U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-1437
Supplement 46

Generic Environmental Impact Statement for License Renewal of Nuclear Plants

Supplement 46

Regarding Seabrook Station

Second Draft Report for Comment

Office of Nuclear Reactor Regulation

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Second Draft Report for Comment

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Office of Nuclear Reactor Regulation

Proposed Action	Issuance of renewed operating license NPF-86 for Seabrook Station in the city of Seabrook, Rockingham County, New Hampshire
Type of Statement	Supplement to Draft Supplemental Environmental Impact Statement
Agency Contact	Lois M. James Office of Nuclear Reactor Regulation Mail Stop O-11F1 U.S. Nuclear Regulatory Commission Washington, D.C. 20555-0001 Telephone: 301-415-3306 E-mail: lois.james@nrc.gov
Comments	Any interested party may submit comments on this supplement to the draft supplemental environmental impact statement (DSEIS). Please specify NUREG-1437, Supplement 46, Volume 2, draft supplement to draft, in your comments. Comments must be received by June 30, 2013. Comments received after the expiration of the comment period will be considered if it is practical to do so, but the U.S. Nuclear Regulatory Commission (NRC) cannot assure that consideration of late comments will be given. Comments may be submitted electronically by searching for docket ID NRC-2010-0206 at the Federal rulemaking Web site, http://www.regulations.gov. Comments may also be mailed to the following address: Chief, Rules, Announcements, and Directives Branch Division of Administrative Services Office of Administration Mail Stop: TWB-05-B01M U.S. Nuclear Regulatory Commission Washington, D.C. 20555-0001 Please be aware that any comments that you submit to the NRC will be considered a public record and entered into the Agencywide Documents Access and Management System (ADAMS). Do not provide information you would not want to be publicly available.

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ABSTRACT

This document supplements the draft supplemental environmental impact statement (DSEIS) which had been prepared in response to an application submitted by NextEra Energy Seabrook, LLC (NextEra) to renew the operating license for Seabrook Station (Seabrook) for an additional 20 years. This supplement incorporates new information that the U.S. Nuclear Regulatory Commission (NRC) staff has obtained since the publication of the DSEIS in August 2011.

This supplement to the DSEIS includes the NRC staff evaluation of revised information provided by NextEra pertaining to the severe accident mitigation alternatives (SAMA) analysis for Seabrook.

In addition, the NRC is taking the opportunity to (1) update the Uranium Fuel Cycle section in light of the June 8, 2012, U.S. Court of Appeals for the District of Columbia Circuit (New York v. NRC, 681 F.3d 471 (D.C. Cir. 2012)) decision to vacate the NRC's Waste Confidence Decision Rule (WCD) (75 Federal Register (FR) 81032, 75 FR 81037) and (2) to provide information on its analysis of new NEPA issues and associated environmental impact findings for license renewal.

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Background

By letter dated May 25, 2010, NextEra Energy Seabrook, LLC (NextEra) submitted an application to the U.S. Nuclear Regulatory Commission (NRC) to issue a renewed operating license for Seabrook Station (Seabrook) for an additional 20-year period.

Pursuant to Title 10, Part 51.20(b)(2) of the *U.S. Code of Federal Regulations* (10 CFR 51.20(b)(2)), the renewal of a power reactor operating license requires preparation of an environmental impact statement (EIS) or a supplement to an existing EIS. In addition, 10 CFR 51.95(c) states that the NRC shall prepare an EIS, which is a supplement to the Commission's NUREG-1437, *Generic Environmental Impact Statement (GEIS) for License Renewal of Nuclear Plants*.

The NRC published its draft supplemental environmental impact statement (DSEIS) for Seabrook in August 2011. Subsequent to the issuance of the DSEIS, by letter dated March 19, 2012, NextEra notified the NRC of significant changes that were made to the severe accident mitigation alternatives (SAMA) analysis related to the Seabrook license renewal application (LRA).

To address this new information, the NRC staff has prepared this supplement to the DSEIS in accordance with 10 CFR 51.72(a)(2) and (b), which address preparation of a supplement to a final environmental impact statement for proposed actions that have not been taken, under the following conditions:

- There are significant new circumstances or information relevant to environmental concerns and bearing on the proposed action or its impacts.
- It is the opinion of the NRC staff that preparation of a supplement will further the purposes of the National Environmental Policy Act of 1969 (NEPA).

In addition, the NRC is taking the opportunity to update the Uranium Fuel Cycle section in light of the June 8, 2012, U.S. Court of Appeals for the District of Columbia Circuit (New York v. NRC, 681 F.3d 471 (D.C. Cir. 2012)) decision to vacate the NRC's Waste Confidence Decision Rule (WCD) (75 Federal Register (FR) 81032, 75 FR 81037). In response to the court's ruling, the Commission (NRC 2012a) determined that it would not issue licenses dependent upon the WCD, until the issues identified in the court's decision are appropriately addressed. The Commission also noted that this determination extends only to final license issuance; all current licensing reviews and proceedings should continue to move forward.

Further, the NRC is also taking the opportunity to provide information on its analysis of new NEPA issues and associated environmental impact findings for license renewal. This is the result of NRC having recently completed, through its rulemaking process, an update and re-evaluation of the potential environmental impacts associated with the renewal of an operating license for a nuclear power reactor for an additional 20 years. A revised Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS), which updates the 1996 GEIS (NRC 1996), provides the technical basis for the revised rule, including the list of NEPA issues and findings contained in Table B-1 in Appendix B to Subpart A of the revised 10 CFR Part 51.

Proposed Action

The proposed action remains the same as that stated in the DSEIS (page 1-1):

> [NextEra] initiated the proposed Federal action by submitting an application for license renewal [for Seabrook], for which the existing license, NPF-86, expires on March 15, 2030. The NRC's Federal action is the decision whether to renew the license for an additional 20 years.

Purpose and Need for Action

The purpose and need for action remains the same as stated in the DSEIS (page 1-1):

> The purpose and need for the proposed action (issuance of a renewed license) is to provide an option that allows for baseload power generation capability beyond the term of the current nuclear power plant operating license to meet future system generating needs. Such needs may be determined by other energy-planning decisionmakers, such as State, utility, and, where authorized, Federal agencies (other than NRC). This definition of purpose and need reflects the NRC's recognition that, unless there are findings in the safety review required by the Atomic Energy Act or findings in the National Environmental Policy Act (NEPA) environmental analysis that would lead the NRC to reject a license renewal application, the NRC does not have a role in the energy-planning decisions of whether a particular nuclear power plant should continue to operate.
>
> If the renewed license is issued, the appropriate energy-planning decisionmakers, along with NextEra, will ultimately decide if the plant will continue to operate based on factors such as the need for power. If the operating license is denied, then the facility must be shut down on or before the expiration date of the current operating license, March 15, 2030.

Environmental Impacts of License Renewal

The changes to this section are highlighted in **redline** and ~~strikeout~~.

The SEIS evaluates the potential environmental impacts of the proposed action. The environmental impacts from the proposed action are designated as SMALL, MODERATE, or LARGE. As set forth in the GEIS, Category 1 issues are those that meet all of the following criteria:

- The environmental impacts associated with the issue are determined to apply either to all plants or, for some issues, to plants having a specific type of cooling system or other specified plant or site characteristics.
- A single significance level (i.e., SMALL, MODERATE, or LARGE) has been assigned to the impacts, except for collective offsite radiological impacts from the fuel cycle and from high-level waste and spent fuel disposal.
- Mitigation of adverse impacts associated with the issue is considered in the analysis, and it has been determined that additional plant-specific mitigation measures are likely not to be sufficiently beneficial to warrant implementation.

SMALL: Environmental effects are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource.

MODERATE: Environmental effects are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.

LARGE: Environmental effects are clearly noticeable and are sufficient to destabilize important attributes of the resource.

For Category 1 issues, no additional site-specific analysis is required in this draft SEIS unless new and significant information is identified. Chapter 4 of this report presents the process for identifying new and significant information. Site-specific issues (Category 2) are those that do not meet one or more of the criterion for Category 1 issues; therefore, an additional site-specific review for these non-generic issues is required, and the results are documented in the SEIS.

Recently, the NRC approved a revision to its environmental protection regulation, 10 CFR Part 51, which governs environmental impact reviews of nuclear power plant operating license renewals. The NRC, through its rulemaking process, has completed an update and re-evaluation of the potential environmental impacts associated with the renewal of an operating license for a nuclear power reactor for an additional 20 years. A revised GEIS, which updates the 1996 GEIS, provides the technical basis for the revised rule. The revised GEIS specifically supports the revised list of NEPA issues and associated environmental impact findings for license renewal contained in Table B-1 in Appendix B to Subpart A of the revised 10 CFR Part 51. The revised rule consolidates similar Category 1 and 2 issues, changes some Category 2 issues into Category 1 issues and consolidates some of those issues with existing Category 1 issues. The revised rule also adds new Category 1 and 2 issues.

The revised rule is expected to be published in 2013; it will become effective 30 days after publication in the *Federal Register*. Compliance by license renewal applicants will not be required until one year from the date of publication (i.e., license renewal environmental reports submitted later than one year after publication must be compliant with the new rule). Nevertheless, under NEPA, the NRC must now consider and analyze, in its license renewal SEISs, the potential significant impacts described by the revised rule's new Category 2 issues, and to the extent there is any new and significant information, the potential significant impacts described by the revised rule's new Category 1 issues.

Table ES-1 summarizes the Category 2 issues applicable to Seabrook, as well as the NRC staff's findings related to those issues. If the NRC staff determined that there were no Category 2 issues applicable for a particular resource area, the findings of the GEIS, as documented in Appendix B to Subpart A of 10 CFR Part 51, stand.

Table ES-1. Summary of NRC conclusions relating to site-specific impact of license renewal

Resource Area	Relevant Category 2 Issues	Impacts
Land Use	None	SMALL
Air Quality	None	SMALL
Surface Water Resources	None	SMALL
Groundwater Resources	**Radionuclides released to groundwater** ~~None~~	SMALL
Aquatic Resources	Impingement Entrainment Heat shock	SMALL to LARGE
Terrestrial Resources	~~None~~**Effects on terrestial resources (non-cooling system impact)**	SMALL
Protected Species and Habitats	Threatened or endangered species	SMALL to LARGE
Human Health	Electromagnetic fields—acute effects (electric shock)	SMALL
Socioeconomics	Housing Impacts Public services (public utilities) Offsite land use Public services (public transportation) Historic and archaeological resources	SMALL
Cumulative Impacts	**Aquatic resources** **All other resource areas**	**MODERATE to LARGE** **SMALL**

No further changes were made to the Executive Summary.

ABBREVIATIONS AND ACRONYMS

°F	degree(s) Fahrenheit
AC	alternating current
ACC	averted cleanup and contamination costs
ADAMS	Agencywide Documents Access and Management System
AEA	**AEA Technology PLC**
AOC	averted offsite property damage cost
AOE	averted offsite occupational exposure
AOSC	averted onsite costs
AOV	air-operated valve
ASME	American Society of Mechanical Engineers
APE	averted public exposure
ATWS	anticipated transient without scram
CBS	containment building spray
CCW	component cooling water
CDF	core damage frequency
CEQ	**Council on Environmental Quality**
CET	containment event tree
CFR	*U.S. Code of Federal Regulations*
CIV	containment isolation valve
CLB	current licensing basis
CMR	***Code of Massachusetts Regulations***
COE	cost of enhancement
CR	control rod
CS	core spray
CST	condensate storage tank
DBA	design-basis accident
DC	direct current
DG	diesel generator
DSEIS	draft supplemental environmental impact statement
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFW	emergency feedwater
EGCA	**East Coast Greenway Alliance**
EIS	environmental impact statement
EOP	emergency operating procedure

EPA	**U.S. Environmental Protection Agency**
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ER	Environmental Report
F&O	facts and observations
FIVE	fire-induced vulnerability evaluation
FPLE	**Florida Power and Light Energy Seabrook, LLC**
FR	***Federal Register***
FSEIS	final supplemental environmental impact statement
FWS	**U.S. Fish and Wildlife Service**
g	gram
GEIS	NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*
GI	generic issue
GIS	geographic information system
GL	generic letter
HCLPF	high confidence low probability of failure
HELB	high-energy line break
HEP	human error probability
HFO	high winds, floods, and other external events
HPI	high-pressure injection
HRA	human reliability analysis
HVAC	heating, ventilation, and air conditioning
IAEA	**International Atomic Energy Agency**
IEEE	**Institute of Electrical and Electronics Engineers**
in.	inch
IPE	individual plant examination
IPEEE	individual plant examination of external events
ISLOCA	interfacing system loss-of-coolant accident
K	thousand
km	kilometer
kV	kilovolt
LERF	large early release frequency
LHSI	low-head safety injection
LL-5	large late containment basemat failure

LLNL	Lawrence Livermore National Laboratory
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LOSP	loss of system pressure
LRA	license renewal application
m	meter
M	million
MAAP	Modular Accident Analysis Program
MAB	Maximum Attainbale Benefit
MACCS2	MELCOR Accident Consequences Code System 2
MACR	maximum averted cost risk
MDFW	**Massachusetts Division of Fisheries and Wildlife**
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
MFGD	**Massachusetts Fish and Game Department**
MFW	main feedwater
MG	motor generator
mi	mile
MIT	**Massachusetts Institute of Technology**
MOV	motor-operated valve
mph	miles per hour
mps	meters per second
MSSV	steam safety valve
MW	megawatt
MWe	megawatt electric
MWt	megawatt thermal
NAESC	North Atlantic Energy Service Corp.
NAI	**Normandeau Associates, Inc.**
NEA	**Nuclear Energy Agency**
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act of 1969
NextEra	NextEra Energy Seabrook, LLC
NHDES	**New Hampshire Department of Environmental Services**
NHDHR	**New Hampshire Division of Historical Resources**
NHDRED	**New Hampshire Department of Resources and Economic Development**
NHFGD	**New Hampshire Fish and Game Department**
NHNHB	**New Hampshire Natural Heritage Bureau**
NHY	New Hampshire Yankee
NIEHS	**National Institute of Environmental Health Sciences**

NMFS	**National Marine Fisheries Service**
NPDES	**National Pollutant Discharge Elimination System**
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NUSCO	**Northeast Utilities Service Company**
PAB	primary auxiliary building
PCC	primary component cooling
PCCW	primary component cooling water system
PNNL	Pacific Northwest National Laboratory
PORV	power-operated relief valve
POST	**Parliamentary Office of Science and Technology**
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
psia	pounds per square inch absolute
RAI	request for additional information
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
ROW	**right-of-way**
RPC	replacement power costs
RRW	risk reduction worth
RSCS	**Radiation Safety & Control Services, Inc.**
RSP	remote shutdown panel
RWST	reactor water storage tank
SAMA	severe accident mitigation alternatives
SAMG	severe accident mitigation guideline
SAR	safety analysis report
SBO	station blackout
SE-3	small early containment penetration failure to isolate
Seabrook	Seabrook Station
SEIS	supplemental environmental impact statement
SELL	small early containment penetration failure to isolate and large late containment basemat failure
SEPS	supplemental electrical power system
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SLOCA	small break LOCA

SRM	staff requirements memorandum
SRP	Standard Review Plan
SUFP	startup feed pump
Sv	Sievert
SW	service water
SWGR	switchgear
SWS	service water system
TDAFW	turbine-driven auxiliary feedwater
TDEFW	turbine-driven emergency feedwater
the Court	U.S. Court of Appeals for the District of Columbia Circuit
TIBL	thermal internal boundary layer
TMDL	**total maximum daily load**
UHS	uniform hazard spectrum
USCB	**U.S. Census Bureau**
USGCRP	**U.S. Global Change Research Program**
USGS	U.S. Geological Survey
WCD	**Waste Confidence Decision and Rule**
WOG	Westinghouse Owner's Group

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) staff prepared this supplement to the draft supplemental environmental impact statement (DSEIS) in accordance with Title 10, Parts 51.72(a)(2) and (b) of the *U.S. Code of Federal Regulations* (10 CFR 51.72(a)(2) and (b)), which address preparation of a supplement to an environmental impact statement for proposed actions that have not been taken, under the following conditions:

- There are significant new circumstances or information relevant to environmental concerns and bearing on the proposed action or its impacts.
- It is the opinion of the NRC staff that preparation of a supplement will further the purposes of the National Environmental Policy Act of 1969 (NEPA).

The NRC staff prepared this supplement to the DSEIS because, subsequent to the issuance of the DSEIS, NextEra Energy Seabrook, LLC (NextEra) (2012) notified the NRC of significant changes that were made to the severe accident mitigation alternatives (SAMA) analysis related to the Seabrook Station (Seabrook) license renewal application (LRA). Specifically, NextEra identified many changes to its SAMA analysis, based on various plant and probabilistic risk assessment (PRA) model changes, that were sufficiently different from what was published in the NRC staff's August 2011 DSEIS to warrant the issuance of this supplement.

In addition, the NRC is taking the opportunity to update the Uranium Fuel Cycle section in light of the June 8, 2012, U.S. Court of Appeals for the District of Columbia Circuit (New York v. NRC, 681 F.3d 471 (D.C. Cir. 2012)) decision to vacate the NRC's Waste Confidence Decision Rule (WCD) (75 *Federal Register* (FR) 81032, 75 FR 81037). In response to the court's ruling, the Commission (NRC 2012a) determined that it would not issue licenses dependent upon the WCD, until the issues identified in the court's decision are appropriately addressed. The Commission also noted that this determination extends only to final license issuance; all current licensing reviews and proceedings should continue to move forward.

Further, on December 6, 2012, the Commission affirmed a decision to publish in the *Federal Register* an amendment that would revise its environmental protection regulation, 10 CFR Part 51, which governs environmental impact reviews of nuclear power plant operating license renewals (NRC 2012b). Specifically, the revised rule will update and re-evaluate the potential environmental impacts associated with the renewal of an operating license for a nuclear power reactor for an additional 20 years. A revised GEIS, which updates the 1996 GEIS, provides the technical basis for the revised rule. The revised GEIS specifically supports the revised list of NEPA issues and associated environmental impact findings for license renewal contained in Table B-1 in Appendix B to Subpart A of the revised 10 CFR Part 51. The revised GEIS and rule reflect lessons learned and knowledge gained during previous license renewal environmental reviews. In addition, public comments received on the draft revised GEIS and rule and during previous license renewal environmental reviews were re-examined to validate existing environmental issues and identify new ones.

The revised rule identifies 78 environmental impact issues, of which 17 will require plant-specific analysis. The revised rule consolidates similar Category 1 and 2 issues, changes some Category 2 issues into Category 1 issues and consolidates some of those issues with existing Category 1 issues. The revised rule also adds new Category 1 and 2 issues. The new Category 1 issues include geology and soils, exposure of terrestrial organisms to radionuclides, exposure of aquatic organisms to radionuclides, human health impact from chemicals, and physical occupational hazards. Radionuclides released to groundwater, effects on terrestrial resources (non-cooling system impacts), minority and low-income populations (i.e., environmental justice), and cumulative impacts were added as new Category 2 issues.

The revised rule is expected to be published in 2013; it will become effective 30 days after publication in the *Federal Register*. Compliance by license renewal applicants will not be required until one year from the date of publication (i.e., license renewal environmental reports submitted later than one year after publication must be compliant with the new rule). Nevertheless, under NEPA, the NRC must now consider and analyze, in its license renewal SEISs, the potential significant impacts described by the revised rule's new Category 2 issues, and to the extent there is any new and significant information, the potential significant impacts described by the revised rule's new Category 1 issues.

Where appropriate, **bold** text indicates specific text corrections or additions to the DSEIS and ~~strikeout~~ indicates deletions from the DSEIS text. This supplement to the DSEIS, and any changes made to it based on public comments, will be incorporated back into the main supplemental environmental impact statement (SEIS) prior to publishing the final document.

2.0 AFFECTED ENVIRONMENT

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

3.0 ENVIRONMENTAL IMPACTS OF REFURBISHMENT

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

4.0 ENVIRONMENTAL IMPACTS OF OPERATION

This chapter addresses potential environmental impacts related to the period of extended operation of Seabrook Station (Seabrook). These impacts are grouped and presented according to resource. Generic issues (Category 1) rely on the analysis provided in the generic environmental impact statement (GEIS) (NRC 1996, 1999) and are discussed briefly. Site-specific issues (Category 2) have been analyzed for Seabrook and assigned a significance level of SMALL, MODERATE, or LARGE, accordingly. Some remaining issues are not applicable to Seabrook because of site characteristics or plant features.

4.1 Land Use

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

4.2 Air Quality

No changes from the DSEIS issued in August 2011.

4.3 Geologic Environment

4.3.1 Geology and Soils

As described in Section 1.0 of this supplement, the U.S. Nuclear Regulatory Commission (NRC) has approved a revision to its environmental protection regulation, Title 10 of the *Code of Federal Regulations*, Part 51 (10 CFR Part 51). With respect to the geologic environment of a plant site, the revised rule amends Table B-1 in Appendix B, Subpart A, to 10 CFR Part 51 by adding a new Category 1 issue, “Geology and soils.” This new issue has an impact level of SMALL. This new Category 1 issue considers geology and soils from the perspective of those resource conditions or attributes that can be affected by continued operations during the renewal term. An understanding of geologic and soil conditions has been well established at all nuclear power plants and associated transmission lines during the current licensing term, and these conditions are expected to remain unchanged during the 20-year license renewal term for each plant. The impact of these conditions on plant operations and the impact of continued power plant operations and refurbishment activities on geology and soils are SMALL for all nuclear power plants and not expected to change appreciably during the license renewal term. Operating experience shows that any impacts to geologic and soil strata would be limited to soil disturbance from construction activities associated with routine infrastructure renovation and maintenance projects during continued plant operations. Implementing best management practices would reduce soil erosion and subsequent impacts on surface water quality. Information in plant-specific SEISs prepared to date, and GEIS reference documents have not identified these impacts as being significant.

Section 2.2.3 of this SEIS describes the local and regional geologic environment relevant to Seabrook. The NRC staff did not identify any new and significant information with regard to this Category 1 (generic) issue based on review of the Environmental Report (ER) (NextEra 2010), the public scoping process, or as a result of the environmental site audit. As discussed in Chapter 3 of this SEIS and as identified in the ER (NextEra 2010), NextEra Energy Seabrook, LLC. (NextEra) has no plans to conduct refurbishment or

construction of new facilities during the license renewal term. Further, it is anticipated that routine plant operation and maintenance activities would continue in areas previously disturbed by construction activities, including existing transmission line rights-of-way (ROWs). Based on this information, it is expected that any incremental impacts on geology and soils during the license renewal term would be SMALL.

~~4.3~~**4.4** Surface Water Resources

No changes to the text from the DSEIS issued in August 2011.

~~4.4~~**4.5** Groundwater Resources

The groundwater issues applicable to Seabrook are listed in **Table 4.5-1**~~Table 4.5-1~~ (also see Table B-1 of Appendix B of 10 CFR Part 51). Groundwater use and water quality relative to Seabrook are described in Sections 2.1.7.2 and~~-~~ 2.2.5 of this SEIS, respectively.

Table 4.5-1. Groundwater use and quality issues

Issues	GEIS sections	Category
Groundwater use conflicts (potable & service water; plants that use <100 gallons per minute)	4.8.1.1	1
Groundwater quality degradation (saltwater intrusion)	4.8.2.1	1
Radionuclides released to groundwater	**To be determined[a]**	**2**

[a]NRC 2012, since the revised GEIS has not been finalized and approved by the Commission, the revised GEIS ssection can not be referenced in this table.

4.5.1 Generic Groundwater Issues

No changes to the text from the DSEIS issued in August 2011.

4.5.2 Groundwater Use Conflicts

No changes to the text from the DSEIS issued in August 2011.

4.5.3 Radionuclides Released to Groundwater

With respect to groundwater quality, the revised rule amends Table B–1 in Appendix B, Subpart A, to 10 CFR Part 51 by adding a new Category 2 issue, “Radionuclides released to groundwater,” with an impact level range of SMALL to MODERATE, to evaluate the potential impact of discharges of radionuclides from plant systems into groundwater. This new Category 2 issue has been added to evaluate the potential impact to groundwater quality from the discharge of radionuclides from plant systems, piping, and tanks. This issue was added because, within the past several years, there have been events at nuclear power reactor sites that involved unknown, uncontrolled, and unmonitored releases of radioactive liquids into the groundwater. A discussion of groundwater quality concerns at Seabrook is included in Section 2.2.5 of the August 2011 DSEIS, and an assessment of the significance of groundwater quality degradation due to tritium contamination is presented in Section 4.10 of the August 2011 DSEIS.

As detailed in Section 2.2.5 of the August 2011 DSEIS, the NRC staff indicated that groundwater with elevated tritium activity concentrations was detected in the annular space around the Unit 1 containment structure in September 1999. In response to the elevated tritium concentrations, NextEra initiated a leak investigation which identified a leak source associated with the cask loading area and transfer canal adjacent to the spent fuel pool. In addition, NextEra has undertaken leak source elimination efforts and other corrective actions, which ultimately involved installation of a groundwater dewatering and pumping system to mitigate contaminated groundwater. An extensive groundwater monitoring network was also installed to provide surveillance of groundwater quality across the Seabrook site.

NextEra has monitored the dewatering system since 2000, the results of which were reviewed by NRC staff in support of the preparation of the August 2011 DSEIS. The highest tritium levels (up to 3,500,000 picocuries per liter (pCi/L) in 2003) were found in water removed from around the Unit 1 containment enclosure ventilation area (CEVA). Since monitoring began, NextEra has found that the tritium levels are trending down. Based on the most recent (2011) dewatering system monitoring data available for the site, tritium concentrations in the CEVA have ranged from 2,150 up to 50,000 pCi/L (NextEra 2011a).

NextEra continues to conduct groundwater monitoring as part of its participation in the Nuclear Energy Institute's Groundwater Protection Initiative (NextEra 2010). Monitoring results obtained through the onsite Groundwater Protection Program are reported in NextEra's radioactive effluent release reports, which are submitted to the NRC. Based on monitoring results from Seabrook's network of 27 groundwater monitoring wells through the end of 2011, the highest concentration of tritium detected was 2,850 pCi/L in well SW-1, a shallow aquifer well located near the Unit 1 containment structure. EPA's drinking water standard (or Maximum Contaminant Level) is 20,000 pCi/L. Several other nearby wells had lower tritium levels, while samples from most wells yielded no tritium above analytical detection limits. Monitoring results from a line of perimeter wells located south and downgradient of the tritium leak source have shown no tritium detections. Finally, NextEra reported no unplanned, unanticipated, or abnormal releases of liquid effluents from the site to unrestricted areas during 2010 and 2011 (NextEra, 2010a, 2011b, 2012).

As noted above and further discussed in the August 2011 DSEIS, the Unit 1 groundwater dewatering system, in combination with pumping from beneath the incomplete Unit 2 containment building, functions at Seabrook to remove and provide hydraulic containment of the tritium-contaminated groundwater by reversing the hydraulic gradient and flow of groundwater offsite. No offsite migration of tritium in groundwater has been observed to date. Further, the only drinking water wells (Town of Seabrook) are located hydraulically upgradient from the Seabrook site, and there is no drinking water pathway onsite.

While tritium continues to be detected above background levels at several onsite locations, the applicant is actively monitoring and controlling the tritium concentrations on site. The tritium-impacted groundwater is sent to the facility's main outfall to the ocean, where it is released in compliance with National Pollutant Discharge Elimination System (NPDES) and NRC's radiological limits. Tritium concentrations in groundwater as measured in onsite monitoring wells have remained well below EPA's 20,000 pCi/L drinking water standard. Based on the information presented above and in Sections 2.2.5 and 4.10 of the August 2011 DSEIS, the NRC concludes that inadvertent releases of tritium have not substantially impaired site groundwater quality or affected

groundwater use downgradient of the Seabrook site. The NRC staff further concludes that groundwater quality impacts would remain SMALL during the license renewal term.

~~4.5~~4.6 Aquatic Resources

Section 2.1.6 of this SEIS describes Seabrook's cooling-water system, and Section 2.2.6 describes the aquatic resources. Category 1 issues in 10 CFR Part 51, Subpart A, Appendix B, Table B–1, which are applicable to the operation of Seabrook's cooling-water systems during the renewed license term, are listed in Table 4.6-1. The NRC staff did not find any new and significant information during the review of the ER (NextEra 2010), the site audit, the scoping process, or the evaluation of other available information; therefore, the NRC staff concludes that there are no impacts related to aquatic resource issues beyond those discussed in the GEIS (NRC 1996) and the revised rule (NRC 2012). Consistent with the GEIS and the revised rule, the NRC staff concludes that the impacts are SMALL, and additional site-specific mitigation measures are unlikely to be sufficiently beneficial to warrant implementation.

Table 4.6-1. Aquatic resources issues

Issues	GEIS sections	Category
For all plants		
Accumulation of contaminants in sediments or biota	4.2.1.2.4	1
Entrainment of phytoplankton & zooplankton	4.2.2.1.1	1
Cold shock	4.2.2.1.5	1
Thermal plume barrier to migrating fish	4.2.2.1.6	1
Distribution of aquatic organisms	4.2.2.1.6	1
Premature emergence of aquatic insects	4.2.2.1.7	1
Gas supersaturation (gas bubble disease)	4.2.2.1.8	1
Low dissolved oxygen in the discharge	4.2.2.1.9	1
Losses from predation, parasitism, & disease among organisms exposed to sublethal stresses	4.2.2.1.10	1
Stimulation of nuisance organisms	4.2.2.1.11	1
Exposure of aquatic organisms to radionuclides	**To be determined[a]**	**1**
For plants with once-through dissipation systems		
Entrainment of fish & shellfish in early life stages	4.1.2	2
Impingement of fish & shellfish	4.1.3	2
Heat shock	4.1.4	2

Table source (61 FR 28467): Table B–1 in Appendix B, Subpart A, to 10 CFR Part 51.

[a] NRC 2012, since the revised GEIS has not been finalized and approved by the Commission, the revised GEIS ssection can not be referenced in this table.~~NRC 2012~~

~~4.5.1~~4.6.1 Exposure of Aquatic Organisms to Radionuclides

As described in Section **1.0 of this SEIS, the NRC has approved a revision to its environmental protection regulation, 10 CFR Part 51. With respect to the aquatic organisms, the revised rule amends Table B-1 in Appendix B, Subpart A, to 10 CFR Part 51 by adding a new Category 1 issue, “Exposure of aquatic organisms to radionuclides,” among other changes. This new Category 1 issue considers the impacts to aquatic organisms from exposure to radioactive effluents discharged from a nuclear power plant during the license renewal term. An understanding of the radiological conditions in the aquatic environment from the discharge of radioactive effluents within NRC regulations has been well established at nuclear power plants during their current licensing term. Based on this information, the NRC concluded that the doses to aquatic organisms are expected to be well below exposure guidelines developed to protect these organisms and assigned an impact level of SMALL.**

The NRC staff has not identified any new and significant information related to the exposure of aquatic organisms to radionuclides during its independent review of Seabrook’s ER, the site audit, and the scoping process. Section 2.1.2 of this SEIS describes the applicant’s Radioactive Waste Management Program to control radioactive effluent discharges to ensure that they comply with NRC regulations in 10 CFR Part 20. Section 4.9.3 of this SEIS contains the NRC staff’s evaluation of Seabrook’s Radioactive Effluent and Radiological Environmental Monitoring programs. Seabrook’s Radioactive Effluent and Radiological Environmental Monitoring programs provide further support for the conclusion that the impacts of aquatic organisms from radionuclides are SMALL.

The NRC staff concludes that there would be no impacts to aquatic organisms from radionuclides beyond those impacts contained in Table B-1 in Appendix B, Subpart A, to 10 CFR Part 51 of the revised rule; therefore, the impacts to aquatic organisms from radionuclides are SMALL.

~~4.5.2~~4.6.2 Generic Aquatic Ecology Issues

No changes to the text from the DSEIS issued in August 2011.

~~4.5.3~~4.6.3 Entrainment and Impingement

No changes to the text from the DSEIS issued in August 2011.

~~4.5.4~~4.6.4 Thermal Shock

No changes to the text from the DSEIS issued in August 2011.

~~4.5.5~~4.6.5 Mitigation

No changes to the text from the DSEIS issued in August 2011.

~~4.5.6~~4.6.6 Combined Impacts

No changes to the text from the DSEIS issued in August 2011.

4.7 Terrestrial Resources

~~The issues related to terrestrial resources~~ **The Category 1 (generic) and Category 2 (site-specific) terrestrial resources issues applicable to Seabrook are listed in Table 4.7-1.** ~~There are no Category 2 issues related to terrestrial resources. The NRC staff did not identify~~

~~any new and significant information during the review of the applicant's ER (NextEra, 2010), the NRC staff's site audit, the scoping process, or the evaluation of other available information. Therefore, there are no impacts related to these issues beyond those discussed in the GEIS. For these issues, the GEIS concluded that the impacts are SMALL, and additional site-specific mitigation measures are not likely to be sufficiently beneficial to warrant implementation.~~

Table 4.7-1. Terrestrial resources issues

Section 2.2.7 provides a description of the terrestrial resources at Seabrook and in the surrounding area.

Issues	GEIS section	Category
Cooling tower impacts on crops & ornamental vegetation	4.3.4	1
Cooling town impacts on native plants	4.3.5.1	1
Bird collisions with cooling towers	4.3.5.2	1
Powerline ROW management (cutting herbicide application)	4.5.6.1	1
Bird collisions with powerlines	4.5.6.1	1
Impacts of electromagnetic fields on flora and fauna (plants, agricultural crops, honeybees, wildlife, livestock)	4.5.6.3	1
Floodplains & wetland on powerline ROW	4.5.7	1
Exposure of terrestrial organisms to radionuclides	**To be determined[a]**	**1**
Effects on terrestrial resources (non-cooling system impacts)	**To be determined[a]**	**2**

[a] NRC 2012, since the revised GEIS has not been finalized and approved by the Commission, the revised GEIS ssection can not be referenced in this table.

4.7.1 Generic Terrestrial Resource Issues

For the Category 1 terrestrial resources issues listed in Table 4.7-1, the NRC staff did not identify any new and significant information during the review of the ER (NextEra 2010), the NRC staff's site audit, the scoping process, or the evaluation of other available information. Therefore, there are no impacts related to these issues beyond those discussed in the GEIS and the revised rule (NRC 2012). For these issues, the GEIS and the revised rule concluded that the impacts are SMALL, and additional site-specific mitigation measures are unlikely to be sufficiently beneficial to warrant implementation.

4.7.1.1 Exposure of Terrestrial Organisms to Radionuclides

As described in Section 1.0 of this SEIS, the NRC has approved a revision to its environmental protection regulation, 10 CFR Part 51. With respect to the terrestrial organisms, the revised rule amends Table B-1 in Appendix B, Subpart A, to 10 CFR Part 51 by adding a new Category 1 issue, "Exposure of terrestrial organisms to radionuclides," among other changes. This new issue has an impact level of SMALL. This new Category 1 issue considers the impacts to terrestrial organisms from exposure to radioactive effluents discharged from a nuclear power plant during the license renewal term. An understanding of the radiological conditions in the terrestrial environment from the discharge of radioactive effluents within NRC regulations has been well established at nuclear power plants during their current licensing term. Based on the revision to the environmental protection guidance and the staff's understanding of radiological conditions, the NRC concluded that the doses to terrestrial organisms are expected to be

well below exposure guidelines developed to protect these organisms and assigned an impact level of SMALL.

The NRC staff has not identified any new and significant information related to the exposure of terrestrial organisms to radionuclides during its independent review of Seabrook's ER, the site audit, and the scoping process. Section 2.1.2 of this SEIS describes the applicant's Radioactive Waste Management Program to control radioactive effluent discharges to ensure that they comply with NRC regulations in 10 CFR Part 20. Section 4.9.3 of this SEIS contains the NRC staff's evaluation of Seabrook's Radioactive Effluent and Radiological Environmental Monitoring programs. Seabrook's Radioactive Effluent and Radiological Environmental Monitoring programs provide further support for the conclusion that the impacts from radioactive effluents are SMALL.

Therefore, the NRC staff concludes that there would be no impact to terrestrial organisms to radionuclides beyond those impacts contained in Table B-1 in Appendix B, Subpart A, to 10 CFR Part 51 of the revised rule; therefore, the impacts to terrestrial organisms from radionuclides are SMALL.

4.7.2 Effects on Terrestrial Resources (Non-cooling System Impacts)

As described in Section 1.0 of this supplement, the NRC has approved a revision to its environmental protection regulation, 10 CFR Part 51. With respect to the terrestrial organisms, the revised rule amends Table B-1 in Appendix B, Subpart A, to 10 CFR Part 51 by expanding the Category 2 issue, "Refurbishment impacts," among others, to include normal operations, refurbishment, and other supporting activities during the license renewal term. This issue remains a Category 2 issue with an impact level range of SMALL to LARGE; however, the revised rule renames this issue "Effects on terrestrial resources (non-cooling system impacts)."

Section 2.2.7 of this SEIS describes the terrestrial resources on and in the vicinity of the Seabrook site, and Section 2.2.8 describes protected species and habitats. During the construction of Seabrook, approximately 22 percent of the plant site (194 ac (79 ha)) was cleared for buildings, parking lots, roads, and other infrastructure. By 2014, NextEra plans to have returned approximately 32 ac (13 ha), which are currently occupied by excavation spoil, to its natural state. The remaining terrestrial habitats have not changed significantly since construction. As discussed in Chapter 3 of this SEIS and according to the applicant's ER (NextEra 2010), NextEra has no plans for refurbishment or other license renewal-related construction activities. Further, it is anticipated that routine plant operation and maintenance activities would continue in areas previously disturbed by construction activities, including existing transmission line ROWs. Based on the staff's independent review, the staff concurs that operation and maintenance activities that NextEra might undertake during the renewal term, such as maintenance and repair of plant infrastructure (e.g., roadways, piping installations, onsite transmission lines, fencing and other security infrastructure), would likely be confined to previously - disturbed areas of the plant site or along the in-scope transmission line corridors. Therefore, the staff expects non-cooling system impacts on terrestrial resources during the license renewal term to be SMALL.

~~4.6~~4.8 Protected Species and Habitats

No changes to the text from the DSEIS issued in August 2011.

~~4.7~~4.9 Human Health

The human health issues applicable to Seabrook are discussed below and listed in **Table 4.9-1**~~Table 4.9-1~~ for Category 1, Category 2, and uncategorized issues.

Table 4.9-1. Human health issues

Table B-1 of Appendix B to Subpart A of 10 CFR Part 51 contains more information on these issues.

Issues	GEIS section	Category
Radiation exposures to the public during refurbishment	3.8.1[a]	1
Occupational radiation exposures during refurbishment	3.8.2[a]	1
Microbiological organisms (occupational health)	4.3.6	1
Microbiological organisms (public health, for plants using lakes or canals or discharging small rivers)	4.3.6[b]	2
Noise	4.3.7	1
Radiation exposures to public (license renewal term)	4.6.2	1
Occupation radiation exposures (license renewal term)	4.6.3	1
Electromagnetic fields—acute effects (electric shock)	4.5.4.1	2
Electromagnetic fields—chronic effects	4.5.4.2	Uncategorized
Human health impact from chemicals	**To be determined[c]**	**1**
Physical occupational hazards	**To be determined[c]**	**1**

[a] Issues apply to refurbishment, an activity that Seabrook does not plan to undertake.

[b] Issue applies to plant features such as cooling lakes or cooling towers that discharge to small rivers. The issue does not apply to Seabrook.

[a]NRC 2012, since the revised GEIS has not been finalized and approved by the Commission, the revised GEIS ssection can not be referenced in this table.

~~4.7.1~~4.9.1 Generic Human Health Issues

Category 1 issues in 10 CFR Part 51, Subpart A, Appendix B, Table B-1, applicable to Seabrook in regard to radiological impacts, are listed in **Table 4.9-2**~~Table 4.9-2~~. NextEra stated in its ER (NextEra 2010) that it was aware of one new radiological issue associated with the renewal of the Seabrook operating license—elevated tritium concentrations in groundwater adjacent to Unit 1. The groundwater monitoring for tritium is discussed later in this section. The NRC staff determined that the issue, while new, is not significant. Section 4.10 contains the discussion of this issue. The NRC staff has not identified any new and significant information, beyond this issue identified by the applicant, during its independent review of NextEra's ER, the site visit, the scoping process, or its evaluation of other available information.

4.9.1.1 New Category 1 Human Health issues

As described in Section 1.4 of this SEIS, the NRC has approved a revision to its environmental protection regulation, 10 CFR Part 51. With respect to the human health, the revised rule amends Table B-1 in Appendix B, Subpart A, to 10 CFR Part 51 by adding two new Category 1 issues, "Human health impact from chemicals" and "Physical occupational hazards." The first issue considers the impacts from chemicals to plant workers and members of the public. The second issue only considers the non-radiological occupational hazards of working at a nuclear power plant. An

understanding of these non-radiological hazards to nuclear power plant workers and members of the public have been well established at nuclear power plants during the current licensing term. The impacts from chemical hazards are expected to be minimized through the applicant's use of good industrial hygiene practices as required by permits and Federal and State regulations. Also, the impacts from physical hazards to plant workers will be of small significance if workers adhere to safety standards and use protective equipment as required by Federal and State regulations. The impacts to human health for each of these new issues from continued plant operations are SMALL.

The NRC staff has not identified any new and significant information related to these non-radiological issues during its independent review of NextEra's ER, the site audit, and the scoping process. Therefore, the NRC staff concludes that there would be no impact to human health from chemicals or physical hazards beyond those impacts described in Table B-1 in Appendix B, Subpart A, to 10 CFR Part 51 of the revised rule; therefore, the impacts are SMALL.

Table 4.9-2. Category 1 issues applicable to radiological impacts of normal operations during the renewal term

Issue—10 CFR Part 51, Subpart A, Appendix B, Table B-1	GEIS section
Human health	
Radiation exposures to public (license renewal term)	4.6.2
Occupational radiation exposures (license renewal term)	4.6.3

According to the GEIS, the impacts to human health are SMALL, and additional plant-specific mitigation measures are **unlikel**y to be sufficiently beneficial to be warranted (Category 1 issues). These impacts are expected to remain SMALL through the license renewal term.

~~4.7.1.1~~4.9.1.2 Radiological Impacts of Normal Operations

No changes to the text from the DSEIS issued in August 2011.

~~4.7.1.2~~4.9.1.3 Seabrook Radiological Environmental Monitoring Program

No changes to the text from the DSEIS issued in August 2011.

~~4.7.1.3~~4.9.1.4 Seabrook Radioactive Effluent Release Program

No changes to the text from the DSEIS issued in August 2011.

~~4.7.2~~4.9.2 Microbiological Organisms

No changes to the text from the DSEIS issued in August 2011.

~~4.7.3~~4.9.3 Electromagnetic Fields—Acute Shock

No changes to the text from the DSEIS issued in August 2011.

~~4.7.4~~4.9.4 Electromagnetic Fields—Chronic Effects

No changes to the text from the DSEIS issued in August 2011.

~~4.8~~4.10 Socioeconomics

No changes to the text from the DSEIS issued in August 2011.

~~4.9~~4.11 Evaluation of New and Potentially Significant Information

No changes to the text from the DSEIS issued in August 2011.

~~4.10~~4.12 Cumulative Impacts

No changes to the text from the DSEIS issued in August 2011.

~~4.11~~4.13 References

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5.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS

This chapter describes the environmental impacts from postulated accidents that Seabrook Station (Seabrook) might experience during the period of extended operation. A more detailed discussion of the severe accident mitigation alternative (SAMA) assessment is provided in Appendix F. The term "accident" refers to any unintentional event outside the normal plant operational envelope that results in a release or the potential for release of radioactive materials into the environment. Two classes of postulated accidents are evaluated in the *Generic Environmental Impact Statement (GEIS) for License Renewal of Nuclear Power Plants* prepared by the U.S. Nuclear Regulatory Commission (NRC) (NRC 1996), as listed in Table 5.1-1. These two classes include the following design-basis accidents (DBAs) and severe accidents.

Table 5.1-1. Issues related to postulated accidents

Two issues related to postulated accidents are evaluated under the National Environmental Policy Act of 1969 (NEPA) in the license renewal review—DBAs and severe accidents.

Issues	GEIS sections	Category
DBAs	5.3.2; 5.5.1	1
Severe accidents	5.3.3; 5.3.3.2; 5.3.3.3; 5.3.3.4; 5.3.3.5; 5.4; 5.5.2	2

5.1 Design-Basis Accidents

In order to receive NRC approval to operate a nuclear power facility, an applicant for an initial operating license must submit a safety analysis report (SAR) as part of its application. The SAR presents the design criteria and design information for the proposed reactor and comprehensive data on the proposed site. The SAR also discusses various hypothetical accident situations and the safety features that prevent and mitigate accidents. The NRC staff reviews the application to determine if the plant design meets the NRC's regulations and requirements and includes, in part, the nuclear plant design and its anticipated response to an accident.

DBAs are those accidents that both the applicant and the NRC staff evaluate to ensure that the plant can withstand normal and abnormal transients and a broad spectrum of postulated accidents, without undue hazard to the health and safety of the public. Many of these postulated accidents are not expected to occur during the life of the plant but are evaluated to establish the design basis for the preventative and mitigative safety systems of the facility. Title 10, Part 50, of the *U.S. Code of Federal Regulations* (10 CFR Part 50) and 10 CFR Part 100 describe the acceptance criteria for DBAs.

The environmental impacts of DBAs are evaluated during the initial licensing process, and the ability of the plant to withstand these accidents is demonstrated to be acceptable before issuance of the operating license. The results of these evaluations are found in license documentation such as the applicant's final safety analysis report, the NRC staff's safety evaluation report, the final environmental statement, and Section 5.1 of this supplement to the draft supplemental environmental impact statement (DSEIS). An applicant is required to maintain the acceptable design and performance criteria throughout the life of the plant, including any extended-life operation. The consequences for these events are evaluated for the hypothetical maximum exposed individual. Because of the requirements that continuous acceptability of the consequences and aging management programs be in effect for license renewal, the environmental impacts, as calculated for DBAs, should not differ significantly from initial licensing assessments over the life of the plant, including the license renewal period. Accordingly, the design of the plant, relative to DBAs during the extended period, is considered to remain acceptable; therefore, the environmental impacts of those accidents were not examined further in the GEIS.

The NRC has determined that the environmental impacts of DBAs are of SMALL significance for all plants because the plants were designed to successfully withstand these accidents. Therefore, for the purposes of license renewal, DBAs are designated as a Category 1 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1. The early resolution of the DBAs makes them a part of the current licensing basis (CLB) of the plant. The CLB of the plant is to be maintained by the applicant under its current license; therefore, under the provisions of 10 CFR 54.30, it is not subject to review under license renewal.

No new and significant information related to DBAs was identified during the review of the NextEra Energy Seabrook (NextEra) Environmental Report (ER), the site visit, the scoping process, or the NRC staff's evaluation of other available information. Therefore, there are no impacts related to DBAs beyond those discussed in the GEIS.

5.2 Severe Accidents

Severe nuclear accidents are those that are more severe than DBAs because they could result in substantial damage to the reactor core, whether or not there are serious offsite consequences. In the GEIS, the staff assessed the impacts of severe accidents during the license renewal period, using the results of existing analyses and information from various sites to predict the environmental impacts of severe accidents for plants during the renewal period. Severe accidents initiated by external phenomena—such as tornadoes, floods, earthquakes, fires, and sabotage—have not traditionally been discussed in quantitative terms in the final environmental impact statements and were not specifically considered for the Seabrook site in the GEIS (NRC 1996). The GEIS, however, did evaluate existing impact assessments performed by the NRC staff and by the industry at 44 nuclear plants in the U.S. It segregated all sites into six general categories and then estimated that the risk consequences calculated in existing analyses bound the risks for all other plants within each category. The GEIS further concluded that the risk from beyond design-basis earthquakes at existing nuclear power plants is designated as SMALL. The Commission believes that NEPA does not require the NRC to consider the environmental consequences of hypothetical terrorist attacks on NRC-licensed facilities. However, the NRC staff's GEIS for license renewal contains a discretionary analysis of terrorist acts in connection with license renewal. The conclusion in the GEIS is that the core damage and radiological release from such acts would be no worse than the damage and release to be expected from internally initiated events. In the GEIS, the NRC staff concludes that the risk from sabotage and beyond design-basis earthquakes at existing nuclear power plants is designated as SMALL and that the risks from other external events are adequately addressed by a generic consideration of internally initiated severe accidents (NRC 1996). Based on information in the GEIS, the staff found the following to be true:

> The generic analysis…applies to all plants and that the probability-weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to groundwater, and societal and economic impacts of severe accidents are of small significance for all plants. However, not all plants have performed a site-specific analysis of measures that could mitigate severe accidents. Consequently, severe accidents are a Category 2 issue for plants that have not

performed a site-specific consideration of severe accident mitigation and submitted that analysis for Commission review.

The staff identified no new and significant information related to postulated accidents during the review of NextEra's ER, the site audit, the scoping process, or evaluation of other available information. Therefore, there are no impacts related to postulated accidents beyond those discussed in the GEIS. In accordance with 10 CFR 51.53(c)(3)(ii)(L), however, the NRC staff has reviewed SAMAs for Seabrook. Review results are discussed in Section 5.3.

5.3 Severe Accident Mitigation Alternatives

Under 10 CFR 51.53(c)(3)(ii)(L), license renewal applicants must consider alternatives to mitigate severe accidents if the staff has not previously evaluated SAMAs for the applicant's plant in an environmental impact statement (EIS) or related supplement or in an environmental assessment. The purpose is to ensure that potentially cost-beneficial, aging-related plant changes (i.e., hardware, procedures, and training) with the potential for improving severe accident safety performance are identified and evaluated. SAMAs have not been previously considered by NextEra, for Seabrook; therefore, the remainder of Section 5.3 addresses those alternatives.

NextEra submitted an assessment of SAMAs for Seabrook as part of the ER (NextEra 2010), based on the most recently available Seabrook probabilistic risk assessment (PRA). This assessment is supplemented by a plant-specific offsite consequence analysis performed using the Methods for Estimation of Leakages and Consequences of Releases (MELCOR) Accident Consequence Code System 2 (MACCS2) computer code and insights from the Seabrook individual plant examination (IPE) (NHY 1991) and individual plant examination of external events (IPEEE) (North Atlantic Energy Service Corp. (NAESC) 1992). In identifying and evaluating potential SAMAs, NextEra considered SAMAs that addressed the major contributors to core damage frequency (CDF) and large early release frequency (LERF) at Seabrook, as well as a generic list of SAMA candidates for pressurized-water reactor (PWR) plants identified from other industry studies. **In the original ER,** NextEra identified 191 potential SAMA candidates. This list was reduced to 74 SAMA candidates by eliminating SAMAs for the following reasons:

- Seabrook has a different design.
- The SAMA has already been implemented at Seabrook.
- The intent of the SAMA has already been met at Seabrook.
- The SAMA has been combined with another SAMA candidate that is similar in nature.
- Estimated implementation costs would exceed the dollar value associated with eliminating all severe accident risk at Seabrook.
- The SAMA would be of very low benefit as it is related to a non-risk significant system.

NextEra assessed the costs and benefits associated with each of these 74 potential SAMAs and concluded in the ER that several of the candidate SAMAs evaluated are potentially cost beneficial.

Based on its review, the NRC staff issued requests for additional information (RAIs) to NextEra (NRC 2010a, 2011b). NextEra's responses addressed the NRC staff's concerns and resulted in the identification of additional potentially cost-beneficial SAMAs (NextEra 2011a, 2011b; NRC 2011a).

Subsequent to the RAI responses, NextEra submitted a supplement to the ER that incorporated updates to the PRA model (NextEra 2012a). NextEra identified four additional SAMA candidates that could be cost beneficial. The supplement to the ER assessed the costs and benefits of these additional SAMA candidates and reassessed the costs and benefits of the previously-identified SAMA candidates. The result of this analysis and reassessment is one additional potentially cost-beneficial SAMA. Based on its review of this supplement, the NRC staff issued RAIs to NextEra (NRC 2012a). NextEra's responses addressed the NRC staff's concerns (NextEra 2012b; NRC 2012b).

5.3.1 Risk Estimates for Seabrook

NextEra combined two distinct analyses to form the basis for the risk estimates used in the SAMA analysis—(1) the Seabrook Level 1 and 2 PRA model, which is an updated version of the IPE (NHY 1991), and (2) a supplemental analysis of offsite consequences and economic impacts (essentially a Level 3 PRA model) developed specifically for the SAMA analysis.[1] The SAMA analysis is based on the most recent Seabrook Level 1 and Level 2 PRA models available at the time of the ER, referred to as SSPSS-**2011** (the model-of-record used to support SAMA evaluation). The scope of this Seabrook PRA includes both internal and external events.

Table 5.3-1 indicates the Seabrook CDF, based on initiating events, for internal events (plus internal **and external** flooding **and severe weather**), fires, and seismic events (NextEra **2012a**, **2012b**).

Table 5.3-1. Seabrook CDF for internal and external events

Initiating event	CDF (per year)~~(a)~~	% Contribution to total CDF(a~~b~~)
Loss of offsite power (LOOP)—due to weather[(e)]	**6.8×10^{-7}** ~~1.5×10^{-6}~~	**6** ~~10~~
Flood in relay room from high-energy line break (HELB)[(e)]	**5.9×10^{-7}** ~~9.5×10^{-7}~~	**5** ~~6~~
Steam generator tube rupture (SGTR)	**5.7×10^{-7}**	**5**
Reactor trip—condenser available	**5.4×10^{-7}** ~~9.3×10^{-7}~~	**4** ~~6~~
Medium loss-of-coolant accident (LOCA)	**5.3×10^{-7}**	**4**
LOOP due to grid-related events	**4.5×10^{-7}**~~9.0×10^{-7}~~	**4** ~~6~~
Flood in yard due to service water (SW) common return rupture[(e)]	**4.1×10^{-7}**~~8.1×10^{-7}~~	**3** ~~5~~
Loss of essential alternating current (AC) power 4 kV bus	**3.2×10^{-7}**~~7.3×10^{-7}~~	**3** ~~5~~
~~Steam generator tube rupture (SGTR)~~	~~5.9×10^{-7}~~	~~4~~
Loss of primary component cooling **water** system (**PCCW**) **B** train	**3.0×10^{-7}**~~5.3×10^{-7}~~	**3**~~4~~
Loss of **PCCW system A train**	**2.3×10^{-7}**~~3.9×10^{-7}~~	**2** ~~3~~

[1] The NRC uses PRA to estimate risk by computing real numbers to determine what can go wrong, how likely is it, and what are its consequences. Thus, PRA provides insights into the strengths and weaknesses of the design and operation of a nuclear power plant. For the type of nuclear plant currently operating in the U.S., a PRA can estimate three levels of risk. A Level 1 PRA estimates the frequency of accidents that cause damage to the nuclear reactor core. This is commonly called CDF. A Level 2 PRA, which starts with the Level 1 core damage accidents, estimates the frequency of accidents that release radioactivity from the nuclear power plant. A Level 3 PRA, which starts with the Level 2 radioactivity release accidents, estimates the consequences in terms of injury to the public and damage to the environment. (http://www.nrc.gov/about-nrc/regulatory/risk-informed/pra.html)

Initiating event	CDF (per year)~~[a]~~	% Contribution to total CDF[a~~b~~]
Major flood, rupture of SW Train A in primary auxiliary building (PAB)[e]	**2.2×10^{-7}** ~~3.5×10^{-7}~~	2
LOOP due to switchyard	**2.1×10^{-7}** ~~3.4×10^{-7}~~	2
Large flood, rupture SW Train A piping in PAB[e]	**2.0×10^{-7}** ~~3.4×10^{-7}~~	2
Large flood, rupture SW Train B piping in PAB[e]	**2.0×10^{-7}** ~~3.3×10^{-7}~~	2
Major flood, rupture of SW Train B in PAB[e]	**2.0×10^{-7}** ~~2.5×10^{-7}~~	2
Major flood, rupture of fire protection piping in turbine building impacting offsite power[e]	**1.8×10^{-7}** ~~2.5×10^{-7}~~	2
Loss of Train B essential AC Power (4 kV Bus ~~-~~E6)	**1.6×10^{-7}** ~~1.9×10^{-7}~~	1
Large flood, rupture of SW common return piping in PAB[e]	**1.4×10^{-7}** ~~1.7×10^{-7}~~	1
Large LOCA	**3.4×10^{-7}**	**2**
Other internal events[b~~c~~]	**1.6×10^{-6}** ~~1.0×10^{-6}~~	**13** ~~7~~
Total internal events CDF[e~~b~~]	**7.8×10^{-6}** ~~1.1×10^{-5}~~	**64** ~~70~~
	Fire Initiating Event	
Fire in control room—power~~-~~-operated relief valve (PORV) LOCA	**3.6×10^{-7}** ~~3.7×10^{-7}~~	**3** ~~2~~
Fire in switchgear (SWGR) room B—loss of Bus E6	**3.5×10^{-7}** ~~3.7×10^{-7}~~	**3** ~~2~~
Fire SWGR room A—loss of Bus E5	**3.1×10^{-7}** ~~2.1×10^{-7}~~	**2** ~~1~~
Fire control room—AC power loss	**1.8×10^{-7}** ~~1.4×10^{-7}~~	1
Other fire events[c]	**3.8×10^{-7}** ~~2.3×10^{-7}~~	2
Total fire events CDF~~(d)~~	**1.4×10^{-6}** ~~1.3×10^{-6}~~	**11** ~~9~~
	Seismic Initiating Event	
Seismic 0.7 g transient event	**9.3×10^{-7}** ~~9.2×10^{-7}~~	**8** ~~6~~
Seismic 1.0 g transient event	**8.9×10^{-7}** ~~8.7×10^{-7}~~	**7** ~~6~~
Seismic 1.4 g transient event	3.6×10^{-7}	**3** ~~2~~
~~Seismic 1.0 g anticipated transient without scram (ATWS)~~	~~1.1×10^{-7}~~	~~1~~
~~Seismic 1.4 g large LOCA~~	~~1.1×10^{-7}~~	~~1~~
~~Seismic 0.7 g ATWS~~	~~1.0×10^{-7}~~	~~1~~
~~Seismic 1.0 g large LOCA~~	~~8.9×10^{-8}~~	~~1~~
Other seismic events[d~~f~~]	**8.8×10^{-7}** ~~4.9×10^{-7}~~	**7** ~~3~~
Total seismic events CDF~~(d)~~	**3.1×10-6**	**25** ~~21~~
Total CDF (internal and external events)~~[g]~~	**1.2×10^{-5}** ~~1.5×10^{-5}~~	**100**

Initiating event	CDF (per year)~~[a]~~	% Contribution to total CDF[a~~b~~]

[References were revised, and only new text is provided below.]

[a] **Individual percent contributions may not sum exactly to subtotals due to round off.**

[b] **Obtained by subtracting the sum of the internal initiating event contributors to internal event CDF from the total internal events CDF.**

[c] **Obtained by subtracting the sum of the fire initiating event contributors to fire event CDF from the total fire events CDF.**

[d] **Obtained by subtracting the sum of the seismic initiating event contributors to seismic event CDF from the total seismic events CDF.**

[e] **NextEra explained in response to an RAI the difference in the frequencies reported for many initiating events for the 2006 and 2011 PRA models. The total internal events CDF in the 2011 model decreased slightly as a result of model enhancements, the internal flooding CDF increased as result of a more detailed flooding analysis, and the severe weather CDF decreased primarily due to the incorporation of more recent data (NextEra 2012b).**

The Level 2 Seabrook PRA model that forms the basis for the SAMA evaluation is an updated version of the Level 2 IPE model (New Hampshire Yankee (NHY1991) and IPEEE model (NAESC 1992), using a single containment event tree (CET) to address both phenomenological and systemic events. The Level 1 core damage sequences are linked directly with the CET, for which the quantified sequences are binned into a set of **21**~~14~~ release categories, which are subsequently grouped into **13**~~10~~ source term categories that provide the input to the Level 3 consequence analysis (NextEra **2012a**). Source terms were developed using the results of Modular Accident Analysis Program (MAAP), Version 4.0.**7** computer code calculations. The offsite consequences and economic impact analyses use the MACCS2 code to determine the offsite risk impacts on the surrounding environment and public. Inputs for these analyses include plant-specific and site-specific input values for core radionuclide inventory, source term and release characteristics, site meteorological data, projected population distribution within an 50-mi (80-km) radius for the year 2050, emergency response evacuation planning, and economic parameters. The core radionuclide inventory corresponds to the end-of-cycle values for Seabrook operating at 3,659 MWt, which is slightly above the current licensed power level of 3,648 MWt. The magnitude of the onsite impacts (in terms of cleanup and decontamination costs and occupational dose) is based on information provided in NUREG/BR-0184 (NRC 1997a). NextEra estimated the dose to the population within 80 km (50 mi) of the Seabrook site to be approximately **37.8** ~~10.7~~ person-rem (0.**378** ~~107~~ person-Sievert (Sv)) per year, as shown in Table 5.3-2 (NextEra **2012a**).

Table 5.3-2. Breakdown of population dose by containment release mode

Containment release mode	Population dose (Person-rem[a] per year)	% Contribution
Small early releases	**1.7** ~~5.3~~	**5** ~~49~~
Large early releases	**1.7** ~~1.6~~	**4** ~~15~~
Large late releases[b]	**34.4** ~~3.8~~	**91** ~~36~~
Intact containment	negligible	negligible
Total	**37.8** ~~10.7~~	**100**

[a] One person-rem = 0.01 person-Sv

[b] **Includes small early containment penetration failure to isolate and large late containment basemat failure (SELL).**

5.3.2 Adequacy of Seabrook PRA for SAMA Evaluation

The first Seabrook PRA was completed in December 1983 to provide a baseline risk assessment and an integrated plant and site model for use as a risk management tool. This model was subsequently updated in 1986, 1989, and 1990, with the last update used to support the IPE. Based on its review of the Seabrook IPE, as described in an NRC report dated March 1, 1992 (NRC 1992), the NRC staff concluded that the IPE submittal met the intent of generic letter (GL) 88-20, “Individual Plant Examination for Severe Accident Vulnerabilities” (NRC 1988). Although no severe accident vulnerabilities were identified in the Seabrook IPE, 14 potential plant improvements were identified. Four of the improvements have been implemented. Each of the 10 improvements not implemented is addressed by a SAMA in the current evaluation. The internal events CDF value from the 1991 Seabrook IPE (6.1×10^{-5} per year) is near the average of the range of the CDF values reported in the IPEs for Westinghouse four-loop plants, which ranges from about 3×10^{-6} per year to 2×10^{-4} per year, with an average CDF for the group of 6×10^{-5} per year (NRC 1997b). It is recognized that plants have updated the values for CDF subsequent to the IPE submittals to reflect modeling and hardware changes. Based on CDF values reported in the SAMA analyses for LRAs, the internal events CDF result for Seabrook used for the SAMA analysis (**7.8×10^{-6}** ~~1.1×10^{-5}~~ per year, including internal **and external** flooding) is somewhat lower than that for most other plants of similar vintage and characteristics.

There have been **11**~~10~~ revisions to the IPE model since the 1991 IPE submittal, and 3 revisions to the PRA model, from the original 1983 PRA model to the 1990 update used to support the IPE submittal. The SSPSA-**2011** model was used for the SAMA analysis. NextEra identified the major changes in each revision of the PRA, with the associated change in internal and external event CDF (NextEra 2010, 2011a, **2012a**). A comparison of the internal events CDF between the 1991 IPE and the **2011** PRA model used for the SAMA evaluation indicates a decrease of approximately **87**~~82~~ percent (from 6.1×10^{-5} per year to **7.8×10^{-6}**~~1.1×10^{-5}~~ per year). The external events CDF has increased by approximately 25 percent since the 1993 IPEEE (from 3.6×10^{-5} per year to 4.5×10^{-5} per year).

The Seabrook PRA model is an integrated internal and external events model that has integrated seismic-initiated, fire-initiated, and external flooding-initiated events with internal events since the initial 1983 PRA (NextEra 2011a). The external events models used in the SAMA evaluation are essentially those used in the IPEEE, with the exception of the seismic PRA model, which underwent a major update for the SSPSA-2005 model. The Seabrook IPEEE was submitted on October 2, 1992 (NAESC 1992), in response to Supplement 4 of GL 88-20 (NRC 1991). The submittal used the same PRA as was used for the IPE (i.e., SSPSA-1990) except for updates to the external events. No fundamental weaknesses or vulnerabilities to severe accident risk with regard to external events were identified. Improvements that have already been realized as a result of the IPEEE process minimized the likelihood of there being cost-beneficial enhancements as a result of the SAMA analysis, especially with the inclusion of a multiplier to account for the additional risk of seismic events. In a letter dated May 2, 2001, the NRC staff concluded that the submittal met the intent of Supplement 4 to GL 88-20, and the applicant’s IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities (NRC 2001).

Internal Events CDF

NextEra identified **three** peer reviews that have been performed on the PRA—a 1999 Westinghouse Owner’s Group (WOG) certification peer review, a 2005 focused peer review against the American Society of Mechanical Engineers (ASME) PRA standard (ASME 2003; NextEra 2010) **and a 2009 peer review of the internal flood model against the ASME PRA standard (ASME 2009; NextEra 2012a). None of the peer reviews** included examination of external flooding, fire, or seismic hazards. The 1999 certification peer review identified 30 Category A and B facts and observations (F&O), and the 2005 focused peer review identified 4 Category A and B F&Os.[2] NextEra provided the resolution of each of the 34 F&Os and stated that all have been dispositioned and implemented in the PRA model **(NextEra 2010)**. NextEra also **stated that there were no Category A and three Category B F&Os from the 2009 peer review, all of which were resolved and implemented in the PRA model (NextEra 2012a). NextEra** explained that many other internal reviews including vendor-assisted reviews have been performed on specific model updates and that comments from these reviews, along with plant changes and potential model enhancements, are tracked through a model change database to ensure that the comments are addressed in the periodic update process (NextEra 2011a).

Consistent with the requirements of the ASME 2009 PRA standard (ASME 2009), NextEra maintains PRA quality control at Seabrook via an existing administrative procedure that defines the quality control process for PRA updates and ensures that the PRA model accurately reflects the current Seabrook plant design, operation, and performance (NextEra 2011a). The quality control process includes monitoring PRA inputs for new information, recording new applicable information, assessing significance of new information, performing PRA revisions, and controlling computer codes and models. NextEra also stated that the PRA training qualification is performed as part of the Engineering Support Personnel Training Program. Given that the Seabrook internal events PRA model has been peer-reviewed, and the peer review findings were all addressed, and that NextEra has satisfactorily addressed NRC staff questions regarding the PRA, the NRC staff concludes that the internal events Level 1 PRA model is of sufficient quality to support the SAMA evaluation.

Seismic CDF

The Seabrook IPEEE seismic analysis used a seismic PRA following NRC guidance (NRC 1991). The seismic PRA included the following:

- a seismic hazard analysis (based on the EPRI (1988) and the Lawrence Livermore National Laboratory (LLNL) (NRC 1994) hazard curves),
- a seismic fragility assessment,
- seismic quantification to yield initiating event frequencies and conditional system failure probabilities, and
- plant model assembly to integrate seismic initiators and seismic-initiated component failures with random hardware failures and maintenance unavailabilities.

The seismic CDF resulting from the Seabrook IPEEE was calculated to be 1.2×10^{-5} per year using a site-specific seismic hazard curve, with sensitivity analyses yielding 1.3×10^{-4} per year using the LLNL seismic hazard curve and 6.1×10^{-6} per year using the EPRI seismic hazard curve. The Seabrook IPEEE did not identify any vulnerability due to seismic events but did identify two plant improvements to reduce seismic risk. Neither of the two improvements has

[2] Now termed a "Finding," a Category A or B F&Os is an "observation (an issue or discrepancy) that is necessary to address to ensure: [1] the technical adequacy of the PRA ... [2] the capability/robustness of the PRA update process, or [3] the process for evaluating the necessary capability of the PRA technical elements (to support applications)." (Nuclear Energy Institute (NEI) 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, 2008)

been implemented. Each of the two improvements is addressed by a SAMA in the current evaluation.

Subsequent to the IPEEE, NextEra updated the seismic PRA analysis. These updates included expanding fragility analysis, with additional components; using the more current EPRI uniform hazard spectrum (UHS); and improving modeling and documentation of credited operator actions.

NextEra stated that extensive internal technical reviews of the seismic PRA analysis were performed for the original 1983 PRA and again when the seismic analysis was revised for the IPEEE and when the seismic analysis was revised for the SSPSA-2005 PRA model update. No significant comments were documented from these reviews, and no formal peer reviews have been conducted on the seismic PRA model (NextEra 2011a). In response to an NRC staff request to assess the impact on the SAMA evaluation of updated seismic hazard curves developed by the U.S. Geological Survey (USGS) in 2008 (USGS 2008), NextEra provided a revised SAMA evaluation using a multiplier of 2.1 to account for the maximum estimated seismic CDF for the Seabrook of 2.2×10^{-5} per year. This was noted in the attachments to NRC Information Notice 2010[-18], generic issue (GI) 199, “Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants” (NRC 2010a, 2010b; NextEra 2011a, 2011b, **2012a**). Note that, in the process of estimating an appropriate multiplier, NextEra considered that the estimated seismic CDF of 2.2×10^{-5} per year did not credit the installation of the supplemental electrical power system (SEPS) diesel generators (DGs) in 2004, which, based on a subsequent PRA estimate, reduced seismic CDF by 26 percent. Therefore, in estimating the multiplier, NextEra first reduced the 2.2×10^{-5} per year estimate for seismic CDF by 26 percent to 1.6×10^{-5} per year. **Using a seismic CDF of 1.6×10^{-5} per year, the total CDF equates to 2.5×10^{-5} per year or 2.1 times the total CDF from Table 5.3-1 (1.2×10^{-5} per year).**

The NRC staff concludes that the seismic PRA model, in combination with the use of a seismic events multiplier **of 2.1**, provides an acceptable basis for identifying and evaluating the benefits of SAMAs. This conclusion is based on the fact that the Seabrook seismic PRA model is integrated with the internal events PRA, the seismic PRA has been updated to include additional components and to extend the fragility screening threshold, the SAMA evaluation was updated using a multiplier to account for a potentially higher seismic CDF, and NextEra has satisfactorily addressed NRC staff RAIs regarding the seismic PRA.

Fire CDF

The Seabrook IPEEE fire analysis employed EPRI’s fire-induced vulnerability evaluation (FIVE) methodology (Electric Power Research Institute (EPRI) 1992) based on definitions of Appendix R fire areas for Seabrook. Qualitative and quantitative screening was performed to determine that 13 of the 73 fire areas contained important equipment (pumps, valves, and cabling, etc.). These were further assessed. Final quantification used the Seabrook IPE PRA model to calculate a fire-induced CDF of 1.2×10^{-5} per year. While no physical plant changes were found to be necessary as a result of the IPEEE fire analysis, potential plant improvements to reduce fire risk were identified—of which, four have been implemented. The one improvement not implemented is addressed by a SAMA in the current evaluation.

NextEra updated the fire PRA, subsequent to the IPEEE, in support of the **SSPSS**-2004 PRA update. NextEra stated that the fire analysis methodology used was essentially the same, with some variations, as that described previously for the IPEEE fire analysis (NextEra 2011a).

NextEra stated that extensive internal technical reviews of the fire PRA analysis were performed for the original 1983 PRA and, again, when the fire analysis was revised for the IPEEE and when the fire analysis was revised for the **SSPSS**-2005 PRA model update. No significant comments were documented from these reviews, and no formal peer reviews have been conducted on the fire PRA model (NextEra 2011a). Considering that the Seabrook fire PRA model is integrated with the internal events PRA, that the fire PRA has been updated to include more current data, and that NextEra has satisfactorily addressed NRC staff RAIs regarding the fire PRA, the NRC staff concludes that the fire PRA model provides an acceptable basis for identifying and evaluating the benefits of SAMAs.

“Other” External Event CDF

The Seabrook IPEEE analysis of “other” external events included high winds, external floods, transportation accidents, etc. (HFO events), and it followed the screening and evaluation approaches specified in Supplement 4 to GL 88-20 (NRC 1991), concluding that Seabrook met the 1975 Standard Review Plan (SRP) criteria (NRC 1975b). The following external event frequencies exceeded the 1.0×10^{-6} per year screening criterion (NAESC 1992):

- flooding resulting from a storm surge caused by a hurricane, which is modeled in the PRA (NextEra 2010) and reported to contribute 2×10^{-8} per year to the total Seabrook CDF, and
- a truck crash into the SF6 transmission lines, which has been mitigated by the installation of jersey barriers and guard rails and that, as a result, has been screened from the PRA model (NextEra 2011a).

While no physical plant changes were found to be necessary as a result of the IPEEE HFO analysis, one plant improvement based on HFO analysis was recommended, but this has already been implemented (NextEra 2011a). The Seabrook IPEEE submittal also stated that, as a result of the Seabrook IPE, cost-benefit analyses were being performed for many potential plant improvements, which may also collaterally reduce external event risk. Four of these five potential plant improvements have been implemented, and the fifth is addressed by a SAMA in the current evaluation.

Level 2 and LERF

To translate the results of the Level 1 PRA into containment releases, as well as the results of the Level 2 analysis, NextEra significantly revised the 2005 PRA update (i.e., PRA model **SSPSS**-2005) from that used in the IPE to reflect the Seabrook plant as designed and operated as of 2006. NextEra explained that the quantification of the Level 1 and Level 2 models is done using a linked event tree method approach that does not employ plant damage states (NextEra 2011a). Therefore, all Level 1 sequences are evaluated by the CET. The Level 2 model is a single CET and evaluates the phenomenological progression of all the Level 1 sequences including internal, fire, and seismically initiated events. It has 37 branching events, for each of which the split fraction is determined based on the type of event. End states resulting from the combinations of the branches are then assigned to one of **21**~~16~~ release categories based on characteristics that determine the timing and magnitude of the release, whether or not the containment remains intact, and isotopic composition of the released material. The quantified CET sequences are subsequently grouped into ~~10~~**13** source term categories by grouping those that occur due to different phenomena but for which the consequence is essentially the same. **Eight of the release categories were mapped one-to-one into a corresponding source term category while 13 release categories were**

mapped into five combined source term categories. These 13 source term categories provide the input to the Level 3 consequence analysis.

Source terms were developed for each of the source term categories. The release fractions and timing for ~~5 of the 10~~ source term categories are based on the results of plant-specific calculations using the MAAP Version 4.0.**7.** NextEra generally selected the representative MAAP case based on that which resulted in the most realistic timing and source term release. **For four of the combined source term categories, the source term for the release category having the highest (dominant) release frequency was used as the source term for the combined category. In the fifth combined source term category, one of the contributors had the most significant source term and the highest frequency so it was selected as the representative case.**

The current Seabrook Level 2 PRA model is an update of that used in the IPE, which did not identify any severe accident vulnerabilities associated with containment performance. The NRC staff's review of the IPE back-end (i.e., Level 2) model concluded that it appeared to have addressed the severe accident phenomena normally associated with large dry containments, that it met the IPE requirements, and that there were no obvious or significant problems or errors. The LERF model was included in the 1999 industry peer review. All F&Os from this review have been dispositioned and implemented in the PRA model. NextEra explained that the apparently very low LERF for Seabrook (1.2×10^{-7} per year in the SSPSS-2006 model, which is less than 1 percent of the CDF) results from the very large-volume and strong containment building in comparison to most other nuclear power plant containment designs (NextEra 2011a), such that there are no conceivable severe accident progression scenarios that result in catastrophic failure early in the accident sequence. The NRC staff considers NextEra's explanation reasonable. Based on the NRC staff's review of the Level 2 methodology, the NRC staff concludes that NextEra has adequately addressed NRC staff RAIs, that the LERF model was reviewed in more detail as part of the 1999 WOG certification peer review, and that all F&Os have been resolved. Therefore, the NRC staff concludes that the Level 2 PRA provides an acceptable basis for evaluating the benefits associated with various SAMAs.

Level 3—Population Dose

NextEra extended the containment performance (Level 2) portion of the PRA to assess offsite consequences (essentially a Level 3 PRA) via Version 1.13.1 of the MACCS2 code, including consideration of the source terms used to characterize fission product releases for the applicable containment release categories and the major input assumptions used in the offsite consequence analyses (NRC 1998). Plant-specific input to the code included the following:

- the source terms for each release category,
- the reactor core radionuclide inventory,
- site-specific meteorological data for the year 2005,
- projected population distribution within an 80-km (50-mi) radius for the year 2050, based on year 2000 census data from SECPOP2000 (NRC 2003),
- emergency evacuation planning, using only 95 percent of the population (conservative relative to NUREG-1150, which assumed 99.5 percent (NRC 1990)), and
- economic parameters including agricultural production.

Multiple sensitivity cases were run, including the following:

- releases at ground level and 25 percent, 50 percent, and 75 percent of the containment building height (baseline is release at the top of containment),
- release plumes with 1 and 10 MW heat release,
- factor-of-two scaling of containment building wake effects,
- annual meteorological data from 2004 through 2008,
- variations in evacuation parameters, such as percent of population, evacuation speed, and delay time, and
- **variations in sea-breeze circulation assumptions.**

NextEra's results showed only minor variations from the baseline for these sensitivities, which is consistent with previous SAMA analyses. The NRC staff concludes that the methodology used by NextEra to estimate the offsite consequences for Seabrook provides an acceptable basis from which to proceed with an assessment of risk reduction potential for candidate SAMAs. Accordingly, the NRC staff based its assessment of offsite risk on the CDF and offsite doses reported by NextEra.

5.3.3 Potential Plant Improvements

NextEra's process for identifying potential plant improvements (SAMAs) consisted of the following elements:

- review of the most significant basic events from the **2011** plant-specific PRA, which was the most current PRA model at the time the SAMA evaluation was performed,
- review of potential plant improvements identified in the Seabrook IPE and IPEEE,
- review of other industry documentation discussing potential plant improvements, and
- insights from Seabrook personnel.

Based on this process, an initial set of **195** ~~191~~ candidate "Phase I" SAMAs was identified, for which NextEra performed a qualitative screening to eliminate ones from further consideration using the following criteria:

- The SAMA is not applicable to Seabrook due to design differences (19 SAMAs screened).
- The SAMA has already been implemented at Seabrook or the Seabrook meets the intent of the SAMA (87 SAMAs screened).
- The SAMA is similar to another SAMA under consideration (11 SAMAs screened).
- The SAMA has estimated implementation costs that would exceed the dollar value associated with eliminating all severe accident risk at Seabrook (no SAMA screened).
- The SAMA was determined to provide very low benefit (no SAMA screened).

Based on this screening, 117 SAMAs were eliminated, leaving **78** ~~74~~ for detailed evaluation in Phase II. In Phase II, a detailed evaluation was performed for each of the remaining **78** SAMA candidates. NextEra accounted for the potential risk reduction benefits associated with each SAMA by quantifying the benefits using the integrated internal and external events PRA model.

The NRC staff reviewed NextEra's process for identifying and screening potential SAMA candidates, as well as the methods for quantifying the benefits associated with potential risk reduction. This included reviewing insights from the plant-specific risk studies, reviewing plant improvements considered in previous SAMA analyses, **and explicitly treating** external events in the SAMA identification process. The NRC staff concludes that NextEra used a systematic and comprehensive process for identifying potential plant improvements for Seabrook, and the set of SAMAs evaluated in the ER, together with those evaluated in response to NRC staff inquiries, is reasonably comprehensive; therefore, it is acceptable.

5.3.3.1 Risk Reduction

NextEra evaluated the risk-reduction potential of the **78** SAMAs retained for the Phase II evaluation**, which includes the risk-reduction potential of additional SAMAs identified in the 2012 SAMA supplement (NextEra 2012a) and in response to NRC staff RAIs (NextEra 2012b).** NextEra used model re-quantification to determine the potential benefits based on the SSPSS-**2011** PRA model. The majority of the SAMA evaluations were performed in a bounding fashion in that the SAMA was assumed to eliminate the risk associated with the proposed enhancement. On balance, such calculations overestimate the benefit and are conservative. The NRC staff reviewed NextEra's bases for calculating the risk reduction for the various plant improvements and concludes that the rationale and assumptions are reasonable and generally conservative (i.e., the estimated risk reduction is higher than what would actually be realized). Accordingly, the NRC staff based its estimates of averted risk for the various SAMAs on NextEra's risk reduction estimates.

5.3.3.2 Cost Impacts

NextEra developed plant-specific costs of implementing the **78** Phase II candidate SAMAs using an expert panel—composed of senior plant staff from the PRA group, the design group, operations, and license renewal—with experience in developing and implementing modifications at Seabrook. In most cases, detailed cost estimates were not developed because of the large margin between the estimated SAMA benefits and the estimated implementation costs (NextEra 2011a). The cost estimates, conservatively, did not specifically account for inflation, contingencies, implementation obstacles, or replacement power costs (RPC). The NRC staff reviewed the bases for the applicant's cost estimates and, for certain improvements, compared the cost estimates to estimates developed elsewhere for similar improvements, including estimates developed as part of other applicants' analyses of SAMAs for operating reactors and advanced light-water reactors. The NRC staff concludes that the cost estimates provided by NextEra are sufficient and appropriate for use in the SAMA evaluation.

5.3.3.3 Cost-Benefit Comparison

The methodology used by NextEra was based primarily on NRC's guidance for performing cost-benefit analysis (i.e., NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook* (NRC 1997a)). The guidance involves determining the net value for each SAMA according to the following formula:

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

where:

APE = present value of averted public exposure ($)

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

AOE = present value of averted occupational exposure costs ($)

AOSC = present value of averted onsite costs ($)

COE = cost of enhancement ($)

If the net value of a SAMA is negative, the cost of implementing the SAMA is larger than the benefit associated with the SAMA, and it is not considered cost beneficial. Present values for both a 3 percent and 7 percent discount rate were considered. Using the NUREG/BR-0184 methods, NextEra estimated the total present dollar value equivalent associated with eliminating severe accidents from internal and external events at Seabrook to be about $**3.05 million** ~~819,000~~**. Use of a multiplier of 2.1 to account for the additional risk from seismic events in the sensitivity analysis increases the value to $6.4 million. This represents the dollar value associated with completely eliminating all internal and external event severe accident risk at Seabrook, and it is** also referred to as the maximum averted cost risk (MACR).

If the implementation costs for a candidate SAMA exceeded the calculated benefit, the SAMA was considered not to be cost beneficial. In the baseline analysis (using a 7 percent discount rate), NextEra identified **three** ~~one~~ potentially cost-beneficial SAMA**s** (SAMA **157,** 165, **and 192,** see Table 5.3-3). Based on the consideration of analysis uncertainties, NextEra identified **three** ~~one~~ additional potentially cost-beneficial SAMA**s** (SAMA **164, 172,** and ~~193~~**195**, see Table 5.3-3). In addition, **as a result** of **the sensitivity analysis using** a multiplier **of 2.1** to account for **the additional risk from seismic events, NextEra identified one** additional **cost-beneficial SAMA (**SAMA **193, see** Table 5.3-3**).**

The **seven** ~~four~~ potentially cost-beneficial SAMAs are discussed in Section 5.3.4. The NRC staff notes that these are included within the set of SAMAs that NextEra plans to enter into the Seabrook long-range plan development process for further implementation consideration. The NRC staff concludes that, with the exception of the **seven** ~~four~~ potentially cost-beneficial SAMAs, the costs of the other SAMAs evaluated would be higher than the associated benefits.

5.3.4 Cost-Beneficial SAMAs

Highlighted in ***bold italics*** in Table 5.3-3 are the potentially cost-beneficial SAMAs (157, **164,** 165, **172,** 192, 193, and **195**).

Table 5.3-3. SAMA cost-benefit Phase-II analysis for Seabrook

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No station blackout (SBO): **Six** ~~Five~~ SAMAs analyzed	Eliminate failure of the emergency diesel generators (EDGs)	**22** ~~27~~	**6** ~~12~~	**220K (470K)** ~~160K (330K~~	**525K (1.1M)** ~~300K (620K~~	**1.75M** ~~>1.0M~~ (minimum of six)

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No LOOP: **Three** ~~Five~~ SAMAs analyzed	Eliminate LOOP events	**18** ~~42~~	**17** ~~36~~	**530K (1.2M**) ~~340K (700K~~	**1.2M (2.7M)** ~~640K (1.3M)~~	>**3M** ~~2.4M~~ (minimum of three)
No loss of 4 kV in-feed breakers: #21—Develop procedures to repair or replace failed 4 kV breakers	Eliminate failure of the 4 kV bus in-feed breakers	1	<1	8K (17K)	15K (32K)	Screened
No loss of high-pressure injection (HPI): **Two** ~~Three~~ SAMAs analyzed	Eliminate failure of the HPI system	**22** ~~68~~	**34** ~~52~~	**1.1M (2.3M**) ~~470K (980K~~	**2.5M (5.3M**) ~~890K (1.9M~~	**8.8M**~~>5.0M~~ (minimum of **both**~~three~~)
No loss of low-pressure injection: #28—Add a diverse low-pressure injection system	Eliminate failure of the low-pressure injection system	**2** ~~11~~	**2** ~~29~~	**68K (140K**) ~~160K (340K~~	**160K (340K**) ~~300K (640K~~	>**1M** ~~1.0M~~
No depletion of reactor water storage tank (RWST): Two SAMAs analyzed	Eliminate RWST running out of water	**13** ~~28~~	**10** ~~12~~	**310K (655K**) ~~160K (330K~~	**730K (1.5M**) ~~300K (630K~~	>**3M** ~~1.0M~~ (minimum of both)
Reduce common cause failure of the safety injection (~~safety injection~~SI) system: **#39—Replace two of the four electric SI pumps with diesel-powered pumps**	**Eliminate dependency of the existing intermediate head SI pump trains on AC power**	**<1**	**0**	**<1K (<1K)**	**<1K (<1K)**	**>5M**
No small LOCAs: #41—Create a reactor coolant depressurization system	Eliminate all small LOCA events	**2** ~~7~~	**1** ~~2~~	**27K (57K**) ~~33K (70K~~	**64K** (130K) ~~63K~~	>**1M** ~~1.0M~~
No direct current (DC) dependence for SW: #43—Add redundant DC control power for SW pumps	Eliminate the dependence of the SW pumps on DC power	**<2** ~~1~~	**0** ~~1~~	**11K (24K**) ~~10K (21K~~	**26K (55K**) ~~19K (40K~~	>100K

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No loss of component cooling water (CCW): **#44—Replace emergency core cooling system (-ECCS) pump motors with air-cooled motors**	**Eliminate failure of the CCW pumps**	**14**	**31**	**920K (1.9M)**	**2.15M (4.6M)**	**>6M**
No failure of support systems for core spray (CS) division B of HPI: **Six**~~Two~~ SAMAs analyzed	**Eliminate failures of support systems (e.g., AC and DC power, cooling) for division B of HPI**	**28** ~~25~~	**34** ~~23~~	**1.0M (2.2M**) ~~180K (380K~~	**2.45M (5.2M**) ~~350K (730K~~	>**6.4M**~~-1.0M~~ (minimum of **six**~~-both~~)
No CCW pump failure when AC/DC power available: **#59—Install a digital feed water upgrade**	**Eliminate CCW pump failure when AC and DC power support is available**	**4**	**11**	**335K (700K)**	**785K (1.7M)**	**>6.1M**
No plant risk **Two**~~Seven~~ SAMAs analyzed	Eliminate all **plant risk**	**100** ~~11~~	**100** ~~12~~	**3.05M (6.4M**) ~~92K (170K~~	**7.15M (15M**) ~~180K (370K~~	>**15M** ~~500K~~ (minimum of **two** ~~seven~~)
No **PORV failures**: #79—Install bigger pilot operated relief valve so only one is required	Eliminate all **PORV failures**	**<1** ~~12~~	**0** ~~7~~	**1.7K (4K**) ~~73K (150K~~	**4.1K (9K**) ~~140K (290K~~	>**2.7M** ~~1.0M~~
No heating, ventilation, and air conditioning (HVAC) dependence for CS, SI, RH, & containment building spray (CBS): #80—Provide a redundant train or means of ventilation	Eliminate the dependence of CS, SI, residual heat removal (RHR), & CBS pumps on HVAC	**3** ~~8~~	**5** ~~1~~	**150K (320K**) ~~32K (67K~~	**360K (750K**) ~~61K (130K~~	>**1M** ~~500K~~
No HVAC dependence for emergency feedwater (EFW): #84—Switch for EFW room fan power supply to station batteries	Eliminate loss of EFW ventilation	<1	**0** ~~<1~~	<1K (<**2K**~~-1K~~)	<**2K** ~~1K~~ (<**4K**~~-2K~~)	>250K

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No CBS support system or common cause failures: **Two SAMAs analyzed**	**Eliminate CBS power, signal, and cooling support system failures, and common cause failure among similar components for one division of CBS**	**0**	**58**	**1.7M (3.5M)**	**4.0M (8.3M)**	**>10M (minimum of two)**
No failure of human action to vent containment: **#93—Install an unfiltered hardened containment vent** ~~Four SAMAs analyzed~~	Eliminate ~~all~~**failure of the human action to vent** containment	0	**1** ~~36~~	**39K (82K)** ~~160K (340K~~	**92K (190K)** ~~310K (650K~~	**>3M** ~~3.0M (minimum of six)~~
No release from containment venting and reduced release from basemat melt-through: **#94—Install a filtered containment vent to remove decay heat**	**Eliminate release category LL3 (containment vent) and prevent 80 percent of release category LL5 (basemat melt-through)**	**0**	**69**	**2.0M (4.1M)**	**4.6M (9.7M)**	**>20M**
Reduced likelihood of non-recovery off off-site power: **#99—Strengthen primary & secondary containment (e.g., add ribbing to containment shell)**	**Reduce by a factor of 10 the non-recovery of off-site power before late containment pressure failure occurs**	**0**	**4**	**120K (245K)**	**270K (570K)**	**11.5M**
Reduced failure of CBS: **#107—Install a redundant containment spray system**	**Add redundant train of CBS**	**0**	**1**	**29K (62K)**	**69K (140K)**	**>10M**
No hydrogen burns or detonations: Three SAMAs analyzed	Eliminate all hydrogen ignition and burns	0	**1** ~~0~~	**18K (39K)** ~~<1K (<1K~~	**43K (90K)** ~~<1K (<1K~~	>100K (minimum of three)

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No failure of operator action to transfer to long-term recirculation following large LOCA: #105—Delay containment spray actuation after a large LOCA	Eliminate the human failure to complete/ ensure the RHR/low-head safety injection (LHSI) transfer to long-term recirculation during large LOCA events	**3** ~~2~~	**0** ~~<1~~	**12K (25K)** ~~7.2K (15K~~	**27K (58K)** ~~14K (29K~~	>100K
No high-pressure core ejection: **#110—Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure**	**Eliminate high-pressure core ejection occurrences**	**0**	**0**	**<1K (<1K)**	**1K (2K)**	**>10M**
No containment isolation valve (CIV) failures: Two SAMAs analyzed	**Eliminate CIV failures**	0	**6** ~~19~~	**115K (240K)** ~~100K (220K~~	**270K (570K)** ~~200K (420K~~	>**1M** ~~500K~~ (minimum of both)
~~Reduce ISLOCA risk by half~~	~~Reduce ISLOCA event risk by 50%~~	~~1~~	~~3~~	~~14K (30K)~~	~~27K (60K)~~	~~>100K~~
No interfacing system loss-of-coolant accidents (ISLOCAs): **Three** ~~Two~~SAMAs analyzed	Eliminate all ISLOCAs	**<1** ~~2~~	**3** ~~7~~	**48K (100K)** ~~28K (60K~~	110K **(240K)** ~~53K~~	>**500** ~~190K~~ (minimum of ~~both~~**three**)
No SGTRs: Five SAMAs analyzed	Eliminate all SGTR events	**5** ~~3~~	**2** ~~17~~	**67K (140K)** ~~86K (180K~~	160K (**330K**~~345K~~)	>500K (minimum of five)
No **anticipated transient without scrams (**ATWSs**)**: Four SAMAs analyzed	Eliminate all ATWS events	**4** ~~3~~	**2** ~~11~~	**60K (130K)** ~~70K (150K~~	**140K (290K)** ~~130K (280K~~	>500K (minimum of four)
No piping system LOCAs: #147—Install digital large break LOCA protection system	Eliminate all piping failure LOCAs	**9** ~~10~~	**2** ~~12~~	**77K (160K)** ~~100K (220K~~	**180K (380K)** ~~200K (410K~~	>500K

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No secondary side depressurization from stem line break upstream of main steam isolation valves: #153—Install secondary side guard pipes up to the main steam isolation valves	Eliminate all steam line break events	**<1** ~~0~~	**0** ~~<1~~	**5K (11K)** ~~3K (7K~~	**11K (24K)** ~~6K (13K~~	>500K
No operator error when aligning & loading SEPS DGs: #154—Modify SEPS design to accommodate: (a) automatic bus loading, (b) automatic bus alignment	Eliminate failure of all operator actions to align and load the SEPS DGs	**8** ~~NP*~~	**2** ~~NP~~	**64K (135K)** ~~33K (68K~~	**150K (320K)** ~~62K (130K~~	>750K
Provide independent AC power to battery chargers: ***#157—Provide independent AC power source for battery chargers; for example, provide portable generator to charge station battery***	***Eliminate failure of operator action to shed DC loads to extend batteries to 12 hours & eliminate failure to recover offsite power for plant-related, grid-related, & weather-related LOOP events***	***<2*** ~~4~~	***1*** ~~2~~	***34K (72K)*** ~~23K (48K~~	***80K (170K)*** ~~45K (95K~~	***30K***
#159—Install additional batteries						>1.0M
No depletion of condensate storage: **#162—Increase the capacity margin of the condensate storage tank (CST)** ~~Two SAMAs analyzed~~	Eliminate CST running out of water	**<2** ~~1~~	1	**35K (73K)** ~~9K (18K~~	**81K (170K)** ~~16K (34K~~	**>2.5M**
#164—Modify 10" condensate filter flange to have a 2-1/2" ~~-inch~~ ***female fire hose adapter with isolation valve***						>40K ~~(minimum of both)~~

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No loss of turbine-driven auxiliary feedwater (TDAFW): #163—Install third EFW pump (steam-driven)	Eliminate failure of the TDAFW train	5 ~~19~~	12 ~~9~~	360K (750K) ~~100K (210K~~	835K (1.8M) ~~190K (400K~~	>2.0M
Guaranteed success of RWST long-term makeup without recirculation: ***#165—RWST fill from firewater during containment injection—Modify 6" RWST Flush Flange to have a 2½" female fire hose adapter with isolation valve***	***Guaranteed success of RWST makeup for long-term sequences where recirculation is not available***	*5 ~~10~~*	*2 ~~8~~*	*57K (120K) ~~75K (160K~~*	*130K (280K) ~~120K (300K~~*	***50K***
No loss of reactor coolant pump (RCP) seal cooling and no failure of RCP seals following a plant transient: ***#172—Evaluate installation of a "shutdown seal" in the RCPs being developed by Westinghouse***	***Eliminate failure of RCP seal cooling initiating event and RCP seal failures subsequent to a plant transient***	***34***	***49***	***1.5M (3.2M)***	***2.5M (7.4M)***	***>2M***
No fire in turbine building at west wall or relay room: #175—Improve fire detection in turbine building relay room	This SAMA has been implemented (NextEra 2011b)					
No failure of operator action to close PORV block valve during a control room fire: #179—Fire~~-~~induced LOCA response procedure from alternate shutdown panel	Eliminate **operator failure to close PORV block valve during a** control room fire	0 ~~1~~	0 ~~<1~~	<1K (<1K) ~~4K (8K~~	<1K (<2K) ~~7K (15K~~	>20K
No failures due to seismic relay chatter: #181—Improve relay chatter fragility	Eliminate all seismic relay chatter failures	12 ~~9~~	3 ~~12~~	87K (180K) ~~100K (210K~~	200K (470K~~410K~~)	>600K

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No seismic-induced loss of DGs or turbine-driven emergency feedwater (TDEFW): #182—Improve seismic capacity of EDGs & steam-driven EFW pump	Eliminate all seismic failures of EDGs or TDEFW	**<1** ~~0~~	0	**2.4K (6K**) ~~<1K (<1K~~	**5.6K (12K**) ~~<1K (<1K~~	>500K
Containment purge valves are always closed: #184—Control/reduce time that the containment purge valves are in open position	Eliminate possibility of containment purge valves being open at the time of an event	0	~~≈~~0	<1K (<1K)	<1K (<**2K**~~1K~~)	>20K
No CDF contribution from pre-existing containment leakage: #186—Install containment leakage monitoring system	Eliminate all CDF contribution from pre-existing containment leakage	**0** ~~NP~~	**0** ~~NP~~	**4.4K (12K**) ~~11K (23K~~	**10K (27K**) ~~20K (43K~~	>500K
Benefits of SEPS success criteria change, from two of two SEPS DGs to one of two SEPS DGs: #189—Modify or analyze SEPS capability; one of two SEPS for loss of system pressure (LOSP) non-SI loads, two of two for LOSP SI loads	Modify fault tree so that one of two SEPS DGs are required rather than both SEPS DGs being required	**6** ~~7~~	**2** ~~1~~	**63K (130K**) ~~30K (60K~~	**150K (310K**) ~~60K (120K~~	>**2M** ~~300K~~
No inadvertent failures of redundant temperature logic during loss of PCCW: #191—Remove the 135 ºF temperature trip of the PCCW pumps	Eliminate inadvertent failure of the redundant temperature element/logic of the associated primary component cooling (PCC) division for both loss of PCCW initiating events & loss of PCCW mitigative function	<1	**0** ~~<1~~	<1K (<1K)	<1K (<**2K** ~~1K~~)	>100K

Analysis case & applicable SAMAs (where multiples, only number & minimum cost are listed)	Modeling assumptions	% Risk reduction		Total benefit ($)		Cost ($)
				Baseline (with 2.1 multiplier)		
		CDF	Pop. dose	Internal + External	with uncertainty	
No flooding in control building due to fire protection system actuation: ***#192—Install a globe valve or flow limiting orifice upstream in the fire protection system***	***Eliminate control building fire protection flooding initiators***	***24*** ***~~25~~***	***11*** ***~~6~~***	***470K (990K)*** ***~~160K (340K~~***	***1.1M (2.3M)*** ***~~310K (640K~~***	***370K ~~200K~~***
No failure of operator action to close CIV CS-V-167: ***#193—Hardware change to eliminate motor-operated valve (MOV) AC power dependency***	***Eliminate operator failure to close CIV CS-V-167***	***0***	***5*** ***~~35~~***	***86K (180K)*** ***~~190K (400K~~***	***200K (420K)*** ***~~365K (770K~~***	***300K***
No failure of main steam safety valves (MSSVs) to reseat: **#194—Purchase or manufacture a “gagging device” that could be used to close a stuck-open steam generator safety valve**	**Eliminate failure of MSSVs to reseat**	**0**	**0**	**<1K (<1K)**	**<1K (<2K)**	**>30K**
No failure of temperature elements for PCC Trains A and B: ***#195—Make improvements to PCCW temperature control reliability***	***Eliminate failure of temperature control and modulation for PCC Trains A and B that could fail PCCW***	***3***	***5***	***140K (300K)***	***340K (710K)***	***300K***

5.3.5 Conclusions

NextEra compiled a list of 191 SAMAs **in the ER and 4 additional SAMAs in the 2012 SAMA supplement (NextEra 2012a)** based on a review of the most significant basic events from the plant-specific PRA, insights from the plant-specific IPE and IPEEE, review of other industry documentation, and insights from Seabrook personnel. Of these, 117 SAMAs were eliminated qualitatively, leaving **78 candidate SAMAs for evaluation.** These underwent more detailed design and cost estimates to show that **three**~~two~~ were potentially cost beneficial in the baseline analysis (SAMAs 157, 165, and **192**). NextEra also performed additional analyses to evaluate the impact of parameter choices and uncertainties, resulting in **three additional** potentially cost-beneficial SAMAs **(SAMAs 164, 172, and 195). In addition**, NextEra **performed a**

sensitivity analysis accounting for the additional risk of seismic events and identified **one** additional **SAMA** (SAMA 193) as being potentially cost beneficial. NextEra has indicated that all **seven**~~four~~ potentially cost-beneficial SAMAs will be entered into the Seabrook long-range plan development process for further implementation consideration.

The NRC staff reviewed the NextEra analysis and concludes that the methods used and their implementation were acceptable. The treatment of SAMA benefits and costs support the general conclusion that the SAMA evaluations performed by NextEra are reasonable and sufficient for the license renewal submittal.

The NRC staff agrees with NextEra's identification of areas in which risk can be further reduced in a potentially cost-beneficial manner through the implementation of the identified, potentially cost-beneficial SAMAs. Given the potential for cost-beneficial risk reduction, the NRC staff agrees that further evaluation of **these** SAMAs by NextEra is **warranted. However,** the applicant **stated that** the **seven**~~four~~ potentially cost-beneficial SAMAs are not aging-related **in that they** do not involve aging management of passive, long-lived systems, structures, or components during the period of extended operation. Therefore, **the NRC staff concludes that** they need not be implemented as part of license renewal pursuant to 10 CFR Part 54.

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6.0 ENVIRONMENTAL IMPACTS OF THE URANIUM FUEL CYCLE, SOLID WASTE MANAGEMENT, AND GREENHOUSE GAS

6.1 The Uranium Fuel Cycle

This section addresses issues related to the uranium fuel cycle and solid waste management during the period of extended operation (listed in Table 6.1-1). The uranium cycle includes uranium mining and milling, the production of uranium hexafluoride, isotopic enrichment, fuel fabrication, reprocessing of irradiated fuel, transportation of radioactive materials, and management of low-level wastes and high-level wastes related to uranium fuel cycle activities. The generic potential impacts of the radiological and nonradiological environmental impacts of the uranium fuel cycle and transportation of nuclear fuel and wastes are described in detail in the *Generic Environmental Impact Statement* (GEIS) (NRC 1996, 1999). They are based, in part, on the generic impacts provided in Title 10, Part 51.51(b) of the *Code of Federal Regulations* (10 CFR 51.51(b)), Table S-3, "Table of Uranium Fuel Cycle Environmental Data," and in 10 CFR 51.52(c), Table S-4, "Environmental Impact of Transportation of Fuel and Waste to and from One Light-Water-Cooled Nuclear Power Reactor."

Table 6.1-1. Issues related to the uranium fuel cycle and solid waste management.

There are nine generic issues related to the fuel cycle and waste management. There are no site-specific issues.

Issues	GEIS Sections	Category
Offsite radiological impacts (individual effects from other than the disposal of spent fuel ~~and~~ **&** high-level waste)	6.1; 6.2.1; 6.2.2.1; 6.2.2.3; 6.2.3; 6.2.4; 6.6	1
Offsite radiological impacts (collective effects)	6.1; 6.2.2.1; 6.2.3; 6.2.4; 6.6	1
Offsite radiological impacts (spent fuel ~~and~~ **&** high-level waste disposal)	6.1; 6.2.2.1; 6.2.3; 6.2.4; 6.6	1
Nonradiological impacts of the uranium fuel cycle	6.1; 6.2.2.6; 6.2.2.7; 6.2.2.8; 6.2.2.9; 6.2.3; 6.2.4; 6.6	1
Low-level waste storage ~~and~~ **&** disposal	6.1; 6.2.2.2;6.4.2; 6.4.3; 6.4.3.1; 6.4.3.2; 6.4.3.3; 6.4.4; 6.4.4.1; 6.4.4.2; 6.4.4.3; 6.4.4.4; 6.4.4.5; 6.4.4.5.1; 6.4.4.5.2; 6.4.4.5.3; 6.4.4.5.4; 6.4.4.6;6.6	1
Mixed waste storage ~~and~~ **&** disposal	6.4.5.1; 6.4.5.2; 6.4.5.3; 6.4.5.4; 6.4.5.5; 6.4.5.6; 6.4.5.6.1; 6.4.5.6.2; 6.4.5.6.3; 6.4.5.6.4; 6.6	1
Onsite spent fuel	6.1; 6.4.6; 6.4.6.1; 6.4.6.2; 6.4.6.3; 6.4.6.4; 6.4.6.5; 6.4.6.6; 6.4.6.7; 6.6	1
Nonradiological waste	6.1; 6.5; 6.5.1; 6.5.2; 6.5.3; 6.6	1
Transportation	6.1; 6.3.1; 6.3.2.3; 6.3.3; 6.3.4; 6.6, Addendum 1	1

The U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the environmental impacts associated with spent nuclear fuel is addressed in two issues in Table 6-1, "Offsite radiological impacts (spent fuel and high-level waste disposal)" and "Onsite spent fuel." However, as explained later in this section, the scope of the evaluation of

these two issues in this supplemental environmental impact statement (SEIS) has been revised. The issue, "Offsite radiological impacts (spent fuel and high-level waste disposal)," is not evaluated in this SEIS. In addition, the issue, "Onsite spent fuel," only evaluates the environmental impacts during the license renewal term.

For the term of license renewal, the staff did not find any new and significant information related to the remaining uranium fuel cycle and solid waste management issues listed in Table 6–1 during its review of the Seabrook Station Environmental Report (ER) (NextEra 2010), the site visit, and the scoping process. Therefore, there are no impacts related to these issues beyond those discussed in the GEIS. For these Category 1 issues, the GEIS concludes that the impacts are SMALL, except for the issue, "Offsite radiological impacts (collective effects)," which the NRC has not assigned an impact level. This issue assesses the 100-year radiation dose to the U.S. population (i.e., collective effects or collective dose) from radioactive effluents released as part of the uranium fuel cycle for a nuclear power plant during the license renewal term compared to the radiation dose from natural background exposure. It is a comparative assessment for which there is no regulatory standard to base an impact level.

For the offsite radiological impacts resulting from spent fuel and high-level waste disposal and the on-site storage of spent fuel, which will occur after the reactors have been permanently shutdown, the NRC's Waste Confidence Decision and Rule represented the Commission's generic determination that spent fuel can continue to be stored safely and without significant environmental impacts for a period of time after the end of the licensed life for operation. This generic determination meant that the NRC did not need to consider the storage of spent fuel after the end of a reactor's licensed life for operation in NEPA documents that support its reactor and spent fuel storage application reviews.

The NRC first adopted the Waste Confidence Decision and Rule in 1984. The NRC amended the decision and rule in 1990, reviewed them in 1999, and amended them again in 2010 (49 FR 34694; 55 FR 38474; 64 FR 68005; and 75 FR 81032 and 81037). The Waste Confidence Decision and Rule are codified in 10 CFR 51.23.

On December 23, 2010, the Commission published in the Federal Register a revision of the Waste Confidence Decision and Rule to reflect information gained from experience in the storage of spent fuel and the increased uncertainty in the siting and construction of a permanent geologic repository for the disposal of spent nuclear fuel and high-level waste (75 FR 81032 and 81037). In response to the 2010 Waste Confidence Decision and Rule, the States of New York, New Jersey, Connecticut, and Vermont along with several other parties challenged the Commission's NEPA analysis in the decision, which provided the regulatory basis for the rule. On June 8, 2012, the United States Court of Appeals, District of Columbia Circuit in New York v. NRC, 681 F.3d 471 (D.C. Cir. 2012) vacated the NRC's Waste Confidence Decision and Rule, after finding that it did not comply with NEPA.

In response to the court's ruling, the Commission, in CLI-12-16 (NRC 2012a), in which the Commission determined that it would not issue licenses that rely upon the Waste Confidence Decision and Rule, until the issues identified in the court's decision are appropriately addressed by the Commission. In CLI-12-16, the Commission also noted that the decision not to issue licenses only applied to final license issuance; all licensing reviews and proceedings should continue to move forward.

In addition, the Commission directed in SRM-COMSECY-12-0016 (NRC 2012b) that the NRC staff proceed with a rulemaking that includes the development of a generic EIS to

support a revised Waste Confidence Decision and Rule and to publish both the EIS and the revised decision and rule in the Federal Register within 24 months (by September 6, 2014). The Commission indicated that both the EIS and the revised Waste Confidence Decision and Rule should build on the information already documented in various NRC studies and reports, including the existing environmental assessment that the NRC developed as part of the 2010 Waste Confidence Decision and Rule. The Commission directed that any additional analyses should focus on the issues indentified in the court's decision. The Commission also directed that the NRC staff provide ample opportunity for public comment on both the draft EIS and the proposed Waste Confidence Decision and Rule.

The revised rule and supporting EIS are expected to provide the necessary NEPA analyses of waste confidence-related human health and environmental issues. As directed by the Commission, the NRC will not issue a renewed license before the resolution of waste confidence-related issues. This will ensure that there would be no irretrievable or irreversible resource commitments or potential harm to the environment before waste confidence impacts have been addressed.

If the results of the Waste Confidence Decision and Rule and supporting EIS identify information that requires a supplement to this SEIS, the NRC staff will perform any appropriate additional NEPA review for those issues before the NRC makes a final licensing decision.

6.2 Greenhouse Gas Emissions

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

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7.0 ENVIRONMENTAL IMPACTS OF DECOMMISSIONING

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

8.0 ENVIRONMENTAL IMPACTS OF ALTERNATIVES

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

9.0 CONCLUSION

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

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APPENDIX A.
COMMENTS RECEIVED ON THE SEABROOK STATION ENVIRONMENTAL REVIEW

A. COMMENTS RECEIVED ON THE SEABROOK STATION ENVIRONMENTAL REVIEW

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

APPENDIX B.
NATIONAL ENVIRONMENTAL POLICY ACT ISSUES FOR LICENSE RENEWAL OF NUCLEAR POWER PLANTS

B. NATIONAL ENVIRONMENTAL POLICY ACT ISSUES FOR LICENSE RENEWAL OF NUCLEAR POWER PLANTS

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

APPENDIX C.
APPLICABLE REGULATIONS, LAWS, AND AGREEMENTS

C. APPLICABLE REGULATIONS, LAWS, AND AGREEMENTS

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

APPENDIX D.
CONSULTATION CORRESPONDENCE

D. CONSULTATION CORRESPONDENCE

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

APPENDIX E.
CHRONOLOGY OF ENVIRONMENTAL REVIEW

E. CHRONOLOGY OF ENVIRONMENTAL REVIEW CORRESPONDENCE

No changes from the draft supplemental environmental impact statement (DSEIS) issued in August 2011.

APPENDIX F
U.S. NUCLEAR REGULATORY COMMISSION STAFF EVALUATION OF SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR SEABROOK STATION UNIT 1 IN SUPPORT OF LICENSE RENEWAL APPLICATION REVIEW

F U.S. NUCLEAR REGULATORY COMMISSION STAFF EVALUATION OF SEVERE ACCIDENT MITIGATION ALTERNATIVES FOR SEABROOK STATION UNIT 1 IN SUPPORT OF LICENSE RENEWAL APPLICATION REVIEW

F.1 Introduction

NextEra Energy Seabrook, LLC (NextEra), submitted an assessment of severe accident mitigation alternatives (SAMAs) for the Seabrook Station (Seabrook), Unit 1, as part of its Environmental Report (ER) (NextEra 2010). This assessment was based on the most recent Seabrook probabilistic risk assessment (PRA) available at that time, a plant-specific offsite consequence analysis performed using the Methods for Estimation of Leakages and Consequences of Releases (MELCOR) Accident Consequence Code System 2 (MACCS2) computer code (NRC 1998a), and insights from the Seabrook individual plant examination (IPE) (New Hampshire Yankee (NHY) 1991) and individual plant examination of external events (IPEEE) (North Atlantic Energy Service Corp. (NAESC) 1992). In identifying and evaluating potential SAMAs, NextEra considered SAMA candidates that addressed the major contributors to core damage frequency (CDF) and large early release frequency (LERF) at Seabrook, as well as a generic list of SAMA candidates for pressurized-water reactor (PWR) plants identified from other industry studies. **In the initial ER,** NextEra identified 191 potential SAMA candidates. This list was reduced to 74 SAMA candidates by eliminating SAMAs for the following reasons:

- Seabrook has a different design.
- The SAMA has already been implemented at Seabrook.
- The intent of the SAMA has already been met at Seabrook.
- The SAMA has been combined with another SAMA candidate that is similar in nature.
- Estimated implementation costs would exceed the dollar value associated with eliminating all severe accident risk at Seabrook.
- The SAMA would be of very low benefit as it is related to a non-risk significant system.

NextEra assessed the costs and benefits associated with each of these 74 potential SAMAs and concluded in the ER that several of the candidate SAMAs evaluated are potentially cost beneficial.

Based on a review of the SAMA assessment, the U.S. Nuclear Regulatory Commission (NRC) issued requests for additional information (RAIs) to NextEra by letters dated November 16, 2010 (NRC 2010a), and March 4, 2011 (NRC 2011b). Key questions in these RAIs concerned the following:

- additional details regarding the plant-specific PRA model and changes to internal and external event CDF and LERF since the IPE,

- the process used to map Level 1 PRA results into the Level 2 analysis and group containment event tree (CET) end states into release categories,[1]
- the process for selecting the representative Modular Accident Analysis Program (MAAP) case for each release category and the release characteristics of each representative case,
- changes to the fire and seismic PRA models since the IPEEE,
- the impact of updated seismic hazard curves,
- the sensitivity of the SAMA results to assumptions used in the Level 3 analysis,
- the use of Level 2 importance analysis and industry SAMA analyses in identifying plant-specific SAMAs, and
- further information on the cost-benefit analysis of several specific candidate SAMAs and low-cost alternatives.

NextEra submitted additional information to the NRC by letters dated January 13, 2011 (NextEra 2011a), and April 18, 2011 (NextEra 2011b). NextEra provided additional information in a telephone conference call with the NRC staff on February 15, 2011 (NRC 2011a). In response to the RAIs, NextEra provided the following:

- the internal and external event contribution to CDF and LERF for each version of the Seabrook PRA model and model changes that most impacted CDF and LERF,
- a description of the CET and the process for determining the frequency of each release category,
- a description of the process for selecting representative MAAP cases for each release category and the characteristics of each plume in each release category,
- changes to the fire and seismic PRA models since the IPEEE,
- a sensitivity analysis of the impact on the SAMA analysis from updated seismic hazard curves,
- the results of the sensitivity analyses performed on the assumptions used in the Level 3 analysis,
- listings of the important basic events for the most risk-significant release categories,
- **evaluation of additional** SAMA candidates **based on basic events** important **to CDF and release frequency**,
- a review of the applicability of industry cost-effective SAMA candidates to Seabrook, and
- additional information regarding several specific SAMAs.

[1] The NRC uses PRA to estimate risk by computing real numbers to determine what can go wrong, how likely is it, and what are its consequences. Thus, PRA provides insights into the strengths and weaknesses of the design and operation of a nuclear power plant. For the type of nuclear plant currently operating in the U.S., a PRA can estimate three levels of risk. A Level 1 PRA estimates the frequency of accidents that cause damage to the nuclear reactor core. This is commonly called CDF. A Level 2 PRA, which starts with the Level 1 core damage accidents, estimates the frequency of accidents that release radioactivity from the nuclear power plant. A Level 3 PRA, which starts with the Level 2 radioactivity release accidents, estimates the consequences in terms of injury to the public and damage to the environment. (http://www.nrc.gov/about-nrc/regulatory/risk-informed/pra.html)

NextEra's responses addressed the NRC staff's concerns and resulted in the identification of additional potentially cost-beneficial SAMAs.

Subsequent to the RAI responses, NextEra submitted a supplement to the ER that incorporates updates made to the PRA model (NextEra 2012a). NextEra identified additional SAMA candidates, assessed the costs and benefits of these SAMAs, and reassessed the costs and benefits of the previously-identified SAMA candidates, which resulted in additional potentially cost-beneficial SAMAs.

The NRC staff reviewed this supplement and issued RAIs to NextEra by letter dated July 16, 2012 (NRC 2012a). Key questions in these RAIs concerned the following:

- **additional initiating event contributors to total CDF,**
- **additional basic events presented in the CDF and release category importance lists,**
- **justification for the implementation cost estimates for certain SAMAs, and**
- **clarification of apparent inconsistencies in the risk reduction and cost-benefit evaluation of certain SAMAs.**

NextEra submitted additional information to the NRC by letter dated September 13, 2012 (NextEra 2012b). NextEra also provided additional information in a telephone conference call with the NRC staff on October 3, 2012 (NRC 2012b). In response to the RAIs, NextEra provided the following:

- **initiating events that contribute one percent and greater to CDF,**
- **additional risk-significant release category basic events and evaluation of SAMA candidates for each,**
- **justification for the increase in the implementation costs for selected SAMAs since the ER and original RAI responses were submitted to the NRC, and**
- **additional information regarding the cost-benefit evaluation of certain SAMAs.**

NextEra's responses addressed the NRC staff's concerns.

The NRC staff notes that many of the original RAIs asked regarding the SAMA analysis in the ER, and associated RAI responses, were superseded by the updated information provided in the 2012 SAMA supplement (NextEra 2012a). For this reason, many of the RAI responses on the original ER submittal are not specifically discussed in this review since they were determined to not be needed to support the conclusions presented in Section F.7.

An assessment of SAMAs for Seabrook is presented below.

F.2 Estimate of Risk for Seabrook

NextEra's estimates of offsite risk at Seabrook are summarized in Section F.2.1. The summary is followed by the NRC staff's review of NextEra's risk estimates in Section F.2.2.

F.2.1 NextEra's Risk Estimates

Two distinct analyses are combined to form the basis for the risk estimates used in the SAMA analysis—(1) the Seabrook Level 1 and 2 PRA model, which is an updated version of the IPE (NHY 1991), and (2) a supplemental analysis of offsite consequences and economic impacts (essentially a Level 3 PRA model) developed specifically for the SAMA analysis. The SAMA analysis is based on the most recent Seabrook Level 1 and Level 2 PRA models available**, model** SSPSS-2006 **for** the **ER (NextEra 2010) updated by** model **SSPSS-2011 in the 2012** SAMA **supplement (NextEra 2012a**). The scope of this Seabrook PRA includes both internal and external events.

The Seabrook CDF is approximately **1.2×10^{-5}**~~1.5×10^{-5}~~ per year for both internal and external events, as determined from quantification of the Level 1 PRA model. A truncation level of 1×10^{-14} per year was used when quantifying event trees, and a truncation value of 1×10^{-12} per year was used when quantifying fault trees, except for the service water system (SWS) (NextEra 2011a). The SWS was divided into two trains, which were each solved at a truncation level of 1×10^{-8} per year. The CDF is based on the risk assessment for internally initiated events, which include internal flooding, and external events, which include fire and seismic events. The internal events CDF is approximately **7.8×10^{-6}**~~1.1×10^{-5}~~ per year **(internal events modeling includes external flooding),** and the external events CDF (fire and seismic events) is approximately 4.5×10^{-6} per year (NextEra **2012a**).

The breakdown of CDF by initiating event is provided in Table F-1 and includes internal, fire, and seismic initiating events. As shown in Table F-1, the largest single contributor to the total CDF is loss of offsite power (LOOP) due to weather. NextEra **clarified in response to an NRC staff RAI (NextEra 2012a)** that station blackout (SBO) contributes approximately **3.3×10^{-6}**~~5.3×10^{-6}~~ per year, or **27**~~35~~ percent, and anticipated transients without scram (ATWS) contribute approximately **4.7×10^{-7}**~~4.6×10^{-7}~~ per year, or **4**~~3~~ percent, to the total internal and external events CDF.

The Level 2 Seabrook PRA model that forms the basis for the SAMA evaluation is an updated version of the Level 2 IPE model (NHY 1991) and IPEEE model (NAESC 1992). The current Level 2 model uses a single CET that is used to address internal, fire, and seismic events. The CET addresses both phenomenological and systemic events. The Level 1 core damage sequences are linked directly with the CET, so all Level 1 sequences are evaluated by the CET (NRC 2011a). The CET probabilistically evaluates the progression of the damaged core with respect to release to the environment. CET nodes are evaluated using supporting fault trees and logic rules. The CET end states are then examined for considerations of timing and magnitude of release and assigned to release categories.

Table F-1. Seabrook CDF for internal and external events

Internal initiating event	CDF (per year)	% contribution to total CDF [a]
LOOP due to weather[e]	6.8×10^{-7} ~~1.5×10^{-6}~~	6 ~~10~~
Flood in relay room from high-energy line break (HELB)[e] ~~Loss of essential alternating current (AC) power 4 kilovolt (kV) bus~~	5.9×10^{-7} ~~9.5×10^{-7}~~	5 ~~6~~
Steam generator tube rupture (SGTR)	5.7×10^{-7}	5
Reactor trip—condenser available	5.4×10^{-7} ~~9.3×10^{-7}~~	4 ~~6~~
Medium loss-of-coolant accident (LOCA)	5.3×10^{-7}	4
LOOP due to grid-related events	~~9.0×10^{-7}~~	4 ~~6~~
Flood in yard due to service water (SW) common return rupture[e] ~~LOOP due to hardware or maintenance~~	4.1×10^{-7} ~~8.1×10^{-7}~~	3 ~~5~~
Loss of essential alternating current (AC) power 4 ~~kilovolt (kV)~~ bus ~~Flood in turbine building~~	3.2×10^{-7} ~~7.3×10^{-7}~~	3 ~~5~~
~~Steam generator tube rupture (SGTR)~~	~~5.9×10^{-7}~~	~~4~~
Loss of primary component cooling **water** system (~~CS)~~**PCCW) B** train	3.0×10^{-7} ~~5.3×10^{-7}~~	3 ~~4~~
Loss of PCCW system A train ~~Loss of essential direct current (DC) power 125V DC bus~~	2.3×10^{-7} ~~3.9×10^{-7}~~	2 ~~3~~
Major flood, rupture of SW Train A in PAB[e] ~~Reactor trip—during shutdown~~	2.2×10^{-7} ~~3.5×10^{-7}~~	2
LOOP due to switchyard ~~Interfacing systems loss-of-coolant accident (ISLOCA)~~	2.1×10^{-7} ~~3.4×10^{-7}~~	2
Large flood, rupture SW Train A piping in primary auxiliary building (PAB)[e] ~~Large loss-of-coolant accident (LOCA)~~	2.0×10^{-7} ~~3.4×10^{-7}~~	2
Large flood, rupture SW Train B piping in PAB[e] ~~Medium LOCA~~	2.0×10^{-7} ~~3.3×10^{-7}~~	2
Major flood, rupture of SW Train B in PAB[e] ~~Excessive LOCA~~	2.0×10^{-7} ~~2.5×10^{-7}~~	2
Major flood, rupture of fire protection piping in turbine building impacting offsite power[e] ~~Inadvertent safety injection (SI)~~	1.8×10^{-7} ~~2.5×10^{-7}~~	2
Loss of Train B Essential AC Power (4 kV Bus E6) ~~Small LOCA~~	1.6×10^{-7} ~~1.9×10^{-7}~~	1
Large flood, rupture of SW common return piping in PAB[e] ~~Reactor trip with no condenser cooling~~	1.4×10^{-7} ~~1.7×10^{-7}~~	1
Large LOCA	3.4×10^{-7}	2
Other internal events[b]	1.6×10^{-6} ~~1.0×10^{-6}~~	13 ~~7~~
Total internal events CDF[ee]	7.8×10^{-6} ~~1.1×10^{-5}~~	64 ~~70~~
	Fire initiating event	
Fire in control room—power-operated relief valve (PORV) LOCA ~~Fire switchgear (SWGR) room B—loss of bus E6~~	3.6×10^{-7} ~~3.7×10^{-7}~~	3 ~~2~~
Fire in switchgear (SWGR) room B—loss of Bus E6~~Fire SWGR room A—loss of bus E5~~	3.5×10^{-7} ~~3.7×10^{-7}~~	3 ~~2~~

Fire SWGR room A—loss of Bus E5~~Fire control room—AC power loss~~	3.1×10^{-7} ~~2.1×10^{-7}~~	2 ~~1~~
Fire control room—AC power~~-operated relief valve (PORV) LOCA~~ loss	1.8×10^{-7} ~~1.4×10^{-7}~~	1
Other fire events[c~~d~~]	3.8×10^{-7} ~~2.3×10^{-7}~~	2
Total fire events CDF~~[e]~~	**1.4×10^{-6}** ~~1.3×10^{-6}~~	**11** ~~9~~
Seismic initiating event		
Seismic 0.7 g transient event	9.3×10^{-7} ~~9.2×10^{-7}~~	8 ~~6~~
Seismic 1.0 g transient event	8.9×10^{-7} ~~8.7×10^{-7}~~	7 ~~6~~
Seismic 1.4 g transient event	3.6×10^{-7}	3 ~~2~~
~~Seismic 1.0 g ATWS~~	~~1.1×10^{-7}~~	~~1~~
~~Seismic 1.4 g large LOCA~~	~~1.1×10^{-7}~~	~~1~~
~~Seismic 0.7 g ATWS~~	~~1.0×10^{-7}~~	~~1~~
~~Seismic 1.0 g large LOCA~~	~~8.9×10^{-8}~~	~~1~~
Other seismic events[d~~f~~]	8.8×10^{-7} ~~4.9×10^{-7}~~	7 ~~3~~
Total seismic events CDF~~[e]~~	**3.1×10^{-6}**	**25** ~~21~~
Total CDF (internal and external events)~~[g]~~	**1.2×10^{-5}** ~~1.5×10^{-5}~~	**100**

[a] ~~May~~**Individual percent contributions may** not ~~total~~**sum exactly** to ~~100 percent~~**subtotals** due to round off.

[b] Obtained by subtracting the sum of the internal initiating event contributors to internal event CDF from the total internal events CDF.

~~[c] Obtained from percentage contribution of internal events provided in response to RAI 1.b.1 (NextEra, 2011a) times the total internal and external events CDF~~

~~[d]~~ [c] Obtained by subtracting the sum of the fire initiating event contributors to fire event CDF from the total fire events CDF.

~~[e] Provided in response to conference call clarification #2 (NRC, 2011a)~~

~~[f]~~ [d] Obtained by subtracting the sum of the seismic initiating event contributors to seismic event CDF from the total seismic events CDF.

~~[g] Provided in response to RAI 1.b.1 (NextEra, 2011a)~~ [e] **NextEra explained in response to an RAI the difference in the frequencies reported for** many **initiating events for the 2006 and 2011 PRA models. The total internal events CDF in the 2011 model decreased slightly as a result of model enhancements, the internal flooding CDF increased as results of a more detailed flooding analysis, and the severe weather CDF decreased primarily due to the incorporation of more recent data (NextEra ~~,~~ 2012b).**

Per the 2012 SAMA supplement (NextEra 2012a), the quantified CET sequences are binned into a set of **21** ~~14~~ release categories, which are subsequently grouped into **13** ~~10~~ source term categories that provide the input to the Level 3 consequence analysis (NextEra **2012a**). The frequency of each source term category was obtained by summing the frequency of the individual accident progression CET endpoints, or release categories, assigned to each source term category. Source terms were developed using the results of MAAP Version 4.0.**7** computer code calculations (NextEra **2012a).**

The offsite consequences and economic impact analyses use the MACCS2 code to determine the offsite risk impacts on the surrounding environment and public. Inputs for these analyses include plant-specific and site-specific input values for core radionuclide inventory, source term and release characteristics, site meteorological data, projected population distribution within an 80-km (50-mi) radius for the year 2050, emergency response evacuation planning, and economic parameters. The core radionuclide inventory corresponds to the end-of-cycle values for Seabrook operating at 3,659 MWt, which is slightly above the current licensed power level of 3,648 MWt. The magnitude of the onsite impacts (in terms of cleanup and decontamination costs and occupational dose) is based on information provided in NUREG/BR-0184 (NRC 1997a).

In the 2012 SAMA supplement (NextEra 2012a), NextEra estimated the dose to the population within 80 km (50 mi) of the Seabrook site to be approximately 0.**378** person-Sievert (Sv) (**37.8** person-rem) per year. The breakdown of the total population dose by containment release mode is summarized in **Table F-2, below, and in** Table F-2 **of the SAMA supplement** (NextEra **2012a). The** large late releases are the dominant contributors to population dose risk at Seabrook.

Table F-2. Breakdown of population dose by containment release mode

Containment release mode	Population dose (person-rem[a] per year)	% contribution
Small early releases	**1.7** ~~5.3~~	**5** ~~49~~
Large early releases	**1.7**~~1.6~~	**4** ~~15~~
Large late releases[b]	**34.4** ~~3.8~~	**91** ~~36~~
Intact containment	negligible	negligible
Total	**37.8** ~~10.7~~	**100**

[a] One person-rem = 0.01 person-Sv

[b] **Includes small early containment penetration failure to isolate and large late containment basemat failure (SELL).**

F.2.2 Review of NextEra's Risk Estimates

NextEra's determination of offsite risk at Seabrook is based on the following major elements of analysis:

- the Level 1 and 2 risk models that form the bases for the 1991 IPE submittal (NHY 1991) and the external event analyses of the 1992 IPEEE submittal (NAESC 1992),
- the major modifications to the IPE and IPEEE models that have been incorporated in the Seabrook PRA, including a complete revision of the Level 2 risk model, and
- the MACCS2 analyses performed to translate fission product source terms and release frequencies from the Level 2 PRA model into offsite consequence measures (essentially this equates to a Level 3 PRA).

Each of these analyses was reviewed to determine the acceptability of the Seabrook risk estimates for the SAMA analysis, as summarized below.

The first Seabrook PRA was completed in December 1983, its purpose being to provide a baseline risk assessment and an integrated plant and site model for use as a risk management tool. This model was subsequently updated in 1986, 1989, and 1990, with the last update used to support the IPE.

The NRC staff's review of the Seabrook IPE is described in an NRC report dated March 1, 1992 (NRC 1992). Based on a review of the original IPE submittal and responses to RAIs, the NRC staff concluded that the IPE submittal met the intent of generic letter (GL) 88-20 (NRC 1988). That is, the applicant demonstrated an overall appreciation of severe accidents, had an understanding of the most likely severe accident sequences that could occur at Seabrook, and had gained a quantitative understanding of core damage and fission product release. Although no severe accident vulnerabilities were identified in the Seabrook IPE, 14 potential plant improvements were identified. Four of the improvements have been implemented. Each of the 10 improvements not implemented is addressed by a SAMA in the current evaluation and is discussed further in Section F.3.2.

The internal events CDF value from the 1991 Seabrook IPE (6.1×10^{-5} per year) is near the average of the range of the CDF values reported in the IPEs for Westinghouse four-loop plants. Figure 11.6 of NUREG-1560 shows that the IPE-based internal events CDF for these plants range from about 3×10^{-6} per year to 2×10^{-4} per year, with an average CDF for the group of 6×10^{-5} per year (NRC 1997b). It is recognized that plants have updated the values for CDF subsequent to the IPE submittals to reflect modeling and hardware changes. Based on CDF values reported in the SAMA analyses for license renewal applications (LRAs), the internal events CDF result for Seabrook used for the SAMA analysis (**7.8×10^{-6}** ~~1.1×10^{-5}~~ per year, including internal **and external** flooding) is somewhat lower than that for most other plants of similar vintage and characteristics.

There have been **11**~~10~~ revisions to the IPE model since the 1991 IPE submittal, and **4** ~~3~~ revisions to the PRA model, as discussed previously, from the original 1983 PRA model to the 1990 update used to support the IPE submittal. The SSPSS-2006 model was used for the SAMA analysis **presented in the ER (NextEra 2010) but was updated by the SSPSS-2011 model used in the 2012 SAMA supplement (NextEra 2012a).** A listing of the major changes in each revision of the PRA, and the associated change in internal and external event CDF, was provided in the ER (NextEra 2010) in response to an NRC staff RAI (NextEra 2011a)**, in the 2012 SAMA supplement (NextEra 2012a),** and is summarized in Table F-3. A comparison of the internal events CDF between the 1991 IPE and the **2011** PRA model used for the **2012** SAMA **supplement** indicates a decrease of approximately **87** ~~82~~ percent (from 6.1×10^{-5} per year to **7.8×10^{-6}**~~1.1×10^{-5}~~ per year). This decrease results from the significant changes shown, while the external events CDF has increased by approximately 25 percent since the 1993 IPEEE (from 3.6×10^{-5} per year to 4.5×10^{-5} per year).

Table F-3. Seabrook PRA historical summary

PRA version	Summary of significant changes from prior model[a]	Total CDF (per year)	Internal events CDF (per year)[b]	External events CDF (per year)[b]
SSPSA-PLG-0300 (1983)	Original model—includes internal, fire, and seismic events	2.3×10^{-4}	1.8×10^{-4}	4.6×10^{-5}
SSPSS-1986	• Updated allowed outage times to reflect current technical specifications • Revised models of the inservice test pump test frequency; turbine driven emergency feedwater (EFW) pump atmospheric relief valves; boron injection tank, pump, and lines; enclosure building air handling system; reactor trip breakers; & reactor coolant pump (RCP) thermal barrier core spray (CS)	2.9×10^{-4}	Not provided	Not provided

PRA version	Summary of significant changes from prior model[a]	Total CDF (per year)	Internal events CDF (per year)[b]	External events CDF (per year)[b]
	• Improved quantification traceability & documentation • Updated seismic fragilities • Expanded common cause treatment			
SSPSS-1989	• Updated initiating event frequencies • Updated common cause & maintenance distributions • Revised electric power recovery model using current data • Added recovery actions into event model	1.4×10^{-4}	$9.5\text{x}10^{-5}$	$4.5\text{x}10^{-5}$
SSPSS-1990	IPE submittal • Added modeling of ATWS mitigation system • Updated electric power recovery model • Updated RCP seal LOCA analysis • Added new recovery actions • Revised CET to explicitly model induced SGTR & direct containment heating	1.1×10^{-4}	6.1×10^{-5}	5.0×10^{-5}
SSPSS-1993	IPEEE submittal • Added plant-specific data for main safety pumps & diesel generators (DGs) • Improved fire event modeling, including modeling operator actions & addition of new fire hazard initiating events • Revised startup feed pump (SUFP) model to conservatively require manual startup • Improved modeling of high-pressure injection (HPI) and event tree logic	8.0×10^{-5}	4.4×10^{-5}	3.6×10^{-5}
SSPSS-1996	• Improved common cause modeling of primary component cooling (PCC) with opposite PCC train failure • Updated ATWS model to account for change from an 18-month to 24-month fuel cycle • Increased use of plant-specific data • Changed definition of LERF to include steam leak from SGTR • Increased failure likelihood for small containment penetrations in seismic sequences • Added credit for manual operator action to close RCP seal return line motor-operated valve (MOV)	4.3×10^{-5}	2.1×10^{-5}	2.2×10^{-5}
SSPSS-1999	• Updated LOCA initiator frequencies • Updated ATWS model to account for change from a 24-	4.6×10^{-5}	2.7×10^{-5}	1.9×10^{-5}

PRA version	Summary of significant changes from prior model[a]	Total CDF (per year)	Internal events CDF (per year)[b]	External events CDF (per year)[b]
	month to an 18-month fuel cycle & to use more current failure rates • Updated event tree to explicitly incorporate RCP seal LOCA model & related power recovery models • Changed emergency diesel generator (EDG) mission time from 6 hours to 24 hours for weather-related LOOP & similar initiators • Moved LOOP & internal flooding models from external to internal events model • Modified common cause factors & mission times for PCC system & SWS • Updated human error probability (HEP) event tree rules & quantification			
SSPSS-2000	• Transitioned PRA software from DOS-based RISKMAN 9.2 to Windows-based RISKMAN 3.0	4.6×10^{-5}	2.7×10^{-5}	1.9×10^{-5}
SSPSS-2001	• Changed system initiator models	4.8×10^{-5}	2.8×10^{-5}	2.0×10^{-5}
SSPSS-2002	• Integrated shutdown & low power risk models into all-modes model	4.8×10^{-5}	2.5×10^{-5}	2.0×10^{-5}
SSPSS-2004	• Updated the human reliability analysis (HRA) • Added credit for the supplemental electric power system (SEPS) DG • Updated the LERF model to include consequential SGTR	3.0×10^{-5}	1.7×10^{-5}	1.3×10^{-5}
SSPSS-2005	• Revised success criteria & operator timing • Updated the seismic PRA • Updated DG failure rate & unavailability data • Updated the Level 2 analysis including modeling of severe accident management guideline (SAMG) actions	1.4×10^{-5}	9.5×10^{-6}	4.5×10^{-6}
SSPSS-2006~~[c]~~	• Updated the Mode 4, 5, & 6 shutdown model • Revised modeling of PCC & SWS initiators	1.5×10^{-5}	1.1×10^{-5}	4.5×10^{-6}
SSPSS-2009	• Updated plant-specific data & generic data distributions • Incorporated electric power convolution model • Expanded the steam generator (SG) model to include condenser cooling, circulating water, & condenser steam dump • Revised operator action modeling	1.2×10^{-5}	7.1×10^{-6}	4.9×10^{-6}
SSPSS-2011[c]	• **Updated the internal flood model to incorporate plant changes, EPRI data and guidance, and to meet current PRA standards for internal flooding[d]**	**1.2×10^{-5}**	**7.8×10^{-6}**	**4.5×10^{-6}**

PRA version	Summary of significant changes from prior model[(a)]	Total CDF (per year)	Internal events CDF (per year)[(b)]	External events CDF (per year)[(b)]
	• **Revised release category and source term based on more detailed modeling using MAAP 4.0.7** • **Added new breakers and buses to reflect a switchyard upgrade**			

[(a)] Summarized from information provided in the ER and in response to an NRC staff RAI (NextEra 2011a).

[(b)] Estimated from percent contribution to total CDF provided in response to an NRC staff RAI (NextEra 2011a).

[(c)] PRA model revision used in the **2012** SAMA **supplement (NextEra 2012a).**

[(d)] NextEra confirmed in response to an RAI that flow orifices installed in the plant and credited in the internal flooding model passed startup acceptance testing (NextEra 2012b).

The NRC staff considered the peer reviews performed for the Seabrook PRA and the potential impact of the review findings on the SAMA evaluation. In the ER (NextEra 2010), NextEra identifies two peer reviews that have been performed on the PRA—a 1999 Westinghouse Owner's Group (WOG) certification peer review and a 2005 focused peer review against the American Society of Mechanical Engineers (ASME) PRA standard (ASME 2003). **The 2012 SAMA supplement (NextEra 2012a) identifies an additional peer review—a 2009 peer review of the internal flood model against the ASME PRA standard (ASME 2009). There were no Category A facts and observations (F&Os) from that 2009 focused peer review, and the three Category B F&Os were addressed in the SSPSS-2011 PRA model update.** In response to an NRC staff RAI, NextEra clarified the scope of these **1999 and 2005** peer reviews**. The** 1999 review provided a full review of the technical elements of the Level 1 and 2 LERF internal events models, including internal flooding and the 2005 peer review providing a focused scope examination of Level 1 internal events accident sequences, success criteria, post-initiating event HRA, and configuration control (NextEra 2011a). Neither the 1999 nor the 2005 peer review included examination of external flooding, fire, or seismic hazards. The 1999 certification peer review identified 30 Category A and B F&Os, and the 2005 focused peer review identified 4 Category A and B F&Os.[2] The applicant provides the resolution of each of the 34 F&Os in the ER and states that all have been dispositioned and implemented in the PRA model.

The NRC staff requested that NextEra clarify how the resolution to F&O 3 (aggressive load shedding and the available cross tie can extend battery life from 8 to 12 hours) addresses the F&O. The NRC asked NextEra to assess the ability of the operators to successfully cool the core using the EFW pump without underfeeding the SGs (NRC 2010a). In response to the RAI, NextEra clarified that during an extended SBO condition, the normal control instrumentation and procedures for which operators are trained and with which they are familiar would be used to maintain long-term control of SG water level (NextEra 2011a).

The NRC staff asked NextEra to summarize the scope and unresolved findings from any other reviews performed on the Seabrook PRA (NRC 2010a). In response to the RAI, NextEra explained that many other internal reviews—including vendor-assisted reviews—have been

[2] Now termed a "Finding," a Category A or B F&Os is an "observation (an issue or discrepancy) that is necessary to address to ensure: [1] the technical adequacy of the PRA ... [2] the capability/robustness of the PRA update process, or [3] the process for evaluating the necessary capability of the PRA technical elements (to support applications)." (NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, 2008)

performed on specific model updates, and comments from these reviews—along with plant changes and potential model enhancements—are tracked through a model change database to assure that the comments are addressed in the periodic update process (NextEra 2011a). **NextEra specifically explains in the 2012 SAMA supplement (NextEra 2012a) that the source term analysis was performed by the PRA group and reviewed by industry experts from a vendor, and the Level 3 model was prepared by experts from a vendor and independently reviewed**.

The NRC staff asked NextEra to identify any changes to the plant, including physical and procedural modifications, since the SSPSS-2006 PRA model that could have a significant impact on the results of the SAMA analysis (NRC 2010a). In response to the RAI, NextEra stated that there have been no major plant changes since PRA model SSPSS-2006 was issued that could significantly impact the SAMA analysis **but did identify** specific plant and model changes made ~~to the PRA model~~ that resulted in the 2009 periodic update of the model, referred to as PRA model SSPSS-2009 (NextEra 2011a). NextEra explained that the model changes resulted in a total CDF decrease of about 19 percent **but** resulted in no significant shift in the relative importance of initiating events or components. **Since then,** NextEra **has updated** the **SSPSS-**2011 **PRA model, which uses source estimates based on more detailed MAAP modeling and meets** the internal flooding requirements in the ASME PRA standard (ASME 2009**). The 2012 SAMA supplement (NextEra 2012a) is based on the SSPSS-2011 model** and **calculates an increase in the CDF, compared to** the SSPSS**-2009 model, by about 5 percent.**

The NRC staff asked NextEra to describe the PRA quality control process used at Seabrook (NRC 2010a). NextEra responded that an existing administrative procedure defines the quality control process for updates to the Seabrook PRA, and the process is consistent with requirements of the ASME 2009 PRA standard (ASME 2009) and ensures that the PRA model accurately reflects the current Seabrook plant design, operation, and performance (NextEra 2011a). The quality control process includes monitoring PRA inputs for new information, recording new applicable information, assessing the significance of new information, performing PRA revisions, and controlling computer codes and models. NextEra also stated that the PRA training qualification is performed as part of the Engineering Support Personnel Training Program.

Given that the Seabrook internal events PRA model has been peer-reviewed and the peer review findings were all addressed, and that NextEra has satisfactorily addressed NRC staff questions regarding the PRA, the NRC staff concludes that the internal events Level 1 PRA model is of sufficient quality to support the SAMA evaluation.

The Seabrook PRA model is an integrated internal and external events model in that it includes seismic-initiated, fire-initiated, and external flooding-initiated events as well as internal initiating events. The external events models have been integrated with the internal events model since the initial 1983 PRA (NextEra 2011a). The external events models used in the SAMA evaluation are essentially those used in the IPEEE, with the exception of the seismic PRA model, which underwent a major update for the SSPSS-2005 model. The updated external events CDF results are described in a response to an NRC staff RAI (NextEra 2011a) and are included in Table F-3 along with the internal events results.

The Seabrook IPEEE was submitted October 2, 1992 (NAESC 1992), in response to Supplement 4 of GL 88-20 (NRC 1991). The submittal used the same PRA as was used for the IPE (i.e., SSPSS-1990) except for updates to the external events. No fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events were identified. Improvements that have already been realized as a result of the IPEEE process minimized the likelihood of there being cost-beneficial enhancements as a result of the SAMA analysis, especially with the inclusion of a multiplier to account for the additional risk of seismic events. In a letter dated May 2, 2001, the NRC staff concluded that the submittal met the intent of Supplement 4 to GL 88-20 and the applicant's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities (NRC 2001).

The Seabrook IPEEE seismic analysis used a seismic PRA following NRC guidance (NRC 1991a). The seismic PRA included a seismic hazard analysis, a seismic fragility assessment, seismic quantification to yield initiating event frequencies and conditional system failure probabilities, and plant model assembly to integrate seismic initiators and seismic-initiated component failures with random hardware failures and maintenance unavailabilities.

The seismic hazard analysis estimated the annual frequency of exceeding different levels of ground motion. Seabrook seismic CDFs were determined for site-specific, Electric Power Research Institute (EPRI) (EPRI 1989) and Lawrence Livermore National Laboratory (LLNL) (NRC 1994) hazard curves. The seismic fragility assessment was performed by walkdowns that were conducted at the time of the original seismic PRA in 1982 through 1983, walkdowns performed for a revised fragility analysis in 1986, and supplemental walkdowns performed in 1991 for the IPEEE, using procedures and screening caveats in EPRI's seismic margin assessment methodology (EPRI 1988). Fragility calculations were made for about 82 components using a screening criterion of median peak ground acceleration of 2.0 g, which corresponds to a high confidence (95 percent) low probability (5 percent) of failure (HCLPF) capacity. A total of 15 components and 2 sets of relay groups were further assessed. Fragility calculations were also made for eight buildings and structures, and HCLPF values were determined. The seismic systems analysis defined the potential seismic induced structure and equipment failure scenarios that could occur after a seismic event and lead to core damage. The Seabrook IPE event tree and fault tree models were used as the starting point for the seismic analysis. Quantification of the seismic models consisted of convoluting the seismic hazard curve with the appropriate structural and equipment seismic fragility curves to obtain the frequency of the seismic damage state. The conditional probability of core damage, given each seismic damage state, was then obtained from the IPE models with appropriate changes to reflect the seismic damage state. The CDF was given based on the product of the seismic damage state probability and the conditional core damage probability.

Quantification of the seismic CDF for Seabrook was performed in nine discrete ground acceleration ranges between 0.1 g to 2.0 g. The seismic CDF resulting from the Seabrook IPEEE was calculated to be 1.2×10^{-5} per year using a site-specific seismic hazard curve, with sensitivity analyses yielding 1.3×10^{-4} per year using the LLNL seismic hazard curve and 6.1×10^{-6} per year using the EPRI seismic hazard curve. The Seabrook IPEEE did not identify any vulnerability due to seismic events but did identify two plant improvements to reduce seismic risk. Neither of the two improvements has been implemented. Each of the two improvements is addressed by a SAMA in the current evaluation and is discussed further in Section F.3.2.

Subsequent to the IPEEE, NextEra updated the seismic PRA analysis. The NRC staff asked NextEra to describe the changes to the seismic analysis incorporated in the PRA model SSPSS-2005 update and to explain the reasons for any significant changes to the seismic CDF (NRC 2011a). In response to the RAI, NextEra stated that the most significant changes to the

IPEEE seismic model made in the SSPSS-2005 update of the Seabrook PRA were as follows (NextEra 2011a):

- The fragility analysis was updated to extend the fragility screening of equipment from greater than 2.0 g to the range from 2.0 g to 2.5 g and greater than 2.5 g to better capture seismic risk.
- The EPRI hazard curve was adopted and used to update the equipment fragilities. The site-specific hazard curve was replaced with the EPRI hazard curve because the EPRI uniform hazard spectrum (UHS) developed for the Seabrook site is more current and realistic than that used in the original 1983 and the IPEEE PRA. In response to a followup NRC staff RAI, NextEra further clarified that the EPRI UHS was judged to be more realistic and representative of the best estimate hazard because of overall general improvement in seismic technology from the early 1980s to 1989, when the EPRI hazard curve was developed (NextEra 2011b). The probabilistic estimates of seismic capacity of structures and components were updated to reflect component-specific fragility information and the EPRI UHS.
- Several new component fragilities were added to the seismic PRA model, including seismic fragilities for the SEPS DGs, which had been added to the plant since the IPEEE.
- Modeling and documentation of operator actions credited in the seismic PRA were improved.

NextEra stated that the most recognizable conservatism in the seismic model is the use of complete correlation of the fragility between identical components, such as both EDGs are assumed to fail at the same seismic hazard level (NextEra 2011a). NextEra further stated that extensive internal technical reviews of the seismic PRA analysis were performed for the original 1983 PRA, when the seismic analysis was revised for the IPEEE, and when the seismic analysis was revised for the SSPSS-2005 PRA model update. No significant comments were documented from these reviews, and no formal peer reviews have been conducted on the seismic PRA model (NextEra 2011a).

The NRC staff noted that, in the attachments to NRC Information Notice 2010-18, generic issue (GI) 199 (NRC 2010b), the NRC staff estimated a seismic CDF for Seabrook of between 5.9×10^{-6} per year and 2.2×10^{-5} per year using updated seismic hazard curves developed by the U.S. Geological Survey (USGS) in 2008 (USGS 2008). The NRC staff asked that NextEra provide an assessment of the impact of the updated USGS seismic hazard curves on the SAMA evaluation (NRC 2010a). In response to the RAI, NextEra provided a revised SAMA evaluation ~~using multipliers of 2.1 and 2.6~~ to account for the maximum GI-199 seismic CDF of 2.2×10^{-5} per year, which is discussed further below (NextEra 2011a, 2011b). **The 2012 SAMA supplement (NextEra 2012a) uses a multiplier of 2.1 to account for a higher seismic hazard than assessed in the PRA.**

Considering the following points, the NRC staff concludes that the seismic PRA model, in combination with the use of a seismic events multiplier, provides an acceptable basis for identifying and evaluating the benefits of SAMAs:

- The Seabrook seismic PRA model is integrated with the internal events PRA.
- The seismic PRA has been updated to include additional components and to extend the fragility-screening threshold.

- The SAMA evaluation was updated using a multiplier to account for a potentially higher seismic CDF.
- NextEra has satisfactorily addressed NRC staff RAIs regarding the seismic PRA.

The Seabrook IPEEE fire analysis, which was significantly updated from the original fire analysis completed in 1983, employed EPRI's FIVE methodology (EPRI 1992) to calculate area fire frequencies, quantitatively screen areas, and provide hazards analysis for resulting critical areas. The quantification of CDF was obtained by propagating fire-induced initiating events through the PRA used for the IPE.

The IPEEE fire areas were based on definitions of Appendix R fire areas for Seabrook. Qualitative screening was performed using a spatial database specifically developed for the IPEEE fire analysis that identified equipment important in initiating or mitigating an accident. Of the 73 fire areas, 13 were determined to contain important equipment (pumps, valves, and cabling, etc.) and were further assessed. Quantitative screening used industry fire data and the assumption that a fire in a compartment damaged all equipment and cables in the compartment. The resulting fire-initiating events are propagated through the appropriate event tree models. Using fire frequencies and conditional core damage probabilities from the internal events PRA, all but eight fire areas were screened as contributing less than 1×10^{-6} per year to the CDF.

Based on the FIVE fire methodology analysis, the unscreened areas were assessed by considering possible targets, fire sources and combustibles, possible fire scenarios (e.g., target-in-plume), and detection and suppression systems to determine the probability of damage given a fire. Credit was explicitly taken for automatic and manual fire suppression. Calculation of automatic fire suppression unavailability was supported by fault tree modeling. Calculation of manual suppression unavailability was supported by HRA. Consideration of fires on containment performance was also addressed. Final quantification used the Seabrook IPE PRA model to determine plant responses and CDFs. The resulting fire-induced CDF was calculated to be 1.2×10^{-5} per year. While no physical plant changes were found to be necessary as a result of the IPEEE fire analysis, fire potential plant improvements to improve fire risk were identified. Four of the plant improvements have been implemented. The one improvement not implemented is addressed by a SAMA in the current evaluation and is discussed further in Section F.3.2.

NextEra updated the fire PRA subsequent to the IPEEE. The NRC staff asked NextEra to describe the changes to the fire analysis since the IPEEE and to explain the reasons for any significant changes to the fire CDF (NRC 2011a). In response to the RAI, NextEra explained that the most recent update of the fire PRA was in support of the **SSPSS**-2004 PRA update, and the fire analysis methodology used is essentially the same, with some variations, as that described previously for the IPEEE fire analysis (NextEra 2011a). Specific changes made to the Seabrook fire PRA since the IPEEE are listed below:

- including current plant data and procedures,
- performing detailed walkdowns to verify locations of the major fire sources and important targets,
- updating data to the EPRI fire database that includes fire records through December 2000,
- developing updated severity factors for cabinets, pumps, control room panels, and transients,

- revisiting the quantitative screening results,
- using new data on cabinet heat release rates, and
- quantitatively evaluating the total area heat-up rate.

NextEra stated that the most significant conservatism in the fire analysis is the assumption that small fires, typical of the generic fire events database, are assumed to grow to cause the maximum damage **(NextEra 2010).** However, because these fire sequences have such low frequencies and large uncertainties, NextEra claimed that the impact of this conservatism on the overall fire CDF is difficult to determine (NextEra 2011a). NextEra further stated that extensive internal technical reviews of the fire PRA analysis were performed for the original 1983 PRA, when the fire analysis was revised for the IPEEE and when the fire analysis was revised for the **SSPSS**-2005 PRA model update. No significant comments were documented from these reviews, and no formal peer reviews have been conducted on the fire PRA model (NextEra 2011a).

In a followup RAI, the NRC staff asked NextEra to clarify if fire-induced failures of components and human actions credited with mitigating the initiator were assessed and to describe how hot short probabilities were considered in the fire analysis (NRC 2011b). In response to the RAI, NextEra clarified that, for fire initiators that are not screened and are evaluated in detail, the probability of fire damage to components due to the fire is included in the analysis and that this probability is dependent upon the presence of combustible material and the success of suppression (NextEra 2011b). NextEra stated that the probability of additional failures needed for core damage was also evaluated, including unavailability of redundant systems and components and failure of operator actions, and component failures not impacted by the fire are modeled as random. Regarding the hot short probability question, NextEra explained that a hot short probability of 0.1 was used in the screening evaluation for important valves and components. NextEra also described the results of an evaluation to assess the sensitivity of the SAMA results to using a hot short probability of 0.6. This evaluation determined that the fire event screening evaluation is insensitive to this increase in the potential for hot shorts and that, while the contribution to CDF does increase due to the higher probability, the contribution compared to the CDF contribution of similarly modeled internal events remains relatively low. Specifically, NextEra evaluated 18 fire events and determined that 3 of the events contributed in the range of 10 to 20 percent of the corresponding internal events CDF, and the remaining 15 fire events contributed less than 10 percent. Based on this result, NextEra determined that the increase in hot short potential does not have a significant effect on the SAMA analysis (NextEra 2011b).

The NRC staff noted that the fire ignition frequencies for a fire in SWGR room B—Loss of Bus E6 and SWGR room A—Loss of Bus E5, which were reported to be about 1.0×10^{-3} per year each, appeared to be low unless the fire only involved the associated buses. The NRC staff asked that NextEra justify these values (NRC 2010a). NextEra responded that the ignition frequency for SWGR room B—Loss of Bus E6 includes the cumulative fire ignition frequencies for 21 Bus E6 cabinets and 170 other electrical cabinets. SWGR room A—Loss of Bus E5 similarly includes the cumulative fire ignition frequencies for 21 Bus E5 cabinets and 86 other electrical cabinets (NextEra 2011a). NextEra explained that the cited value of 1.0×10^{-3} per year was more than just “frequency” (i.e., it included not only fire ignition frequency of 4.6×10^{-5} per year per cabinet but also a severity factor of 0.2 and a manual non-suppression probability of 0.1 for fires in the other electrical cabinets). Therefore, the calculated total fire ignition frequency for each of the two SWGR rooms is the same as that reported in the ER. The NRC staff considers NextEra’s assumptions reasonable.

Considering that the Seabrook fire PRA model is integrated with the internal events PRA, that the fire PRA has been updated to include more current data, and that NextEra has satisfactorily addressed NRC staff RAIs regarding the fire PRA, the NRC staff concludes that the fire PRA model provides an acceptable basis for identifying and evaluating the benefits of SAMAs.

The Seabrook IPEEE analysis of high winds, tornadoes, external floods, and other (HFO) external events followed the screening and evaluation approaches specified in Supplement 4 to GL 88-20 (NRC 1991) and concluded that Seabrook meets the 1975 Standard Review Plan (SRP) criteria (NRC 1975). Two external event frequencies exceeded the 1.0×10^{-6} per year screening criterion (NAESC 1992). One of these events is flooding resulting from a storm surge caused by a hurricane, which is modeled in the PRA and described in the ER (NextEra 2010) as event EXFLSW in which the SW pumps are flooded. This sequence was reported in the ER to contribute just 2×10^{-8} per year to the total Seabrook CDF. The second event is an external initiating event involving a truck crash into the SF6 transmission lines. In response to an NRC staff RAI, NextEra explained that this event has been mitigated by the installation of jersey barriers and guard rails that further limit the possibility of a truck crash impacting the transmission lines and that, as a result, this initiating event has been screened from the PRA model (NextEra 2011a).

While no physical plant changes were found to be necessary as a result of the IPEEE HFO analysis, one plant improvement based on HFO analysis was recommended—modify several exterior doors so that they will be able to withstand the design pressure differential resulting from high winds. NextEra clarified in response to an NRC staff RAI that this suggested improvement has been implemented (NextEra 2011a).

The NRC staff noted that while the risk of flooding resulting from a storm surge caused by a hurricane is included in the PRA, the impact of hurricane-force winds does not appear to be addressed, and the staff requested that NextEra provide an assessment of the risk of this event on the Seabrook site (NRC 2010a). In response to the RAI, NextEra explained that the high winds associated with a hurricane that might accompany a storm surge are screened from consideration because the site design basis criteria for high winds and tornadoes meets the 1975 SRP criteria (NextEra 2011a). The NRC staff considered this explanation acceptable.

The Seabrook IPEEE submittal also stated that as a result of the Seabrook IPE, cost-benefit analyses are being performed for many potential plant improvements, which may also reduce external event risk because they address functional failures. Five potential plant improvements to improve internal event risk that may also reduce external event risk were identified. Four of the plant improvements have been implemented. The one improvement not implemented is addressed by a SAMA in the current evaluation and is discussed further in Section F.3.2.

NextEra estimated the benefits for both internal and external events using the integrated Seabrook PRA model. However, as discussed previously, an NRC staff assessment of the USGS 2008 seismic hazard curves yielded an upper bound seismic CDF for Seabrook of 2.2×10^{-5} per year, which is substantially greater than the 3.1×10^{-6} per year seismic CDF used in the SAMA evaluation. The NRC staff requested that NextEra provide an assessment of the impact of this higher seismic CDF on the SAMA evaluation (NRC 2010a, 2011b). In response to the RAIs, NextEra noted that the NRC staff's estimate of the seismic CDF using the USGS 2008 seismic hazard curves did not include credit for the SEPS DGs installed at Seabrook in 2004, which have a median seismic fragility of 1.23 g (NextEra 2011b). NextEra stated that the SEPS DGs were modeled in the Seabrook seismic PRA in 2005 and reduced the seismic CDF by approximately 26 percent by avoiding SBO sequences, and a corresponding reduction in the

NRC staff estimate of the seismic CDF using the USGS 2008 seismic hazard curves to 1.6×10^{-5} per year would be expected. NextEra also provided a sensitivity analysis using a multiplier of 2.1 to account for the revised higher seismic CDF. This multiplier is based on an increased seismic CDF of 1.3×10^{-5} per year (upper bound seismic CDF of 1.6×10^{-5} per year minus seismic CDF of 3.1×10^{-6} per year used in the SAMA evaluation) and a total estimated CDF of 1.2×10^{-5} per year for PRA model **SSPSS**-2009 (NextEra 2011b). The NRC staff **agrees** that a seismic CDF of 1.6×10^{-5} per year for Seabrook is reasonable and agrees that the applicant's use of a multiplier of 2.1**, which was used in the 2012 SAMA supplement (NextEra 2012a),** to account for the additional risk from seismic events is reasonable for the purposes of the SAMA evaluation. This is discussed further in Section F.6.2.

The NRC staff reviewed the general process used by NextEra to translate the results of the Level 1 PRA into containment releases, as well as the results of the Level 2 analysis, as described in the ER and in response to NRC staff RAIs (NextEra, 2011a). The Level 2 model was significantly revised in the 2005 PRA update (i.e., PRA model SSPSS-2005) from that used in the IPE and reflects the Seabrook plant as designed and **currently** operated. In response to an NRC staff RAI **(NextEra 2010),** NextEra identified the following major changes to the PRA that most impacted the LERF (NextEra 2011a):

- change in definition of LERF to include steam leak from a SGTR,
- higher failure likelihood for small containment penetrations in seismic sequences,
- update to credit manual operator action to close the RCP seal return line MOV,
- expansion of the LERF model by adding a steam line break to SGTR and consideration of ATWS sequences,
- updates to the Level 2 analysis to reflect current state of knowledge including SAMGs,
- revisions to incorporate plant-specific data,
- update of data distributions, and
- revisions to operator action modeling.

No Level 2 design or plant changes were identified in the 2012 SAMA supplement (NextEra 2012a).

In response to an NRC staff RAI, NextEra explained that the quantification of the Level 1 and Level 2 models is done using a linked event tree method approach and does not employ plant damage states (NextEra 2011a). Therefore, all Level 1 sequences are evaluated by the CET, making it unnecessary to summarize and group similar sequences into Level 1 plant damage states before they are input to the CET. The Level 2 model is a single CET and evaluates the phenomenological progression of all the Level 1 sequences including internal, fire, and seismically initiated events. In response to another NRC staff RAI, NextEra clarified that the CET has 37 branching events, which include 10 hardware-related, 13 human action-related, and 13 phenomena-related events, along with a single mapping event (NextEra 2011a). CET branch point split fraction numerical values are determined based on the type of event. The CET event success criterion is defined, and split fraction logic rules are used to apply the correct event split fraction values during CET quantification. Included in the response to the NRC staff RAI, NextEra provided a description of each of the 37 CET branching events. End states resulting from the combinations of the branches are then assigned to one of 16 release categories based on characteristics that determine the timing and magnitude of the release, whether or not the containment remains intact, and isotopic composition of the released material. In response to another NRC staff RAI, NextEra clarified that the frequency of each release category was obtained by summing the frequency of the individual accident progression CET end states binned into the release category (NextEra 2011a).

The quantified CET sequences binned into the **21**~~16~~ release categories are subsequently grouped into **13**~~10~~ source term categories that provide the input to the Level 3 consequence analysis **(NextEra 2012a).** In response to an NRC staff RAI, NextEra explained that the ~~16~~ release categories were reduced to~~ 10~~ source term categories by grouping release categories that occur due to different phenomena, but the consequence is essentially the same (e.g., thermally induced SGTR and pressure-induced SGTR) (NextEra 2011a). **Eight of the release categories were mapped one-to-one into a corresponding source term category. For three of the source term categories, three release categories were binned together to form the combined source term category, and for** two of the source term categories, two release categories were binned together to form the combined source term category.

Source terms were developed for each of the source term categories. In **the 2012 SAMA supplement**, NextEra **explains** that the release fractions and timing for source term categories are based on the results of plant-specific calculations using the MAAP Version 4.0.**7** and **represent more realism and an upgrade from** the source **terms presented in** the **ER** (NextEra **2010**). NextEra generally selected the representative MAAP case based on that which resulted in the most realistic timing and source term release. In **four of** the **combined source term categories, the source term** for the **release category having the highest (dominant)** release **frequency was used as the source term** for the **combined category. The consequences from the contributors were considered similar. In one of the four categories, the total frequency was very low (approximately 1E-9 per year). In the fifth combined source term category (i.e., SELL), one of the contributors had the most significant source term and the highest frequency so it was selected as the representative case.** The source term categories and their frequencies and release characteristics are presented in **tables on pages 12, 13,** and **18** of the **2012 SAMA supplement** (NextEra **2012a**).

As indicated above, the current Seabrook Level 2 PRA model is an update of that used in the IPE. The IPE did not identify any severe accident vulnerabilities associated with containment performance. Risk-related insights and improvements discussed in the IPE submittal were discussed previously. The NRC staff review of the IPE back-end (i.e., Level 2) model concluded that it appeared to have addressed the severe accident phenomena normally associated with large dry containments, it met the IPE requirements, and there were no obvious or significant problems or errors.

The LERF model was included in the 1999 industry peer review discussed previously. Seven of the F&Os from this review addressed the LERF analysis. The applicant provides in the ER the resolution of each of the seven F&Os and states that all have been dispositioned and implemented in the PRA model. NextEra noted that the Seabrook radiological source terms were significantly revised for the SSPSS-2005 PRA model based on Level 2 analysis by Westinghouse Electric Company. In addition, NextEra noted that the source terms were further revised during the SSPSS-2011 PRA model and are reflected in the 2012 SAMA supplement (NextEra 2012a).

The NRC staff noted that the LERF reported for Seabrook is less than 1 percent of the CDF and asked NextEra to explain this apparently very low LERF (NRC 2010a). In response to the RAI, NextEra explained that Seabrook has a very large-volume and strong containment building in comparison to most other nuclear power plant containment designs (NextEra 2011a). As a result of the containment design median failure pressure of 187 pounds per square inch absolute (psia) (dry) and 210 psia (wet), there are no conceivable severe accident progression scenarios that result in catastrophic failure early in the accident sequence. The NRC staff considers NextEra's explanation reasonable.

The NRC staff requested that NextEra explain how fire-induced interfacing system loss-of-coolant accidents (ISLOCAs) and fire-induced containment impacts are addressed in the fire analysis (NRC 2010a, 2011b). In response to the RAIs, NextEra explained that containment performance was evaluated in three areas: (1) containment structure, (2) containment response to a core damage event, and (3) containment isolation failure (NextEra 2011a). Fires were determined to have no impact on containment structure integrity. Fire-initiated core damage events were determined to have the same impact on containment response as internal-initiated events; thus, they are handled through the CET. The potential for containment isolation failure was assessed by evaluating the potential for fire-induced failure of important isolation valves, as follows:

- Because the containment isolation valves (CIVs) are located both inside and outside containment, NextEra concluded that only a fire in the control room or cable spreading room could affect CIVs both inside and outside containment and that, in this event, important CIVs could be controlled locally at the valve or from the remote shutdown panel (RSP). CIVs located outside containment could be controlled both locally at the valve and from the RSP, CIVs located inside containment could be controlled from the RSP, and no credit is taken for local control of valves inside containment (NextEra 2011b).
- Because the letdown system has three normally open, air-operated valves (AOVs) in series, NextEra concluded that hot shorting in all three valves is not credible. NextEra clarified that failure to isolate the letdown system for an extended period of time is judged to not be credible for the following reasons (NextEra 2011b):
  - There are three AOVs inside containment and one AOV outside containment.
  - All four AOVs fail to the closed position upon loss of air or control power.
  - Shorts to ground in the control cables for these AOVs will also result in the AOVs failing to the closed position.
  - There are two MOVs inside containment that are available to provide isolation.
- The potential for fire-induced failures of several other potential isolation pathways was also evaluated (e.g., large residual heat removal (RHR) suction line MOVs, RCP seal return line isolation valves, and containment on-line purge valves) and determined to not be credible.

Based on the information above, NextEra concluded that the only credible impact of fires on containment performance is to fail a single train of isolation. For isolation failure of one or more valves in a single train, either redundant isolation would be available or the ability to remove power from fail closed valves to provide isolation is available (NextEra 2011a). NextEra further clarified that, since Seabrook is designed with divisional cable separation, power to the fail closed valves can be removed, if necessary, by removing its divisional power supply, thus ensuring that the valves fail closed and are prevented from being failed opened due to hot shorting (NextEra 2011b). NextEra further concluded that the frequency of fires that could cause this level of damage is sufficiently low compared to hardware failures that this scenario does not contribute significantly to containment isolation failure and that, as a result, no fire impacts on containment isolation components are included in the PRA (NextEra 2011a).

Based on the NRC staff's review of the Level 2 methodology, the NRC staff concludes that NextEra has adequately addressed NRC staff RAIs, that the LERF model was reviewed in more detail as part of the 1999 WOG certification peer review, and that all F&Os have been resolved. Therefore, the NRC staff concludes that the Level 2 PRA provides an acceptable basis for evaluating the benefits associated with various SAMAs.

As indicated in the ER, the reactor core radionuclide inventory used in the consequence analysis corresponds to the end-of-cycle values for Seabrook operating at 3,659 MWt. This bounds the current Seabrook rated power of 3,648 MWt. The core radionuclide inventory is provided in Table F.3.4.3-1 of Appendix F of the ER (NextEra 2010). In response to an NRC staff RAI, NextEra clarified that a Seabrook-specific core inventory was calculated using ORIGEN2.1 except for Cobalt-58 and Cobalt-60 (NextEra 2011a). NextEra noted that the ORIGEN calculations did not provide isotopic inventories for Cobalt-58 and Cobalt-60. Therefore, these isotope inventories were estimated using the MACCS2 sample problem inventory corrected by the ratio of Seabrook's power level to the MACCS2 sample problem A power level (i.e., 3,659 MWt/3,412 MWt). Based on this clarification, the NRC staff concludes that the reactor core radionuclide inventory assumptions for estimating consequences are reasonable and acceptable for purposes of the SAMA evaluation.

The NRC staff reviewed the process used by NextEra to extend the containment performance (Level 2) portion of the PRA to an assessment of offsite consequences (essentially a Level 3 PRA). This included consideration of the source terms used to characterize fission product releases for the applicable containment release categories and the major input assumptions used in the offsite consequence analyses. Version 1.13.1 of the MACCS2 code was used to estimate offsite consequences (NRC 1998) **based on the results of the SSPSS-2011 PRA model (NextEra 2012a).** Plant-specific input to the code includes the source terms for each release category and the reactor core radionuclide inventory (both discussed above), site-specific meteorological data, projected population distribution within an 80-km (50-mi) radius for the year 2050, emergency evacuation planning, and economic parameters including agricultural production. This information is provided in Section F3.4 of Attachment F to the ER (NextEra 2010**) and was unchanged by the 2012 SAMA supplement (NextEra, 2012a**).

All releases were modeled as occurring at the top height of the containment building. **In the ER, sensitivity** cases were run assuming ground level release, as well as releases at 25 percent, 50 percent, and 75 percent of the containment building height. In response to an NRC staff RAI, NextEra reported that decreasing the release height from the top of the reactor building to ground level decreased the population dose risk and offsite economic cost risk by up to 3 percent and 4 percent, respectively (NextEra 2011a). The thermal content of each of the releases was assumed to be the same as ambient (that is a non-buoyant plume). A sensitivity analysis was performed in the ER assuming a 1 MW and 10 MW heat release plume. In response to an NRC staff RAI, NextEra reported that increasing the release heat decreased the population dose risk by 2 percent and 12 percent, and the offsite economic cost risk decreased by 1 percent and 9 percent for the 1 MW and 10 MW heat release, respectively (NextEra 2011a). Wake effects for the containment building were included in the model. A sensitivity analysis was performed in the ER assuming the wake size was one-half and double the baseline wake size. In response to an NRC staff RAI, NextEra reported that decreasing the wake size by one-half decreased the population dose risk by 1 percent and did not change the offsite economic cost risk, while doubling the wake size increased both the population dose risk and offsite economic cost risk by 1 percent (NextEra 2011a). **While these sensitivity analyses were not re-performed for the 2012 SAMA supplement, NextEra concluded that the results in the ER would be representative of the updated SAMA evaluation (NextEra 2012a).** The NRC staff notes that these results are consistent with previous SAMA analyses that have shown only minor sensitivities to release height, buoyancy, and building wake effects. Based on the information provided, the staff concludes that the release parameters used are acceptable for the purposes of the SAMA evaluation.

NextEra used site-specific meteorological data for the year 2005 as input to the MACCS2 code. The development of the meteorological data is discussed in Section F.3.4.5 of the ER (NextEra 2010). Data from 2004 through 2008 were also considered, but the 2005 data were chosen because the results of a MACCS2 sensitivity analysis indicated that the 2005 data produced more conservative results (i.e., the 2005 data set was found to result in the largest population dose risk and offsite economic cost risk). In response to an NRC staff RAI, NextEra reported that the results of the meteorological data sensitivity analysis, which was performed for each of the years 2004 through 2008, showed a decrease in population dose risk in the range of 5 to 13 percent and a range of 3 to 12 percent decrease in offsite economic cost risk (NextEra 2011a). **NextEra repeated this sensitivity study for the 2012 SAMA supplement (NextEra 2012a), and the 2005 data set was again found to result in the largest population dose risk and offsite economic cost risk.** Missing data were estimated using data substitution methods. These methods include substitution of missing data with corresponding data from another level on the meteorological tower, interpolation between data from the same level, or data from the same hour and a nearby day of a previous year. Hourly stability was classified according to the system used by the NRC (NRC 1983). The baseline analysis assumes perpetual rainfall in the 40 to 50 mi segment surrounding the site. A sensitivity analysis was performed **for the 2012 SAMA supplement** assuming measured rainfall rather than perpetual rainfall in the 40 to 50 mi spatial segment **(NextEra 2012a).** This resulted in a decrease in population dose risk of 14 percent and a decrease in offsite economic cost risk of **15**~~17~~ percent. The NRC staff notes that these results are consistent with previous SAMA analyses that have shown little sensitivity to year-to-year differences in meteorological data. Based on the information provided, the NRC staff concludes that the use of the 2005 meteorological data in the SAMA analysis is reasonable.

The population distribution the licensee used as input to the MACCS2 analysis was estimated for the year 2050 using year 2000 census data as accessed by SECPOP2000 (NRC 2003). The baseline population was determined for each of 160 sectors, consisting of the 16 directions for each of 10 concentric distance rings with outer radii at 1, 2, 3, 4, 5, 10, 20, 30, 40, and 50 mi surrounding the site. County population growth estimates were applied to year 2000 census data to develop year 2050 population distribution. The distribution of the population is given for the 10-mi radius from Seabrook and for the 50-mi radius from Seabrook in the ER (NextEra 2010). In response to an NRC staff RAI, NextEra clarified that the year 2000 population was exponentially extrapolated to year 2050 **(NextEra 2011a).** The NRC staff noted that the total population of 4,157,215, identified in Section 2.6.1 of the ER, was different than the 4,232,394 reported in ER Table F.3.4.1 (NRC 2010a). In response to the NRC staff RAI, this difference was attributed to the following factors (NextEra 2011a):

- the choice of distribution centroids between the two references,
- the inclusion of transient population in the population extrapolation for ER Table F.3.4.1-1 but not in ER Section 2.6.1, and

- the assumption that the population fraction is equal to the land area fraction where the 50-mi radius bisects the census block groups.

The NRC staff also requested clarification of why some sectors showed zero or (small) negative population growth (NRC 2010a). NextEra clarified that this was attributed to the geographic information system (GIS) land layers not being detailed enough to account for the existence of some small islands, and the GIS water sectors were projected as zero populations (NRC 2011a). Also, the direction distribution used in the 2050 projection was slightly offset from the existing population, resulting in some sectors being considered all water and, thus, zero population. In fact, a portion of those sectors include the coastline; therefore, they have a population. The population projections were refined to account for the above and to include the most recent county population growth rates (the sensitivity case above). A sensitivity analysis was performed using the refined population projections and the population distribution centroid for ER Table F.3.4.1-1 **(NextEra 2010).** This resulted in an overall population decrease of about 4 percent, resulting in a corresponding decrease in population dose risk and economic cost risk of 5 percent and 6 percent, respectively. The NRC staff considers the methods and assumptions for estimating population reasonable and acceptable for purposes of the SAMA evaluation.

The emergency evacuation model was modeled as a single evacuation zone extending out 16 km (10 mi) from the plant. NextEra assumed that 95 percent of the population would evacuate. This assumption is conservative relative to the NUREG-1150 study (NRC 1990), which assumed evacuation of 99.5 percent of the population within the emergency planning zone (EPZ). The evacuated population was assumed to move at an average speed of approximately 0.4 mps (0.9 mph) with a delayed start time of 120 minutes after declaration of a general emergency. The evacuation speed was derived from the projected time to evacuate the entire EPZ under adverse weather conditions during the year 2000 (NextEra 2010) and then adjusted by the ratio of the year 2000 EPZ population to the projected year 2050 EPZ population. **In the ER,** NextEra performed sensitivity analyses in which the evacuation speed, the delayed start time or preparation time for evacuation of the EPZ, and the emergency declaration time were each individually decreased by 50 percent and doubled relative to the base case. In response to an NRC staff RAI, NextEra reported that the decrease in evacuation speed increased the population dose risk by 3 percent, and the increase in evacuation speed decreased the population dose risk by 4 percent. Additionally, the decrease in delay time decreased the population dose risk by 9 percent, the increase in delay time decreased the population dose risk by 2 percent, the decrease in emergency declaration time decreased the population dose risk by 6 percent, and the increase in emergency declaration time decreased the population dose risk by 3 percent (NextEra 2011a). For all three parameters, both the increase and decrease in the base values resulted in no change to the offsite economic cost risk. In the ER, NextEra explained that an increase in delay time or emergency declaration time could decrease population dose risk if the evacuation and plume release are simultaneous. NextEra also performed a sensitivity analysis **in the 2012 SAMA supplement (NextEra 2012a)** assuming that the population does not evacuate for a severe accident resulting in a small, early containment penetration failure with no source term scrubbing, representative of a seismically induced severe accident event. **This** resulted in **an increase in** population dose risk **of less than 1** percent **and** no change in offsite economic cost risk. The NRC staff concludes that the evacuation assumptions and analysis are reasonable and acceptable for the purposes of the SAMA evaluation.

In an NRC staff RAI, NextEra clarified that sea-breeze circulation was included in the SAMA evaluation only to the extent that this is included in the onsite meteorological data

(NextEra 2011a). NextEra further explained that there are two major mechanisms associated with sea-breezes, a mixing front and thermal internal boundary layer (TIBL). A mixing front results in increased plume mixing and dispersion, resulting in a potential decrease in population dose. This was conservatively ignored in the SAMA evaluation. However, TIBL could decrease dispersion and increase population dose. Given this, NextEra performed a sensitivity study assuming 25 percent of the year with TIBL formation (data for year 2005 identified a TIBL was present 7 percent of the year). The increase in TIBL formation increased the population dose risk and offsite economic cost risk by 4 percent and 7 percent, respectively. **NextEra re-performed this sensitivity study in the 2012 SAMA supplement (NextEra 2012a). The results of the evaluation indicate that the population dose and offsite economic cost risks increase by less than 1 percent each. NextEra clarified that the previous results were calculated in MACCS2 using the Monte Carlo random bin sampling technique. The revised evaluation summarized above used the MACCS2 sequential hour analysis technique, which provides a more accurate result compared to the Monte Carlo bin sampling technique. Thus, the latest results are shown to be less than previous results despite the increase in release category source terms. In both the original RAI response and the 2012 SAMA supplement, NextEra performed a sensitivity study of the TIBL lid height by changing the lid height from 110 m to 100 m. The decrease in TIBL lid height, in both sensitivity studies,** resulted in an increase in population dose risk and offsite economic cost of less than 1 percent each. The NRC staff concludes that sea-breeze affects have a minor impact on the SAMA analysis results.

Much of the site-specific economic and agricultural data were provided from SECPOP2000 (NRC 2003) by specifying the data for each of the 13 counties surrounding Seabrook, to a distance of 80 km (50 mi). SECPOP2000 uses county economic and agriculture data from the 2000 National Census of Agriculture. This included the fraction of land devoted to farming, annual farm sales, the fraction of farm sales resulting from dairy production, and the value of non-farmland. In response to an NRC staff RAI, NextEra identified that the recent, three known errors in SECPOP2000 were corrected for the SAMA evaluation (NextEra 2011a).

NRC staff asked NextEra to explain its assertion in the 2012 SAMA supplement (NextEra 2012a) that sensitivities to variation in other Level 3 parameters (not explicitly re-evaluated in the 2012 SAMA supplement) are expected to be consistent with the ER sensitivity analysis results. NextEra explained (NextEra 2012b) that except for the difference in source term release, the Level 3 parameters used in the SAMA analysis supplement did not change. In addition, NextEra further noted (a) that greater meteorology specification (imposed as 40 to 50 mi (approximately 64 to 80 km) rather than following the site boundary) produces 15 percent more conservative dose and cost risks, (b) that the re-evaluated sea-breeze effect for the 2012 SAMA supplement (NextEra 2012a) showed only small change in dose and cost risk, and (c) that non-evacuation rather than delayed evacuation for extreme seismic events (release category LE4) results in only a small increase in total LE4 dose consequences.

The NRC staff concludes that the methodology used by NextEra to estimate the offsite consequences for Seabrook provides an acceptable basis from which to proceed with an assessment of risk reduction potential for candidate SAMAs. Accordingly, the NRC staff based its assessment of offsite risk on the CDF and offsite doses reported by NextEra.

F.3 Potential Plant Improvements

The process for identifying potential plant improvements, an evaluation of that process, and the improvements evaluated in detail by NextEra are discussed in this section.

F.3.1 Process for Identifying Potential Plant Improvements

NextEra's process for identifying potential plant improvements (SAMAs) consisted of the following elements:

- review of the most significant basic events from the plant-specific PRA **used in** the **2012** SAMA **supplement (NextEra 2012a)**,
- review of potential plant improvements identified in the Seabrook IPE and IPEEE,
- review of other industry documentation discussing potential plant improvements, and
- insights from Seabrook personnel.

Based on this process, an initial set of 191 candidate SAMAs **was identified in the ER (NextEra 2010), and 4 additional SAMA candidates were identified in the 2012 SAMA supplement. A total of 195 candidate SAMAs, which are** referred to as Phase I SAMAs, was identified. In Phase I of the evaluation, NextEra performed a qualitative screening of the initial list of SAMAs and eliminated SAMAs from further consideration**. The screening was performed** using the following criteria:

- The SAMA is not applicable to Seabrook due to design differences (19 SAMAs screened).
- The SAMA has already been implemented at Seabrook or Seabrook meets the intent of the SAMA (87 SAMAs screened).
- The SAMA is similar to another SAMA under consideration (11 SAMAs screened).
- The SAMA has estimated implementation costs that would exceed the dollar value associated with eliminating all severe accident risk at Seabrook (no SAMA screened).
- The SAMA was determined to provide very low benefit (no SAMA screened).

In response to an NRC staff RAI (NRC 2012a), NextEra clarified that Phase I SAMAs screened on the basis of the first three criteria were not re-reviewed in the 2012 SAMA supplement since this supplement was based on modeling changes that did not change the conclusions of earlier qualitative screening of Phase 1 SAMAs (NextEra 2012b). Based on this screening, 117 SAMAs were eliminated, leaving **78**~~74~~ for **reevaluation, including the 4 new SAMAs identified in the 2012 SAMA supplement (NextEra 2012a). These SAMAs are** referred to as Phase II SAMAs and are listed in Table **1 of the 2012 SAMA supplement (NextEra 2012a). As part of** Phase II, a detailed evaluation was performed for each of **these 78** SAMA candidates, as discussed in Sections F.4 and F.6 below. The estimated benefits for these SAMAs include the risk reduction from both internal and external events.

As previously discussed, NextEra accounted for the potential risk reduction benefits associated with each SAMA by quantifying the benefits using the integrated internal and external events PRA model. In response to NRC staff RAIs, NextEra performed a sensitivity analysis to account for the potential additional risk reduction benefits associated with the additional risk from seismic events **(NextEra 2011a), which was also performed in the 2012 SAMA supplement (NextEra 2012a),** NextEra multiplied the estimated benefits for internal and external events by a factor of 2.~~16 for those Phase II SAMAs that were qualitatively screened on high implementation costs and by a factor of 2.1~~ for all other Phase II SAMAs for which a detailed evaluation was performed (NextEra **2012a**).

F.3.2 Review of NextEra's Process

NextEra's efforts to identify potential SAMAs focused primarily on areas associated with internal initiating events but also included explicit consideration of potential SAMAs for fire and seismic events. The initial list of SAMAs generally addressed the accident sequences considered to be important to CDF from functional, initiating event, and risk reduction worth (RRW) perspectives at Seabrook.

NextEra's SAMA identification process began with a review of the list of potential PWR enhancements in Table 14 of NEI 05-01 (NEI 2005). As a result of this review, 153 SAMAs were identified. In response to NRC staff **RAIs**, NextEra clarified that **25 SAMAs were identified from previous reviews of internal and external events from the Seabrook plant-specific PRA and an additional 13 SAMAs were identified** as a result of a general solicitation of Seabrook staff for possible SAMA candidates by an expert panel**. As mentioned previously, four** additional SAMAs were identified **in the 2012 SAMA supplement, of which three SAMAs were suggested by plant personnel and one SAMA was identified in response to an NRC staff RAI (NextEra 2012a).**

In the ER and subsequent RAI responses, NextEra provided tabular listings of both the Level 1 and LERF PRA internal, fire, and seismic basic events sorted according to their RRW (NextEra 2010), listings of the Level 2 non-LERF basic events that contribute 90 percent of the population dose risk, and a review all of these basic events for potential SAMAs.

These importance analyses were subsequently updated in the 2012 SAMA supplement (NextEra 2012a) based on the SSPSS-2011 PRA model. In this supplement, NextEra provided a tabular listing of the top 15 initiating events contributing to each of CDF and LERF, the top 15 basic events contributing to each of CDF and LERF, and the basic events for the Level 2 release categories that cumulatively contribute to approximately 90 percent of the total public risk (i.e., dose and economic cost risk). As a result, existing SAMAs or new SAMAs were identified for a total of 29 initiating events (one initiating event contributes to both CDF and LERF) and 43 basic events (some basic events contribute to multiple release categories). In response to an NRC staff RAI on the supplement to provide importance analysis down to a level that all potentially cost-beneficial SAMAs could be identified, NextEra provided listings of basic event contributors to non-LERF release categories LL-5 (large late containment basemat failure), SE-3 (small early containment penetration failure to isolate), and SELL (small early RCS release with large late containment failure), down to RRW values of 1.005, 1.003, and 1.003, respectively (NextEra 2012b). For release categories SE-3 and SELL, all of the basic events were already identified and evaluated in the 2012 SAMA supplement. For release category LL-5, 28 new basic events were identified, and a SAMA (either already existing or new) was correlated to each of these basic events. NextEra explained that differences in basic events and corresponding RRW values to those presented in the ER (NextEra 2010) and associated RAI responses (NextEra 2011a) were in general due to an accumulation of small changes including an updated HRA performed in 2009. In

response to a separate RAI (NextEra 2012b), NextEra also provided a listing of all CDF and LERF initiating events contributing greater than 1 percent of the total CDF and 0.3 percent of the total LERF. All of the LERF initiating events were already identified and evaluated in the 2012 SAMA supplement, while 11 new CDF initiating events and a SAMA (either already existing or new) was correlated to each of these initiating events. The newly identified SAMAs, and the results of their evaluation, are discussed further in Section F.6.2.

NextEra states in the ER that no SAMAs were identified to address the operator actions in the Level 1 and LERF basic events importance lists because the current plant procedures and training meet current industry standards, and no plant-specific procedure improvements were identified that would affect the results of the HEP calculations. The NRC staff asked NextEra to consider the feasibility of non-procedural and training SAMAs for the human error basic events (NRC 2011a). In response to **RAIs**, NextEra identified and evaluated **three** operator actions included in the top 15 Level 1 basic events and to automate or install additional alarm indication for the operator action having the highest LERF-related RRW (NextEra 2011a**). Subsequently in the 2012 SAMA supplement (NextEra 2012a), NextEra included an evaluation of SAMAs for 15 different operator failures covered by the importance analyses.**

The NRC staff estimated that a risk reduction of ~~3.3~~ **1** percent, corresponding to the **least bounding cut-off** of the **different importance analysis listings (i.e., CDF initiating event listing) produced by NextEra**, equates to a maximum **baseline** benefit of approximately $**30**,000**, or** approximately $**64**,000 after the benefits have been multiplied by a factor of 2.1 to account for the additional risk from seismic events, which is less than the minimum implementation cost of $100,000 associated with a hardware change.

Based on this, and NextEra's statement discussed previously that procedure and training improvements have been considered but that no improvements were identified that would reduce plant risk, the NRC staff concludes that it is unlikely that additional cost-beneficial SAMAs would be found from a further review of initiating events having lower contribution to CDF.

In response to an NRC staff RAI, NextEra reviewed the cost-beneficial SAMAs from prior SAMA analyses for five Westinghouse four-loop PWR sites (NextEra 2011a). NextEra's review determined that all but two of these cost-beneficial SAMAs were already represented by a SAMA, have intent that was already met at Seabrook, have low potential for risk reduction at Seabrook (e.g., do not address risk-important basic events), or were not applicable to Seabrook. Two SAMAs were identified and evaluated further as a result of this review and are further discussed in Section F.6.2. The two SAMAs are "procedure change to ensure that the reactor coolant system (RCS) cold leg water seals are not cleared" and "installation of redundant parallel service water valves to the emergency diesel generators (EDGs)."

The NRC staff noted that both SAMA 173, identified from the IPEEE review, and SAMA 185 are described as "improve procedural guidance for directing depressurization of RCS," and it asked NextEra to clarify the difference between these two SAMAs (NRC 2010a). In response to the RAI, NextEra clarified that SAMA 173 was to improve procedural guidance directing operators to depressurize the RCS before core damage, while SAMA 185 was to improve procedural guidance directing operators to depressurize the RCS after core damage. The NRC staff considers NextEra's clarification reasonable **and the screening of those Phase I SAMAs acceptable**.

Although the IPE did not identify any fundamental vulnerabilities or weaknesses related to internal events, 14 potential plant improvements were identified. NextEra reviewed these potential improvements for consideration as plant-specific candidate SAMAs. In response to an NRC staff RAI, NextEra clarified that the following 13 SAMAs were identified from the review of the potential plant improvements identified in the IPE (NextEra 2011a):

- Phase II SAMA 167, “install independent seal injection pump (low volume pump) with automatic start,”
- Phase II SAMA 168, “install independent seal injection pump (low volume pump) with manual start,”
- Phase II SAMA 169, “install independent charging pump (low volume pump) with manual start,”
- Phase I SAMA 155, “install alternate emergency AC power source (e.g., swing diesel),”
- Phase II SAMA 156, “install alternate offsite power source that bypasses switchyard, for example, use campus power source to energize Bus E5 or E6,”
- Phase II SAMA 174, “provide alternate scram button to remove power from motor generator (MG) sets to control rod (CR) drives,”
- Phase II SAMA 157, “provide independent AC source for battery chargers, for example, provide portable generator to charge station battery,”
- Phase I SAMA 158, “provide enhanced procedural direction for cross-tie of batteries within each train,”
- Phase II SAMA 159, “install additional batteries,”
- Phase II SAMA 184, “control/reduce time that the containment purge valves are in open position,”
- Phase I SAMA 185, “improve procedural guidance to directing depressurization of RCS,”
- Phase II SAMA 186, “install containment leakage monitoring system,” and
- Phase II SAMA 187, “install RHR isolation valve leakage monitoring system.”

In addition, the improvement identified in the IPE for “alternate, independent EFW pump (e.g., diesel firewater pump hard piped to discharge of startup feed pump),” is already addressed by Phase I SAMA 29, “provide capability for alternate injection via diesel-driven fire pump,” and Phase II SAMA 163, “install third EFW pump (steam-driven).” Phase I SAMA 29 and Phase II SAMA 163 were previously identified from the review of the list of potential PWR enhancements in Table 14 of NEI 05-01 (NEI 2005). Phase I SAMAs 29, 155, 158, and 185 were screened in the Phase I evaluation as having already been implemented.

Based on this information, the NRC staff concludes that the set of SAMAs evaluated in the ER **and 2012 SAMA supplement (NextEra 2012a),** together with those identified in response to NRC staff RAIs, addresses the major contributors to internal event CDF.

As described previously, NextEra’s importance analysis considered both fire and seismic basic events from the internal and external event integrated Level 1 and Level 2 PRA model. The NRC staff noted that since the importance analyses did not separately consider the importance of internal, fire, and seismic events, SAMAs identified to address the important basic events may not address the more important initiator (e.g., fire), and it asked NextEra to explain how the identified SAMAs address this issue (NRC 2010a). In response to the RAI, NextEra explained that the importance analysis considers the contribution from all hazards, and the contribution from the individual hazards will be a subset of the total risk contribution **(NextEra 2011a).** Additionally, based on evaluations provided in response to the NRC staff RAIs discussed above, in which SAMAs were identified to address each of the important Level 1 and 2 basic events, hardware changes to address the individual hazard contributors would not, in NextEra's judgement, be cost beneficial based on a conservative minimum cost for a hardware change of $100,000 (NextEra 2011a). Based on the NRC staff conclusions above regarding NextEra's systematic process for identifying SAMAs for each important Level 1 and 2 basic event and NextEra's statement that procedure/training improvements have been considered but that no improvements were identified that would reduce plant risk, the NRC staff agrees that it is unlikely that additional cost-beneficial SAMAs would be found from a further review of basic events.

Although the IPEEE did not identify any fundamental vulnerabilities or weaknesses related to external events, two potential plant improvements were identified to improve seismic CDF, and five potential plant improvements were identified to improve fire CDF. Additionally, five potential plant improvements were identified that were being evaluated to improve internal event risk but which may also reduce external event risk because they address functional failures. In response to an NRC staff RAI, NextEra clarified that the following 12 SAMAs were identified from the review of the potential plant improvements identified in the IPEEE (NextEra 2011a):

- SAMAs to improve seismic CDF:
  - Phase II SAMA 181, "improve relay chatter fragility," and
  - Phase II SAMA 182, "improve seismic capacity of EDGs and steam-driven EFW pump."
- SAMAs to improve fire CDF:
  - Phase II SAMA 175, "install fire detection in turbine building relay room,"
  - Phase I SAMA 176, "install additional suppression at west wall of turbine building,"
  - Phase I SAMA 177, "improve fire response procedure to indicate that PCCW can be impacted by PAB fire event,"
  - Phase I SAMA 178, "improve the response procedure to indicate important fire areas including control room, PCCW pump area, and cable spreading room," and
  - Phase I SAMA 180, "modify SW pump house roof to allow scuppers to function properly."
- Other SAMAs identified from the IPEEE review:
  - Phase I SAMA 160, "enhancements to address loss of SF6-type sequences,"
  - Phase I SAMA 171, "install high temperature O-rings in RCPs,"
  - Phase I SAMA 173, "improve procedural guidance for directing depressurization of RCS,"
  - Phase II SAMA 179, "fire-induced LOCA response procedure from Alternate Shutdown Panel," and
  - Phase I SAMA 183, "Turbine Building internal flooding improvements."

Phase I SAMAs 160, 171, 173, 176, 177, 178, 180, and 183 were screened in the Phase I evaluation as having already been implemented.

The NRC staff questioned whether SAMA 162, "increase the capacity margin of the CST" addressed basic event COTK25.RT, "CST CO-TK-25 ruptures/excessive leakage" (NRC 2010a). In response to the RAI, NextEra explained that the CST has a median seismic fragility of 1.65 g and a HCLPF of 0.65, without crediting the concrete shield structure surrounding the CST (NextEra 2011a). Therefore, NextEra identified and evaluated a SAMA to make "seismic upgrades to the CST." This is discussed further in Section F.6.2.

The NRC staff asked NextEra to clarify how additional fire barriers for fire areas were considered since SAMA 143, "upgrade fire compartment barriers," was screened in the Phase I evaluation based on the Seabrook plant design including 3-hour rated fire barriers (NRC 2010a). NextEra responded with a review of the fire risk by plant location and explained that it is not physically possible to install additional fire barriers in the control room, which contribute 52 percent of the fire CDF. Additionally, NextEra stated that additional fire barriers in the essential SWGR rooms, which contribute 41 percent of the fire CDF, would have no impact on the fire risk since these rooms are already separated (NextEra 2011a). Other lower risk fire areas were also similarly evaluated with similar conclusions. In a response to a followup NRC staff RAI, NextEra further clarified that additional fire barriers were not considered for the essential SWGR rooms because a review of fire scenarios in these rooms did not identify impacts to any redundant safety train cables (NextEra 2011b). The NRC staff concludes that the applicant's rationale for eliminating fire barrier enhancements from further consideration is reasonable.

Based on the licensee's IPEEE, the review of the results of the Seabrook PRA, which includes seismic and fire events, and the expected cost associated with further risk analysis and potential plant modifications, the NRC staff concludes that the opportunity for seismic and fire-related SAMAs has been adequately explored, and it is unlikely that there are any additional cost-beneficial seismic or fire-related SAMA candidates.

As stated earlier, other external hazards (i.e., high winds, external floods, transportation and nearby facility accidents, and chemical releases) are below the IPEEE threshold screening frequency, or met the 1975 SRP design criteria, and are not expected to represent opportunities for cost-beneficial SAMA candidates. Nevertheless, NextEra reviewed the IPEEE results and identified no additional Phase I SAMAs to reduce HFO risk (NextEra 2010).

For many of the Phase II SAMAs listed in the ER, the information provided did not sufficiently describe the proposed modification. Therefore, the NRC staff asked the applicant to provide more detailed descriptions of the modifications for several of the Phase II SAMA candidates (NRC 2010a). In response to the RAI, NextEra provided the requested information on the modifications for SAMAs 44, 59, 94, 112, 114, 163, 186, and 187 (NextEra 2011a).

The NRC staff questioned NextEra about lower cost alternatives to some of the SAMAs evaluated (NRC 2010a) to include using a portable generator to extend the coping time in loss of AC power events (to power selected instrumentation and DC power to the turbine-driven auxiliary feedwater (TDAFW) pump provide alternate DC feeds (using a portable generator) to panels supplied only by DC bus and purchasing or manufacturing a "gagging device" that could be used to close a stuck-open SG safety valve for a SGTR event prior to core damage.

In response to the RAIs, NextEra clarified that the first alternative to use a portable generator was already represented by SAMA 157, "provide independent AC power

source for battery chargers; for example, provide portable generator to charge station battery" (NextEra 2011a). The second alternative was addressed in the 2012 SAMA supplement (NextEra 2012a) as SAMA 194, "purchase or manufacture of a "gagging device" that could be used to close a stuck-open steam generator safety valve." Both of these SAMAs were assessed in the Phase II cost-benefit evaluation. The NRC staff concludes that these alternatives have been adequately addressed.

The NRC staff requested NextEra to clarify the Phase I screening criteria, which was described in the ER as including two criteria that appear to not have been used—(1) excessive implementation cost, and (2) very low benefit (NRC 2010a). NextEra responded that these criteria, while they could have been used in the Phase I evaluation, were not used in the Phase I screening evaluation in order to force evaluation of more SAMA candidates into the Phase II evaluation so that the merit of each could be judged based on associated costs and benefits (NextEra 2011a).

The NRC staff asked NextEra to provide justification for the screening of SAMA 29, "provide capability for alternate injection via diesel-driven fire pump," in the Phase I evaluation on the basis that it has already been implemented through an existing alternate mitigation strategy (NRC 2010a). In response to the RAI, NextEra responded that Seabrook has the capability to use its diesel-driven fire pump to provide injection to the SGs through implementation of existing SAMGs (NextEra 2011a). NextEra also stated that two portable diesel-driven pumps are also available to provide injection using suction from the fire protection system, the cooling tower basin, and the Browns River. Based on this clarification, the NRC staff considers NextEra's basis for screening SAMA 29 reasonable.

The NRC staff noted that SAMA 64, "implement procedure and hardware modification for a CCW header cross-tie," was screened in the Phase I evaluation because a cross-tie already exists to support a maintenance activity. The staff asked NextEra to clarify if the cross-tie between divisions A and B of the PCCW system is already provided for in existing plant procedures (NRC 2010a). In response to the RAI, NextEra clarified that the Seabrook operating procedures do provide explicit instructions for alignment of the PCCW division A and B cross-tie. Additionally, while the cross-tie is primarily used during maintenance activities, it could be used during an off-normal event involving a failure of heat sink in one division with failure of frontline components in the opposite division, provided that adequate time is available (NextEra 2011a). **Based on this clarification, the NRC staff considers NextEra's basis for screening SAMA 64 reasonable.**

The NRC staff questioned why SAMA 79, "install bigger pilot operated relief valve so only one is required," was screened in the Phase I evaluation based on the intent of the SAMA having already been implemented when the success criterion is two of two PORVs needed for intermediate head SI (NRC 2010a). NextEra responded that the context of SAMA 79 was to increase the capacity of the pressurizer PORVs such that opening of only one PORV would satisfy the feed and bleed success criteria for all loss of feedwater-type sequences, which is all that is needed at Seabrook if feed and bleed is provided by one of two high head charging pumps (NextEra 2010). However, since opening of two PORVs is needed if feed is provided by one of two SI pumps, NextEra provided a Phase II evaluation of this SAMA, the results of which are further discussed in Section F.6.2.

The NRC staff asked NextEra to provide justification for the screening of SAMA 82, "stage backup fans in switchgear rooms," and SAMA 84, "switch for emergency feedwater room fan power supply to station batteries," in the Phase I evaluation on the basis that they are not applicable to Seabrook (NRC 2010a). In response to the RAI, NextEra explained that the context of SAMA 82 was to enhance the availability and reliability of ventilation to the essential SWGR rooms in the event of a loss of SWGR room ventilation. Additionally, this SAMA is more accurately screened as its intent having been already implemented at Seabrook since procedures already exist for maintaining acceptable SWGR room temperatures when ventilation becomes unavailable, which includes opening doors and setting up portable fans (NextEra 2011a). The NRC staff considers NextEra's **basis** for **screening** SAMA 82 reasonable.

Regarding SAMA 84, NextEra explained that the context of this SAMA was to enhance the availability and reliability of ventilation to the EFW pump house, in the event of a loss of pump house ventilation, by switching the pump house ventilation fan(s) power supply to station batteries. NextEra further stated that the initial screening of "not applicable" is incorrect (NextEra 2011a). NextEra explained that since procedures already exist for maintaining acceptable EFW pump house room temperatures when ventilation becomes unavailable, failure of the already reliable ventilation system is not a significant contributor to CDF. Nevertheless, NextEra provided a Phase II evaluation of this SAMA, the results of which are further discussed in Section F.6.2.

The NRC staff noted that SAMA 92, "use a fire water system as a backup source for the containment spray system," was screened in the Phase I evaluation because the containment spray function is not important early, yet basic events RCPCV456A.FC and RCPCV456B.FC, "spray valves fail to open on demand," appear on the LERF importance list (NRC 2010a). In response to the RAI, NextEra explained that these two basic events refer to modeling of the PORVs and not the containment spray valves, that descriptions of these two events in the ER inadvertently referred to the PORVs as PORV spray valves, that the PORV function is unrelated to the containment spray function, and that, therefore, no SAMA is necessary. The NRC staff considers NextEra's **basis for screening SAMA 92** reasonable.

The NRC staff also asked NextEra to provide justification for the screening of SAMA 105, "delay containment spray actuation after a large LOCA," and SAMA 191, "remove the 135 °F temperature trip of the PCCW pumps," in the Phase I evaluation on the basis that they would violate the CLB for Seabrook (NRC 2010a). In response to the RAI, NextEra provided a Phase II evaluation of these SAMAs, the results of which are further discussed in Section F.6.2 (NextEra 2011a).

The NRC staff requested that NextEra clarify the basis for screening SAMA 127, "revise emergency operating procedures (EOPs) to direct isolation of a faulted steam generator," in the Phase I evaluation on the basis that it is already implemented (NRC 2010a). NextEra responded that the context of SAMA 127 was to have specific EOPs for isolation of the SG for the purpose of reducing the consequences of a SGTR, and existing EOPs direct specific operator actions to diagnose a SGTR and to perform its isolation. Additionally, existing plant EOPs also specifically provide actions for the identification and isolation of a faulted SG (NextEra 2011a). The NRC staff considers NextEra's **basis for screening SAMA 127** reasonable.

The NRC staff asked NextEra to clarify the screening of SAMA 188, "containment flooding—modify the containment integrated leak rate test (ILRT) 10-in. test flange to include a 5-in. adapter with isolation valve" based on the statement that "flange and procedures exist" (NRC 2010a). NextEra responded that the 10-in. flange with fire hose adapter has been pre-fabricated, is stored in a designated and controlled area, and is available for attaching to the

10-in. ILRT flange to provide containment flooding via Severe Accident Guideline instructions (NextEra 2011a). NextEra further explained that pre-installation of the flange adapter will provide no significant time savings in light of the containment flooding scenario evolution via the fire hose connection which takes several days. The NRC staff considers NextEra's **basis for screening SAMA 188** reasonable.

The NRC staff notes that the set of SAMAs submitted is not all-inclusive since additional, possibly even less expensive, design alternatives can always be postulated. However, the NRC staff concludes that the benefits of any additional modifications are unlikely to exceed the benefits of the modifications evaluated and that the alternative improvements would be unlikely to cost less than the least expensive alternatives evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered.

The NRC staff concludes that NextEra used a systematic and comprehensive process for identifying potential plant improvements for Seabrook, and the set of SAMAs evaluated in the ER **and 2012 SAMA supplement (NextEra 2012a),** together with those evaluated in response to NRC staff inquiries, is reasonably comprehensive and, therefore, acceptable. This search included reviewing insights from the plant-specific risk studies and reviewing plant improvements considered in previous SAMA analyses.

F.4 Risk Reduction Potential of Plant Improvements

NextEra evaluated the risk-reduction potential of the **78** SAMAs retained for the Phase II evaluation in the ER and **the 2012 SAMA supplement (NextEra 2012a). NextEra also evaluated the risk-reduction potential of the additional SAMAs discussed in Section F.3 that were identified in the 2012 SAMA supplement (NextEra 2012a) and in response to NRC staff RAIs (NextEra 2012b).** The majority of the SAMA evaluations were performed in a bounding fashion in that the SAMA was assumed to eliminate the risk associated with the proposed enhancement. On balance, such calculations overestimate the benefit and are conservative.

NextEra used model re-quantification to determine the potential benefits. The CDF, population dose, and offsite economic cost reductions were estimated using the SSPSS-**2011** PRA model with a truncation level of 1×10^{-14} per year. The changes made to the model to quantify the impact of SAMAs are detailed in **Tables 1 and 2 of the 2012 SAMA supplement (NextEra 2012a) and in Tables 3-2, 3-3 and 4-1 of the response to NRC staff RAIs on the 2012 SAMA supplement (NextEra 2012b).** Tables F-6 and F-7 list the assumptions considered to estimate the risk reduction for each of the evaluated **SAMA** analysis cases, the estimated risk reduction in terms of percent reduction in CDF and population dose, the estimated total benefit (present value) of the averted risk, and the Phase II SAMAs evaluated for each analysis case. The estimated benefits reported in Tables F-6 and F-7 reflect the combined benefit in both internal and external events. The Phase II SAMAs included in **Table F-4** are the **78** Phase II SAMAs **identified from industry sources, plant experts, or** the **IPE or IPEEE. The Phase II** SAMAs **included in Table F-5 are from plant-specific importance analyses**. The determination of the benefits for the various SAMAs is further discussed in Section F.6.

The NRC staff questioned the assumptions used in evaluating the benefits or risk reduction estimates of certain SAMAs **(NRC 2012a). For example, Table 1 of the 2012 SAMA supplement (NextEra 2012a) presents SAMA case CONTX1, which eliminates AC, DC, and PCCW support for one division of CBS. The NRC staff asked how eliminating these support system failures bounds the hardware improvement SAMAs represented by this**

case (i.e., SAMA 91-Install Passive Containment Spray System, SAMA 94-Install Filtered Containment Vent System, SAMA 99-Strengthen Containment, SAMA 102-Construct Containment Ventilation System, and SAMA 107-Install Redundant Containment Spray System). In response to the RAI (NextEra 2012b), NextEra provided a revised evaluation of these SAMAs using more differentiated SAMA analysis cases (i.e., CBSP, FVENT, CONST, and CBSR). Descriptions of these SAMA analysis cases and revised results for the corresponding SAMAs are provided in Table F-6. The NRC staff also asked NextEra to explain the basis for using SAMA analysis case NOATWS to evaluate the risk reduction of potential modifications addressing initiating events (IE) #23, #24, #25, #26, #27, and #28. Initiating events #23 through IE #27 are seismic initiators of different seismic acceleration levels (0.7 g, 1.0 g, 1.4 g, 1.8 g, and 2.5 g) which lead to ATWS while IE #28 is loss of main feedwater (MFW) that also leads to ATWS. In response to the RAI, NextEra clarified (NextEra 2012b) that SAMA analysis case NOATWS assumes all ATWS initiating events (both seismic and non-seismic initiators) are eliminated; therefore, it is a conservative evaluation for all of these initiating events. NextEra further clarified that the description of IE #28 is incorrect and should be ATWS with loss of MFW initially available, which provides further support for the assessment that the use of the SAMA analysis case NOATWS for this initiating event is conservative. The NRC staff considers NextEra's explanations reasonable.

The NRC staff has reviewed NextEra's bases for calculating the risk reduction for the various plant improvements and concludes that the rationale and assumptions for estimating risk reduction are reasonable and generally conservative (i.e., the estimated risk reduction is higher than what would actually be realized). Accordingly, the NRC staff based its estimates of averted risk for the various SAMAs on NextEra's risk reduction estimates.

Table F-46. SAMA cost and benefit screening analysis for Seabrook[a]

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
NOSBO1 2—Replace lead-acid batteries with fuel cells	Eliminate failure of the EDGs	**22** ~~27~~	**6** ~~12~~	**220K (470K)** ~~160K (330K2~~	**525K (1.1M)** ~~300K (620K~~	**>1.75M[w]** ~~>1M~~
14[m]—Install a gas turbine generator						>**2M[w]** ~~1M~~
16[m]—Improve uninterruptable power supplies						>**2M[w]** ~~1M~~
20—Add a new backup source of diesel cooling						>**2M[w]** ~~1M~~

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
161—Modify EDG jacket heat exchanger SW supply & return to allow timely alignment of alternate cooling water source (supply & drain) from firewater, reactor makeup water, dewatering, etc.						>**2M**[w] ~~1M~~
190—Add synchronization on capability to SEPS diesel						>~~1M~~**6.4M**[w]
NOLOSP 13—Install an additional buried offsite power source	Eliminate LOOP events	**18** ~~42~~	**17**~~36~~	**530K (1.2M)** ~~340K (700K~~	**1.2M (2.7M)**~~640K (1.3M)~~	**>3**[w] ~~2.4M[l]~~
24—Bury offsite powerlines						>3M[l]
156—Install alternate offsite power source that bypasses the switchyard; for example, use campus power source to energize Bus E5 or E6						>7M[l]
BREAKER 21—Develop procedures to repair or replace failed 4 kV breakers	Eliminate failure of the 4 KV bus infeed breakers	1	<1	8K (17K)	15K (32K)	Screened[n]
CSBX ~~LOCA02~~ 25—Install an independent active or passive **HPI** system	Eliminate failure of the **HPI** system	**22**~~68~~	**34**~~52~~	**1.1M (2.3M)** ~~470K (980K~~	**2.5M (5.3M)** ~~890K (1.9M~~	**8.8M**[w] ~~>5M[l]~~
26—Provide an additional **HPI** pump with independent diesel						**8.8M**[~~l~~w] ~~>5M[l]~~
~~39—Replace two of the four electric SI pumps with diesel-powered pumps~~						~~>5M[l]~~
LOCA03 28—Add a diverse low-pressure injection system	Eliminate failure of the low-pressure injection system	**2** ~~11~~	**2** ~~29~~	**68K (140K)** ~~160K (340K~~	**160K (340K)** ~~300K (640K~~	>1M

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
LOCA04 35—Throttle low-pressure injection pumps either in medium or large-break LOCAs to maintain RWST inventory	Eliminate RWST running out of water	**13** ~~28~~	**10** ~~12~~	**310K (655K)** ~~160K (330K~~	**730K (1.5M)** ~~300K (630K)~~	**>3M**[w] ~~>1M[l~~
106—Install automatic containment spray pump header throttle valves						>**3M**[w] ~~1M[l~~
DSIPP **39—Replace two of the four electric SI pumps with diesel-powered pumps**	**Eliminate dependency of the existing intermediate head SI pump trains on AC power**	**<1**	**0**	**<1K (<1K)**	**<1K (<1K)**	**>5M**[l]
LOCA01 41—Create a reactor coolant depressurization system	Eliminate all small LOCA events	**2** ~~7~~	**1** ~~2~~	**27K (57K)** ~~33K (70K~~	**64K** ~~63K~~ (130K)	>1M
SW01 43—Add redundant DC control power for SW pumps	Eliminate the dependency of the SW pumps on DC power	**<2** ~~1~~	**0** ~~1~~	**11K (24K)** ~~10K (21K~~	**26K (55K)** ~~19K (40K~~	>100K
CCW01 44—Replace ECCS pump motors with air-cooled motors	Eliminate failure of the component cooling water (CCW) pumps	**14** ~~25~~	**31** ~~23~~	**920K (1.9M)** ~~180K (380K~~	**2.15M (4.6M)** ~~350K (730K~~	>**6M**[w] ~~4M[l~~
PCCABCD 59—Install a digital feed water upgrade	**Eliminates CCW pump failure when AC & DC power support is available**	**4**	**11**	**335K (700K)**	**785K (1.7M)**	>**6.**1M[~~l~~w]
CSBX 55—Install an independent RCP seal injection system with dedicated diesel	**Eliminate failures of support systems (e.g., AC & DC power, cooling) for division B of high–pressure injection**	**28** ~~11~~	**34** ~~12~~	**1.0M (2.2M)** ~~92K (170K~~	**2.45M (5.2M)** ~~180K (370K~~	**>6.4M**[w] ~~>1M~~
56[b]—Install an independent RCP seal injection system without dedicated diesel						**>6.4M**[w] ~~3M~~

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
167—Install independent seal injection pump (low volume pump) with automatic start						>**6.4M**[w] ~~1M~~
168—Install independent seal injection pump (low volume pump) with manual start						>**6.4M**[w] ~~1M~~
169—Install independent charging pump (high volume pump) with manual start						>**6.4M**[w] ~~500K~~
170—Replace the positive displacement pump (PDP) with a 3rd centrifugal pump; consider low volume & cooling water independence						>**6.4M**[w] ~~500K~~
MAB[r] **65—Install digital feed water upgrade**	**Eliminate all plant risk**	**100**	**100**	**3.05M (6.4M)**	**7.15M (15M)**	**>30M**
77—Provide a passive, secondary-side heat-rejection loop consisting of a condenser & heat sink						>**15M**[w] ~~1M~~
PORV ~~FW01~~ 79[d]—Install bigger pilot operated relief valve so only one is required	Eliminate all **PORV failures**	**<1** ~~12~~	**0** ~~7~~	**1.7K (4K)** ~~73K (150K~~	**4.1K (9K)** ~~140K (290K~~	>**2.7M**[w] ~~1M(f~~
HVAC2 80—Provide a redundant train or means of ventilation	Eliminate the dependency of the CS, SI, RHR, & CBS pumps on HVAC	**3** ~~8~~	**5** ~~1~~	**150K (320K)** ~~32K (67K~~	**360K (750K)** ~~61K (130K~~	>**1M**[w] ~~500K~~
OEFWVS 84[e]—Switch for EFW room fan power supply to station batteries	Eliminate loss of EFW ventilation	<1	**0** ~~<1~~	<1K (<~~1K~~**2K**)	**<2K (<4K)**~~<1K (<2K)~~	>250K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
CBSP ~~CONT01~~ 91[b)(g]—Install a passive containment spray system	Eliminate CBS power, signal, ~~and~~& cooling support system failures, ~~and~~& common cause failure among similar components for one division of CBS	0	58 ~~36~~	1.7M (3.5M) ~~160K (340K~~	4.0M (8.3M) ~~310K (650K~~	>10M[w] ~~>3–6M~~
102[b)(g)(v]—Construct a building to be connected to primary & secondary containment & maintained at a vacuum						>56.7M ~~3M~~
FVENT 94[g]—Install a filtered containment vent to remove decay heat; Option 1: Gravel Bed Filter; Option 2: Multiple Venturi Scrubber	Eliminate release category LL3 (containment vent) ~~and~~& prevents 80 percent of release category LL5 (basemat melt-through)	0	69	2.0M (4.1M)	4.6M (9.7M)	>20M[w] ~~5M(l~~
CONST 99[b~~)~~__)(g]—Strengthen primary & secondary containment (e.g., add ribbing to containment shell)	Reduce by a factor of 10 the non-recovery of off-site power before late containment pressure failure occurs	0	4	120K (245K)	270K (570K)	11.5M[w] ~~>10M~~
CBSR 107[b)(g]—Install a redundant containment spray system	Add redundant train of CBS	0	1	29K (62K)	69K (140K)	>10M[w]
XOVNTS 93[b]—Install ~~a redundant~~an unfiltered hardened containment vent	Eliminate failure of the human action to vent containment[u]	0	1	39K (82K)	92K (190K)	>3M ~~3–4M~~
H2Burn 96—Provide post-accident containment inerting capability	Eliminate all hydrogen ignition & burns	0	1 ~~0(g)~~	18K (39K) ~~<1K (<1K~~	43K (90K) ~~<1K (<1K~~	>100K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
108—Install an independent power supply to the hydrogen control system using either new batteries, a nonsafety grade portable generator, existing station batteries, or existing AC/DC independent power supplies, such as the security system diesel						>100K
109—Install a passive hydrogen control system						>100K
OLRP[t] 105[f]—Delay containment spray actuation after a large LOCA	Eliminate the human failure to complete & ensure the RHR & **low-head safety injection** (LHSI**)** transfer to long-**t-**~~t~~erm recirculation during large LOCA events	**3** ~~2~~	**0** ~~<1~~	**12K (25K)** ~~7.2K (15K~~	**27K (58K)** ~~14K (29K~~	>100K
HPME **110—Erect a barrier that would provide enhanced protection of the containment walls (shell) from ejected core debris following a core melt scenario at high pressure**	**Eliminate high-pressure core ejection occurrences**	**0**	**0**	**<1K (<1K)**	**1K (2K)**	**>10M**
CONT02[p] 112—Add redundant & diverse limit switches to each CIV	**Eliminate CIV failures**	0	**6** ~~19~~	**115K (240K)** ~~100K (220K~~	**270K (570K)** ~~200K (420K~~	**>1M[w]** ~~>500K~~
114—Install self-actuating CIVs						>**2M[w]** ~~500K~~
LOCA06[q] 113—Increase leak testing of valves in ISLOCA paths	**Eliminate ISLOCA contribution**	**<**1	3	**48K (100K)** ~~14K (30K~~	**110K (240K)** ~~27K (60K~~	**>1M[w]** ~~>100K~~
115—Locate RHR inside containment		~~2~~	~~7~~	~~28K (60K)~~	~~53K (110K)~~	>1M
187—Install RHR isolation valve leakage monitoring system						>**500K[w]** ~~190K~~

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
NOSGTR 119—Institute a maintenance practice to perform a 100% inspection of SG tubes during each refueling outage	Eliminate all SGTR events	**5** ~~3~~	**2** ~~17~~	**67K** (**140K**) ~~86K (180K~~	160K (**330K**) ~~345K~~	>500K
121—Increase the pressure capacity of the secondary side so that a SGTR would not cause the relief valves to lift						>500K
125—Route the discharge from the main steam safety valves (MSSVs) through a structure where a water spray would condense the steam & remove most of the fission products						>500K
126—Install a highly reliable (closed loop) SG shell-side heat removal system that relies on natural circulation & stored water sources						>**15M**[w] ~~500K~~
129—Vent MSSVs in containment						>500K
NOATWS 130—Add an independent boron injection system	Eliminate all ATWS events	**4** ~~3~~	**2** ~~11~~	**60K** (**130K**) ~~70K (150K~~	**140K** (**290K**)	>500K
131—Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS						>500K
133—Install an ATWS sized filtered containment vent to remove decay heat						>500K
174—Provide alternate scram button to remove power from MG sets to CR drives						>500K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
LOCA05 147—Install digital large break LOCA protection system	Eliminate all piping failure LOCAs	9 ~~10~~	2 ~~12~~	77K (160K) ~~100K (220K~~	180K (380K) ~~200K (410K~~	>500K
NOSLB 153—Install secondary side guard pipes up to the main steam isolation valves	Eliminate all steam line break events	<1 ~~0~~	0 ~~<1~~	5K (11K) ~~3K (7K~~	11K (24K) ~~6K (13K~~	>500K
OSEPALL 154[k]—Modify SEPS design to accommodate automatic bus loading & automatic bus alignment	Eliminate failure of all operator actions to align & load the SEPS DGs	8 ~~Not Provided~~	2 ~~Not Provided~~	64K (135K) ~~33K (68K~~	150K (320K) ~~62K (130K~~	>750K
Case INDEPAC **157—Provide independent AC power source for battery chargers; for example, provide portable generator to charge station battery**	**Eliminate failure of operator action to shed DC loads to extend batteries to 12 hours. Also, eliminate failure to recover offsite power for plant-related, grid-related, & weather-related LOOP events**[h]	**<2 ~~4~~**	**1 ~~2~~**	**34K (72K) ~~23K (48K~~**	**80K (170K) ~~45K (95K~~**	**30K**
159—Install additional batteries						>1M
CST01 162—Increase the capacity margin of the CST	Eliminate CST running out of water	<2 ~~1~~	1	35K (73K) ~~9K (18K~~	81K (170K) ~~16K (34K~~	>2.5M[w] ~~100K~~
164—Modify 10" condensate filter flange to have a 2½" female fire hose adapter with isolation valve						**>40K**
TDAFW 163—Install third EFW pump (steam-driven)	Eliminate failure of the TDAFW train	5 ~~19~~	12 ~~9~~	360K (750K) ~~100K (210K~~	835K (1.8M) ~~190K (400K~~	>2M[l]

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)(j)		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
NORMW **165—RWST fill from firewater during containment injection—modify 6" RWST flush flange to have a 2½" female fire hose adapter with isolation valve**	**Guaranteed success of RWST makeup for long-term sequences where recirculation is not available**	**5 ~~10~~**	**2 ~~8~~**	**57K (120K) ~~75K (160K~~**	**130K (280K) ~~120K (300K~~**	**50K**
RCPL **172—Evaluate installation of a "shutdown seal" in the RCPs being developed by Westinghouse**	**Eliminate loss of RCP seal cooling initiating event & RCP seal failures subsequent to a plant transient**	**34**	**49**	**1.5M (3.2M)**	**2.5M (7.4M)**	**>2M(l)**
FIRE2 175—Improve fire detection in turbine building relay room	This SAMA has been implemented (NextEra, 2011b).					
FIRE1A 179—Fire-induced LOCA response procedure from alternate shutdown panel	Eliminate **operator failure to close PORV block valve during a** control room fire	**0** ~~1~~	**0** ~~<1~~	**<1K (<1K)** ~~4K (8K~~	**<1K (<2K)** ~~7K (15K~~	>20K(l)
SEISMIC01 181—Improve relay chatter fragility	Eliminate all seismic relay chatter failures	**12** ~~9~~	**3** ~~12~~	**87K (180K)** ~~100K (210K~~	**204K (470K)** ~~200K (410K~~	>600K(l)
SEISMIC02 182—Improve seismic capacity of EDGs & steam-driven EFW pump	Eliminate all seismic failures of EDGs or turbine-driven emergency feedwater (TDEFW)	**<1** ~~0~~	0	**2.4K (6K)** ~~<1K (<1K~~	**5.6K (12K)** ~~<1K (<1K~~	>500K
COP 184—Control & reduce time that the containment purge valves are in open position	Eliminate possibility of containment purge valves being open at the time of an event	0	~~≈~~0	<1K (<1K)	<1K (<**2K**)	>20K
CISPRE 186(o)—Install containment leakage monitoring system	Eliminate all CDF contribution from pre-existing containment leakage	**0** ~~Not Provided~~	**0** ~~Not Provided~~	**4.4K (12K)** ~~11K (23K~~	**10K (27K)** ~~20K (43K~~	>500K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction: CDF	% Risk reduction: Population dose	Total benefit ($)[j]: Baseline (internal + external)	Total benefit ($)[j]: Baseline with uncertainty	Cost ($)
SEPS 189—Modify or analyze SEPS capability; one of two SEPS for LOOP non-SI loads, two of two for LOOP SI loads	Modify fault tree so that one of two SEPS DGs are required rather than both SEPS DGs being required	**6** ~~7~~	**2** ~~1~~	**63K (130K)** ~~30K (60K~~	**150K (310K)** ~~60K (120K~~	**>2M**[w] ~~300K~~
PCTES 191[f]—Remove the 135 °F temperature trip of the PCCW pumps	Eliminate inadvertent failure of the redundant TE/logic of the associated PCC division for both loss of PCCW initiating events & loss of PCCW mitigative function	<1	**0** ~~<1~~	<1K (<1K)	<1K (<**2K** ~~1K~~)	>100K
NOCBFLD **192[i]—Install a globe valve or flow limiting orifice upstream in the fire protection system**	**Eliminate control building fire protection flooding initiators**	**24** ~~25~~	**11** ~~6~~	**470K (990K)** ~~160K (340K~~	**1.1M (2.3M)** ~~310K (640K~~	**370K**[w] ~~200K~~
CSV167 **193[c]—Hardware change to eliminate MOV AC power dependency**	**Eliminate operator failure to close CIV CS-V-167 locally**~~Eliminate MOV AC power dependency by replacing the MOV with a fail-closed AOV~~	**0**	**5** ~~35~~	**86K (180K)** ~~190K (400K~~	**200K (420K)** ~~365K (770K~~	**300K**
MSSVRS **194—Purchase or manufacture a "gagging device" that could be used to close a stuck-open SG safety valve**	**Eliminate failure of MSSVs to reseat**	**0**	**0**	**<1K (<1K)**	**<1K (<2K)**	**>30K**
CCTE1 **195[s]—Make improvements to PCCW temperature control reliability**	**Eliminate failure of temperature control & modulation for PCC Trains A & B that could fail PCCW**	**3**	**5**	**140K (300K)**	**340K (710K)**	**300K**

[a] SAMAs in bold are potentially cost beneficial. This table summarizes the results of the revised SAMA analysis provided in the 2012 SAMA supplement (NextEra 2012a), which revised all results reported for "% Risk reduction" and "Total benefit ($)," and included changes to "Analysis case & applicable SAMAs," "Modeling assumptions," and "Cost ($)."

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[j]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	

(b) This is retained as a quantitatively evaluated Phase II SAMA in response to NRC staff RAI 3.g (NextEra **2011a**).

(c) This is a new SAMA identified in response to NRC staff RAI 2.f (NextEra 2011a) and conference call clarification #7 (NRC 2011a).

(d) Evaluation of this SAMA is provided in response to NRC staff RAIs 5.g (NextEra 2011a) and conference call clarification #14 (NRC 2011a**), and it was subsequently updated in the 2012 SAMA supplement (NextEra 2012a**).

(e) Evaluation of this SAMA is provided in response to NRC staff RAI 5.j (NextEra 2011a**) and was subsequently updated in the 2012 SAMA supplement (NextEra 2012a**).

(f) Evaluation of these SAMAs is provided in response to NRC staff RAI 5.n (NextEra 2011a) and conference call clarification #15 (NRC 2011a**), and it was subsequently updated in the 2012 SAMA supplement (NextEra 2012a).**

(g) In response to an NRC staff RAI, NextEra subdivided previous SAMA analysis case, CONTX1, into separate SAMA analysis cases CBSP (SAMA s 91 and 102), FVENT (SAMA 94), CONST (SAMA 99), and CBSR (SAMA 107) given the potentially high benefits (NextEra 2012b). NextEra refers to these as sensitivity cases.

(h) Information is provided **for SAMA157** in response to NRC staff RAI 6.h (NextEra 2011a**), and it subsequently updated in the 2012 SAMA supplement (NextEra 2012a**).

(i) This is a new SAMA **(#192)** identified and evaluated in response to NRC staff RAI 1.a (NextEra 2011a) and conference call clarification #1 (NRC 2011a**) and subsequently updated in the 2012 SAMA supplement (NextEra 2012a**).

(j) Values in parenthesis are the results of the sensitivity analysis applying a multiplier of 2.1 to account for the additional risk of seismic events (NextEra 2011b).

(k) The analysis case for SAMA 154 changed from NOSBO to OSEPALL in response to followup NRC staff RAI 4 (NextEra 2011b).

(l) Cost updated in supplement to response to followup NRC staff RAI 4 (NextEra 2011c).

(m) The analysis case for SAMAs 14 and 16 changed from NOLOSP to NOSBO in response to followup NRC staff RAI 4 (NextEra 2011b).

(n) In response to followup NRC staff RAI 4, NextEra determined that detailed procedures already exist for inspection and repair of the Seabrook 4 kV breakers, and this SAMA was, therefore, screened from further consideration (NextEra 2011b).

(o) The analysis case for SAMA 186 changed from CONT01 to CISPRE in response to followup NRC staff RAI 4 (NextEra 2011b).

(p) NextEra notes (NextEra 2010) that although calculated as eliminating all CIV failures, the limit switches actually contribute no more than 50 percent to the containment isolation function; thus, the upper bound benefit is more accurately $566*0.5 = $283K (NextEra 2012a).

(q) NextEra notes (NextEra 2010) that although calculated as eliminating all ISLOCAs pressure isolation valve testing could be assumed to reduce ISLOCA by half, thus the upper bound benefit is more accurately $240K * 0.5 = $120K (NextEra 2012a).

(r) In response to NRC staff RAI **4, NextEra clarified that the analysis case for SAMAs designated MAB are evaluated using the MACR (NextEra 2012b).**

(s) In response to an NRC staff, NextEra clarified that SAMA analysis case CCTE1 addresses both the reliability of PCCW and loss of CCW as an initiator (NextEra 2012b).

(t) Although the name of this SAMA analysis case was changed from "OLPRS" in the 2012 SAMA supplement to "OLPR" in the ER, the modeling assumptions are unchanged (NextEra 2012a).

(u) Description of analysis case provided in response to NRC staff RAI 2f (NextEra ~~,~~2011b).

(v) The **analysis case CBSR was used to represent this SAMA because CBSP would prevent containment overpressure (NextEra 2012b).**

(w) Cost updated in 2012 SAMA supplement (NextEra 2012a).

F.5 Cost Impacts of Candidate Plant Improvements

NextEra developed plant-specific costs of implementing the **78 Phase II candidate SAMAs evaluated in the ER and the 2012 SAMA supplement (NextEra 2012a). This SAMA group consisted of SAMAs identified from industry, by plant experts, by identifying important**

failures, and by plant improvements identified in the Seabrook IPE and IPEEE. NextEra also developed implementation cost for the additional SAMAs discussed in Section F.3 that were identified in the 2012 SAMA supplement (NextEra 2012a) and in response to NRC staff RAIs (NextEra 2012b). An expert panel—composed of senior plant staff from the PRA group, the design group, operations, and license renewal—developed the cost estimates based on their experience with developing and implementing modifications at Seabrook. The NRC staff requested that NextEra describe the level of detail used to develop the cost estimates (NRC 2010a). In response to the RAI, NextEra explained that the cost estimates were based on the experience and judgment of the plant staff serving on the expert panel and that, in most cases, detailed cost estimates were not developed because of the large margin between the estimated SAMA benefits and the estimated implementation costs (NextEra 2011a). The cost estimates conservatively did not specifically account for inflation, contingencies, implementation obstacles, or replacement power costs (RPC).

The NRC staff reviewed the bases for the applicant's cost estimates **provided in the ER** (presented in Section F.7.2 and Table F.7-1 of Attachment F to the ER). For certain improvements, the NRC staff also compared the cost estimates to estimates developed elsewhere for similar improvements, including estimates developed as part of other applicants' analyses of SAMAs for operating reactors and advanced light-water reactors. In response to an RAI requesting a more detailed description of the changes associated with Phase II SAMAs 44, 59, 94, 112, 114, 163, 186, and 187, NextEra provided additional information detailing the analysis and plant modifications included in the cost estimate of each improvement (NextEra 2011a). The staff reviewed the costs and found them to be reasonable and generally consistent with estimates provided in support of other plants' analyses. **In many cases, the cost estimates and their descriptions were superseded by the estimates performed for the 2012 SAMA supplement (NextEra 2012b), and they were generally higher than the cost estimates provided in the ER and associated RAI responses. Based on its review of this supplement, the NRC staff requested more detailed justification of the cost estimates for Phase II SAMAs 162 and 189 (NRC 2012a). In response to the RAI, NextEra provided additional justification as to why the cost estimates increased for these SAMAs (NextEra 2012b). For SAMA 162, NextEra explained that the original cost estimate of greater than $100,000 was made to represent a non-complex hardware change because a detailed estimate was not needed due to the low benefit estimated for the SAMA, but that the higher benefit estimated in the 2012 SAMA supplement necessitated reassessing the implementation cost to reflect the expected scope of the modification. Similarly, for SAMA 189, NextEra explained that the original cost estimate of greater than $300,000 was a conservative minimum estimate made based on the assumption that the SAMA would primarily be an analytical task, while the higher benefit estimate in the 2012 SAMA supplement for this SAMA necessitated the development of a more detailed cost estimate of the expected scope of the modification, which includes engineering analysis, hardware modifications, and testing.**

The NRC staff also asked NextEra to provide the basis for the implementation cost estimates for the plant modifications to address IE #23, #24, #25, #26, #27, and #28. Initiating events #23 through IE #27 are seismic initiators of different seismic acceleration levels (0.7 g, 1.0 g, 1.4 g, 1.8 g, and 2.5 g), which lead to ATWS while IE #28 is loss of MFW that also leads to ATWS. In response to the RAI, NextEra clarified (NextEra 2012b) that modifications to reduce risk from IE #23 through IE #27 all include structural upgrades to the reactor internals to increase seismic capacity, which would be expected to significantly exceed the $500,000 cost estimate for this SAMA case. Additionally, NextEra clarified that IE #28 is dominated by failure of control rods to insert

and failure to initiate emergency boration of RCS and that a hardware modification to upgrade reactor internals and emergency boration system are expected to significantly exceed $500,000. The NRC staff considers NextEra's clarification reasonable.

The NRC staff noted that Phase I SAMA 65, "install a digital feed water upgrade," has an estimated implementation cost of $30 million, which is much larger than the estimated implementation cost of more than $500,000 for Phase II SAMA 147, "install digital large break LOCA protection system." The NRC staff asked NextEra to explain the reason for this difference between what appear to be similar modifications (NRC 2010a). NextEra responded that the estimated implementation cost of $30 million for Phase I SAMA 65 was based on a detailed assessment of the costs associated with the Seabrook long-range plan for a digital upgrade of the feedwater control system, while the estimated cost of more than $500,000 for SAMA 147 was based on the judgment of the expert panel (NextEra 2011a). NextEra also noted that since the conservatively estimated benefit for SAMA 147 was much less than the estimated implementation cost, developing a more detailed cost estimate for this SAMA was not necessary. The NRC staff considers NextEra's clarification reasonable.

The NRC staff also requested additional clarification on the estimated cost of $30,000 for implementation of Phase II SAMA 157, "provide independent AC power source for battery chargers," which seems low for what is described as a hardware change (NRC 2010a). In response to the RAI, NextEra explained that the cost estimate is based on expert panel judgment and includes procurement of a small portable, nonsafety-related 480 V generator and associated connection cables, operation guideline development, and storage onsite in a convenient location for ease in moving into position/connected if ever needed during an extended SBO event (NextEra 2011a). The NRC staff considers NextEra's clarification reasonable.

As discussed in Section F.2.2, NextEra provided the results of a sensitivity analysis that applied a multiplier of 2.1 to account for the additional risk reduction from seismic events (NextEra 2011b, **2012a**). In **these analyses**, NextEra revised the implementation costs for several SAMAs in which the estimated costs were determined to be overly conservative. The revised implementation costs are **reflected** in Tables F-6 and F-7. The staff reviewed the basis for each of the revised costs and found them to be reasonable and, generally, consistent with estimates provided in support of other plants' analyses.

The NRC staff concludes that the cost estimates provided by NextEra are sufficient and appropriate for use in the SAMA evaluation.

F.6 Cost-Benefit Comparison

NextEra's cost-benefit analysis and the NRC staff's review are described in the following sections.

F.6.1 NextEra's Evaluation

The methodology used by NextEra was based primarily on NRC's guidance for performing cost-benefit analysis (i.e., NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook* (NRC 1997a)). The guidance involves determining the net value for each SAMA according to the following formula:

Net Value = (APE + AOC + AOE + AOSC) – COE where, APE = present value of averted public exposure ($)

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

AOE = present value of averted occupational exposure costs ($)

AOSC = present value of averted onsite costs ($)

COE = cost of enhancement ($)

If the net value of a SAMA is negative, the cost of implementing the SAMA is larger than the benefit associated with the SAMA, and it is not considered cost beneficial. NextEra's derivation of each of the associated costs is summarized below**, which reflects updated values provided in the 2012 SAMA supplement (NextEra 2012a).**

NUREG/BR-0058 has recently been revised to reflect the NRC's policy on discount rates. Revision 4 of NUREG/BR-0058 states that two sets of estimates should be developed, one at 3 percent and one at 7 percent (NRC 2004). NextEra provided a base set of results using the 7 percent discount rate and a sensitivity study using the 3 percent discount rate (NextEra **2012a**).

Averted Public Exposure (APE) Costs

The APE costs were calculated using the following formula:

APE = Annual reduction in public exposure (Δperson-rem/year)

x monetary equivalent of unit dose ($2,000 per person-rem)

x present value conversion factor (10.76 based on a 20-year period with a 7 percent discount rate)

As stated in NUREG/BR-0184 (NRC 1997a), the monetary value of the public health risk after discounting does not represent the expected reduction in public health risk due to a single accident. Rather, it is the present value of a stream of potential losses extending over the remaining lifetime (in this case, the renewal period) of the facility. Thus, it reflects the expected annual loss due to a single accident, the possibility that such an accident could occur at any time over the renewal period, and the effect of discounting these potential future losses to present value. For the purposes of initial screening, which assumes elimination of all severe accidents caused by internal and external events, NextEra calculated an APE of approximately $**815,100** ~~230,400~~ for the 20-year license renewal period (NextEra **2012b**).

Averted Offsite Property Damage Costs (AOC)

The AOCs were calculated using the following formula:

AOC = Annual CDF reduction

x offsite economic costs associated with a severe accident (on a per-event basis)

x present value conversion factor

This term represents the sum of the frequency-weighted offsite economic costs for each release category, as obtained for the Level 3 risk analysis. For the purposes of initial screening, which assumes elimination of all severe accidents caused by internal events, NextEra calculated an annual offsite economic cost of about $23,500 based on the Level 3 risk analysis (NextEra 2011a). This results in a 7 percent-discounted value of approximately $**1,950,600** ~~253,300~~ for the 20-year license renewal period **(NextEra 2012b).**

Averted Occupational Exposure (AOE) Costs

The AOE costs were calculated using the following formula:

AOE = Annual CDF reduction
x occupational exposure per core damage event
x monetary equivalent of unit dose
x present value conversion factor

NextEra derived the values for AOE from information provided in Section 5.7.3 of the *Regulatory Analysis Technical Evaluation Handbook* (NRC 1997a). Best estimate values provided for immediate occupational dose (3,300 person-rem) and long-term occupational dose (20,000 person-rem over a 10-year cleanup period) were used. The present value of these doses was calculated using the equations provided in the handbook in conjunction with a monetary equivalent of unit dose of $2,000 per person-rem, a real discount rate of 7 percent, and a time period of 20 years to represent the license renewal period. For the purposes of initial screening, which assumes elimination of all severe accidents caused by internal events, NextEra calculated an AOE of approximately $**4,600** ~~5,500~~ for the 20-year license renewal period (NextEra **2012b**).

Averted Onsite Costs (AOSC)

AOSC include averted cleanup and decontamination costs (ACC) and averted power replacement costs. Repair and refurbishment costs are considered for recoverable accidents only and not for severe accidents. NextEra derived the values for AOSC based on information provided in Section 5.7.6 of NUREG/BR-0184, the *Regulatory Analysis Technical Evaluation Handbook* (NRC 1997a).

NextEra divided this cost element into two parts—the onsite cleanup and decontamination cost, also commonly referred to as ACC, and the RPC.

ACC were calculated using the following formula:

ACC = Annual CDF reduction
x present value of cleanup costs per core damage event
x present value conversion factor

The total cost of cleanup and decontamination subsequent to a severe accident is estimated in NUREG/BR-0184 to be $$1.5\times10^9$ (undiscounted). This value was converted to present costs over a 10-year cleanup period and integrated over the term of the proposed license extension. For the purposes of initial screening, which assumes elimination of all severe accidents caused by internal events, NextEra calculated an ACC of approximately $**141,700** ~~167,200~~ for the 20-year license renewal period.

Long-term RPC were calculated using the following formula:

RPC = Annual CDF reduction

x present value of replacement power for a single event

x factor to account for remaining service years for which replacement power is required

x reactor power scaling factor

NextEra based its calculations on the rated Seabrook gross electric output of 1,290 MWe and scaled up from the 910 MWe reference plant in NUREG/BR-0184 (NRC 1997a). Therefore, NextEra applied a power scaling factor of 1,290/910 to determine the RPC. For the purposes of initial screening, which assumes elimination of all severe accidents caused by internal events, NextEra calculated an RPC of approximately $**136,500** and an AOSC **(AOSC = ACC + RPC)** of approximately $**278**,200 ~~and RPC of $162,300)~~ for the 20-year license renewal period **(NextEra 2012b).**

Using the above equations, NextEra estimated the total present dollar value equivalent associated with eliminating severe accidents from internal and external events at Seabrook to be about $**3,048,500.** Use of a multiplier of 2.1 to account for the additional risk from seismic events in the sensitivity analysis increases the **value, as estimated by the NRC staff, to $6.4 million. This represents the dollar value associated with completely eliminating all internal and external event severe accident risk at Seabrook, and it is also referred to as the maximum averted cost risk (MACR). NextEra explained (NRC 2012b) that the value of $3,048,500, reported in a response to an RAI (NextEra 2012b), was slightly updated from the value of $3,051,800 reported in the 2012 SAMA supplement (NextEra 2012a). The value was updated because of refinements in the calculation that were made related to time used to declare a general emergency. The small reduction had negligible impact on the SAMA cost benefit analysis. The NRC staff agrees that this change would have negligible impact on the SAMA cost benefit analysis.**

NextEra's Results

If the implementation costs for a candidate SAMA exceeded the calculated benefit, the SAMA was considered not to be cost beneficial. In the baseline analysis contained in the **2012 SAMA supplement (NextEra 2012a),** using a 7 percent discount rate, NextEra identified **three**~~one~~ potentially cost-beneficial **SAMAs (SAMAs 157,** 165**, and 192**). Based on the consideration of analysis uncertainties, NextEra identified **three** additional potentially cost-beneficial SAMAs **(SAMAs 164, 172, and 195). In addition, as a result of the sensitivity analysis using a multiplier of 2.1 to account for the additional risk** from **seismic events, NextEra identified one additional cost-beneficial SAMA (SAMA 193).** The potentially cost-beneficial SAMAs for Seabrook are listed below:

- SAMA 157—provide independent AC power source for battery chargers,
- **SAMA 164—modify condensate filter flange to incorporate a 2.5-in female hose adapter and isolation valve,**
- SAMA 165—RWST fill from firewater during containment injection—modify 6-in. RWST flush flange to have a 2½-in. female fire hose adapter with isolation valve,
- **SAMA 172—evaluate installation of a RCP "shutdown seal" being developed by Westinghouse,**

- SAMA 192—install a globe valve or flow limiting orifice upstream in the fire protection system,
- SAMA 193—hardware change to eliminate MOV AC power dependency, and
- **SAMA 195—make improvement to PCCW temperature control.**

The potentially cost-beneficial SAMAs, and NextEra's plans for further evaluation of these SAMAs, are discussed in more detail in Section F.6.2.

F.6.2 Review of NextEra's Cost-Benefit Evaluation

The cost-benefit analysis performed by NextEra was based primarily on NUREG/BR-0184 (NRC 1997a) and discount rate guidelines in NUREG/BR-0058 (NRC 2004), and it was executed consistently with this guidance. **Three SAMAs were** determined to be cost beneficial in NextEra's baseline analysis in the **2012** SAMA **supplement (SAMAs 157,** 165, **and 192,** as described above). NextEra stated that **these SAMAs** would be entered into the Seabrook long-range plan development process for further implementation consideration (NextEra **2012a**).

NextEra considered the impact that possible increases in benefits from analysis uncertainties would have on the results of the SAMA assessment. In the **2012 SAMA supplement (NextEra 2012a),** NextEra presents an uncertainty **multiplier of 2.35 based on the ratio** of the CDF **mean value of 1.23×10^{-5} per year to** the 95th percentile value **of 2.86×10^{-5} per year.** Since none of the Phase I SAMAs were screened based on excessive cost or very low benefit, a **reexamination** of the Phase I SAMAs based on the **95th percent** upper bound benefits was not necessary. NextEra **examined** the Phase II SAMAs to determine if any would be potentially cost beneficial if the baseline benefits were increased by a factor of **2.35. As a result, three SAMAs** became cost beneficial **(SAMAs 164, 172, and 195**, as described above). Although not cost beneficial in the baseline analysis, NextEra stated that **these SAMAs** would be entered into the Seabrook long-range plan development process for further implementation consideration (NextEra **2012a**).

The NRC staff asked NextEra to describe how the uncertainty distribution was developed to derive the 95th percentile CDF value and how the distribution is different for internal, fire, and seismic CDF (NRC 2010a). In response to the RAI, NextEra explained that the uncertainty distribution was developed using a Monte Carlo sample size of 10,000 and a sequence bin cutoff of 1×10^{-9}, that the distribution included the integrated contribution from both internal and external events, and that individual contributions for internal, fire, and seismic events were not developed (NextEra 2011a). In response to a followup RAI, NextEra further clarified that the uncertainty analysis included uncertainty distributions for fire-initiating events, seismic-initiating events, component seismic fragilities, operator actions, and component random failures (NRC 2011b). NextEra also noted that, while uncertainty distributions were not specifically considered for hot short probabilities and non-suppression probabilities, numerous sensitivity studies were performed to support the fire events and seismic events models to ensure the reasonableness of key input parameters. The results of these sensitivity studies indicate that the baseline fire and seismic results are relatively insensitive to reasonable variations in key input parameters. Based on the results of these studies and the level of uncertainty applied in the fire and seismic events analyses, NextEra concluded that the uncertainty distribution used for the SAMA evaluation adequately reflects the uncertainty for both internal and external events.

NextEra provided the results of additional sensitivity analyses in the ER, including the use of 3 percent and 8.5 percent discount rates, variations in MACCS2 input parameters (as discussed in Section F.2.2), and a 41-year analysis period representing the remaining operating life of the plant accounting for the expected 20-year period of extended operation. **Cost benefits are determined using the 3 percent discount rate, as clarified in an RAI response, and the 41-year extended period are bounded by the cost benefits determined using 95 percent upper bound MACR.** These analyses did not identify any additional potentially cost-beneficial SAMAs.

SAMAs identified primarily on the basis of the internal events analysis could provide benefits in certain external events, in addition to their benefits in internal events. Since the SSPSS-**2011** PRA model is an integrated internal and external events model, NextEra's evaluation accounted for the potential risk reduction benefits associated with both internal and external events. The NRC staff asked NextEra to assess the impact of updated 2008 seismic hazard curves by the USGS on the Seabrook SAMA analysis (NRC 2010a). As indicated in Section F.2.2, NextEra responded with a sensitivity analysis in which a **2.1** multiplier is applied to the estimated benefits for internal **and external** events to account for the higher seismic CDF developed from the 2008 USGS seismic hazard curves (NextEra 2011a). **This same multiplier was subsequently used in the 2012 SAMA supplement (NextEra 2012a).** Since no SAMAs were screened in the Phase I analysis on very low benefit or excessive implementation cost, NextEra did not reexamine the Phase I SAMAs.

However, NextEra did provide a sensitivity analysis that reexamined the Phase II SAMAs to determine if any would be potentially cost beneficial if the baseline **(7 percent real discount rate), uncertainty benefits (95th uncertainty percentile), and a 2.1 seismic multiplier were considered together (NextEra 2012a). As a result of this sensitivity analysis, one additional SAMA (SAMA 193) became cost beneficial. Although not cost beneficial in the baseline analysis, NextEra stated that this SAMA would be entered into the Seabrook long-range plan development process for further implementation consideration (NextEra 2012a).**

As indicated in Section F.3.2, in response to NRC staff RAIs and followup RAIs **related to the ER (NextEra 2010) and 2012 SAMA supplement (NextEra 2012a), NextEra performed cost-benefit analyses on risk-significant** Level 1 and Level 2 basic events**, including** human error basic events **and risk-significant initiating events**. The additional SAMAs and NextEra's evaluation of each is summarized in Table F-7 (NextEra **2012a**, **2012b**). This table also provides the results of the sensitivity analysis applying the multiplier of 2.1 to account for the additional risk of seismic events (NextEra **2012a, 2012b). While these analyses did not identify any additional potentially cost-beneficial SAMAs, two** of the SAMAs were determined to be cost beneficial **but were already identified as such** in the baseline **SAMA analyses after accounting for uncertainties (SAMA 195) and after accounting for** the **seismic multiplier of 2.1 (SAMA 193).**

Table F-5~~7~~. SAMAs identified and evaluated for risk-significant basic events and initiating events[a]

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[b]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
OALTO Provide automatic alignment of alternate cooling based on applicable signals	Eliminate failure of operator to align alternate cooling	4	11	340K (710K)	800K (1.7M)	>2.4M
PCCABCD Install a diverse & independent CCW pump, reduce to reduce potential for common mode failure	Eliminate CCW pump failure if AC & DC power are available	4	11	335K (700K)	785K (1.65M)	>6M
SWG11AB Improve Bus 11A/B reliability to reduce common mode failure	Eliminate bus failures that could fail associated division during mission	3	10	290K (610K)	680K (1.4M)	>1.8M
XOINEO Implement hardware change to improve reliability of containment injection for sequences where containment pressure is low	Eliminate all failures of operators to perform early injection during AC power scenarios	<1	10	290K (610K)	680K (1.4M)	>1.5M
Implement hardware change in support of automatic initiation of containment injection gravity drain						>1.5M
OHSBO Implement hardware change to improve ability to maintain stable primary & secondary conditions with plant in hot standby	Eliminate all operator failures related to maintaining stable hot standby conditions for extended cooling using the SG	4	5	140K (300K)	335K (705K)	>1.5M[c]
ZZSY12 Provide power system upgrades that would significantly reduce or prevent consequential LOOP events	Eliminate LOOP events that occur subsequent to a plant trip	7	5	140K (300K)	340K (710K)	>2M

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[b]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
CCTE1 Install hardware to improve the reliability of the CCW to reduce the potential for loss of CCW initiators (SAMA 195)[f]	Eliminate PCCW temperature element failures towards the temperature control function	3	5	140K (300K)	340K (710K)	>300K
CCE17 Improve Primary Closed Cooling (PCC) heat exchanger reliability related to tube leakage	The Intent of this SAMA has already been implemented (NextEra 2012a)					
ORHP10 Improve reliability or capability of the operator to restore RCS makeup after support systems are made available	Eliminate failure of all actions to restore high pressure for long term	2	4	110K (230K)	260K (550K)	>5M
SWAFN Improve reliability of SW Cooling Tower SWGR Room Ventilation fans	Eliminate failures related to ventilation fan FN-64 & associated damper & temperature switch when support systems are available	1	3	91K (190K)	210K (445K)	>480K
	Eliminate failures related to ventilation fan FN-51A & associated damper & temperature switch	1	2	74K (160K)	170K (340K)	>1M
	Eliminate failures related to ventilation fan FN-64 & associated damper & temperature switch when support systems are available	1	2	91K (190K)	210K (445K)	>480K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[b]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
XOSMPO Implement hardware modification for automatic control of containment sump recirculation after core melt	Eliminate operator failure to align containment sump recirculation after core melt given recovery of CBS	<1	3	61K (130K)	140K (230K)	>1.5M
CISPRE Install containment leakage monitoring system	Eliminate all pre-existing small & large containment leakage events	0	<1	4K (12K)	10K (27K)	50K to 100K
NOSBO1 Install additional DG to improve overall reliability of onsite emergency power	Elimination of all SBO events	22	6	220K (470K)	525K (1.1M)	>2M
OSEPS Implement hardware change in support of auto closure of supplemental electrical power system (SEPS) breaker to replace operator action	Eliminate operator failures associated with align & load the SEPS DGs	8	2	64K (135K)	151K (320K)	>750K
SEPS Install or modify a SEPS DG to substantially improve reliability of DG start & run failures	Eliminate SEPS DG hardware failures	6	2	63K (130K)	148K (310K)	>2M
OC12 Implement hardware modification (additional signals or remote capability) to allow closure of MOV CS-V-167	This SAMA is address by SAMA 193 and SAMA analysis case CSV167 (NextEra 2012a)					
CSV167 Implement hardware change to eliminate MOV AC power dependencies (SAMA 193)	Eliminate operator failure to close CIV CS-V-167 locally.	0	5	86K (180K)	200K (420K)	>300K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)(b)		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
TDAFW Install additional steam driven EFW pump	Eliminates all failures of the motor-driven EFW independent of AC power	5.3	12	360K (750K)	835K (1.75M)	>2M
OTS10 Implement hardware change to improve reliability of SGTR control to eliminate or reduce operator failure to terminate safety injection (SI)	Eliminate operator failure to terminate SI	3	1	26K (55K)	61K (130K)	>300K
OLPR Implement hardware change to improve reliability of ECCS transfer to long-term recirculation	Eliminate operator failure to complete transfer of RHR/LHSI to long-term recirculation following a LOCA	3	0	12K (25K)	27K (58K)	>100K
OHSB670 Implement hardware change to improve ability to maintain stable & secondary conditions with SG cooling with plant in hot standby during CR fire events	Eliminate operator failures related to evacuation & control at the remote safe shutdown panel after fire-induced transients & LOCAs	3	1	29K (61K)	68K (140K)	>420K
OSGLC0 Implement hardware change to improve operator reliability or provide automatic feature to control SG levels using the EFW discharge pathway	Eliminate operator failures related to controlling SG level via a EFW SUFP and EFW with the EFW discharge & SUFP with the MFW discharge	2	1	29K (62K)	68K (140K)	>500K
SWGE561 Improve 4 KV emergency Bus E6 reliability to eliminate potential for bus fault	Eliminate Bus 5 and 6 random failures in the initiating event model or eliminate associated power division failure or both (d)	6	3	100K (220K)	240K (510K)	>1.2M

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[b]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
XOEFW Implement hardware change to improve operator reliability to feed a failed SG during a SGTR	Eliminate operator failures related to feeding the SG to back pressure the leak	0	1	21K (44K)	50K (100K)	>500K
ORWMZ Implement hardware change to improve operator reliability or provide automatic feature to throttle ECCS RCS to minimize leak for small break LOCA (SLOCA) and ISLOCA sequences	Eliminate operator failure to throttle ECCS flow for scenarios where the containment sump Is not available during SLOCA or ISLOCA	2	0	15K (32K)	35K (74K)	>500K
ORWCD1 Implement hardware change to improve operator reliability or provide automatic features to cool & depressurize the RCS to minimize leak for SLOCA and ISLOCA sequences	Eliminate operator failure control RCS cooldown & depressurization in scenarios where the containment sump is not available during SLOCA & ISLOCA	<1	0	5.3K (11K)	12K (26K)	>500K
ORWLT1 Implement hardware change to improve operator reliability or provide automatic features to maintain stable plant conditions for extended SG cooling after a LOCA or SGTR	Eliminate operator failure to maintain stable primary & secondary conditions to extend SG cooling following SLOCA, ISLOCA, or ISLOCA[e]	<1	0	5.3K (11K)	11K (24K)	>500K
ORWIN Implement hardware change to improve operator reliability or provide automatic feature to initiate RWST makeup	Eliminate operator failure to initiate makeup to the RWST to extend ECCS injection during SLOCA & ISLOCA with recirculation failed	<1	0	4K (8.4K)	9.3K (20K)	>500K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)(b)		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
PS40XA Implement hardware change to improve reliability of the low-pressure permissive signal need to align RHR suction	Eliminate failure of Train A & B low-pressure permissive signals	2	0	9K (20K)	21K (44K)	>500K
RCVR Implement hardware change to improve RHR Train A suction relief valve opening on demand	Eliminate failures of both RHR Train A relief valves to open & reclose	<1	2	24K (50K)	55K (120K)	>500K
CST01 Implement hardware & procedural changes to improve reliability of makeup to CST for long-term SG cooling	Eliminate failures of condensate storage tank (CST) source for EFW	1	1	35K (73K)	81K (170K)	>500K
SWOC6 Implement hardware & procedural changes to improve reliability of transferring SW from the ocean to the cooling tower	Eliminate failure to transfer SW from the ocean to the cooling tower	<1	1	28K (59K)	66K (140K)	>1.5M
SWA6 Implement hardware changes to improve reliability of the SW cooling tower SWGR ventilation	Eliminate failure to transfer SW from the ocean to the cooling tower	<1	1	22K (46K)	52K (110K)	>240K
OFCR0 Implement hardware and procedural changes to improve operator capacity to restore PCCW at the remote shutdown panel	Eliminate failure to restore PCCW at the remote shutdown panel	<1	1	27K (56K)	62K (130K)	>200K
SW64 Implement hardware changes to reduce the probability of spurious SW intake return valve opening	Reduces to a low probability that SW intake return valve spuriously opens	<1	1	25K (52K)	58K (120K)	>300K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[b]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
SW7071C Implement hardware changes to improve reliability of SW cooling tower pump or SWGR room ventilation fans to reduce potential for common mode failure	Eliminate failure of CW cooling tower pump or SWGR room ventilation fans when support systems are available	1	3	84K (180K)	200K (410K)	>480K
EA180C Implement hardware changes to improve reliability of the emergency air handing ventilation fans by eliminating potential for common mode failure	Eliminate failure of emergency air handing ventilation fans when support systems are available	1	2	58K (120K)	140K (285K)	>480K
SW51C Implement hardware changes to improve reliability of SW cooling tower fans to reduce potential for common mode failure	Eliminate failure of SW cooling tower fans when support systems are available	1	3	87K (180K)	205K (430K)	>1M
E7T Implement hardware changes to reduce or eliminate impact of 0.7 g seismic events	Eliminate the 0.7 g seismic initiator	8	2	77K (160K)	180K (380K)	>500K
NOLOSP Implement hardware changes to reduce the risk of weather-related loss of system pressure (LOSP)	Eliminate the LOSP initiator	18	17	530K (1.2M)	1.2M (2.7M)	>3M
F4TREL Provide analysis & hardware changes to protect relay room structure from postulated turbine bay flooding due to an HELB	Eliminate the HELB flooding initiator in the turbine bay	5	1	46K (97K)	110K (225K)	>300K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[b]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
NOSGTR Install upgrades that would reduce or eliminate SGTR	Eliminate the SGTR initiator in addition to pressure and thermo-induced tube rupture	5	2	67K (140K)	160K (330K)	>500K
RXT1 Improve overall Seabrook reliability by installing digital control systems to reduce plant trip initiating frequency	Eliminate the plant trip initiator	4	7	205K (430K)	480K (1.0M)	>19M
LOCA05 Implement hardware changes to reduce or eliminate pipe break LOCA events	Eliminate all small, medium, & large pipe break LOCA events	9	2	77K (160K)	180K (380K)	>500K
F1SWCY Implement hardware changes to reduce the risk of SW common return line rupture event	Eliminate the SW common return line rupture event	3	9	260K (550K)	620K (1.3M)	>5M
FIRE1 Implement hardware change to reduce potential for PORV LOCA caused by fire in the control room	Eliminate spurious or fire-induced actuation of the PORV	3	0	14K (31K)	34K (71K)	>100K
FSGBE6 Implement hardware change to reduce potential for loss of electrical Bus E6 caused by fire in SWGR room B	Eliminate fire initiating events in SWGR room B that result in loss of electrical Bus E6	3	1	28K (58K)	65K (140K)	>500K
Implement hardware change to reduce potential for loss of electrical Bus E6 caused by fire in SWGR room A						

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[b]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
LACPA Improve Bus E5 reliability & eliminate or reduce bus faults contributing	Eliminate the loss of the Train A essential 4 KV power (Bus 5E) initiator	3	1	44K (92K)	100K (220K)	>3M
LOCA06 Implement hardware changes to reduce or eliminate ISLOCA risk in the RHR injection path	Eliminate ISLOCA events	<1	3	48K (101K)	110K (240K)	>500K
LOCA05 Implement hardware changes to reduce or eliminate impact of 2.5 g seismically induced LOCA (by installing digital large break LOCA protection system)	Eliminates pipe break LOCAs	9	2	77K (160K)	180K (380K)	>500K
E18T Implement hardware changes to reduce or eliminate impact of 1.8 g seismic transient event	Eliminate the 1.8 g seismic transient initiator	<1	3	48K (100K)	110K (240K)	>500K
NOATWS Implement seismic upgrades to the ATWS system to withstand up to a 2.5 g seismic event	Eliminate the ATWS initiator	4	2	60K (130K)	140K (290K)	>500K[g]
Implement hardware upgrades to ATWS to reduce potential for ATWS with loss of MFW						>500K
NOSLB Install secondary side guard pipes to up to the main steam isolation valves	Eliminate steam line breaks	<1	0	5K (11K)	11K (24K)	>500K
MSSVO Install “gagging device” to close a stuck open MSSV	Eliminate stuck open MSSV initiator	<1	0	1K (2K)	2K (4.5K)	>500K

Analysis case & applicable SAMAs	Modeling assumptions	% Risk reduction		Total benefit ($)[b]		Cost ($)
		CDF	Population dose	Baseline (internal + external)	Baseline with uncertainty	
LOSPP **Implement hardware upgrades to reduce LOSP**	**Eliminate all plant centered LOSP events**	**2**	**2**	**80K (170K)**	**190K (395K)**	**>7M**
F4TFPB **Implement hardware changes to provide flood and spray protection of non-safety bus duct in turbine bay**	**Eliminate all flooding scenarios due to rupture of fire protection piping in the turbine bay impacting offsite power**	**1**	**0**	**14K (30K)**	**33K (70K)**	**>100K[h]**
FCRAC **Implement hardware changes to provide fire protection features to eliminate or reduce the potential for fire on the Main Control Room panel**	**Eliminate all scenarios where fire in the Main Control Room leads to AC power loss**	**1**	**0**	**15K (31K)**	**35K (70K)**	**>100K[h]**
LOC1LG **Implement hardware changes to eliminate or reduce the potential for large LOCA events**	**Eliminate all large LOCAs**	**1**	**0**	**15K (31K)**	**35K (70K)**	**>100K[h]**

[a] SAMAs in bold are potentially cost beneficial. This table summarizes the results of the revised SAMA analysis provided in the 2012 SAMA supplement (NextEra 2012a).

[b] Values in parenthesis are the results of the sensitivity analysis applying a multiplier of 2.1 to account for the additional risk of seismic events (NextEra 2011a).

[c] In response to an NRC staff RAI, NextEra clarified that the cost reported in the "Expected Cost" column of the 2012 SAMA supplement (NextEra 2012a) was incorrect, but it was reported correctly (i.e., $1.5M) in the "Evaluation" column (NextEra 2012b).

[d] In response to an NRC staff RAI, NextEra clarified that PRA case SWGE61 eliminated both the initiating and basic events associated with 4 kV essential buses E5 and E6 (NextEra 2012b).

[e] In response to an NRC staff RAI, NextEra clarified that PRA case ORWLT1 applied to small LOCA, interfacing LOCA, and SGTR (NextEra 2012b).

[f] In response to an NRC staff RAI, NextEra clarified that PRA case CCTE1 addresses both the reliability of PCCW and loss of CCW as an initiator (NextEra 2012b).

[g] In response to an NRC staff RAI, NextEra clarified that the cost of PRA case NOATWS reflects structural upgrades to reactor internals to reduce seismic capacity as well as non-seismically related reactor internals and emergency boration system upgrades (NextEra 2012b).

[h] NextEra explained in a telephone clarification meeting (NRC 2012b) that $100K is a nominal value used because of the very low calculated benefit. This value reflects the minimum cost of a hardware change.

As indicated in Section F.3.2, in response to an NRC staff RAI, NextEra identified and evaluated a SAMA to make "seismic upgrades to the CST" (NextEra 2011a). This SAMA was estimated to have an implementation cost of more than $100,000. NextEra performed a bounding analysis of the benefit of this SAMA by assuming that it eliminated structural failures of the CST during all seismic-initiating events. The total baseline benefit (using a 7 percent real discount rate) was estimated to be $1,000 and, after accounting for uncertainties, to be $2,000. Based on this result, NextEra concluded that this SAMA was not cost beneficial in either the baseline or the uncertainty analysis. **This SAMA was not re-evaluated in the 2012 SAMA supplement (NextEra 2012a). However, based on the very low potential benefit for this SAMA, the** NRC staff concludes that this SAMA would not be cost beneficial **even after accounting for the higher MACR in the 2012 SAMA supplement, which is about a factor 3.7 increase over the MACR presented in the ER, and** after applying the multiplier of 2.1 to account for the additional risk from seismic events.

Also, in response to an NRC staff RAI, NextEra provided a Phase II evaluation of the following SAMAs, which were originally screened in the Phase I evaluation (NextEra 2011a, 2011b):

- SAMA 79—install bigger pilot operated relief valve so only one is required,
- SAMA 84—switch for EFW room fan power supply to station batteries,
- SAMA 105—delay containment spray actuation after a large LOCA, and
- SAMA 191—remove the 135 °F temperature trip of the PCCW pumps.

The **2012 SAMA supplement (NextEra 2012a) evaluated** these SAMAs **(**in Table F-6**), and** determined **them** to not be cost beneficial in either the baseline or uncertainty analysis or in the sensitivity analysis applying the **seismic** multiplier **of 2.1**.

As indicated in Section F.3.2, in response to an NRC staff RAI, NextEra provided an evaluation of the following two SAMAs identified as a result of its review of the cost-beneficial SAMAs from prior SAMA analyses for five Westinghouse four-loop PWR sites (NextEra 2011a):

- SAMA "procedure change to ensure that the RCS cold leg water seals are not cleared" has an estimated implementation cost of $15,000 to $20,000. NextEra performed a bounding analysis of the benefit of this SAMA by assuming that it eliminated all thermally induced SGTR events (Analysis Case XSGTIS). The total baseline benefit (using a 7 percent real discount rate) was estimated to be less than $1,000 and, after accounting for uncertainties, to be less than $1,000. Based on this result, NextEra concluded that this SAMA was not cost beneficial in either the baseline or the uncertainty analysis. NextEra also concluded that this SAMA would not be cost beneficial after applying the multiplier of 2.1 to account for the additional risk from seismic events (NextEra 2011b). **This SAMA was not re-evaluated in the 2012 SAMA supplement (NextEra 2012a). However, based on the very low potential benefit for this SAMA, the NRC staff concludes that this SAMA would not be cost beneficial even after accounting for the higher MACR in the 2012 SAMA supplement, which is about a factor 3.7 increase over the MACR presented in the ER.**

- SAMA "installation of redundant parallel service water valves to the EDGs" **was** estimated **to have an** implementation cost **similar to SAMA 161 (NextEra 2011b), or $2** million (NextEra **2012a). In response to RAIs on the ER,** NextEra performed a bounding analysis of the benefit of this SAMA by assuming that it eliminated all SBO events **(SAMA analysis case NOSBO1). It** concluded that this SAMA was not cost beneficial either **in** the **baseline (using a 7 percent real discount rate) nor after accounting for uncertainties and the seismic risk** multiplier of 2.1 **(NextEra 2011a,** 2011b). **This SAMA was not re-evaluated in the 2012 SAMA supplement (NextEra 2012a). However, using the benefit results for SAMA analysis case NOSBO1 provided in Table F-6, the NRC staff estimates** total baseline benefit (using a 7 percent real discount rate **and the seismic multiplier of 2.1)** to be

$470,000 and, after accounting for uncertainties, to be **$1.1 million**. The NRC staff concludes that this **SAMA is not cost beneficial.**

Based on review of the ER (NextEra 2010), the NRC staff noted that the evaluation of SAMA 80, "provide a redundant train or means of ventilation," assumes removal of HVAC dependence for CS, SI, RHR, and CBS pumps. The NRC staff asked NextEra to provide an evaluation of a SAMA to remove the HVAC dependency for just the highest risk system (NRC 2010a). In response to the RAI, NextEra explained that the estimated implementation cost to install a redundant HVAC train to **either a single ECCS pump/system or multiple** ECCS pumps and systems was **estimated** to be greater than $500,000**. NextEra further noted that this cost estimate is significantly** greater than the **estimated benefit,** after accounting for uncertainties and **the seismic multiplier of 2.1, and which conservatively assumes elimination of 100 percent of the ECCS dependency on HVAC during long-term recirculation sequences. The analysis of this SAMA was updated in the 2012 SAMA supplement (NextEra 2012a), which shows a maximum benefit of $750,000, after accounting for uncertainties and the seismic multiplier of 2.1) and an updated cost estimate of greater than $1 million. NextEra points out (NextEra 2012a) that this cost is judged to be comparable to other plants that do not have this redundancy. The NRC staff concludes that this SAMA has been adequately addressed**.

The NRC staff notes that all of the potentially cost-beneficial SAMAs (SAMAs 157, **164,** 165**, 172**, 192, 193, and **195**) identified in **the 2012 SAMA supplement (NextEra 2012a)** are included within the set of SAMAs that NextEra plans to enter into the Seabrook long-range plan development process for further implementation consideration. The NRC staff concludes that, with the exception of the potentially cost-beneficial SAMAs discussed above, the costs of the other SAMAs evaluated would be higher than the associated benefits.

F.7 Conclusions

NextEra compiled a list of 191 SAMAs **in the ER (NextEra 2010) and 4 additional SAMAs in the 2012 SAMA supplement (NextEra 2012a)** based on a review of the most significant basic events from the plant-specific PRA, insights from the plant-specific IPE and IPEEE, review of other industry documentation, and insights from Seabrook personnel. A qualitative screening removed SAMA candidates that had modified features not applicable to Seabrook due to design differences, that were determined to have already been implemented at Seabrook or Seabrook meets the intent of the SAMA, or that could be combined with another similar SAMA under consideration. Based on this screening, 117 SAMAs were eliminated, leaving **78** ~~74~~ candidate SAMAs for evaluation.

For the remaining SAMA candidates, more detailed design and cost **estimates** were developed, as shown in Table F-4. The cost-benefit analyses showed that ~~two~~**three** of the SAMA candidates were potentially cost beneficial in the baseline analysis (SAMAs 157, 165, and **192**). NextEra performed additional analyses to evaluate the impact of parameter choices and uncertainties on the results of the SAMA assessment. As a result, ~~no~~**three** additional SAMAs were identified as potentially cost beneficial in the **2012 SAMA supplement (SAMAs 164, 172, and 195).** In **addition, NextEra performed a sensitivity analysis accounting for the** additional **risk of seismic events and identified one additional SAMA (SAMA** 193) as being potentially cost beneficial. NextEra has indicated that all ~~four~~**seven** potentially cost-beneficial SAMAs would be entered into the Seabrook long-range plan development process for further implementation consideration.

The NRC staff reviewed the NextEra analysis and concludes that the methods used and their implementation were sound. **In reviewing insights from plant-specific risk studies, the SAMA evaluation included explicit consideration of external as well as internal hazards.** The treatment of SAMA benefits and costs support the general conclusion that the SAMA evaluations performed by NextEra are reasonable and sufficient for the license renewal submittal.~~ Although the treatment of SAMAs for external events was somewhat limited, the likelihood of there being cost beneficial enhancements in this area was minimized by improvements that have been realized as a result of the IPEEE process and inclusion of a multiplier to account for the additional risk of seismic events.~~

The NRC staff **agrees** with NextEra's identification of **areas in which risk can be further reduced in a cost-beneficial manner through the implementation of the identified,** potentially cost-beneficial SAMAs. Given the potential for cost beneficial risk reduction, the NRC staff agrees that further evaluation of **these** SAMAs by NextEra is **warranted. However,** the applicant **stated that** the **seven**~~four~~ potentially cost-beneficial SAMAs are not aging-related **in that they** do not involve aging management of passive, long-lived systems, structures, and components during the period of extended operation. Therefore, **the NRC staff concludes that the potential cost beneficial SAMAs are not aging related and** they need not be implemented as part of license renewal pursuant to 10 CFR Part 54.

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11. ABSTRACT (200 words or less)
This document supplements the draft supplemental environmental impact statement (DSEIS) which had been prepared in response to an application submitted by NextEra Energy Seabrook, LLC (NextEra) to renew the operating license for Seabrook Station (Seabrook) for an additional 20 years. This supplement incorporates new information that the U.S. Nuclear Regulatory Commission (NRC) staff has obtained since the publication of the DSEIS in August 2011.

This supplement to the DSEIS includes the NRC staff evaluation of revised information provided by NextEra pertaining to the severe accident mitigation alternatives (SAMA) analysis for Seabrook.

In addition, the NRC is taking the opportunity to (1) update the Uranium Fuel Cycle section in light of the June 8, 2012, U.S. Court of Appeals for the District of Columbia Circuit (New York v. NRC, 681 F.3d 471 (D.C. Cir. 2012)) decision to vacate the NRC's Waste Confidence Decision Rule (WCD) (75 Federal Register (FR) 81032, 75 FR 81037) and (2) to provide information on its analysis of new NEPA issues and associated environmental impact findings for license renewal.

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