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Subject: **ESBWR Severe Accident Management Design Alternatives**

Please find enclosed GE Licensing Topical Report NEDO-33306, "ESBWR Severe Accident Management Design Alternatives." A Severe Accident Management Design Alternatives (SAMDA) evaluation is not part of a Design Certification Application. SAMDA information is required per 10 CFR 52.79(a)(2) for the Environmental Report submitted with a Combined Operating License Application (COLA). NEDO-33306 provides an evaluation of SAM Design alternatives that are applicable to the certified ESBWR design. GE requests NRC review of this LTR because it is intended that it be referenced in the Environmental Report for COLAs referencing the certified ESBWR design. These COLAs would then be supplemented by an evaluation of SAMDA that are site specific.

This LTR provides the more rigorous SAMDA assessment of design alternatives or enhancements requested in NRC Request for Additional Information No 19.4.0-1 transmitted in the Reference 1 Letter. Submittal of this Licensing Topical Report constitutes GE's response to NRC Request for Additional Information No 19.4.0-1. The SAMDA information for the ESBWR will reside in NEDO-33306 not in the ESBWR DCD.

If you have any questions or require additional information regarding the information provided here, please contact me.

D068

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

Reference:

1. MFN 05-156, Letter from NRC to David Hinds, *Request for Additional Information Letter No. 3 for the ESBWR Design Certification Application*, dated December 8, 2005

Enclosures:

1. Licensing Topical Report, NEDO-33306, ESBWR Severe Accident Management Design Alternatives

cc: AE Cubbage USNRC (with enclosures)
David Hinds GE/Wilmington (with enclosures)
eDRF 0000-0062-7426

Enclosure 1

MFN 07-062

Licensing Topical Report

NEDO-33306

ESBWR Severe Accident Management Design Alternatives

GE Nuclear Energy

3901 Castle Hayne Rd
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Licensing Topical Report
ESBWR Severe Accident Management Design Alternatives

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Approved:

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INFORMATION NOTICE

This document, NEDO-33306, Rev 0, contains no proprietary information.

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

PLEASE READ CAREFULLY

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 NEDC-33201P. 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 Management 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.

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 (NEDC-33201P).

3 EVALUATIONS OF RADIOLOGICAL RISK FROM NUCLEAR POWER PLANTS

3.1 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.

3.2 Cost/Benefit Standard for Evaluation of SAMDAs

The NRC updated its recommended approach for the monetary conversion of radiation exposures. Previous guidance specified that 1 person-rem of exposure should be valued at \$1000. This conversion factor for offsite doses was intended to account for both health effects and offsite property damage, and exposures incurred in future years were not to be discounted. The guidance given in the NRC's regulatory analysis guidelines (NUREG/BR-0058, Revision 2), recommends using \$2000 per person-rem of exposure as the monetary conversion factor. In addition, future exposures are to be discounted to arrive at their present worth to assess values and impacts. Offsite property damage from nuclear accidents is to be valued separately, and is not part of the \$2000 per person-rem value. A criterion of \$3000 per person-rem averted was added to account for offsite property damage and other related costs for severe accidents.

3.3 Socio-Economic Risks for Severe Accidents

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, a general discussion of socio-economic risks for the ESBWR design, based in large measure on the discussion of such risks in NUREG-1437, "Draft Generic Environmental Impact Statement for License Renewal of Nuclear Plants" is provided in the remainder of this subsection.

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 off-site costs. The off-site costs considered in NUREG-1437 (see Vol. 1 at 5-90) 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 there from (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 off-site costs were estimated to be as high as $6 billion to $8 billion for severe accidents with a probability of once in one million operating years. Higher costs were estimated for severe accidents with much lower probabilities. The projected costs of adverse health effects from deaths and illnesses were estimated to average about 10-20% of off-site 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, were costs associated with the termination of economic activities in a contaminated area. This could create adverse economic impacts in wider regional markets and sources of supplies outside the contaminated area. The predicted conditional land contamination was estimated to be small (10 acres/year at most), see NUREG-1437, pp. 5-90 through 5-93.

NUREG-1437 provides the bases for the Commission's proposed amendments to 10 CFR Part 51 concerning the environmental impacts of license renewal. The proposed amendments find 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, see 56 Fed. Reg. 47,016, 47,019, 47,034-35 (September 17, 1991).

The socio-economic risks contained in NUREG-1437 are bounding for plants of ESBWR design. First, the core damage frequency for plants of ESBWR design is less than 10-7 per year. Thus, no accidents, and hence no off-site costs, are expected at probabilities at or greater than once in one million years. Second, plants of ESBWR design meet the safety goals set forth by the NRC.

3.4 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 alternatives to reduce the radiological risks from normal operation of a plant of ESBWR design is not warranted in order to satisfy NEPA. Moreover, the radiological impacts from normal operation of an ESBWR are environmentally insignificant.

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

NEDC-33201P 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 NEDC-33201P:

- 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 NEDC-33201P evaluations are as follows:

- Core Damage Frequency: The ESBWR core damage frequency was determined to be less than 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-8 per reactor year.

4.2 Dominant Severe Accidents Sequences for Plants of ESBWR Design

In performing the PRA for the ESBWR design, GE identified and evaluated many severe accident sequences. For each sequence, the analysis identified an initiating event and traced the accident's progression to its end. For sequences involving core damage, offsite consequences were estimated.

Only the sequences with frequencies greater than 1E-9 per reactor year were considered. The complete radiological consequence analysis of the dominant sequences can be found in NEDC-33201P.

Sequences with probabilities of occurrence less than 1E-9 were considered remote and speculative. While the Commission has not yet specified a quantitative point at which it will consider severe accident probabilities as remote and speculative, it has indicated that a decision to consider severe accidents remote and speculative would be based upon the accident probabilities and the accident scenarios being analyzed. See Vermont Yankee Nuclear Power Corporation, (Vermont Yankee Nuclear Power Station), CLI-90-07, 32 NRC 129, 132 (1990). GE believes that the severe accident analysis in NEDC-33201P provides a sufficient basis for the Commission to find that ESBWR sequences with frequencies less than 1E-9 per reactor year can be deemed remote and speculative.

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. 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. 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. 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 SAMDA DEFINITION APPLIED TO PLANTS OF ESBWR DESIGN

This subsection considers whether the ESBWR 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 8c 52 and the Severe Accident Policy Statement. The cost/benefit evaluation of SAMDAs to plants of ESBWR design uses the expanded definition of SAMDAs, design alternatives that could prevent and/or mitigate the consequences of a severe accident.

5.1 Cost/Benefit Standard for Evaluation of ESBWR SAMDAs

As discussed earlier, the cost/benefit ratio of $2,000 per person-rem averted is viewed by the NRC and the nuclear industry as an acceptable standard for the purposes of evaluating SAMDAs. This standard was used as a surrogate for all off-site costs in the cost/benefit evaluation of SAMDAs to plants of ESBWR design. In order to accurately reflect the costs associated with prevention of severe accidents, averted on-site costs were incorporated for SAMDAs that were at least partially preventative in nature. On-site costs resulting from a severe accident include replacement power, on-site cleanup costs, and economic loss of the facility.

The equation used to determine the cost/benefit ratio is:

$$R = \frac{\text{Cost of SAMDA Implementation} - \text{Averted On-Site Costs}}{\text{Reduction in Residual Risk}}$$

A plant life time of 60 years was assumed to maximize the reduction in residual risk.

5.2 Cost Estimates of Potential Modifications to the ESBWR Design

All previous evaluations of design alternatives (e.g., the Limerick and Comanche Peak FES Supplements, NUREG-1437, and the ABWR SSAR) have reported design alternative costs which, at a minimum, are in the hundreds of thousands of dollars. The high cost of design alternatives that have the potential for proven risk reduction is also demonstrated in several state-of-the-art surveys (e.g., NUREG/CR-3908, NUREG/CR-4025 and NUREG/CR-4920). In fact, most proposed design alternatives cost in the millions of dollars to implement.

This analysis uses a representative design alternative implementation cost of $200,000 (which is below the cost of all design alternatives which would be expected to provide a non-negligible reductions in risk) to determine if additional analysis needs to be performed for plants of ESBWR design.

For design alternatives that can prevent core damage, averted on-site costs will also be considered. A conservative estimate of averted on-site costs can be obtained by multiplying the frequency of core damage, the number of years the plant will be licensed to operate, and the sum of plant construction cost and cleanup costs. By assuming a plant life of 60 years, a construction cost of $3B and cleanup costs of $3B and an implementation cost of $200,000, the resulting frequency of core damage would be about 5.6E-7. The frequency of core damage from the

ESBWR PRA is about an order of magnitude less than this value. Therefore the implementation of a design alternative that would have an impact on the core damage frequency would have to cost significantly less than $200,000.

6 ANALYSIS OF SEVERE ACCIDENT MANAGEMENT DESIGN ALTERNATIVES

Tables 1,2, and 3 comprise a list of severe accident management design alternative candidates that have been compiled from the list of SAMA issues from the ABWR SAMA study (Reference13), from a generic list compiled for License Renewal Environmental Reports (Reference 12), and also from the ESBWR PRA insights and assumptions in Table 19.2-3 of DCD Tier 2 Chapter 19. This list is screened to eliminate activities that do not apply to the ESBWR design or have no significant benefit. 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-informed design alternatives have 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. All Criterion 3 items will be considered by the COL Holder for SAMA guidance.
4. No significant safety benefit.
   PRA importance of SAMA is not significant.
5. Candidate for Risk-Benefit Consideration

The list of 230 SAMA candidates has been analyzed to determine if there are cost-beneficial design alternatives that should be considered for the ESBWR. The screening analysis determined that there are there are 50 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. Of the remaining 180 alternatives, 30 are classified as Criterion 3, (i.e., a procedural or administrative issue with potential benefit) and these items will be reviewed by the COL Holder for consideration in their SAMA guidance. These items are listed in Table 4, and will be incorporated into a COL Action Item. There are 97 candidate design alternatives that are similar to, or are already incorporated into the ESBWR design. The remaining 53 candidates do not have a significant safety benefit. These design alternatives typically are improvements for functions that do not have high risk achievement worth or Fussell-Vesely Importance values in the ESBWR PRA. In many cases, the ESBWR design has reduced their risk significance. For example, the severe accident management design alternatives for adding redundant DC power sources are not risk-significant because the ESBWR design includes four divisions of DC power with more capacity and less service to large loads. There are no alternatives classified as Criterion 5 that provide a clear risk significant benefit beyond the current ESBWR design.

7 SUMMARY AND CONCLUSIONS

ESBWR design alternatives that only provide severe accident mitigation must cost significantly less than the $200,000, which is a conservatively estimated cost for a design alternative that has the potential for a measurable reduction in severe accident risk. This low cost limitation is a result of the ESBWR's 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.

8 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.

8 GE Nuclear Energy, "Survey of the State of the Art in Mitigation Systems," NUREG/CR-3908, December 1985.

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.

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	The ESBWR PSW motors are self-cooled.
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.

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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.	#1 - N/A	Loss of Service Water is not a significant initiating event.
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.
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.

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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.	#1 - N/A	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.
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.	#4 – No Significant Safety Benefit	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.	#4 – No Significant Safety Benefit.	Control Room HVAC is not risk significant.
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	SAMA would provide for	#3 – Not a	Loss of HVAC is not a

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
	loss of HVAC.	improved credit to be taken for loss of HVAC sequences (improved affected electrical equipment reliability upon a loss of control building HVAC).	design alternative	significant initiating event.
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.
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 – No Significant Safety Benefit.	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 – No Significant Safety Benefit.	BWR issue. Drywell spray is not risk significant due to redundant methods for containment cooling.

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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 – No Significant Safety Benefit.	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 – No Significant Safety Benefit.	The ESBWR drywell vent is scrubbed by the Suppression Pool.
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 – No Significant Safety Benefit.	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	#1 – N/A	BWR issue. ESBWR containment failure due to hydrogen detonation is not risk significant.
35A	Install hydrogen recombiners.	SAMA would provide a means to reduce the chance of hydrogen detonation.	#1 – N/A	BWR issue. ESBWR containment failure due to hydrogen detonation is not risk significant.
36	Create a passive design hydrogen ignition system.	SAMA would reduce hydrogen denotation system without requiring electric power.	#1 – N/A	BWR issue. ESBWR containment failure due to hydrogen detonation is not risk significant.
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.

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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.
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.
42	Create other options for reactor cavity flooding.	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	#1 – N/A	BiMAC device.
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.

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 – No Significant Safety Benefit.	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 without high cost.	#4 – No Significant Safety Benefit.	Containment spray is not a highly risk significant 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.	#1 – N/A	Containment overpressure failures are not risk significant.
53	Not used.	N/A	N/A	N/A
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 – No Significant Safety Benefit.	The importance of alternate AC power is significantly less for the ESBWR.

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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.	#4 – No Significant Safety 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.	#4 – No Significant Safety 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.
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 – No Significant Safety Benefit.	The importance of diesel generators is significantly less for the ESBWR.

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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 – No Significant Safety Benefit.	The importance of alternate AC power is significantly less for the ESBWR.
71	Not used.	N/A	N/A	N/A
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 – No Significant Safety Benefit.	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 source 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 – No Significant Safety Benefit.	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 – No Significant Safety Benefit.	The importance of alternate AC power is significantly less for the ESBWR.

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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.
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. N/A to BWR. 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. N/A to BWR
81	Add other SGTR coping abilities.	SAMA would decrease the consequences of an SGTR.	#1 - N/A	PWR issue. N/A to BWR
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. N/A to BWR
83	Replace steam generators (SG) with a new design.	SAMA would lower the frequency of an SGTR.	#1 - N/A	PWR issue. N/A to BWR
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. N/A to BWR
85	Direct SG flooding after a	SAMA would provide for	#1 - N/A	PWR issue. N/A to BWR

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
	SGTR, prior to core damage.	improved scrubbing of SGTR releases.		
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. N/A to BWR
87	Locate residual heat removal (RHR) inside of containment.	SAMA would prevent intersystem LOCA (ISLOCA) out the RHR pathway.	#4 – No Significant Safety 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
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.	#4 – No Significant Safety 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.	#4 – No Significant Safety Benefit.	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.	#4 – No Significant Safety Benefit	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.	#4 – No Significant Safety Benefit.	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.

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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.	#4 – No Significant Safety Benefit.	ISLOCA is not risk significant for the ESBWR because of the design requirements for SSCs connected to the primary system.
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.	#4 – No Significant Safety Benefit.	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 – No Significant Safety Benefit.	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.	COL Holder to consider inspection frequency for expansion joints within the Reliability Assurance Program.
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. N/A to BWR
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

SAMA ID No.	SAMA title	Result of Potential Enhancement	Screening Criteria	Disposition
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. N/A to BWR
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. N/A to BWR
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. N/A to BWR
105	Proceduralize intermittent operation of HPCI.	SAMA would allow for extended duration of HPCI availability.	#1 – N/A	BWR 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.	#4 – No Significant Safety Benefit.	Total CDF contribution from LOCA is less than 1%, which is insignificant.
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.	#4 – No Significant Safety Benefit.	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.

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.	#4 – No Significant Safety Benefit.	Total CDF contribution from ATWS is less than 1%, which is 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.	#3 – Not a design alternative	COL Holder to consider for potential applicability.

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 incorporated into 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 incorporated into 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 incorporated into 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 incorporated into 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 incorporated into 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 incorporated into 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 – No Significant Safety Benefit	PCCS supplies condensate to Suppression Pool for additional inventory. Also, the Suppression Pool is less significant as a source of water in the ESBWR due to lack of

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title		Result of Potential Enhancement	Screening Criteria	Disposition
				ECCS pumps.
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 incorporated into 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 incorporated into the ESBWR design.	ESBWR uses Isolation Condensers and CRD pumps.
2.f.	Improved Low Pressure System (Fire pump)	SAMA would provide fire protection system pump(s) for use in low pressure scenarios.	#2 – Already incorporated into 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 incorporated into the ESBWR design.	PCCS provides containment heat removal in addition to suppression pool cooling.
2.h.	Safety Related Condensate Storage Tank	SAMA will improve availability of CST following a Seismic event	#4 – No Significant Safety Benefit	Seismic fragilities have been evaluated for the ESBWR SSCs.
2.i.	16 hour Station Blackout Injection	SAMA includes improved capability to cope with longer station blackout scenarios.	#2 – Already incorporated into 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 – No Significant Safety Benefit	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	#4 – No Significant Safety Benefit	
3.c.	Improved Vacuum Breakers (redundant valves in each line)	SAMA reduces the probability of a stuck open vacuum breaker.	#2 – Already incorporated into the ESBWR design.	ESBWR vacuum breakers are designed with in-line isolation valves.

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
3.d. Increased Temperature Margin for Seals	This SAMA would reduce the potential for containment failure under adverse conditions.	#4 – No Significant Safety 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.
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.	#4 – No Significant Safety Benefit	The contribution to CDF from LOCAs is less than 1%.
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 incorporated into the ESBWR design.	Drywell vent path through suppression pool.
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 incorporated into 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 incorporated into 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 incorporated into 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 – No Significant Safety Benefit	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	#4 – No Significant Safety Benefit	ESBWR CSET release frequencies for containment overpressurization are

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
	greater potential for reducing the release of radioactive material than through a random break.		insignificant.
5.a/d. Unfiltered Vent	SAMA would provide an alternate decay heat removal method with the released fission products not being scrubbed.	#4 – No Significant Safety Benefit	Redundant containment heat removal functions are available.
5.b/c. Filtered Vent	SAMA would provide an alternate decay heat removal method with the released fission products being scrubbed.	#2 – Already incorporated into 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 incorporated into the ESBWR design.	ESBWR design incorporates an inerted containment.
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.	ESBWR containment failure due to hydrogen detonation is not risk significant.
6.c. Pre-inerting	SAMA would reduce likelihood of gas combustion inside containment	#4 – No Significant Safety Benefit	ESBWR containment failure due to hydrogen detonation 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.	#4 – No Significant Safety Benefit	ESBWR containment failure due to hydrogen detonation is not risk significant.
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	#2 – Already incorporated into 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
	head seal would not fail.		
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 incorporated into 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 – No Significant Safety Benefit	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.	#4 – No Significant Safety Benefit	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.	#4 – No Significant Safety 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 incorporated into 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.
9.b. Alternate Pump	This SAMA would provide a small	#4 – No	Restoration of condensate

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
Power Source	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.	Significant Safety Benefit	for low pressure injection does not provide a significant benefit.
9.d. Additional Diesel Generator	SAMA would reduce the SBO frequency.	#4 – No Significant Safety Benefit	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 incorporated into 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 incorporated into 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 incorporated into 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 – No Significant Safety Benefit	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 – No Significant Safety Benefit	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 incorporated into the ESBWR design.	ESBWR passive design reduces the dependence on motive power.
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 incorporated into 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.c. Fuel Cells	SAMA would extend DC power availability in an SBO.	#4 – No Significant Safety Benefit	ESBWR safety-related batteries are sized to accommodate SBO events.
10.d. DC Cross-ties	This SAMA would improve DC power reliability.	#2 – Already incorporated into the ESBWR design.	ESBWR DC design has 4 divisions.
10.e. Extended Station Blackout Provisions	SAMA would provide reduction in SBO sequence frequencies.	#4 – No Significant Safety 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 – No Significant Safety Benefit	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.	#4 – No Significant Safety 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 incorporated into 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 incorporated into the ESBWR design.	BiMAC device
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.	
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 incorporated into the ESBWR design.	
13.c. Reduction in Reactor Building Flooding	This SAMA reduces the Reactor Building Flood Scenarios contribution to core damage and	#2 – Already incorporated into the ESBWR	Internal flood prevention and mitigation features are incorporated into the

Table 2 ABWR SAMA Design Alternatives

ABWR SAMA Title	Result of Potential Enhancement	Screening Criteria	Disposition
	release.	design.	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 incorporated into 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 incorporated into the ESBWR design.	GDCS/BiMAC.
14.c. Basaltic Cements	SAMA minimizes carbon dioxide production during core concrete interaction.	#4 – No Significant Safety Benefit	BiMAC device siginificantly reduces the probability of CCI.

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
1a	Loss of Preferred Power is an important initiating event.	Reliability of offsite power sources cannot be completely controlled by the plant. However, to assure that plant equipment does not contribute to power losses, inspection of equipment that connects the plant to the switchyard should be performed periodically in accordance with the RAP.	#2 – Already incorporated into the ESBWR design.	The ESBWR design for AC Transmission and distribution incorporates best practices for design reliability.
1b	Total Loss of Feedwater is an important initiating event.	The design of the feedwater system precludes any single failure from causing this initiator. The digital feedwater control system is a key ingredient to keeping this initiator low. It should be monitored in accordance with the RAP.	#2 – Already incorporated into the ESBWR design.	The ESBWR design for feedwater controls incorporates best practices for design reliability.
1c	Failure to Recognize Need for Low Pressure Makeup is an important post-initiator operator action.	This insight is covered in operator training and operating procedures.	#3 – Not a design alternative.	
1d	Failure to Recover Offsite Power is an important recovery factor.	This insight is covered in operator training and operating procedures.	#3 – Not a design alternative.	
1e	The GDCS pools and the suppression pool are instrumented and alarmed such that inadvertent draining of a pool would be immediately obvious to the crew and pool level would be restored.	This is an important indicator for prompt operator recovery action. Should be covered in operator training and operating procedures.	#3 – Not a design alternative.	
1f	Important common cause failures are CCF of squib valves in GDCS lines.	The GDCS squib valve pyrotechnic charges shall be replaced during refueling in accordance with the RAP.	#2 – Already incorporated into the ESBWR design.	The ESBWR design for squib valves incorporates best practices for design reliability.
1g	An alarm located within the control room alerts the operator if the battery connection switch is inadvertently left open after	This insight is covered in operator training and operating procedures.	#3 – Not a design alternative.	

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
	test or maintenance.			
1h	Condensate pump discharge header valve F018 and Condensate pump discharge header bypass valve F016 are assumed to be air operated valves that fail to remain open on a loss of air supply.	Design requirement to support condensate system operation.	#2 – Already incorporated into the ESBWR design.	
1i	The opening of SRVs alone is sufficient for reactor depressurization to allow low pressure injection.	It is a design requirement for this function to be able to be accomplished, including margin.	#2 – Already incorporated into the ESBWR design.	
1j	RCCW heat exchanger discharge valves open, heat exchanger bypass valves close, and cross connection valves fail as is, given a loss of pneumatic supply.	Design requirement to support PRA success criteria for RCCW.	#2 – Already incorporated into the ESBWR design.	
2a	Use of AC independent components of the fire water system to provide long term cooling.	The fire water system is capable of providing long term cooling water to the ICS, PCCS, and spent fuel pools that is independent of the safety-related systems and the onsite AC power system. Operator alignment of these systems should be covered in training and emergency response procedures. Flow and flow monitoring instrumentation from the fire protection system to these pools should be monitored and tested according to the RAP. All piping that provides this function of the FPS should be monitored and tested in the RAP.	#2 – Already incorporated into the ESBWR design.	Design requirement. COL Holder to develop operating procedures and training.
2b	Logic to Prevent Spurious Actuation of GDCS Deluge Subsystem.	In order to ensure a dry cavity at the time of vessel failure, it is important to prevent premature or spurious actuation of the passive deluge valves. Reliability of the Deluge Squib valves and actuation logic are in the RAP.	#2 – Already incorporated into the ESBWR design.	

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
2c	The nitrogen supply and battery capacity are sufficient to allow vessel depressurization after potential ICS failures.	Insights that are covered in operator training and operating procedures.	#2 – Already incorporated into the ESBWR design	
2d	An inerted containment prevents hydrogen-oxygen concentrations from reaching combustible levels.	Insight from Level 2 PRA.	#2 – Already incorporated into the ESBWR design.	
2e	Containment isolation prevents or mitigates releases.	Tested in accordance with Tech Spec Surveillance Requirements	#2 – Already incorporated into the ESBWR design.	
2f	Upgraded low pressure piping for reactor coolant pressure boundary prevents interfacing systems LOCAs.	Design Requirement	#2 – Already incorporated into the ESBWR design.	
2g	Drywell-Wetwell Vacuum Breakers ensure containment integrity and sufficient pressure differential to drive PCCS.	The failures of these vacuum breakers to close can be kept to an acceptably low probability if they are incorporated into the RAP.	#2 – Already incorporated into the ESBWR design.	
2h	The failure rate of the GDCS deluge system is not to exceed 1E-3 per demand. It is assumed that the system will be sufficiently independent from any core damage prevention systems to maintain this level of reliability.	Design requirement to support the mitigation effectiveness of GDCS/BiMAC.	#2 – Already incorporated into the ESBWR design.	
2i	Basemat Internal Melt Arrest and Coolability Device (BiMAC) mitigates vessel melt-through scenarios.	Inspection of the BiMAC device should be in the RAP.	#2 – Already incorporated into the ESBWR design.	
2j	For ATWS mitigation, Automatic initiation of ADS is inhibited after there is a coincident low reactor water level signal and an average power range monitor (APRM) ATWS permissive signal (i.e., APRM signal	Logic functional testing to be in accordance with technical specifications.	#2 – Already incorporated into the ESBWR design.	

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
	above a specified setpoint.) The same inhibit condition applies to GDCS function.			
2k	Check valves in the GDCS injection lines prevent backflow from the reactor vessel into the GDCS pools during the time when injection squib valves have actuated on low reactor vessel level and reactor vessel is depressurizing, but pressure is higher than drywell pressure.	The testable GDCS check valves shall be tested periodically to ensure the disk readiness to function, both to open, if required, and to close in case of spurious opening of the squib valves. During refueling, an inspection of the strainers of the GDCS equalizing lines connected to the suppression pool shall be performed to prevent potential undetected obstructions.	#2 – Already incorporated into the ESBWR design.	
2l	The flow rate for each RCCW heat exchanger train is regulated by the bypass valves and the exchanger discharge valves. Both valves are pneumatic.	Risk significant These valves should be tested in accordance with the RAP.	#2 – Already incorporated into the ESBWR design.	
2m	Service Water Pumps supply cooling water to RCCW and TCCW. Cooling Tower Fans provide heat removal for service water.	Risk significant These pumps and fans should be tested in accordance with the RAP.	#2 – Already incorporated into the ESBWR design.	
2n	Loss of incoming transmission lines results in loss of preferred power scenario. If the main generator trips, the low voltage generator breaker opens and power to the unit auxiliary transformers is backfed from the normal preferred power (utility power grid).	Risk significant Reliability of offsite power sources cannot be completely controlled by the plant. However, to assure that plant equipment does not contribute to power losses, inspection of switchyard equipment should be performed periodically in accordance with the RAP.	#2 – Already incorporated into the ESBWR design.	
2p	The safety-related DC distribution system is arranged in four divisional class 1E 250V DC power supplies.	Risk significant Station emergency batteries receive periodic checks in accordance with plant Technical Specifications. These checks are adequate to ensure that the batteries will have the reliability	#2 – Already incorporated into the ESBWR design.	

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
		assumed in safety analyses and that the possibility of common cause failures is minimized.		
2q	Diesel Generators provide backup source of AC power for loss of preferred power events.	Risk significant Maintenance for the emergency diesel generators is expected to be performed in accordance with site procedures and the manufacturer's recommendations. Surveillance testing is required in accordance with manufacturer recommendations and best industry practices.	#2 – Already incorporated into the ESBWR design.	
2r	Redundant motor operated valves interconnecting reactor well pool with IC/PCC pools to extend water inventory from 24 to 72 hours have been identified as important components.	These valves should be tested in accordance with the RAP.	#2 – Already incorporated into the ESBWR design.	
2s	A high DW temperature could be caused by accidents such as LOCAs, inadvertent opening of one DPV, or loss of the drywell cooling system. The instrumentation is assumed to be designed for the maximum temperature attainable in the DW.	Design requirement that level sensors are designed for the maximum temperature attainable in the drywell.	#2 – Already incorporated into the ESBWR design.	
2t	An FPS pump is assumed to have the same head and flow capacity as an FAPCS pump.	This supports the assumption that FPS can provide makeup injection to the vessel in some low pressure sequences.	#2 – Already incorporated into the ESBWR design.	
2u	One FAPCS system train can also accomplish the long-term heat removal.	FAPCS suppression pool cooling provides long-term decay heat removal in certain sequences.	#2 – Already incorporated into the ESBWR design.	
2v	The FAPCS and FPS injection capability provides adequate core cooling for transients given successful DPV or ADS valve operation, even if	This assures that injection from FAPCS and FPS is available in sequences with high containment pressure.	#2 – Already incorporated into the ESBWR design.	

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
	containment pressure is at the ultimate containment pressure.			
2w	Room cooling is not required for first 24 hours of the accident.	Design requirement to support PRA success criteria for RCCW.	#2 – Already incorporated into the ESBWR design.	
2x	All loads needed for the PRA are specified/required to be cooled with at least n+1 redundancy.	General design requirement to support PRA success criteria for cooling functions.	#2 – Already incorporated into the ESBWR design.	
2y	Specific guidance for the use of the suppression pool vent has not been developed, thus, venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength.	Design requirement to support successful containment venting.	#2 – Already incorporated into the ESBWR design.	COL Holder to develop procedures for controlled venting.
2z	Vent operation is modeled using the operator action for venting containment. It is assumed that the vent can be operated (manually) independently of any Level 1 mitigation systems.	Design requirement to prevent venting failure due to common causes.	#2 – Already incorporated into the ESBWR design.	
3a	Fire in Turbine Building is an important fire initiating event.	Fire barriers, including penetrations, are tested in accordance with fire protection requirements.	#2 – Already incorporated into the ESBWR design.	
3b	Fire in Reactor Building	Fire barriers in the Reactor Building are important for keeping the fire risk low.	#2 – Already incorporated into the ESBWR design.	
3c	Reactor Building flood design	Penetrations to the Control Building are located at elevations above the worst-case flood level to prevent flooding into the Control Building.	#2 – Already incorporated into the ESBWR design.	
3d	Control Building flood design	Fire main pipes are located external to the building to minimize flood potential.	#2 – Already incorporated into the ESBWR design.	

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
3e	If automatic actuations fail, the operators may manually perform the necessary recovery actuations from the remote shutdown panels.	The remote shutdown panel should be tested periodically to show that it can perform its functions that will lead to safe shutdown. Operators should be trained and instructed in the use of controls in the remote shutdown panels. Instruction should be prepared to decide in which condition the control room must be evacuated.	#2 – Already incorporated into the ESBWR design.	
3f	A Circulating Water System pipe break in the turbine building is an important flood initiating event.	Periodically, room barriers should be inspected to ensure that they will prevent the spread of flooding; room drain lines should be checked to ensure no blockage exists; Circulating Water isolation valves should be stroke tested (normally accomplished by switching from an operating pump to a standby pump in a given loop); the ability of Circulating Water pump circuit breakers to trip upon receipt of a trip signal should be demonstrated; and level sensors in the turbine building must be periodically tested to show their functionality.	#2 – Already incorporated into the ESBWR design.	
3g	Risk from tornado strikes on the plant is acceptably low.	Site response procedures address actions to take for high winds. No additional controls are warranted.	#4 – No Significant Safety Benefit.	
3h	The HCLPF analysis identifies seismic-induced loss of DC power and seismic-induced ATWS due to seismic-induced failure of both the fuel channels and the SLC tank as important scenarios.	No maintenance activities other than those already associated with the in-service surveillance of the seismic instruments are needed for seismic events.	#4 – No Significant Safety Benefit.	

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
4a	Important initiating events in the internal events shutdown PRA are: • Instrument Line Break Below TAF – Mode 6. • RWCU/SDC Drain Line Break Below TAF – Mode 6, Flooded. • Instrument Line Break Below TAF – Mode 6, Unflooded. • LOPP – Mode 6. • CRD line break in the Reactor Bldg. – Mode 6.	Piping integrity is assured by the in-service inspection and testing programs. Given the importance of LOPP to shutdown PRA, inspection and testing of the AC-independent fire protection system in vessel injection mode should be included in RAP. Also, due to the importance of manual alignment, lining up the firewater should be included in the training programs.	#2 – Already incorporated into the ESBWR design.	
4b	An important operator action in the shutdown internal flooding PRA is the failure to recognize the need for low-pressure makeup.	Insights that are covered in operator training and operating procedures.	#3 – Not a design alternative.	
4c	It is assumed that both RWCU/SDCS trains are running, because the time periods in which only one is running occurs when the reactor well is flooded. Consequently, failure of one of the trains is not considered an initiating event.	Insights that are covered in operator training and operating procedures.	#3 – Not a design alternative.	

Table 3 ESBWR SAMA Design Alternatives

Item	Insight/Assumption	Comment	Screening Criteria	Disposition
4d	Breaks in lines connected to the vessel below the core elevation are important scenarios during shutdown Mode 5. In this mode, the lower drywell equipment hatch or personnel hatch is likely to be open to facilitate work in the lower drywell.	The loss of reactor coolant inventory control function during mode 5 underscores the importance of keeping the lower drywell personnel and equipment hatches closed as long as possible. The lower drywell hatches (equipment and personnel) must remain open only when personnel are working inside the lower drywell, and not left open otherwise. Whenever the hatches are open, procedures shall require personnel to be available and in close proximity to the hatches, with the purpose of providing fast closure of the containment in the case of a water leak.	#2 – Already incorporated into the ESBWR design.	There is a COL Action Item in DCD Tier 2 Chapter 19 to implement procedural controls for the drywell hatches.
4e	During shutdown, Mode 5, with the reactor cavity not flooded, it is important to ensure that the GDCS squib valves do not inadvertently actuate.	Controls on maintenance on GDCS components during Mode 5 when reactor cavity has not been flooded are managed in accordance with 10 CFR 50.65 (a)(4), i.e., Maintenance Rule.	#3 – Not a design alternative.	
4f	Relative insights from the shutdown Fire PRA assume the proper functioning of fire barriers to prevent propagation of fires to adjacent zones.	Fire barriers are inspected and maintained in accordance with Fire Protection Program procedures.	#2 – Already incorporated into the ESBWR design.	

Table 4 SAMA Items for COL Holder Consideration

SAMA ID No.	SAMA title	Result of Potential Enhancement
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	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.
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.)
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.
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.
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.
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.
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.
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).
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.
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.

Table 4 SAMA Items for COL Holder Consideration

SAMA ID No.	SAMA title	Result of Potential Enhancement
67	Emphasize steps in recovery of offsite power after an SBO.	SAMA would reduce human error probability during offsite power recovery.
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.
69	Develop procedures for replenishing diesel fuel oil.	SAMA would allow for long-term diesel operation.
73	Use fire protection system as a backup source for diesel cooling.	This SAMA would provide a redundant and diverse source of cooling for the diesel generators, which would contribute to enhanced diesel reliability.
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.
113	Use control rod drive (CRD) for alternate boron injection.	SAMA provides an additional system to address ATWS with SLC failure or unavailability.
E1c	Failure to Recognize Need for Low Pressure Makeup is an important post-initiator operator action.	This insight is covered in operator training and operating procedures.
E1d	Failure to Recover Offsite Power is an important recovery factor.	This insight is covered in operator training and operating procedures.
E1e	The GDCS pools and the suppression pool are instrumented and alarmed such that inadvertent draining of a pool would be immediately obvious to the crew and pool level would be restored.	This is an important indicator for prompt operator recovery action. Should be covered in operator training and operating procedures.
E1g	An alarm located within the control room alerts the operator if the battery connection switch is inadvertently left open after test or maintenance.	This insight is covered in operator training and operating procedures.
E4b	An important operator action in the shutdown internal flooding PRA is the failure to recognize the need for low-pressure makeup.	Insights that are covered in operator training and operating procedures.
E4c	It is assumed that both RWCU/SDCS trains are running, because the time periods in which only one is running occurs when the reactor well is flooded. Consequently, failure of one of the trains is not considered an initiating event.	Insights that are covered in operator training and operating procedures.

Table 4 SAMA Items for COL Holder Consideration

SAMA ID No.	SAMA title	Result of Potential Enhancement
E4e	During shutdown, Mode 5, with the reactor cavity not flooded, it is important to ensure that the GDCS squib valves do not inadvertently actuate.	Controls on maintenance on GDCS components during Mode 5 when reactor cavity has not been flooded are managed in accordance with 10 CFR 50.65 (a)(4), i.e., Maintenance Rule.
A1.a	1.a Severe Accident EPGs/AMGs	Develop plant-specific guidance beyond the generic scope.
A1.h	1.h. Simulator Training for Severe Accident	SAMA would lead to improved arrest of core melt progress and prevention of containment failure
A6.b	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.
A8.b	8.b. Improved Operating Response	This SAMA would improve likelihood of success of operator actions taken in response to an abnormal condition.
A8.d	8.d. Operating Experience Feedback	This SAMA would provide information on the effectiveness of maintenance practices and equipment reliability.
A13.a	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.