose (10 times the base case)	36,000
SC-3	Realistic delta-F (SAMDA reduces CDF by 50% of total)	10,000
SC-4	Twice the base CDF	41,000
SC-5	10 times the benefit of base case	207,000

Table 1B-5

DESIGN ALTERNATIVES FOR SAMDA

No.	Design Alternative	Cost
1	Upgrade chemical, volume, and control system for small LOCA	1,500,000
2	Containment filtered vent	5,000,000
3	Self-actuating containment isolation valves	33,000
4	Safety grade passive containment spray	3,900,000
6	Steam generator shell-side heat removal	1,300,000
7	Steam generator relief flow to IRWST	620,000
8	Increased steam generator pressure capability	8,200,000
9	Secondary containment ventilation with filtration	2,200,000
10	Diverse IRWST injection valves	570,000
11	Diverse containment recirculation valves	Already Implemented
12	Ex-vessel core catcher	1,660,000
13	High-pressure containment design	50,000,000
14	More reliable diverse actuation system	470,000