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

August 10, 2021

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SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1—CORRECTION TO AMENDMENT NO. 219 TO MODIFY TECHNICAL SPECIFICATION 6.9.1.11, "CORE OPERATING LIMITS REPORT" (EPID L-2020-LLA-0124)

Dear Mr. Sartain:

On June 30, 2021, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21112A108), the U.S. Nuclear Regulatory Commission (the Commission) has Amendment No. 219 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit 1. The amendment consist of changes to the technical specifications (TSs) in response to your application dated June 4, 2020 (ADAMS Accession No. ML20156A303), as supplemented by a letter dated January 7, 2021 (ADAMS Accession No. ML21007A339).

Following the issuance of the approval, the licensee informed the NRC of administrative errors in the numbering schemes and references. The NRC staff determined that the errors were entirely administrative in nature and were inadvertently introduced.

This correction letter provides a revised SE which incorporates changes to the corrected pages or sections of the SE using temporary change bars to identify the corrections being made. These corrections do not change the amendment pages of the license or any of the conclusions in the safety evaluation associated with the amendment issued by letter dated June 30, 2021, and do not affect the associated *Federal Register* Notice of Issuance of the amendment, 85 FR 48568, published on August 11 2020.

Consistent with NRC staff guidance dated January 16, 1997 (ADAMS Accession No. ML103260096), based on the NRC's policy established by SECY-96-238, these errors can be corrected by a letter to the licensee from the NRC staff.

If you have any questions regarding this matter, please contact me at 301-414-5897.

Sincerely,

/RA/

Vaughn V. Thomas, Project Manager
Plant Licensing Branch II-I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Corrected Safety Evaluation, ML21188A332

cc: Listserv

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

ENCLOSURE

CORRECTED SAFETY EVALUATON

RELATED TO AMENDMENT 219 DATED JUNE 30, 2021

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 219 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOMINION ENERGY SOUTH CAROLINA, INC.

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By letter dated June 4, 2020 (Reference 1), as supplemented by letter dated January 7, 2021 (Reference 2), Dominion Energy South Carolina, Inc. (the licensee) submitted a license amendment request to modify technical specifications (TS) related to the core operating limits report for Virgil C. Summer Nuclear Station (Summer), Unit 1. Specifically, the licensee proposed to revise the analytical method Item c. of TS 6.9.1.11, "Core Operating Limits Report," to apply the full spectrum of break sizes to the loss-of-coolant analysis (LOCA) methodology (FULL SPECTRUM LOCA Methodology), hereafter referred to as FULL SPECTRUM™.

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 Volume 85 of the *Federal Register* (FR), page 48568, on August 11, 2020 (85 FR 48568).

Proposed Change

The proposed changes would replace current analytical method Item c. of TS 6.9.1.11, "Core Operating Limits Report," with an approved FULL SPECTRUM™ loss-of-coolant-accident (FSLOCA™) approach. Specifically, the amendment would replace the current statistically-based best estimate large-break loss-of-coolant accident (LBLOCA) and deterministically-based small-break loss-of-coolant accident (SBLOCA) with an approved FSLOCA™ approach.

UFSAR Section 6.3.2, "System Design"

The emergency core cooling system (ECCS) components are designed such that a minimum of two accumulators, one charging pump, and one residual heat removal pump, together with their associated valves and piping, will ensure adequate core cooling in the event of a design-basis accident. The redundant onsite emergency diesels ensure adequate emergency power to all cooling," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 and 10 CFR 50.36, "Technical specifications."

2.1 10 CFR 50.46

Regulatory requirements specified in 10 CFR 50.46 that are relevant to the proposed license amendment states, in part, that:

- Each boiling-water or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an ECCS that must be designed so that its calculated cooling performance following postulated LOCAs set forth in 10 CFR 50.46(b)

- ECCS cooling performance must be calculated with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and different properties sufficient to provide assurance that the most severe LOCAs are calculated.

- The evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during LOCAs.

- Uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to 10 CFR 50.46(b), there is a high level of probability that the criteria would not be exceeded.

- Appendix K to 10 CFR Part 50, ECCS Evaluation Models, provides required and acceptable features of the evaluation models and required documentation.

The regulations in 10 CFR 50.46(b) require, in part, that, during LOCA events, the following criteria are met:

(1) For peak cladding temperature, the calculated maximum fuel element cladding temperature shall not exceed 2200 °F [degrees Fahrenheit].

(2) For maximum cladding oxidation, the calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) For maximum hydrogen generation, the calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) For coolable geometry, the calculated changes in core geometry shall be such that the core remains amenable to cooling.

As described in the safety evaluation (SE) for WCAP-16996-P-A, Revision 1 (Reference 4), the NRC staff clarified with Westinghouse (the vendor of WCAP-16996) that "the FSLOCA methodology does not treat boric acid precipitation, and long-term cooling cannot be completely addressed with this methodology. Therefore, the long-term cooling criterion defined in

2.1.4 Applicable Regulatory Guides and Standard Review Plan Guidance

In its review of this LAR, the NRC staff considered the following guidance:

- Regulatory Guide (RG) 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” issued May 1989 (Reference 6) , which describes acceptable models, correlations, data, model evaluation procedures, and methods for meeting the realistic (best estimate) EM requirements for calculating ECCS performance during a LOCA as set forth in 10 CFR 50.46.

- RG 1.203, “Transient and Accident Analysis Methods,” issued December 2005 (Reference 7), which provides guidance to licensees and applicants for use in developing and assessing EMs for accident and transient analyses.

- NRC Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

- NRC Information Notice (IN) 2011-21, Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation,” (Reference 8)

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 15.0.2, “Review of Transient and Accident Analysis Methods,” (Reference 9) and SRP Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary,” (Reference 10) describes for NRC reviewers the review scope, acceptance criteria, review procedures, and findings relevant to ECCS analyses.

2.2 Licensee’s Proposed Changes

The licensee plans to transition from the current statistically based best estimate large-break LOCA (Reference 11), deterministically based small-break LOCA ((Reference 12) and (Reference 13)) methods to a state-of-the-art, unified, and approved FSLOCA™ EM approach (Reference 14). The proposed change will involve a change to Summer’s current TS 6.9.1.11 Analytical Methods Item (c). The proposed change in analysis methods also will fulfill the Dominion Energy South Carolina, Inc (DESC) commitment (Reference 15) to address fuel pellet thermal conductivity degradation (TCD) as described in IN 2011-21 regarding thermal conductivity degradation, by replacing the previous PAD3.4 and PAD4.0 fuel thermal performance Codes with the updated and approved PAD5 Code (Reference 16) in the LOCA analyses. Section 3.2.2.2 of this SE provides further details.

Westinghouse adopted the FSLOCA™ EM approach (Reference 14) to complete an analysis with the FSLOCA™ EM for Summer. This LAR seeks NRC approval to apply the approved Westinghouse FSLOCA™ EM for the licensing analysis of record at Summer.

As described in the evaluation below, the NRC staff reviewed the licensee’s implementation of the FSLOCA™ EM for Summer to ensure compliance with applicable regulatory requirements. The NRC staff’s review activities associated with the LOCA analysis using the FSLOCA™ EM focused on the review of pertinent sections of the licensee’s submittals (particularly

Attachment 4 in Reference 1). The NRC staff conducted a regulatory audit from October 21 to November 30, 2020 (Reference 17).

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

The licensee proposed to revise the analytical method required by TS 6.9.1.11.c under the title "CORE OPERATING LIMITS REPORT,"

Current TS 6.9.1.11.c. states:

> c. WCAP-12945-P-A, Volume 1 (Revision 2) through Volumes 2 through 5 (Revision 1) "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary) (Reference 11).
>
> Liparulo, N. (W) to NRC Document Control Desk, NSD-NRC-96-4746, "Re-Analysis Work Plans Using Final Best Estimate Methodology" dated June 13,1996 (Reference 18).
>
> (Methodology for Specification 3.2.2—Heat Flux Hot Channel Factor.)

Revised TS 6.9.1.11.c. would state:

> c. WCAP-16996-P-A Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," November 2016, (Westinghouse Proprietary) (Reference 14).

3.2 NRC Staff Evaluation

3.2.1 Applicability of FSLOCA EM to Summer

Since the FSLOCA™ EM is developed to comply with 10 CFR 50.46, the first key requirement of 10 CFR 50.46 as summarized in Section 2.2.1 constitutes one condition for FSLOCA™ EM to be applied to Summer. The Executive Summary for WCAP-16996-P-A/WCAP-16996-NP-A, Volumes I, II, and III, Revision 1, (Reference 19) states that "the FULL SPECTRUM™ LOCA EM is intended to be applicable to all pressurized-water reactors (PWR) fuel designs with Zirconium alloy cladding." Table 1 of Attachment 4 to the LAR listed the fuel type used in Summer as 17x17 Vantage+ fuel with Optimized ZIRLO™ cladding material. The NRC staff finds the FSLOCA™ EM applicable because Summer uses Zirconium alloy clad fuel as specified in 10 CFR 50.46.

The SE for FSLOCA™ EM (Reference 13) established the following applicability limitations and conditions:

- Limitation No. 1: FSLOCA™ EM applicability with regard to LOCA transient phases
- Limitation No. 2: FSLOCA™ EM applicability with regard to type of PWR plants

In Section 3.2.3 of this SE, the NRC evaluates both limitations and finds the applicability conditions of FSLOCA™ EM to Summer as developed acceptable. Section 3.2.3 of this SE also shows the details for the applicability conditions.

While the NRC previously approved the FSLOCA™ EM that the licensee proposes to use to support its proposed analysis method transition (Reference 14), the NRC staff reviewed the licensee's implementations of this EM for Summer to ensure the following:

- confirmation of acceptable plant-specific inputs to the EM (Section 3.2.2.1 of this SE)
- confirmation of adherence to the approved EM (Sections 3.2.2.2 and 3.2.2.3 of this SE)
- confirmation that results calculated using the EM satisfy regulatory acceptance criteria and conform to expected outcomes (Section 3.2.2.4 of this SE)
- verification of acceptable responses to limitations and conditions specified in the NRC staff's SE (Section 3.2.3 of this SE)

3.2.2.1 Summer Plant-Specific Inputs

Acceptance criteria in SRP Section 6.3, "Emergency Core Cooling System," (Reference 20) state that one of the requirements for a realistic or best estimate EM for ECCS performance is to identify and account for uncertainties in the analysis method and inputs such that there is a high level of probability that the acceptance criteria will not be exceeded.

Tables 1 to 6 of Attachment 4 to the LAR listed the Summer data used in the FSLOCA™ EM calculation. The NRC staff finds the data as provided reasonable for a set of PWR design and/or operation data to meet the FSLOCA™ EM analysis need. For example, the licensee provided the Summer containment design data in Table 2 to support FSLOCA™ EM to calculate a conservatively low containment pressure during the Region II LOCA analysis because minimum containment back pressure maximizes the calculated PCT. NRC considers this is a conservative approach based on the FSLOCA™ methodology. The NRC staff also finds that Tables 4 and 5 in LAR Attachment 4 provided the minimum safety injection (SI) flow (Table 1, Item 5e) for the FSLOCA™ EM analysis, which is also conservative because it maximizes the calculated PCT.

3.2.2.2 Summer Plant FSLOCA™ EM Model

Summer is a three-loop plant containing Westinghouse 17x17 Vantage+ fuel with intermediate flow mixers, integral fuel burnable absorbers, and Optimized ZIRLO™ cladding. The licensee is authorized to operate the facility at reactor core power levels not to exceed 2,900 megawatts thermal. Summer also has Westinghouse DELTA-75 replacement steam generators. The vessel and loop portions of Summer are modeled with significant details for WCOBRA/TRAC-TF2 within the FSLOCA™ EM.

The core is modeled with rod types representing hot rod, hot assembly average rod, outer low-powered assembly average rod, and two inner assembly average rods (under or not under guide tube). The nuclear fuel rods are initialized with internal gas compositions and fuel average temperatures from the PAD5 code (Reference 16). The licensee complied with Limitation and Condition 6 of FSLOCA™ EM methodology and stated that the fuel performance data for analyses with the FSLOCA™ EM would be based on the PAD5 Code, which includes the effect of TCD. In its letter dated January 7, 2021, the licensee stated that this satisfies its commitment to address fuel pellet TCD as described in NRC IN 2011-21 (Reference 8), by replacing the previous PAD3.4 and PAD4.0 fuel thermal performance Codes with the updated and approved PAD5 Code in the LOCA analyses.

The NRC staff reviewed the Westinghouse Letter, LTR-NRC-14-17, "Submittal of Westinghouse Responses to 'WCAP-16996-P, 'Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)" Request for Additional Information – RAIs 36-39,'" dated March 24, 2014 (Reference 21), and finds the approved PAD5 explicitly models TCD and is benchmarked to high burnup data. The nominal fuel pellet average temperatures and rod internal pressures would be the maximum values, and the generation of all the PAD5 fuel performance data would adhere to the NRC-approved PAD5 methodology.

Loop 1 contains the pressurizer, hot leg, steam generator, feedwater connection, downcomer of the steam generator, feedwater FILL, main steam isolation valve, or the main steam safety valve. Continuing around Loop 1 are the crossover leg, reactor coolant pump (RCP), cold leg (divided into two TEE components to model SI FILL) and accumulator injection PIPE. Loop 2 is set up much the same way, except that the hot leg is modeled as a PIPE module in the absence of the pressurizer, and the SI and accumulator injection points are reversed based on the actual plant ECCS configuration. Loop 3 is modeled the same as Loop 2.

The SI system for Summer consists of three accumulator tanks, two charging/SI pumps, and two low head residual heat removal pumps. Each pump is connected to injection lines that inject directly into each cold leg. This modeling is consistent with Limitation and Condition 2.

The accumulators are modeled to inject at a nominal pressure, a nominal temperature, and a nominal water volume. The pumped SI flow is modeled assuming a nominal temperature and the loss of one train of SI pumps (one SI and one residual heat removal). The loss of one train of SI is considered as the limiting single-failure assumption. The NRC staff finds the model acceptable, as it contains the appropriate details (components and systems) for the intended analysis for Summer.

Table 1 of LAR Attachment 4 provides the initial conditions for the FSLOCA™ EM analysis. In LAR-20-176 (Reference 2), the licensee provided information describing how the axial power distributions based on Summer TS LCO 3.2.1, "Axial Power Difference," are addressed in the FSLOCA™ EM axial power distribution library. The NRC staff finds that the licensee verified that the power distributions created in the FSLOCA™ EM library for the Summer analysis are consistent with, or bound, those allowed by Summer TS LCO 3.2.1 and satisfy the applicable acceptance criteria of 10 CFR 50.46.

3.2.2.3 Break Spectrum (Region I and Region II) Analyses

In its submittal dated June 4, 2020, the licensee described its Region 1 and 2 analyses. The review evaluated whether the entire break spectrum (break size and location) has been explored to identify the limiting break through sufficient analyses to determine the worst break peak cladding temperature (PCT), the worst local clad oxidation, and the highest core-wide oxidation (CWO) percentage. The small-break spectrum should be evaluated with sufficient resolution to locate these limiting conditions.

The licensee stated in the LAR that the entire break size spectrum has been divided into two regions. Region I encompasses breaks that are typically defined as small-break LOCAs. Region II includes break sizes that are typically defined as large-break LOCAs.
As to Limitation and Condition 10, the licensee stated that the minimum sampled break area for the Region II analysis was 1 square foot. Summer is one of the pilot plants the FSLOCA EM

vendor (Westinghouse) used to develop the FSLOCA EM. Based on the demonstration plants' analysis results, Section 4.7.4 of the SE for WCAP-16996 provided the following evaluation results on the FSLOCA™ EM's treatment of the break spectrum:

> For the proposed approach of modeling breaks in Region II of the FSLOCA™ EM as either split breaks of a variable area with a uniform break area distribution or as a constant-size DEG break with an equal probability of choosing between a DEG break and a split break is consistent with the approved ASTRUM method and was found acceptable to the NRC staff. The proposed treatment of breaks in Region I of the FSLOCA™ EM as split breaks of a variable area is consistent with modeling of smaller breaks other than a DEG break in Region II and was also found appropriate.

However, based on its review of the FSLOCA™ EM methodology, the NRC staff determined that there should be a limiting break size (due to the PCT) determined from the Region I (small-break LOCA) break spectrum analysis (see Section 4.6 of SE for WCAP-16996), but this limiting break size was not reported in either the LAR or WCAP-16996. During the audit (Reference 17), the NRC staff reviewed the limiting size and verified that it is close to the current licensing break size, based on a different approved LOCA analysis methodology (NOTRUMP EM) (see current UFSAR Section 15.3.1) (Reference 3).

Based on the above, the NRC staff finds the LOCA break spectrum analyses for Summer acceptable, because similar break spectrum analyses had been reviewed and evaluated during FSLOCA EM methodology development, and the PCT behavior was evaluated as being similar on either side of the boundary between Region I and Region II.

3.2.2.4 Single-Failure Assumption

The NRC reviewed the licensee's submittals to determine if sufficient analysis of possible failure modes of ECCS equipment and the effects of the failure modes on the ECCS performance was provided. As described in WCAP-16996, Volume III, Section 26.2.1.4, the loss of one train of SI is the limiting single-failure assumption for Summer. The NRC staff confirmed that this assumption was listed in LAR Table 1 "Plant Operating Range Analyzed and Key Parameters for VCSNS [Virgil C. Summer Nuclear Power Station]," Item 5.a.

If loss of an entire train is assumed, then the flow to the RCS will be minimized, since this assumption results in the loss of both high-head and low-head SI flows. However, this assumption also leads to reduced cooling of the containment since the train includes containment sprays and fan coolers. If the assumed single failure is the loss of a low-head pump only, this results in a higher flow to the RCS but also increases containment cooling, which reduces containment pressure.

The failure of a single train of the ECCS is assumed for the Summer LOCA transient calculation, but all trains of containment spray, fan coolers, and similar components are assumed, as in WCAP-16996-P-A, to be in operation for the calculation of the containment pressure in order to further reduce the calculated containment pressure (see LAR Attachment 4, Table 1, Item 5.a). The values for inputs pertinent only to the containment model are typically selected to provide a minimum containment pressure for conservative evaluation of LOCA reactor response. This has the effect of maximizing the break flow rates late in the transient, which is conservative.

3.2.2.5 LOCA Sequence of Events and Reactor System Response

The NRC performed this review to evaluate the LOCA sequence of events, including time delays before and after emergency power actuation; the calculation of the power, pressure, flow and temperature transients; the functional and operational characteristics of the reactor protective and ECCS systems in terms of how they affect the sequence of events; and operator actions required to mitigate the consequences of the accident.

Table 8 of LAR Attachment 4 contains a sequence of events for the transient that produced the Region I analysis PCT result. Figures 1 through 13 illustrate the calculated key transient response parameters for this transient. Control rod drop is modeled for breaks less than 1 square foot, assuming a 2.0-second signal delay time and a 4.0-second rod drop time. According to FSLOCA™ EM methodology, a maximum control rod drop time should be assumed. Summer TS 3.1.3.4 requires the individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds, which is consistent with the FSLOCA™ EM methodology recommendation. The RCP trip is modeled coincident with the reactor trip on the low pressurizer pressure setpoint for loss-of-offsite power transients. When the low pressurizer pressure SI setpoint is reached, there is a delay to account for emergency diesel generator startup, filling headers, and other steps, after which SI is initiated into the RCS. Figures 1 through 13 of LAR Attachment 4 illustrate the calculated key transient response parameters for this transient.

Table 9 of LAR Attachment 4 contains the sequence of events for a transient analyzed in Region II using similar assumptions along with the most limiting assumption for offsite power availability. Figures 14 through 27 of LAR Attachment 4 illustrate the key response parameters for this type of transient.

The transient responses as shown in the figures are found to be consistent with the sequence of events. The transients appropriately incorporate the intended functions of the reactor protection and ECCS systems. One significant difference between the analyses is that credit for control rod insertion is taken for the Region I (small-break LOCA) analysis to preclude re-criticality in the core, while credit for control rod insertion is not taken for the Region II (large-break LOCA) analysis due to the presence of large reactor core voiding during large-break LOCAs. Both sequences of events are justifiable, as based upon the expected values for the relevant monitored parameters and equipment functions. The problem end times are appropriate to have either the top of the core be covered or at least all fuel rods be quenched, ensuring that the maximum PCT and cladding oxidization have been captured.

Based on the above, the NRC staff concludes that no operator action, such as RCP trip or the low head SI pumps switchover from the reactor water storage supply to the containment sump, is required for Summer to mitigate LOCAs during the time spans considered in the analyses, specifically less than 3,000 seconds and less than 300 seconds for Regions I and II, respectively. Based on the licensee’s statement in LAR-20-176 (Reference 2) that the safety system actuation occurs automatically due to the reactor trip and engineering safety feature actuation systems, the NRC staff finds it acceptable that the LOCA mitigation does not require any new operator actions in order to satisfy the applicable acceptance criteria of 10 CFR 50.46.

3.2.2.6 LOCA Analysis Results to Comply with 10 CFR 50.46 Acceptance Criteria

The NRC performed this review to ensure that the LOCA analysis results based on FSLOCA™ EM will directly address the criteria in 10 CFR 50.46(b)(1), (b)(2), and (b)(3); the determination of PCT; maximum local oxidation; and CWO. The 10 CFR 50.46(b)(4) criterion (coolable geometry) is satisfied by meeting the first three criteria. The last criterion (long-term cooling) is satisfied by other means and is not part of the scope of this LAR.

In 10 CFR 50.46(a)(1)(i), the NRC requires that "uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded." In 10 CFR 50.46(b), the NRC lists the acceptance criteria, but does not explicitly specify in 10 CFR 50.46 how this probability should be evaluated or what its value should be.

Section 4 of RG 1.157 (Reference 6) provides additional clarification on one method to meet NRC's expectations for the acceptable implementation of the "high probability" requirement, stating that "a 95% probability is considered acceptable to the NRC staff." Further, RG 1.157 introduced the concept of confidence level as a possible refinement to the uncertainty.

In WCAP-16996-P-A, Section 31, (Reference 14) Westinghouse demonstrated the FSLOCA analysis methodology using the statistical method and WCOBRA/TRAC-TF2 code to determine the 95/95 (95-percent probability/95-percent confidence) figures of merit (PCT, maximum local oxidation, and CWO) and used them to demonstrate compliance with the 10 CFR 50.46 acceptance criteria. During the audit (Reference 17), the NRC staff reviewed and verified that compliance with the 10 CFR 50.46 acceptance criteria had been demonstrated for the Summer analyses.

The licensee presented the analysis results for Region I and Region II in LAR Attachment 4, Table 7 (as summarized in Table 1 below). The licensee stated that, with respect to Limitation and Condition 2, it discovered two errors in the WCOBRA/TRAC-TF2 Code after completion of the analysis for Summer. The errors were due to an incorrect calculation on the gamma energy redistribution that resulted in a 0- to 5-percent deficiency in the modeled hot rod and hot assembly rod-linear heat rates on a run-specific basis, depending on the as-sampled value for the multiplier uncertainty. These errors led to an impact on the PCTs of Region I and Region II with +12 degrees Fahrenheit (°F) and +31 °F, respectively. Since the final PCT as summarized in LAR Attachment 4, Table 7 (see Table 2 below), is determined conservatively by adding the impact of errors on the PCT to the analyzed PCT values, the NRC staff finds the final PCTs are less than 2200 °F, which meets the 10 CFR 50.46 (b)(1) acceptance criteria and is, therefore, acceptable.

A parenthetical in TS 6.9.1.11.c requires that the TS 3.2.2, "HEAT FLUX HOT CHANNEL FACTOR - $F_Q(z)$," be determined using the analysis methodology in a Westinghouse letter dated June 13, 1996 (Reference 18). Westinghouse Nuclear Safety Advisory Letter (NSAL)-09-5 Revision 1, "Relaxed Axial Offset Control FQ Technical Specification Actions," dated September 23, 2009 (Reference 22), stated that this methodology is the Relaxed Axial Offset Control (RAOC) methodology described in WCAP-10217-A, Revision 1A, "Relaxation of Constant Axial Offset Control – FQ Surveillance Technical Specification," February 1994 (Reference 23). NSAL-09-5 (Reference 22) also stated that for plants that have implemented the RAOC methodology, the peaking factor basis assumed in the current licensing-basis analysis may not be maintained under all conditions if the transient F_Q limit is not met. Since Summer used the RAOC methodology and is listed as an affected plant in NSAL-09-5, the NRC staff confirmed during the audit (Reference 10) that the impact of NSAL-09-5 on the application of FSLOCA™ EM to Summer, especially NSAL-09-5 (Reference 11) Safety Significance Items 4 and 5 regarding Heat Flux Hot Channel Factor $F_Q(z)$, have been evaluated and dispositioned. Specifically, the licensee verified that power distributions created in the FSLOCA™ EM library for the Summer analysis are consistent with, or bound, those allowed by Summer TS LCO 3.2.1 and satisfy the applicable acceptance criteria of 10 CFR 50.46. The analysis approach used for TS 3.2.2 is described in FSLOCA™ EM (Reference 6) Section 4.6.3.4 "Treatment of Core Power Distributions and Peaking Factors."

Based on the above, the NRC staff concluded that Summer would continue to comply with the criteria in 10 CFR 50.46 with WCAP-16996-P-A on the list of approved methodologies for determining core operating limits.

Table 1 Predicted Figures of Merit for Summer FSLOCA™ EM Analysis Results

Figure of Merit	Region I	Region II		Acceptance Criterion
		LOOP	OPA	
Peak Cladding Temperature (°F)	1,096 + 12 = 1,108	1,848 + 31 = 1879	1,837 + 31 = 1868	≤ 2200 °F
Maximum (Local) Cladding Oxidation	8.43%	9.13%	9.06%	≤ 17 %
Maximum (Corewide) Hydrogen Generation	0.00%	0.36%	0.33%	≤ 1 %

3.2.3 Conformance with Limitations and Conditions

The NRC staff's SE for WCAP-16996-P-A, Revision 1 (Reference 4), contains 15 limitations and conditions that must be satisfied to acceptably use a FSLOCA-based EM.

Limitation and Condition 1 specifies that FSLOCA must not be used to analyze the long-term core cooling phase of LOCA transients for the purpose of satisfying the requirements of 10 CFR 50.46(b)(5). In Section 2.3, "Compliance with FSLOCA™ EM Limitations and Conditions," of Attachment 4, "License Amendment Request Technical Evaluation," of the LAR, the licensee stated that the FSLOCA EM is not approved to demonstrate compliance with 10 CFR 50.46(b)(5). The NRC staff finds this to be acceptable and, therefore, considers Limitation and Condition 1 to be satisfied.

Limitation and Condition 2 specifies that FSLOCA™ is only to be used to analyze Westinghouse-designed three- or four-loop PWRs with cold-side ECCS injection only. The licensee stated that Summer is a three-loop Westinghouse PWR with cold-side injection. The NRC staff verified this by review of the licensee's UFSAR, as updated. Limitation and Condition 2 also states that, in plant-specific applications of the FSLOCA™ methodology, licensees should summarize the extent to which the approved methodology was followed and justify any departures from the approved methodology. In its application dated June 4, 2020, the licensee stated that the approved methodology was followed but it contains three known errors. Two of these errors are negligible and did not need to be accounted for, as discussed in LTR-NRC-19-6 (Reference 24), the NRC staff reviewed these errors and confirmed the licensee's assessment. The third error is related to the programming of the gamma energy redistribution during the transient. Though this error has a negligible effect on the system response (since it only affects the hot assembly), the effect on the peak cladding temperature

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the staff notified the South Carolina State official of the proposed issuance of the amendment on March 31, 2021. On March 31, 2021 the State official confirmed the State of Carolina had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement for the installation or use of a facility component located within the restricted area as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for protection against radiation." 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 off site, 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, and there has been no public comment on such finding published in the *Federal Register* (85 FR 48568; August 11 2020). Accordingly, the amendment meets the eligibility criteria for categorical exclusion in 10 CFR 51.22(c)(9). In accordance with 10 CFR 51.22(b), the NRC staff does not need to prepare an environmental impact statement or environmental assessment in connection with the issuance of the amendment.

8.0 REFERENCES

1 Dominion Energy, South Carolina, Inc., letter to U. S. Nuclear Regulatory Commission (NRC), "Dominion Energy South Carolina (DESC), Virgil C. Summer Nuclear Station (VCSNS) Unit 1, License Amendment Request LAR-20-142, Request for Technical Specification Change, Technical Specification 3.6.4, "Containment Isolation Valves"," April 30, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20121A185).

2 Dominion Energy South Carolina, Inc., (LAR-20-176) letter to U. S. Nuclear Regulatory Commission (NRC), "Dominion Energy South Carolina (DESC), Virgil C. Summer Nuclear Station (VCSNS) Unit 1, License Amendment Request LAR-20-142, Request for Technical Specification Change, Technical Specification 3.6.4, "Containment Isolation Valves," Response to," Request for Additional Information (RAI), Jaunary 7, 2021 (ADAMS Pkg Accession No. ML21007A339).

3 U. S. Nuclear Regulatory Commission (NRC), "VC Summer, Updated Final Safety Analysis Report (FSAR), Chapter 15 - Accident Analysis: Section 15.3 - Condition III - Infrequent Faults, Section 15:4 - Condition IV - Limiting Faults," August 2, 1984 (ADAMS Accession No. ML20094D127).

4 U. S. Nuclear Regulatory Commission (NRC), "Revised Final Safety Evaluation for Westinghouse Electric Company Topical Report WCAP-16996-P/WCAP-16996-NP, Volumes I, II and III, Revision 1, Realistic Loss-of-Coolant Accident Evaluation Methodology Applied to the Full Spectrum of Break Sizes," September 12, 2017 (ADAMS Accession No. ML17277A132).

5 U. S. Nuclear Regulatory Commission (NRC), "Virgil C.Summer, Updated Final Safety Analysis (UFSAR) Chapter 3 - Design of Structures, Components, Equipment and Systems, Section 3.1 - Conformance with NRC General Design Criteria," August 2, 1984 (ADAMS Accession No. ML20094C975).

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22 Westinghouse Electric Company (WEC), "Nuclear Safety Advisory Letter (NSAL)-09-05 - Relaxed Axial Offset Control FQ Technical Specification Actions," September 23, 2009 (ADAMS Accession No. ML17167A233.

23 Westinghouse Electric Company (WEC), "WCAP-10217-A, Revision 1A - Relaxation of Constant Axial Offset Control - FQ Surveillance Technical Specification," February 1994 (ADAMS Accession No. ML2006L190).

24 Westinghouse Electric Company (WEC), letter to NRC, "LTR-NRC-19-6, U. S. Nuclear Regulatory Commission 10 CFR 50-46 Annual Notification and Reporting for 2018; Changes or errors in ECCS evaluation models," February 7, 2019 (ADAMS Accession No. ML19042A379).

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SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1—CORRECTION TO AMENDMENT NO. 219 TO MODIFY TECHNICAL SPECIFICATION 6.9.1.11, “CORE OPERATING LIMITS REPORT” (EPID L-2020-LLA-0124), DATED AUGUST 10, 2021

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