# TECHNICAL SUPPORT DOCUMENT FOR AMENDMENTS TO 10 CFR PART 51 CONSIDERING SEVERE ACCIDENTS UNDER NEPA FOR PLANTS OF SYSTEM 80+ DESIGN (Rev.2)

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TECHNICAL SUPPORT DOCUMENT
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10 CFR PART 51
CONSIDERING SEVERE ACCIDENTS UNDER NEPA
FOR PLANTS OF SYSTEM 80+ DESIGN

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## EXECUTIVE SUMMARY

The term "severe accident" refers to those events which are "beyond the substantial coverage of design basis events" and includes those for which there is substantial damage to the reactor core whether or not there are serious offsite consequences[1].

The Nuclear Regulatory Commission (NRC), in satisfaction of its severe accident safety requirements and guidance, is requiring for new reactor designs, such as the System 80+ Standard Design, among other things, the evaluation of Design Alternatives (DAs) to reduce the radiological risk from a severe accident by preventing substantial core damage (i.e., preventing a severe accident) or by limiting releases from the containment in the event that substantial core damage occurs (i.e., mitigating the impacts of a severe accident).

The National Environmental Policy Act (NEPA) requires the consideration of reasonable alternatives to proposed major Federal actions significantly affecting the quality of the human environment, including alternatives to mitigate the impacts of the proposed action. In 1989, a Federal Court of Appeals determined that NEPA required consideration of certain Design Alternatives; namely, Severe Accident Mitigation Design Alternatives (SAMDAs). (Limerick Ecology Action v. NRC, 869 F.2d 719, 3rd Cir. 1989). The court indicated that "[SAMDAs] are, as the name suggests, possible plant design modifications that are intended not to prevent an accident, but to lessen the severity of the impact of an accident should one occur." Id. at 731. The court rejected the use of a policy statement as an acceptable basis for closing out NEPA consideration of SAMDAs in a licensing proceeding, because, among other things, it was not a rulemaking. Id. at 739.

Recently, the NRC staff expanded the concept of SAMDAs to encompass Design Alternatives to prevent severe accidents, as well as mitigate them[2]. By doing so, the staff makes the set of SAMDAs considered under NEPA the same as the set of alternatives to prevent or mitigate severe accidents considered in satisfaction of the Commission's severe accident requirements and policy.

# 1.0 INTRODUCTION

## 1.1 Background

The term "severe accident" refers to those events that are "beyond the substantial coverage of design basis events" and includes those for which there is substantial damage to the reactor core whether or not there are serious offsite consequences[1]. The Nuclear Regulatory Commission (NRC), in satisfaction of its severe accident safety requirements, is requiring for new reactor designs, such as the System 80+, among other things, the evaluation of Design Alternatives (DAs) to reduce the radiological risk from a severe accident by preventing substantial core damage (i.e., preventing a severe accident) or by limiting releases from the containment in the event that substantial core damage occurs (i.e., mitigating the impacts of a severe accident).

The Commission's severe accident safety requirements for new reactor designs are set forth in 10 CFR Part 52, §52.47(a)(1)(ii), (iv) and (v). Paragraph 52.47(a)(1)(ii) references the Commission's Three Mile Island safety requirements in §50.34(f). Paragraph 52.47(a)(1)(iv) concerns the treatment of unresolved safety issues and generic safety issues. Paragraph 52.47(a)(1)(v) requires the performance of a design-specific Probabilistic Risk Assessment (PRA). The Commission's Severe Accident Policy Statement elaborates what the Commission is requiring for new designs. The Commission's Safety Goal Policy Statement[3] sets goals and objectives for determining an acceptable level of radiological risk.

As part of its application for certification of the System 80+ design, ABB-CE has prepared a Standard Safety Analysis Report - Design Certification (System 80+ CESSAR-DC[4]). Chapter 19 of the System 80+ CESSAR-DC, "Probabilistic Risk Assessment", demonstrates how the System 80+ design meets the Commission's severe accident safety requirements and policies. In particular, Chapter 19 includes:

(1) Identification of the dominant severe accident sequences and associated source terms for the System 80+ design,

(2) Descriptions of modifications that have been made to the System 80+ design, based on the results of the Probabilistic Risk Assessment (PRA), to prevent or mitigate severe accidents and reduce the risk of a severe accident,

(3) Bases for concluding that "all reasonable steps [have been taken] to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur," (Severe Accident Policy Statement, 50 Fed. Reg. 32,139), and

(4) Bases for concluding that System 80+ meets the Commission's safety goals and objectives as set forth in the Safety Goal Policy Statement.

Consequently, the conclusions are drawn in Chapter 19 that further modifications to the System 80+ design to reduce severe accident risk are not warranted. The National Environmental Policy Act (NEPA) requires the consideration of reasonable alternatives to proposed major Federal actions significantly affecting the quality of the human environment, including alternatives to mitigate the impacts of the proposed action. In 1989, a Federal Court of Appeals determined that NEPA required consideration of certain Design Alternatives; namely, Severe Accident Mitigation Design Alternatives (SAMDAs).(Limerick Ecology Action v. NRC, 869 F.2d 719, 3rd Cir. 1989). The court indicated that "[SAMDAs] are, as the name suggests, possible plant design modifications that are intended not to prevent an accident, but to lessen the severity of the impact of an accident should one occur" (Id. at 731). The court rejected the use of a policy statement as an acceptable basis for closing out NEPA consideration of SAMDAs in a licensing proceeding, because, among other things, it was not a rulemaking (Id. at 739).

Subsequent to the Limerick decision, the NRC issued Supplemental Final Environmental Impact Statements (FES) for the Limerick and Comanche Peak facilities that considered whether there were any cost-effective SAMDAs that should be added to these facilities ("NEPA/SAMDA FES Supplements"). On the basis of the evaluations in the supplements (called "NEPA/SAMDA evaluations"), the NRC determined that further modifications would not be cost-effective and were not necessary in order to satisfy the mandates of NEPA.

In recognition of the Limerick decision, the Commission is requiring NEPA consideration in 10 CFR Part 52 licensing of whether there are cost-effective SAMDAs that should be added to a new reactor design to reduce severe accident risk. While this consideration could be done later on a facility-specific basis for each combined license application under Subpart C to 10 CFR Part 52, the Commission has decided that maintenance of design standardization will be enhanced if this is done on a generic basis for each standard design in conjunction with design certification (SECY-91-229[5]). That is, the Commission has decided to resolve the NEPA/SAMDA question through rulemaking at the time of certification in a so-called unitary proceeding, rather than in the context of later licensing proceedings.

Recently, the NRC Staff expanded the definition of SAMDAs to encompass Design Alternatives to prevent severe accidents, as well as mitigate them[2]. By doing so, the Staff makes the set of SAMDAs considered under NEPA the same as the set of DAs to prevent or mitigate severe accidents considered in satisfaction of the Commission's severe accident requirements and policies.

### 1.2 Purpose

The purpose of this Technical Support Document is to provide a basis for determining the status of severe accident closure under NEPA for the System 80+ design. The document supports a determination, which could be codified in a manner similar to the format of the Waste Confidence Rule (10 CFR § 51.23), as proposed in amendments to 10 CFR Part 51. These amendments would provide that:

(1) For the System 80+ design all reasonable steps have been taken to reduce the occurrence of a severe accident involving substantial damage to the core and to mitigate the consequences of such an accident should one occur.

(2) No further cost-effective SAMDAs to the System 80+ design have been identified to mitigate the consequences of or prevent a severe accident involving substantial damage to the core, and

(3) No further evaluation of severe accidents for the System 80+ design, including SAMDAs to the design, is required in any environmental report, environmental assessment, environmental impact statement or other environmental analysis prepared in connection with issuance of a Combined License for a nuclear power plant referencing the System 80+ design.

The evaluation presented in this document is modeled after that found in the NEPA/SAMDA Final Environmental Statement (FES) Supplements for the Limerick[6] and Comanche Peak[7] facilities. This document presents the bases for concluding that further modifications to the System 80+ design are not warranted in order to reduce the risk of a severe accident through the addition of design features to prevent or mitigate a severe accident.

### 1.3 Description of Technical Support Document

Section 2.0 of this report provides an overview of the radiological risks from nuclear power plants and evaluations of SAMDAs under NEPA. Section 3.0 provides a NEPA/SAMDA evaluation of the radiological risks from normal operations and severe accidents for the System 80+ design. Chapter 4.0 presents the discussion and results of the cost-effectiveness evaluation of the potential SAMDA modifications. Section 5.0 presents the summary and conclusions, and references are included in Section 6.0.

## 2.0 EVALUATIONS OF RADIOLOGICAL RISK FROM NUCLEAR POWER PLANTS

### 2.1 Evaluation of SAMDAs under NEPA and Limerick Ecology Action

Limerick Ecology Action stands for two propositions. First, NEPA requires explicit consideration of SAMDAs unless the Commission makes a finding that the severe accidents being mitigated are remote and speculative. Second, the Commission may not make this finding and dispose of NEPA consideration of SAMDAs by means of a policy statement. The purpose of evaluating SAMDAs under NEPA is to assure that all reasonable means have been considered to mitigate the impacts of severe accidents that are not remote and speculative. As discussed above, the Commission has indicated that it will resolve the NEPA/SAMDA issue for a new reactor design in the same proceeding, called a unitary proceeding, in which it certifies that design.

The Commission's Severe Accident and Safety Goal policy statements require the Commission to make certain findings about each new reactor design. For evolutionary designs, of which the System 80+ is one, this must be done by the Staff in conjunction with issuance of Final Design Approval (FDA) and by the Commission in conjunction with Design Certification. First, the Commission must find that an evolutionary plant meets the safety goals and objectives; i.e., that the radiological risk from operating an evolutionary plant will be acceptable, meaning that any further reduction in risk will not be substantial.

Second, the Commission must find that all reasonable means have been taken to reduce severe accident risk in the evolutionary plant design. As part of the basis for making this finding, the cost-effectiveness of risk reduction alternatives of a preventive or mitigative nature must be evaluated.

Chapter 19 of the System 80+ CESSAR-DC and this Technical Support Document demonstrate that these findings can be made for the System 80+ design. Given the nature and findings of these severe accident and safety goal evaluations, ABB-CE believes that a sufficient basis exists for finding by rule that further consideration of severe accidents, including evaluation of SAMDAs pursuant to NEPA, is neither necessary nor reasonable.

### 2.2 Cost/benefit Standard for NEPA Evaluation of SAMDAs

The Limerick decision interpreted NEPA to require evaluation of SAMDAs for their risk reduction potential. In implementing the court's decision, the NRC considered the cost-effectiveness of each candidate SAMDA in mitigating the impact of a severe accident, using the $1,000 per person-rem averted standard. This standard is a surrogate for all offsite consequences.

The basic approach in this study is to rank the SAMDAs in terms of their cost-effectiveness in mitigating the impact of a severe

accident. The criterion applied is the $1,000 per person-rem averted standard, which is what the Commission has historically used in distinguishing among, and ranking, Design Alternatives, including SAMDAs.

The Commission has used this standard in the context of both safety and NEPA analyses. For example, in the context of safety analysis, the standard has been used to perform evaluations associated with implementation of the Safety Goal Policy Statement; the Severe Accident Policy Statement; and § 50.34(f) requirements. In the context of environmental analysis, it has been used in the Limerick and Comanche Peak NEPA/SAMDA FES Supplements[6,7] and in NUREG-1437[2].

As indicated above, the Commission is preparing a Generic Environmental Impact Statement for License Renewal of Nuclear Plants. The draft statement, NUREG-1437, makes clear that the use of this standard in the evaluation of severe accident risk reduction alternatives, which include SAMDAs, is acceptable (see NUREG-1437, p. 5-108).

On the basis of these considerations, the cost/benefit ratio of $1,000 per person-rem averted is viewed as an acceptable standard for the purposes of evaluating SAMDAs under NEPA.

## 2.3 Socio-Economic Risks for Severe Accidents

As discussed above in Section 2.2, the Commission uses the $1,000/person-rem averted standard as a surrogate for all offsite consequences[8]. However, Environmental Impact Statements (EIS) for nuclear power plants provide separate, general discussions of the socio-economic risks from severe accidents. In keeping with this precedent, ABB-CE is providing a general discussion of socio-economic risks for the System 80+ design, based in large measure on the discussion of such risks in NUREG-1437.

The term "socio-economic risk from a severe accident" means the probability of a severe accident multiplied by the socio-economic impacts of a severe accident. "Socio-economic impacts," in turn, relate to offsite costs. The offsite costs considered in NUREG-1437 are:

- Evacuation costs
- Value of crops or milk contaminated and condemned
- Costs of decontaminating property where practical
- Indirect costs due to the loss of the use of property or incomes derived therefrom (including interdiction to prevent human injury), and
- Impacts in wider regional markets and on sources of supply outside the contaminated area.

NUREG-1437 estimated the socio-economic risks from severe accidents. The estimates were based on 27 FESs for nuclear power plants that contain analyses considering the probabilities and consequences of severe accidents. For these plants, the offsite costs were estimated to be as high as $6 billion to $8 billion dollars for severe accidents with a probability of once in one million operating years of occurring. Higher costs were estimated for severe accidents with much lower probabilities. The projected cost of adverse health effects from deaths and illnesses were estimated to average about 10-20% of offsite mitigation costs and were not included in the $6-$8 billion dollar estimate.

Another source of costs, which NUREG-1437 indicated could reach into the billions of dollars, was costs associated with the termination of economic activities in a contaminated area, which would create adverse economic impacts in wider regional markets and sources of supplies outside the contaminated area. The predicted land contamination was estimated to be small (10 acres/yr at most).

NUREG-1437 provides the bases for concluding that the socio-economic risks from severe accidents are predicted to be small and the residual impacts of severe accidents so minor that detailed consideration of mitigation alternatives is not warranted (56 Fed. Reg. 47,016, 47,019, 47,034-35, September 17, 1991).

## 3.0 RADIOLOGICAL RISK FROM SEVERE ACCIDENTSIN PLANTS OF SYSTEM 80+ DESIGN

### 3.1 Severe Accidents in Plants of System 80+ Design

Chapter 19 of the System 80+ CESSAR-DC, "Probabilistic Risk Assessment," establishes that the Commission's severe accident safety requirements have been met for the System 80+ design, including treatment of internal and external events, uncertainties, performance of sensitivity studies, and support of conclusions by appropriate deterministic analyses and the evaluations required by 10 CFR §50.34(f). It also establishes that the Commission's safety goals have been met.

Specifically, the following topics were addressed in Chapter 19:

(1) Consideration of the contributions of internal events (Section 19.4), Shutdown events (Section 19.8) and external events (Section 19.7) to severe accident risks, including a seismic risk analysis based on the application of the seismic margins methodology (Section 19.7.5),

(2) Identification of the System 80+ dominant accident sequences,

(3) Identification of severe accident risk reduction features which were included in the System 80+ design to achieve accident prevention and mitigation (Section 19.15.1), and

Chapter 19 of CESSAR-DC addresses how the goals of the Severe Accident Policy Statement have been met for plants of System 80+ design. These goals include:

- Prevention of core damage,
- Prevention of early containment failure for dominant accident sequences,
- Evaluation of the effects of hydrogen generation,
- Heat removal to reduce the probability of containment failure,
- Prevention of hydrogen deflagration and detonation
- Offsite dose, and
- Containment conditional failure probability.

Specific conclusions concerning severe accidents for the System 80+ design based on the Chapter 19 evaluations and this document are as follows:

(1) Core Damage Frequency. The System 80+ core damage frequency for power operation, including the scoping values for fire and flood, was determined to be 1.96E-6 per reactor year (Table 4-2 of this document).

(2) Conditional Containment Failure Probability. The conditional containment failure probability was shown to be 0.04 (Section 19.12.2.3). This is significantly below the goal of 0.1,

(3) Probability of Large Offsite Dose. The probability of exceeding a whole body dose of 25 rem at a distance of one-half mile from the System 80+ design site boundary was determined to be less than 5.3E-8 per reactor year (Section 19.13), and

(4) Residual Radiological Risk. Residual radiological risk from severe accidents in plants of System 80+ design is summarized in Table 4-3 of this document. The cumulative exposure risk to the population within 50 miles of the plant of System 80+ design site boundary is approximately 17 person-rem for an assumed plant life of 60 years (2.82E-1 mr/yr * 60 yrs).

In addition, consideration of design modifications in accordance with §50.34(f)(1) is presented in Section 4 of this document.

## 3.2 Dominant Severe Accident Sequences for Plants of System 80+ Design

In performing the Probabilistic Risk Assessment (PRA) for the System 80+ design, many severe accident sequences were identified and evaluated . For each sequence, the analysis identified an initiating event and traced the accident's progression to its end. For sequences involving core damage, conditional containment failure probabilities and offsite consequences were estimated. The accident scenarios were binned according to radiological release (source term) parameters, and twenty-three release classes were defined and quantified. The dominant cases in terms of offsite risk are containment bypass sequences. Table 4-1 of this document defines the release classes. The complete radiological consequence analysis of the dominant sequences can be found in Section 19.13 of CESSAR-DC.

ABB-CE believes that the severe accident analysis in Chapter 19 of CESSAR-DC is complete and core damage sequences not included in that analysis are not dominant can be deemed remote and speculative.

### 3.3 Overall Conclusions from Chapter 19 of CESSAR-DC

The specific conclusions about severe accident risk discussed above support the overall conclusion that the environmental impacts of severe accidents for plants of System 80+ design represent a low risk to the population and to the environment. For the System 80+ design, all reasonable steps have been taken to reduce the occurrence of a severe accident involving substantial damage to the core and to mitigate the consequences of such an accident should one occur. No further cost-effective modifications to the System 80+ design have been identified to reduce the risk from a severe accident involving substantial damage to the core. No further evaluation of severe accidents for the System 80+ design is required to demonstrate compliance with the Commission's severe accident requirements or its policy or safety goals.

## 4.0 COST/BENEFIT EVALUATION OF SAMDAS FOR PLANTS OF SYSTEM 80+ DESIGN

This document considered whether the System 80+ design should be modified in order to prevent or mitigate the consequences of a severe accident in satisfaction of the NRC's severe accident requirements in 10 CFR Parts 50 & 52 and the Severe Accident Policy Statement. The cost/benefit evaluation of SAMDAs to plants of System 80+ design used the expanded definition of SAMDAs set forth in NUREG-1437, i.e., DAs that could prevent and/or mitigate the consequences of a severe accident.

### 4.1 Cost/Benefit Standard for Evaluation of System 80+ SAMDAs

As discussed in Section 2.2 above, the cost/benefit ratio of $1,000 per person-rem averted is viewed by the NRC and the nuclear industry as an acceptable standard for the purposes of evaluating SAMDAs under NEPA. This standard was used as a surrogate for all offsite costs in the cost/benefit evaluation of SAMDAs to plants of System 80+ design. Averted Onsite Costs (AOCs) were incorporated for SAMDAs that were at least partially preventive in nature. Onsite costs resulting from a severe accident include replacement power, onsite cleanup costs, and economic loss of the facility. The equation used to determine the cost/benefit ratio is:

$$\text{Cost/Benefit Ratio (\$/per.-rem)} = \frac{[\text{cost of SAMDA} - (60\text{yr} * \text{AOC/yr})]}{60\text{yr} * \text{reduction in person-rem/yr}}$$

A plant lifetime of 60 years was assumed to maximize the reduction in residual risk. This equation neglects any additional costs associated with the maintenance and testing of the additional DAs. It also neglects the time effect on the cost of capital. Assessments of AOCs are provided for information only. It is ABB-CE's position that the NRC is not required to account for these costs.

The Design Alternative evaluation followed the format and procedure used by the NRC in evaluating DAs for Limerick[6]. The DAs were evaluated in terms of cost/benefit where the cost of the additional equipment is compared with the savings in terms of a reduced exposure risk to the general population in person-rems per year. The risk of the base System 80+ design is described in Section 19.15 of CESSAR-DC.

### 4.2 Risk Reduction

Risk (person-rem/year) in this analysis is the product of the frequency of core damage for each type of accident (events/yr) times the consequence of the accident (person-rem/event). The total risk is the sum of the risks from all types of accidents. For each DA,

the reduction in total risk is the difference between the risk of the base System 80+ design and the risk with the DA added.

Risk is defined as the product of frequency and consequence. The frequency of core damage for various accident sequences are calculated. These sequences are then grouped ("binned") into release classes depending on the timing of the accident and the conditions of the core, vessel, containment, and release characteristics for the sequence. Each DA is evaluated in terms of how it might affect each release class. For this analysis it is assumed that each DA is perfect, that is, if installed it eliminates all failures associated with the systems for which it is designed to be an alternative or addition. This implies that each DA is also tied to perfect support systems. This is a conservative upper limit approach since it overestimates the benefits associated with any design addition. If a DA is cost beneficial using this screening approach, then a more detailed analysis could be performed.

The DAs can be divided into two groups. One group prevents core damage and the other group protects the containment or reduces the releases. For the DAs that prevent core damage, the frequency of affected release classes was decreased based on the sequences that were binned and the risk reduction was calculated. For example, an Alternative Pressurizer Auxiliary Spray (DA B3) is assumed to eliminate all core melt risk of a Steam Generator Tube Rupture (SGTR) by always getting the plant depressurized and into shutdown cooling. Therefore, the frequency of core damage for the Plant Damage States (PDSs) with failure to cooldown aggressively was reduced to zero and a risk reduction was calculated.

Some DAs protect the containment or reduce the amount of radioactive material that is released in an accident. These DAs reduce the consequences of the accident and, therefore, reduce the risk (risk = frequency x consequence). Using the S80SOR, a modified version of the ZISOR Code, the consequence in terms of dose to the general population is calculated for the ALWR site. The ALWR site was described in the May 1989 version of the KAG[9] and was to represent 80% of the potential sites. The site was an existing site in South Carolina with the population increased to represent most potential sites[10]. For DAs that prevent containment failure, the releases were assumed to be reduced to zero, and the risk was reevaluated.

### 4.3 Cost Estimates

In order to evaluate the effectiveness of the DAs, the benefits were compared to the costs of the DAs. Conservatively low cost estimates were made for each potential modification. These costs represent the incremental costs that would be incurred in incorporating the alternative in a new plant. The cost estimate for each of the modifications is given where the modification is discussed.

The cost estimates were intentionally biased on the low side, but all known or reasonably expected costs were accounted for in order that a reasonable assessment of the minimum cost would be obtained. Actual plant costs are expected to be higher than indicated in this evaluation. All costs for the DAs are in 1993 U.S. dollars.

The analysis presented here conservatively neglects any annual costs associated with the operation of the DAs. These DAs would have to be tested and maintained at regular intervals. Regular training would also be required. In a more detailed analysis, such costs would be converted to an annual cost and be used to reduce the annual benefits.

### 4.4 Averted Onsite Cost

For the plant modifications that reduced the Core Damage Frequency (CDF), the cost was decreased by an amount equal to sixty times the AOC (no correction for present worth). Modifications that reduced the probability of containment failure, or reduced the amount of fission products leaving the site, were assumed to have no significant AOC reduction.

AOC included replacement power costs, direct accident costs (including cleanup), and the economic loss of the plant. Credit is given for property and replacement-power insurance. Evaluation of the AOC includes the following considerations:

1) The replacement power costs used ($386,000/day) is a replacement power cost for the Palo Verde Reactor (a System 80 plant) averaged for 1993 as predicted by ANL[22]. This cost is applied for a three-year period because it is assumed that the utility will contract with another producer prior to expiration of that period for power at a comparable cost as that incurred for the nuclear plant. Currently, Independent Power Producers (IPPs) can build new facilities in approximately 12 months and IPP capacity and energy rates are competitive relative to nuclear plant rates. Therefore, a three-year replacement power period is a reasonable assumption. Replacement power costs are estimated at $423 Million (M) per year but will be partially offset by total replacement power insurance of $365M[23].

2) Direct accident costs, including cleanup costs, were assumed to be $2 Billion (B). This is partially offset by the primary and excess nuclear-property insurance of $1.625B[23,24]. Most new, large plants and publicly owned plants carry the maximum amount of coverage. The NRC requires the plant owners to carry over $1B of private coverage.

3) The economic value of the facility at the time of the accident was calculated assuming that the initial plant invested cost was $1.4B based on DOE cost guidelines. It is also assumed that a straight line depreciation value is used over a twenty-year period and the accident is equally probable during any year in the plant's sixty year life. The economic value of the plant averages $233M and is assumed lost. The inclusion of both the value of the plant and its output (replacement power) is conservatively exaggerate the size of the AOC.

The total AOC is estimated at $666M. This figure neglects the credits for premature decommissioning insurance and/or elimination of annual capital expenditures. Such credits would further reduce the AOC.

### 4.5 Cost Benefit Comparison

As described in Section 4.2, the benefit of a DA is a risk reduction, which was evaluated in terms of reduced exposure of the general population (in units of person-rem/yr). The cost of additional equipment is in dollars, a one-time initial capital cost. To compare these two numbers, a common measure must be used. In this analysis, the risk reduction was converted to a single capital benefit which can be directly compared with the capital cost.

The benefits of a particular DA were defined as the risk reduction to the general public. Offsite factors evaluated were limited to whole body dose to the general public. Consistent with the standard used by the NRC to evaluate radiological impacts, health and economic effects costs were evaluated based on a value of $1,000 per offsite person-rem averted[11] due to the design modification.

The offsite costs for other items, such as relocation of local residents, elimination of land use and decontamination of contaminated land are considered as part of the $1000/person-rem figure.

### 4.6 PRA Release Classes

In assessing the risk reduction of each DA, the potential for each DA to reduce the frequency of occurrence or the consequence of each Release Class (RC) is assessed. To do this, an understanding of each RC is required.

In Section 19.12 of CESSAR-DC, the containment event analysis describes the possible accident pathways in a Containment Event Tree (CET). This CET was developed so that each end point of an accident sequence uniquely specified the mode of containment

failure and the status of the various phenomena which have the potential to affect the source term characteristics. Therefore, each of the accident end points is a distinct release class.

A release class (RC) can be fully characterized by the following parameters:

A) Its frequency of occurrence,
B) the isotopic content and magnitude of the release,
C) the energy of the release,
D) the time of the release,
E) the duration of the release, and
F) the location of the release.

The RC frequency is determined directly from the cumulative frequency for its respective Containment Event Tree end point. The location of the release was assigned as follows:

1) For overpressure containment failure RCs, the release was assumed to occur at the top of the containment building. This is at an elevation of 52.8 meters above grade,

2) For containment bypass RCs initiated by an interfacing systems LOCA and for containment melt-through RCs, the release from containment occurs in the region of the auxiliary building located below the containment sphere. The actual release to the environment occurs at grade level, and

3) For all other RCs, the releases are assumed to occur at grade level.

S80SOR analyses were used to determine the isotopic content and magnitude of the source term and the time of the release. In general, releases were calculated for a period of 24 hours from the time of containment failure or from the time of vessel failure for containment bypass and containment isolation failure RCs.

Table 4-1 presents a brief description for each release class with a frequency greater than or equal to 1.0E-10. This table is used to identify the effect of mitigation equipment (more details of each RC is given in Section 19.12.3). Table 4-2 gives the mapping of each PDS into each release class. Also given in this table are the mapping of the Core Damage Frequency (CDF) sequences into the PDSs. In addition, the description of each sequence and the sequence CDF is also presented. This table is used to reduce each RC frequency (column 2 of Table 4-2) for preventative DAs.

The sequence CDF (last column of Table 4-2) was used to calculate the risk reduction associated with DAs that prevent core damage. It was assumed that any preventive DA would completely eliminate the sequence that the DA would address. For example, a Safety Injection DA would reduce the RC1.1E by 55%. Safety Injection System (SIS) failure appears in five of the sequences with a total

sequence frequency of 6.63E-7. The sum of all the sequences contributing to RC1.1E is 1.21E-6 and, therefore, the DA is assumed to reduce this RC by 55% (6.63E-7/1.21E-6). Each release class is evaluated in this manner for each preventive DA.

Table 4-3 gives the ranking of the release classes in terms of risk to the general population (mr/yr). It also gives the base frequency, and population dose for each RC that is used in the risk reduction analysis. The first three sequences are associated with Steam Generator Tube Ruptures (SGTRs). This table, in conjunction with the previous two tables, was used in selecting the DAs because it highlights the importance of the failure modes. Table 4-4 gives the ranking of the Level I sequences in terms of CDF. This table is useful for selecting DAs for preventing core damage.

Each release class was evaluated for total person-rem exposure using MACCS[12]. Table 4-3 gave the initiating frequency, and total person-rem dose for the twenty-three release classes with initiating frequencies greater than 1.0E-10. The lifetime doses were calculated for people within 50 miles of the site and assumes the evacuation strategy used in NUREG-1150[13]. The risk for each release class is the product of frequency (events/year) times the total person-rem exposure per event. This product gives person rem per year and is a measure of risk. The total risk of the dominant release classes is 0.135 person-rem/yr. These results are for the ALWR site which is representative of most of the current U.S. sites[10].

Table 4-1 summarizes the accident characteristics for each release class. These are the dominant sequences of the binned accidents. For each DA, the release class was reviewed assuming that the DA worked perfectly (failure rate = 0.0). This means that each DA had perfect support systems, power supplies and heat sinks. In addition, for each DA, no other failure modes were considered when the DA was employed. For example, when the Pressurizer Auxiliary Spray DA is employed to ensure that the primary coolant pressure can be decreased to enter SCS operation, the Shutdown Cooling System (SCS) is assumed always to work. This represents an upper limit scoping analysis and maximizes the benefit of each DA. If a DA is cost beneficial in this analysis, then a more detailed analysis addressing the actual failure rate of the DA can be undertaken.

### 4.7 Design Alternatives

Potential modifications to the System 80+ design were derived from a survey of the dominant failure modes shown in Table 4-1 through Table 4-4. Others were suggested by the PRA or the design engineering staff. Some of the DAs were suggested by a foreign utility interested in System 80+ design. Table 4-5 gives the DAs considered and shows how they were treated.

The risk reduction values of twenty-seven DAs were quantified. These were selected based on the SAMDAs for the Limerick plant[6], Comanche Peak SAMDA[7], NUREG/CR-4920[15], GI-163[16], and a review of the dominant failure modes for the System 80+ plant. In addition, suggestions from ABB-CE personnel with technical expertise in containment response were employed. Design Alternatives from earlier plant studies were also considered.

The Design Alternatives can be divided into five groups (see Table 4-6). The first four groups prevent core damage by:

A) Increase primary and secondary boundary integrity,
B) Increase decay heat removal reliability,
C) Improved electrical power reliability, and
D) General reduction of CDF.

The fifth group protects the containment or reduces the releases:

E) Reduce Radioactive Releases.

For the DA that prevents core damage, the frequency of affected release classes is reduced by the fraction that the sequence contributes to the RC, and the total risk reduction is calculated. This group includes the high capacity Safety Injection System (SIS), improved DC Battery and Emergency Feedwater System (EFWS), Anticipated Transients Without Scram (ATWS) Pressure Relief Valves, improved Pressurizer Auxiliary Spray, improved Primary Depressurization System, and alternative RCP Seal Cooling System.

At the beginning of the design process, it was recognized that steam generator integrity was important to safety and plant economics. The risk of a SGTR in the System 80+ design is two orders of magnitude below current plants, but SGTR nevertheless represents over half the offsite risk. System 80+ is designed to prevent Main Steam Safety Valve (MSSV) actuation following SGTR, as described below, and also includes new or enhanced features for the prevention of SGTRs.

Features to prevent SGTRs include:

- Steam generator tubes made of thermally treated Inconel 690, which has favorable corrosion resistance properties including superior resistance to primary and secondary stress corrosion cracking,

- A deaerator in the condensate/feedwater system for removal of oxygen,

- Condensate system with full flow condensate polisher to remove dissolved and suspended impurities,

- Main condenser with provisions for early detection of tube leaks, and segmented design permitting repair of leaks while operating at reduced power,

- Steam, feedwater and condensate blowdown system and steam generator secondary side recirculation system for chemistry control during wet layup, and

- N-16 monitors to provide early detection and an opportunity to shutdown prior to SGTR.

The response to Unresolved Safety Issue A-4 (Chapter 20 of CESSAR-DC) further describes design features to assure steam generator (SG) tube integrity. New or enhanced System 80+ features which help to mitigate SGTRs include:

- Larger steam generator secondary volume,

- Larger pressurizer,

- Four train Safety Injection System,

- Four train Emergency Feedwater System,

- Electrical system upgrades including alternate AC combustion turbine and 8-hour batteries,

- Safety Depressurization and Vent System,

- Component Cooling Water System upgrade to four 100% capacity pumps and heat exchangers,

- Highly reliable Turbine Bypass System, discharging all steam to condenser, not partially to atmosphere as in earlier designs,

- Radiation monitors on the steam lines, and

- N-16 monitors for the steam generators.

The System 80+ design meets the Electric Power Research Institute's (EPRI's) ALWR requirement of preventing MSSV actuation following a SGTR. A reactor trip on high SG water level, actuation of the Turbine Bypass System and controlled depressurization of the RCS using the Reactor Coolant Gas Vent System (RCGVS) limit secondary side pressure below the MSSV setpoint. The turbine bypass valves discharge steam to the main condenser, which minimizes the radioactive release to the environment. The intent of the EPRI Advanced Light Water Reactor Utility Requirement Document (ALWR URD[9]) was to meet the above requirement on a best-estimate basis (i.e., credit for operator action and use of control-grade equipment is acceptable) to provide an effective and economical design.

The consequences of a SGTR with Loss of Offsite Power (LOOP) where the containment is bypassed due to malfunction of an Atmospheric Dump Valve (ADV) has been analyzed and presented in Section 19.15.6.3.3 of CESSAR-DC. The analysis simulated a double-ended break of one SG tube and calculated the worst-case releases for an SGTR event with LOOP and a stuck open ADV on the affected steam generator. The analysis contained conservative assumptions regarding atmospheric dispersion factors, initial RCS and SG activity levels, and iodine spiking. Mitigating operator actions based on the approved ABB-CE Emergency Procedure Guidelines (EPGs), CEN-152, were simulated. The analysis showed that no fuel failures were expected for this event.

The ADV on the affected SG was assumed to stick open when the operator tried to reseat the ADV to isolate the affected SG. After 30 minutes of steaming through the stuck-open ADV, the operator isolated this path by closing the ADV block valve. However, the leak of RCS liquid through the tube break continues for the duration of the analysis (8 hours) due to the conservative nature of the analysis models. In order to avoid overfilling the SG, the operator periodically steams from the affected SG per the EPGs. This additional steaming increased the total radiation dose. The total releases are well within regulatory limits.

An additional analysis was performed that looked at SGTRs that are beyond design bases[17]. This analysis looks at failure of up to five tubes in a SG concurrent with a stuck open MSSV. The analysis credits the Turbine Bypass Control System and shows that the operator has thirty minutes to take action before the MSSVs lift. This analysis also evaluated eleven design changes including; automatic bypass of Main Steam Isolation Signal (MSIS) on high SG level, automatic initiation of Auxiliary Pressurizer Spray, and automatic opening of the Reactor Coolant Gas Vent System (RCGVS) valves. The referenced report contains a technical discussion of the advantages and disadvantages of each modification and concluded that none of them are worth including in the design.

It was recognized that the SGTR event represented a significant fraction of the offsite risk and, therefore, DAs were selected specifically to address these sequences. These DAs include the Alternative Pressurizer Auxiliary Spray, Ideal 100% SG inspection, MSSV and ADV Scrubbing, Alternative SIS, and Diesel SIS Pump. The last two DAs address failure to inject for RC4.30E. DA B8 specifically addresses refilling the Refueling Water Storage Tank (RWST) during a SGTR. Secondary side guard pipes (DA A6) are also evaluated.

## A. INCREASE PRIMARY AND SECONDARY BOUNDARY INTEGRITY

This group of Design Alternatives were grouped because they address primary or secondary coolant boundary integrity.

### A1 RCP SEAL COOLING

The System 80+ employs a type of Reactor Coolant Pump (RCP) seal which can withstand a loss of cooling and not result in a LOCA. This type of seal design has been employed in the operating ABB-CE plants and experience has shown that the seals do not fail when seal cooling is lost[18]. The reliability of the reactor coolant pump seal cooling could be improved by adding a small dedicated positive displacement pump for diverse seal injection. This design addition will provide additional diversity for RCP seal cooling and provide a seal cooling system that is not dependent on CCW. Such a RCP seal cooling pump has been added to the System 80+ plant as a result of NRC's questions on testing of the RCP seals and, therefore, a cost/benefit analysis is not needed.

### A2 100% SG INSPECTION

Inspection of 100% of the tubes in a steam generator is not really a DA but is a maintenance practice. Inspection was selected because it has reasonable costs, and can be executed with a management decision. This DA was introduced to specifically address a SGTR, which is the initiating event for the largest three RCs. The analysis assumes that all SGTRs are eliminated. This reduces the risk due to a SGTR in the System 80+ design for six RCs (Table 4-7).

The increased cost of performing eddy current testing on 100% of the steam generator tubes compared to a 20% random inspection of the steam generator tubes is $1.5 million per refueling outage. Assuming an eighteen month fuel cycle, this would cost $1.0M/yr. This amount is used directly to calculate cost/benefit, $/p-rem.

### A3 N-16 MONITORS

N-16 monitors have been added to the System 80+ design. Their purpose is to assist the operators in identifying SGTR events. They also help prevent SGTR by offering an alternative method on detecting a leaking tube before rupture occurs. This DA was not quantified since it has been included in the design.

### A4 INCREASE SECONDARY SIDE PRESSURE

Upgrading the design pressure of the secondary system, including the MSSVs, to 1500 psia from the current 1200 psia was considered early in the System 80+ design process. It was determined that an increased design pressure would not significantly reduce the

probability of containment bypass and release to the environment during a SGTR event.

During a SGTR with loss of offsite power, the condenser is not available for plant cooldown. The decay heat of the core and the stored energy in components are released to the atmosphere via the MSSVs, then via the ADVs. The steaming will continue until reaching Shutdown Cooling System (SCS) entry conditions. The total heat to be removed (or the total steam release) is only slightly reduced by increasing the secondary design pressure and MSSV setpoints. Hence, using conservative safety analysis assumptions and methods, the overall radiation release would be essentially unchanged.

During a SGTR with offsite power available, the operator will act to mitigate this event according to the Emergency Procedures Guidelines, using both control grade and safety grade equipment, if required. Therefore, for a "real-world" scenario, an increased design pressure would not significantly decrease the likelihood of lifting the MSSVs.

There are several technical disadvantages of increasing the secondary system design pressure to 1500 psia:

1. Steam generator weight would increase by up to 100 tons each. The added weight would increase containment heat sinks and increase thermal stresses on the steam generator shell and main steam piping. These factors would likely impact the volume and arrangement of the containment. The additional weight would also increase the handling difficulties during fabrication;

2. The RCS support system would need to be redesigned and/or reevaluated to accommodate the increased loads. Any contribution to containment sizing must also be assessed;

3. For decreased heat removal events, RCS temperature and pressure would rise to a higher value than in current plants. Pressurizer safety valve actuation would be more likely;

4. Unless the entire steam system and turbine are upgraded to 1500 psia, a second set of secondary side relief valves would be required downstream of the Main Steam Isolation Valves (MSIVs) to protect the low pressure portion of the steam system; and

5. Feedwater systems would have to be compatible with the higher design pressure. Increasing secondary design pressure would require a major redesign effort and increase design complexity,which are not consistent with the evolutionary ALWR design goals.

In summary, including an upgrade to the secondary side design pressure was considered from design considerations. Based on this review, this DA poses serious design drawbacks with limited benefits. A cost benefit analysis was not performed for this DA because very limited benefits were expected for extensive costs.

## A5 PASSIVE SECONDARY SIDE COOLERS

Secondary heat rejection for System 80+ has been considered at the conceptual level. Passive secondary heat rejection was included in the conceptual design for SIR, a much smaller plant.

The passive secondary heat rejection concept that is often discussed consists of an elevated condenser designed to full secondary side pressures. The heat sink for this condenser can be either water or air. If the heat sink is water, there is an elevated water tank that supplies water to the condenser and water is allowed to boil to the atmosphere. Use of air in natural circulation results in a large increase of the surface area of the condenser but it has the potential of continuous long term operation without support. The water tank concept requires a periodic refill.

The base system is relatively simple. However, several supporting functions are required to initiate the system. Isolation of the affected steam generator will be required, otherwise one must assume the entire cooling loop will go water solid with pressures equal to RCS pressure. An alternative is to have a continuous drain system that maintains a suitable free surface in the steam generator. This requires coordination with the RCS makeup system. Control of cooldown rates is expected to be required, adding additional complexity. Heat rejection capacity sufficient to avoid early releases is expected to result in excessive cooldown rates later. While simple in concept, the implementation of secondary closed loop cooling is expected to require major changes in the plant structures. A workable system will be more complex than the conceptual presentations being offered. Because of the redundancies in the current System 80+ design, and the potential high cost of this DA, this DA was not further studied.

## A6 SECONDARY SIDE GUARD PIPES

The secondary side guard pipe was proposed to address a Main Steam Line Break (MSLB) outside containment. This event is postulated to trigger multiple steam generator tube failures which could then result in a core melt because of depletion of coolant inventory. This sequence also bypasses the containment. The guard pipe would extend from the containment to the MSIVs and would be designed to prevent depressurization, given a MSLB in the specific section of pipe. MSLB represented 0.5% of the CDF for System 80+. It was assumed that this DA would halve the risk associated with Interfacing Systems LOCAs (RC5.1E) and halve the risk associated with all steam line break sequences because it is assumed that half

of the lengths of main steam lines are guarded. Table 4-8 quantifies the risk reduction value of this DA.

The cost for the guard pipes was taken from GI-163[16] and adjusted for the different number and size. The original estimate of $1.1M was for a four loop plant. Since the System 80+ has four steam lines leaving the two SGs, this estimate was used for the DA analysis. The final cost of $1,100,000 was used in this analysis. This cost neglects the increased inspection and maintenance cost of the main steam lines because they are no longer accessible.

## A7 AUTOMATIC OVERPRESSURE PROTECTION

ABB-CE conducted an extensive evaluation of the System 80+ standard design to respond to Interfacing System LOCA (ISLOCA) challenges to address the NRC Staff concerns raised in SECY-90-016 and SECY-93-087. ABB-CE and the Staff worked closely in the development of an acceptance criteria and performance of a system-by-system evaluation of ISLOCA challenges. The evaluation was documented in an ABB-CE special report, which has been incorporated as Appendix 5E in CESSAR-DC. Table 2-1 of Appendix 5E summarizes the design changes made to achieve ISLOCA responses acceptable to the Staff. Section 4 of Appendix 5E presents the evaluation of DAs and rationale for the selected design approach for each potential ISLOCA pathway. Section 5 of Appendix 5E presents the Nuplex 80+ indication and control availability requirements supporting ISLOCA detection and diagnosis. Since this issue has been designated by the Staff as technically resolved, no further evaluation or reporting will be provided.

## A8 DIGITAL LBLOCA PROTECTION

The likelihood of Plant Protection System (PPS) or Engineered Safety Feature (ESF) component system failure has been made extremely low through redundancy, hardware qualification, and a rigorous quality assurance program which has been reviewed by the NRC (Section 7.2.1.1.2.5). Large Break LOCAs represent only 6.6% of the CDF, and steam line breaks represent 0.5% of the CDF. These events are not major contributors to offsite risk because they tend to be inside the containment. Therefore, only minor benefits in terms of public risk would be expected. The Large Break LOCA (LBLOCA) and steam line break within containment events can be assured through operator action in response to symptoms of precursor leakage (Leak Before Break, LBB). The instrumentation available to detect the leakage includes:

- Acoustic leak monitoring system alarm and trending,
- Containment Temperature Level,
- Containment Radiation, and
- Containment Humidity.

Nuplex 80+ makes possible tracking of leakage within containment and correlation of multiple symptoms. In addition to increased costs and complexity of additional trips and ESF actuation paths, the additional trips could decrease plant availability and increase the potential for equipment challenge (false actuation leading to transients) for a negligible improvement in plant safety. Because of the small public risk associated with the LBLOCA and the sophistication of the current protection system, this DA was not further considered.

**B. INCREASE DECAY HEAT REMOVAL RELIABILITY**

This group of Design Alternatives has been grouped together as improved decay heat removal reliability. The SIS and Safety Depressurization System (SDS) were grouped here because of their feed and bleed capability.

B1 ALTERNATIVE DC BATTERIES AND EFWS

This DA addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. The System 80+ design already has an improved battery system that will carry the DC loads for 8 hours. However, there are still accident sequences where the batteries are depleted and emergency feedwater is lost leading to core damage. The improved DC batteries and EFWS DA is assumed to have the capability to remove decay heat using batteries and the turbine-driven feedwater pump for whatever time period that is required (without any failures). This DA prevents core damage and, therefore, removes two of the release classes (Table 4-9).

Design of a battery system with unlimited capacity is not possible. However, to increase the existing battery capacity for the EFWS pumps from the current System 80+ design capacity of 8 hours to 72 hours will require 9 times the number of current battery cells and thus approximately 9 times the space for building storage. The increased building space will also increase the Heating, Ventilation and Air Conditioning (HVAC) requirements. The cost for the extra battery cells, building volume and increased HVAC requirements is estimated to exceed $2 million. In the Comanche Peak SAMDA, additional batteries were estimated to cost between $1.3M and $3M.

B2 12-HOUR BATTERIES

This DA is similar to the DA described in Section B1 (an ideal battery system). This DA is for a specific and technically realistic DA of using a battery system that would maintain load for twelve hours. Such an improvement would decrease the probability of failure to restore offsite power from 0.081 to 0.031[9], a 62% improvement. In terms of risk reduction benefits, this DA reduces the risk of two RCs (Table 4-10).

Increasing the current battery size to accommodate a 12-hour duty cycle for station blackout loads rather than an 8-hour duty cycle would require more plates per cell (minimum of 25% increase). Preliminary estimates show that the existing 8-hour duty cycle requires a large number of plates per cell (assuming 60 cell battery). Therefore, a 25% increase in plates per cell may exceed the number of plates that can be placed in a typical cell and may not be possible. However, if cells are available in sufficient size, they would be larger per cell and would require an additional mounting rack, which would require, at a minimum, 1.5 times existing battery building space. The more likely scenario would require another 60-cell battery or two 58-cell batteries connected in parallel. Thus, the required space would be 2 times existing space. The cost of this modification would be in excess of $300,000.

## B3 ALTERNATIVE PRESSURIZER AUXILIARY SPRAY

This DA was introduced to specifically address a SGTR, which is the initiating event for the largest three RCs. The analysis assumes that during a SGTR, the auxiliary spray will always depressurize the primary system to the SCS operation mode with sufficient speed and the SCS will always remove decay heat. This reduces the risk due to a SGTR in the System 80+ design for six RCs (Table 4-11).

Designing a perfect Pressurizer Auxiliary Spray System is not possible. However, increased reliability and diversity can be obtained by increasing the redundancy and diversity of the pressurizer spray valves and providing a diverse positive displacement charging pump that is powered from a diverse power source. The reliability of the SCS can be improved by providing a diverse shutdown cooling pump with a diverse power source and providing a diverse heat sink. The cost for the additional components, piping, power supplies, instrumentation and building volume is estimated to exceed $5 million.

## B4 ALTERNATIVE HIGH PRESSURE SAFETY INJECTION

The System 80+ design has a very reliable four train Safety Injection System to begin with. The high pressure Safety Injection DA assumes that all sequences with SIS failures can be eliminated (Table 4-12). As shown in Table 19.6.3.6-5 of the PRA, the dominant failure mode (80% of the total for small break LOCA) is common cause failure of the four check valves or four motor operated isolation valves. The Alternative SIS would have eight additional valves, each one with piping to parallel the existing valves. The estimated cost for this modification is $2.2M. It is assumed that these valves are not subject to common cause failures. Testing and maintenance has not been considered.

B5 ALTERNATIVE RCS DEPRESSURIZATION

The System 80+ design has motor-operated relief valves (MORVs) that permit residual heat removal using the valves and SIS pumps in a "feed and bleed" mode of operation. This DA models a perfect MORV system that permits the primary coolant system to be quickly depressurized so that the SIS pumps are effective in getting coolant into the core and removing decay heat. This DA eliminates all sequences in Table 4-2 where the Safety Depressurization System (SDS) or bleed fails. The risk reduction is shown in Table 4-13.

Designing a perfect SDS is not possible. However, increased reliability and diversity of the system can be obtained by increasing the redundancy of the safety depressurization valves and/or providing valves that are diverse. Providing the additional valves, piping and instrumentation is estimated to exceed $500,000. In the Comanche Peak SAMDA, an Alternate Depressurization System was estimated to cost between $1.9M and $3.7M.

B6 DIESEL SI Pumps (2)

The System 80+ design has a very reliable four train Safety Injection System (SIS) to begin with. The high pressure Safety Injection DA (DA B4, Table 4-12) assumes that all sequences with SIS failures can be eliminated. This DA is more specific. It assumes that two of the motor-driven SIS pumps are replaced with diesel pumps. This reduces common cause failures of all four pumps and also reduces the risk of station blackout. Using the failure rates and common cause dependencies in Reference 9, the reliability of the SIS would be increased by a factor of 60. Station blackout was assumed to be eliminated. Table 4-14 shows that nineteen RCs are reduced.

This modification would require replacing the electric motors on two of the safety injection pumps with diesel engines. The diesel engines will also require additional support systems and additional building volume to house the diesel drives and support systems compared to electric motor drives. The cost of this modification would be in excess of $2 million.

B7 ALTERNATIVE STARTUP FEEDWATER SYSTEM

The Startup Feedwater System introduces the feedwater upstream of the main feedwater control valves and is assumed to be unavailable for transients such as loss of feedwater. The Alternative Startup

Feedwater System is available as a backup to the EFWS. It is assumed to eliminate the sequences in Table 4-2 where the EFWS fails. The System 80+ Startup Feedwater System has been modified such that it can be utilized as a back up to the Emergency Feedwater System. The System 80+ startup feedwater pumps are powered from the Combustion Turbine such that they are available on a Loss of Offsite Power event. The Condensate Storage Tank provides the water source for the startup feedwater pumps. Since the Startup Feedwater System is a non-safety grade system, the water from the startup feedwater pump is supplied upstream of the main feedwater isolation valves. Should the transient cause the main feedwater isolation v[illegible]es to close on a Main Steam Isolation Signal (MSIS), the sign[illegible] be bypassed and the valves reopened. The instrument air com[illegible]s are also powered from the Combustion Turbine. Therefore, [illegible]ey will be available to provide the air source for reopening the main feedwater isolation valves. Since this DA is included in the System 80+ design, no cost benefit analysis is necessary.

B8 EXTENDED REFUELING WATER STORAGE TANK SOURCE

In the important SGTR sequences (RC4.36L), the Refueling Water Storage Tank (RWST) source depletes as a makeup source. This DA consists of a ground level tank of borated water, and a pump and piping to pump the water to the RWST. It is assumed that the supply of water is sufficient to permit corrective actions before it also is exhausted. This DA is assumed to eliminate RC4.36L (Table 4-16). A detailed design for the extended RWST source has not been performed but it would require a ground level tank of borated water and a pump and piping to pump the water to the elevated RWST together with an instrumentation and control system. This is estimated to cost in excess of $1 Million.

## C. IMPROVED ELECTRICAL POWER RELIABILITY

This group of Design Alternatives addresses the station blackout scenario. Battery depletion was considered under improved decay heat removal because it was assumed to enable longer operation of the steam-driven emergency feedwater trains.

C1 THIRD DIESEL GENERATOR

The System 80+ plant is designed to have two Diesel Generators (DGs), a Combustion Turbine and two independent switchyards. Many plants are using a third DG as a swing unit or during a refueling when one DG is out for maintenance. This DA was selected to address the risk reduction of installing an additional unit. It was assumed that the unit was affected by common cause failure and had a conditional failure rate[9] ($\gamma$) of 0.76/d given that the other

DGs had failed. This reduced the risk of the two RCs for station blackout by 24% (Table 4-17).

Addition of a third diesel generator to lower the probability of station blackout would require the addition of a 6.4 MW diesel generator, its associated support systems, additional component cooling water piping to and from the diesel generator cooling water heat exchanger, an addition of a swing bus, additional cabling for connecting the diesel generator to the Electrical Distribution System, an additional diesel generator building to house the diesel, an additional fuel oil storage tank and storage tank structure, and additional HVAC systems for the diesel generator building and fuel oil storage tank structure. A study conducted for Duke Power Company's McGuire Nuclear Station estimates the cost of adding a similar swing diesel to be in excess of $25 Million. This McGuire study investigated the cost that other utilities incurred in installing additional diesel generators. Pennsylvania Power and Light installed a swing diesel at their Susquehanna plant. This job was originally bid at $30 Million; however, final installation ended up costing $130 Million. Northern States Power added additional diesel generators at the Prairie Island site. The initial bid for the project was $60 Million; however, the final price was around $78 Million. The cost estimate for an additional diesel was $18.4M to $19M in the Comanche Peak SAMDA. For this analysis the additional diesel will be estimated to cost $25 million.

### C2 TORNADO-PROTECTION FOR COMBUSTION TURBINE

The PDSs in Table 4-2 with the designator "TRND" are for tornados and it was assumed that offsite power was lost and the Combustion Turbine was not available. For these sequences, it was assumed that the DA completely protected the Combustion Turbine and it was available to supply AC with a failure rate[9] of 0.025/d. This reduced the risk of two RCs (Table 4-18).

The cost of this DA was estimated at over $3M and includes protection of the Combustion Turbine, fuel tank, and tunneling for cooling line. The cost could be as high as $4M depending on tunneling distances.

### C3 FUEL CELLS

In addition, alternative battery types to the traditional lead battery were investigated. Alternative battery types such as lithium or zinc are not commercially available in the necessary sizes to provide the capacity required by System 80+. Fuel cells are available in the size required for System 80+; however, they are not proven technologies in nuclear station applications and are not available as Class 1E equipment. In addition, the use of fuel

cells presents the problem of heat generation since a typical fuel cell will operate at a temperature of 300 to 1000 °C. HVAC systems would have to be capable of removing the heat. Also, a safety related fuel delivery and exhaust system would be required for each battery. Since fuel cells use combustible materials such as hydrogen or methane, their presence would increase the risk of fires in the plant. Design, development and installation of this type of fuel cell system would cost well over $2 million more than a conventional lead acid battery arrangement.

This DA addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. This DA is assumed to have the capability to remove decay heat using the turbine-driven feedwater pump for whatever time period that is required (without any failures). This DA prevents core damage and therefore removes two of the release classes (same as Alternative DC Batteries and EFWS, Table 4-9).

C4 HOOKUP FOR PORTABLE GENERATORS

Instead of increasing the battery capacity for the turbine-driven EFW pump train, portable generators could be brought in and hooked up for continued operation of the turbine-driven EFW pump train after the batteries are depleted. This would require temporary hookup connections so that the portable generators could be connected in a timely manner. These temporary hook up connections would need to be located in an area that was easily accessible for installing the portable generators and would have to be located in an appropriate environment for running the generators during station blackout conditions. The cost of adding these temporary hookup connections, including the cabling to an appropriate location for hookup, would be in excess of $10,000.

The System 80+ design includes numerous means of supplying feedwater to the steam generators. It has main feedwater runback, automatic actuation of the startup feedwater train, four emergency feedwater trains (two motor-driven and two steam-driven). The electric trains are supported by two diesel generators and a standby AC source (Combustion Turbine) in addition to the two sources of offsite power. In addition, the design has the capability to remove decay heat by a feed and bleed process using one of four safety injection pumps. Because of the redundancy in feedwater and feed and bleed capability, the use of the diesel-driven fire pump as an alternate feedwater source was not quantified because of the small risk reduction capability.

The DA addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. This DA prevents core damage and, therefore, removes two of the release classes (same as Alternative DC Batteries and EFWS, Table 4-9).

## D. ATWS AND EXTERNAL EVENTS, REDUCTION OF CDF

This group of Design Alternatives consists of ATWS prevention or mitigation alternatives and protection of some external events.

### D1 ALTERNATIVE ATWS PRESSURE RELIEF VALVES

This DA was selected because the System 80+ design uses an advanced digital Plant Protection System that has raised much interest. It consists of a system of relief valves that can prevent any equipment damage from a primary coolant pressure spike in an ATWS accident sequence. This DA is assumed to eliminate all the ATWS core damage sequences. ATWS does not show up as a dominant PDS but represents 3% of the CDF. Therefore, the risk of all release classes from transients was reduced by 3% (Table 4-19).

To implement this DA, the safety relief valve sizes and discharge piping size would need to be increased. It may also require additional safety relief valves and thus additional safety relief valve discharge piping and supports. In addition, the size and possibly the number of safety valve nozzles on top of the pressurizer would need to be increased. The cost of this DA is estimated to exceed $1 million.

### D2 ATWS INJECTION SYSTEM

An "ink" injection system was proposed for the planned federal heavy water New Production Reactor as a shutdown system diverse from the mechanical rods. Also, a foreign utility also showed some interest in this concept. Therefore, this DA was selected for evaluation. In terms of risk reduction benefits, this DA has the same advantage as the ATWS pressure relief valves (Table 4-19).

For estimating the cost of this DA, it was assumed that the RCP seal cooling pump could be used with existing sources of boron and existing piping and valves. The cost of this DA is $300,000 and is associated with the instrumentation and control system to activate the pump and align it.

### D3 DIVERSE PPS

This DA was selected because the System 80+ design uses an advanced digital Plant Protection System. A foreign utility also inquired about this DA. In this analysis, it was assumed that the redundant PPS eliminated all ATWS. The System 80+ has an Alternate Protection System (APS) in order to meet the ATWS rule. The APS contains an alternate scram system and a Diverse Emergency Feedwater Actuation System (DEFAS). The DA considered here is a third, diverse PPS to resolve Instrumentation and Control (I&C) diversity concerns. This DA has the same risk reduction as the

ATWS pressure relief valves (see Table 4-19). The cost of a diverse PPS was estimated to be $3,000,000.

### D4 SEISMIC CAPABILITY

The System 80+ Plant is designed for a Safe Shutdown Earthquake (SSE) of 0.3g acceleration. The Seismic Margins Analysis (Section 19.7.5 of CESSAR-DC) addresses the margins associated with the seismic design and demonstrates that the plant High Confidence of Low Probability of Failure (HCLPF) value is 0.6g acceleration. Therefore, there is a 95% confidence that existing equipment has less than a 5% probability of failure at twice the SSE level. To meet this stringent design goal, the containment design and SG support design may be modified. Recent Nuclear Regulatory Commission policy decisions state that ALWRs need to demonstrate a HCLPF only 0.5g. The seismic capability is considered adequate for the System 80+ design and no additional changes are considered.

### D5 FIRE AND FLOOD CAPABILITY

The System 80+ Plant is designed with four quadrants, two in each of two divisions with permanent barriers between the divisions. Also, sources of flooding were reduced in the annex building and drains were specifically designed to reduce flooding potential. These design features are described in Sections 9.5 (Fire Protection) and 3.4 (Flood Design) of CESSAR-DC. This capability is considered adequate for the System 80+ design and no additional changes are considered for fire and flood.

## E. DESIGN ALTERNATIVES TO REDUCE RADIOACTIVE RELEASES

The fifth group of Design Alternatives is the alternatives that were selected to protect the containment or reduce the release fractions. For the DAs that protect the containment, the releases are set to zero and then the risk is reevaluated. These DAs include the improved containment sprays, filtered vent, concrete composition, reactor vessel exterior cooling, and H2 igniters.

### E1 ALTERNATIVE CONTAINMENT SPRAY

An Alternative Containment Spray System is assumed to prevent the high pressure containment failures caused by slow steam pressurization (RC2.2M) and to eliminate the sequences where scrubbing does not occur. This system is assumed to have a perfect power supply and heat sink and work in all release classes where the containment is challenged regardless of the sequence of events or equipment failures that led to core damage and containment challenge. These assumptions overestimate the benefits of this DA. This DA also reduces the releases in all the release classes where no scrubbing of fission products was initially predicted. This DA reduces the risk of six of the release classes (Table 4-20).

The above analysis assumes that the system is completely successful in terminating the accident by protecting the containment. The benefit is proportional to the reliability of the system. For example, if the design had a conditional reliability of 0.5 in these accident sequences, then the DA would have half of the benefit. In addition, any annual operating costs (testing or maintenance) would have to be subtracted from the annual risk reduction benefits.

Estimating the cost to design and build a perfect Containment Spray System is not realistically possible. However, one option would be to provide piping from the containment spray header to the exterior of the Nuclear Annex for a temporary hook-up of a skid-mounted pump or fire truck should all containment spray and shutdown cooling pumps be unavailable. The cost of the additional Class 2 piping, pipe supports, valves, onsite pump or fire truck with the required pumping capacity and pump head and building to store the fire truck is estimated to exceed $1.5 million. This design modification has been included in the design.

E2 FILTERED VENT (CONTAINMENT)

The Filtered Vent DA prevents all slow high pressure containment failures and therefore reduces the doses in RC2.2M (Table 4-21).

The cost estimates for a Filtered Vent System ranges from $2.8 Million to $25 Million. IDCOR Technical Report 19.1, July 1983, estimated a cost of $25M for larger systems than our design and sized to handle ATWS. In the Advanced Boiling Water Reactor (ABWR) SAMDA, a cost of $3M was quoted. This is probably a smaller design taking credit for scrubbing in the BWR suppression pool.

The Comanche Peak SAMDA estimated the cost from $15M to $22.3M and the Limerick SAMDA gives a range from $2.8M to $11.3M. The System 80+ design estimate of $10M is for a non-ATWS sized, fully Category I facility and is bounded by the previous estimates.

E3 ALTERNATIVE CONCRETE COMPOSITION

The containment building for System 80+ uses a spherical containment with rooms below it that are part of the nuclear annex building. It is assumed that in accident sequences where corium/concrete interaction are not stopped, containment failure would lead to releases through the nuclear annex building. This DA assumes that an ideal concrete composition could be developed that prevents basemat melt-through. This would eliminate seven RCs where basemat melt-through is modeled (Table 4-22).

An advanced concrete composition to prevent corium/concrete interaction is not currently available. However, additional concrete could be added to increase the time before containment failure would occur. Currently, additional concrete can not be added to the reactor cavity, since there would be an interference with the incore instrumentation tubes which exit the bottom of the reactor vessel. In order to add an additional two feet of concrete the Nuclear Steam Supply System (NSSS) would have to be raised by two feet to avoid interference with the incore instrumentation tubes. Raising the NSSS would also require the crane wall height to be increased by two feet in order to have adequate clearance to lift the reactor head and service other NSSS components. In order to increase the crane wall height the containment diameter would have to be increased by approximately two feet in order to avoid an interference between the crane wall and containment vessel and to allow adequate space for spray coverage. An increase in containment diameter may also require an increase in containment plate thickness. An increase in containment plate thickness will require post-weld heat treatment for the construction of the containment vessel since the current thickness is at the limit allowed by the ASME Code before post-weld heat treatment is required. An increase in containment diameter will also require an increase in the diameter of the concrete shield building. The added cost for an additional two feet of concrete in the reactor cavity floor is relatively small. However, the added cost of additional steel for the increased containment diameter and thickness, post-weld heat treatment required for the increased containment plate thickness, additional concrete and rebar for the increase in crane wall height, and shield building diameter is estimated to exceed $5 million.

Because the dominant risks are associated with containment bypass events, the risk reduction associated with the additional thickness of the containment was not quantified. In events where no decay heat removal is available, the containment failure would still be postulated.

## E4 REACTOR VESSEL EXTERIOR COOLING

A reactor vessel exterior cooling system is assumed to prevent vessel melt-through and subsequent basemat attack or steam explosions. This DA reduces the consequences of eleven RCs (Table 4-23). The current arrangement for the In-Containment Refueling Water Storage Tank (IRWST) will not allow wetting of the

reactor vessel. The elevation of the IRWST was selected to ensure that wetting of the vessel would not occur should the holdup volume and cavity flood valves inadvertently open during power operation. This will prevent thermal shock of the vessel. However, water can be induced into the reactor cavity for exterior vessel cooling from external sources such as the Boric Acid Tank which provides a makeup source to the IRWST or by inducing water through the temporary hookup on the containment spray line discussed in DA E1 above and cost $1.5M. However, to utilize this option it must first be demonstrated that the reactor vessel will not breach due to thermal shock of the vessel from the cold water. The analysis to demonstrate this is estimated to cost $1M. This is based on the FERC prudence hearings for Yankee Atomic Electric Co. where it was reported that demonstration of vessel integrity would be a "multi-million dollar cost"[19]. The total cost would be $2.5 M.

If there was an inadvertent wetting of the reactor vessel during power, the event would require extensive testing and inspection before the plant would be permitted to startup. Such costs and additional economic risks have not been quantified but it is believed that these economic risks would far outweigh any advantage of vessel flooding.

E5 ALTERNATIVE H2 IGNITERS

Ideal hydrogen (H2) igniters would prevent release classes associated with containment failures from hydrogen burns or explosions. The System 80+ design has two different hydrogen control systems as described in Section 6.2.5 of CESSAR-DC. The Containment Hydrogen Recombiner System (CHRS) is designed to control the H2 concentrations in the containment following a LOCA. The CHRS prevents the concentration of hydrogen from reaching the lower flammability limit of 4% by volume in air or steam-air mixtures. During a degraded core accident, hydrogen will be produced at a greater rate than that of a design basis LOCA. The Hydrogen Mitigation System (HMS) is designed to accommodate the hydrogen produced from 100% fuel clad metal-water reaction and limit the average hydrogen concentration in the containment to a 10% for a degraded core accident. The HMS consists of 80 Glow Plug Igniters distributed throughout the containment. Their placement is based on a detailed assessment of the flow paths to fully cover all of the containment. Section 19.11.4.1.3 of CESSAR-DC discussed hydrogen igniters in severe accidents. System 80+ already has a degraded core H2 control system and only two release classes (RC2.1E and 2.2E) have containment failure from hydrogen burning. This DA reduces the risk of these RCs (Table 4-24).

Providing perfect hydrogen igniters which have no probability of failure is not possible. However, the reliability of the hydrogen igniters could be improved by either providing dedicated batteries for the existing design (Glow Plug Igniters) or by providing

catalytic hydrogen recombiners which do not require a power source. Since catalytic hydrogen recombiners are not fully developed, possible failure modes, including common cause failure modes, are not known. Therefore, they are not being selected for the System 80+ design at this time. The addition of dedicated batteries for the hydrogen igniters along with the additional equipment such as battery chargers and invertor and the additional building space to store this equipment is estimated to exceed $1 million. In the Comanche Peak SAMDA, additional batteries were estimated to cost between $1.3M and $3M and an ignition system was estimated to cost $5.8M to $8M.

E6 PASSIVE AUTOCATALYTIC RECOMBINERS (PARs)

Passive Autocatalytic Recombiners (PARs) are arrays of a palladium catalyst that will combine molecular hydrogen and oxygen gases into water. These units are currently in the development stage and have not been used in existing U.S. plants. They have a low conversion efficiency and, therefore, would have to be used in combination with existing H2 igniters. The advantage of the PARs is that they require no electrical power and, therefore, would operate during a station blackout. The success of the PARs to prevent a H2 burn would depend on the speed of the production and release of the H2. For this analysis, it was conservatively assumed that the PARs worked perfectly and, therefore, would prevent release classes associated with containment failures from hydrogen burns or explosions. The System 80+ design already has H2 igniters with redundant power backup via either DGs, batteries, or Combustion Turbine. Therefore, only two release classes (RC2.1E and 2.2E) have containment failure from hydrogen burning. This DA reduces the risk of these RCs (Table 4-24, Alternative H2 Igniters).

EPRI[20] has been developing PARs technology and estimates that 40 units would be needed for large dry containments. EPRI estimates the units would cost $19,000 each, and the cost for the PARs in System 80+ would be $760,000. This costs neglects any annual costs of cleaning, inspection and testing. Also, both NRC and the Advisory Committee on Reactor Safeguards (ACRS) have expressed concern about the expected relative slow response time of the PARs.

E7 MSSV AND ADV SCRUBBING

The discharges of the Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs) could be scrubbed by routing the discharges through a structure with a water spray condense the steam and remove most of the fission products. This DA was introduced to specifically address a SGTR, where isolation fails (the largest three RCs). Table 4-25 gives the risk reduction of this DA.

This modification would require building a structure over the valve discharges and installing a header system to distribute water. In addition, a pump, piping, water supply and instrumentation and drain system would be needed. Conceptually, this system is similar to a Containment Spray System for which a cost estimate of $9.5M was given in the Commanche Peak SAMDA analysis, and that cost estimate will be used in this analysis.

E8 ALTERNATIVE CONTAINMENT MONITORING SYSTEM

The alternative containment monitoring system was selected to address the RCs where containment bypass is predicted. It does not address a SGTR, where failure to isolate the SG is predicted. This DA is assumed to eliminate the containment bypass RC4.8E and the Interfacing Systems LOCA RC5.1E (Table 4-26).

This modification would require the addition of a redundant and diverse limit switch to each containment isolation valve, and the addition of control and fiber optic cabling to the plant computer. The cost of this modification would be in excess of $1 million. In the Comanche Peak SAMDA, an alternative bypass instrumentation system was estimated to cost $2.7M.

E9 CAVITY COOLING

The Cavity Cooling DA uses the existing SCS heat exchangers in the IRWST to cool the reactor vessel cavity under natural circulation. It uses existing piping and equipment but only increases the size of the pipes to ensure that natural circulation is effective. In the upper limit as modeled here, this DA assumes the existing SCS equipment always works and it is assumed to eliminate vessel failure, steam explosions and concrete interactions. It has the same advantages and risk reduction worth as the reactor vessel cooling system (DA E4, Table 4-23) but has a lower capital cost because it uses existing equipment.

This modification would require increasing the size of the existing cavity flood lines and performing analysis that there is adequate mixing between the reactor cavity and IRWST. The cost of this modification would be in excess of $50,000. In the Comanche Peak SAMDA, an Alternative Cavity Flooding System was estimated to cost between $1.2M and $2.3M.

E10 VENTING THE MSSV IN CONTAINMENT

System 80+ design does not divert MSSV steam releases back to the containment. While such a system would reduce radiological releases to the environment for selected accident scenarios, such a system does not significantly reduce public risk and does carry

several disadvantages. It should be noted that this feature does not eliminate releases to the environment.

The technical disadvantages of the MSSV containment steam return system are summarized below for two hypothetical systems. In the first system, the steam is simply returned to the containment atmosphere. In the second system, the steam is discharged into the IRWST where it would be condensed.

Direct discharge of MSSV into containment has several serious disadvantages:

1. The secondary system return will place an additional loading burden on the containment and restrict plant operators in responding to accidents when containment sprays are unavailable. This could lead to the addition of a containment vent to address those concerns which in itself introduces another means of inadvertent containment bypass;

2. Any condensed steam discharge will drain to the IRWST, diluting the boron concentration. A minimum IRWST boron concentration for safety injection is necessary for mitigating LOCA and non-LOCA events; and

3. The release to containment atmosphere has the potential to cause personal injury.

An MSSV return system directed to the IRWST has similar drawbacks to Items 1 and 2 described above and poses the additional complication that discharge of steam flows typical of the MSSVs may produce excessive loadings within the IRWST.

Either return path would require a major redesign effort and increase design complexity, each of which is not consistent with evolutionary ALWR goals. Also, this provision will not eliminate radiological releases to the environment from a SGTR.

In summary, including a MSSV discharge return to the containment was evaluated in design considerations. Based on this review, this DA poses serious design drawbacks. ABB-CE does not believe that the secondary steam should be piped and vented inside containment. These events are characterized as Anticipated Operational Occurrence (AOO) events, and filling the containment with steam during these events would be both damaging to the equipment and dangerous to operators. A cost/benefit analysis was not performed for this DA because as it would require an assessment of equipment degradation, injuries, and loss of plant availability after secondary side venting into containment.

E11 <u>HYDROGEN PURGE LINE</u>

An existing System 80+ design feature that could be utilized in venting the containment is the hydrogen purge vent. System 80+ is

equipped with two 3-inch diameter hydrogen purge vents which can be used for purposes of containment venting. This design feature is shown in Figure 6.2.5-1 of CESSAR-DC. The vents are intended for use in post-LOCA conditions for diverting hydrogen to the secondary containment (annulus), should the hydrogen recombiners be inoperative. The Annulus Ventilation System then collects and filters the secondary containment atmosphere before release.

An analysis of the potential application of the venting capabilities of the hydrogen purge piping was performed using the MAAP computer code. This analysis conservatively simulated hydrogen purge as a 0.049 $ft^2$ equivalent area opening in the containment. A hypothetical accident management strategy was considered in which the hydrogen purge system is used to vent containment when the containment pressure reaches 80 psia. This will enable maintenance of the containment pressure well below the containment failure threshold.

Since there are four AC electric motor-operated valves in series on each division that must be opened to purge the containment, and since the Annulus Ventilation System requires AC power for operation, this feature can not be credited for mitigating severe accidents resulting from a complete loss of AC power. This DA has already been included in the System 80+ design and no cost benefit analysis is necessary.

E12 <u>WATER COOLED RUBBLE BED</u>

The purpose of the water cooled rubble bed is to achieve a coolable debris bed below the vessel and remove decay heat. This DA consists of a floodable rubble bed in the bottom of the vessel cavity. The rubble bed would be kept dry until the corium had penetrated into it, thus minimizing the potential for steam explosion. This DA would have the same risk reduction potential as the ideal, Alternative Concrete Composition (DA E3). This DA would eliminate seven RCs where basemat melt-through is modeled (see Table 4-22).

The cost of the water cooled rubble bed is estimated[6] to be between $35.5 and $38.5 Million. Another source[14] estimated the cost to be $18.8 Million. Neither source included the cost of actually developing the system. Periodic testing and maintenance of the device could be significant. For this analysis, the lower cost of $18.8M was used.

E13 <u>REFRACTORY LINED CRUCIBLE</u>

The purpose of the refractory lined crucible is to achieve a coolable debris bed below the vessel and remove decay heat. This DA consists of a ceramic lined crucible with cooling located in the

vessel cavity. This DA would have the same risk reduction potential as the ideal, Alternative Concrete Composition (DA E3). This DA would eliminate seven RCs where basemat melt-through is modeled (Table 4-22).

The cost of the refractory linned crucible is estimated[6] to be between $108 and $119 Million. Another source[14] estimated the cost to be $18.8 Million. Neither source included the cost of actually developing the system. Periodic testing and maintenance of the device which could be significant. For this analysis, the lower cost of $108M was used.

E14 VACUUM BUILDING

ABB-CE developed a conceptual design for a vacuum building which was designed to reduce emissions from severe accidents and is described in Reference 21. The cost was estimated as $30 M in 1983. A separate IDCOR sponsored study (IDCOR Technical Report 19.1, July, 1983) also estimated the cost to be $30M (approximately $42M in 1993 dollars). Because of the high costs, and because most of the significant releases are bypass events for which the vacuum building would not help, this DA was not quantified.

E15 RIBBED CONTAINMENT

A ribbed containment was considered to address increased capacity against potential failure of the containment from buckling during a seismic event coupled with an inadvertent actuation of the containment spray. This conservative combination of events would lead to a negative pressure in containment and increase the potential for buckling. However, the free standing unreinforced containment design meets or exceeds all design criteria for this design basis event. Also, the probability of simultaneous occurrence of maximum seismic excitation and maximum negative pressure is exceedingly low. The cost of this DA is in the tens of millions of dollars because the ribs complicate manufacturing and construction and would require field heat treating. Given that this DA has a high cost and no appreciable benefit, it was not further quantified.

| TABLE 4-1<br><br>**SUMMARY DESCRIPTION OF SYSTEM 80+ RELEASE CLASSES** | |
|---|---|
| Release Class | Release Class Definition |
| RC1.1E | Early core melt, intact containment, annulus filtering, Core damage less than 8 hrs |
| RC1.1M | Mid core melt, intact containment, annulus filtering, Core damage 8 to 24 hrs |
| RC2.1E | Early core melt, late containment failure, in-vessel scrubbing, no vaporization or revaporization releases, core damage less than 8 hrs, H2 burn |
| RC2.2E | Early core melt, in-vessel scrubbing, no vaporization releases, revaporization releases, releases scrubbed, late containment failure, H2 burn |
| RC2.4E | Early core melt, late containment failure, in-vessel scrubbing, vaporization releases, no revaporization releases, releases scrubbed, basemat melt-thru |
| RC2.5E | Early core melt, late containment failure, in-vessel scrubbing, vaporization releases, no revaporization releases, releases not scrubbed, basemat melt-thru |
| RC2.6E | Early core melt, late containment failure, in-vessel scrubbing, vaporization releases and revaporization releases, releases scrubbed, basemat melt-thru |
| RC2.7E | Early core melt, late containment failure, in-vessel scrubbing, vaporization releases and revaporization releases, revaporization releases scrubbed, vaporization releases not scrubbed, basemat melt-thru |
| RC2.2M | Mid core melt, late containment failure, in-vessel scrubbing, no vaporization releases, revaporization releases, releases scrubbed, CSS fails, steam explosion |
| RC2.5M | Mid core melt, late containment failure, in-vessel scrubbing, vaporization releases, no revaporization releases, releases not scrubbed, Basemat melt-thru |

**TABLE 4-1**

**SUMMARY DESCRIPTION OF SYSTEM 80+ RELEASE CLASSES**

| Release Class | Release Class Definition |
|---|---|
| RC2.6M | Mid core melt, late containment failure, in-vessel scrubbing, vaporization releases and revaporization releases, releases scrubbed, basemat melt-thru |
| RC2.7M | Mid core melt, late containment failure, in-vessel scrubbing, vaporization releases and revaporization releases, revaporization releases scrubbed, vaporization releases not scrubbed, basemat melt-thru |
| RC3.1E | Early core melt, early containment failure, in-vessel scrubbing, no vaporization or revaporization releases, steam explosion |
| RC3.2E | Early core melt, early containment failure, in-vessel scrubbing, no vaporization release, revaporization releases, releases scrubbed, steam explosion |
| RC3.4E | Early core melt, early containment failure, in-vessel scrubbing, vaporization release, no revap releases, releases scrubbed, steam explosion |
| RC3.6E | Early core melt, early containment failure, in-vessel scrubbing, vaporization release, revaporization releases, releases scrubbed, steam explosion |
| RC3.2M | Mid core melt, early containment failure, in-vessel scrubbing, no vaporization release, revap. releases, releases scrubbed, CSS failed, steam explosion |
| RC3.6M | Mid core melt, early containment failure, in-vessel scrubbing, vaporization release, revap. releases, releases scrubbed, CSS failed, steam explosion |
| RC4.22E | Early core melt(SGTR), in-vessel scrubbing, vaporization releases, no revaporization releases, releases scrubbed, isolation failure |

TABLE 4-1

SUMMARY DESCRIPTION OF SYSTEM 80+ RELEASE CLASSES

| Release Class | Release Class Definition |
|---|---|
| RC4.8E | Early core melt, isolation failure, in-vessel scrubbing, vaporization releases, revaporization releases, vaporization releases scrubbed, revaporization releases not scrubbed |
| RC4.30E | Early core melt (SGTR), isolation failure, in-vessel scrubbing, vaporization releases, revaporization releases, releases scrubbed |
| RC4.36L | Late core melt (SGTR), isolation failure, in-vessel scrubbing, vaporization releases, revaporization releases, releases not scrubbed |
| RC5.1E | Early core melt, containment bypass, vaporization releases, releases scrubbed/attenuated in auxiliary building |

## TABLE 4-2

## MAPPING SEQUENCES INTO RELEASE CLASSES

| RC | RC FREQ | PDS | SEQ | DESCRIPTION | SEQ FREQ* |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-6 | PDS235 | LSSB-9A | (LSSB)(Safety Injection OK)(Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails) | 1.8E-09 |
| | | | LOFW-9A | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 3.7E-07 |
| | | | TOTH-9A | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 1.9E-09 |
| | | | LOOP-9A | (LOOP)(Failure to Deliver Emergency Feedwater) (Safety Depressurization for Bleed Fails) | 3.3E-09 |
| | | PDS184 | SGTR-16A | (SGTR) (Safety Injection Fails) (Aggressive Cooldown OK) (SCS Injection Fails) | 1.3E-08 |
| | | | SGTR-17A | (SGTR) (Injection Fails) (Aggressive Secondary Cooldown Fails) | 2.5E-07 |
| | | PDS3 | LL-3A | (LLOCA) (SITs Inject OK) (Safety Injection Fails) | 1.0E-07 |
| | | | LL-4A | (LLOCA) (SITs Fail to Inject) | 4.4E-09 |
| | | | VR-A | (Vessel Rupture) | 1.0E-07 |
| | | PDS85 | ML2-3A | (Medium LOCA 2) (Safety Injection Fails) | 1.5E-07 |
| | | PDS201 | SL-11A | (SLOCA) (Safety Injection Fails) (Aggressive Cooldown Fails) | 1.5E-07 |
| RC1.1M | 3.83E-7 | PDS148 | LOFW-4E | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 1.5E-09 |
| | | | TOTH-4E | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 2.3E-09 |
| | | | TRND-4E | (Tornado) (PSV Reseats) (EFWS OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.2E-07 |
| | | PDS136 | LOFW-4A | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 3.2E-08 |
| | | | TOTH-4A | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 6.3E-08 |
| | | | TRND-4A | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.0E-08 |

## TABLE 4-2

## MAPPING SEQUENCES INTO RELEASE CLASSES

| RC | RC FREQ | PDS | SEQ | DESCRIPTION | SEQ FREQ* |
|---|---|---|---|---|---|
| RC2.1E | 3.44E-9 | PDS3 | LL-3A | (LLOCA) (SITs Inject OK) (Safety Injection Fails) | 1.0E-07 |
| | | | LL-4A | (LLOCA) (SITs Fail to Inject) | 4.4E-09 |
| | | | VR-A | (Vessel Rupture) | 1.0E-07 |
| | | PDS235 | LSSB-9A | (LSSB) (Safety Injection OK) (Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails) | 1.8E-09 |
| | | | LOFW-9A | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 3.7E-07 |
| | | | TOTH-9A | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 2.9E-09 |
| | | | LOOP-9A | (LOOP) (Failure to Deliver Emergency Feedwater) (Safety Depressurization for Bleed Fails) | 3.3E-09 |
| RC2.2E | 3.31E-9 | PDS201 | SL-11A | (SLOCA) (Safety Injection Fails) (Aggressive Cooldown Fails) | 1.5E-07 |
| | | PDS235 | LSSB-9A | (LSSB) (Safety Injection OK) (Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails) | 1.8E-09 |
| | | | LOFW-9A | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 3.7E-07 |
| | | | TOTH-9A | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 1.9E-09 |
| | | | LOOP-9A | (LOOP) (Failure to Deliver Emergency Feedwater) (Safety Depressurization for Bleed Fails) | 3.3E-09 |
| RC2.4E | 3.76E-8 | PDS233 | LOFW-9B | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 4.6E-07 |
| | | PDS83 | ML2-3B | (Medium LOCA 2) (Safety Injection Fails) | 7.4E-09 |
| | | PDS18 | ML1-3B | (Medium LOCA 1) (Safety Injection Fails) | 6.6E-09 |
| | | PDS3 | LL-3A | (LLOCA) (SITs Inject OK) (Safety Injection Fails) | 1.0E-07 |
| | | | LL-4A | (LLOCA) (SITs Fail to Inject) | 4.4E-09 |
| | | | VR-A | (Vessel Rupture) | 1.0E-07 |
| | | PDS1 | LL-3B | (LLOCA) (SITs Inject OK) (Safety Injection Fails) | 1.1E-07 |

TABLE 4-2

MAPPING SEQUENCES INTO RELEASE CLASSES

| RC | RC FREQ | PDS | SEQ | DESCRIPTION | SEQ FREQ* |
|---|---|---|---|---|---|
| RC2.5E | 2.84E-8 | PDS241 | LOFW-9F | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 4.3E-08 |
| RC2.6E | 3.33E-8 | PDS181 | SGTR-17B | (SGTR) (Injection Fails) (Aggressive Secondary Cooldown Fails) | 1.3E-08 |
| | | PDS199 | SL-11B | (SLOCA) (Safety Injection Fails) (Aggressive Cooldown Fails) | 8.2E-09 |
| | | PDS233 | LOFW-9B | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 1.9E-08 |
| RC2.7E | 1.62E-8 | PDS241 | LOFW-9F | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 4.3E-08 |
| RC2.2M | 4.05E-9 | PDS148 | LOFW-4E | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 1.5E-09 |
| | | | TOTH-4E | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 2.3E-09 |
| | | | TRND-4E | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.2E-07 |
| | | PDS136 | LOFW-4A | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 3.2E-08 |
| | | | TOTH-4A | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 6.3E-08 |
| | | | TRND-4A | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.0E-08 |
| RC2.5M | 3.95E-9 | PDS242 | SBOBD-F | (Station Blackout with Battery Depletion) | 3.4E-09 |
| | | | TRND-SBF | (Tornado) (Station Blackout with Battery Depletion) | 2.9E-09 |
| RC2.6M | 9.08E-9 | PDS134 | LOFW-4B | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 1.9E-09 |

**TABLE 4-2**

**MAPPING SEQUENCES INTO RELEASE CLASSES**

| RC | RC FREQ | PDS | SEQ | DESCRIPTION | SEQ FREQ* |
|---|---|---|---|---|---|
| | | | TOTH-4B | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 3.0E-09 |
| | | PDS148 | LOFW-4E | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 1.5E-09 |
| | | | TOTH-4E | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 2.3E-09 |
| | | | TRND-4E | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.2E-07 |
| RC2.7M | 1.22E-8 | PDS145 | TRND-4F | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 1.0E-08 |
| | | PDS242 | SBGBD-F | (Station Blackout with Battery Depletion) | 3.4E-09 |
| | | | TRND-SBF | (Tornado) (Station Blackout with Battery Depletion) | 2.9E-09 |
| RC3.1E | 4.64E-9 | PDS235 | LSSB-9A | (LSSB) (Safety Injection OK) Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails) | 1.8E-09 |
| | | | LOFW-9A | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 3.7E-07 |
| | | | TOTH-9A | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 1.9E-09 |
| | | | LOOP-9A | (LOOP) (Failure to Deliver Emergency Feedwater) (Safety Depressurization for Bleed Fails) | 3.3E-09 |
| | | PDS85 | ML2-3A | (Medium LOCA 2) (Safety Injection Fails) | 1.5E-07 |
| | | PDS3 | LL-3A | (LLOCA) (SITs Inject OK) (Safety Injection Fails) | 1.0E-07 |
| | | | LL-4A | (LLOCA) (SITs Fail to Inject) | 4.4E-09 |
| | | | VR-A | (Vessel Rupture) | 1.0E-07 |
| RC3.2E | 3.16E-9 | PDS184 | SGTR-16A | (SGTR) (Safety Injection Fails) (Aggressive Cooldown OK) (SCS Injection Fails) | 1.3E-08 |
| | | | SGTR-17A | (SGTR) (Safety Injection Fails) (Aggressive Secondary Cooldown Fails) | 2.5E-07 |

**TABLE 4-2**

**MAPPING SEQUENCES INTO RELEASE CLASSES**

| RC | RC FREQ | PDS | SEQ | DESCRIPTION | SEQ FREQ* |
|---|---|---|---|---|---|
| | | PDS235 | LSSB-9A | (LSSB) (Safety Injection OK) (Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails) | 1.8E-09 |
| | | | LOFW-9A | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 3.7E-07 |
| | | | TOTH-9A | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 1.9E-09 |
| | | | LOOP-9A | (LOOP) (Failure to Deliver Emergency Feedwater) (Safety Depressurization for Bleed Fails) | 3.3E-09 |
| RC3.4E | 6.52E-9 | PDS235 | LSSB-9A | (LSSB) (Safety Injection OK) (Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails) | 1.8E-09 |
| | | | LOFW-9A | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 3.7E-07 |
| | | | TOTH-9A | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 1.9E-09 |
| | | | LOOP-9A | (LOOP) (Failure to Deliver Emergency Feedwater) (Safety Depressurization for Bleed Fails) | 3.3E-09 |
| | | PDS20 | ML1-3A | (Medium LOCA 1) (Safety Injection Fails) | 1.3E-07 |
| | | PDS3 | LL-3A | (LLOCA) (SITs Inject OK) (Safety Injection Fails) | 1.0E-07 |
| | | | LL-4A | (LLOCA) (SITs Fail to Inject) | 4.4E-09 |
| | | | VR-A | (Vessel Rupture) | 1.0E-07 |
| | | PDS85 | ML2-3A | (Medium LOCA 2) (Safety Injection Fails) | 1.5E-07 |
| RC3.6E | 3.21E-9 | PDS184 | SGTR-16A | (SGTR) (Safety Injection Fails) (Aggressive Cooldown OK) (RHR Injection Fails) | 1.3E-08 |
| | | | SGTR-17A | (SGTR) (Injection Fails) (Aggressive Secondary Cooldown Fails) | 2.5E-07 |
| | | PDS235 | LSSB-9A | (LSSB) (Safety Injection OK) (Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails) | 1.8E-09 |
| | | | LOFW-9A | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 3.7E-07 |
| | | | TOTH-9A | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 1.9E-09 |

TABLE 4-2

MAPPING SEQUENCES INTO RELEASE CLASSES

| RC | RC FREQ | PDS | SEQ | DESCRIPTION | SEQ FREQ* |
|---|---|---|---|---|---|
| | | | LOOP-9A | (LOOP) (Failure to Deliver Emergency Feedwater) (Safety Depressurization for Bleed Fails) | 3.3E-09 |
| RC3.2M | 1.82E-9 | PDS148 | LOFW-4E | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 1.5E-09 |
| | | | TOTH-4E | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 2.3E-09 |
| | | | TRND-4E | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.2E-07 |
| | | PDS136 | LOFW-4A | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 3.2E-08 |
| | | | TOTH-4A | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 6.3E-08 |
| | | | TRND-4A | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.0E-08 |
| RC3.6M | 1.83E-9 | PDS148 | LOFW-4E | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 1.5E-09 |
| | | | TOTH-4E | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 2.3E-09 |
| | | | TRND-4E | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.2E-07 |
| | | PDS136 | LOFW-4A | (LOFW) (Emergency Feedwater OK) (Long-Term Decay Heat Removal Fails) (Bleed OK) (Safety Injection for Feed Fails) | 3.2E-08 |
| | | | TOTH-4A | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (Safety Injection for Feed Fails) | 6.3E-08 |
| | | | TRND-4A | (Tornado) (PSV Reseats) (EFW OK) (LTDHR Fails) (Bleed OK) (Feed Fails) | 2.0E-08 |

TABLE 4-2

MAPPING SEQUENCES INTO RELEASE CLASSES

| RC | RC FREQ | PDS | SEQ | DESCRIPTION | SEQ FREQ* |
|---|---|---|---|---|---|
| RC4.22E | 5.99E-9 | PDS184 | SGTR-16A | (SGTR) (Safety Injection Fails) (Aggressive Cooldown OK) (RHR Injection Fails) | 1.3E-08 |
| | | | SGTR-17A | (SGTR) (Injection Fails) (Aggressive Secondary Cooldown Fails) | 2.5E-07 |
| RC4.8E | 1.06E-9 | PDS235 | LSSB-9A | (LSSB) (Safety Injection OK) (Failure to Deliver Feedwater) (Safety Depressurization for Bleed Fails | 1.8E-09 |
| | | | LOFW-9A | (LOFW) (Emergency Feedwater Fails) (Safety Depressurization for Bleed Fails) | 3.7E-07 |
| | | | TOTH-9A | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 1.9E-09 |
| | | | LOOP-9A | (LOOP) (Failure to Deliver Emergency Feedwater) (Safety Depressurization for Bleed Fails) | 3.3E-09 |
| | | PDS20 | ML1-3A | (Medium LOCA 1) (Safety Injection Fails) | 1.3E-07 |
| | | PDS3 | LL-3A | (LLOCA) (SITs Inject OK) (Safety Injection Fails) | 1.0E-07 |
| | | | LL-4A | (LLOCA) (SITs Fails to Inject) | 4.4E-09 |
| | | | VR-A | (Vessel Rupture) | 1.0E-07 |
| | | PDS85 | ML2-3A | (Medium LOCA 2) (Safety Injection Fails) | 1.5E-07 |
| RC4.30E | 6.55E-9 | PDS184 | SGTR-16A | (SGTR) (Safety Injection Fails) (Aggressive Cooldown OK) (RHR Injection Fails) | 1.3E-08 |
| | | | SGTR-17A | (SGTR) (Injection Fails) (Aggressive Secondary Cooldown Fails) | 2.5E-07 |
| | | PDS181 | SGTR-17B | (SGTR) (Injection Fails) (Aggressive Secondary Cooldown Fails) | 1.3E-08 |
| RC4.36L | 3.08E-8 | PDS194 | SGTR-9F | (SGTR) (Safety Injection OK) (Deliver Feedwater OK) (RCS Pressure Control Fails) (SG not Isolated) Failure to Refill IRWST) | 3.1E-08 |
| RC5.1E | 5.10E-10 | | ISLOCA | (Failure of Check & Isolation Valves in One SCS Line) | 5.1E-10 |

* Frequency for Core Damage (Level 1)

## TABLE 4-3

## RANKING OF RELEASE CLASSES BY OFFSITE RISK

| Rank | Release Class | Frequency Events/yr | Mean Dose mr/event | Dose Risk mr/yr |
|---|---|---|---|---|
| 1 | RC4.36L | 3.08E-08 | 5.90E+06 | 1.82E-01 |
| 2 | RC4.30E | 6.55E-09 | 5.07E+06 | 3.32E-02 |
| 3 | RC4.22E | 5.99E-09 | 5.24E+06 | 3.14E-02 |
| 4 | RC3.4E | 6.52E-09 | 1.20E+06 | 7.82E-03 |
| 5 | RC3.1E | 4.64E-09 | 1.02E+06 | 4.73E-03 |
| 6 | RC3.2E | 3.16E-09 | 1.32E+06 | 4.17E-03 |
| 7 | RC3.6E | 3.21E-09 | 1.27E+06 | 4.08E-03 |
| 8 | RC3.6M | 1.83E-09 | 1.97E+06 | 3.61E-03 |
| 9 | RC3.2M | 1.82E-09 | 1.81E+06 | 3.29E-03 |
| 10 | RC2.7M | 1.22E-0[illegible] | 1.38E+05 | 1.68E-03 |
| 11 | RC5.1E | 5.10E-10 | 2.87E+06 | 1.46E-03 |
| 12 | RC2.4E | 3.76E-08 | 2.38E+04 | 8.95E-04 |
| 13 | RC2.6E | 3.33E-08 | 2.35E+04 | 7.83E-04 |
| 14 | RC2.5E | 2.84E-08 | 2.35E+04 | 6.67E-04 |
| 15 | RC2.2M | 4.05E-09 | 1.31E+05 | 5.31E-04 |
| 16 | RC2.1E | 3.44E-09 | 1.37E+05 | 4.71E-04 |
| 17 | RC2.2E | 3.31E-09 | 1.37E+05 | 4.53E-04 |
| 18 | RC2.7E | 1.62E-08 | 2.35E+04 | 3.81E-04 |
| 19 | RC2.6M | 9.08E-09 | 3.02E+04 | 2.74E-04 |
| 20 | RC4.8E | 1.06E-09 | 1.86E+05 | 1.97E-04 |
| 21 | RC2.5M | 3.95E-09 | 4.73E+04 | 1.87E-04 |
| 22 | RC1.1E | 1.36E-06 | 1.19E+02 | 1.62E-04 |
| 23 | RC1.1M | 3.83E-07 | 1.09E+02 | 4.17E-05 |
| | | SUM = 1.96E-06 | | 2.82E-01 |

**TABLE 4-4**

**RANKING OF SEQUENCES BY CORE DAMAGE FREQUENCY (CDF)**

| SEQUENCE CODE | SEQUENCE | CDF EV/YEAR |
|---|---|---|
| LOFW-9 | (LOFW) (Emergency Feedwater Fails) (SDS for Bleed Fails) | 4.4E-7 |
| SGTR-17 | (SGTR) (Injection Fails) (Aggressive Secondary Cooldown Fails) | 2.7E-7 |
| SL-11 | (SLOCA) (Safety Injection Fails) (Aggressive Cooldown Fails) | 1.6E-7 |
| ML2-3 | (Medium LOCA 2) (Safety Injection Fails) | 1.6E-7 |
| ML1-3 | (Medium LOCA 1) (Safety Injection Fails) | 1.4E-7 |
| LL-3 | (LLOCA) (SITs Inject OK) (Safety Injection Fails) | 1.1E-7 |
| VR | Vessel Rupture | 1.0E-7 |
| TOTH-4 | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (SIS for Feed Fails) | 6.9E-8 |
| ATWS-29 | (ATWS) (Adverse MTC) | 4.7E-8 |
| LOFW-4 | (LOFW) (Emergency Feedwater OK) (Long-Term DHR Fails) (Bleed OK) (SIS for Feed Fails) | 3.6E-8 |
| SGTR-9 | (SGTR) (Safety Injection OK) (EFW OK)(Unisolable Leak in Ruptured SG) (Failure to Refill IRWST) | 3.1E-8 |
| TOTH-12 | (TOTH) (PSV Fails to Reseat) (SI Injection Fails) | 2.7E-8 |
| SBO | Station Blackout with Battery Depletion | 2.1E-8 |
| LOFW-8 | (LOFW) (Emergency Feedwater Fails) (Bleed OK) (Safety Injection for Feed Fails) | 2.1E-8 |
| SGTR-16 | (SGTR) (Safety Injection Fails) (Aggressive Cooldown OK) (RHR Injection Fails) | 1.5E-8 |
| SL-10 | (SLOCA) (Safety Injection Fails) (Aggressive Cooldown) (RHR Injection Fails) | 9.0E-9 |
| SL-4 | (SLOCA) (Safety Injection OK) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (SDS Fails) | 8.9E-9 |
| TOTH-5 | (Other Transients) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (SDS Fails) | 6.9E-9 |
| SGTR-12 | (SGTR) (Safety Injection OK) (Feedwater Fails) (SDS - Bleed Fails) | 6.9E-9 |
| LOFW-5 | (LOFW) (Emergency Feedwater OK) (Long-Term DHR Fails) (SDS for Bleed Fails) | 5.1E-9 |
| LL-4 | (LLOCA) (SITs Fail to Inject) | 4.7E-9 |
| LOOP-9 | (LOOP) (Failure to Deliver Emergency Feedwater) (SDS for Bleed Fails) | 3.9E-9 |
| LHV-5 | (LHVAC) (Deliver Feedwater OK) (Long-Term Decay Heat Removal Fails) (SDS for Bleed Fail) | 4.0E-9 |
| ATWS-9 | (ATWS) (PSVs Open & Reclose OK) (No Consequential SGTR) (Deliver Feedwater OK) (Failure to Borate by Charging Pumps) (Safety Depressurization Fails) | 2.9E-9 |
| TOTH-9 | (Other Transients) (Feedwater Fails) (Safety Depressurization Fails) | 2.3E-9 |
| LSSB-9 | (LSSB) (Safety Injection OK) (EFW Failure) (Safety Depressurization for Bleed Fails) | 2.1E-9 |

## TABLE 4-5

## DESIGN ALTERNATIVES CONSIDERED

| | DESIGN ALTERNATIVE | CATEGORY* |
|---|---|---|
| 1. | LARGER PRESSURIZER | 1 |
| 2. | LARGER STEAM GENERATORS | 1 |
| 3. | HIGH-PRESSURE SHUTDOWN COOLING SYSTEM (SCS) | 1 |
| 4. | FUNCTIONALLY INTERCHANGEABLE SCS AND CONTAINMENT SPRAY SYSTEM (CSS) PUMPS | 1 |
| 5. | MULTIPLE INDEPENDENT CONNECTIONS TO THE GRID | 1 |
| 6. | TURBINE-GENERATOR RUNBACK CAPABILITY | 1 |
| 7. | DEDICATED STARTUP FEEDWATER SYSTEM | 1 |
| 8. | IMPROVED CONTROL ROOM DESIGN | 1 |
| 9. | IMPROVED NORMALLY OPERATING COMPONENT COOLING WATER SYSTEM (CCWS)/STATION SERVICE WATER SYSTEM (SSWS) | 1 |
| 10. | FOUR TRAIN SAFETY INJECTION SYSTEM (SIS) WITH DIRECT VESSEL INJECTION | 1 |
| 11. | SAFETY DEPRESSURIZATION SYSTEM (SDS) | 1 |
| 12. | FOUR TRAIN EMERGENCY FEEDWATER SYSTEM | 1 |
| 13. | TWO EMERGENCY DIESEL GENERATORS AND A STANDBY ALTERNATE AC SOURCE (COMBUSTION TURBINE) | 1 |
| 14. | SIX VITAL BATTERIES | 1 |

**TABLE 4-5**

**DESIGN ALTERNATIVES CONSIDERED**

| | DESIGN ALTERNATIVE | CATEGORY* |
|---|---|---|
| 15. | IMPROVED CONTROL ROOM DESIGN | 1 |
| 16. | CROSS-CONNECTED CSS AND SCS TRAINS | 1 |
| 17. | IMPROVED CONTROL ROOM DESIGN | 1 |
| 18. | LARGE SPHERICAL CONTAINMENT | 1 |
| 19. | REACTOR CAVITY DESIGNED FOR CORIUM DISENTRAINMENT | 1 |
| 20. | REACTOR CAVITY DESIGNED FOR DEBRIS COOLABILITY | 1 |
| 21. | IRWST AND SDS INTERCONNECTED | 1 |
| 22. | HYDROGEN MITIGATION SYSTEM | 1 |
| 23. | ALTERNATIVE CONTAINMENT SPRAY (E1)** | 2 |
| 24. | FILTERED VENT (E2) | 2 |
| 25. | ALTERNATIVE DC BATTERIES AND EFWS (B1) | 2 |
| 26. | RCP SEAL COOLING (A1) | 1 |
| 27. | ALTERNATIVE PRESSURIZER AUXILIARY SPRAY(B3) | 2 |
| 28. | ALTERNATIVE ATWS PRESSURE RELIEF VALVES(D1) | 2 |
| 29. | ALTERNATIVE CONCRETE COMPOSITION (E3) | 2 |
| 30. | REACTOR VESSEL EXTERIOR COOLING (E4) | 2 |
| 31. | ALTERNATIVE H2 IGNITERS (E5) | 2 |
| 32. | ALTERNATIVE HIGH PRESSURE SAFETY INJECTION (B4) | 2 |
| 33. | ALTERNATIVE RCS DEPRESSURIZATION (B5) | 2 |
| 34. | 100% SG INSPECTION (A2) | 2 |
| 35. | MSSV SCRUBBING (E7) | 2 |

**TABLE 4-5**

**DESIGN ALTERNATIVES CONSIDERED**

| | DESIGN ALTERNATIVE | CATEGORY* |
|---|---|---|
| 36. | THIRD DIESEL GENERATOR (C1) | 2 |
| 37. | BORON INJECTION SYSTEM (ATWS) (D2) | 2 |
| 38. | DIVERSE PPS (D3) | 2 |
| 39. | ALTERNATIVE CONTAINMENT MONITORING SYSTEM (E8) | 2 |
| 40. | ALTERNATIVE CAVITY COOLING (E9) | 2 |
| 41. | 12 HOUR BATTERIES (B2) | 2 |
| 42. | TORNADO PROTECTION FOR COMBUSTION TURBINE (C2) | 2 |
| 43. | DIESEL SI PUMPS (2) (B6) | 2 |
| 44. | ALTERNATIVE STARTUP FEEDWATER SYSTEM (B7) | 1 |
| 45. | VACUUM BUILDING (E14) | 3 |
| 46. | RIBBED CONTAINMENT (E15) | 3 |
| 47. | EXTENDED RWST SOURCE (B8) | 2 |
| 48. | N-16 MONITOR (A3) | 1 |
| 49. | INCREASE SECONDARY SIDE PRESSURE (A4) | 3 |
| 50. | PASSIVE SECONDARY SIDE COOLERS (A5) | 3 |
| 51. | VENTING MSSV TO CONTAINMENT (E10) | 3 |
| 52. | SECONDARY SIDE GUARD PIPES (A6) | 2 |
| 53. | PASSIVE AUTOCATALYTIC RECOMBINERS (PARS) (E6) | 2 |
| 54. | HYDROGEN PURGE LINE (E11) | 1 |
| 55. | FUEL CELLS (C3) | 2 |
| 56. | HOOKUP FOR PORTABLE GENERATOR (C4) | 2 |
| 57. | WATER COOLED RUBBLE BED (E12) | 2 |

**TABLE 4.5**

**DESIGN ALTERNATIVES CONSIDERED**

| | DESIGN ALTERNATIVE | CATEGORY* |
|---|---|---|
| 58. | REFRACTORY LINED CRUCIBLE (E13) | 2 |
| 59. | AUTOMATIC OVERPRESSURE PROTECTION (A7) | 3 |
| 60. | DIGITAL LBLOCA PROTECTION (A8) | 3 |
| 61. | SEISMIC CAPABILITY (D4) | 3 |
| 62. | FIRE AND FLOOD CAPABILITY (D5) | 3 |

* Category: 1 Modification is applicable to System 80+ and is already incorporated in the design. No further discussions are included for the first 22 items because they are described in CESSAR-DC. A discussion is included for four DAs (DA # 26, 44, 48, 54) that are included in the System 80+ design because they address important safety issues but their cost benefits were not quantified.

2 Modification for 27 DAs were quantified in this report and not included in the System 80+ design.

3 Nine modifications were discussed in this report but were not quantified because of high costs or small benefits.

** Designation refers to the Section where this DA is described.

## TABLE 4-6

## DESIGN ALTERNATIVES EVALUATED

| NUMBER | DESIGN ALTERNATIVE |
|---|---|
| A: INCREASE PRIMARY AND SECONDARY BOUNDARY INTEGRITY | |
| A1 | RCP SEAL COOLING |
| A2 | 100% SG INSPECTION |
| A3 | N-16 MONITOR |
| A4 | INCREASE SECONDARY SIDE PRESSURE |
| A5 | PASSIVE SECONDARY SIDE COOLERS |
| A6 | SECONDARY SIDE GUARD PIPES |
| A7 | AUTOMATIC OVERPRESSURE PROTECTION |
| A8 | DIGITAL LBLOCA PROTECTION |
| B: INCREASE DECAY HEAT REMOVAL RELIABILITY | |
| B1 | ALTERNATIVE DC BATTERIES AND EFWS |
| B2 | 12 HOUR BATTERIES |
| B3 | ALTERNATIVE PRESSURIZER AUXILIARY SPRAY |
| B4 | ALTERNATIVE HIGH PRESSURE SAFETY INJECTION |
| B5 | ALTERNATIVE RCS DEPRESSURIZATION |
| B6 | DIESEL SI PUMPS (2) |
| B7 | ALTERNATIVE STARTUP FEEDWATER SYSTEM |
| B8 | EXTENDED RWST SOURCE |
| C: IMPROVE ELECTRICAL POWER RELIABILITY | |
| C1 | THIRD DIESEL GENERATOR |
| C2 | TORNADO PROTECTION FOR COMBUSTION TURBINE |
| C3 | FUEL CELLS |
| C4 | HOOKUP FOR PORTABLE GENERATOR |
| D: ATWS AND EXTERNAL EVENTS | |
| D1 | ALTERNATIVE ATWS PRESSURE RELIEF VALVES |
| D2 | ATWS INJECTION SYSTEM |
| D3 | DIVERSE PPS |
| D4 | SEISMIC CAPABILITY |
| D5 | FIRE AND FLOOD CAPABILITY |

## TABLE 4-6

## DESIGN ALTERNATIVES EVALUATED

| NUMBER | DESIGN ALTERNATIVE |
|---|---|
| E: REDUCE RADIOACTIVE RELEASES | |
| E1 | ALTERNATIVE CONTAINMENT SPRAY |
| E2 | FILTERED VENT (CONTAINMENT) |
| E3 | ALTERNATIVE CONCRETE COMPOSITION |
| E4 | REACTOR VESSEL EXTERIOR COOLING |
| E5 | ALTERNATIVE H2 IGNITERS |
| E6 | PASSIVE AUTOCATALYTIC RECOMBINERS (PARS) |
| E7 | MSSV AND ADV SCRUBBING |
| E8 | ALTERNATIVE CONTAINMENT MONITORING SYSTEM |
| E9 | CAVITY COOLING |
| E10 | VENTING MSSV TO CONTAINMENT |
| E11 | HYDROGEN PURGE LINE |
| E12 | WATER COOLED RUBBLE BED |
| E13 | REFRACTORY LINED CRUCIBLE |
| E14 | VACUUM BUILDING |
| E15 | RIBBED CONTAINMENT |

TABLE 4-7

RISK REDUCTION EVALUATION OF
A2, 100% SG INSPECTION

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.21 | 2.86E-07 | 1.19E+02 | 3.40E-05 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 0.00 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.00 | 0 | 4.73E+04 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.00 | 0 | 1.38E+05 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 0.00 | 0 | 1.02E+06 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 0.38 | 1.20E-09 | 1.32E+06 | 1.59E-03 |
| RC3.4E | 6.52E-09 | 0.00 | 0 | 1.20E+06 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 0.38 | 1.22E-09 | 1.27E+06 | 1.55E-03 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 1.00 | 5.99E-09 | 5.24E+06 | 3.14E-02 |
| RC4.8E | 1.06E-09 | 0.00 | 0 | 1.86E+05 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 1.00 | 6.55E-09 | 5.07E+06 | 3.32E-02 |
| RC4.36L | 3.08E-08 | 1.00 | 3.08E-08 | 5.90E+06 | 1.82E-01 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 | | 3.31E-07 | | 2.49E-01 |
| | CD/YR | | CDF REDUCTION | | P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $220.69

TABLE 4-8

RISK REDUCTION EVALUATION OF
A6, SECONDARY SIDE GUARD PIPES

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.001 | 1.36E-09 | 1.19E+02 | 1.62E-07 |
| RC1.1M | 3.83E-07 | 0.000 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.002 | 6.88E-12 | 1.37E+05 | 9.43E-07 |
| RC2.2E | 3.31E-09 | 0.002 | 6.62E-12 | 1.37E+05 | 9.07E-07 |
| RC2.4E | 3.76E-08 | 0.000 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.000 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.000 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.000 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.000 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.000 | 0 | 4.73E+04 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 0.000 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.000 | 0 | 1.38E+05 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 0.001 | 4.64E-12 | 1.02E+06 | 4.73E-06 |
| RC3.2E | 3.16E-09 | 0.001 | 3.16E-12 | 1.32E+06 | 4.17E-06 |
| RC3.4E | 6.52E-09 | 0.001 | 6.52E-12 | 1.20E+06 | 7.82E-06 |
| RC3.6E | 3.21E-09 | 0.001 | 3.21E-12 | 1.27E+06 | 4.08E-06 |
| RC3.2M | 1.82E-09 | 0.000 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.000 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.000 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 0.000 | 0 | 1.86E+05 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 0.000 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 0.000 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.500 | 2.55E-10 | 2.87E+06 | 7.32E-04 |
| SUM = | 1.96E-06 CD/YR | | 1.65E-09 CDF REDUCTION | | 7.55E-04 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $ 1.10

TABLE 4-9

RISK REDUCTION EVALUATION* OF
B1, ALTERNATIVE DC BATTERIES AND EFWS

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.00 | 0 | 1.19E+02 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 0.00 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 1.00 | 3.95E-09 | 4.73E+04 | 1.87E-04 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 1.00 | 1.22E-08 | 1.38E+05 | 1.68E-03 |
| RC3.1E | 4.64E-09 | 0.00 | 0 | 1.02E+06 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 0.00 | 0 | 1.[illegible]2E+06 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 0.00 | 0 | 1.20E+06 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 0.00 | 0 | 1.27E+06 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.00 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 0.00 | 0 | 1.86E+05 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 0.00 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 1.62E-08 CDF REDUCTION | | 1.87E-03 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $ 10.76

* EVALUATION USED FOR C3, FUEL CELL, AND C4, HOOKUP FOR PORTABLE GENERATOR

TABLE 4-10

RISK REDUCTION EVALUATION OF
B2, 12 HOUR BATTERIES

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.00 | 0 | 1.19E+02 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 0.00 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.62 | 2.45E-09 | 4.73E+04 | 1.16E-04 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.62 | 7.56E-09 | 1.38E+05 | 1.04E-03 |
| RC3.1E | 4.64E-09 | 0.00 | 0 | 1.02E+06 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 0.00 | 0 | 1.32E+06 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 0.00 | 0 | 1.20E+06 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 0.00 | 0 | 1.27E+06 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.00 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 0.00 | 0 | 1.86E+05 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 0.00 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 | | 1.00E-08 | | 1.16E-03 |
| | CD/YR | | CDF REDUCTION | | P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $ 6.66

TABLE 4-11

RISK REDUCTION EVALUATION OF
B3, ALTERNATIVE PRESSURIZER AUXILIARY SPRAY

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.21 | 2.86E-07 | 1.19E+02 | 3.40E-05 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 0.00 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.00 | 0 | 4.73E+04 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.00 | 0 | 1.38E+05 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 0.00 | 0 | 1.02E+06 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 0.36 | 1.14E-09 | 1.32E+06 | 1.50E-03 |
| RC3.4E | 6.52E-09 | 0.00 | 0 | 1.20E+06 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 0.36 | 1.16E-09 | 1.27E+06 | 1.47E-03 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.95 | 5.69E-09 | 5.24E+06 | 2.98E-02 |
| RC4.8E | 1.06E-09 | 0.00 | 0 | 1.86E+05 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 0.97 | 6.35E-09 | 5.07E+06 | 3.22E-02 |
| RC4.36L | 3.08E-08 | 0.78 | 2.40E-08 | 5.90E+06 | 1.42E-01 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 3.24E-07 CDF REDUCTION | | 2.07E-01 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $215.76

TABLE 4-12

RISK REDUCTION EVALUATION OF
B4, ALTERNATIVE HIGH PRESSURE SAFETY INJECTION

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.55 | 7.48E-07 | 1.19E+02 | 8.90E-05 |
| RC1.1M | 3.83E-07 | 1.00 | 3.83E-07 | 1.09E+02 | 4.17E-05 |
| RC2.1E | 3.44E-09 | 0.16 | 5.50E-10 | 1.37E+05 | 7.54E-05 |
| RC2.2E | 3.31E-09 | 0.38 | 1.26E-09 | 1.37E+05 | 1.72E-04 |
| RC2.4E | 3.76E-08 | 0.48 | 1.80E-08 | 2.38E+04 | 4.30E-04 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.48 | 1.60E-08 | 2.35E+04 | 3.76E-04 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 1.00 | 4.05E-09 | 1.31E+05 | 5.31E-04 |
| RC2.5M | 3.95E-09 | 0.00 | 0 | 4.73E+04 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 1.00 | 9.08E-09 | 3.02E+04 | 2.74E-04 |
| RC2.7M | 1.22E-08 | 0.87 | 1.06E-08 | 1.38E+05 | 1.46E-03 |
| RC3.1E | 4.64E-09 | 0.32 | 1.48E-09 | 1.02E+06 | 1.51E-03 |
| RC3.2E | 3.16E-09 | 0.38 | 1.20E-09 | 1.32E+06 | 1.59E-03 |
| RC3.4E | 6.52E-09 | 0.42 | 2.74E-09 | 1.20E+06 | 3.29E-03 |
| RC3.6E | 3.21E-09 | 0.38 | 1.22E-09 | 1.27E+06 | 1.55E-03 |
| RC3.2M | 1.82E-09 | 1.00 | 1.82E-09 | 1.81E+06 | 3.29E-03 |
| RC3.6M | 1.83E-09 | 1.00 | 1.83E-09 | 1.97E+06 | 3.61E-03 |
| RC4.22E | 5.99E-09 | 1.00 | 5.99E-09 | 5.24E+06 | 3.14E-02 |
| RC4.8E | 1.06E-09 | 0.42 | 4.45E-10 | 1.86E+05 | 8.28E-05 |
| RC4.30E | 6.55E-09 | 1.00 | 6.55E-09 | 5.07E+06 | 3.32E-02 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 1.21E-06 CDF REDUCTION | | 8.30E-02 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $808.43

TABLE 4-13

RISK REDUCTION EVALUATION OF
B5, ALTERNATIVE RCS DEPRESSURIZATION

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.36 | 4.90E-07 | 1.19E+02 | 5.83E-05 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.69 | 2.37E-09 | 1.37E+05 | 3.25E-04 |
| RC2.2E | 3.31E-09 | 0.75 | 2.48E-09 | 1.37E+05 | 3.40E-04 |
| RC2.4E | 3.76E-08 | 0.42 | 1.58E-08 | 2.38E+04 | 3.76E-04 |
| RC2.5E | 2.84E-08 | 1.00 | 2.84E-08 | 2.35E+04 | 6.67E-04 |
| RC2.6E | 3.33E-08 | 0.52 | 1.73E-08 | 2.35E+04 | 4.07E-04 |
| RC2.7E | 1.62E-08 | 1.00 | 1.62E-08 | 2.35E+04 | 3.81E-04 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.00 | 0 | 4.73E+04 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.00 | 0 | 1.38E+05 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 0.56 | 2.60E-09 | 1.02E+06 | 2.65E-03 |
| RC3.2E | 3.16E-09 | 0.62 | 1.96E-09 | 1.32E+06 | 2.59E-03 |
| RC3.4E | 6.52E-09 | 0.48 | 3.13E-09 | 1.20E+06 | 3.76E-03 |
| RC3.6E | 3.21E-09 | 0.62 | 1.99E-09 | 1.27E+06 | 2.53E-03 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+05 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.00 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 0.48 | 5.09E-10 | 1.86E+05 | 9.46E-05 |
| RC4.30E | 6.55E-09 | 0.00 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 | | 5.82E-07 | | 1.42E-02 |
| | CD/YR | | CDF REDUCTION | | P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $ 387.85

TABLE 4-14

RISK REDUCTION EVALUATION OF
B6, DIESEL SI PUMPS (2)

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.55 | 7.48E-07 | 1.19E+02 | 8.90E-05 |
| RC1.1M | 3.83E-07 | 1.00 | 3.83E-07 | 1.09E+02 | 4.17E-05 |
| RC2.1E | 3.44E-09 | 0.16 | 5.50E-10 | 1.37E+05 | 7.54E-05 |
| RC2.2E | 3.31E-09 | 0.38 | 1.26E-09 | 1.37E+05 | 1.72E-04 |
| RC2.4E | 3.76E-08 | 0.48 | 1.80E-08 | 2.38E+04 | 4.30E-04 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.48 | 1.60E-08 | 2.35E+04 | 3.76E-04 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 1.00 | 4.05E-09 | 1.31E+05 | 5.31E-04 |
| RC2.5M | 3.95E-09 | 1.00 | 3.95E-09 | 4.73E+04 | 1.87E-04 |
| RC2.6M | 9.08E-09 | 1.00 | 9.08E-09 | 3.02E+04 | 2.74E-04 |
| RC2.7M | 1.22E-08 | 1.00 | 1.22E-08 | 1.38E+05 | 1.68E-03 |
| RC3.1E | 4.64E-09 | 0.32 | 1.48E-09 | 1.02E+06 | 1.51E-03 |
| RC3.2E | 3.16E-09 | 0.38 | 1.20E-09 | 1.32E+06 | 1.59E-03 |
| RC3.4E | 6.52E-09 | 0.42 | 2.74E-09 | 1.20E+06 | 3.29E-03 |
| RC3.6E | 3.21E-09 | 0.38 | 1.22E-09 | 1.27E+06 | 1.55E-03 |
| RC3.2M | 1.82E-09 | 1.00 | 1.82E-09 | 1.81E+06 | 3.29E-03 |
| RC3.6M | 1.83E-09 | 1.00 | 1.83E-09 | 1.97E+06 | 3.61E-03 |
| RC4.22E | 5.99E-09 | 1.00 | 5.99E-09 | 5.24E+06 | 3.14E-02 |
| RC4.8E | 1.06E-09 | 0.42 | 4.45E-10 | 1.86E+05 | 8.28E-05 |
| RC4.30E | 6.55E-09 | 1.00 | 6.55E-09 | 5.07E+06 | 3.32E-02 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 1.22E-06 CDF REDUCTION | | 8.34E-02 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $812.12

TABLE 4-15

THIS TABLE HAS BEEN INTENTIONALLY LEFT BLANK

TABLE 4-16

RISK REDUCTION EVALUATION OF
B8, EXTENDED RWST SOURCE

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.00 | 0 | 1.19E+02 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 0.00 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.00 | 0 | 4.73E+04 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.00 | 0 | 1.38E+05 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 0.00 | 0 | 1.02E+06 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 0.00 | 0 | 1.32E+06 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 0.00 | 0 | 1.20E+06 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 0.00 | 0 | 1.27E+06 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.00 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 0.00 | 0 | 1.86E+05 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 0.00 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 1.00 | 3.08E-08 | 5.90E+06 | 1.82E-01 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 3.08E-08 CDF REDUCTION | | 1.82E-01 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $ 20.51

TABLE 4-17

RISK REDUCTION EVALUATION OF
C1, THIRD DIESEL GENERATOR

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.00 | 0 | 1.19E+02 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 0.00 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.24 | 9.48E-10 | 4.73E+04 | 4.48E-05 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.24 | 2.93E-09 | 1.38E+05 | 4.04E-04 |
| RC3.1E | 4.64E-09 | 0.00 | 0 | 1.02E+06 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 0.00 | 0 | 1.32E+06 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 0.00 | 0 | 1.20E+06 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 0.00 | 0 | 1.27E+06 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.00 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 0.00 | 0 | 1.86E+05 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 0.00 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 3.88E-09 CDF REDUCTION | | 4.49E-04 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $ 2.58

TABLE 4-18

RISK REDUCTION EVALUATION OF
C2, TORNADO-PROTECTION FOR COMBUSTION TURBINE

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.00 | 0 | 1.19E+02 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 0.00 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.43 | 1.70E-09 | 4.73E+04 | 8.03E-05 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.90 | 1.10E-08 | 1.38E+05 | 1.52E-03 |
| RC3.1E | 4.64E-09 | 0.00 | 0 | 1.02E+06 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 0.00 | 0 | 1.32E+06 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 0.00 | 0 | 1.20E+06 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 0.00 | 0 | 1.27E+06 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.00 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 0.00 | 0 | 1.86E+05 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 0.00 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 1.27E-08 CDF REDUCTION | | 1.60E-03 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $ 8.44

TABLE 4-19

RISK REDUCTION EVALUATION* OF
D1, ALTERNATIVE ATWS PRESSURE RELIEF VALVES

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.03 | 4.08E-08 | 1.19E+02 | 4.86E-06 |
| RC1.1M | 3.83E-07 | 0.03 | 1.15E-08 | 1.09E+02 | 1.25E-06 |
| RC2.1E | 3.44E-09 | 0.03 | 1.03E-10 | 1.37E+05 | 1.41E-05 |
| RC2.2E | 3.31E-09 | 0.03 | 9.93E-11 | 1.37E+05 | 1.36E-05 |
| RC2.4E | 3.76E-08 | 0.03 | 1.13E-09 | 2.38E+04 | 2.68E-05 |
| RC2.5E | 2.84E-08 | 0.03 | 8.52E-10 | 2.35E+04 | 2.00E-05 |
| RC2.6E | 3.33E-08 | 0.03 | 9.99E-10 | 2.35E+04 | 2.35E-05 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.03 | 1.22E-10 | 1.31E+05 | 1.59E-05 |
| RC2.5M | 3.95E-09 | 0.00 | 0 | 4.73E+04 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 0.03 | 2.72E-10 | 3.02E+04 | 8.23E-06 |
| RC2.7M | 1.22E-08 | 0.00 | 0 | 1.38E+05 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 0.03 | 1.39E-10 | 1.02E+06 | 1.42E-04 |
| RC3.2E | 3.16E-09 | 0.03 | 9.48E-11 | 1.32E+06 | 1.25E-04 |
| RC3.4E | 6.52E-09 | 0.03 | 1.96E-10 | 1.20E+06 | 2.35E-04 |
| RC3.6E | 3.21E-09 | 0.03 | 9.63E-11 | 1.27E+06 | 1.22E-04 |
| RC3.2M | 1.82E-09 | 0.03 | 5.46E-11 | 1.81E+06 | 9.88E-05 |
| RC3.6M | 1.83E-09 | 0.03 | 5.49E-11 | 1.97E+06 | 1.08E-04 |
| RC4.22E | 5.99E-09 | 0.00 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 0.03 | 3.18E-11 | 1.86E+05 | 5.91E-06 |
| RC4.30E | 6.55E-09 | 0.00 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 0.00 | 0 | 2.87E+06 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 5.65E-08 CDF REDUCTION | | 9.65E-04 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $37.65

* EVALUATION USED FOR D2, ATWS INJECTION SYSTEM, AND D3, DIVERSE PPS

TABLE 4-20

RISK REDUCTION EVALUATION OF
E1, ALTERNATIVE CONTAINMENT SPRAY

| RELEASE CLASS | CDF FREQUENCY | DOSE P-REM | RISK P-REM/YR | FRACTION REDUCTION | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 1.19E+02 | 1.62E-04 | 0.00 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 1.09E+02 | 4.17E-05 | 0.00 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 1.37E+05 | 4.71E-04 | 0.00 | 0.00F+00 |
| RC2.2E | 3.31E-09 | 1.37E+05 | 4.53E-04 | 0.00 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 2.38E+04 | 8.95E-04 | 0.00 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 2.35E+04 | 6.67E-04 | 0.05 | 3.34E-05 |
| RC2.6E | 3.33E-08 | 2.35E+04 | 7.83E-04 | 0.00 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 2.35E+04 | 3.81E-04 | 0.00 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 1.31E+05 | 5.31E-04 | 1.00 | 5.31E-04 |
| RC2.5M | 3.95E-09 | 4.73E+04 | 1.87E-04 | 0.36 | 6.73E-05 |
| RC2.6M | 9.08E-09 | 3.02E+04 | 2.74E-04 | 0.00 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 1.38E+05 | 1.68E-03 | 0.78 | 1.31E-03 |
| RC3.1E | 4.64E-09 | 1.02E+06 | 4.73E-03 | 0.00 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 1.32E+06 | 4.17E-03 | 0.00 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 1.20E+06 | 7.82E-03 | 0.00 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 1.27E+06 | 4.08E-03 | 0.00 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 1.81E+06 | 3.29E-03 | 0.78 | 2.57E-03 |
| RC3.6M | 1.83E-09 | 1.97E+06 | 3.61E-03 | 0.78 | 2.81E-03 |
| RC4.22E | 5.99E-09 | 5.24E+06 | 3.14E-02 | 0.00 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 1.86E+05 | 1.97E-04 | 0.00 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 5.07E+06 | 3.32E-02 | 0.00 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 5.90E+06 | 1.82E-01 | 0.00 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 2.87E+06 | 1.46E-03 | 0.00 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 2.82E-01 P-REM/RY | | 7.33E-03 P-REM/YR |

TABLE 4-21

RISK REDUCTION EVALUATION OF
E2, FILTERED VENT (CONTAINMENT)

| RELEASE CLASS | CDF FREQUENCY | DOSE P-REM | RISK P-REM/YR | FRACTION REDUCTION | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 1.19E+02 | 1.62E-04 | 0.00 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 1.09E+02 | 4.17E-05 | 0.00 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 1.37E+05 | 4.71E-04 | 0.00 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 1.37E+05 | 4.53E-04 | 0.00 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 2.38E+04 | 8.95E-04 | 0.00 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 2.35E+04 | 6.67E-04 | 0.00 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 2.35E+04 | 7.83E-04 | 0.00 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 2.35E+04 | 3.81E-04 | 0.00 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 1.31E+05 | 5.31E-04 | 1.00 | 5.31E-04 |
| RC2.5M | 3.95E-09 | 4.73E+04 | 1.87E-04 | 0.00 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 3.02E+04 | 2.74E-04 | 0.00 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 1.38E+05 | 1.68E-03 | 0.00 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 1.02E+06 | 4.73E-03 | 0.00 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 1.32E+06 | 4.17E-03 | 0.00 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 1.20E+06 | 7.82E-03 | 0.00 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 1.27E+06 | 4.08E-03 | 0.00 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 1.81E+06 | 3.29E-03 | 0.00 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 1.97E+06 | 3.61E-03 | 0.00 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 5.24E+06 | 3.14E-02 | 0.00 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 1.86E+05 | 1.97E-04 | 0.00 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 5.07E+06 | 3.32E-02 | 0.00 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 5.90E+06 | 1.82E-01 | 0.00 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 2.87E+06 | 1.46E-03 | 0.00 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 2.82E-01 P-REM/RY | | 5.31E-04 P-REM/YR |

TABLE 4-22

RISK REDUCTION EVALUATION* OF
E3, ALTERNATIVE CONCRETE COMPOSITION

| RELEASE CLASS | CDF FREQUENCY | DOSE P-REM | RISK P-REM/YR | FRACTION REDUCTION | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 1.19E+02 | 1.62E-04 | 0.00 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 1.09E+02 | 4.17E-05 | 0.00 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 1.37E+05 | 4.71E-04 | 0.00 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 1.37E+05 | 4.53E-04 | 0.00 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 2.38E+04 | 8.95E-04 | 1.00 | 8.95E-04 |
| RC2.5E | 2.84E-08 | 2.35E+04 | 6.67E-04 | 1.00 | 6.67E-04 |
| RC2.6E | 3.33E-08 | 2.35E+04 | 7.83E-04 | 1.00 | 7.83E-04 |
| RC2.7E | 1.62E-08 | 2.35E+04 | 3.81E-04 | 1.00 | 3.81E-04 |
| RC2.2M | 4.05E-09 | 1.31E+05 | 5.31E-04 | 0.00 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 4.73E+04 | 1.87E-04 | 1.00 | 1.87E-04 |
| RC2.6M | 9.08E-09 | 3.02E+04 | 2.74E-04 | 1.00 | 2.74E-04 |
| RC2.7M | 1.22E-08 | 1.38E+05 | 1.68E-03 | 1.00 | 1.68E-03 |
| RC3.1E | 4.64E-09 | 1.02E+06 | 4.73E-03 | 0.00 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 1.32E+06 | 4.17E-03 | 0.00 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 1.20E+06 | 7.82E-03 | 0.00 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 1.27E+06 | 4.08E-03 | 0.00 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 1.81E+06 | 3.29E-03 | 0.00 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 1.97E+06 | 3.61E-03 | 0.00 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 5.24E+06 | 3.14E-02 | 0.00 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 1.86E+05 | 1.97E-04 | 0.00 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 5.07E+06 | 3.32E-02 | 0.00 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 5.90E+06 | 1.82E-01 | 0.00 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 2.87E+06 | 1.46E-03 | 0.00 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 2.82E-01 P-REM/RY | | 4.87E-03 P-REM/YR |

* EVALUATION USED FOR E12, WATER COOLED RUBBLE BED, AND E13, REFRACTORY LINED CRUCIBLE

TABLE 4-23

RISK REDUCTION EVALUATION* OF
E4, REACTOR VESSEL EXTERIOR COOLING

| RELEASE CLASS | CDF FREQUENCY | DOSE P-REM | RISK P-REM/YR | FRACTION REDUCTION | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 1.19E+02 | 1.62E-04 | 0.00 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 1.09E+02 | 4.17E-05 | 0.00 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 1.37E+05 | 4.71E-04 | 0.00 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 1.37E+05 | 4.53E-04 | 0.00 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 2.38E+04 | 8.95E-04 | 1.00 | 8.95E-04 |
| RC2.5E | 2.84E-08 | 2.35E+04 | 6.67E-04 | 1.00 | 6.67E-04 |
| RC2.6E | 3.33E-08 | 2.35E+04 | 7.83E-04 | 1.00 | 7.83E-04 |
| RC2.7E | 1.62E-08 | 2.35E+04 | 3.81E-04 | 1.00 | 3.81E-04 |
| RC2.2M | 4.05E-09 | 1.31E+05 | 5.31E-04 | 0.00 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 4.73E+04 | 1.87E-04 | 0.00 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 3.02E+04 | 2.74E-04 | 1.00 | 2.74E-04 |
| RC2.7M | 1.22E-08 | 1.38E+05 | 1.68E-03 | 0.00 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 1.02E+06 | 4.73E-03 | 1.00 | 4.73E-03 |
| RC3.2E | 3.16E-09 | 1.32E+06 | 4.17E-03 | 1.00 | 4.17E-03 |
| RC3.4E | 6.52E-09 | 1.20E+06 | 7.82E-03 | 1.00 | 7.82E-03 |
| RC3.6E | 3.21E-09 | 1.27E+06 | 4.08E-03 | 1.00 | 4.08E-03 |
| RC3.2M | 1.82E-09 | 1.81E+06 | 3.29E-03 | 1.00 | 3.29E-03 |
| RC3.6M | 1.83E-09 | 1.97E+06 | 3.61E-03 | 1.00 | 3.61E-03 |
| RC4.22E | 5.99E-09 | 5.24E+06 | 3.14E-02 | 0.00 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 1.86E+05 | 1.97E-04 | 0.00 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 5.07E+06 | 3.32E-02 | 0.00 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 5.90E+06 | 1.82E-01 | 0.00 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 2.87E+06 | 1.46E-03 | 0.00 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 2.82E-01 P-REM/RY | | 3.07E-02 P-REM/YR |

* EVALUATION USED FOR E9, CAVITY COOLING

TABLE 4-24

RISK REDUCTION EVALUATION* OF
E5, ALTERNATIVE H2 IGNITORS

| RELEASE CLASS | CDF FREQUENCY | DOSE P-REM | RISK P-REM/YR | FRACTION REDUCTION | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 1.19E+02 | 1.62E-04 | 0.00 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 1.09E+02 | 4.17E-05 | 0.00 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 1.37E+05 | 4.71E-04 | 1.00 | 4.71E-04 |
| RC2.2E | 3.31E-09 | 1.37E+05 | 4.53E-04 | 1.00 | 4.53E-04 |
| RC2.4E | 3.76E-08 | 2.38E+04 | 8.95E-04 | 0.00 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 2.35E+04 | 6.67E-04 | 0.00 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 2.35E+04 | 7.83E-04 | 0.00 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 2.35E+04 | 3.81E-04 | 0.00 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 1.31E+05 | 5.31E-04 | 0.00 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 4.73E+04 | 1.87E-04 | 0.00 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 3.02E+04 | 2.74E-04 | 0.00 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 1.38E+05 | 1.68E-03 | 0.00 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 1.02E+06 | 4.73E-03 | 0.00 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 1.32E+06 | 4.17E-03 | 0.00 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 1.20E+06 | 7.82E-03 | 0.00 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 1.27E+06 | 4.08E-03 | 0.00 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 1.81E+06 | 3.29E-03 | 0.00 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 1.97E+06 | 3.61E-03 | 0.00 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 5.24E+06 | 3.14E-02 | 0.00 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 1.86E+05 | 1.97E-04 | 0.00 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 5.07E+06 | 3.32E-02 | 0.00 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 5.90E+06 | 1.82E-01 | 0.00 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 2.87E+06 | 1.46E-03 | 0.00 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 2.82E-01 P-REM/YR | | 9.25E-04 P-REM/YR |

* EVALUATION USED FOR E6, PARs

TABLE 4-25

RISK REDUCTION EVALUATION OF
E7, MSSV AND ADV SCRUBBING

| RELEASE CLASS | CDF FREQUENCY | DOSE P-REM | RISK P-REM/YR | FRACTION REDUCTION | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 1.19E+02 | 1.62E-04 | 0.00 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 1.09E+02 | 4.17E-05 | 0.00 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 1.37E+05 | 4.71E-04 | 0.00 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 1.37E+05 | 4.53E-04 | 0.00 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 2.38E+04 | 8.95E-04 | 0.00 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 2.35E+04 | 6.67E-04 | 0.00 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 2.35E+04 | 7.83E-04 | 0.00 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 2.35E+04 | 3.81E-04 | 0.00 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 1.31E+05 | 5.31E-04 | 0.00 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 4.73E+04 | 1.87E-04 | 0.00 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 3.02E+04 | 2.74E-04 | 0.00 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 1.38E+05 | 1.68E-03 | 0.00 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 1.02E+06 | 4.73E-03 | 0.00 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 1.32E+06 | 4.17E-03 | 0.00 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 1.20E+06 | 7.82E-03 | 0.00 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 1.27E+06 | 4.08E-03 | 0.00 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 1.81E+06 | 3.29E-03 | 0.00 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 1.97E+06 | 3.61E-03 | 0.00 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 5.24E+06 | 3.14E-02 | 1.00 | 3.14E-02 |
| RC4.8E | 1.06E-09 | 1.86E+05 | 1.97E-04 | 0.00 | 0.00E+00 |
| RC4.30E | 6.55E-09 | 5.07E+06 | 3.32E-02 | 1.00 | 3.32E-02 |
| RC4.36L | 3.08E-08 | 5.90E+06 | 1.82E-01 | 1.00 | 1.82E-01 |
| RC5.1E | 5.10E-10 | 2.87E+06 | 1.46E-03 | 0.00 | 0.00E+00 |
| SUM = | 1.96E-06 CD/YR | | 2.82E-01 P-REM/RY | | 2.46E-01 P-REM/YR |

TABLE 4-26

RISK REDUCTION EVALUATION OF
E8, ALTERNATIVE CONTAINMENT MONITORING SYSTEM

| RELEASE CLASS | CDF FREQUENCY | FRACTION REDUCTION | CDF REDUCTION | DOSE P-REM | RISK REDUCTION P-REM/YR |
|---|---|---|---|---|---|
| RC1.1E | 1.36E-06 | 0.00 | 0 | 1.19E+02 | 0.00E+00 |
| RC1.1M | 3.83E-07 | 0.00 | 0 | 1.09E+02 | 0.00E+00 |
| RC2.1E | 3.44E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.2E | 3.31E-09 | 0.00 | 0 | 1.37E+05 | 0.00E+00 |
| RC2.4E | 3.76E-08 | 0.00 | 0 | 2.38E+04 | 0.00E+00 |
| RC2.5E | 2.84E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.6E | 3.33E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.7E | 1.62E-08 | 0.00 | 0 | 2.35E+04 | 0.00E+00 |
| RC2.2M | 4.05E-09 | 0.00 | 0 | 1.31E+05 | 0.00E+00 |
| RC2.5M | 3.95E-09 | 0.00 | 0 | 4.73E+04 | 0.00E+00 |
| RC2.6M | 9.08E-09 | 0.00 | 0 | 3.02E+04 | 0.00E+00 |
| RC2.7M | 1.22E-08 | 0.00 | 0 | 1.38E+05 | 0.00E+00 |
| RC3.1E | 4.64E-09 | 0.00 | 0 | 1.02E+06 | 0.00E+00 |
| RC3.2E | 3.16E-09 | 0.00 | 0 | 1.32E+06 | 0.00E+00 |
| RC3.4E | 6.52E-09 | 0.00 | 0 | 1.20E+06 | 0.00E+00 |
| RC3.6E | 3.21E-09 | 0.00 | 0 | 1.27E+06 | 0.00E+00 |
| RC3.2M | 1.82E-09 | 0.00 | 0 | 1.81E+06 | 0.00E+00 |
| RC3.6M | 1.83E-09 | 0.00 | 0 | 1.97E+06 | 0.00E+00 |
| RC4.22E | 5.99E-09 | 0.00 | 0 | 5.24E+06 | 0.00E+00 |
| RC4.8E | 1.06E-09 | 1.00 | 1.06E-09 | 1.86E+05 | 1.97E-04 |
| RC4.30E | 6.55E-09 | 0.00 | 0 | 5.07E+06 | 0.00E+00 |
| RC4.36L | 3.08E-08 | 0.00 | 0 | 5.90E+06 | 0.00E+00 |
| RC5.1E | 5.10E-10 | 1.00 | 5.10E-10 | 2.87E+06 | 1.46E-03 |
| SUM = | 1.96E-06 CD/YR | | 1.57E-09 CDF REDUCTION | | 1.66E-03 P-REM/RY |

AOC/EVENT= 6.66E+08

AOC/YR= $ 1.05

## 5.0 SUMMARY AND CONCLUSIONS

The System 80+ design is an evolutionary Advanced Light Water Reactor (ALWR) design with improved design features to reduce the risk of core damage and mitigate the consequences if core damage should occur. The design process was integrated with the PRA to ensure that the risk is very low and distributed over all of the safety related systems (i.e., no single system carries a disproportional responsibility for plant safety). The design ensured that no single accident sequence dominated the plant risk and the lessons learned from previous PRAs were addressed.

Sixty-two Design Alternatives were considered and the expected risk reduction from twenty-seven of those alternatives were quantified. These were selected based on the DAs evaluated for the Limerick plant[6] , Comanche Peak[7], NUREG/CR-4920[15], GI-163[16], and the results from the System 80+ PRA performed by ABB-CE. The DAs were selected to address the sequences that either have the largest risk to the public or sequences that have high CDF. The analysis used a bounding technique. It was assumed that each DA worked perfectly and completely eliminated the accident sequences that the DA was to address. This approach maximizes the benefits associated with each DA. The benefits were the reduction in risk in terms of whole body person-rems per year received by the total population around the ALWR site. Consistent with the standard used by NRC to evaluate offsite impacts, health and economic effect costs were evaluated based on a value of $1000 per offsite person-rem averted.

Table 5-1 summarizes the results of the DA quantification including AOC. The second column is a capital cost estimate for the given DAs. The third column is the annual risk reduction to the Combined License (CL) applicant for each DA for dose risk to the general population using in person-rem/year reduction. The next column gives the AOC savings and the cost per person rem is given in the last column.

The System 80+ plant was designed to meet the stringent design goals in the EPRI ALWR Utility Requirements Document[9]. The System 80+ design has a core damage frequency approximately two orders of magnitude lower than existing plants. The analysis presented in this report conservatively estimated the benefits of the DAs by assuming that they would work perfectly to eliminate the type of accident they are designed to address and would require no maintenance or testing. Because of the small initial risk associated with the System 80+ design, none of the DAs was found to be cost beneficial.

Therefore, after evaluating a reasonable and comprehensive set of candidate SAMDAs relevant to the System 80+ design in terms of costs, averted onsite costs and potential benefits using a screening criterion of $1,000 per person-rem averted, it was concluded that none of the SAMDAs is cost effective. Given the

low residual risk profile of the System 80+ design, SAMDAs cannot reasonably be incorporated in a cost effective manner.

On the basis of the foregoing analysis, further incorporation of SAMDAs into the System 80+ design is not warranted. No further screening of SAMDAs is needed and no further SAMDAs need be incorporated into the System 80+ design in satisfaction of NEPA.

TABLE 5-1

**NEPA COST BENEFIT ANALYSIS FOR SYSTEM 80+**

| | DESIGN ALTERNATIVE | CAPITAL COST | PERSON-REM REDUCTION (P-REM/YR) | AOC SAVINGS ($/YR) | COST/BENEFIT RATIO ($/P-REM) |
|---|---|---|---|---|---|
| A2 | 100% SG INSPECTION | $1,000,000* | 0.249 | $220.69 | $4,015,178 |
| A6 | SECONDARY SIDE GUARD PIPES | $1,100,000 | 0.00076 | $1.10 | $24,121,360 |
| B1 | ALT. DC BATTERY AND EFWS | $2,000,000 | 0.00187 | $10.76 | $17,819,558 |
| B2 | 12-HOUR BATTERIES | $300,000 | 0.00116 | $6.66 | $4,304,603 |
| B3 | ALT. PRESSURIZER AUX SPRAY | $5,000,000 | 0.207 | $215.76 | $401,534 |
| B4 | ALT. HPSI | $2,200,000 | 0.083 | $808.43 | $432,027 |
| B5 | ALT. RCS DEPRESSURIZATION | $500,000 | 0.0142 | $387.85 | $559,541 |
| B6 | DIESEL SI PUMPS (2) | $2,000,000 | 0.0834 | $812.12 | $389,943 |
| B8 | EXTENDED RWST SOURCE | $1,000,000 | 0.182 | $20.51 | $91,462 |
| C1 | THIRD DIESEL GENERATOR | $25,000,000 | 0.00045 | $2.58 | $925,920,193 |
| C2 | TORNADO-PROTECTION AAC | $3,000,000 | 0.0016 | $8.44 | $31,244,725 |
| C3 | FUEL CELLS | $2,000,000 | 0.00187 | $10.76 | $17,819,558 |
| C4 | HOOKUP FOR PORTABLE GENERATR | $10,000 | 0.00187 | $10.76 | $83,373 |
| D1 | ALT. ATWS RELIEF VALVES | $1,000,000 | 0.000965 | $37.65 | $17,232,142 |
| D2 | ATWS INJECTION SYSTEM | $300,000 | 0.000965 | $37.65 | $5,142,332 |
| D3 | DIVERSE PPS SYSTEM | $3,000,000 | 0.000965 | $37.65 | $51,774,456 |
| E1 | ALT. CONTAINMENT SPRAY | $1,500,000 | 0.00733 | $0.00 | $3,410,641 |
| E2 | FILTERED VENT (CONTAINMENT) | $10,000,000 | 0.00053 | $0.00 | $314,465,409 |
| E3 | ALT. CONCRETE COMPOSITION | $5,000,000 | 0.00487 | $0.00 | $17,111,567 |
| E4 | RV EXTERIOR COOLING | $2,500,000 | 0.0307 | $0.00 | $1,357,220 |
| E5 | ALT. H2 IGNITERS | $1,000,000 | 0.000925 | $0.00 | $18,018,018 |
| E6 | PASSIVE AUTOCA. REC. (PARS) | $760,000 | 0.000925 | $0.00 | $13,693,694 |
| E7 | MSSV AND ADV SCRUBBING | $9,500,000 | 0.246 | $0.00 | $643,631 |
| E8 | ALT. CONT. MON. SYS. | $1,000,000 | 0.00166 | $1.05 | $10,039,528 |
| E9 | CAVITY COOLING | $50,000 | 0.0307 | $0.00 | $27,144 |
| E12 | WATER COOLED RUBBLE BED | $18,800,000 | 0.00487 | $0.00 | $64,339,493 |
| E13 | REFRACTORY LINED CRUCIBLE | $108,000,000 | 0.00487 | $0.00 | $369,609,856 |

* 100% SG costs are an annual cost and used directly to calculate $/p-rem averted.

## 6.0 REFERENCES


1. 50FR32138, Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, August, 1985.

2. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Draft for Comment.

3. 50FR30028, Safety Goals for the Operations of Nuclear Power Plants; Policy Statement, August 1986.

4. CESSAR-DC, "Combustion Engineering Standard Safety Analysis Report - Design Certification," Amendment V, April 29, 1994.

5. SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Design," July 31, 1991.

6. Varga, S.A., "Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2," Docket nos. 50-352/353, August 16, 1989.

7. Memorandum from A. C. Thadani, et al, to C. I. Grimes, "Supplement to the Final Environmental Statement - Comanche Peak Generating Station, Units 1 and 2," October 23, 1989.

8. SECY-89-102, "Implementation of Safety Goal Policy," March 30, 1989.

9. "Advanced Light Water Reactor Utility Requirements Document," Volume II, ALWR Evolutionary Plant, Chapter 1, Appendix A, PRA Key Assumptions and Groundrules, Revision 3, EPRI, November 1991.

10. Hedrick, G.E., Duke Power, Letter to Sugnet, W.R., EPRI, "ALWR PRA Key Assumptions and Groundrules (KAG) Reference Site CRAC2 Input Files and Narrative Duke File: ASI-1407," April 17, 1989.

11. Memorandum from W. J. Dircks to the Commission, "Bases for Quantification of Offsite Costs," October 23, 1985.

12. NUREG/CR-4691, "MELCOR Accident Consequence Code System (MACCS) - User's Guide," Prepared for U.S. Nuclear Regulatory Commission by Sandia National laboratories, February 1990.

13. NUREG-1150, "Severe Accident Risks : An Assessment for Five US Nuclear Power Plants," January 1991.

14. NUREG/CR-3908, "Survey of the State of Art in Mitigation Systems," Prepared for U.S. Nuclear Regulatory Commission by R&D Associates, December 1985.

15. NUREG/CR-4920, "Assessment of Severe Accident Prevention and Mitigation Features: PWR, Large Dry Containment Design," July 1988.

16. Memorandum from C. J. Heltemes to F. P. Gillespie, "GI-163, Multiple Steam Generator Tube Leakage," September 28, 1992.

17. Letter from Brinkman, C., ABB-CE, to NRC, "System 80+ Steam Generator Tube Rupture Report," LD-93-145, October 6, 1993.

18. CEN-408, "Generic Issue - 23, Evaluation of the Reactor Pump Seal Integrity Issue," Combustion Engineering Inc., September, 1991.

19. "FERC Staff Recommends Allowing Recovery of Most Yankee Costs," Nucleonics Week, September 9, 1993, Page 2.

20. "Quantification of Passive Autocatalytic Recombiners for Combustible Gas Control in ALWR Containments," EPRI ALWR Program, April 8, 1993.

21. West, John, et. al, "Conceptual Design of a Post Accident Vacuum Containment System," ANS Transactions, Washington, D.C., November 1984.

22. Nucleonics Week, December 3, 1992, Page 13.

23. "Nuclear Insurance Newsletter," Johnson & Higgins Inc., January, 1990 (90-1).

24. "Nuclear Insurance Newsletter," Johnson & Higgins Inc., July, 1990 (90-2).

ATTACHMENT 2

**TECHNICAL SUPPORT DOCUMENT**
**FOR AMENDMENTS TO**
**10 CFR PART 51**
**CONSIDERING SEVERE ACCIDENTS UNDER NEPA**
**FOR PLANTS OF SYSTEM 80+ DESIGN**
**(Rev. ~~1~~)**
2

ABB-COMBUSTION ENGINEERING
WINDSOR, CONNECTICUT
~~OCTOBER 7, 1994~~
JANUARY ~~4~~, 1995
5

TECHNICAL SUPPORT DOCUMENT
FOR AMENDMENTS TO
10 CFR PART 51
CONSIDERING SEVERE ACCIDENTS UNDER NEPA
FOR PLANTS OF SYSTEM 80+ DESIGN

(4) Bases for concluding that System 80+ meets the Commission's safety goals and objectives as set forth in the Safety Goal Policy Statement.

Consequently, the conclusions are drawn in Chapter 19 that further modifications to the System 80+ design to reduce severe accident risk are not warranted. The National Environmental Policy Act (NEPA) requires the consideration of reasonable alternatives to proposed major Federal actions significantly affecting the quality of the human environment, including alternatives to mitigate the impacts of the proposed action. In 1989, a Federal Court of Appeals determined that NEPA required consideration of certain Design Alternatives; namely, Severe Accident Mitigation Design Alternatives (SAMDAs).(Limerick Ecology Action v. NRC, 869 F.2d 719, 3rd Cir. 1989). The court indicated that "[SAMDAs] are, as the name suggests, possible plant design modifications that are intended not to prevent an accident, but to lessen the severity of the impact of an accident should one occur" (Id. at 731). The court rejected the use of a policy statement as an acceptable basis for closing out NEPA consideration of SAMDAs in a licensing proceeding, because, among other things, it was not a rulemaking (Id. at 739).

Subsequent to the Limerick decision, the NRC issued Supplemental Final Environmental Impact Statements (FES) for the Limerick and Comanche Peak facilities that considered whether there were any cost-effective SAMDAs that should be added to these facilities ("NEPA/SAMDA FES Supplements"). On the basis of the evaluations in the supplements (called "NEPA/SAMDA evaluations"), the NRC determined that further modifications would not be cost-effective and were not necessary in order to satisfy the mandates of NEPA.

In recognition of the Limerick decision, the Commission is requiring NEPA consideration in 10 CFR Part 52 licensing of whether there are cost-effective SAMDAs that should be added to a new reactor design to reduce severe accident risk. While this consideration could be done later on a facility-specific basis for each combined license application under Subpart C to 10 CFR Part 52, the Commission has decided that maintenance of design standardization will be enhanced if this is done on a generic basis for each standard design in conjunction with design certification (SECY-91-229[5]). That is, the Commission has decided to resolve the NEPA/SAMDA question through rulemaking at the time of certification in a so-called unitary proceeding, rather than in the context of later licensing proceedings.

Recently, the NRC Staff expanded the definition of SAMDAs to encompass Design Alternatives to prevent severe accidents, as well as mitigate them[2]. By doing so, the Staff makes the set of SAMDAs considered under NEPA the same as the set of DAs to prevent or mitigate severe accidents considered in satisfaction of the Commission's severe accident requirements and policies.

### 1.2 Purpose

The purpose of this Technical Support Document is to provide a basis for determining the status of severe accident closure under NEPA for the System 80+ design. The document supports a determination, which could be codified in a manner similar to the format of the Waste Confidence Rule (10 CFR § 51.23), as proposed in amendments to 10 CFR Part 51. These amendments would provide that:

(1) For the System 80+ design all reasonable steps have been taken to reduce the occurrence of a severe accident involving substantial damage to the core and to mitigate the consequences of such an accident should one occur. Additionally, all reasonable steps have been taken to reduce the radiological environmental impacts from normal reactor operations, including expected operational occurrences, to As Low As Reasonably Achievable (ALARA). (See Appendix 19A of the System 80+ CESSAR-DC),

(2) No further cost-effective SAMDAs to the System 80+ design have been identified to mitigate the consequences of or prevent a severe accident involving substantial damage to the core, and

(3) No further evaluation of severe accidents for the System 80+ design, including SAMDAs to the design, is required in any environmental report, environmental assessment, environmental impact statement or other environmental analysis prepared in connection with issuance of a Combined License for a nuclear power plant referencing the System 80+ design.

The evaluation presented in this document is modeled after that found in the NEPA/SAMDA Final Environmental Statement (FES) Supplements for the Limerick[6] and Comanche Peak[7] facilities. Additional information concerning the radiological risk from severe accidents for the System 80+ design presented in this document is based on Chapter 19 of the System 80+ CESSAR-DC and Appendix 19A to CESSAR-DC.

Insert (A) from pg. 8

### 1.3 Description of Technical Support Document

Section 2.0 of this report provides an overview of the radiological risks from nuclear power plants and evaluations of SAMDAs under NEPA. Section 3.0 provides a NEPA/SAMDA evaluation of the radiological risks from normal operations and severe accidents for the System 80+ design. Chapter 4.0 presents the discussion and results of the cost-effectiveness evaluation of the potential SAMDA modifications. Section 5.0 presents the summary and conclusions, and references are included in Section 6.0.

## 2.0 EVALUATIONS OF RADIOLOGICAL RISK FROM NUCLEAR POWER PLANTS

### 2.1 Evaluation of SAMDAs under NEPA and Limerick Ecology Action

Limerick Ecology Action stands for two propositions. First, NEPA requires explicit consideration of SAMDAs unless the Commission makes a finding that the severe accidents being mitigated are remote and speculative. Second, the Commission may not make this finding and dispose of NEPA consideration of SAMDAs by means of a policy statement. The purpose of evaluating SAMDAs under NEPA is to assure that all reasonable means have been considered to mitigate the impacts of severe accidents that are not remote and speculative. As discussed above, the Commission has indicated that it will resolve the NEPA/SAMDA issue for a new reactor design in the same proceeding, called a unitary proceeding, in which it certifies that design.

The Commission's Severe Accident and Safety Goal policy statements require the Commission to make certain findings about each new reactor design. For evolutionary designs, of which the System 80+ is one, this must be done by the Staff in conjunction with issuance of Final Design Approval (FDA) and by the Commission in conjunction with Design Certification. First, the Commission must find that an evolutionary plant meets the safety goals and objectives; i.e., that the radiological risk from operating an evolutionary plant will be acceptable, meaning that any further reduction in risk will not be substantial.

Second, the Commission must find that all reasonable means have been taken to reduce severe accident risk in the evolutionary plant design. As part of the basis for making this finding, the cost-effectiveness of risk reduction alternatives of a preventive or mitigative nature must be evaluated.

Chapter 19 of the System 80+ CESSAR-DC and this Technical Support Document demonstrate~~s~~ that these findings can be made for the System 80+ design. Given the nature and findings of these severe accident and safety goal evaluations, ABB-CE believes that a sufficient basis exists for finding by rule that further consideration of severe accidents, including evaluation of SAMDAs pursuant to NEPA, is neither necessary nor reasonable.

### 2.2 Cost/benefit Standard for NEPA Evaluation of SAMDAs

The Limerick decision interpreted NEPA to require evaluation of SAMDAs for their risk reduction potential. In implementing the court's decision, the NRC considered the cost-effectiveness of each candidate SAMDA in mitigating the impact of a severe accident, using the $1,000 per person-rem averted standard. This standard is a surrogate for all offsite consequences.

The basic approach in this study is to rank the SAMDAs in terms of their cost-effectiveness in mitigating the impact of a severe

accident. The criterion applied is the $1,000 per person-rem averted standard, which is what the Commission has historically used in distinguishing among, and ranking, Design Alternatives, including SAMDAs.

The Commission has used this standard in the context of both safety and NEPA analyses. For example, in the context of safety analysis, the standard has been used to perform evaluations associated with implementation of ~~10 CFR Part 50, Appendix I;~~ the Safety Goal Policy Statement; the Severe Accident Policy Statement; and § 50.34(f) requirements. In the context of environmental analysis, it has been used in the Limerick and Comanche Peak NEPA/SAMDA FES Supplements[6,7] and in NUREG-1437[2].

As indicated above, the Commission is preparing a Generic Environmental Impact Statement for License Renewal of Nuclear Plants. The draft statement, NUREG-1437, makes clear that the use of this standard in the evaluation of severe accident risk reduction alternatives, which include SAMDAs, is acceptable (see NUREG-1437, p. 5-108). Additionally, Appendix I determinations are used to satisfy NEPA requirements with respect to radiological impacts from normal operations.

On the basis of these considerations, the cost/benefit ratio of $1,000 per person-rem averted is viewed as an acceptable standard for the purposes of evaluating SAMDAs under NEPA.

### 2.3 Socio-Economic Risks for Severe Accidents

As discussed above in Section 2.2, the Commission uses the $1,000/person-rem averted standard as a surrogate for all offsite consequences[8]. However, Environmental Impact Statements (EIS) for nuclear power plants provide separate, general discussions of the socio-economic risks from severe accidents. In keeping with this precedent, ABB-CE is providing a general discussion of socio-economic risks for the System 80+ design, based in large measure on the discussion of such risks in NUREG-1437.

The term "socio-economic risk from a severe accident" means the probability of a severe accident multiplied by the socio-economic impacts of a severe accident. "Socio-economic impacts," in turn, relate to offsite costs. The offsite costs considered in NUREG-1437 are:

- Evacuation costs
- Value of crops or milk contaminated and condemned
- Costs of decontaminating property where practical
- Indirect costs due to the loss of the use of property or incomes derived therefrom (including interdiction to prevent human injury), and
- Impacts in wider regional markets and on sources of supply outside the contaminated area.

## 3.0 RADIOLOGICAL RISK FROM NORMAL OPERATIONS AND SEVERE ACCIDENTS IN PLANTS OF SYSTEM 80+ DESIGN

### 3.1 Radiological Risk from Normal Operations of a System 80+ Plant

Sections 50.34a and 50.36a of 10 CFR Part 50 require, in effect, that nuclear power reactors be designed and operated to keep levels of radioactive materials in gaseous and liquid effluents during normal operations, including expected operational occurrences, "As Low As Reasonably Achievable (ALARA)". Compliance with the guidelines in Appendix I to 10 CFR Part 50 is deemed a conclusive showing of compliance with these ALARA requirements.

In addition to specifying numerical limits, Appendix I also requires an applicant to include in the radwaste system "all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost/benefit return, can, for a favorable cost/benefit ratio, affect reductions in dose to the population reasonably expected to be within 50 miles of the reactor." The standard to be used in making this assessment is the cost/benefit ratio of $1,000 per person-rem averted.

The System 80+ design complies with the guidance of Appendix I, as documented in Chapter 12 of the System 80+ CESSAR-DC. Consequently, further consideration of alternatives to reduce the radiological risks from normal operation of a plant of System 80+ design is not warranted in order to satisfy NEPA. Moreover, the radiological impacts from normal operation of a System 80+ standard plant are environmentally insignificant.

Non-radiological impacts from operation of a System 80+ plant include those from the site-specific circulating system that removes heat from the reactor (e.g., cooling towers, cooling lakes, etc.), intake systems for the water in the circulating systems, discharge systems for the water in the circulating system, biocide treatment in circulating water to prevent fouling by organisms, chemical waste treatment and disposal, sanitary waste treatment system, and electrical transmission facilities. Each of these systems is part of that portion of the System 80+ design that is not being certified because it is site-specific, although it may interface with the certified portion. It may be appropriate to consider DAs for non-radiological systems under NEPA. However, the choice of alternative will not have an effect on that portion of the System 80+ design that is being certified.

### 3.2 Severe Accidents in Plants of System 80+ Design

Chapter 19 of the System 80+ CESSAR-DC, "Probabilistic Risk Assessment," establishes that the Commission's severe accident safety requirements have been met for the System 80+ design, including treatment of internal and external events, uncertainties, performance of sensitivity studies, and support of conclusions by appropriate deterministic analyses and the evaluations required by

10 CFR §50.34(f). It also establishes that the Commission's safety goals have been met.

Specifically, the following topics were addressed in Chapter 19:

(1) Consideration of the contributions of internal events (Section 19.4), Shutdown events (Section 19.8) and external events (Section 19.7) to severe accident risks, including a seismic risk analysis based on the application of the seismic margins methodology (Section 19.7.5),

to pg. 3

(2) Identification of the System 80+ dominant accident sequences,

(3) Identification of severe accident risk reduction features which were included in the System 80+ design to achieve accident prevention and mitigation (Section 19.15.1), and

~~(4) Consideration of additional modifications, evaluated in accordance with §50.34(f)(1) (Appendix 19A).~~

A

This document ~~Appendix 19A of CESSAR-DC~~ presents the bases for concluding that further modifications to the System 80+ design are not warranted in order to reduce the risk of a severe accident through the addition of design features to prevent or mitigate a severe accident.

Chapter 19 of CESSAR-DC addresses how the goals of the Severe Accident Policy Statement have been met for plants of System 80+ design. These goals include:

- Prevention of core damage,
- Prevention of early containment failure for dominant accident sequences,
- Evaluation of the effects of hydrogen generation,
- Heat removal to reduce the probability of containment failure,
- Prevention of hydrogen deflagration and detonation
- Offsite dose, and
- Containment conditional failure probability.

Specific conclusions concerning severe accidents for the System 80+ design based on the Chapter 19 evaluations and this document are as follows:

(1) Core Damage Frequency. The System 80+ core damage frequency for power operation, including the scoping values for fire and flood, was determined to be 1.96E-6 per reactor year (Table 4-2 ~~19.15.4-2~~).

(2) Conditional Containment Failure Probability. The conditional containment failure probability was shown to be 0.0~~6~~ 4 (Section 19.12.2.3). This is significantly below the goal of 0.1,

(3) Probability of Large Offsite Dose. The probability of exceeding a whole body dose of 25 rem at a distance of one-



In addition, consideration of design modifications in accordance with §50.34(f)(1) is presented in Section 4 of this document.

half mile from the System 80+ design site boundary was determined to be less than 5.3E-8 per reactor year (Section 19.13), and

(4) ~~Residual Radiological Risk.~~ Residual radiological risk from severe accidents in plants of System 80+ design is summarized in Table 19A.4-3 of ~~Appendix 19A of CESSAR-DC~~ this document. The cumulative exposure risk to the population within 50 miles of the plant of System 80+ design site boundary is approximately 17 person-rem for an assumed plant life of 60 years. (0.282 mr/yr * 60)

### 3.~~5~~2 Dominant Severe Accident Sequences for Plants of System 80+ Design

In performing the Probabilistic Risk Assessment (PRA) for the System 80+ design, many severe accident sequences were identified and evaluated . For each sequence, the analysis identified an initiating event and traced the accident's progression to its end. For sequences involving core damage, conditional containment failure probabilities and offsite consequences were estimated. The accident scenarios were binned according to radiological release (source term) parameters, and twenty-three release classes were defined and quantified. The dominant cases in terms of offsite risk are containment bypass sequences. Table 19A.4-1 of ~~Appendix 19A of CESSAR-DC~~ this document defines the release classes. The complete radiological consequence analysis of the dominant sequences can be found in Section 19.13 of CESSAR-DC.

ABB-CE believes that the severe accident analysis in Chapter 19 of CESSAR-DC is complete and core damage sequences not included in that analysis are not dominant can be deemed remote and speculative. ~~The analysis provides a sufficient basis for the Commission to review.~~

### 3.~~6~~3 Overall Conclusions from Chapter 19 of CESSAR-DC

The specific conclusions about severe accident risk discussed above support the overall conclusion that the environmental impacts of severe accidents for plants of System 80+ design represent a low risk to the population and to the environment. For the System 80+ design, all reasonable steps have been taken to reduce the occurrence of a severe accident involving substantial damage to the core and to mitigate the consequences of such an accident should one occur. No further cost-effective modifications to the System 80+ design have been identified to reduce the risk from a severe accident involving substantial damage to the core. No further evaluation of severe accidents for the System 80+ design is required to demonstrate compliance with the Commission's severe accident requirements or its policy or safety goals.

Nuplex 80+ makes possible tracking of leakage within containment and correlation of multiple symptoms. In addition to increased costs and complexity of additional trips and ESF actuation paths, the additional trips could decrease plant availability and increase the potential for equipment challenge (false actuation leading to transients) for a negligible improvement in plant safety. Because of the small public risk associated with the LBLOCA and the sophistication of the current protection system, this DA was not further considered.

## B. INCREASE DECAY HEAT REMOVAL RELIABILITY

This group of Design Alternatives has been grouped together as improved decay heat removal reliability. The SIS and Safety Depressurization System (SDS) were grouped here because of their feed and bleed capability.

### B1 ALTERNATIVE DC BATTERIES AND EFWS

This DA addresses the release classes where emergency feedwater is lost after battery depletion during a station blackout. The System 80+ design already has an improved battery system that will carry the DC loads for 8 hours. However, there are still accident sequences where the batteries are depleted and emergency feedwater is lost leading to core damage. The improved DC batteries and EFWS DA is assumed to have the capability to remove decay heat using batteries and the turbine-driven feedwater pump for whatever time period that is required (without any failures). This DA prevents core damage and, therefore, removes two of the release classes (Table 4-9).

Design of a battery system with unlimited capacity is not possible. However, to increase the existing battery capacity for the EFWS pumps from the current System 80+ design capacity of 8 hours to 72 hours will require 9 times the number of current battery cells and thus approximately 9 times the space for building storage. The increased building space will also increase the Heating, Ventilation and Air Conditioning (HVAC) requirements. The cost for the extra battery cells, building volume and increased HVAC requirements is estimated to exceed $2 million. In the Comanche Peak SAMDA, additional batteries were estimated to cost between $1.3M and $3M.

### B2 12-HOUR BATTERIES

The DA described in Section 18A.B1 is for an ideal battery system. This DA is for a specific and technically realistic DA of using a battery system that would maintain load for twelve hours. Such an improvement would decrease the probability of failure to restore

This DA is similar to