5.0 Environmental Impacts of Postulated Accidents

Environmental issues associated with postulated accidents are discussed in the *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (GEIS), NUREG-1437, Volumes 1 and 2 (NRC 1996, 1999)[a]. The GEIS includes a determination of whether the analysis of the environmental issue could be applied to all plants and whether additional mitigation measures would be warranted. Issues are then assigned a Category 1 or a Category 2 designation. As set forth in the GEIS, Category 1 issues are those that meet all of the following criteria:

(1) The environmental impacts associated with the issue have been 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 characteristic.

(2) 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).

(3) Mitigation of adverse impacts associated with the issue has been 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.

For issues that meet the three Category 1 criteria, no additional plant-specific analysis is required unless new and significant information is identified.

Category 2 issues are those that do not meet one or more of the criteria for Category 1, and therefore, additional plant-specific review of these issues is required.

This chapter describes the environmental impacts from postulated accidents that might occur during the license renewal term.

5.1 Postulated Plant Accidents

Two classes of accidents are evaluated in the GEIS. These are design-basis accidents (DBAs) and severe accidents, as discussed below.

(a) The GEIS was originally issued in 1996. Addendum 1 to the GEIS was issued in 1999. Hereafter, all references to the “GEIS” include the GEIS and Addendum 1.

5.1.1 Design-Basis Accidents

To receive U.S. Nuclear Regulatory Commission (NRC) approval to operate a nuclear power facility, an applicant for an initial operating license (OL) 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 are provided to prevent and mitigate accidents. The NRC staff reviews the application to determine whether the plant design meets the Commission's regulations and requirements and includes, in part, the nuclear plant design and its anticipated response to an accident.

DBAs are accidents that both the licensee 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. A number 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 preventive and mitigative safety systems of the facility. The acceptance criteria for DBAs are described in 10 CFR Part 50 and 10 CFR Part 100.

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 OL. The results of these evaluations are found in license documentation such as the applicant's Final Safety Analysis Report (FSAR), the staff's Safety Evaluation Report (SER), and the Final Environmental Statement (FES). A licensee 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 maximally exposed individual; as such, changes in the plant environment will not affect these evaluations. 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 and the environmental impacts of those accidents were not examined further in the GEIS.

The Commission 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, design-basis events are designated as a Category 1 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1. This issue, applicable to St. Lucie Units 1 and 2, is listed in Table 5-1. The early resolution of the DBAs

Table 5-1. Category 1 Issue Applicable to Postulated Accidents During the Renewal Term

ISSUE—10 CFR Part 51, Subpart A, Appendix B, Table B-1	GEIS Section
POSTULATED ACCIDENTS	
Design-basis accidents (DBAs)	5.3.2; 5.5.1

makes them a part of the current licensing basis of the plant; the current licensing basis of the plant is to be maintained by the licensee under its current license and, therefore, under the provisions of 10 CFR 54.30, is not subject to review under license renewal.

Based on information in the GEIS, the Commission found that

> The NRC staff has concluded that the environmental impacts of design basis accidents are of small significance for all plants.

Florida Power and Light Company (FPL) stated in its Environmental Report (ER; FPL 2001) that it is not aware of any new and significant information associated with the renewal of the St. Lucie Units 1 and 2 OLs. The staff has not identified any significant new information during its independent review of the ER (FPL 2001), the staff's site visit, the scoping process, or its evaluation of other available information. Therefore, the staff concludes that there are no impacts related to DBAs beyond those discussed in the GEIS.

5.1.2 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 site-specific information to conservatively predict the environmental impacts of severe accidents for each plant 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 FESs and were not specifically considered for the St. Lucie site in the GEIS (NRC 1996). However, in the GEIS, the staff did evaluate existing impact assessments performed by NRC and by the industry at 44 nuclear plants in the United States and concluded that the risk from sabotage and beyond design-basis earthquakes at existing nuclear power plants is SMALL. Additionally, the staff concluded that the risks from other external events are adequately addressed by a generic consideration of internally initiated severe accidents.

Based on information in the GEIS, the Commission found that

> The probability weighted consequences of atmospheric releases, fallout onto open bodies of water, releases to groundwater, and societal and economic impacts from severe accidents are small for all plants. However, alternatives to mitigate severe accidents must be considered for all plants that have not considered such alternatives.

Therefore, the Commission has designated mitigation of severe accidents as a Category 2 issue in 10 CFR Part 51, Subpart A, Appendix B, Table B-1. This issue, applicable to St Lucie Units 1 and 2, is listed in Table 5-2.

Table 5-2. Category 2 Issue Applicable to Postulated Accidents During the Renewal Term

ISSUE—10 CFR Part 51, Subpart A, Appendix B, Table B-1	GEIS Sections	10 CFR 51.53(c)(3)(ii) Subparagraph	SEIS Section
POSTULATED ACCIDENTS			
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	L	5.2

The staff has not identified any significant new information with regard to the consequences from severe accidents during its independent review of the ER (FPL 2001), the staff's site visit, the scoping process, or its evaluation of other available information. Therefore, the staff concludes that there are no impacts of severe accidents beyond those discussed in the GEIS. However, in accordance with 10 CFR 51.53(c)(3)(ii)(L), the staff has reviewed severe accident mitigation alternatives (SAMAs) for St. Lucie Units 1 and 2. The results of its review are discussed in Section 5.2.

5.2 Severe Accident Mitigation Alternatives

10 CFR 51.53(c)(3)(ii)(L) requires that license renewal applicants 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 of this consideration is to ensure that 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 for St. Lucie Units 1 and 2; therefore, the remainder of Chapter 5 addresses those alternatives.

5.2.1 Introduction

FPL submitted an assessment of SAMAs for St. Lucie as part of the ER (FPL 2001). This assessment was based on the current St. Lucie Probabilistic Safety Assessment (PSA), a plant-specific offsite consequence analysis performed using the MELCOR Accident Consequence Code System (MACCS), and insights from the St. Lucie Individual Plant Examination of External Events (IPEEE) (FPL 1994). In identifying and evaluating potential SAMAs, FPL considered several SAMA analyses for other plants and advanced light-water reactor designs, including Watts Bar, Calvert Cliffs, Oconee, Turkey Point, and CE System 80+, and other documents that discuss potential plant improvements, such as NUREG-1560 (NRC 1997a). FPL identified and evaluated 169 potential SAMA candidates. This list was reduced to 50 unique SAMA candidates by eliminating SAMAs that either were not applicable to St. Lucie or were already implemented at the plant. FPL assessed the costs and benefits associated with each of the potential SAMAs and concluded that none of the candidate SAMAs evaluated would be cost-beneficial for St. Lucie.

Based on a review of the SAMA assessment, the NRC issued a request for additional information (RAI) to FPL by letter dated May 7, 2002 (NRC 2002a). Key questions concerned: differences between the PSA used for the SAMA analysis and earlier risk assessments for St. Lucie, the potential impact of uncertainties and external event initiators on the study results, detailed information on several candidate SAMAs, and the applicability of some SAMAs proposed at another Combustion Engineering plant. FPL submitted additional information on June 25, 2002, in response to the RAIs (FPL 2002a). In these responses, FPL included supplemental tables showing the impacts of uncertainties, risk reduction worth importance measures, results of sensitivity analysis, and additional information on specific SAMAs. FPL provided further information during a teleconference on July 15, 2002, clarifying the remaining issues (NRC 2002b). In these responses, FPL provided additional information on its use of importance analysis and cut set information, on regional population projections, and on use of the MAAP code in its consequence assessment. FPL's responses addressed the staff's concerns and reaffirmed that none of the SAMAs would be cost-beneficial.

An assessment of SAMAs for St. Lucie is presented below.

5.2.2 Estimate of Risk for St. Lucie Units 1 and 2

FPL's estimates of offsite risk at St. Lucie are summarized in Section 5.2.2.1. The summary is followed by the staff's review of FPL's risk estimates in Section 5.2.2.2.

5.2.2.1 FPL's Risk Estimates

Two distinct analyses are combined to form the basis for the risk estimates used in the SAMA analysis: (1) the St. Lucie Level 1 and 2 PSA model, which is an updated version of the St. Lucie Individual Plant Examination (IPE) (FPL 1993), and (2) a supplemental analysis of offsite consequences and economic impacts (essentially a Level 3 PSA model) developed specifically for the SAMA analysis. The St. Lucie PSA, dated April 2001, is indicated in the ER (FPL 2001) to be more advanced than the St. Lucie IPE submittal of 1993 (FPL 1993) and is considered a "living" plant risk model that reflects periodic updates to incorporate (1) additional data on equipment performance, (2) changes in plant configuration, and (3) PSA model refinements.

The baseline core damage frequencies (CDFs) for the purpose of the SAMA evaluation are approximately 3.0 x 10^{-5} per reactor-year (ry) and 2.4 x 10^{-5}/ry for St. Lucie Units 1 and 2, respectively. These CDFs are based on the risk assessment for internally initiated events, including internal floods. These values represent only small changes from the original IPE CDF values of 2.3 x 10^{-5}/ry and 2.6 x 10^{-5}/ry for St. Lucie Units 1 and 2, respectively. Although FPL did not include the contribution of risk from external events within the St. Lucie risk estimates, it did account for the potential risk reduction benefits associated with external events by applying a factor of 2 margin in the SAMA screening process. It is FPL's position that this approach is conservative because the external events contributions to core damage are small relative to the internal events contributions (FPL 2001). This is discussed further in Section 5.2.2.2.

The breakdown of CDFs is provided in Table 5-3. It is noted that the total CDFs in Table 5-3 are slightly different than the total CDFs given above. This is because the values are based on the use of a top event model, which was also used for the purpose of screening SAMAs. The top event model accounts for 95 percent of the CDF for Unit 1 and 99 percent of the CDF for Unit 2. As shown in Table 5-3, containment bypass events (i.e., interfacing system loss-of-coolant accident [ISLOCA] and steam generator tube rupture [SGTR]) contribute about 13 percent and 24 percent to the total internal events CDF for Units 1 and 2, respectively. Transients (including loss-of-offsite power [LOOP] and anticipated transient without scram [ATWS]) contribute about 35 percent and 20 percent, respectively. The contribution of loss-of-coolant accidents (LOCAs) to the total CDFs is large at both plants (29 percent and 32 percent, respectively). The station blackout (SBO) contribution to the transients was not explicitly provided in the submittal; however, in response to a RAI, FPL stated that the LOOP sequences are predominantly SBO sequences (FPL 2002a). The CDFs that were used in the SAMA analysis and that are cited here are best-estimate values. The uncertainty analysis for the updated PSA indicates 95 percent confidence level (upper) CDF values of 6.15 x 10^{-5}/ry and 6.11 x 10^{-5}/ry for Units 1 and 2, respectively. The impact of this uncertainty on the SAMA analysis is discussed in Section 5.2.6.2.

Table 5-3. St. Lucie Core Damage Frequency[a]

Initiating Event	Frequency (per reactor-year)		% Contribution to CDF	
	Unit 1	Unit 2	Unit 1	Unit 2
Loss of Offsite Power (LOOP)/Station Blackout (SBO)	4.63×10^{-6}	2.67×10^{-6}	16	11
Transients	4.55×10^{-6}	1.84×10^{-6}	16	8
Anticipated Transient Without Scram (ATWS)	8.23×10^{-7}	3.31×10^{-7}	3	1
Loss-of-Coolant Accident (LOCA)	8.22×10^{-6}	7.82×10^{-6} [b]	29	32[b]
Interfacing Systems LOCA (ISLOCA)	2.89×10^{-6}	5.64×10^{-6}	10	23
Steam Generator Tube Rupture (SGTR)	9.58×10^{-7}	2.78×10^{-7}	3	1
Internal floods	5.00×10^{-7}	5.00×10^{-7}	2	2
Others	6.03×10^{-6}	5.22×10^{-6} [b]	21	22[b]
Total CDFs (from internal events)	**2.86×10^{-5}**	**2.43×10^{-5}**	**100**	**100**

(a) CDF calculated using a single top event model that included all plant damage states and containment bypass sequences.
(b) The Unit 2 LOCA value, originally provided in the FPL RAI responses (FPL 2002a), was in error and subsequently corrected in a communication with NRC (FPL 2002b).

The major difference in the CDFs for St. Lucie Units 1 and 2 is attributed to the following:

- Unit 2 has larger power-operated relief valves (PORVs), thus only one PORV is required for once-through cooling. This is the main reason Unit 1 has a larger SGTR CDF than Unit 2.

- Unit 2 has a larger capacity condensate storage tank than Unit 1. Thus, Unit 1 has a slightly higher contribution from long-term decay heat removal related scenarios such as transients.

- The Unit 2 shutdown cooling line has one more configuration of an ISLOCA path due to crosstie capability. This increases the ISLOCA frequency for Unit 2.

The CDF results were obtained using two cases for 4.16-kV AB-bus alignment. Case 1 is when the AB-bus is aligned to the A-bus, and Case 2 is when the AB-bus is aligned to the B-bus. FPL states that the SAMA evaluation uses the most conservative cases for the baseline risk model, which are Case 2 for Unit 1 and Case 1 for Unit 2 (FPL 2001).

The Level 2 PSA model is based on the containment event tree and source terms from the IPE (FPL 1993). The conditional probabilities associated with each release category are provided in Table E.1-1 of the ER (FPL 2001). The fission product release fractions and characteristics for each release category are provided in Table E.2-1 of the ER.

The offsite consequences and economic impact analyses use the MELCOR Accident Consequence Code System 2 (MACCS2) code, Version 1.12, to determine the offsite risk impacts on the surrounding environment and public. Inputs for this analysis include plant-specific and site-specific input values for core radionuclide inventory, source term and release fractions, meteorological data, projected population distribution, emergency response evacuation modeling, and economic data.

FPL estimated the dose to the population within 80 km (50 mi) of the St. Lucie site to be approximately 0.15 person-Sv (15 person-rem) per year for Unit 1 and 0.14 person-sievert (14 person-rem) per year for Unit 2. The breakdown of the total population dose by containment release mode is summarized in Table 5-4. ISLOCAs dominate the population dose risk at St. Lucie. The ISLOCAs are followed in contribution by late containment failure.

Table 5-4. Breakdown of Population Dose by Containment Release Mode

	Population Dose Person-Sv (Person-Rem) Per Year	
Containment Release Mode	**Unit 1**	**Unit 2**
SGTR (steam generator tube rupture) (Late and Early)	0.009 (0.9)	0.001 (0.1)
Interfacing Systems LOCAs (loss-of-coolant accidents)	0.087 (8.7)	0.113 (11.3)
Early containment failure	~0 (~0.0)	~0 (~0.0)
Late containment failure	0.057 (5.7)	0.026 (2.6)
No containment failure	0.0 (0.0)	0.0 (0.0)
Total	0.153 (15.3)	0.14 (14.0)

5.2.2.2 Review of FPL's Risk Estimates

FPL's determination of offsite risk at St. Lucie is based on the following three major elements of analysis:

- the Level 1 and 2 risk models that form the bases for the 1993 IPE and 1994 IPEEE submittals (FPL 1993, 1994)

- the major modifications to the IPE model that have been incorporated in the St. Lucie PSA

- the MACCS2 analyses performed to translate fission product release frequencies from the Level 2 PSA model into offsite consequence measures.

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

The staff's review of the St. Lucie IPE is described in an NRC report dated July 21, 1997 (NRC 1997b). In that review, the staff evaluated the methodology, models, data, and assumptions used to estimate the CDF and characterize containment performance and fission product releases. The staff concluded that FPL's analysis met the intent of Generic Letter 88-20 (NRC 1988); that is, the IPE was of adequate quality to be used to look for design or operational vulnerabilities. The staff's review primarily focused on the licensee's ability to examine St. Lucie Units 1 and 2 for severe accident vulnerabilities and not specifically on the detailed findings or quantification estimates. Overall, the staff concluded that the St. Lucie IPE was of adequate quality to be used as a tool in searching for areas with high potential for risk reduction and to assess such risk reductions, especially when the risk models are used in conjunction with insights, such as those from risk importance, sensitivity, and uncertainty analyses.

A comparison of risk profiles between the original IPE, which was reviewed by the NRC staff, and the PSA used in the SAMA analysis indicates a small increase in the St. Lucie Unit 1 CDF and small decrease in the St. Lucie Unit 2 CDF. The specific changes to the St. Lucie PSA include (FPL 2001):

- Changed to a "one-top" model rather than solving individual sequences.

- Updated software to allow use of a recovery rule file that allows automatic application of recovery rules consistently to every appropriate cut set.

- Refined common-cause failure modeling by the use of a basic event for common causes only. The original model normally used an "A" train event with the common-cause factor. This practice overemphasized the importance of the "A" train components, because all common-cause failures were tied to "A" (and none to "B" train components).

- Added test and maintenance basic events for various components as further improvements to the model.

- Improved treatment of reactor coolant pump (RCP) seal failures depending on operator action or failure to act, according to the latest Combustion Engineering Owners Group (CEOG) information.

- Updated LOCA and Main Steam Line Break initiating event frequencies per the latest CEOG methodologies.

- Updated the Unit 2 ISLOCA analysis to reflect a Unit 2 design change. This change increased the calculated probability of ISLOCA while reducing the probability of pressure locking of the shutdown cooling isolation valves (which would prevent the use of shutdown cooling).

The changes from the IPE version to the current April 2001 version appear to be reasonable and have a relatively small effect on PSA results.

In an RAI, the staff questioned whether the current St. Lucie PSA has been subjected to peer review (NRC 2002a). In response to the RAI, FPL noted that the PSA conforms to the FPL Quality Assurance Program procedures and the FPL Reliability and Risk Assessment Group standards. Further, the Level 1 model was compared to the CEOG plants via the CEOG PSA subcommittee cross comparison project (FPL 2002a). While these activities do not constitute a thorough external peer review, they do enhance the quality of a PSA.

The IPE and updated CDF values for the two FPL units are lower than most of the original IPE values estimated for other pressurized water reactors (PWRs) with a large dry containment. Figure 11.6 of NUREG-1560 shows that the IPE-based total internal events CDF for Combustion Engineering plants ranges from 1×10^{-5} to 3×10^{-4}/ry (NRC 1997a). While it is recognized that other plants have reduced the values for CDF subsequent to the IPE submittals, due to modeling and hardware changes, the CDF results for St. Lucie confirm that the overall risks from these units are lower than or comparable to other plants of similar vintage and characteristics.

FPL submitted an IPEEE by letter dated December 15, 1994 (FPL 1994), in response to Supplement 4 of Generic Letter 88-20 (NRC 1999). FPL did not identify any fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, or other external events. The St. Lucie hurricane, tornado, and high winds analyses show that the plant is adequately designed or procedures exist to cope against the effects of these natural events. Additionally, the St. Lucie IPEEE demonstrated that transportation and nearby facility accidents were not considered to be significant vulnerabilities at the plant. However, a number of areas were identified for improvement in both the seismic and fire areas. In a letter dated January 25, 1999 (NRC 1999), the staff concluded that the submittal met the intent of Supplement 4 to Generic Letter 88-20, and that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities.

The ER (FPL 2001) acknowledges that the methods used for the St. Lucie IPEEE do not provide the means to determine the numerical estimates of the CDF contributions from seismic initiators (i.e., the seismic IPEEE uses a reduced scope margins method emphasizing plant walkdown) and fire initiators (i.e., the fire IPEEE uses the Fire Vulnerability Evaluation method). However, the risk associated with external events at St. Lucie is very low. The IPEEE fire CDF estimates are considered by FPL to be extremely conservative and overestimate the fire risk for screening purposes (FPL 2001). FPL states in the ER that recent preparatory work in support of OL amendments to extend the allowed outage time (AOT) for emergency diesel generators (EDGs) has refined and revised the fire risk estimates for the cable spreading rooms and the control rooms, and the current estimates are now about two orders of magnitude lower than reported in the original St. Lucie IPEEE (FPL 1994). Furthermore, as part of the OL amendment, FPL committed to perform several actions that would ensure low risk due to external and internal fire events for each unit if an EDG is to be removed from service for maintenance for an extended AOT (i.e., more than 72 hours) during Modes 1, 2, and 3. In addition, the submittal states that improvements continue to be made in St. Lucie Units 1 and 2 fire protection features as a result of ongoing (10 CFR Part 50) Appendix R evaluations. Accordingly, the staff finds that the FPL fire assessment is adequate for the purpose of the SAMA review and that the fire vulnerabilities at St. Lucie are not major contributors to plant risk.

Because of the small expected contribution of external events to the overall risk profile for St. Lucie, the risk reduction estimates for the SAMAs were evaluated based on consideration of the internal events risk profile. However, in the SAMA screening process described in Section 5.2.3.1, FPL screened out SAMAs from further consideration only if their implementation cost would be much greater than twice the estimated benefit (based on internal events). This provides a factor of two margin in the analysis. The contribution of external events to total risk would be bounded by this factor of two if (1) the total contribution from external events is a small fraction of the contribution from internal events, and (2) there are no external event vulnerabilities that can be eliminated or mitigated by cost-effective SAMAs. FPL presents an adequate case that the external risk contribution is relatively small. FPL also states that a search for SAMAs yielded no SAMA that would provide redundancy to plant safe shutdown capabilities in order to reduce the external event contribution. Accordingly, the staff concludes that FPL's consideration of external events within the SAMA analysis is acceptable.

The staff reviewed the process used by FPL to extend the containment performance (Level 2) portion of the PSA to an assessment of offsite consequences (essentially a Level 3 PSA). This included consideration of the source terms used to characterize fission product releases for each containment release category and the major input assumptions used in the offsite consequence analyses. The MACCS2 code was used to estimate offsite consequences.

Plant-specific input to the code includes the St. Lucie reactor core radionuclide inventory, emergency evacuation modeling, release category source terms from the St. Lucie IPE, site-specific meteorological data, and projected population distribution within a 80-km (50-mi) radius for the year 2025. This information is provided in Appendix E.2 of the ER (FPL 2001).

The applicant used source term release fractions for 48 different release modes defined for the St. Lucie site. Forty-five of the release modes were previously identified in the St. Lucie IPE. Three additional containment bypass release modes were added: two SGTR cases and one ISLOCA case. The staff reviewed FPL's source term estimates for the major release categories and, with the exception of SGTR noted below, found the release fractions to be consistent with those of like plants and of expected magnitudes when considering early versus late containment failures and rupture versus leak-type failures. The staff questioned FPL regarding the release fractions for SGTR events, which were relatively low and did not include tellurium releases (NRC 2002a). FPL indicated that large amounts of radionuclides (including all tellurium) are retained in the intact containment after vessel failure, thus mitigating release to the environment (FPL 2002a). The staff finds this explanation to be reasonable, and further notes that since the SGTR contribution to CDF is relatively low for St. Lucie (3 percent for Unit 1 and 1 percent for Unit 2), higher release fractions for the SGTR sequences than those estimated by FPL would not have a significant impact on the plant risk. The staff concludes that the assignment of source terms is acceptable for use in the SAMA analysis.

The applicant used site-specific meteorological data processed from hourly measurements for the 1999 calendar year as input to the MACCS2 code. Supplementary information derived from meteorological data obtained from the National Climatic Data Center of the National Oceanographic and Atmospheric Administration (NOAA) for Vero Beach Airport was used where data were missing in the source file. A sensitivity analysis was performed using meteorological data from 1998. The impact on population dose was a 10 percent decrease. Therefore, the staff considers use of the 1999 data in the base case to be conservative.

The population distribution the applicant used as input to the MACCS2 analysis was initially prepared using the computer program SECPOP90 (NRC 1997c). The output from SECPOP90 is a file based on a reference database for the specified site. The applicant extrapolated population projections from the years 1990 and 2015 to year 2025 using the U.S. Census Bureau (USCB) data. The MACCS2 calculations were based on the population in year 2025 because 2025 was the latest data produced by the USCB and because 2025 is the midterm year for the Unit 1 license renewal period. It is noted that the midterm year for the license renewal period for Unit 2 would be 2033. If a year later than 2025 were used, it is expected that the population dose would increase proportionately with the projected increase in population. Based on information provided in Section 2.5 of the ER, the population in two areas surrounding the plant is expected to increase at an average rate of 1.5 percent per year. If the year 2033 was chosen for the population projection, an increase in the population (over the base case year 2025 population dose) of approximately 13 percent would be expected. The applicant, in Section E.2.4.2 of the ER (FPL 2001), presents sensitivity analyses that show a 2.5 percent and 10 percent increase in population results in approximately a 3 percent and 11 percent increase in the population dose. Thus, the population dose estimates for Unit 2 would be approximately 15 percent higher if the dose estimates were based on the population in 2033 rather than 2025. FPL pointed out that other conservative aspects of the analysis more than compensate for this apparent non-conservatism (NRC 2002b). The 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. It was assumed that 100 percent of the population would move at an average speed of approximately 1.8 m/s (6 ft/s) with a delayed start time of 7,200 seconds with no sheltering. The results of a sensitivity analysis presented in Section E.2.4.2 of the ER (FPL 2001) show that if only 95 percent of the people within the evacuation zone would participate in the evacuation, there would be only about a 1 percent increase in population dose. 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. Additionally, a sensitivity analysis was performed in which the evacuation speed was reduced to approximately 0.3 m/s (1 ft/s). This resulted in an increase in population dose of about 2 percent. Accordingly, the evacuation assumptions and analysis are deemed reasonable and acceptable for the purposes of the SAMA evaluation.

Much of the site-specific economic data were provided by SECPOP90 (NRC 1997c) and used in the MACCS2 analyses. SECPOP90 contains a database extracted from USCB CD-ROMs (1990 census data), the 1992 Census of Agriculture CD-ROM Series 1B, the 1994 U.S. Census County and City Data Book CD-ROM, the 1993 and 1994 Statistical Abstract of the United States, and other minor sources. These regional economic values were updated to 1999 for nine Florida counties within 80 km (50 mi) of the plant. The staff questioned whether FPL made any adjustments to the analysis to account for higher economic areas in the vicinity of the plant such as resorts (NRC 2002a). In response, FPL stated that the site file prepared for St. Lucie contained updated values (from 1999) for each county including contributions from resort areas (FPL 2002a).

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

5.2.3 Potential Design Improvements

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

5.2.3.1 Process for Identifying Potential Design Improvements

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

- review of plant-specific improvements identified in the St. Lucie Units 1 and 2 IPE and IPEEE

- review of SAMA analyses submitted in support of original licensing and license renewal activities for other operating nuclear power plants

- review of other NRC and industry documentation discussing potential plant improvements, e.g., NUREG-1560 (NRC 1997a), and review of the top 100 cut sets of the updated Level 1 PSA.

Based on this process, an initial list of 169 candidate SAMAs was identified, as reported in Table E.3-1 in Appendix E to the ER (FPL 2001).

FPL performed a qualitative screening of the initial list of SAMAs. SAMAs were eliminated from further consideration at St. Lucie if the SAMA enhancement was for a boiling water reactor, the Westinghouse AP600 design, an ice condenser containment, or for a plant-specific application not applicable to St. Lucie. SAMAs were also eliminated from further consideration if the SAMA had already been implemented at St. Lucie or the plant design meets the intent of the SAMA.

Based on the qualitative screening, 119 SAMAs were eliminated leaving 50 for further evaluation. Of the 119 SAMAs, 29 were eliminated because they were not applicable to St. Lucie, and 90 were eliminated because they already had been implemented at St. Lucie (or the design met the intent of the SAMA). The 50 remaining SAMAs are listed in Table 4.15-2 of the ER (FPL 2001) and were subjected to a final screening and evaluation process.

The final screening process was conducted in two steps: (1) identifying and eliminating those SAMAs whose cost exceeded the maximum attainable benefit (MAB) approximated at $1,382,000, and (2) performing a benefits analysis on the remaining SAMAs. Of the 50 SAMAs, 29 were screened from further evaluation because the SAMA was estimated to have a single unit cost of implementation that exceeded the MAB of $1,382,000. Each of the 21 remaining SAMAs was further evaluated and subsequently eliminated, as described in Sections 5.2.4 and 5.2.6 below.

5.2.3.2 Staff Evaluation

FPL's efforts to identify potential SAMAs focused primarily on areas associated with internal initiating events. The initial list of SAMAs generally addressed the accident categories that are dominant CDF contributors or issues that tend to have a large impact on a number of accident sequences at St. Lucie Units 1 and 2.

The preliminary review of FPL's SAMA identification process raised some concerns regarding the completeness of the set of SAMAs identified and the inclusion of plant-specific risk contributors. The staff also requested specific information about several of the final SAMA candidates. The staff requested clarification regarding the portion of risk represented by the top 100 cut sets and whether an importance analysis was used to confirm the adequacy of the SAMA identification process. A review of the importance ranking of basic events in the PSA has the potential to identify SAMAs that may not be apparent from a review of the top cut sets. In response to the RAI, FPL stated that the top 100 cut sets examined account for about 55 percent of the CDF for Unit 1 and about 68 percent of the CDF for Unit 2 (FPL 2002a). In a follow-up teleconference, FPL clarified that although it did not specifically use the importance measures (risk reduction worth [RRW]) to identify potential SAMAs, it performed a supplementary review of the importance measures, which did not reveal any new SAMAs. FPL indicated that the risk significant basic events are contained in the top 100 cut sets, particularly SGTR and ISLOCA.

The staff questioned FPL about considering lower cost alternatives to a couple of the SAMAs evaluated (NRC 2002a). In response to the RAI, FPL stated that either the design and modification costs for "lower cost alternatives" were prohibitive or the reduction in risk was insufficient to warrant the implementation (FPL 2002a). The staff also questioned FPL about six SAMAs that were proposed at another Combustion Engineering plant and whether those SAMAs might be applicable to St. Lucie (NRC 2002a). In response to the RAI, FPL noted that four of the six planned SAMAs were related to SBO or LOOP. These SAMAs would provide less risk reduction benefit for St. Lucie because St. Lucie is equipped with four EDGs and has cross-tie capability. As for the other two planned SAMAs, one is already addressed by the St. Lucie emergency operating procedures network, and the other involving an improvement to refueling water tank level indication is not applicable because the recirculation actuation system at St. Lucie does not depend on instrument air.

The 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 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 not likely cost less than the least expensive alternatives evaluated, when the subsidiary costs associated with maintenance, procedures, and training are considered.

The staff concludes that FPL used a systematic and comprehensive process for identifying potential plant improvements for St. Lucie Units 1 and 2. While explicit treatment of external events in the SAMA identification process was limited, it is recognized that the absence of external event vulnerabilities reasonably justifies examining primarily the internal events risk results for this purpose.

5.2.4 Risk Reduction Potential of Design Improvements

FPL evaluated the risk-reduction potential of the 21 remaining SAMA candidates that were applicable to St. Lucie. Each SAMA evaluation was performed in a bounding fashion in that the SAMA was assumed to completely eliminate the risk associated with the proposed enhancement. Such bounding calculations overestimate the benefit and are conservative. FPL used two types of evaluations to determine the benefit of the SAMAs: model re-quantification and importance measure analysis. Some of the SAMAs were evaluated by making simple bounding changes to one or more system models and re-quantifying the full model. Some of the SAMAs were more quickly evaluated by examining importance measures such as RRW. In such cases, it was assumed that the benefit is approximately proportional to the reduction in CDF.

For many of the SAMAs, the CDF reduction was estimated from a model (referred to as PDS TOP), which used a single top event that included all plant damage states (PDSs) and containment bypass sequences. This resulted in a manageable number of cut sets and accounted for about 95 percent of the total baseline CDF for Unit 1 and about 99 percent of the total baseline CDF for Unit 2. For specific cases such as SGTR and ISLOCA, full-risk model cases were used.

Seven SAMA evaluation scenarios were developed to accomplish this effort (Cases 1 through 4 plus three cases addressing elimination of ISLOCA, SGTR, and high-pressure safety injection failures). Each of the 21 SAMAs were binned into one of the seven scenarios. (Note that although Case 2 was defined and quantified, all of the SAMAs applicable to this case were screened out prior to the final evaluation. Thus, none of the 21 SAMAs were assigned to this case). Table 5-5 lists the evaluation scenario performed to estimate the risk reduction for each of the 21 SAMAs, the estimated risk reduction in terms of percent-reduction in CDF and person dose, and the estimated total benefit (present value) of the averted risk. The determination of the benefits for the various SAMAs is discussed in Section 5.2.6.

In response to an RAI, FPL considered the uncertainties associated with the calculated CDF, and it was found that if the 95th percentile value of the CDF were to be used in the cost-benefit

Table 5-5. SAMA Cost-Benefit Screening Analysis

Evaluation Scenario and Applicable SAMAs	Assumptions	% Risk Reduction (Unit 1/Unit2)		Total Benefit in $ (Unit 1/Unit2)	Cost (2001 dollars)
		CDF	Population Dose		
Case 1 48-Install a passive containment spray system (CSS)	The containment spray system will be perfectly reliable, thus eliminating those PDSs representing loss of sprays. The logic for CSS injection and recirculation is removed from the fault tree.	0.2 / 0.2	22 / 13	200,400 / 112,200	$20M
Case 2 None	The reactor coolant pump (RCP) seal LOCA does not occur, and the operator does not fail to secure the RCPs. A few logic changes are imposed on the baseline model.	14 / 19	6 / 8	129,700 / 145,700	
Case 3 123-Upgrade chemical and volume control system (CVCS) to mitigate small-small loss-of-coolant accident (LOCA)	Small-small LOCA does not occur. A few logic changes are imposed on the baseline model.	23 / 27	11 / 12	225,300 / 216,600	>>2 x Benefit
Case 4 8-Eliminate RCP thermal barrier dependence on component cooling water (CCW) such that a loss of CCW does not result directly in core damage 10-Create an independent RCP seal injection system, with dedicated diesel 11-Create an independent RCP seal injection system without dedicated diesel 12-Use existing hydro test pump for RCP seal injection 16-Prevent charging pump flow diversion from the relief valves	The RCP seal LOCA does not occur. A few logic changes are imposed on the baseline model.	5 / 6	2 / 3	44,300 / 50,100	8 - >>2 x Benefit 10 - >>2 x Benefit 11 - >>2 x Benefit 12 - >>2 x Benefit 16 - >>2 x Benefit

Table 5-5. (cont'd)

Evaluation Scenario and Applicable SAMAs	Assumptions	% Risk Reduction (Unit 1/Unit 2) CDF	Population Dose	Total Benefit in $ (Unit 1/Unit2)	Cost (2001 dollars)
No ISLOCA	ISLOCA will be eliminated. PDSs that represent ISLOCA are set to zero to represent the impact of eliminating the event	10 / 23	26 / 55	251,500 / 487,400	
89-Install additional instrumentation for interfacing systems LOCA (ISLOCA) sequences					89 - $2.3M
90-Increase frequency of valve leak testing					90 - >>2 x Benefit
95-Ensure all ISLOCA releases are scrubbed					95 - >>2 x Benefit
96-Add redundant and diverse limit switch to each containment isolation valve					96 - >>2 x Benefit
159-Provide auxiliary building (AB) vent/seal structure					159 - >>2 x Benefit
160-Add charcoal filters on the AB exhaust					160 - >>2 x Benefit
No SGTR	All SGTRs will be eliminated. PDSs that represent SGTR (i.e., SGTR1 and SGTR2) are set to zero.	4 / 1	14 / 2	111,300 / 12,600	
80-Improve instrumentation to detect SGTR, or add systems to scrub fission product releases					80 - $9.5M
81-Add other SGTR coping features					81 - >>2 x Benefit
82-Increase secondary-side pressure capacity such that an SGTR would not cause the relief valves to lift					82 - >>2 x Benefit
83-Replace steam generators (SGs) with new design					83 - $100M
85-Establish a maintenance practice that inspects 100% of the tubes in an SG					85 - $500K - $750K per inspection
HPSI	Eliminate HPSI failures	18 / 20	18 / 20	249,100 / 242,400	
13-Replace emergency core cooling system pump motors with air-cooled motors					13 - >>2 x Benefit
117-Provide an additional high-pressure safety injection (HPSI) pump with independent diesel					117 - >>2 x Benefit
118-Install an independent alternating current (AC) HPSI system					118 - >>2 x Benefit

analysis, instead of the best-estimate CDF value, the benefits would be about a factor of 2 greater.

The staff has reviewed FPL'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 staff based its estimates of averted risk for the various SAMAs on FPL's risk-reduction estimates.

5.2.5 Cost Impacts of Candidate Design Improvements

FPL estimated the costs of implementing the 50 SAMAs, which were not initially screened out, through the application of engineering judgment, estimates from other licensees' submittals, and site-specific cost estimates. The cost estimates conservatively did not include the cost of replacement power during extended outages required to implement the modifications, nor did they include contingency costs associated with unforeseen implementation obstacles. Estimates based on modifications implemented or estimated in the past were presented in terms of dollar values at the time of implementation and were not adjusted to present-day dollars. The depth of analysis performed varied depending on the magnitude of the expected benefit. For most of the SAMAs considered, the cost estimates were sufficiently greater than the benefits calculated such that no detailed evaluation was required. Detailed cost estimating was only applied in those situations in which the benefit is significant and application of judgment would be questioned. The minimum cost of making a procedural change (including training) was estimated at $30,000. The minimum hardware modification package was assumed to be $70,000.

The staff reviewed the bases for the applicant's cost estimates. For certain improvements, the staff also compared the cost estimates (presented in Table 4.15-2 of the ER) to estimates developed elsewhere for similar improvements, including estimates developed as part of other licensees' analyses of SAMAs for operating reactors and advanced light-water reactors. A majority of the SAMAs were screened from further consideration on the basis that the expected implementation cost would be much greater than twice the estimated risk-reduction benefit. This is reasonable for the SAMAs considered given the relatively small estimated benefit for the SAMAs (a maximum benefit of about $250,000), and the large implementation costs typically associated with major hardware changes and hardware changes that impact safety-related systems. In previous SAMA evaluations the implementation costs for such hardware changes were generally estimated to be $1 million or more. Where specific cost estimates were provided in the ER (FPL 2001), these were typically obtained from previous licensees' ERs or from other industry submittals, most of which have been previously reviewed by the NRC. Accordingly, the cost estimates were found to be consistent with previous estimates. The staff concludes that the cost estimates are sufficient and appropriate for use in the SAMA evaluation.

5.2.6 Cost-Benefit Comparison

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

- **FPL Evaluation**

The methodology used by FPL was based primarily on NRC's guidance for performing cost-benefit analysis, i.e., NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook* (NRC 1997d). 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. FPL's derivation of each of the associated costs is summarized below.

<u>Averted Public Exposure (APE) Costs</u>

The APE costs were calculated using the following formula:

APE	=	Annual reduction in public exposure (Δperson-rem/ry)
	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 1997d), it is important to note that 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, FPL calculated an APE of approximately $330,000 for the 20-year license renewal period, which assumes elimination of all severe accidents.

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.

For the purposes of initial screening, which assumes all severe accidents are eliminated, FPL calculated an annual offsite economic risk of $42,542 based on the Level 3 risk analysis. This results in a discounted value of approximately $458,000 for the 20-year license renewal period.

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.

FPL derived the values for averted occupational exposure from information provided in Section 5.7.3 of the regulatory analysis handbook (NRC 1997d). Best estimate values provided for immediate occupational dose (3300 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 all severe accidents are eliminated, FPL calculated an AOE of approximately $11,400 for the 20-year license renewal period.

Averted Onsite Costs (AOSC)

Averted onsite costs (AOSC) include averted cleanup and decontamination costs and averted power replacement costs. Repair and refurbishment costs are considered for recoverable accidents only and not for severe accidents. FPL derived the values for AOSC based on information provided in Section 5.7.6 of the regulatory analysis handbook (NRC 1997d).

FPL divided this cost element into two parts – the onsite cleanup and decontamination costs, also commonly referred to as averted cleanup and decontamination costs, and the replacement power cost.

Averted cleanup and decontamination costs (ACCs) 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 the regulatory analysis handbook to be $1.5 billion (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 all severe accidents are eliminated, FPL calculated an ACC of approximately $347,000 for the 20-year license renewal period.

Long-term replacement power costs (RPCs) 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.

For conservatism, FPL based its calculations on the 910-MWe reference plant in NUREG/BR-0184, and did not scale down for the 800-MWe output of St. Lucie. For the purposes of initial screening, which assumes all severe accidents are eliminated, FPL calculated an RPC of approximately $236,000 for the 20-year license renewal period.

Using the above equations, FPL estimated the total present dollar value equivalent associated with completely eliminating severe accidents at St. Lucie to be about $1,382,000 for each unit.

FPL's Results

If the single unit implementation costs were greater than the MAB of $1.38 million, then the SAMA was screened from further consideration. Twenty-nine SAMAs were screened from further consideration in this way. A more refined look at the costs and benefits was performed for the remaining 21 SAMAs. If the expected cost for one of the 21 SAMAs exceeded twice the calculated benefit, the SAMA was considered not to be cost-beneficial. The cost-benefit results for the individual analysis of the 21 SAMA candidates are presented in Table 5-5. As a result, all 50 SAMAs that were evaluated were eliminated because the cost was expected to exceed the estimated benefit.

FPL performed sensitivity analyses to evaluate the impact of parameter choices on the analysis results (FPL 2001, 2002a). The sensitivity analyses included the calculation of candidate SAMA benefits using a 3-percent discount rate as recommended in NUREG/BR-0184 (NRC 1997d). This sensitivity case resulted in less than a factor of 1.4 increase in the benefit calculation. Thus, the FPL conclusion that none of the candidate SAMAs would be cost-beneficial remains unchanged.

- **Staff Evaluation**

The cost-benefit analysis performed by FPL was based primarily on NUREG/BR-0184 (NRC 1997d) and was executed appropriately.

In response to an RAI, FPL considered the uncertainties associated with the calculated CDF (see Table 5-6 below). The uncertainty values provided are for "parameter value" uncertainties. The calculated CDF used for the uncertainty analysis is based on the PDS TOP model whereby approximately 95 percent (99 percent for Unit 2) of the baseline CDF is captured. The best-estimate CDFs calculated using the PDS TOP model are 2.86×10^{-5}/ry and 2.43×10^{-5}/ry for Units 1 and 2, respectively. If the 95th percentile values of the CDF were used in the cost-benefit analysis instead of the best-estimate CDF values cited above, the estimated benefits of the SAMAs would increase by about a factor of two. However, a more detailed examination by FPL found that the initial conclusion (that none of the candidate SAMAs evaluated would be cost-beneficial for St. Lucie) would still be valid (FPL 2002a).

Table 5-6. Uncertainty in the Calculated CDF for St. Lucie Units 1 and 2

	CDF (per reactor-year)	
Percentile	**Unit 1**	**Unit 2**
5th	8.21×10^{-6}	9.64×10^{-6}
50th	1.52×10^{-5}	1.73×10^{-5}
95th	6.15×10^{-5}	6.11×10^{-5}

In addition, FPL performed sensitivity analyses to address assumptions made in other parts of the cost-benefit analysis, including variations in discount rate, weather, percent of population evacuating, evacuation speed, population, and source terms. None of these parametric variations were found to have a significant impact on results.

The staff concludes that the costs of the 50 candidate SAMAs assessed would be higher than the associated benefits. This conclusion is upheld despite a number of uncertainties and non-quantifiable factors in the calculations summarized as follows:

- Uncertainty in the internal events CDF was not explicitly included in the calculations, which employed best-estimate values to determine the benefits. The 95-percent confidence level for internal events CDF is approximately 2 times the best-estimate CDF. However, the results of the cost-benefit analysis show that all of the SAMAs evaluated would cost much more than twice the associated benefit. Therefore, consideration of CDF uncertainty is not expected to alter the conclusions of the analysis.

- External events were similarly not included in the St. Lucie risk profile. However, given that the expected external events contribution to CDF is small, and the observation that there are no particular vulnerabilities in the external event risk profile at St. Lucie, any additional benefits that might accrue due to external events would fall within the factor of 2 margin used in the screening analysis.

- Risk reduction and cost estimates were generally found to be conservative. As such, uncertainty in the costs of any of the contemplated SAMAs would not likely have the effect of making them cost-beneficial.

- A number of sensitivity calculations were performed with respect to the discount rate (as low as 3 percent) and various MACCS2 parameters, including evacuation percentage and speed, meteorological data, population distribution, and source terms. The results of these calculations showed that none of the risk benefits were increased by more than a factor of 1.2. Since this is less than the margin between cost and benefit for the SAMAs considered, the uncertainties in these parameters would not alter the conclusions.

5.2.7 Conclusions

FPL compiled a list of 169 SAMA candidates using the SAMA analyses as submitted in support of licensing activities for other nuclear power plants, NRC and industry documents discussing potential plant improvements, and the plant-specific insights from the FPL IPE, IPEEE, and current PSA model. A qualitative screening removed SAMA candidates that (1) did not apply to St. Lucie Units 1 and 2 due to design differences, or (2) the SAMA had already been implemented at St. Lucie (or the design meets the intent of the SAMA, as determined by plant review of each SAMA). A total of 119 SAMA candidates were eliminated based on the above criteria, leaving 50 SAMA candidates for further evaluation.

Using guidance in NUREG/BR-0184 (NRC 1997d), the current PSA model, and a Level 3 analysis developed specifically for SAMA evaluation, a maximum attainable benefit of about $1,382,000 was calculated, representing the total present dollar value equivalent associated with completely eliminating severe accidents at St. Lucie. Twenty-nine of the 50 SAMAs were screened from further evaluation because their single unit implementation costs were greater than this maximum attainable benefit. Each of the remaining 21 SAMAs was eliminated because their implementation cost exceeded twice the estimated benefit for that specific SAMA. The factor of two was used to account for uncertainties in the analysis and the potential impact of external events on the results of the SAMA evaluations. The end result was that no SAMA candidates were found to be cost-beneficial.

The staff reviewed the FPL analysis and has preliminarily concluded that the methods used and the implementation of those methods were sound. The treatment of SAMA benefits and costs, the generally large negative net benefits, and the inherently small baseline risks support the general conclusion that the SAMA evaluations performed by FPL are reasonable and sufficient for the license renewal submittal. The unavailability of a seismic and fire PSA model precluded a quantitative evaluation of the SAMAs specifically aimed at reducing risk of these initiators; however, significant improvements have been realized as a result of the IPEEE process at St. Lucie that would minimize the likelihood of identifying cost-beneficial enhancements in this area.

Based on its review of the FPL SAMA analyses, the staff concurs that none of the candidate SAMAs are cost-beneficial. This is based on conservative treatment of costs and benefits. This conclusion is consistent with the low residual level of risk indicated in the St. Lucie PSA and the fact that St. Lucie has already implemented many plant improvements identified from the IPE and IPEEE process.

5.3 References

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10 CFR 51. Code of Federal Regulations, Title 10, *Energy,* Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.”

10 CFR 54. Code of Federal Regulations, Title 10, *Energy,* Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.”

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