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).(1) 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) 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).

(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.

(1) 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.

The DBAs are evaluated by both the licensee and the NRC staff 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). The 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 maximum exposed individual; as such, changes in the plant environment will not affect these evaluations. Because of the requirement that aging management programs be in effect for license renewal and the requirement that the consequences of any DBA remain below specified acceptable levels at all times during plant operation, 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 period of extended operation 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. The early resolution of the DBAs 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.

This issue, applicable to the Virgil C. Summer Nuclear Station (V.C. Summer) is listed in Table 5-1.

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

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.

South Carolina Electric and Gas Company (SCE&G) stated in its Environmental Report (ER) (SCE&G 2002) that it is not aware of any new and significant information associated with the renewal of the V.C. Summer OL. The staff has not identified any significant new information during its independent review of the ER (SCE&G 2002), 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 design basis accidents 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 V.C. Summer site in the GEIS (NRC 1996). However, in the GEIS, the staff did evaluate existing impact assessments performed by the NRC and by the industry for 44 nuclear plants in the United States. As set forth in the GEIS, the staff 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 V.C. Summer, 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 (SCE&G 2002), 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 V.C. Summer. The results of its review are discussed in Section 5.2.

5.2 Severe Accident Mitigation Alternatives (SAMAs)

10 CFR 51.53(c)(3)(ii)(L) requires that license renewal (LR) 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 V.C. Summer; therefore, the remainder of Chapter 5 addresses those alternatives.

5.2.1 Introduction

This section presents a summary of the SAMA evaluation for V.C. Summer conducted by SCE&G and described in the ER (SCE&G 2002) and of the NRC's review of that evaluation. The details of the review are described in the NRC staff evaluation that was prepared by the staff with contract assistance from Information Systems Laboratories, Inc. The entire evaluation is presented in Appendix G.

The SAMA evaluation for V.C. Summer was a multi-step process. In the first step, SCE&G quantified the level of risk associated with potential reactor accidents using the plant-specific probabilistic risk assessment (PRA) and other risk models.

The second step was the examination of the major risk contributors to identify areas where plant improvements might have the greatest chance to reduce risk. Then, possible ways of reducing those risks were identified. Common ways of reducing risk are changes to components, systems, procedures, and training. SCE&G identified 268 potential SAMAs. Using a set of screening criteria, the number of SAMAs warranting further consideration was reduced to 32. Of these remaining SAMAs, 20 were screened from further analysis because, based on plant-specific PRA insights, they did not provide a significant safety benefit, or because the cost of implementation would be greater than the benefits associated with implementing the SAMA.

In the third step, the benefits and costs for the 12 remaining candidate SAMAs were estimated. Estimates were made of how much each proposed SAMA could reduce risk. Those estimates were developed in terms of dollars in accordance with NRC guidance for performing regulatory analyses (NRC 1997). The costs of implementing the proposed SAMAs were also estimated.

Finally in the fourth step, the costs and benefits of each of the 12 final SAMAs were compared to determine whether the SAMA was cost-beneficial, meaning the benefits of the SAMA were greater than the costs (a positive cost-benefit). In the final analysis, none of these 268 SAMAs were determined to be cost-beneficial for V.C. Summer.

Each of these four steps is discussed in more detail in the sections that follow.

5.2.2 Estimate of Risk

SCE&G submitted an assessment of SAMAs for V.C. Summer as part of the ER (SCE&G 2002). This assessment was based on the most recent V. C. Summer Probabilistic Risk Analysis (PRA) available at that time, a plant-specific offsite consequence analysis performed using the MELCOR Accident Consequence Code System 2 (MACCS2), and insights from the

V. C. Summer Individual Plant Examination (IPE) (SCE&G 1993) and Individual Plant Examination of External Events (IPEEE) (SCE&G 1995). The baseline core damage frequency (CDF) for the purpose of the SAMA evaluation is approximately 5.6×10^{-5} per year, and the baseline large early release frequency (LERF) is approximately 7.0×10^{-7} per year. The CDF and LERF are based on the risk assessment for internally-initiated events. The CDF represents a sizeable change from the original IPE CDF value of 2.0×10^{-4} per year. SCE&G did not include the contribution of risk from external events within the V. C. Summer risk estimates, but in response to a request for additional information, SCE&G applied a factor of two multiplier to the estimated internal events benefits to account for additional benefits in external events. The breakdown of CDF by initiating event/accident class is summarized in Table 5-3. Transients and loss of offsite power events are the dominant contributors to the CDF.

Table 5-3. V. C. Summer Core Damage Frequency

Initiating Event/Accident Class	CDF (Per Year)	Contribution to CDF (%)
Loss of Offsite Power (LOOP)	3.9×10^{-5}	70
Transients	7.5×10^{-6}	13
Special Initiators	4.4×10^{-6}	8
Loss-of-Coolant Accident (LOCA)	1.7×10^{-6}	3
Steam Generator Tube Rupture (SGTR)	1.7×10^{-7}	<1
Interfacing Systems LOCA (ISLOCA)	1.8×10^{-7}	<1
Others	2.6×10^{-6}	5
Total CDF (from internal events)	**5.6×10^{-5}**	**100**

SCE&G estimated the dose from all postulated accidents to the population within 80 km (50 mi) of the V.C. Summer site to be approximately 0.01 person-Sv (1.0 person-rem). The breakdown of the population dose by containment release mode is summarized in Table 5-4. Bypass events (SGTR, interfacing system LOCA) dominate the population dose.

SCE&G's determination of offsite risk at V. C. Summer 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 1995 IPEEE submittals (SCE&G 1993 and SCE&G 1995),
- the major modifications to the IPE model that have been incorporated in the V. C. Summer PRA, and

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

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

Containment Release Mode	Population Dose (Person-Rem[a] Per Year)	Contribution (%)
SGTR	0.27	27
Interfacing Systems LOCAs	0.63	63
Containment isolation failure	0.05	5
Early containment failure	0	0
Late containment failure	0.05	5
Total	1.0	100

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

The staff has reviewed SCE&G's data and evaluation methods and concludes that the quality of the risk analyses is adequate to support an assessment of the risk reduction potential for the candidate SAMAs. Specifically, the staff concludes that the Level 1 and Level 2 PRA models are of sufficient quality, SCE&G's consideration of external events is acceptable, and the methods, assumptions, and analyses applied in the estimation of offsite consequences are reasonable and acceptable for the purposes of SAMA evaluation. Accordingly, the staff based its assessment of offsite risk on the CDF and offsite doses provided by SCE&G.

5.2.3 Potential Plant Improvements

Once the most risk significant parts of the plant design and operation were identified, SCE&G searched for ways to reduce those risks. To identify potential plant improvements, SCE&G's process consisted of the following elements:

- review of plant-specific improvements identified in the V.C. Summer IPE and IPEEE and subsequent PRA revisions

- 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.

SCE&G identified 268 potential risk-reducing improvements to plant components, systems, procedures, and training (SAMAs).

All but 69 of these SAMAs were removed from further consideration because: (1) the SAMA was not applicable at V.C. Summer due to design differences, (2) the SAMA had already been implemented at V.C. Summer, (3) the SAMA was sufficiently similar to another SAMA such that they could be combined, or (4) the SAMA would not provide a significant safety benefit. A preliminary cost estimate was prepared for each of the remaining 69 SAMAs.

The preliminary cost estimate of each of these 69 remaining SAMAs was compared to the maximum attainable benefit (MAB) of 1.2 million dollars. The MAB is the dollar value of the benefit that would be achieved if the plant risk and population dose from postulated accidents could be reduced to zero. If the cost of a SAMA exceeded the MAB, it could not be cost-beneficial because no single SAMA could eliminate all the risk. Using this comparison, 37 of the candidate SAMAs were eliminated from further consideration, leaving 32 candidate SAMAs for further evaluation in Phase 2. Of these remaining SAMAs, 20 were screened from further analysis because, based on plant-specific PRA insights, they did not provide a significant safety benefit, or because the cost of implementation would be greater than the benefits associated with implementing the SAMA. This culminated in identification of 12 candidate SAMAs.

The staff questioned SCE&G about lower cost alternatives to several of the SAMAs evaluated, including the use of: (1) portable battery chargers to supply power to the steam generator instrument panels, (2) a cross-tie to the existing non-safety station batteries, (3) a direct-drive diesel emergency feedwater pump, and (4) an automatic safety injection pump trip on low refueling water storage tank (RWST) level as an alternative to an automatic swap to recirculation (NRC 2003a). In response, SCE&G provided estimated benefits and implementation costs for each alternative (SCE&G 2003a). These are discussed further in Appendix G.

The staff concludes that SCE&G used a systematic and comprehensive process for identifying potential plant improvements for V. C. Summer, and that the set of potential plant improvements identified by SCE&G is reasonably comprehensive and therefore acceptable. This search included reviewing insights from the IPE and IPEEE, and plant improvements considered in previous SAMA analyses. While explicit treatment of external events in the SAMA identification process was limited, the staff recognizes 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 Plant Improvements

SCE&G evaluated the risk-reduction potential of the 12 Phase 2 SAMAs applicable to V. C. Summer, as well as several additional SAMAs suggested by the staff. In response to a staff request, SCE&G further examined several SAMAs including those closest to being cost beneficial to determine the extent to which the SAMAs might reduce external event risk

(SCE&G 2003b). The SAMAs considered include: Phase 2 SAMA 3, Phase 2 SAMA 10, use of portable 120V DC generator to supply power to steam generator level instrumentation, installation of direct-drive diesel emergency feedwater pump, and use of the fire service water for make-up to steam generators. This assessment included consideration of both seismic and fire risk.

Based on this assessment, SCE&G concluded that although some credit may be taken for these SAMAs in external events, the benefit is more limited than in the internal events analysis. For example, power recovery in fire events may create additional difficulties not present for the initiators addressed in the internal events model. Also, the low cost alternatives would not be required to meet the rigors of a seismically-qualified component, and therefore, may not be useable following a seismic event. Nevertheless, SCE&G conservatively increased the benefit for these SAMAs by a factor of two to account for external events. In addition, the estimated benefit for all SAMAs was increased by 15% to account for the resolution of peer review comments.

The staff has reviewed SCE&G'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, for the above reasons, are 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 SCE&G's risk reduction estimates.

5.2.5 Cost Impacts of Candidate Plant Improvements

SCE&G estimated the costs of implementing the 12 SAMAs which were not initially screened out. The cost estimates conservatively did not include the cost of replacement power during any extended outages that might be needed to implement the modifications. Estimates that were taken from prior SAMA analyses were not adjusted to present-day dollars. For many of the SAMAs considered, the cost estimates were significantly greater than the benefits calculated such that a detailed evaluation was not necessary and a specific dollar value was not reported.

The staff reviewed the bases for the applicant's cost estimates. For certain improvements, the staff also compared the cost estimates (presented in Table F.6-1 of Appendix F to 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 eliminated from further consideration on the basis that the expected implementation cost would be much greater than the estimated risk reduction benefit.

The staff notes that the cost to implement a direct-drive diesel EFW pump at another plant was estimated to be about $200K. However, SCE&G estimated the cost of the modification at V. C. Summer to be about $800K based on the following: $200K for design, $200K for evaluations, $100K for materials, $200K for implementation, $30K for training, and $80K for documentation and closeout (SCE&G 2003c). To verify the validity of the $800K cost, the staff reviewed the costs for similar modifications evaluated in other plants' SAMA analyses. These costs ranged from $300K to $2M.

Although SCE&G's cost estimate is significantly greater than $200K, it does not appear to be unreasonable relative to the cost estimates for similar modifications. The staff concludes that the cost estimates provided by SCE&G are sufficient and appropriate for use in the SAMA evaluation.

5.2.6 Cost-Benefit Comparison

The methodology used by SCE&G 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 staff reviewed the SCE&G SAMA analysis and questioned the treatment of uncertainties associated with the calculated CDF. SCE&G revisited the cost-benefit analyses for the 12 Phase 2 SAMAs and found SAMAs 3 and 10 potentially cost beneficial (SCE&G 2003a). SAMA 3 was further evaluated and SCE&G estimated the total benefit to be approximately $24K and the cost of implementation to be approximately $150K to $170K. Accordingly, this SAMA is not cost-beneficial. Similarly, SAMA 10 was evaluated further. SCE&G noted that this SAMA would require modification to controls in the main control room. Costs associated with this aspect were not considered in the original cost estimate provided, nor were costs associated with the engineering analysis needed to support the modification. When these additional costs factors are included, the implementation costs would be substantially greater than $50K. The total benefit for this SAMA was estimated to be approximately $48K, accordingly, this SAMA is not cost-beneficial.

The staff questioned SCE&G about lower cost alternatives to several of the SAMAs evaluated, including the use of: (1) portable 120V DC generator to supply power to the steam generator instrument panels, (2) a cross-tie to the existing non-safety station batteries, (3) a direct-drive diesel emergency feedwater pump, and (4) an automatic safety injection pump trip on low RWST level as an alternative to an automatic swap to recirculation (NRC 2003a). SCE&G provided estimated benefits and implementation costs for each alternative. Based on these estimates, none of these alternatives appear cost beneficial.

The staff concludes that the costs of all of the SAMAs assessed would be higher than the associated benefits. This conclusion is supported by sensitivity analysis and upheld despite a

number of additional uncertainties and non-quantifiable factors in the calculations, summarized as follows:

- Uncertainty in the internal events CDF was not initially included in the calculations, which employed best-estimate values to determine the benefits. Even upon considering benefits at the 95th percentile value, no SAMAs were judged to be cost-beneficial.

- External events were similarly not included in the V. C. Summer risk profile. However, given that the expected external events contribution to CDF is calculated in a conservative fashion and is expected to be on the same order of magnitude as the internal events contribution to CDF, a factor of two increase in the estimated internal events benefits to account for the external events should be conservative

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

Based on its review of the SCE&G SAMA analysis, as set forth above, the staff finds that none of the candidate SAMAs are cost-beneficial. Therefore, they need not be implemented as part of license renewal pursuant to 10 CFR Part 54.

5.2.7 Conclusions

SCE&G compiled a list of 268 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 V. C. Summer IPE, IPEEE, and current PRA model. A qualitative screening removed SAMA candidates that (1) were not applicable at V. C. Summer due to design differences, (2) had already been implemented at V. C. Summer, (3) were sufficiently similar to another SAMA such that they could be combined, or (4) did not provide a significant safety benefit. A total of 199 SAMA candidates were eliminated based on the above criteria, leaving 69 SAMA candidates for further evaluation.

Using guidance in NUREG/BR-0184 (NRC 1997d), the current PRA model, and a Level 3 analysis developed specifically for SAMA evaluation, a maximum attainable benefit of about $1.2M was calculated, representing the total present dollar value equivalent associated with completely eliminating severe accidents at V. C. Summer. Thirty-seven of the 69 SAMAs were eliminated from further evaluation because their implementation costs were greater than this maximum attainable benefit. An additional 20 SAMAs were eliminated because, based on plant-specific PRA insights, they did not provide a significant safety benefit, or because the cost of implementation would be greater than the benefits associated with implementing the SAMA.

For the remaining 12 SAMA candidates and several additional alternatives identified by the staff, more detailed conceptual designs and cost estimates were developed. The cost-benefit analyses showed that none of the candidate SAMAs were cost-beneficial.

The staff reviewed the SCE&G analysis and 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 small baseline risks support the general conclusion that the SAMA evaluations performed by SCE&G are reasonable and sufficient for the license renewal submittal. The unavailability of a seismic and fire PRA model precluded a quantitative evaluation of SAMAs specifically aimed at reducing risk of these initiators; however, improvements have been realized as a result of the IPEEE process at V. C. Summer that would minimize the likelihood of identifying further cost-beneficial enhancements in these areas. To assess the potential impact of uncertainties in the analysis or the inclusion of additional benefits in external events, SCE&G applied a factor of two multiplier to the estimated benefits based on internally-initiated events, and confirmed that even when considering the increase in the benefits, none of the SAMAs become cost beneficial.

Based on its review of the SCE&G SAMA assessment, and as explained above, the staff finds 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 V. C. Summer PRA and the fact that V. C. Summer has already implemented plant improvements identified from the IPE and IPEEE processes.

5.3 References

South Carolina Electric & Gas Company (SCE&G). 1993. Letter from John L. Skolds (SCE&G) to Nuclear Regulatory Commission Document Control Desk. "Virgil C. Summer Nuclear Station (VCSNS) Docket No. 50-395 Operating License No. NPF-12 Transmittal of IPE Report; Generic Letter 88-20, LTR 880020," June 18, 1993.

South Carolina Electric & Gas Company (SCE&G). 1995. Letter from Gary J. Taylor (SCE&G) to Nuclear Regulatory Commission Document Control Desk. "Virgil C. Summer Nuclear Station Docket No. 50-395 Operating License No. NPF-12 Transmittal of IPEEE Report; Generic Letter 88-20, Supplement 4, (LTR 880020-4)," June 30, 1995.

South Carolina Electric & Gas Company (SCE&G). 2002. *Applicant's Environmental Report–Operating License Renewal Stage, Virgil C. Summer Nuclear Station.* South Carolina Electric & Gas Company, Columbia, South Carolina. August 2002.

U.S. Nuclear Regulatory Commission (NRC). 1988. Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," November 23, 1988.

U.S. Nuclear Regulatory Commission. 1996. *Generic Environmental Impact Statement for License Renewal of Nuclear Plant*s. NUREG-1437. Office of Nuclear Regulatory Research. Washington, D.C.