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Subject: **ESBWR Severe Accident Mitigation Design Alternatives, NEDO-33306, Revision 3**

The purpose of this letter is to submit Revision 3 of the GE Hitachi Nuclear Energy (GEH) Licensing Topical Report NEDO-33306, *ESBWR Severe Accident Mitigation Design Alternatives* (Enclosure 1).

If you have any questions or require additional information, please contact me.

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

Enclosure:

1. MFN 07-062 Supplement 4, NEDO-33306, Revision 3, *ESBWR Severe Accident Mitigation Design Alternatives*.

cc:	AE Cubbage	USNRC (with enclosure)
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	CW Bagnal	GEH/Wilmington (with enclosure)
	eDRF Section	0000-0123-4257
	eCM Section	0000-0123-4572

MFN 07-062 Supplement 4

Enclosure 1

ESBWR Severe Accident Mitigation Design Alternatives
NEDO-33306 Revision 3

GE Hitachi Nuclear Energy

NEDO-33306
Revision 3
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September 2010

Licensing Topical Report
ESBWR Severe Accident Mitigation Design Alternatives

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This document, NEDO-33306, Revision 3, contains no proprietary information.

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The information contained in this document is furnished as reference to the NRC Staff for the purpose of obtaining NRC approval of the ESBWR Certification and implementation. The only undertakings of GE Hitachi Nuclear Energy (GEH) with respect to information in this document are contained in contracts between GEH and participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone other than that for which it is intended is not authorized; and with respect to any unauthorized use, GEH makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

Table of Contents

1 INTRODUCTION

1.1 Background

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 off-site consequences, see Severe Accident Policy Statement, 50 Federal Register 32,138,32,139 (August 8,1985). For new reactor designs, such as the ESBWR, the Nuclear Regulatory Commission (NRC), in satisfaction of its severe accident safety requirements, is requiring, among other things, the evaluation of design alternatives 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 designs are set forth in 10 CFR Part 52, paragraph 52.47(a) (1) (ii), (iv) and (v). Paragraph 52.47(a) (1) (ii) references the Commission's Three Mile Island safety requirements in 10 CFR 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 Severe Accident Policy Statement elaborates what the Commission is requiring for new designs. The Safety Goal Policy Statement sets goals and objectives for determining an acceptable level of radiological risk.

GE performed a probabilistic risk assessment (PRA) for the ESBWR design to achieve the following objectives:

- Identify the dominant severe accident sequences and associated source terms for the design.
- Modify the design, on the bases of PRA insights, to prevent or mitigate severe accidents and reduce the risk of severe accidents.
- Provide a basis for concluding that all reasonable steps have been taken to reduce the chances of occurrence, and to mitigate the consequences, of severe accidents.
- Provide a basis for concluding that the NRC safety goals are met by the plant design.

The ESBWR PRA analysis is provided in NEDO-33201. The PRA was performed in accordance with the requirements of 10 CFR 52 and 10 CFR 50.34(f)(1)(i) which requires the performance of a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.

The U.S. Court of Appeals decision, in Limerick Ecology Action v. NRC, 869 F.2d 719 (3rd Cir. 1989), effectively requires the NRC to include consideration of certain severe accident mitigation design alternatives (SAMDAs) in the environmental impact review performed under Section 102(2)(c) of NEPA.

These two requirements share a common purpose to consider alternatives to the proposed design, to evaluate potential alternative improvements in the plant design that increase safety performance during severe accidents, and to prevent reasonable alternatives from being foreclosed. As a matter of discretion, the Commission has determined that considering SAMDAs is consistent with the intent of 10 CFR Part 52 for early resolution of issues, finality of design issues resolution, and achieving the benefits of standardization.

Recently, the NRC Staff expanded the concept of SAMDAs to encompass design alternatives to prevent severe accidents, as well as mitigate them. See NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," (Volume I, p. 5-100). By doing so, the Staff makes the set of SAMDAs considered under NEPA the same as the set of SAMDAs considered in satisfaction of the Commission's severe accident requirements and policies.

1.2 Purpose

The purpose of this subsection is to demonstrate that all cost effective steps have been taken to reduce the risk associated with operation of plants of ESBWR design. The basis for determining the status of severe accident closure under NEPA for the ESBWR design is also provided. 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 amendments to 10 CFR Part 51. These amendments would provide that:

- For the ESBWR 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 were taken to reduce the radiological environmental impacts from normal reactor operations, including expected operational occurrences, to as low as reasonably achievable (ALARA).
- No further cost-effective SAMDAs to the ESBWR design have been identified to mitigate the consequences of or prevent a severe accident involving substantial damage to the core; and,
- No further evaluation of severe accidents for the ESBWR 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 a certified ESBWR design.

The evaluation presented in this document is modeled after that found in the Limerick and Comanche Peak NEPA/SAMDA Final Environmental Statement (FES) Supplements for those facilities. Additional information concerning the radiological risk from severe accidents for those plants is not found in the supplements, but in the FESs for the Limerick and Comanche Peak facilities. That information with respect to the ESBWR design is presented in this document. The discussion herein of the radiological risk from severe accidents is based on the ESBWR PRA (NEDO-33201).

2 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, that NEPA requires explicit consideration of SAMDAs unless the Commission makes a finding that the severe accidents being mitigated are remote and speculative. Second, that 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 in the same proceeding, called a unitary proceeding, in which it certifies a new reactor 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 ESBWR is one, this must be done by the Staff in conjunction with NEPA approval and by the Commission in conjunction with 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.

2.2 Cost/Benefit Evaluation of SAMDAs

The net value of a design alternative is the difference between the averted cost benefit due to the modification, and the cost of the enhancement. The methodology to calculate averted costs is based on the NRC's guidance for performing cost-benefit analysis in the Regulatory Analysis Technical Evaluation Handbook, NUREG/BR-0184 (Reference 14). The NRC’s method determines the net value for each design alternative according to the following equations:

Net Value = (APE + AOC + AOE + AOSC) – COE, where

APE = present value of averted public exposure ($)

AOC = present value of averted offsite property damage costs ($)

AOE = present value of averted occupational exposure ($)

AOSC = present value of averted onsite costs ($)

COE = cost of enhancement ($)

2.2.1 Averted Public Exposure (APE)

The baseline annual offsite exposure risk is converted to dollars using the NRC's conversion factor of \$2,000 per person-rem, and discounting to present value using the NRC standard formula (Reference 14):

$W_{pha} = C \times Z_{pha}$

Where:

W_{pha} = Monetary value of public health risk after discounting (Averted Public Exposure)

$C = [\exp(-rt_i)-\exp(-rt_f)]/r$ Present-value discount factor

t_f = Facility life = 60 years

t_i = Years before facility begins operating

r = Real discount rate (as fraction) = 0.03/year

Z_{pha} = Monetary value of public health (accident) risk per year before discounting (\$/year)

For the purpose of this analysis, the net present value costs are calculated relative to a plant in its first year of operation. That is, t_i (years before facility begins operating) = 0. This simplifies the discount factor formula to:

$C = [1-\exp(-rt_f)]/r.$

The calculated value for C using 60 years and a 7 percent discount rate is 14.07. Using a 3 percent discount rate, C is 27.82. A discount rate of .03 per year will be used because it represents a more conservative estimate.

The ESBWR PRA Level 3 analysis estimates an annual offsite population dose risk (W_{pha}) of 0.035 sievert per year (Table 5 Total Weighted Population Dose.) This is based on the Level 1 and 2 internal events results, which are used to calculate whole-body person-rem per year received by the total population within a 50-mile radius of the site. It is the product of the predicted radiological release fraction and the radionuclide inventory for each release category. Therefore, calculating the discounted monetary equivalent of accident risk involves by converting the dose to rem per year and multiplying by \$2,000 and by the C value (present-value discount factor) of 27.82. The calculated averted public exposure (APE) cost is \$194,740.

2.2.2 Averted Offsite Property Damage Costs (AOC)

Averted offsite property damage costs are calculated using the following formula:

AOC = [offsite economic costs associated with a severe accident (on a per event basis)]

x [present value discount factor]

The maximum theoretical CDF reduction is the total CDF of 1.12 E-7/year. The total CDF includes internal and external events for at-power and shutdown conditions (reference Table 4.) It represents the upper bound value for CDF reduction, which assumes that a design change could eliminate all core damage risk. Typically, a design change with a high risk reduction worth would not affect the entire risk profile, and would reduce risk by significantly less. The Level 3 analysis, which uses the MACCS2 code in conjunction with generic site data in the EPRI Utility Requirements Document (Reference 15), calculates an annual offsite economic risk of $1,931 (Table 5 Total Weighted Offsite Cost.) The calculated value for offsite economic costs caused by severe accidents must be discounted to present value as well by applying the present-value discount factor (27.82). The resulting present value AOC is $53,720.

2.2.3 Averted Occupational Exposure (AOE) Costs

The NRC methodology in Reference 14, Section 5.7.3, involves separately evaluating immediate and long-term doses.

Immediate Dose - For the case where the plant is in operation, the equation that NRC recommends using (Reference 14) is:

$W_{IO} = R \text{ x } F \text{ x } D_{IO} \text{ x } C$

Where:

W_{IO} = Monetary value of accident risk avoided due to immediate doses, after discounting

R = Monetary equivalent of unit dose ($/person-rem)

F = Accident frequency (events/yr)

D_{IO} = Immediate occupational dose (person-rem/event)

C = Present-value discount factor

The values used in the analysis are:

R = $2,000/person-rem

F = 1.12×10^{-7} /yr (total core damage frequency)

D_{IO} = 3,300 person-rem/accident (best estimate), 14,000 person-rem/accident (upper bound estimate) (Reference 14)

C = 27.82

The estimate of the immediate dose cost is:

$W_{IO} = 2{,}000 \times 1.12\times10^{-7} \times 3{,}300\times 27.82$

W_{IO}= \$87 (using 14,000 p-rem/acc.)

Long-Term Dose - For the case where the plant is in operation, the NRC equation (Reference 14) is:

$W_{LTO} = R \text{ x } F \text{ x } D_{LTO} \text{ x } C \text{ x } \{[1 - \exp(-rm)]/rm\}$

Where:

W_{LTO} = monetary value of accident risk avoided long-term doses, after discounting, \$

D_{LTO} = long-term occupational dose

The values used in the analysis are:

R = \$2,000/person-rem

r = 0.03

D_{LTO} = 20,000 person-rem/accident (best estimate), 30,000 person-rem/accident (upper bound estimate) (Reference 14)

m = 10 year clean-up period (Reference 14)

The best estimate of the long-term dose is:

$W_{LTO} = 2{,}000 \times 1.12\times10^{-7} \times 30{,}000 \times 27.82 \times \{[1 - \exp(-0.03\times10)]/(0.03\times10)\}$

W_{LTO} = \$162.

Total Occupational Exposure - Combining the equations for immediate dose (W_{IO}) and long-term dose (W_{LTO}) and using the above numerical values, the total accident related on-site (occupational) exposure avoided is:

AOE = W_{IO} + W_{LTO} = \$ \$249.

2.2.4 Averted Onsite Costs (AOSC)

Averted onsite costs include cleanup and decontamination costs. Repair and refurbishment costs are considered for recoverable accidents only and not for severe accidents. The net present value that NRC provides for cleanup and decontamination for a single event is \$1.1 billion, discounted over a 10-year cleanup period, with an upper bound estimate of \$1.5 billion. The NRC uses the following equation in integrating the net present value over the maximum number of remaining facility years:

$U_{CD} = PV_{CD} \text{ x } C$

Where:

PV_{CD} = Net present value of a single event

$PV_{CD} = \$1.5\times10^9 * 1.12\times10^{-7}$

The resulting net present value of cleanup integrated over the license renewal term must be multiplied by the total core damage frequency to determine the expected value of cleanup and decontamination costs. The resulting best estimate present value monetary equivalent is $4,674.

2.2.5 Replacement Power Cost (RPC)

Long-term replacement power costs are determined following the NRC methodology in Reference 14. The net present value of replacement power for a single event, PVrp, is determined using the following equation:

$PV_{rp} = [B/r] * [1 - \exp(-rt_f)]^2$

Where B is a constant representing a string of replacement power costs that occur over the lifetime of a reactor after an event. For a 910 MWe "generic" reactor, section 5.7.6.2 of Reference 14 uses a value of $1.4E+9/year, assuming a 3% discount rate.

PV_{rp} = net present value of replacement power for a single event, ($)

For the ESBWR, B is scaled to the following value:

B = 1.4E+9/year x (1585/910) = 2.4E+9/year

The net present value of replacement power for a single event is:

PV_{rp} = 5.57 E+10/year. To attain a summation of the single-event costs over the entire facility lifetime, the following equation is used:

$U_{rp} = [PV_{rp} /r] * [1 - \exp(-rt_f)]^2$

Where:

U_{rp} = net present value of replacement power over life of facility ($/year)

Multiplying the result by the CDF results in a present value replacement power cost of $144,480.

2.2.6 Maximum Averted Risk Benefit

Averted Public Exposure Cost:	\$194,740
Averted Offsite Property Damage Cost:	\$53,720
Averted Occupational Exposure Cost:	\$249
Averted Onsite Cost:	\$4,674
Replacement Power Cost:	\$144,480
Total (Maximum Averted Cost Benefit):	\$397,863

In order to be cost-beneficial, a SAMDA Cost of Enhancement must be less than \$397,863. The estimated cost of replacement power has a large effect (36%) on averted costs, but it is not a radiological cost. Design modifications that reduce core damage frequency or offsite releases do not typically have an effect on the cost of replacement power. The averted offsite property damage cost is based on the total core damage frequency, which is the sum of internal and external event calculations for at-power and shutdown conditions. The external event calculations were developed for the design certification PRA to be bounding analyses in order to be applicable to the various U.S site locations. Site-specific core damage frequencies are expected to be lower, thus, the total core damage frequency used in this assessment is considered to be conservative.

3 RADIOLOGICAL RISK FROM NORMAL OPERATIONS OF AN ESBWR PLANT

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, effect reductions in dose to the population expected to be within 50 miles of the reactor". The standard to be used in making this assessment is the cost/benefit ratio of $2000 per person-rem averted.

The ESBWR design complies with the guidance of Appendix I, therefore further consideration of cost beneficial alternatives to reduce the radiological risks from normal operation of a plant of ESBWR design is not warranted in order to satisfy NEPA.

Non-radiological impacts from operation of an ESBWR plant include those from the circulating system which 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 ESBWR design which is not being certified because it is site-specific.

It may be appropriate to consider design alternatives for non-radiological systems under NEPA. However, the choice of alternatives will not have an effect on the portion of the ESBWR design that is being certified. Consideration of alternative designs to systems affecting non-radiological impacts must be done on a site-specific basis. 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.

4 SEVERE ACCIDENTS

4.1 Severe Accidents in Plants of ESBWR Design

NED0-33201 establishes that the Commission's severe accident safety requirements have been met for the ESBWR 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 Part 50.34(f). It also establishes that the Commission's safety goals have been met.

Specifically, the following topics were addressed in NEDO-33201:

- Consideration of the contributions of internal events and external events to severe accident risks, including a seismic risk analysis based on the application of the seismic margins methodology;
- Identification of the ESBWR dominant accident sequences;

Section 19.1 of Chapter 19 of the ESBWR DCD addresses how the goals of the Severe Accident Policy Statement have been met for plants of ESBWR design.

Specific conclusions concerning severe accidents for plants of ESBWR design based on the NEDO-33201 evaluations are as follows:

- Core Damage Frequency: The ESBWR total core damage frequency was determined to be approximately 1E-7 per reactor year.
- Individual Risk (Prompt Fatality Risk). The prompt fatality risk to a biologically average individual within one mile of an ESBWR site boundary was determined to be significantly less than the goal of one-tenth of one percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. Population are generally exposed.
- Societal Risk (Latent Fatality Risk): The latent fatality risk to the population in the vicinity of an ESBWR was determined to be significantly less than the goal of one-tenth of one percent of the sum of the cancer fatality risks resulting from all other causes.
- Probability of Large Off-Site Dose: The probability of exceeding a whole body dose of 25 rem at a distance of one-half mile from an ESBWR was determined to be less than 1E-6 per reactor year.

4.2 Dominant Severe Accidents Sequences for Plants of ESBWR Design

Insights from the ESBWR PRA are examined to determine which design features are most important relative to risk, and whether or not design improvements could yield a significant risk reduction. The ESBWR risk profile is balanced and does not contain dominating contributors to risk. The most important systems, based on Risk Achievement Worth importance values, are the 6.9 kV AC PIP buses. Loss of a PIP bus during at-power operation would result in a plant trip and loss of one division of active mitigation systems. Other important components include

SLCS valves and accumulators. The top common cause failures are the mechanical binding of control rods, nonsafety-related UPS transformers, wetwell to drywell vacuum breakers, RPS scram valves, depressurization squib valves, and GDCS check valves and squib valves, feedwater line check valves, and digital I&C software. These common cause failures correspond to failures of RPS, passive functions or multiple injection functions. In each case, at least one diverse backup system is available.

The important operator actions involve recognizing the need to makeup ICS/PCCS Pool level; recognizing the need for depressurization or providing low pressure injection in particular scenarios; failure to restart feedwater pumps during certain ATWS scenarios; failure to open 6 out of 10 SRVs; and pre-initiator valve mispositioning events in the FAPCS, CRD, and RCCW systems. Information on important operator actions is incorporated into the human factors engineering program.

The ESBWR is designed to be safe with respect to internal fire events. All potential fires have been analyzed and it has been shown that the plant can be safely shut down at low risk to plant personnel and the general public.

The ESBWR is designed to be safe with respect to internal flood events and no operator actions are required to mitigate postulated floods. Although timely operator action can reduce damage to equipment and flood events, these actions are not yet developed and their benefit is not included. It has been shown that the plant can be safely shut down at low risk to plant personnel and the general public.

The ESBWR high wind analysis explicitly quantifies accident sequences initiated by tornado winds. Straight winds are lesser velocity winds that pose minimal challenges to the plant design. Hurricane winds are quantified using a bounding analysis. Due to the strength of construction of the ESBWR Category I buildings, the effects of a tornado strike are limited to Loss of Preferred Power events with a potential loss of the Condensate Storage Tank. Overall risk from tornados and high winds is further minimized by design features such as the diesel driven fire protection pump for alternate injection, and the DC batteries with a 72-hour operational life. Administrative controls will ensure that the Drywell Hatches can be closed during shutdown conditions if a loss of coolant event is initiated in the drywell.

The largest Fussell-Vesely (F-V) value for a component failure, (F-V is a measure of risk reduction), is less than 0.4. A value of 0.4 yields a change in CDF of approximately 7 E-9/year. This represents the change in CDF if all components in a functional group are made perfect, which is the theoretical maximum benefit. Actual design improvements would yield significantly lower changes in CDF. Adding diversity would require significantly higher costs, with marginal benefit. For example, adding a second vendor to supply diverse squib valves would be costly due to the first-time engineering and qualification testing costs that would be incurred.

The dominant F-V values from initiating events are less than 0.2. That is, if the dominant initiating event were eliminated, CDF would be only reduced by approximately 4 E-9/year.

The dominant insight from the shutdown PRA is the need to close the lower drywell hatches during certain shutdown conditions following a LOCA. The net benefit would be a CDF reduction of approximately 7 E-9/year.

The insights from the external events and Level 2 analysis of severe accident sequences do not identify any other potential design enhancements that could significantly reduce risk.

A PRA-based Seismic Margins Analysis is used to derive seismic vulnerability insights. Therefore, there are no CDF calculations performed. The Seismic Margins Analysis concludes that the most significant HCLPF sequences are seismic-induced loss of DC power and seismic-induced ATWS due to seismic-induced failure of the fuel channels and seismic-induced failure of the SLC tank. Based on previous industry seismic analyses, seismic risk is dominated by seismic-induced SSC failures, and not by random SSC failures or human actions. Human actions are typically not necessary until the long-term.

Overall, the maximum theoretical risk reduction from any design change is considered to be extremely low, and no further examination of cost-benefit is warranted.

4.3 Conclusions from the ESBWR PRA

The specific conclusions about severe accident risk discussed above support the conclusion that the environmental impacts of severe accidents for plants of ESBWR design represent a low and acceptable risk to the population and to the environment. As shown in Table 3, the ESBWR design already incorporates numerous plant features oriented toward reducing CDF and risk. The PRA has been used to minimize the effects of initiating events and accident sequences that have been important contributors to risk in previous BWR PRAs. No further cost-effective modifications to the ESBWR design have been identified to reduce the risk from a severe accident involving substantial damage to the core. It is unlikely that any future design changes would be justifiable on the basis of person-rem exposure because the estimated CDF changes would remain low on an absolute scale. No further evaluation of severe accidents for the ESBWR design is required to demonstrate compliance with the Commission's severe accident requirements or policy, SECY-90-016 or the EPRI ALWR Utility Requirements Document.

5 ANALYSIS OF SEVERE ACCIDENT MITIGATION DESIGN ALTERNATIVES

5.1 Screening Process

Tables 1 and 2 are lists of severe accident management design alternatives that have been compiled from a generic list compiled for License Renewal Environmental Reports (Reference 12), and from the list of SAMA issues from the ABWR SAMA study (Reference13). These lists are screened to identify candidate design alternatives for further risk-benefit consideration. The following screening criteria are applied:

1. Not applicable.
   An issue that only pertains to another class of reactors, even on a functional level.
2. Already incorporated into the ESBWR design.
   Cases where the risk-beneficial design features have already been applied to the ESBWR.
3. Not a design alternative.
   The proposed activity does not involve a design change; it is for procedural or administrative changes only.
4. Excessive Implementation Cost.
   If a SAMA requires extensive changes that obviously would exceed the maximum averted risk benefit, it is not retained.
5. Very Low Benefit.
   If the change in reliability is known to have a negligible effect on risk, it is not retained.
6. Candidate For Cost-Benefit Consideration.
   If a SAMA does not screen out from the above criteria, it is a candidate for cost-benefit analysis.

The list of 177 items has been analyzed to determine if there are cost-beneficial design alternatives that should be considered for the ESBWR. The screening analysis identifies 39 alternatives which are not applicable, primarily due to issues involving either loss of reactor coolant pump seals, which is an issue with current PWRs, or BWR-specific issues, for example, reactor core isolation cooling pump operations. There are 71 design alternatives that are similar to, or are already incorporated into the ESBWR design. A summary of these types of design features is provided below. There are 28 items that are procedural or administrative, and thus are not design features. The benefits offered by these changes are not likely to exceed those for the design modifications that are evaluated. Also, the costs of these non-design changes, such as engineering, procedure development, maintenance, and training, are likely to be substantial. There are 27 items that are not feasible because their cost would clearly outweigh any risk-benefit consideration. Finally, there are 12 issues that are considered to have very low benefit due to their insignificant contribution to reducing risk. In summary, several design enhancements relative to severe accident mitigation have already been incorporated into the ESBWR design. Potential design enhancements from generic BWR SAMA reports and from the ABWR have been evaluated on a risk-benefit basis. The relatively minor economic impacts of radiological consequences, and the very low probability of a severe accident, yield an overall risk that is significantly lower than current operating reactors. Therefore, no additional design modifications yield a positive cost benefit.

6 SUMMARY AND CONCLUSIONS

Due to the low probability of a severe accident, and the mitigating capabilities in the ESBWR design, the economic risks of a severe accident are not significant, and do not justify design modifications. This is attributed to multiple layers of reliable safety functions that provide significant protection to the public and the environment. A detailed analysis of specific design alternatives from previous industry studies and from ESBWR PRA insights supports the conclusion that there are no additional design changes warranting further consideration. In order to be cost-beneficial, a SAMDA Cost of Enhancement must be less than $397,863 for a change that reduces the core damage frequency by approximately 1 E-7/year. Based on this analysis, that none of the SAMDA candidates are cost-beneficial.

Insights from the ESBWR PRA determine which design features are most important relative to risk. Due to the low absolute value of core damage and offsite release risk, there are no design improvements that could yield a significant risk reduction.

Severe accident mitigation design alternatives that have been compiled from a generic list for License Renewal Environmental Reports (Reference 12), and from SAMA issues from the ABWR SAMA study (Reference13) have been evaluated. No design alternatives warrant further risk-benefit consideration. The ESBWR design already incorporates numerous plant features oriented toward reducing CDF and risk. Several examples are shown in Table 3. The PRA has been used to minimize the effects of initiating events and accident sequences that have been important contributors to risk in previous BWR PRAs. No further cost-effective modifications to the ESBWR design have been identified to reduce the risk from a severe accident involving substantial damage to the core. It is unlikely that any future design changes would be justifiable on the basis of person-rem exposure because the estimated CDF changes would remain low on an absolute scale.

7 REFERENCES

1 Brookhaven National Laboratory, "Assessment of Severe Accident Prevention and Mitigation Features," NUREG/CR-4920, July 1988.

2 R&D Associates, "Design and Feasibility of Accident Mitigation Systems for Light Water Reactors," NUREG/CR-4025, August 1985.

3 GE Nuclear Energy, "Evaluation of Proposed Modifications to the GESSARII Design," NEDE 30640, Class III, San Jose, CA, June 1984.

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

5 "Issuance of Supplement to the Final Environmental Statement - Comanche Peak Steam Electric Station, Units 1 and 2," NUREG-0775 Supplement, December 15, 1989.

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

7 "Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2," NUREG-0974 Supplement, August 16, 1989.

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9 "Technical Guidance for Siting Criteria Development," NUREG/CR-2239, Sandia National Laboratories, December 1982.

10 Title 10, Code of Federal Regulations, Part 50 and 52.

11 50FR32138, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," August 1985.

12 "License Renewal Application – Peach Bottom Atomic Power Station, Units 2 and 3," July 2001.

13 GE Nuclear Energy, "Technical Support Document for the ABWR" 25A5680, Revision 1, January 1995.

14 "Regulatory Analysis Technical Evaluation Handbook," NUREG/BR-0184, 1977.

15 Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document" Issued December 1995.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
1	Cap downstream piping of normally closed component cooling water drain and vent valves.	SAMA would reduce the frequency of a loss of component cooling event, a large portion of which was derived from catastrophic failure of one of the many single isolation valves.	#1 - N/A	PWR RCP seal leakage issue.
2	Enhance loss of component cooling procedure to facilitate stopping reactor coolant pumps.	SAMA would reduce the potential for reactor coolant pump (RCP) seal damage due to pump bearing failure.	#3 – Not a Design Alternative.	PWR RCP seal leakage issue.
3	Enhance loss of component cooling procedure to present desirability of cooling down reactor coolant system (RCS) prior to seal LOCA.	SAMA would reduce the potential for RCP seal failure.	#3 – Not a Design Alternative.	PWR RCP seal leakage issue.
4	Provide additional training on the loss of component cooling.	SAMA would potentially improve the success rate of operator actions after a loss of component cooling (to restore RCP seal damage.)	#3 – Not a Design Alternative.	PWR RCP seal leakage issue.
5	Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals.	SAMA would reduce effect of loss of component cooling by providing a means to maintain the centrifugal charging pump seal injection after a loss of component cooling.	#1 - N/A	PWR RCP seal leakage issue.
5A	Procedure changes to allow cross connection of motor cooling for RHRSW pumps.	SAMA would allow continued operation of both RHRSW pumps on a failure of one train of PSW.	#1 - N/A	RHR Service Water Booster Pumps are not used.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
6	Proceduralize shedding component cooling water loads to extend component cooling heatup on loss of essential raw cooling water.	SAMA would increase time before the loss of component cooling (and reactor coolant pump seal failure) in the loss of essential raw cooling water sequences.	#3 – Not a Design Alternative.	PWR RCP seal leakage issue.
7	Increase charging pump lube oil capacity.	SAMA would lengthen the time before centrifugal charging pump failure due to lube oil.	#1 - N/A	This SAMA would improve the charging pump mission time, which affects RCP seal injection. There is no equivalent function for the ESBWR.
8	Eliminate the RCP thermal barrier dependence on component cooling such that loss of component cooling does not result directly in core damage.	SAMA would prevent the loss of recirculation pump seal integrity after a loss of component cooling.	#1 – N/A	PWR RCP seal leakage issue.
9	Add redundant DC control power for PSW pumps C & D.	SAMA would increase reliability of PSW and decrease core damage frequency due to a loss of SW.	#2 - Already in the design	PSW design incorporates reliability principles.
10	Create an independent RCP seal injection system, with a dedicated diesel.	SAMA would add redundancy to RCP seal cooling alternatives, reducing CDF from loss of component cooling or service water or from a station blackout event.	#1 - N/A	PWR RCP seal leakage issue.
11	Use existing hydro-test pump for RCP seal injection.	SAMA would provide an independent seal injection source, without the cost of a new system.	#1 - N/A	PWR RCP seal leakage issue.
12	Replace ECCS pump motor with air-cooled motors.	SAMA would eliminate ECCS dependency on component cooling system (but not on room cooling).	#1 - N/A	The ESBWR emergency cooling systems (e.g., GDCS, ADS, ICS, PCCS) do not rely on motor-driven pumps.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
13	Install improved RCS pumps seals.	SAMA would reduce probability of RCP seal LOCA by installing RCP seal O-ring constructed of improved materials	#1 - N/A	PWR RCP seal leakage issue.
14	Install additional component cooling water pump.	SAMA would reduce probability of loss of component cooling leading to RCP seal LOCA.	#1 - N/A	PWR RCP seal leakage issue.
15	Prevent centrifugal charging pump flow diversion from the relief valves.	SAMA modification would reduce the frequency of the loss of RCP seal cooling if relief valve opening causes a flow diversion large enough to prevent RCP seal injection.	#1 - N/A	PWR RCP seal leakage issue.
16	Change procedures to isolate RCP seal letdown flow on loss of component cooling, and guidance on loss of injection during seal LOCA.	SAMA would reduce CDF from loss of seal cooling.	#3 – Not a Design Alternative.	PWR RCP seal leakage issue.
17	Implement procedures to stagger high-pressure safety injection (HPSI) pump use after a loss of service water.	SAMA would allow HPSI to be extended after a loss of service water.	#3 – Not a Design Alternative.	The ESBWR emergency cooling systems do not rely on motor-driven pumps.
18	Use fire protection system pumps as a backup seal injection and high-pressure makeup.	SAMA would reduce the frequency of the RCP seal LOCA and the SBO CDF.	#2 - Already in design	The ESBWR Fire protection pumps are capable of supplying injection and makeup via dedicated lines.
19	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	SAMA would reduce the frequency of the loss of component cooling water and service water.	#3 – Not a design alternative	The ESBWR design for PSW and RCCW maintains the capability to cross-tie pumps/headers.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
20	Procedure enhancements and operator training in support system failure sequences, with emphasis on anticipating problems and coping.	SAMA would potentially improve the success rate of operator actions subsequent to support system failures.	#3 – Not a design alternative	General procedure guidance.
21	Improved ability to cool the residual heat removal heat exchangers.	SAMA would reduce the probability of loss of decay heat removal by implementing procedure and hardware modifications to allow manual alignment of the fire protection system or CCW cross-tie.	#2 – Already in the design.	In addition to the RWCU heat exchangers, the ESBWR design has hardware in place to allow manual alignment of fire protection water for ICC/PCC pool makeup and alternate injection.
22	Provide reliable power to control building fans.	SAMA would increase the availability of control room ventilation on loss of power.	#2 – Already in the design.	Control Room emergency HVAC is not dependent on AC power.
23	Provide a redundant train of ventilation.	SAMA would increase the availability of components dependent on room cooling.	#2 – Already in the design.	ESBWR Reactor Building ventilation uses redundant trains.
24	Procedures for actions on loss of HVAC.	SAMA would provide for improved credit to be taken for loss of HVAC sequences (improved affected electrical equipment reliability upon a loss of control building HVAC).	#3 – Not a design alternative	General recovery actions.
25	Add a diesel building switchgear room high temperature alarm.	SAMA would improve diagnosis of a loss of switchgear room HVAC.	#2 – Already in the design.	The ESBWR design incorporates room high temperature alarms.
26	Create ability to switch fan power supply to DC in an SBO event.	SAMA would allow continued operation in an SBO event. This SAMA was created for reactor core isolation cooling system room at Fitzpatrick Nuclear Power Plant.	#1 – N/A	BWR issue – RCIC room cooling. The Isolation Condenser does not require room cooling.
27	Delay containment spray actuation after large LOCA.	SAMA would lengthen time of RWST availability.	#1 – N/A	PWR issue - stored water capacity.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
28	Install containment spray pump header automatic throttle valves.	SAMA would extend the time over which water remains in the RWST, when full CS flow is not needed	#1 – N/A	PWR issue - stored water capacity.
29	Install an independent method of suppression pool cooling.	SAMA would decrease the probability of loss of containment heat removal.	#2 – Already in the design.	Passive Containment Cooling is independent method of containment heat removal. Loss of suppression pool cooling is not risk significant due to redundant methods for containment cooling.
30	Develop an enhanced drywell spray system.	SAMA would provide a redundant source of water to the containment to control containment pressure, when used in conjunction with containment heat removal.	#4 – Excessive Cost	BWR issue. Drywell spray is not risk significant due to redundant methods for containment cooling.
31	Provide dedicated existing drywell spray system.	SAMA would provide a source of water to the containment to control containment pressure, when used in conjunction with containment heat removal. This would use an existing spray loop instead of developing a new spray system.	#4 – Excessive Cost	BWR issue. Drywell spray is not risk significant due to redundant methods for containment cooling.
32	Install an unfiltered hardened containment vent.	SAMA would provide an alternate decay heat removal method for non-ATWS events, with the released fission products not being scrubbed.	#4 – Excessive Cost	The ESBWR drywell vent is scrubbed by the Suppression Pool.
33	Install a filtered containment vent to remove decay heat.	SAMA would provide an alternate decay heat removal method for non-ATWS events, with the released fission products being scrubbed. Option 1: Gravel Bed Filter Option 2: Multiple Venturi Scrubber	#4 – Excessive Cost	The ESBWR drywell vent is scrubbed by the Suppression Pool.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
34	Install a containment vent large enough to remove ATWS decay heat.	Assuming that injection is available, this SAMA would provide alternate decay heat removal in an ATWS event.	#4 – Excessive Cost	ESBWR ATWS sequences are not risk significant.
35	Create/enhance hydrogen recombiners with independent power supply.	SAMA would reduce hydrogen detonation at lower cost, Using a new independent power supply	#2 – Already in the design.	Hydrogen igniters using a control platform independent from Q-DCIS and recombiners are already in ESBWR design.
35A	Install hydrogen recombiners.	SAMA would provide a means to reduce the chance of hydrogen detonation.	#2 – Already in the design.	Hydrogen control system already in ESBWR design.
36	Create a passive design hydrogen ignition system.	SAMA would reduce hydrogen denotation system without requiring electric power.	#2 – Already in the design.	Passive Auto-catalytic recombiners are already in ESBWR design for PCCS heat exchangers and containment atmosphere.
37	Create a large concrete crucible with heat removal potential under the basemat to contain molten core debris.	SAMA would ensure that molten core debris escaping from the vessel would be contained within the crucible. The water cooling mechanism would cool the molten core, preventing a melt-through of the basemat.	#2 – Already in the design.	BiMAC device.
38	Create a water-cooled rubble bed on the pedestal.	SAMA would contain molten core debris dropping on to the pedestal and would allow the debris to be cooled.	#2 – Already in the design.	BiMAC device.
39	Provide modification for flooding the drywell head.	SAMA would help mitigate accidents that result in the leakage through the drywell head seal.	#2 – Already in the design.	BWR issue. ESBWR drywell head is under pool of water.
40	Enhance fire protection system and/or standby gas treatment system hardware and procedures.	SAMA would improve fission product scrubbing in severe accidents.	#1 – N/A	BWR issue – secondary containment. A radiation release into the ESBWR Reactor Building is not risk significant. Scrubbing would not be risk significant.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
41	Create a reactor cavity flooding system.	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	#2 – Already in the design.	BiMAC device and GDCS Deluge.
42	Create other options for reactor cavity flooding.	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	#2 – Already in the design.	BiMAC device and GDCS Deluge.
43	Enhance air return fans (ice condenser plants).	SAMA would provide an independent power supply for the air return fans, reducing containment failure in SBO sequences.	#1 - N/A	
44	Create a core melt source reduction system.	SAMA would provide cooling and containment of molten core debris. Refractory material would be placed underneath the reactor vessel such that a molten core falling on the material would melt and combine with the material. Subsequent spreading and heat removal form the vitrified compound would be facilitated, and concrete attack would not occur.	#2 – Already in the design.	BiMAC device.
45	Provide a containment inerting capability.	SAMA would prevent combustion of hydrogen and carbon monoxide gases.	#2 – Already in the design.	Containment is inerted during normal operation.
46	Use the fire protection system as a backup source for the containment spray system.	SAMA would provide redundant containment spray function without the cost of installing a new system.	#2 – Already in the design.	The ESBWR FPS is capable of supplying drywell spray.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
47	Install a secondary containment filter vent.	SAMA would filter fission products released from primary containment.	#4 – Excessive Cost	BWR issue – secondary containment. A radiation release into the ESBWR Reactor Building is not risk significant. Scrubbing would not be risk significant.
48	Install a passive containment spray system.	SAMA would provide redundant containment spray method.	#4 – Excessive Cost	Containment spray is not modeled as a mitigation function for the ESBWR.
49	Strengthen primary/secondary containment.	SAMA would reduce the probability of containment overpressurization to failure.	#2 – Already in the design.	The ESBWR containment is designed with a higher design margin to maximum pressure and ultimate strength.
50	Increase the depth of the concrete basemat or use an alternative concrete material to ensure melt- through does not occur.	SAMA would prevent basemat melt-through.	#2 – Already in the design.	BiMAC device.
51	Provide a reactor vessel exterior cooling system.	SAMA would provide the potential to cool a molten core before it causes vessel failure, if the lower head could be submerged in water.	#1 – N/A	This option is not compatible with the ESBWR design. Exterior cooling puts water on the lower drywell floor, which, in some scenarios, would increase the risk of ex-vessel steam explosions.
52	Construct a building to be connected to primary/secondary containment that is maintained at a vacuum.	SAMA would provide a method to depressurize containment and reduce fission product release.	#4 – Excessive Cost	
53	Not used.	N/A	N/A	N/A

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
54	Proceduralize alignment of spare diesel to shutdown board after loss of offsite power and failure of the diesel normally supplying it.	SAMA would reduce the SBO frequency.	#3 – Not a Design Alternative.	The importance of alternate AC power is significantly less for the ESBWR.
55	Not used.	N/A	N/A	N/A
56	Provide an additional diesel generator.	SAMA would increase the reliability and availability of onsite emergency AC power sources.	#4 – Excessive Cost	The importance of alternate AC power is significantly less for the ESBWR.
57	Provide additional DC battery capacity.	SAMA would ensure longer battery capability during an SBO, reducing the frequency of long-term SBO sequences.	#2 – Already in the design.	The ESBWR design for DC power uses improved redundancy and capacity.
58	Use fuel cells instead of lead-acid batteries.	SAMA would extend DC power availability in an SBO.	#5 – Very Low Benefit	The ESBWR design for DC power uses improved redundancy.
59	Procedure to cross-tie high-pressure core spray diesel.	SAMA would improve core injection availability by providing a more reliable power supply for the high-pressure core spray pumps.	#1 - N/A	BWR-5/6 issue.
60	Improve 4.16-kV bus cross-tie ability.	SAMA would improve AC power reliability.	#2 – Already in the design.	AC power distribution design uses cross-tie capability between 6.9kV buses.
61	Incorporate an alternate battery charging capability.	SAMA would improve DC power reliability by either cross-tying the AC busses, or installing a portable diesel-driven battery charger.	#5 - Very Low Benefit	The ESBWR design for DC power uses improved redundancy and capacity.
62	Increase/improve DC bus load shedding.	SAMA would extend battery life in an SBO event.	#1 – N/A	The ESBWR design for DC power does not require DC load shedding.
63	Replace existing batteries with more reliable ones.	SAMA would improve DC power reliability and thus increase available SBO recovery time.	#2 – Already in the design.	More reliable batteries to be installed.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
63A	Mod for DC Bus A reliability.	SAMA would increase the reliability of AC power and injection capability. Loss of DC Bus A causes a loss of main condenser, prevents transfer from the main transformer to offsite power, and defeats one half of the low vessel pressure permissive for LPCI/CS injection valves.	#2 – Already in the design.	ESBWR design has 4 divisions of safety-related DC buses. No loss of a single DC bus leads to loss of condenser. Transfer from main transformer to offsite power also not affected.
64	Create AC power cross-tie capability with other unit.	SAMA would improve AC power reliability.	#1 – N/A	The importance of alternate AC power is significantly less for the ESBWR.
65	Create a cross-tie for diesel fuel oil.	SAMA would increase diesel fuel oil supply and thus diesel generator, reliability.	#4 – Excessive Cost	The importance of diesel generators is significantly less for the ESBWR.
66	Develop procedures to repair or replace failed 4-kV breakers.	SAMA would offer a recovery path from a failure of the breakers that perform transfer of 4.16-kV non-emergency busses from unit station service transformers, leading to loss of emergency AC power.	#3 – Not a design alternative	The importance of alternate AC power is significantly less for the ESBWR.
67	Emphasize steps in recovery of offsite power after an SBO.	SAMA would reduce human error probability during offsite power recovery.	#3 – Not a design alternative.	Restoring power from offsite sources after SBO to be proceduralized by COL Holder.
68	Develop a severe weather conditions procedure.	For plants that do not already have one, this SAMA would reduce the CDF for external weather-related events.	#3 – Not a design alternative	Guidelines for preparation for severe weather to be provided by COL Holder.
69	Develop procedures for replenishing diesel fuel oil.	SAMA would allow for long-term diesel operation.	#3 – Not a design alternative	Guidelines for replenishing diesel fuel oil to be provided by COL Holder.
70	Install gas turbine generator.	SAMA would improve onsite AC power reliability by providing a redundant and diverse emergency power system.	#4 – Excessive Cost	The importance of alternate AC power is significantly less for the ESBWR.
71	Not used.	N/A	N/A	N/A

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
72	Create a backup source for diesel cooling. (Not from existing system)	This SAMA would provide a redundant and diverse source of cooling for the diesel generators, which would contribute to enhanced diesel reliability.	#4 – Excessive Cost	The importance of alternate AC power is significantly less for the ESBWR.
73	Use fire protection system as a backup source for diesel cooling.	This SAMA would provide a redundant and diverse method of cooling for the diesel generators, which would contribute to enhanced diesel reliability.	#3 – Not a design alternative	The importance of alternate AC power is significantly less for the ESBWR.
74	Provide a connection to an alternate source of offsite power.	SAMA would reduce the probability of a loss of offsite power event.	#4 – Excessive Cost	The importance of alternate AC power is significantly less for the ESBWR.
75	Bury offsite power lines.	SAMA could improve offsite power reliability, particularly during severe weather.	#4 – Excessive Cost	The importance of alternate AC power is significantly less for the ESBWR.
76	Replace anchor bolts on diesel generator oil cooler.	Millstone Nuclear Power Station found a high seismic SBO risk due to failure of the diesel oil cooler anchor bolts. For plants with a similar problem, this would reduce seismic risk. Note that these were Fairbanks Morse DGs.	#1 – N/A	Plant-specific issue.
77	Change undervoltage (UV), auxiliary feedwater actuation signal (AFAS) block and high pressurizer pressure actuation signals to 3-out-of-4, instead of 2- out-of-4 logic.	SAMA would reduce risk of 2/4 inverter failure.	#2 – Already in the design.	The ESBWR design uses improved redundancy in actuation logic.
78	Provide DC power to the 120/240-V vital AC system from the Class 1E station service battery system instead of its own battery.	SAMA would increase the reliability of the 120-VAC Bus.	#2 – Already in the design.	The importance of alternate AC power is significantly less for the ESBWR. The 125 VAC safety-related power is supplied by inverted DC power.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
79	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture (SGTR).	SAMA would enhance depressurization during a SGTR.	#1 - N/A	PWR issue. Isolation Condenser tube ruptures are not analogous to SGTR because they are isolable.
80	Improve SGTR coping abilities.	SAMA would improve instrumentation to detect SGTR, or additional system to scrub fission product releases.	#1 - N/A	PWR issue.
81	Add other SGTR coping abilities.	SAMA would decrease the consequences of an SGTR.	#1 - N/A	PWR issue.
82	Increase secondary side pressure capacity such that an SGTR would not cause the relief valves to lift.	SAMA would eliminate direct release pathway for SGTR sequences.	#1 - N/A	PWR issue.
83	Replace steam generators (SG) with a new design.	SAMA would lower the frequency of an SGTR.	#1 - N/A	PWR issue.
84	Revise emergency operating procedures to direct that a faulted SG be isolated.	SAMA would reduce the consequences of an SGTR.	#1 - N/A	PWR issue.
85	Direct SG flooding after a SGTR, prior to core damage.	SAMA would provide for improved scrubbing of SGTR releases.	#1 - N/A	PWR issue
86	Implement a maintenance practice that inspects 100% of the tubes in a SG.	SAMA would reduce the potential for an SGTR.	#1 - N/A	PWR issue.
87	Locate residual heat removal (RHR) inside of containment.	SAMA would prevent intersystem LOCA (ISLOCA) out the RHR pathway.	#5 Very Low Benefit	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.
88	Not used.	N/A	N/A	N/A

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
89	Install additional instrumentation for ISLOCAs.	SAMA would decrease ISLOCA frequency by installing pressure of leak monitoring instruments in between the first two pressure isolation valves on low-pressure inject lines, RHR suction lines, and HPSI lines.	#5 – Very Low Benefit	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.
90	Increase frequency for valve leak testing.	SAMA could reduce ISLOCA frequency.	#3 – Not a Design Alternative.	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.
91	Improve operator training on ISLOCA coping.	SAMA would decrease ISLOCA effects.	#3 – Not a Design Alternative.	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.
92	Install relief valves in the CC System.	SAMA would relieve pressure buildup from an RCP thermal barrier tube rupture, preventing an ISLOCA.	#1 - N/A	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.
93	Provide leak testing of valves in ISLOCA paths.	SAMA would help reduce ISLOCA frequency. At Kewaunee Nuclear Power Plant, four MOVs isolating RHR from the RCS were not leak tested.	#3 – Not a Design Alternative.	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system. Normal valve in-service inspections are adequate.
94	Revise EOPs to improve ISLOCA identification.	SAMA would ensure LOCA outside containment could be identified as such. Salem Nuclear Power Plant had a scenario where an RHR ISLOCA could direct initial leakage back to the pressurizer relief tank, giving indication that the LOCA was inside containment.	#3 – Not a Design Alternative.	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
95	Ensure all ISLOCA releases are scrubbed.	SAMA would scrub all ISLOCA releases. One example is to plug drains in the break area so that the break point would cover with water.	#3 – Not a Design Alternative.	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.
96	Add redundant and diverse limit switches to each containment isolation valve.	SAMA could reduce the frequency of containment isolation failure and ISLOCAs through enhanced isolation valve position indication.	#4 – Excessive Cost	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.
97	Modify swing direction of doors separating turbine building basement from areas containing safeguards equipment.	SAMA would prevent flood propagation, for a plant where internal flooding from turbine building to safeguards areas is a concern.	#2 – Already in the design.	Flood propagation is considered in the ESBWR layout. Flooding from Turbine Building does not affect adjacent buildings.
98	Improve inspection of rubber expansion joints on main condenser.	SAMA would reduce the frequency of internal flooding, for a plant where internal flooding due to a failure of circulating water system expansion joints is a concern.	#3 – Not a Design Alternative.	Inspection frequency for expansion joints is directed by normal Maintenance controls.
99	Implement internal flood prevention and mitigation enhancements.	This SAMA would reduce the consequences of internal flooding.	#2 – Already in the design.	Internal flood prevention and mitigation features are incorporated into the ESBWR layout.
100	Implement internal flooding improvements such as those implemented at Fort Calhoun.	This SAMA would reduce risk by preventing or mitigating rupture in the RCP seal cooler of the component cooling system.	#1 - N/A	PWR issue.
101	Install a digital feedwater upgrade.	This SAMA would reduce the chance of a loss of main feedwater following a plant trip.	#2 – Already in the design.	The ESBWR design will incorporate a digital feedwater control system
102	Perform surveillances on manual valves used for backup AFW pump suction.	This SAMA would improve success probability for providing alternative water supply to the AFW pumps.	#1 - N/A	PWR issue.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
103	Install manual isolation valves around AFW turbine-driven steam admission valves.	This SAMA would reduce the dual turbine-driven AFW pump maintenance unavailability.	#1 - N/A	PWR issue.
104	Install accumulators for turbine-driven AFW pump flow control valves (CVs).	This SAMA would provide control air accumulators for the turbine-driven AFW flow CVs, the motor-driven AFW pressure CVs and SG power-operated relief valves (PORVs). This would eliminate the need for local manual action to align nitrogen bottles for control air during a LOOP.	#1 - N/A	PWR issue.
105	Proceduralize intermittent operation of HPCI.	SAMA would allow for extended duration of HPCI availability.	#1 – N/A	BWR issue, not ESBWR issue.
106	Increase the reliability of safety relief valves by adding signals to open them automatically.	SAMA reduces the probability of a certain type of medium break LOCA. Hatch evaluated medium LOCA initiated by an MSIV closure transient with a failure of SRVs to open. Reducing the likelihood of the failure for SRVs to open, subsequently reduces the occurrence of this medium LOCA.	#2 – Already in the design.	CDF contribution from LOCA is insignificant. SRVs and DPVs have capability to be opened automatically.
107	Install motor-driven feedwater pump.	SAMA would increase the availability of injection subsequent to MSIV closure.	#2 – Already in the design.	The ESBWR feedwater pumps are motor-driven.
108	Enhance procedure to instruct operators to trip unneeded RHR/CS pumps on loss of room ventilation.	SAMA increases availability of required RHR/CS pumps. Reduction in room heat load allows continued operation of required RHR/CS pumps, when room cooling is lost.	#1 – N/A	The ESBWR emergency cooling systems (e.g., GDCS, ADS, ICS, PCCS) do not rely on motor-driven pumps.
109	Increase available net positive suction head (NPSH) for injection pumps.	SAMA increases the probability that these pumps will be available to inject coolant into the vessel by increasing the available NPSH for the injection pumps.	#2 – Already in the design.	The ESBWR emergency cooling systems (e.g., GDCS, ADS, ICS, PCCS) do not rely on motor-driven pumps. The CRD pumps have adequate NPSH in all cases.

Table 1 Generic SAMA Design Alternatives

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
110	Increase the safety relief valve (SRV) reseat reliability.	SAMA addresses the risk associated with dilution of boron caused by the failure of the SRVs to reseat after standby liquid control (SLC) injection.	#5 – Very Low Benefit	CDF contribution from ATWS insignificant. High pressure ATWS sequences are less likely due to ICS.
111	Reduce DC dependency between high-pressure injection system and ADS.	SAMA would ensure containment depressurization and high-pressure injection upon a DC failure.	#2 – Already in the design.	Loss of one DC bus cannot disable ADS or CRD injection.
112	Modify Reactor Water Cleanup (RWCU) for use as a decay heat removal system and proceduralize use.	SAMA would provide an additional source of decay heat removal.	#2 – Already in the design.	
113	Use control rod drive (CRD) for alternate boron injection.	SAMA provides an additional system to address ATWS with SLC failure or unavailability.	#4 – Excessive Cost	Routing SLCS piping to CRD suction would be costly and it is uncertain whether the CRD suction would be compatible with high pressure SLCS.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
1.a. Severe Accident EPGs/AMGs	Scope is to develop plant-specific actions that are beyond the generic guidelines	#3 – Not a design alternative.	
1.b. Computer Aided Instrumentation	SAMA will improve prevention of core melt sequences by making operator actions more reliable.	#2 – Already in the ESBWR design.	ESBWR Instrumentation incorporates human factors engineering into the design.
1.c/d. Improved Maintenance Procedures/Manuals	SAMA will improve prevention of core melt sequences by increasing reliability of important equipment	#2 – Already in the ESBWR design.	
1.e. Improved Accident Management Instrumentation	SAMA will improve prevention of core melt sequences by making operator actions more reliable.	#2 – Already in the ESBWR design.	ESBWR Instrumentation incorporates human factors engineering into the design.
1.f. Remote Shutdown Station	This SAMA would allow alternate system control in the event that the control room becomes uninhabitable. This SAMA would reduce the potential for sabotage.	#2 – Already in the ESBWR design.	
1.g. Security System	SAMA would reduce the potential for sabotage	#1 - N/A	Security issues are addressed separately.
1.h. Simulator Training for Severe Accident	SAMA would lead to improved arrest of core melt progress and prevention of containment failure	#3 – Not a design alternative.	
2.a. Passive High Pressure System	SAMA will improve prevention of core melt sequences by providing additional high pressure capability to remove decay heat through an isolation condenser type system	#2 – Already in the ESBWR design.	ESBWR uses Isolation Condensers.
2.b. Improved Depressurization	SAMA will improve depressurization system to allow more reliable access to low pressure systems.	#2 – Already in the ESBWR design.	ESBWR uses S/RVs and DPVs.
2.c. Suppression Pool Jockey Pump	SAMA will improve prevention of core melt sequences by providing a small makeup pump to provide low pressure decay heat removal from the RPV using the suppression pool as a source of water.	#4 – Excessive Cost	PCCS is the heat sink for decay heat removal, not the Suppression Pool.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
2.d. Improved High Pressure Systems	SAMA will improve prevention of core melt sequences by improving reliability of high pressure capability to remove decay heat.	#2 – Already in the ESBWR design.	ESBWR uses Isolation Condensers and CRD pumps.
2.e. Additional Active High Pressure System	SAMA will improve reliability of high pressure decay heat removal by adding an additional system.	#2 – Already in the ESBWR design.	ESBWR uses Isolation Condensers and RWCU/SDC .
2.f. Improved Low Pressure System (Fire pump)	SAMA would provide fire protection system pump(s) for use in low pressure scenarios.	#2 – Already in the ESBWR design.	Fire Pump can provide low pressure injection via FAPCS.
2.g Dedicated Suppression Pool Cooling	SAMA would decrease the probability of loss of containment heat removal.	#2 – Already in the ESBWR design.	PCCS provides containment heat removal.
2.h. Safety Related Condensate Storage Tank	SAMA will improve availability of CST following a Seismic event	#5 – Very Low Benefit.	Seismic fragilities have been evaluated for the ESBWR SSCs.
2.i.1 6 hour Station Blackout Injection	SAMA includes improved capability to cope with longer station blackout scenarios.	#2 – Already in the ESBWR design.	ESBWR is designed to essentially a 72-hour coping period.
3.a. Larger Volume Containment	SAMA increases time before containment failure and increases time for recovery	#4 – Excessive Cost	Redundant containment heat removal features in the ESBWR increase the design margin.
3.b. Increased Containment Pressure Capability (sufficient pressure to withstand severe accidents)	SAMA minimizes likelihood of large releases	#2 – Already in the ESBWR design.	
3.c. Improved Vacuum Breakers (redundant valves in each line)	SAMA reduces the probability of a stuck open vacuum breaker.	#2 – Already in the ESBWR design.	ESBWR vacuum breakers are designed with in-line isolation valves.
3.d. Increased Temperature Margin for Seals	This SAMA would reduce the potential for containment failure under adverse conditions.	#5 – Very Low Benefit.	Reducing the probability of failure at drywell or hatch seals would have a minimal risk effect because containment failure would occur at a higher pressure in a different location.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
3.e. Improved Leak Detection	The intent of this SAMA is to increase piping surveillance in order to identify leaks prior to the onset of complete failure. Improved leak detection would potentially reduce the LOCA frequency.	#3 – Not a Design Alternative.	The contribution to CDF from LOCAs is not significant.
3.f. Suppression Pool Scrubbing	This SAMA would reduce the consequences of venting the containment by directing the vent path through the water contained in the suppression pool.	#2 – Already in the ESBWR design.	Drywell vent path is through the wetwell.
3.g. Improved Bottom Penetration Design	SAMA reduces failure likelihood of RPV bottom head penetrations by changing the Bottom Head drain line transition piece to a material with a higher melting point than carbon steel.	#2 – Already in the ESBWR design.	ESBWR RWCU/SDC Bottom Head drain line at the penetration is stainless steel.
4.a. Larger Volume Suppression Pool (double effective liquid volume)	SAMA would increase the size of the suppression pool so that heatup rate is collapsed, allowing more time for recovery of a heat removal system	#2 – Already in the ESBWR design.	ESBWR containment has larger capacity to remove decay heat.
4.b. CUW Decay Heat Removal	This SAMA provides a means for Alternate Decay Heat Removal.	#2 – Already in the ESBWR design.	The ABWR CUW system is analogous to ESBWR RWCU/SDC system.
4.c. High Flow Suppression Pool Cooling	SAMA would improve suppression pool cooling.	#4 – Excessive Cost	Redundant containment heat removal functions are available.
4.d. Passive Overpressure Relief	This SAMA will prevent catastrophic failure of the containment. Controlled relief through a selected vent path has a greater potential for reducing the release of radioactive material than through a random break.	#4 – Excessive Cost	ESBWR CSET release frequencies for containment overpressurization are insignificant.
5.a/d. Unfiltered Vent	SAMA would provide an alternate decay heat removal method with the released fission products not being scrubbed.	#4 – Excessive Cost	Redundant containment heat removal functions are available.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
5.b/c. Filtered Vent	SAMA would provide an alternate decay heat removal method with the released fission products being scrubbed.	#2 – Already in the ESBWR design.	Vent path from the suppression pool.
6.a. Post Accident Inerting System	SAMA would reduce likelihood of gas combustion inside containment	#2 - Already in the ESBWR design.	ESBWR containment is inerted. Gas combustion in inerted areas is not risk significant.
6.b. Hydrogen Control by Venting	This SAMA will prevent catastrophic failure of the containment due to hydrogen detonation by venting the hydrogen gas prior to reaching detonable concentration.	#3 – Not a design alternative.	This item is an operator action. ESBWR has the capability to vent the containment, but relies on passive hydrogen recombiners and automatically actuated hydrogen igniters to control hydrogen levels.
6.c. Pre-inerting	SAMA would reduce likelihood of gas combustion inside containment	#2 – Already in the ESBWR design.	ESBWR containment failure due to hydrogen detonation in the drywell and wetwell air spaces is not risk significant.
6.d. Ignition Systems	This SAMA will prevent catastrophic failure of the containment due to hydrogen detonation by burning the hydrogen gas prior to reaching detonable concentration.	#2 – Already in the ESBWR design.	ESBWR design includes hydrogen igniters.
6.e. Fire Suppression System Inerting	This SAMA will prevent catastrophic failure of the containment due to hydrogen detonation by inerting the containment with the fire suppression system.	#1 – N/A	
7.a Drywell Head Flooding	SAMA would provide intentional flooding of the upper drywell head such that if high drywell temperatures occurred, the drywell head seal would not fail.	#2 – Already in the ESBWR design.	ESBWR drywell head is underneath a pool of water.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
7.b. Containment Spray Augmentation	SAMA would provide a redundant source of water to the containment to control containment pressure when used in conjunction with containment heat removal.	#2 – Already in the ESBWR design.	Multiple sources of water from FAPCS can supply containment spray.
8.a. Additional Service Water Pump	SAMA might conceivably reduce common cause dependencies from SW system and thus reduce plant risk through system reliability improvement.	#4 – Excessive Cost	Loss of Service Water is not a significant initiating event.
8.b. Improved Operating Response	This SAMA would improve likelihood of success of operator actions taken in response to an abnormal condition.	#3 – Not a design alternative.	
8.c. Diverse Injection System	SAMA will improve prevention of core melt sequences by providing additional injection capabilities.	#2 – Already in the ESBWR design.	ESBWR injection functions are GDCS, ICS, CRD, Feedwater/ Condensate, and other diverse systems.
8.d. Operating Experience Feedback	This SAMA would provide information on the effectiveness of maintenance practices and equipment reliability.	#3 – Not a design alternative.	
8.e. Improved MSIV Design	This SAMA would decrease the likelihood of containment bypass scenarios.	#5 – Very Low Benefit	Improvements in MSIV isolation would be marginal due to redundancy.
8.f. Improved SRV Design	This SAMA would improve SRV reliability, thus increasing the likelihood that sequences could be mitigated using low pressure heat removal.	#2 – Already in the ESBWR design.	DPVs provide additional relief capability for reactor depressurization.
9.a. Steam Driven Turbine Generator	This SAMA would provide a steam driven turbine generator which uses reactor steam and exhausts to the suppression pool. If large enough, it could provide power to additional equipment.	#1 – N/A	Passive ESBWR features have significantly less reliance on AC power.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
9.b. Alternate Pump Power Source	This SAMA would provide a small dedicated power source such as a dedicated diesel or gas turbine for the feedwater or condensate pumps, so that they do not rely on offsite power.	#4 – Excessive Cost	Restoration of condensate for low pressure injection does not provide a significant benefit.
9.d. Additional Diesel Generator	SAMA would reduce the SBO frequency.	#4 – Excessive Cost	Passive ESBWR features have significantly less reliance on AC power.
9.e. Increased Electrical Divisions	SAMA would provide increased reliability of AC power system to reduce core damage and release frequencies.	#2 – Already in the ESBWR design.	ESBWR electrical design incorporates 4 divisions of electrical power.
9.f. Improved Uninterruptible Power Supplies	SAMA would provide increased reliability of power supplies supporting front-line equipment, thus reducing core damage and release frequencies.	#2 – Already in the ESBWR design.	The importance of alternate AC power is significantly less for the ESBWR. The 125 VAC safety-related power is supplied by inverted DC power.
9.g. AC Bus Cross-Ties	SAMA would provide increased reliability of AC power system to reduce core damage and release frequencies.	#2 – Already in the ESBWR design.	ESBWR electrical design has AC bus cross-tie capability.
9.h. Gas Turbine	SAMA would improve onsite AC power reliability by providing a redundant and diverse emergency power system.	#4 – Excessive Cost	Passive ESBWR features have significantly less reliance on AC power.
9.i. Dedicated RHR (bunkered) Power Supply	This SAMA would improve the reliability of the RHR system by enhancing the AC power supply system.	#4 – Excessive Cost	Passive ESBWR features have significantly less reliance on AC power.
10.a. Dedicated DC Power Supply	This SAMA addresses the use of a diverse DC power system such as an additional battery or fuel cell for the purpose of providing motive power to certain components (e.g., RCIC).	#2 – Already in the ESBWR design.	ESBWR passive design reduces the dependence on motive power.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
10.b. Additional Batteries/Divisions	This SAMA addresses the use of a diverse DC power system such as an additional battery or fuel cell for the purpose of providing motive power to certain components (e.g., RCIC).	#2 – Already in the ESBWR design.	ESBWR passive design reduces the dependence on motive power.
10.c. Fuel Cells	SAMA would extend DC power availability in an SBO.	#5 – Very Low Benefit	ESBWR safety-related batteries are sized to accommodate SBO events.
10.d. DC Cross-ties	This SAMA would improve DC power reliability.	#5 – Very Low Benefit Already in the ESBWR design.	ESBWR DC design has 4 divisions. Cross-ties would provide no significant benefit.
10.e. Extended Station Blackout Provisions	SAMA would provide reduction in SBO sequence frequencies.	#5 – Very Low Benefit	The importance of SBO is significantly less for the ESBWR.
11.a. ATWS Sized Vent	This SAMA would provide the ability to remove reactor heat from ATWS events.	#4 – Excessive Cost	ATWS sequences are not a significant risk contributor.
11.b. Improved ATWS Capability	This SAMA includes items which reduce the contribution of ATWS to core damage and release frequencies.	#5 – Very Low Benefit	ATWS sequences are not a significant risk contributor.
12.a. Increased Seismic Margins	This SAMA would reduce the risk of core damage and release during seismic events.	#2 – Already in the ESBWR design.	Seismic fragilities already evaluated and incorporated into the ESBWR design.
12.b. Integral Basemat	This SAMA would improve containment survivability under severe seismic activity.	#2 – Already in the ESBWR design.	Seismic fragilities already evaluated and incorporated into the ESBWR design.
13.a. Reactor Building Sprays	This SAMA provides the capability to use firewater sprays in the reactor building to mitigate release of fission products into the Rx Bldg following an accident.	#3 – Not a design alternative.	Crediting existing fire sprays is not a design change.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
13.b. System Simplification	**This SAMA is intended to address system simplification by the elimination of unnecessary interlocks, automatic initiation of manual actions or redundancy as a means to reduce overall plant risk.**	**#2 – Already in the ESBWR design.**	**Implemented in Human Factors Engineering.**
13.c. Reduction in Reactor Building Flooding	**This SAMA reduces the Reactor Building Flood Scenarios contribution to core damage and release.**	**#2 – Already in the ESBWR design.**	**Internal flood prevention and mitigation features are incorporated into the ESBWR layout.**
14.a. Flooded Rubble Bed	**SAMA would contain molten core debris dropping on to the pedestal and would allow the debris to be cooled.**	**#2 – Already in the ESBWR design.**	**BiMAC device.**
14.b. Reactor Cavity Flooder	**SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.**	**#2 – Already in the ESBWR design.**	**GDCS/BiMAC.**
14.c. Basaltic Cements	**SAMA minimizes carbon dioxide production during core concrete interaction.**	**#2 – Already in the ESBWR design.**	**Basaltic cement will be used in the containment basemat. Also, BiMAC device siginificantly reduces the probability of CCI.**

Table 3
ESBWR Design Features for Severe Accident Mitigation

ESBWR Design Feature	**Severe Accident Mitigation Benefit**
Isolation Condenser with Improved Design	Provides passive, high pressure decay heat removal. Significantly reduces the need for S/RVs to lift on pressure setpoint.
Depressurization Valves	Improves reliability for reactor vessel depressurization. Provides reliable logic to prevent inadvertent depressurization.
Diesel powered pumps and Ancillary DG Bus powered pumps for Makeup and Injection	IC/PCC pool makeup. Spent Fuel Pool makeup. Alternate source for low pressure coolant injection. Alternate source for suppression pool cooling.
Passive Containment Cooling System	Provides reliable, passive containment heat removal.
BiMAC Device and GDCS Deluge Function	Enhances core debris coolability after vessel melt-through. Provides water deluge for core melt source reduction. Reduces probability of core-concrete interaction. Reduces probability of basemat melt-through.

Table 3
ESBWR Design Features for Severe Accident Mitigation

ESBWR Design Feature	**Severe Accident Mitigation Benefit**
DC Power Reliability	Additional battery capacity to reduce common cause failures. Safety-related DC power has 4 divisions Significantly reduces the effects of a loss of a single DC bus on transients and mitigating systems.
Actuation Logic Reliability	Reduces probability of actuation failures and inadvertent actuation failures.
Motor-driven Feedwater Pumps	Available for injection after MSIV closure.
Water Pool Above Drywell Head	Provides cooling to drywell head to prevent or reduce head seal leakage. Mitigates containment releases through the drywell head.
Containment Ultimate Strength and Maximum Design Pressure	Reduces probability and consequences of containment overpressure failure.
Flood Mitigation Incorporated into Design	Flood propagation between buildings is minimized.
RWCU Heat Exchangers Sized to Remove Decay Heat	Provides high pressure decay heat removal.
Station Blackout 72-hour Coping Period	No reliance on AC power for first 72 hours of an event.
Upgraded Low Pressure Piping for Reactor Coolant Pressure Boundary	Provides significant reduction in probability of ISLOCA.

Table 3
ESBWR Design Features for Severe Accident Mitigation

ESBWR Design Feature	Severe Accident Mitigation Benefit
Digital Controls and Instrumentation	Significant improvement in operator actions for: • Monitoring • Diagnosis • Mitigation • Remote shutdown capability • Accident management

Table 4 Core Damage Frequencies (NEDO-33201 Rev.5)	
Category	**CDF /year**
At-Power Internal Events	**1.68 E-8**
At-Power Fire	**1.25 E-8**
At-Power Flood	**3.3 E-9**
At-Power High Winds	**8.5 E-9**
Shutdown Internal Events	**1.7 E-8**
Shutdown Fire	**9.6 E-9**
Shutdown Flood	**5.2 E-9**
Shutdown High Winds	**3.95 E-8**
TOTAL	**1.12 E-7**

Table 5 Release Frequencies and Offsite Consequences					
Release Category	**Release Frequency**	**Population Dose (Sv) at 50 Miles[1]**	**Weighted Pop. Dose (Sv/yr)**	**Offsite Consequence Cost ($)**	**Weighted Offsite Cost ($/yr)**
BOC	**1.1 E-10**	**5.90E+05**	**6.3 E-5**	**2.58E+10**	**2.74**
BYP	**7.38 E-8**	**4.73E+05**	**3.49 E-2**	**2.56E+10**	**1888.14**
CCID	**9.7 E-11**	**2.80E+05**	**2.72 E-5**	**2.87E+10**	**2.79**
CCIW	**8 E-12**	**4.13E+04**	**3.38 E-7**	**6.44E+09**	**.05**
DCH	**<1 E-12**	**3.09E+05**	**0**	**2.03E+10**	**0**
EVE	**1.14 E-9**	**3.56E+05**	**4.06 E-4**	**3.00E+10**	**34.20**
FR	**6.2 E-10**	**1.30E+03**	**8 E-7**	**1.95E+07**	**.01**
OPVB	**5 E-12**	**1.05E+05**	**5.62 E-7**	**8.90E+09**	**.05**
OPW1	**5 E-12**	**2.06E+05**	**1.13 E-6**	**1.94E+10**	**.11**
OPW2	**2.6 E-10**	**7.30E+04**	**1.87 E-5**	**9.93E+09**	**2.54**
TSL	**3.65 E-8**	**6.81E+02**	**2.49 E-5**	**2.17E+07**	**.79**
Total	**1.12 E-7**	**--**	**3.54 E-2**	**--**	**1931.42**

1 Based on input from EPRI Utility Requirements Document (Reference 15)