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

June 30, 2021

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

Dear Mr. Sartain:

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

The amendment revises TS 6.9.1.11, "Core Operating Limits Report [COLR]," to add the Westinghouse Topical Report WCAP-16996-P-A, Revision 1, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," to the list of NRC-approved analytical methodologies approved for reference in the COLR.

A copy of the related safety evaluation and notice and environmental findings are also enclosed. The Commission's monthly *Federal Register* notice will include the notice of issuance.

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. Amendment No. 219 to NPF-12
2. Safety Evaluation

cc: Listserv

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

DOMINION ENERGY SOURTH CAROLINA, INC.

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 219
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 Dominion Energy South Carolina, Inc. (the licensee), dated June 4, 2020, as supplemented by letter dated January 7, 2021, 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 public 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, the license is hereby amended by a page change to the Technical Specifications as indicated in the attachment to this license amendment, and 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. 219, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. Dominion Energy South Carolina, Inc. 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 before restart from the fall 2021 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2021.06.30 15:01:35 -04'00'

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 and Technical Specifications

Date of Issuance: June 30, 2021

VIRGIL C. SUMMER NUCLEAR STATION, UNIT 1

ATTACHMENT TO LICENSE AMENDMENT NO. 219

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the renewed facility operating license with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Page	Insert Page
License Page 3	License Page 3
Technical Specifications 6-16a	Technical Specifications 6-16a

(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 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 Part 30, 40 and 70 to receive, possess and use at any time 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 of 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 m[a]y 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) <u>Maximum Power Level</u>

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 preoccupation 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 renewed license.

(2) <u>Technical Specifications and Environmental Protection Plant</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 219, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

CORE OPERATING LIMITS REPORT (Continued)

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).

d. WCAP-12472-P-A, “BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM,” August 1994, (W Proprietary).

WCAP-12472-P-A, Addendum 1-A, “BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM,” January 2000, (W Proprietary)

(Methodology for Specifications 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio.)

e. WCAP-13749-P-A, "Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement," March 1997, (Westinghouse Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

f. WCAP-12610-P-A, “VANTAGE + Fuel Assembly Reference Core Report,” April 1995 (W Proprietary). WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO™,” July 2006 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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 references to a statistically-based best estimate large-break loss-of-coolant accident (LBLOCA) and a 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

electrically operated components in the event that a loss of offsite power occurs simultaneously with a loss-of-coolant accident (LOCA), even assuming a single failure in the emergency power system such as the failure of one diesel to start.

The operation of the ECCS, following a LOCA, can be divided into two distinct modes:

(1) injection mode, in which any reactivity increase following the postulated accident is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system is replenished

(2) recirculation mode, in which long-term core cooling is provided during the accident recovery period

The principal mechanical components of the ECCS that provide core cooling immediately following a LOCA are the accumulators, the centrifugal charging pumps, and the residual heat removal pumps.

For large pipe ruptures, the reactor coolant system (RCS) is depressurized and voided of coolant rapidly, and a high flow rate of emergency coolant is required to quickly cover the exposed fuel rods and limit possible core damage. This high flow is provided by the passive accumulators, followed by the charging pumps and the residual heat removal pumps discharging into the cold legs of the RCS.

Emergency cooling is provided for small ruptures primarily by high head injection. Small ruptures are those that do not immediately depressurize the RCS below the accumulator discharge pressure. The centrifugal charging pumps deliver borated water at the prevailing RCS pressure to the cold legs of the RCS. During the injection mode, the charging pumps take suction from the refueling water storage tank.

The residual heat removal pumps take suction from the refueling water storage tank and deliver borated water to the RCS. These pumps begin to deliver water to the RCS only after the pressure has fallen below the pump shutoff head.

Core protection is afforded with the minimum engineered safety feature equipment, which is defined by consideration of the single-failure criterion. The minimum design case will ensure the entire break spectrum is accounted for and core cooling design bases are met. Summer Updated Final Safety Analysis Report (UFSAR) Section 15.3, “Condition III—Infrequent Faults,” and Section 15.4, “Condition IV—Limiting Faults,” (Reference 3) present the analyses for this case.

The proposed change is relevant to the ECCS insofar as the methods being changed are used to demonstrate its performance to meet the acceptance criteria set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.”

2.0 REGULATORY EVALUATION

The staff reviewed the proposed TS change for Summer using the requirements in 10 CFR 50.46; 10 CFR Part 50, “Domestic licensing of production and utilization facilities,” Appendix K, “ECCS Evaluation Models;” General Design Criterion (GDC) 35, “Emergency core

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 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 10 CFR 50.46(b)(5)

cannot be stated as being satisfied by application of the FSLOCA methodology." This restriction of the FSLOCA™ evaluation model (EM), which defines its applicability range as being exclusive from analyzing the long-term core cooling phase of postulated LOCA transients for the purpose of demonstrating compliance with 10 CFR 50.46(b)(5), is reflected in the FSLOCA EM Limitation and Condition 1. The licensee has recognized this limitation and condition in its compliance statement in that the analysis for Summer with the FSLOCA™ EM is only being used to demonstrate compliance with 10 CFR 50.46 (b)(1) through (b)(4).

2.1.1 Appendix K to 10 CFR Part 50

Appendix K to 10 CFR Part 50 consists of two parts:

(1) required and acceptable features of LOCA evaluation models
(2) documentation required for LOCA evaluation models

The first part specifies modeling requirements and acceptable methods for simulating significant physical phenomena throughout all phases of a design-basis LOCA event, including relevant heat sources, fuel rod performance, and thermal-hydraulic behavior. These regulatory requirements are not applicable for this LAR, because the licensee chose to apply a realistic method with an explicit accounting for uncertainties, consistent with 10 CFR 50.46(a)(1)(i).

The second part specifies requirements for the documentation of LOCA EMs, including a complete description, a code listing, sensitivity studies, and comparisons against experimental data.

2.1.2 10 CFR Part 50, Appendix A, General Design Criterion 35

GDC 35, Emergency core cooling, in 10 CFR Part 50, Appendix A, requires, A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of coolant and at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Section 3.1 of the Summer UFSAR (Reference 5) describes how the plant was designed to ensure compliance to GDC 35.

2.1.3 Technical Specification Requirements

Regulatory requirements related to the content of TSs are set forth in 10 CFR 50.36. Pursuant to 10 CFR 50.36, TSs are required to include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) surveillance requirements, (4) design features, and (5) administrative controls. The regulation does not specify the particular requirements to include in a plant's TSs.

The regulation at 10 CFR 50.36(c)(5) states that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." TS 6.9.1.11 requires administrative controls associated with the Summer COLR, which ensures core operating parameters are appropriate to demonstrate compliance to 10 CFR 50.36(c)(5).

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 13) 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 3, 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/WCAP-16996-NP, 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 13), 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)

Evaluation Model Implementation for Summer

3.2.1.1 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.

3.2.1.2 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.

3.2.1.3 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.

3.2.1.4 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, 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.

3.2.1.5 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.

3.2.1.6 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, 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 2 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 2 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 %

3.2.2 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

was estimated to be increased by 12 ºF, which maintains a margin far below the required PCT limit. Although the NRC staff finds this increase in PCT acceptable, the licensee should continue to include the error in reports submitted to NRC pursuant to 10 CFR 50.46(a)(3) until a reanalysis is performed that accounts for the error. Based on the above, the NRC staff considers Limitation and Condition 2 to be satisfied.

Limitation and Condition 3 specifies that containment backpressure calculations associated with FSLOCA™ analyses must be performed according to the approved FSLOCA™ methodology, using the containment pressure analysis Code (COCO) or LOTIC Codes with appropriate plant-specific conditions. The licensee stated that the containment pressure calculation was performed consistent with the approved methodology, using appropriate design parameters, design conditions, and engineered safety features. Minimum initial temperature was used to appropriately minimize the containment pressure before the LOCA, and no coatings were credited on any containment structures. Based on the above, the NRC staff finds that the Limitation and Condition 3 to be satisfied.

Limitation and Condition 4 specifies acceptable parameters for decay heat modeling and sampling, specifically restricts the FSLOCA™ EM from being used to run simulations longer than 10,000 seconds following the reactor trip unless the decay heat model is appropriately justified, and asks that the values of the decay heat multiplier used in the limiting Region I and Region II analyses be provided in plant-specific licensing submittals. The licensee stated that the decay heat multiplier used in the analysis is consistent with the NRC-approved methodology and that the simulations were executed for no longer than 10,000 seconds following the reactor trip. The licensee also provided the sampled values of decay heat uncertainty requested in the limitation and condition in Table 10 of the submittal. Based on the above, the NRC staff considers Limitation and Condition 4 to be satisfied.

Limitation and Condition 5 specifies assembly average and peak rod-length average burnup limits for the FSLOCA™ EM. The licensee stated that these limits were met in the analysis. The NRC staff reviewed the licensee's calculations in the regulatory audit and verified that the limits were met appropriately. Based on the above, the NRC staff considers that Limitation and Condition 5 to be satisfied.

Limitation and Condition 6 provides restrictions on the fuel performance code that is used to generate the fuel rod initial conditions, states that the fuel performance shall include the effects of fuel thermal conductivity degradation, and states that the initial fuel stored energy and gap pressure should be maximized. The licensee stated that PAD5 fuel performance data were used to generate the initial conditions and bounding maximum values for both fuel pellet average temperatures and rod internal pressure were used in the analysis. Since PAD5 is the latest version of a Westinghouse fuel performance Code approved for use by the NRC staff and because it contains the effects of thermal conductivity degradation, it is acceptable for use with FSLOCA™. The NRC staff therefore considers Limitation and Condition 6 to be satisfied.

Limitation and Condition 7 specifies the value of the interfacial drag multiplier that is to be used in the FSLOCA™ Region I (small break) analysis. The licensee stated that the appropriate multiplier was used throughout the analysis. Based on the above, the NRC staff considers Limitation and Condition 7 to be satisfied.

Limitation and Condition 8 specifies biased uncertainty contributors for Region I analyses in FSLOCA. The licensee stated that the required values were used. Based on the above, the NRC considers Limitation and Condition 8 to be satisfied.

Limitations and Conditions 9 and 10 are only applicable to plants that are not Westinghouse-designed three-loop PWRs. Since Summer is a Westinghouse-designed three-loop PWR, these limitations and conditions are not applicable to this LAR.

Limitation and Condition 11 provides requirements for declaring and documenting parameters related to the uncertainty analysis before executing FSLOCA™ and specifies that these parameters are not to be changed once documented. It also specifies that the operating ranges considered in the uncertainty analysis are to be provided with the plant-specific licensing submittal for NRC review. The licensee stated that the uncertainty analysis parameters were declared and documented before execution of the analysis and that the parameters were not changed once documented. The licensee also provided the plant operating ranges in Table 1 of the submittal. The NRC staff reviewed Table 1 and found that, in accordance with the limitation and condition, the plant operating ranges used in the analysis appropriately bounded the Summer TS operating limits. The NRC staff's audit of the licensee's documents also confirmed the licensee's statements about declaration and documentation of uncertainty parameters. Based on the above, the NRC considers Limitation and Condition 11 to be satisfied.

Limitation and Condition 12 specifies that the dynamic pressure losses between the main steam generator and the main steam safety valves should be appropriately considered and accounted for in design-basis LOCA analyses performed using FSLOCA™. The licensee stated that the plant model included a bounding dynamic pressure loss. The NRC staff verified the licensee's calculation notes and analysis during its audit. Based on the above, the NRC staff therefore considers Limitation and Condition 12 to be satisfied.

Limitation and Condition 13 provides modeling requirements for the upper head spray nozzle loss coefficients in the analysis. The licensee stated that the specific modeling requirements were met. The NRC staff verified the licensee's modeling during its regulatory audit. Based on the above, the NRC staff considers Limitation and Condition 13 to be satisfied.

Limitation and Condition 14 specifies how compliance should be demonstrated with the oxidation criteria in 10 CFR 50.46 depending on the oxidation relationship used to convert time at temperature to equivalent cladding reacted (ECR). Specifically, if Baker-Just is used, a 17-percent ECR limit is appropriate. If Cathcart-Pawel is used, a 13-percent ECR limit is appropriate. In both cases, the final result needs to include the pre-transient oxidation. The licensee specified that Baker-Just correlation was used and the result was summed with the pre-transient corrosion before comparison to a 17-percent ECR limit. Based on the above, the NRC staff considers Limitation and Condition 14 to be satisfied.

Limitation and Condition 15 specifies how offsite power should be included in the Region II (large break) LOCA analyses using FSLOCA™. Specifically, two separate sets of sampled statistical evaluations are to be performed—one with offsite power available and one without offsite power available. The licensee stated that the Region II uncertainty analysis was performed twice, once with a loss of offsite power and once with offsite power available. Both analyses exceeded the minimum sample size required by Limitation and Condition 11 and met the 10 CFR 50.46 acceptance criteria. Based on the above, the NRC staff considers Limitation and Condition 15 to be satisfied.

4.0 TECHNICAL EVALUATION SUMMARY

The NRC staff reviewed the information in the licensee's submittals supporting the analysis of the spectrum of postulated LOCA events for Summer, including the initial submittal (Reference 1) and responses to requests for additional information (Reference 2). The NRC staff also conducted a regulatory audit (Reference 17) to confirm information referenced in docketed submittals. The NRC staff concludes that the LOCA analysis with FSLOCA™ EM complies with the applicable requirements in 10 CFR 50.36, 10 CFR 50.46, Appendix K to 10 CFR Part 50, and GDC 35 and is, therefore, acceptable for use at Summer. This conclusion is based on the following:

(1) The licensee analyzed the performance of the ECCS with FSLOCA™ EM in accordance with 10 CFR 50.46. The analyses considered a spectrum of postulated break sizes and locations and were performed with an EM that meets the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The results of the analyses (Sections 3.1 through 3.2.3 of this SE) show that the ECCS with FSLOCA™ EM satisfies the 10 CFR 50.46 acceptance criteria.

(2) The licensee's evaluation shows that the use of the FSLOCA™ EM continues to demonstrate that the ECCS meets the requirements of GDC 35 with respect to the provision of abundant emergency core cooling that will transfer heat from the reactor core in the event of a LOCA, and that suitable redundancy of components and features has been provided so that the safety function can be accomplished, assuming a single failure by doing the following:

   a. demonstrating with the LOCA analysis performed with FSLOCA™ EM for Summer that abundant emergency core cooling has been provided to transfer heat from the reactor core filled in the event of a LOCA, and showing that suitable redundancy of components and features is provided so that the safety function is able to be accomplished, assuming a single failure, irrespective of whether its electrical power is supplied from offsite or onsite sources; and

   b. applying NRC-approved LOCA EM and methodology for the LOCA analysis with FSLOCA™ EM and meeting the limitations and conditions listed in the NRC staff SEs for the applied topical reports.

(3) The evaluation meets the Appendix K to 10 CFR Part 50 requirement stated in item (2) of Section 2.1.1 because the FSLOCA EM is NRC-approved and is appropriately documented.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that public health and safety 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 amendment will not be inimical to the common defense and security or to public health and safety.

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.

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).

6 U. S. Nuclear Regulatory Commission (NRC), "Regulatory Guide (RG) 1.157 - Best-Estimate Calculations of Emergency Core Cooling System Performance," May 31, 1989 (ADAMS Accession No. ML003739584).

7 U. S. Nuclear Regulatory Commission (NRC), "Regulatory Guide (RG) 1.203 - Transient and Accident Analysis Methods," December 30, 2005 (ADAMS Accession No. ML053500170).

8 U. S. Nuclear Regulatory Commission (NRC), "Information Notice 2011-21: Realistic Emergency Core Cooling System Evaluation Model Effects Resulting From Nucler Fuel Thermal Conductivity Degradation," December 13, 2011 (ADAMS Accession No. ML113430785).

9 U. S. Nuclear Regulatory Commission (NRC), "NUREG-0800 - Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition, Chapter 15, Section 15.0.2, Review of Transient and Accident Analysis Methods, Revision 0," March 2007 (ADAMS Accession No. ML070820123).

10 U. S. Nuclear Regulatory Commission (NRC), "NUREG-0800 - Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition, Chapter 15, Section 15.6.5, Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the ," Reactor Coolant Pressure Boundary, Revision 3, March 2007 (ADAMS Accession No. ML070550016).

11 Westinghouse Electric Company (WEC), "WCAP-12945-P-A, Volume 1, Revision 2 through Volume 5, Revision 1 - Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary - Non-Public).

12 Westinghouse Electric Company (WEC), "WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Mode Using the NOTRUMP Code," August 1985 (Non-Public).

13 Westinghouse Electric Company (WEC), "WCAP-10079-P-A, NOTRUMP, A Nodal Transient Small Break and General Network Code," August 1985 (Non-Public).

14 Westinghouse Electric Company (WEC), "WCAP-16996-P-A, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," Revision 1," November 2016 (Westinghouse Proprietary - Non-Public).

15 Dominion Energy South Carolina, Inc. (DESC) (formerly South Carolina Electric & Gas (SCE&G)) letter to NRC, "RC-12-0104: ECCS [Emergency Core Cooling System) Evaluation Model Revisions 30-Day Report; B.C. Summer 10 CFR 50.46 Reporting Text and Peak Cladding Temperature Impacts," October 16, 2012 (ADAMS Accession No. ML12293A071).

16 Westinghouse Electric Company (WEC), "WCAP-17462-P-A, Revision 1, Westinghouse Performance Analysis and Design Model (PAD5)," November 2017 (Non-Public).

17 U. S. Nuclear Regulatory Commission (NRC) letter to , "Virgil C. Summer Nuclear Station, Unit 1 - Audit Summary: Proposed License Amendment Request for the Addition of Westinghouse Topical Report WCAP-16996-P-A, Revision 1, to the Core Operating Limits Report," January 25, 2021 (ADAMS Accession No. ML21012A345).

18 N. J Liparulo letter to NRC, "Re-analysis Work Plans Using Final Best-Estimate Methodology," June 13, 1996 (ADAMS Accession No. ML20117M224).

19 U. S. Nuclear Regulatory Commission (NRC) letter to James A. Gresham, Westinghouse Electric Company, "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. ML17207A124).

20 U. S. Nuclear Regulatory Commission (NRC), "NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [light-water reactors] Edition, Section 6.3 - Emergency Core Cooling System, Revision 3," March 2007 (ADAMS Accession No. ML070550068).

21 Westinghouse Electric Company (WEC) Letter to NRC (LTR-NRC-14-17), "Submittal of Westinghouse Responses to WCAP-16996-PP, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Breack Sizes (FULL SPECTRUM LOCA Methodology) Request for Additional Information RAIs 36-39," March 24, 2014 (ADAMS Accession No. ML14090A022 - Proprietary/Non-Proprietary).

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—ISSUANCE OF AMENDMENT NO. 219 TO MODIFY TECHNICAL SPECIFICATION 6.9.1.11, "CORE OPERATING LIMITS REPORT" (EPID L-2020-LLA-0124), DATED JUNE 30, 2021

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