UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 16, 2016

Mr. George A. Lippard, III
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P.O. Box 88, Mail Code 800
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT TO CHANGE LICENSING BASIS TO INCORPORATE A SUPPLEMENTAL ANALYSIS TO THE STEAM GENERATOR TUBE RUPTURE ACCIDENT (CAC NO. MF4699)

Dear Mr. Lippard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 205 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1, in response to your application dated August 27, 2014, as supplemented by letters dated October 31, 2014; February 12, May 12, September 10, and November 5, 2015; January 14 and March 4, 2016.

This amendment approves a change to the Virgil C. Summer Nuclear Station, Unit No. 1, licensing basis to incorporate a supplemental analysis to the steam generator tube rupture accident.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Shawn Williams

Shawn A. Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 205 to NPF-12
2. Safety Evaluation

cc w/Enclosures: Distribution via Listserv

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 205
Renewed License No. NPF-12

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Virgil C. Summer Nuclear Station, Unit No. 1 (the facility), Renewed Facility Operating License No. NPF-12 filed by the South Carolina Electric & Gas Company (the licensee), dated August 27, 2014, as supplemented by letters dated October 31, 2014; February 12, May 12, September 10, and November 5, 2015; January 14 and March 4, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations as set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-12 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 205, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-12

Date of Issuance: May 16, 2016.

ATTACHMENT TO LICENSE AMENDMENT NO. 205

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following page of the Renewed Facility Operating License No. NPF-12 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
Page 3	Page 3

(3) SCE&G, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as amended through Amendment No. 33;

(4) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) SCE&G, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

SCE&G is authorized to operate the facility at reactor core power levels not in excess of 2900 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this renewed license. The preoperational tests, startup tests and other items identified in Attachment 1 to this renewed license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 205, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 205 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By application dated August 27, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14245A408), as supplemented by letters dated October 31, 2014 (ADAMS Accession No. ML14308A075); February 12, 2015 (ADAMS Accession No. ML15055A143); May 12, 2015 (ADAMS Accession No. ML15135A238); September 10, 2015 (ADAMS Accession No. ML15258A021); November 5, 2015 (ADAMS Accession No. ML15313A023); January 14, 2016 (ADAMS Accession No. ML16020A498); and March 4, 2016 (ADAMS Accession No. ML16068A176), South Carolina Electric & Gas Company (SCE&G, the licensee) submitted a license amendment request (LAR) for the Virgil C. Summer Nuclear Station, Unit No. 1 (VCSNS).

The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 14, 2014 (79 FR 61661).

The amendment incorporates a supplemental analysis of a steam generator tube rupture (SGTR) accident, which explicitly models operator responses and quantifies their impact on the potential for steam generator overfill and offsite and control room (CR) doses. The licensee stated that the new transient calculations supplement the VCSNS licensing basis analysis by demonstrating margin to steam generator overfill and providing input to a dose analysis that confirms the licensing basis mass transfer input is conservative.

2.0 REGULATORY EVALUATION

2.1 Description

An SGTR results in a loss of primary coolant from the reactor coolant system (RCS) to the secondary side of the affected steam generator (SG). The event is assumed to take place at full power with the reactor coolant contaminated with fission products based on a conservative assumption regarding cladding breach. The flow of radioactive reactor coolant results in contamination of the secondary system. In the event of a coincidental loss of offsite power or failure of the condenser steam dump system, discharge of activity to the environment takes place via the SG power-operated relief valves (PORVs), or safety valves. Operator actions are required to terminate the primary to secondary break flow and the release of steam to the environment. The SGTR analysis is performed to assure that the radiological consequences resulting from an SGTR are within allowable guidelines.

2.2 Description of Changes

The purpose of the submittal was to update the SGTR licensing basis presented in Chapter 15.4.3 of the Updated Final Safety Analysis Report (UFSAR) to include a supplemental analysis that models operator actions in accordance with the VCSNS emergency operating procedures (EOPs), with times confirmed by plant-specific simulator exercises. The modifications include the capability to model operator actions, an improved steam generator secondary side model, and a more realistic tube rupture break flow model.

The current SGTR thermal-hydraulic calculations do not include a computer analysis to determine the plant transient behavior following an SGTR. Instead, simplified calculations are performed, based on the expected SGTR transient response, to determine the primary-to-secondary break flow and the steam release to the environment for use in calculating the radiological consequences due to the event. Although the operator actions were not modeled explicitly in the analysis, it was implicitly assumed that the required operator actions to terminate the break flow and the steam release from the ruptured steam generator can be performed within 30 minutes. The updated analysis supports longer operator action times using revised procedures to terminate the break flow and the steam release from the ruptured steam generator.

2.3 Regulatory Requirements

Radiation Protection and Consequences

The regulatory requirements and guidance that the Radiation Protection and Consequences Branch, Division of Risk Assessment (DRA), Office of Nuclear Reactor Regulation (NRR), NRC staff considered in this review are:

Title 10 of the *Code of Federal Regulation* (10 CFR) 50.67(b)(2), "Accident source term," which states, in part:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product

release, would not receive a radiation dose in excess of 0.25 Sv (25 rem [roentgen equivalent man]) total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Appendix A to 10 CFR Part 50, "General Design Criteria (GDC)," *Criterion 19 - Control room,* which states:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem [0.05 Sv] whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792), provides the methodology for analyzing the radiological consequences of several design-basis accidents (DBAs) to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of alternate source term (AST, also known as the accident source term) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms," July 2000, provides guidance to the staff for the review of alternative source term amendment requests. SRP 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. The dose acceptance criteria for a pressurized-water reactor (PWR) SGTR with fuel damage or pre-incident spike is a TEDE of 25 rem at the exclusion area boundary (EAB) for the maximum 2-hour period, and 25 rem at the outer boundary of the low population zone (LPZ) during the entire period of the postulated radioactive cloud passage. The dose acceptance criteria for a PWR SGTR with a coincident iodine spike is a TEDE of 2.5 rem at the EAB for the maximum

2-hour period, and 2.5 rem at the outer boundary of the LPZ during the entire period of the postulated radioactive cloud passage. The NRC staff also considered relevant information in the VCSNS UFSAR.

License Amendment No. 183, dated October 4, 2010 (ADAMS Accession No. ML102160020), "Virgil C. Summer Nuclear Station, Unit No. 1, Issuance of Amendment Regarding Alternative Source Term Implementation (TAC No. ME0663)," used an AST methodology for analyzing the radiological consequences of six DBAs using RG 1.183. The SGTR was one of the DBAs analyzed.

The regulatory requirements from which the NRC staff based its acceptance are the reference values in 10 CFR 50.67, the accident specific guideline values in Regulatory Position 4.4 of RG 1.183, and Table 1 of SRP Section 15.0.1.

Meteorology

The regulatory requirements and guidance that the Meteorology & Oceanography Team, Division of Site Safety & Environmental Analysis, Office of New Reactors, NRR staff considered in this review are detailed below.

The NRC staff's evaluation of the proposed χ/Q values is based upon the following regulatory guides, codes, and standards:

- 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control Room"
- NUREG–0800, Standard Review Plan (SRP) Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases"
- NUREG–0800, Standard Review Plan (SRP) Section 6.4, "Control Room Habitability Systems"
- Regulatory Guide (RG) 1.23, Revision 1, "Meteorological Monitoring Programs for Nuclear Power Plants"
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants"
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"

Operations and Human Factors

The regulatory requirements and guidance that the Probabilistic Risk Assessment Operations & Human Factors Branch, DRA, NRR, NRC staff considered in this review are as follows:

- Appendix A to 10 CFR Part 50, "General Design Criteria (GDC)," *Criterion 19–Control room*
- 10 CFR 50.120, "Training and qualification of nuclear power plant personnel"
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," specifically:
    - Chapter 13.2.1, Revision 3, "Reactor Operator Requalification Program; Reactor Operator Training"
    - Chapter 13.5.2.1, Revision 2, "Operating and Emergency Operating Procedures"
    - Chapter 18, Revision 2, "Human Factors Engineering"
- NUREG-1764, Revision 1, "Guidance for the Review of Changes to Human Actions"
- Generic Letter 82-33, "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability"
- NUREG-0700, Revision 2, "Human-System Interface Design Review Guidelines"
- NUREG-0711, Revision 3, "Human Factors Engineering Program Review Model"
- Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times"

Reactor Systems

The regulatory requirements and guidance the Reactor Systems Branch, Division of Safety Systems, NRR, NRC staff considered in this review are:

- NUREG-0800, Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms"
- NUREG-0800, Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure"
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

Based on the above, the NRC staff reviewed the licensee's analysis to determine if there was reasonable assurance that the licensing basis analysis provided inputs to the radiological consequences analysis was appropriately bounding and conservative. The review covered: (1) postulated initial core and plant conditions, (2) method of thermal and hydraulic analysis, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the RPS, (6) operator actions consistent with the plant's EOPs, and (7) the results of the accident analysis.

3.0 TECHNICAL EVALUATION

3.1 Reactor Systems

3.1.1 Background

Enclosure 1 to the August 27, 2014, LAR proposes a change to the VCSNS licensing basis to incorporate supplemental analysis of an SGTR accident, which explicitly models operator responses and quantifies their impact on the potential for steam generator overfill and offsite and CR doses. This supplemental analysis uses the LOFTTR2 computer program and is based on the methodology developed in WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill."

The supplemental SGTR analysis are submitted in order to update the Final Safety Analysis Report (FSAR) in recognition, in part, of the following:

1. Changes in EOPs and CR protocols, which have resulted in longer times to complete the SGTR recovery.
2. Extended time required for operating crews to terminate primary to secondary break flow, following an SGTR during plant simulator exercises.
3. The current SGTR methodology does not model operator actions explicitly and is, therefore, insufficient to examine the effects of extended operator action times.
4. An increased potential for SG overfill due to longer duration of primary to secondary break flow.
5. SG overfill may lead to water relief out of a main steam safety/relief valve, and potentially create and un-isolable release path, should the main steam safety/relief valve fail to fully reset.

An SGTR accident will transfer radioactive reactor coolant to the shell side of the SG as a result of the ruptured tube and ultimately to the atmosphere. Therefore, the SGTR analysis for this LAR were performed to show that the resulting onsite and offsite doses will stay within the allowable guidelines, and there was margin available to provide reasonable assurance that SG overfilling is unlikely.

The SGTR analysis consider the complete severance of one SG tube. Other assumptions are a core power level of 2,900 megawatts thermal (MWt), plus uncertainties; nominal RCS pressure; RCS T_{avg} of 572.0 degrees Fahrenheit (°F) to 587.4 °F; 0 percent to 10 percent SG tube plugging; a low pressurizer pressure of safety injection (SI) actuation; and maximum SI flow from the high-head pumps.

3.1.2 Margin to Steam Generator Overfill (MTO) Analysis

The licensing basis radiological consequence analysis for the postulated tube rupture event is based on a scenario where the radiological release is steam-only. The radiological consequence analysis assumes that the secondary side of the SG does not overfill with liquid as a result of the event. This assumption is validated by performing a separate thermal-hydraulic analysis to demonstrate that under a different set of limiting initial conditions, the ruptured SG does not overfill.

The acceptable analytic approach described in WCAP-10698 includes the following assumptions: (1) a concurrent loss of offsite power, (2) initial conditions that are bounding of expected plant operation, and (3) a limiting single failure. The limiting single failure assumed in the WCAP-10698 reference plant analysis is the failure of an SG power-operated relief valve to open. The safety function of the PORV was accomplished by another PORV opening in another unaffected SG. The conclusion set forth in the reference plant analysis was that such an assumption regarding the operability of a single PORV, rather than no PORVs, would require plant-specific validation, as power supplies to PORVs, and the possible susceptibility of any given plant to a common-mode failure of the PORVs, are highly plant-specific. The VCSNS licensing basis does not consider the common-cause failure of all PORVs.

The supplemental thermal hydraulic evaluation provided within the LAR was performed using the LOFTTR2 code in a manner consistent with the methodology described in WCAP-10698-P-A, with the exception of assuming a single failure. The results of this evaluation indicated that there would be 11ft^3 of margin-to-overfill. The analysis assumes that the operators will isolate the ruptured SG within 6 minutes and initiate an RCS cooldown within 15 minutes after the ruptured SG is isolated.

The staff's safety evaluation report for WCAP-10698-P-A contains the condition that five plant-specific inputs were required of any user referencing the topical report. The licensee provided responses to these five conditions in Attachment 1 of the LAR. The staff reviewed the plant-specific inputs and determined that they were acceptable. The staff finds the departure from the approved WCAP-10698-P-A methodology (i.e., the removal of single failure) consistent with the VCSNS licensing basis for SGTR and supported by previous staff approvals of similar reviews. Based on the meeting of the staff's condition for WCAP-10698-P-A, and the basis for the departure, the NRC staff finds that the licensee's MTO analysis was performed using appropriate analytic methods.

The staff reviewed the operator action time input assumptions and found that they are supported by the simulator timeline results provided by the licensee in Table 15 of the LAR. The staff, therefore, finds the operator action time input assumptions acceptable for use in the supplemental analysis. The licensee's MTO analysis demonstrate that in a conservatively limiting scenario, the VCSNS SGs retain adequate margin to SG overfill, thus validating the conservatism of the licensee's radiological release calculations.

The licensee used a supplemental calculation to demonstrate the conservatism of the hand calculation and to validate the assumption that the release would be steam-only. The staff finds that the licensee's analysis accounted adequately for the operation of the plant at the proposed power level and was performed using appropriately conservative analytical methods and approved computer codes. The staff further concludes that the assumptions used in this analysis are conservative and that the event would not likely result in overfill of the ruptured SG. Therefore, the staff concludes the LAR acceptable with respect to the SGTR event MTO analysis.

3.1.3 Thermal and Hydraulic Analysis for Radiological Consequences

The licensee performed an additional thermal and hydraulic analysis to determine the input for the radiological consequences analysis for an SGTR event modeling the operator responses and break flow continuing beyond the 30 minutes considered in the licensing basis analysis. The analysis was performed with LOFTTR2 and the methodology detailed in WCAP-10698-P-A, with the exception that a single failure was not considered, which is consistent with the VCSNS licensing basis.

The limiting T_{avg} and steam generator tube plugging (SGTP) assumptions with respect to the thermal-hydraulic analysis were determined via plant-specific sensitivity runs to identify the mass releases for the radiological consequences analysis. The limiting analysis was based on a T_{avg} of 587.4 °F with no SGTP. The initial SG mass was increased to account for uncertainty in the feedwater temperature.

Other plant inputs and assumptions are the same as for the MTO analysis, except for the following:

- Maximized decay heat using the same methodology from WCAP-10698-P-A.
- Emergency Feedwater (EF) temperature was assumed to be at a maximum per WCAP-10698-P-A methodology.
- Minimum EF flow was delivered to the SGs following reactor trip and loss of offsite power with a maximum 60-second delay. The flow split was selected to minimize flow to the ruptured SG. A maximum purge volume of 68 ft^3 was modeled to delay delivery of cold EF to the SGs and maximize steam release.

The staff reviewed the plant input assumptions and confirmed that they followed the approved methodology (WCAP-10698-P-A), with the exception of the single failure assumption. The staff finds these input assumptions acceptable for the thermal hydraulic analysis.

The operator actions required are largely the same as for the MTO analysis, except as noted in Section 3.3.2.2 of the LAR. The staff reviewed the operator action time assumptions and confirmed that they were conservative and were, therefore, acceptable for this analysis.

The LOFTTR2 results for the limiting input to dose analysis are presented in Section 3.3.3 of the LAR. The limiting case was based on plant operation at the maximum operating temperature (587.4 °F), with the maximum main feedwater temperature (445.0 °F) and the minimum SGTP level (0 percent).

3.1.4 NRC Staff Conclusion

The NRC staff concludes that the licensee used appropriately conservative input and followed the approved methodology, and the calculated mass transfer data results are, therefore, acceptable for input to the radiological consequences analysis.

3.2 Meteorology

3.2.1 Atmospheric Relative Concentrations Estimates

For the SGTR analysis presented in the LAR, the licensee's CR, EAB, and LPZ radiological dose analysis uses atmospheric relative concentration (χ/Q) values taken from their 2009 AST LAR analysis. However, the NRC staff noted that the safety evaluation for the AST LAR (ADAMS Accession No. ML102160020) stated that these data may not be considered acceptable for use in other dose assessments or other meteorological applications without further NRC review and approval. In response to NRC requests for additional information (RAIs), the licensee agreed to provide new meteorological data and develop new χ/Q values for the SGTR analysis and submit them to the NRC staff for review.

3.2.2 Meteorological Data

The licensee provided meteorological data for calendar years 2012, 2013, and 2014. The data were provided in electronic form and included an Excel file listing of meteorological data for 2012, 2013, and 2014, as well as the data formatted for input into the ARCON96 computer code (NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes"). The NRC staff performed extensive screening of the meteorological data and requested additional information from the licensee regarding potential anomalies in the data. The NRC staff reviewed the RAI responses and performed analysis to determine the potential effects of the data anomalies on calculated χ/Q values. The NRC staff concluded that the licensee's results would not be affected significantly by using the 2012 through 2014 meteorological data set as input into its analysis. Therefore, on the basis of this review, the NRC staff found the 2012 through 2014 meteorological data suitable for use in making atmospheric dispersion calculations.

3.2.3 Control Room Atmospheric Relative Concentrations

In response to NRC RAIs, the licensee developed new CR χ/Q values using ARCON96 and supplied meteorological data for the 2012 through 2014 calendar years formatted for ARCON96. RG 1.194 states that ARCON96 is an acceptable methodology for assessing CR χ/Q values for use in DBA radiological analysis. The NRC staff evaluated the applicability of the ARCON96 model and concluded that the use of this model was appropriate to make calculations in support of the LAR.

The NRC staff found the 2012 through 2014 calendar year data to be appropriate for use in creating input data for the ARCON96 code. For confirmatory analysis, the NRC staff used the ARCON96 formatted files for the 2012 through 2014 calendar years that were supplied by the licensee as input. Also, the NRC staff used the Excel file listing of meteorological data supplied by the licensee to create a second set of input files, which were also used to run ARCON96. The results of the NRC staff's confirmatory analysis were consistent with those of the licensee. The licensee listed the χ/Q values in Table 1 of its March 4, 2016, RAI response.

3.2.4 EAB and LPZ Atmospheric Relative Concentrations

In response to NRC RAIs, the licensee used the PAVAN computer code (NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design-Basis Accidental Releases of Radiological Materials from Nuclear Power Stations") to develop new EAB and LPZ χ/Q values using the updated 2012 through 2014 meteorological data. The licensee's analysis was based on a joint frequency distribution using seven wind speed categories. The NRC staff requested that new calculations of χ/Q values be performed using a joint frequency distribution based on a higher number of wind speed categories, with an emphasis on lower wind speeds. In response to the RAI, the licensee calculated χ/Q values based on twelve wind speed categories and determined that in some instances, the χ/Q values would be higher when using twelve wind speed categories. The licensee initially estimated a maximum difference of 18.3 percent higher χ/Q value at the EAB using the updated information. The licensee preferred to continue using the χ/Q values based on the seven wind speed categories and proposed applying a factor of 20 percent to the radiological dose values to account for any difference in the dose values that would result from using the higher χ/Q values in the calculation. The NRC staff performed confirmatory analysis with the new information and estimated a similar, but somewhat higher, increase. Subsequently, the licensee proposed a 25 percent increase to the dose values to take into account the differences between calculations made using the χ/Q values from the seven versus twelve wind speed categories. The staff found this proposal to be acceptable for use in calculating radiological dose values associated with this LAR.

The licensee provided these augmented dose values in Table 2 of the licensee's March 4, 2016, RAI response. However, the licensee stated that it considers the χ/Q values, which were calculated based on the seven wind speed categories of the 2012 through 2014 meteorological data, to be its final χ/Q values. These χ/Q values are provided in Table 1 of the licensee's March 4, 2016, RAI response. The NRC staff found the χ/Q values based on the seven wind speed categories to be acceptable for this LAR because the resulting dose values were increased by 25 percent. However, the χ/Q values may not be considered acceptable for use in other dose assessments or other meteorological applications without further NRC review and approval.

3.2.5 NRC Staff Conclusion

The NRC staff has reviewed the licensee's SGTR LAR atmospheric dispersion analysis. In response to NRC RAIs, the licensee proposed applying a factor of 25 percent to all of the radiological dose values (CR, EAB, and LPZ) for the STGR LAR to compensate for any uncertainties in the methodology used in the calculation of the χ/Q values used in the radiological dose calculations. Based on this adjustment, the NRC staff finds the χ/Q values acceptable for use in calculating the radiological dose values for the SGTR analysis associated with this LAR. However, because of the uncertainties regarding the calculation methodology used to generate these χ/Q values, the NRC staff notes that these χ/Q values may not provide sufficient representation of the atmospheric dispersion environment at the site for other applications without correction factors. Therefore, the NRC concludes χ/Q values may not be considered acceptable for use in other dose assessments or other meteorological applications without further NRC review and approval.

3.3 Radiation Protection and Consequences

3.3.1 Background

VCSNS previously requested, and the NRC approved, use of an AST. In License Amendment No. 183, the SGTR accident was postulated to occur as a complete severance of a single steam generator tube. The accident was assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited amount of defective fuel rods. The SGTR accident allows primary coolant to leak into the secondary system via the SG. The primary coolant transfer causes the pressurizer level to decrease and causes the level in the affected SG to increase. A coincident loss of offsite power causes the release of activity to the atmosphere via the SG power-operated relief valves. The VCSNS operators were expected, within 30 minutes, to determine that an SG tube rupture had occurred and to identify and isolate the ruptured SG in order to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the ruptured SG. Break flow to the secondary system is terminated before water level in the affected SG rises into the main steam pipe.

For a double-ended SG tube rupture, the leak rate exceeds the charging pump capacities and, consequently, the pressurizer level will decrease. The decrease in the pressurizer level and the inability of the heaters to maintain pressurizer pressure causes the RCS pressure to decrease. The drop in the RCS pressure will cause a reactor trip and ensure that the departure from nucleate boiling fuel design limit is not exceeded. No fuel failure resulted from the postulated design-basis SGTR accident at VCSNS.

The SGTR analysis LAR provides simplified thermal-hydraulic calculations based on the expected SGTR transient response to determine the primary-to-secondary break flow and the steam released to the environment for use in calculating the radiological consequences resulting from the event. The existing calculations do not include computer analysis to determine the plant transient behavior following an SGTR. Instead, simplified calculations were performed, based on the expected SGTR transient response, to determine the primary-to-secondary break flow and the steam release to the environment for use in calculating the radiological consequences due to the event.

3.3.2 Revised SGTR Accident

Westinghouse performed a new SGTR dose analysis for VCSNS using the RADTRAD Version 3.03 computer program to evaluate radiation dose from the mass transfer input from the current licensing basis and new transient analysis. In the new transient analysis, the mass transfer is calculated using the LOFTTR2 computer code from the initiation of the event until break flow termination. For conservatism, 10 percent is added to the calculated mass transfer. The mass transfer information is then used to calculate the radiological consequences at the EAB, LPZ, and CR. In addition, the new transient analysis includes the simulation of the operator actions for recovery from an SGTR with reactor trip based on the VCSNS specific EOPs.

The new analysis was performed for the following two cases of postulated activity release following a design-basis SGTR in order to determine the maximum offsite and CR radiation dose:

- Pre-accident iodine spike case: For this case, the licensee assumed a reactor transient has occurred prior to the postulated SGTR and has raised the primary RCS iodine concentration to 60 times the Technical Specification (TS) 3.4.8 limit of 1.0 microcuries per gram (μCi/gm) dose equivalent iodine 131 (DEI), as is consistent with the guidance of RG 1.183 when no fuel failure is assumed.

- Concurrent iodine spike case: For this case, the licensee assumed that the RCS transient associated with the SGTR causes an iodine spike in the primary RCS. It is assumed that the iodine release rate from the fuel rods to the primary RCS increases to a value of 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value. The spike was assumed to continue until 8 hours after the start of the event. This is consistent with RG 1.183 guidance.

In addition, for the two cases considered, the TS maximum secondary coolant iodine concentration is available for release as inferred by RG 1.183 guidance. As shown in VCSNS TS 3.7.1.4, and as input to the SGTR analysis, the maximum secondary coolant iodine concentration at VCSNS is 0.1 μCi/gm DEI. Consistent with RG 1.183 guidance, the chemical form of iodine released from the SGs to the environment is 97 percent elemental and 3 percent organic.

The new analysis assumes that a portion of the primary-to-secondary leakage through the SGTR flashes to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. The leakage that immediately flashes to vapor is immediately released to the environment with no mitigation (i.e., no reduction for scrubbing within the SG bulk water was credited). The licensee did not credit partitioning of this release to the condenser. All leakage that does not immediately flash is mixed with the SG bulk water. The radioactivity within the SG bulk water becomes vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 is used for the leakage from the SG bulk water. This value is consistent with the regulatory guidance of RG 1.183.

The new analysis restrict primary-to-secondary leakage rate to a TS limited rate of 1 gallon per minute (gpm) apportioned between all SGs in such a manner that maximizes the calculated dose consequence. The licensee assumed that prior to the event, this leakage was distributed throughout the three SGs. The activity associated with the 1 gpm leak is released to the environment via the two intact SGs for the 24-hour duration of the event. A partition coefficient for iodine of 100 is assumed for this leakage. For the initial release of secondary coolant activity to the environment, the licensee also uses a partitioning factor of 100. Both values are consistent with the regulatory guidance of RG 1.183.

The radiation doses for both sets of mass transfer data are presented in Table 1 below and compared to the SRP 15.0.1 and RG 1.183 limits. The new analysis incorporates new atmospheric dispersion factors (x/Qs) for the EAB, LPZ, and CR radiation doses. The licensee updated the x/Qs (see Section 3.2.2 of this safety evaluation) to reflect the most recent meteorological input data (January 1, 2012, through December 31, 2014). The licensee concluded that the new radiation dose values are well within the acceptance limits specified in 10 CFR 50.67, SRP 15.0.1, and RG 1.183.

The NRC staff performed an independent analysis to confirm the licensee's results and, based on that analysis, the NRC staff concludes the licensee's assessment is acceptable. The new transient analysis yields radiation doses that are below the regulatory limits. The NRC staff evaluated the licensee's revised SGTR accident analysis and finds that the licensee's revised analysis methods and assumptions are consistent with the guidance of RG 1.183. The NRC review concluded, with reasonable assurance, that the licensee's estimates of the radiation dose consequences of a postulated SGTR at the EAB, LPZ, and CR will comply with (1) 10 CFR 50.67, (2) 10 CFR 50, Appendix A, GDC 19, (3) RG 1.183 dose acceptance criteria; and (4) SRP 15.0.1 dose acceptance criteria.

Table 1
Steam Generator Tube Rupture Radiological Consequence Analysis Results

Scenario	Location	New Transient TEDE Dose (rem)	RG 1.183 and SRP 15.0.1 TEDE Dose Limit (rem)
Pre-Accident Iodine Spike	Exclusion Area Boundary	0.85	25
	Low Population Zone	0.35	25
	Control Room	1.63	5
Concurrent Iodine Spike	Exclusion Area Boundary	0.39	2.5
	Low Population Zone	0.18	2.5
	Control Room	0.63	5

3.3.3 NRC Staff Conclusion

The NRC staff reviewed the regulatory and technical analysis performed by the licensee in support of its proposed license amendment as they relate to the radiological consequences of the SGTR analysis. The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed license amendment. The NRC staff finds that the licensee used analysis methods, inputs, and assumptions that are adequately conservative and consistent with applicable regulatory guidance identified in Section 2.0 of this safety evaluation. The NRC staff performed an independent analysis of the radiological consequences to confirm the licensee's radiation dose results and conservatism of the analysis. The NRC staff found the major parameters and assumptions used by the licensee as presented in Tables 9 through 14 of the LAR and its supplements to be acceptable. The results of the licensee's design-basis radiological consequence calculation are provided in Table 1 above. The EAB, LPZ, and CR doses estimated by the licensee for the SGTR accident were found to meet the applicable accident dose criteria and are, therefore, acceptable.

3.4 Operations and Human Factors

3.4.1 Description of Operator Action(s) and Assessed Safety Significance

The licensee listed the required operator actions for the SGTR scenario in Table 3 of the LAR on page 14 of 53 as follows:

- isolate emergency feedwater flow to ruptured SG
- isolate main steam isolation valve on ruptured SG
- initiate cooldown
- initiate depressurization
- terminate SI following depressurization.

None of these actions will be changing; however, the assumed timing of these actions was questioned due to training experience in the simulator that showed that actual performance times were longer than those assumed in the original analysis. The SGTR scenario was reanalyzed using actual performance times in order to ensure that the original conclusions and offsite dose consequences are still valid and conservative.

In accordance with the generic risk categories established in Appendix A to NUREG-1764, these task sequences are considered "of high risk-importance," due to the fact that they are integral to the overall strategy for reactor cooldown and minimizing the spread of radioactive contamination. Because of the high potential risk, the NRC staff performed a "Level One" review (i.e., the most stringent of the graded reviews possible under the guidance of NUREG-1764).

3.4.2 Precedents

VCSNS identified that the proposed changes are based on analysis similar to the approved license amendments for the Point Beach Nuclear Plant, Units 1 and 2 (ADAMS Package Accession No. ML111170513); Turkey Point Nuclear Generating, Units 3 and 4 (ADAMS Package Accession No. ML 11293A359); and Prairie Island Nuclear Generating Plant, Units 1 and 2 (ADAMS Accession No. ML112521289).

The NRC staff reviewed the approved license amendments mentioned above. Similar to this LAR, the approved license amendments included an SGTR analysis that supplemented the original licensing basis 30-minute hand calculation. The approved license amendments included a calculation to show that with respect to dose inputs, the 30-minute hand calculation is conservative. Although the inputs and assumptions used for the approved license amendments are different, the same general approach, with similar analysis, was used. Based on the above, the staff concludes that SCE&G's comparison to previously approved license amendments is acceptable.

3.4.3 Functional Requirements Analysis and Function Allocation

Because the proposed operator actions are not new actions, functional requirements analysis and function allocations were not necessary. Prior training experience had shown that operators, when assigned this task, were taking longer than was assumed in the original analysis. However, rather than automating these manual actions, the licensee decided to determine the consequence of the delayed actions by reanalysis using actual operator performance times. The staff finds this approach acceptable based on its demonstration of adequate margin to the newly analyzed time constraints.

3.4.4 Task Analysis

Because the proposed operator actions are not new actions, the only aspect requiring reanalysis was the establishment of time constraints for the action sequence. The licensee had originally established, via hand calculations, that 30 minutes was needed for operator action to terminate primary to secondary break flow following an SGTR accident.

For this LAR, simulated operator action walkthroughs, rather than hand calculations, were conducted to determine the more realistic operational values for the timing of the action sequence. During plant simulator exercises, the operating crews were taking slightly longer than 30 minutes to terminate primary to secondary break flow following an SGTR. The staff finds the use of operator action walkthroughs to estimate the task timing acceptable.

3.4.5 Staffing

Normal staffing and qualification are not affected by the proposed LAR. There are no new or additional qualifications required to perform the action sequences within the new time constraints established.

3.4.6 Human-System Interface Design

Human-system interface design, including the design of the Safety Parameter Display System, will not be affected by the proposed LAR.

Local operator actions for the SGTR response are all within accessible areas with acceptable environmental conditions.

3.4.7 Procedure Design

The licensee implemented the following procedure changes to achieve earlier break flow termination:

- In EOP-4.0, (E-3) SGTR, actions to isolate the ruptured SG were removed from the main body of the procedure and included in an attachment. This allows one main control board operator to isolate feedwater to, and steam from, the ruptured SG, while other actions in preparation for RCS cooldown and depressurization are being performed by the remaining CR staff.

- In EOP-4.0, after the RCS cooldown and depressurization is complete, actions were added to confirm the charging/SI pump mini-flow lines are aligned such that SI flow to the RCS can be terminated prior to other recovery actions to restore normal makeup and letdown.

Based on the licensee's validation of these procedure changes as feasible, effective, and reliable (see Section 3.3.9 below), the NRC staff finds these changes acceptable.

3.4.8 Training Program and Simulator Design

The operator action times used in the VCSNS analysis will be added to the plant's operator requalification program, which currently requires both classroom and simulator SGTR training for all licensed operators. The operator action times modeled in the analysis have been validated in VCSNS simulator exercises and are being used as acceptance criteria for continuing training.

3.4.9 Human Factors Verification and Validation

Time testing of the proposed actions was performed to demonstrate sufficient margin to the licensee-established design values. The feasibility of operators completing required actions within the time available was validated using Operations Administrative Procedure 101.3, "Timeline Validation of Required Operator Actions." For the simulator exercises presented, the time from event initiation to SI termination ranged from approximately 31.1 to 34.3 minutes. The required operator action times used in the VCSNS specific SGTR margin-to-overfill have been shown to be realistic via simulator exercises to assure the operators could mitigate the accident within the period of time compatible with overfill prevention. The required times were met by all operating crews and would be adequate to preclude overfill during an SGTR scenario.

The analyzed doses at the EAB, LPZ, and in the CR resulting from an SGTR were well within the applicable limits. It is confirmed that the simplified calculations with a 30-minute break flow and release duration used in the licensing basis analysis produce significantly higher dose results compared to doses calculated with transient data obtained modeling conservative operator response times and break flow continuing beyond 30 minutes. Therefore, past use of the 30-minute time limit was conservative, and delays beyond 30 minutes up to approximately 50 minutes had no negative effect in regard to increasing doses.

Based on the results of these simulator demonstrations, the staff concludes that the actions are feasible and can be reliably performed by the operators using the revised procedures within the operational times established (approximately 50 minutes).

3.4.10 Human Performance Monitoring Strategy

The licensee stated that all plant modifications and/or major changes to the EOPs are assessed for impact to assure the EOPs remain compatible with the plant design and licensing basis, including SGTR scenarios. Major changes to EOPs are also assessed for impact on time critical actions and, when adverse impacts are suspected, the ability of the operator to perform within acceptable time limits is revalidated. This administrative program for assessing the impact of plant changes and EOP changes on the time-critical actions within the EOPs is acceptable to the staff.

Additionally, the licensee has committed to revise the VCSNS FSAR to reflect the updated SGTR analysis. This action assures, through 10 CFR 50.59, that any further changes to the analysis will require screening for unreviewed safety questions and, if necessary, review by the NRC.

3.4.11 NRC Staff Conclusion

Based on the statements provided by SCE&G (i.e., that time-testing results demonstrate significant margin to updated design; that appropriate administrative controls will be applied to procedures; that trained, qualified operators will perform the actions using an unchanged interface; and that all operators have been successfully time-tested, the NRC staff concludes that the proposed LAR is acceptable in regards to human performance.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration published in the *Federal Register* on October 14, 2014 (79 FR 61661) and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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