ATTACHMENT 2

ET-NRC-92-3784
NSRA-APSL-92-0265

AP600

SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVE EVALUATION

DECEMBER 15, 1992

APPENDIX 1B SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

1B.1.0 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 date October 25, 1991 (Reference 8).

1B.2.0 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 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 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.3.0 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 design alternatives were finally selected for further evaluation. These fourteen SAMDAs are:

1. Chemical volume and control system (CVCS) upgraded to mitigate small LOCAs
2. Filtered containment vent
3. Normal residual heat removal system (RHR) located inside containment
4. Self-actuating containment isolation valves
5. Passive containment spray
6. Active high pressure safety injection system
7. Steam generator shell side passive heat removal system
8. Steam generator safety valve flow directed to in-containment refueling water storage tank (IRWST)
9. Increase steam generator secondary side pressure capacity
10. Secondary containment filtered ventilation
11. IRWST discharge valve diversification
12. Ex-vessel core catcher
13. High pressure containment design
14. Diverse actuation system (DAS) improved reliability.

A description of each design alternative evaluated for AP600 is presented in Section 7.0.

Several design alternatives addressed in previous SAMDA evaluations for other plants were excluded from further evaluation because the alternatives are already incorporated 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.4.0 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 section.

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, 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 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 years and a tax life of 15 years, the annual levelized fixed charge rate is 15.4 percent in current U.S. dollars (with inflation).

1B.5.0 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 section.

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 or anticipated transient without scram, leading to core damage
- The primary system pressure at the time of core damage (high or low)
- Timing of core damage (early or late)
- Containment integrity at the time of core damage (intact 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). The containment event tree analysis considers both the containment and associated auxiliary systems. In particular, the following items are considered:

- 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 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:

- 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 sections present a brief description of the accident sequences from the probabilistic risk assessment which represent each AP600 release category.

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 2.5×10^{-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 5.6×10^{-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 7.7×10^{-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.6.0 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 Section 5.0. The MELCOR Accident Consequence Code System (MACCS), Version 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 and median doses in person-sieverts (1 person-sievert equals 100 man-rems) for each release category. Table 1B.6-2 shows the 50-mile population dose risk for each release category as well as the total risk of 3.42 x 10^{-3} man-rem per year for the AP600 plant.

1B.7.0 SAMDA DESCRIPTION AND BENEFIT

This section 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% 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.

1B.7.1 Upgrade the CVCS for Small LOCAs

The chemical, volume, and control system (CVCS) 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 ($\leq$ 3/4" diameter break) loss of coolant accident (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 ($\leq$ 4" diameter breaks) LOCA accidents, as well.

A perfect, upgraded CVCS system is assumed to prevent core damage in all the very small and small LOCAs in each release category. The CVCS 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 5.80 x 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. 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% of the noble gas fission products and approximately 1.0×10^{-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 Inside Containment

This SAMDA consists of placing the entire normal residual heat removal (RHR) system and piping inside the containment pressure boundary. Locating the normal RHR inside the containment would prevent containment bypass due to interfacing system LOCAs (ISLOCA) of the 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 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, Table 7-1). Therefore, relocating the 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 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 indicative of a severe accident.

To evaluate the benefit of this SAMDA, 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 1.13×10^{-3} man-rem per year.

1B.7.5 Passive Containment Sprays

This SAMDA involves adding a passive safety grade spray system and all associated piping and support systems to the AP600 containment. 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 3.39×10^{-3} man-rem per year.

1B.7.6 Active High Pressure Safety Injection System

This SAMDA consists of adding a safety 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 transients and small medium and large LOCAs in each release category. Only excessive LOCA and ATWS are assumed to lead to core damage. Therefore, 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 1.86×10^{-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 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 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. The total risk averted is 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), 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 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 increase 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 section 6.8, 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 section 6.8. The total risk averted is 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 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, 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 1.14×10^{-3} man-rem per year.

1B.7.11 Diversify the IRWST Discharge Valves

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, 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 8.33×10^{-5} man-rem per year.

1B.7.12 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 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.7.13 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. 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.7.14 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, the frequency of the DAS failure are subtracted from the release category frequencies and the risk is requantified. The total risk averted is 7.18×10^{-4} man-rem per year.

1B.8.0 RESULTS

As discussed in Section 7.0, 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 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.

Five of the design alternatives evaluated in other SAMDA analyses are included in the current AP600 design. As the AP600 plant core damage frequency is approximately two orders of magnitude lower than for existing plants, the benefits of design alternatives are very small. 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 3.42×10^{-3} man-rem per year, the capital benefit only amounts to $22.20.

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

1B.9.0 REFERENCES

1. "Simplified Passive Advanced Light Water Reactor Plant Program - AP600 Probabilistic Risk Assessment," Westinghouse Electric Corporation and ENEL, DE-AC03-90SF18495, 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. "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.

TABLE 1B.4-1

SUMMARY OF ANNUAL LEVELIZED FIXED CHARGE RATE ASSUMPTIONS

Type of Security	Value
Discount Rate (before tax)	11.5%/yr
Inflation rate	5.0%/yr
Federal and State Income Tax Rate	38.0%
Investment Tax Credit	0.0%
Property Taxes and Insurance	2.0%
Tax Recovery Period	15 years
Comp[illegible] Book Life	60 years
Total Levelized Fixed Charge Rate	**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^{-3}	1.0×10^{-4}	6.4×10^{-3}	3.4×10^{-1}
CsI	5.6×10^{-7}	2.0×10^{-6}	7.9×10^{-7}	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^{-7}	9.6×10^{-7}	6.5×10^{-7}	1.4×10^{-3}
CsOH	5.8×10^{-7}	2.0×10^{-6}	9.0×10^{-7}	3.7×10^{-2}
BaO	2.9×10^{-7}	6.5×10^{-7}	4.2×10^{-7}	4.8×10^{-4}
La_2O_3	2.0×10^{-8}	5.5×10^{-8}	3.1×10^{-8}	2.0×10^{-5}
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}

AP600

TABLE 1B-[illegible]

AP600 ESTIMATED POPULATION DOSE ESTIMATES
(EDEWBODY DOSES IN PERSON-SIEVERTS)

Release Category	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^{-3}	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.8-1

AP600 SAMDA RESULTS

Design Alternative	Risk Reduction (man-rem per yr)	Capital Benefit ($)	Capital Cost ($)	Net Capital Benefit ($)
Upgrade CVCS for Small LOCA	5.80×10^{-5}	< 1	1,460,000	(1,460,000)
Seif-Actuating Containment Isolation Valves	1.13×10^{-3}	7	60,000	(60,000)
Passive Containment Spray	3.39×10^{-3}	22	3,500,000	(3,500,000)
Active High Pressure Safety Injection System	1.86×10^{-3}	12	20,000,000	(20,000,000)
SG Shell Side Heat Removal	6.70×10^{-4}	4	1,180,000	(1,180,000)
SG Relief Flow to IRWST	6.70×10^{-4}	4	560,000	(560,000)
Increased SG Pressure Capability	6.70×10^{-4}	4	2,720,000	(2,720,000)
Secondary Containment Ventilation with Filtration	1.14×10^{-3}	7	2,000,000	(2,000,000)
Diversity IRWST Valves	8.33×10^{-5}	< 1	300,000	(300,000)
More Reliable DAS/DIS	7.18×10^{-4}	5	390,000	(390,000)