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

May 24, 2022

Mr. Bob Coffey
Executive Vice President, Nuclear
and Chief Nuclear Officer
Florida Power & Light Company
NextEra Energy Seabrook, LLC
Mail Stop: EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 – ISSUANCE OF AMENDMENT NOS. 296 AND 289 REGARDING IMPLEMENTATION OF FULL SPECTRUM LOSS-OF-COOLANT ACCIDENT (FSLOCA) METHODOLOGY (EPID L-2021-LLA-0070)

Dear Mr. Coffey:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 296 to Subsequent Renewed Facility Operating License No. DPR-31 and Amendment No. 289 to Subsequent Renewed Facility Operating License No. DPR-41 for the Turkey Point Nuclear Generating Unit Nos. 3 and 4, respectively (Turkey Point). These amendments revise the Technical Specifications (TS) in response to your application dated April 15, 2021, supplemented by letter dated November 19, 2021.

The amendments revise TS 6.9.1.7 to change the loss-of-coolant accident (LOCA) methodology to reflect the adoption of WCAP-16996-P-A, Revision 1, ''Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM™ LOCA Methodology).''

Enclosure 3 to this letter contains proprietary information, when separated, this letter is decontrolled.

A copy of our related safety evaluation is also enclosed. Notice of issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Michael Mahoney, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 296 to DPR-31
2. Amendment No. 289 to DPR-41
3. Safety Evaluation (Proprietary)
4. Safety Evaluation (Non-Proprietary)

cc: Listserv

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO SUBSEQUENT RENEWED FACILITY OPERATING LICENSE

Amendment No. 296
Subsequent Renewed License No. DPR-31

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated April 15, 2021, supplemented by letter dated November 19, 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 the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Subsequent Renewed Facility Operating License No. DPR-31 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 296, are hereby incorporated into this subsequent renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this subsequent renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented by no later than the next Unit 3 fuel reload campaign.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Subsequent Renewed
Facility Operating License
and Technical Specifications

Date of Issuance: May 24, 2022

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

AMENDMENT TO SUBSEQUENT RENEWED FACILITY OPERATING LICENSE

Amendment No. 289
Subsequent Renewed License No. DPR-41

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

   A. The application for amendment by Florida Power & Light Company (the licensee) dated April 15, 2021, supplemented by letter dated November 19, 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 the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   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 amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Subsequent Renewed Facility Operating License No. DPR-41 is hereby amended to read as follows:

    B. Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 289, are hereby incorporated into this subsequent renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this subsequent renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented by no later than the next Unit 4 fuel reload campaign.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Subsequent Renewed
Facility Operating License
and Technical Specifications

Date of Issuance: May 24, 2022

ATTACHMENT TO LICENSE AMENDMENT NOS. 296 AND 289

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

SUBSEQUENT RENEWED FACILITY OPERATING LICENSE NOS. DPR-31 AND DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace the following pages of the Subsequent Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
DPR-31, page 3	DPR-31, page 3
DPR-41, page 3	DPR-41, page 3

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove	Insert
6-18	6-18
6-19	6-19

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 below:

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 296, are hereby incorporated into this subsequent renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this subsequent renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than July 19, 2012.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Fire Protection

FPL shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment requests dated June 28, 2012 and October 17, 2018 (and supplements dated September 19, 2012; March 18, April 16, and May 15, 2013; January 7, April 4, June 6, July 18, September 12, November 5, and December 2, 2014; and February 18, 2015; October 24, and December 3, 2018; and January 31, 2019), and as approved in the safety evaluations dated May 28, 2015 and March 27, 2019. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the

A. Maximum Power Level

The applicant is authorized to operate the facility at reactor core power levels not in excess of 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 289, are hereby incorporated into this subsequent renewed operating license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this subsequent renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than April 10, 2013.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

D. Fire Protection

FPL shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment requests dated June 28, 2012 and October 17, 2018 (and supplements dated September 19, 2012; March 18, April 16, and May 15, 2013; January 7, April 4, June 6, July 18, September 12, November 5, and December 2, 2014; and February 18, 2015; October 24, and December 3, 2018; and January 31, 2019), and as approved in the safety evaluations dated May 28, 2015 and March 27, 2019. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine F_Q (Z), $F_\Delta H$ and the K(Z) curve shall be those previously reviewed and approved by the NRC in:

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

2. WCAP-12610-P-A, “VANTAGE+ Fuel Assembly Reference Core Report,” S. L. Davidson and T. L. Ryan, April 1995.

3. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, “Optimized ZIRLO™,” July 2006.

The analytical methods used to determine Overtemperature ΔT and Overpower ΔT shall be those previously reviewed and approved by the NRC in:

1. WCAP-8745-P-A, “Design Basis for the Thermal Overtemperature ΔT and Overpower ΔT Trip Functions, ” September 1986

2. WCAP-9272-P-A, “ Westinghouse Reload Safety Evaluation Methodology,” July 1985

The analytical methods used to determine Safety Limits, Shutdown Margin - $T_{avg} > 200°F$, Shutdown Margin - $T_{avg} \leq 200°F$, Moderator Temperature Coefficient, DNB Parameters, Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, “Westinghouse Reload Safety Evaluation Methodology,” July 1985.

The analytical methods used to support the suspension of the measurement of the Moderator Temperature Coefficient in accordance with Surveillance Requirement 4.1.1.3.b shall be those previously reviewed and approved by the NRC in:

1. WCAP-13749-P-A, “Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement,” March 1997.

2. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

3. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," August 2004.

4. WCAP-16045-P-A, Addendum 1-A, "Qualification of the NEXUS Nuclear Data Methodology," August 2007.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 296 AND 289

TO SUBSEQUENT RENEWED FACILITY OPERATING LICENSE NOS. DPR-31 AND DPR-41

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By letter dated April 15, 2021 (Reference 1) as supplemented by letter dated November 19, 2021 (Reference 2), to the U.S. Nuclear Regulatory Commission (NRC), Florida Power & Light Company (FPL, the licensee) submitted a license amendment request (LAR) for Turkey Point Nuclear Generating Unit Nos. 3 and 4 (Turkey Point). The proposed amendment would revise the Turkey Point Technical Specification (TS) 6.9.1.7 to reflect the adoption of topical report (TR) WCAP-16996-P-A, Revision 1, “Realistic LOCA [loss-of-coolant accident] Evaluation Methodology Applied to the Full Spectrum of Break Sizes (Full Spectrum LOCA Methodology),” (Reference 3) as a reference in the Core Operating Limits Report (COLR). The added reference identifies the analytical method used to determine the core operating limits for the large break LOCA (LBLOCA) and the small break LOCA (SBLOCA) events described in the Turkey Point Updated Final Safety Analysis Report (UFSAR) Sections 14.3.2.1 and 14.3.2.2 respectively. The amendment proposes to delete references 1 through 6 in the TS 6.9.1.7 COLR list of analytical methods.

The supplement dated November 19, 2021, provided additional information that clarified and corrected the application, did not expand the scope of the application as originally noticed, and did not change the staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on September 7, 2021 (86 FR 50188).

This safety evaluation contains proprietary information, which is marked with double brackets and bold font such as **[[This is an example.]].**

Enclosure 4

2.0 REGULATORY EVALUATION

2.1 Description of Proposed TS Changes

Additions are shown in double-underline and deletions in double-strike-through.

In current TS 6.9.1.7, the licensee proposes to delete the following 6 references from the COLR used for determining the Heat Flux Hot Channel Factor, $F_Q(Z)$, the Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta}H$, and the K(Z) curve (normalized $F_Q(Z)$ as a function of core height):

1. ~~WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS [Emergency Core Cooling System] Evaluation Model – 1981 Version," February 1982.~~
2. ~~WCAP-10054-P-A, (Proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985~~
3. ~~WCAP-10054-P-A, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.~~
4. ~~WCAP-16009-P-A, "Realistic Large-break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", January 2005.~~
5. ~~USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,'" June 28, 1996.**~~
6. ~~Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology." **~~

Additionally, the licensee proposed to delete the associated footnote (from deleted references 5 and 6 above) as shown below:

> ~~**As evaluated in NRC Safety Evaluation dated December 20, 1997." to above References 5 and 6.~~

For determining $F_Q(Z)$, $F_{\Delta}H$, and the K(Z) curve, the licensee proposes to add the following document in TS 6.9.1.7, as Reference 1, in the COLR list.

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

The licensee proposes to renumber "WCAP-12160-P-A, VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995 (current Reference 7) and "WCAP-12160-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," July 2006 (current Reference 8) in the same COLR list, as References 2 and 3 respectively.

2.2 Description

As stated in Turkey Point UFSAR, revision issued dated May 12, 2021, Chapter 1, Section 1.2.2, “Nuclear Steam Supply System,” each Nuclear Steam Supply System consists of a Pressurized Water Reactor (PWR), reactor coolant system (RCS), and associated auxiliary fluid systems. The RCS is arranged as three closed reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump (RCP) and a steam generator (SG). An electrically heated pressurizer is connected to one of the loops.

Regarding the reactor core, Turkey Point UFSAR, Chapter 3, Section 3.1.1 states:

> The reactor core is a three-region cycled core. The fuel rods are cold worked partially annealed Zircaloy-4, ZIRLO® or Optimized ZIRLO™ tubes containing slightly enriched uranium dioxide fuel. All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life.
>
> The fuel assembly consists of the rod cluster control (RCC) guide thimbles fastened to the grids and the top and bottom nozzles. The fuel rods are held in this assembly at seven points along their length by spring-clip grids which provide a very stiff support for the fuel rods.
>
> The Turkey Point Units are loaded with Westinghouse seven grid 15 x 15 Upgrade Assemblies (Upgrade) starting with Unit 3 Cycle 25 and Unit 4 Cycle 26.

2.3 Regulations and Guidance Documents

Regulations

The NRC staff considered the following regulations in its review of the proposed changes.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR 50) paragraph 36(c)(5), “Administrative controls,” provide provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. This applies to the list of references to approved methods to be used to determine the core operating limits contained in the COLR.

The regulation in 10 CFR 50.46(a)(1)(i) states that an acceptable Emergency Core Cooling System (ECCS) evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor during LOCAs.

The regulations in 10 CFR 50.46(b) require that during a LOCA event, the following criteria are satisfied:

(1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200° F.

(2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) *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) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS [emergency core cooling system], the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The regulation in 10 CFR Part 50, Appendix K, Part II specifies documentation requirements for the emergency core cooling performance evaluation models specified in 10 CFR 50.46(a)(1)(i).

Title 10 of the *Code of Federal Regulations*, Part 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," provides a list of the minimum design requirements for nuclear power plants. Turkey Point was licensed prior to the issuance of the GDC listed in 10 CFR 50, Appendix A. However, Turkey Point committed to the 1967 proposed draft GDC as discussed in its Updated Final Safety Analysis Report (FSAR). The following 1967 proposed draft GDC are applicable:

- GDC 15 states that protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

- GDC 37 states that engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

- GDC 41 states that engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

- GDC 42 states that engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

- GDC 44 states that at least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core

cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe.

- GDC 49 states that the containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

- GDC 52 states that where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component.

Guidance Documents

The NRC staff used the following documents in its review of this license amendment request.

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition,” (Reference 14)

    - Section 15.6.5, Revision 3, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary,” March 2007.

    - Section 6.2.1.5, Revision 3, “Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies,” March 2007.

- NRC Regulatory Guide (RG) 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” May 1989 (Reference 16).

- NRC RG 1.203, “Transient and Accident Analysis Methods,” December 2005 (Reference 17).

- NRC Information Notice (IN) 2011-21, “Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation,” December 13, 2011 (Reference 6).

- NRC IN 1998-29, “Predicted Increase in Fuel Rod Cladding Oxidation” (Reference 15).

- NRC Generic Letter (GL) 1988-16, “Removal of Cycle Specific Parameter Limits from Technical Specifications,” October 4, 1988 (Reference 4).

3.0 TECHNICAL EVALUATION

The NRC staff evaluation of the proposed changes in the LAR consists of assuring the following:

- TS 6.9.1.7 changes are acceptable.
- The Full Spectrum LOCA (FSLOCA) evaluation model (EM) in WCAP-16996-P-A is appropriately applied.
- Application of FSLOCA EM for FSLOCA analysis meets the requirements of 10 CFR 50.46(b)(1) to (b)(4).
- Acceptance criteria of 10 CFR 50.46(b)(5) continue to be satisfied using the current methodology and remain as the licensing basis.
- GDCs listed in Section 2.3 above are satisfied.
- Limitations and conditions listed in WCAP-16996-P-A (Reference 3), Table 22 of the NRC's safety evaluation report (SER), are satisfied.

3.1 Evaluation of TS Changes

TS 6.9.1.7 provides the parameters list for which core operating limits should be established and documented in the COLR before each reload cycle or any remaining part of a reload cycle. It also lists the NRC-approved analytical methods for determining $F_Q(Z)$, $F_{\Delta}H$, and the K(Z) curve.

The proposed change replaces the 6 COLR references listed in Section 2.1 above with the WCAP-16996-P-A (Reference 3) methodology for determining $F_Q(Z)$, $F_{\Delta}H$, and the K(Z) curve. NRC GL 1988-16 (Reference 4) states that it is acceptable for licensees to control reactor physics parameter limits by specifying an NRC-approved calculation methodology. These parameter limits may be removed from the TS and placed in a cycle-specific COLR that is required to be submitted to the NRC every operating cycle or each time it is revised. The guidance in the GL 1988-16 will continue to be met because the proposed change in TS 6.9.1.7 identifies the NRC-approved methodologies that are used to determine the Turkey Point core operating limits. These changes are administrative in nature since the analytical methods of current COLR References 1 through 6 are being replaced by WCAP-16996-P-A, Revision 1, and renumbering existing References 7 and 8 does not impact any applicable requirements.

The NRC staff considers the proposed TS change to be acceptable because the WCAP-16996-P-A is an NRC-approved methodology and therefore would continue to provide administrative controls consistent with 10 CFR 50.36(c)(5).

3.2 FSLOCA Analysis

The FSLOCA EM methodology divides the break spectrum into two regions, termed as Region I and Region II. The Region I analysis is for SBLOCAs and the Region II analysis is for LBLOCAs. According to the FSLOCA EM, the SBLOCA (Region I) and LBLOCA (Region II) analyses are independent, separable, and do not influence each other. The licensee provided analyses for both SBLOCA (Region I) and LBLOCA (Region II) in the LAR.

Methodology

The current NRC-approved Westinghouse (W) best-estimate methodology used for the Turkey Point LBLOCA analysis described in WCAP-16009-P-A (Reference 5) is termed as Automated Statistical Treatment of Uncertainty Method (ASTRUM) EM. This methodology is applicable to Westinghouse designed (a) 3-loop and 4-loop PWRs with ECCS injection into the RCS cold legs, and (b) 2-loop PWRs with upper plenum injection. The ASTRUM EM uses WCOBRA/TRAC as the analysis code and is applicable only for the LBLOCA analysis with a minimum break size of 1.0 ft^2. The ASTRUM EM was developed consistent with the criteria in RG 1.157 (Reference 16).

As stated in Section 1.2.2 of NRC SER for WCAP-16996-P-A (Reference 3), the FSLOCA EM is built on the ASTRUM EM by extending the applicability of the WCOBRA/TRAC code to include the full spectrum of break sizes postulated in the RCS cold leg. The break sizes considered include any size in which the break flow is beyond the capacity of the normal charging pumps, up to and including a double-ended guillotine rupture in the RCS cold leg with a break flow area equal to two times the pipe flow area. For the development of FSLOCA EM, the licensee used the guidance in RG 1.157 (Reference 16) and RG 1.203 (Reference 17).

The FSLOCA EM methodology uses the WCOBRA/TRAC-TF2 code to analyze the RCS thermal-hydraulic response for a full spectrum of break sizes. This methodology is currently applicable to Westinghouse 3- and 4-loop plants with RCS cold leg injections. Since Turkey Point is a Westinghouse designed 3-loop plant with ECCS injection in the RCS cold legs, the FSLOCA EM is applicable.

The FSLOCA EM meets the regulation in 10 CFR 50.46(a)(1)(i) with respect to being an acceptable ECCS evaluation model because it is a best-estimate methodology and realistically describes the behavior of the reactor during LOCAs.

The FSLOCA EM meets the regulation in 10 CFR Part 50, Appendix K, that specifies documentation requirements for the emergency core cooling performance evaluation models specified in 10 CFR 50.46.

NRC Information Notice (IN) 2011-21 (Reference 6) notified addressees of concerns on the impact of irradiation on fuel thermal conductivity and its potential to result in significantly higher predicted peak cladding temperature (PCT) in realistic ECCS EMs. The licensee proposes to use the FSLOCA EM to meet its regulatory commitment in a letter dated March 27, 2018 (Reference 7) by updating its licensing basis analysis to account for the thermal conductivity degradation (TCD) of fuel pellets in the FSLOCA analyses using the NRC-approved for generic use PAD5 code. The FSLOCA EM explicitly accounts for the effects of fuel pellet TCD by using the fuel rod performance input data generated by the PAD5 code described in NRC-approved Topical Report (TR) WCAP-17642-P-A (Reference 8). The fuel pellet thermal conductivity input to the WCOBRA/TRAC-TF2 code used in the FSLOCA EM accounts for the TCD of fuel pellets. Therefore, it addresses IN 2011-21 (Reference 6) and satisfies the licensee's regulatory commitment in the letter dated March 27, 2018 (Reference 7).

For the calculation of the transient containment minimum back pressure as a boundary condition at the break for the FSLOCA analysis, the licensee used the NRC-accepted Westinghouse COCO [Containment Pressure Analysis Code] methodology described in TRs WCAP-8339 (Reference 10) and WCAP-8327 (Reference 11) by integrating it into the

WCOBRA/TRAC-TF2 code. The mass and energy (M&E) release calculated by the WCOBRA/TRAC-TF2 code at the end of each timestep in the LOCA transient is transferred as an input to the COCO code for calculating the containment back pressure as a boundary condition at the break. The Westinghouse COCO code is an NRC-accepted code for calculating the containment pressure response.

The NRC staff finds that the licensee (1) appropriately applied the proposed FSLOCA EM for the FSLOCA analysis in order to satisfy 10 CFR 50.46(b)(1) through (b)(4) as described below, and (2) used NRC-accepted Westinghouse COCO code for calculating the transient minimum containment back pressure at the break.

SBLOCA (Region I) Analysis

In the LAR, Attachment 1, Section 3.1, the licensee described the phenomena occurring during a SBLOCA event by dividing the transient into the following phases: (a) Blowdown, (b) Natural Circulation, (c) Loop Seal Clearance, (d) Boil-Off, and (e) Core Recovery. The data used by the licensee in the SBLOCA event analysis is provided in the following tables in the LAR, Attachment 1:

- Table 1 - Plant operating range analyzed and key parameters
- Table 5 - SI flow versus pressure
- Table 5a - SI flow versus pressure after switchover to cold leg recirculation
- Table 7 - SG MSSV parameters
- Table 9 - Sequence of events for the analysis PCT Case
- Table 11 - Sampled value of decay heat uncertainty multiplier DECAY_HT

The licensee's analysis used the following assumptions:

- Most limiting single failure of one ECCS train.

- The control rod drop is modeled for the small breaks assuming a 2-second signal delay time and a 2.4 second rod drop time.

- RCP trip is modeled coincident with reactor trip on the low pressurizer pressure setpoint for loss of offsite power (LOOP) transients.

- When the low pressurizer pressure safety injection (SI) setpoint is reached, the SI is initiated into the RCS after a delay time that accounts for the emergency diesel generators start-up and filling of headers.

The licensee stated that the transient that produced the most limiting PCT result is a cold leg break with a break diameter of 2.1-inches.

LBLOCA (Region II) Analysis

In the LAR, Attachment 1, Section 4.1, the licensee described the phenomena occurring during an LBLOCA event by dividing the transient into the following phases: (a) Blowdown – Critical Heat Flux (CHF) Phase, (b) Blowdown – Upward Core Flow Phase, (c) Blowdown – Downward Core Flow Phase, (d) Refill Phase, and (e) Reflood Phase. The data used by the licensee in the

LBLOCA event analysis is provided in the following tables in the LAR, Attachment 1:

- Table 1 - Plant operating range analyzed and key parameters
- Table 2 - Containment data used for the calculation of containment back pressure
- Table 3 - Fan cooler data used for the calculation of containment back pressure
- Table 4 - Heat sink data used for the calculation of containment back pressure
- Table 6 - SI flow versus pressure
- Table 10 - Sequence of events for the analysis PCT Case, i.e., LOOP case
- Table 11 - Sampled value of decay heat uncertainty multiplier, DECAY_HT

Westinghouse letters LTR-NRC-18-30, dated July 18, 2018 (Reference 12) and LTR-NRC-19-6, dated February 7, 2019 (References 13) to the NRC reported errors and some changes to be made in the FSLOCA EM. The NRC staff noted that these letters did not report the error that impacts the gamma energy redistribution multiplier identified in Virginia Electric and Power Company letter to NRC dated August 31, 2020 (Reference 9). In a letter dated November 19, 2021 (Reference 2), the licensee confirmed that the FSLOCA EM for Turkey Point used the WCOBRA/TRAC-TF2 code which incorporated the error corrections and changes identified in the above Westinghouse letters. Regarding the error that impacts the gamma energy redistribution multiplier, the licensee stated that this was not an error in the FSLOCA EM or the WCOBRA/TRAC-TF2 code; it was an error in the plant-specific implementation of the methodology for North Anna Units 1 and 2, and therefore was not reported in the Westinghouse communications as a methodology error for the FSLOCA EM. The staff finds that error did not occur in the execution of the Turkey Point analysis using the FSLOCA EM, and as a result, the as-approved methodology related to gamma energy redistribution was correctly applied.

In a letter dated November 19, 2021 (Reference 2), the licensee referred to Section 30 of WCAP-16996-P-A (Reference 3) for the method used for the LBLOCA uncertainty analysis. This method is based on Monte Carlo sampling of all uncertainty contributors which led to the results from which upper tolerance limits are derived for the PCT, maximum local oxidation (MLO) and core-wide oxidation (CWO). **[[**

]]

The licensee performed the LBLOCA analysis for both offsite power available (OPA) and LOOP cases assuming the most limiting single failure using the FSLOCA EM as described in WCAP-16996-P-A, with the exception that the licensee made the changes and error corrections reported in LTR-NRC-18-30 and LTR-NRC-19-6.

The NRC staff noted differences in some of the input parameters between the current analysis of record documented in the UFSAR and the proposed analysis of record presented in the LAR. These differences are shown in Table 1 below.

Table 1: Differences Between LAR and UFSAR Values

Parameter	LAR, Attachment 1		UFSAR	
	Value	Table	Value	Table
Accumulator temperature (T_{ACC})	80°F ≤ T_{ACC} ≤ 130°F	1	85°F ≤ T_{ACC} ≤ 126°F	14.3.2.1-8
Steam generator tube plugging	≤ 15%	1	≤ 5%	14.3.2.1-1
Initial spray temperature	34°F	2	39°F	14.3.2.1-2
RWST Temperature	34°F	2	39°F	14.3.2.1-2
Minimum initial containment pressure at full power operation	14.1 psia	2	13.26 psia	14.3.2.1-2
Minimum initial containment temperature at full power operation	80°F	2	90°F	14.3.2.1-2

In a letter dated November 19, 2021 (Reference 2), the licensee provided the following explanations for the differences noted in Table 1 above:

> … LOCA analysis is relatively insensitive to the steam generator tube plugging input and so it was increased to improve operating margins.
>
> The minimum initial containment pressure and temperature changes are an example of selecting inputs consistent with the approved FSLOCA methodology. Limitation and Condition 3 on the FSLOCA methodology states that for the purpose of calculating "a conservatively low, although not explicitly bounded, containment pressure, the input to the COCO model will be based on appropriate plant-specific containment design parameters and initial conditions." As such, an acceptable plant-specific initial containment temperature and pressure were used for the purpose of modeling the containment pressure response, consistent with the Limitation and Condition. A minimum value based on plant operating data was applied in the FSLOCA EM analysis. In a similar manner, a plant-specific minimum initial containment temperature was established based on plant operating data.
>
> Lastly, the other temperatures (accumulator, initial spray, and RWST) were modified to account for instrument uncertainties, and to add conservatism to safety analysis margin.

Based on the above explanations provided by the licensee, the NRC staff finds the values listed in Table 1 above that are used in the proposed FSLOCA analysis to be acceptable.

Results

Table 2 below shows the licensee's SBLOCA and LBLOCA analysis results for the PCT, MLO, and CWO, which are the same as in LAR Attachment 1, Table 8. The maximum values of PCT, MLO, and CWO are 1981°F, 11.06%, and 0.84%, respectively, which are based on the LOOP case from the LBLOCA analysis. In letter dated November 19, 2021 (Reference 2), the licensee stated that the MLO results were determined based on the pre-transient oxidation plus the transient oxidation.

LAR (Reference 1), Attachment 1, Figure 21 shows the containment back pressure for the transient that produced the limiting PCT result.

In a letter dated November 19, 2021 (Reference 2), the licensee provided the following:

- Figures 1a and 1b, which show PCT versus effective break area for the OPA and the LOOP cases, respectively. These graphs reflect the combined effect of the break size and break flow model uncertainties.

- Figures 2a and 2b, which show transient maximum local oxidation (or transient equivalent cladding reacted (ECR)) versus PCT for the OPA and the LOOP cases, respectively.

- Figures 3a and 3b, which show CWO versus PCT for the OPA and the LOOP cases, respectively.

Table 2: Results

Results	SBLOCA Analysis Value	LBLOCA Analysis Value With OPA	LBLOCA Analysis Value With LOOP	Acceptance Criteria in 10 CFR 50.46
95/95 PCT	1475°F	1947°F	1981°F	(b)(1) 2200°F
95/95 MLO	8.02%	10.91%	11.06%	(b)(2) 17%
95/95 CWO	0.06%	0.73%	0.84%	(b)(3) 1%

NRC Staff Evaluation of the SBLOCA and LBLOCA Analysis and Results

The NRC staff finds that the licensee's SBLOCA and LBLOCA analysis is acceptable based on the following:

- The assumptions and key inputs, which include core parameters, RCS parameters, and containment parameters, are consistent with the plant configuration and current licensing basis as described in the UFSAR with the exception of those identified in Table 1, for which the licensee provided acceptable justification.

- UFSAR Table 14.3.2.1-8 summarizes the peaking factor margin and burndown credits supported by the best-estimate LBLOCA analysis.

- In accordance with the requirement of Limitation and Condition 11 from the staff's safety evaluation on WCAP-16996-P-A, the licensee provided the plant operating ranges over which the uncertainty values are defined for SBLOCA and LBLOCA analysis which are consistent with UFSAR Table 14.3.2.1-1.

- The uncertainty analysis is consistent with the FSLOCA EM.

- The licensee used the updated FSLOCA EM after making changes and correcting errors reported in Westinghouse 10 CFR 50.46 letters LTR-NRC-18-30 (Reference 12) and LTR-NRC-19-6 (Reference 13) to NRC.

- For the calculation of a minimum containment back pressure as a boundary condition at the break, which is conservative for the PCT calculation, the licensee integrated the COCO code into the WCOBRA/TRAC-TF2 thermal-hydraulic code and used appropriate input parameters to include the effects of the installed pressure reducing systems, i.e., assuming all trains of the containment spray system and fan coolers are in operation.

- The licensee performed the LBLOCA analysis for both OPA and LOOP cases, assuming the most limiting single failure of one ECCS train.

- For the LBLOCA uncertainty analysis, the licensee used **[[ ]]** simulations to simultaneously bound the 95th percentile values of PCT, MLO, and CWO with a 95 percent confidence level as predicted by the statistical theory.

- The error identified for some licensees that impacts the gamma energy redistribution multiplier was not an error in the FSLOCA EM or the WCOBRA/TRAC-TF2 code used in the execution of the Turkey Point analysis, and therefore, the FSLOCA EM was correctly applied.

- The licensee calculated the MLO as the sum of pre-transient corrosion and transient oxidation consistent with the position in Information Notice (IN) 1998-29 (Reference 15).

- The licensee has satisfied all limitations and conditions for the analysis as evaluated by the NRC staff in Section 3.5 below.

3.3 Compliance with 10 CFR 50.46

The results in Table 2 for the PCT, MLO, and CWO show significant margins relative to the regulatory requirements in 10 CFR 50.46(b)(1), (b)(2), and (b)(3). Regarding compliance with 10 CFR 50.46(b)(4), the NRC staff finds that coolable core geometry is maintained because of the following:

- The criteria in 10 CFR 50.46(b)(1) through (b)(3) are satisfied.

- Based on Section 32.1 of the FSLOCA EM, the effects of LOCA and seismic loads on the core geometry do not need to be considered unless the fuel assembly grid deformation extends beyond the core periphery (i.e., deformation in a fuel assembly with no sides adjacent to the core baffle plates). In a letter dated November 19, 2021 (Reference 2), the licensee stated

    > The FSLOCA EM analysis does not affect the existing calculations that support the analysis of record related to combined LOCA and seismic loads, and the conclusion is retained from prior calculations and is credited in the current LOCA design basis analyses. That is, the previous calculations on grid deformation due to combined LOCA and seismic loads remain valid. As described in Section 3.2.3 of the Updated Final Safety Analysis Report (UFSAR) regarding the combined LOCA and seismic loads, “Some ZIRLO grid designs experience grid crush during the most severe load conditions of a combined seismic/LOCA event.

However, crushed grid locations are limited to the periphery of the core and coolable geometry is maintained. Control rod insertability and fuel cladding integrity are also maintained."

- As stated in UFSAR Section 14.3.2.1.6, the above conclusion is based on taking credit for the low power generation in the peripheral assemblies, and the observation that any flow redistribution which may occur would tend to benefit the inboard assemblies. The UFSAR Section 14.3.2.1.6 also states that for Turkey Point, grid crushing has been predicted to occur only on the core periphery and low power generation has been confirmed for all core peripheral assemblies. Therefore, the criterion in 10 CFR 50.46(b)(4) is satisfied.

For compliance with 10 CFR 50.46(b)(5), the licensee did not utilize the FSLOCA EM because it is not NRC-approved for the long-term analyses. Therefore, in regard to continued compliance with 10 CFR 50.46(b)(5), because the licensee did not use the FSLOCA for long-term analysis, the current licensing basis is maintained and is not affected by the implementation of the FSLOCA EM.

The NRC staff finds the proposed LBLOCA (Region II) analysis results for Turkey Point are in compliance with 10 CFR 50.46(b)(1) through (b)(4) and acceptable. For the LBLOCA long-term cooling, the NRC staff finds it acceptable that the licensee proposes to maintain the current licensing basis which was previously found to be acceptable by the NRC.

3.4 Compliance with the Applicable 1967 Draft GDCs

The NRC staff finds the proposed FSLOCA analysis using the FSLOCA EM is in compliance with the following applicable 1967 draft GDC:

- GDC 15, because the protection systems that sense the accident condition and initiate the ECCS are unaffected by the proposed change.

- GDC 37, because the ECCS design described in UFSAR Section 6.3, which is currently in conformance with this GDC, remains unaffected.

- GDC 41, because the ECCS and containment heat removal system performance capability is unaffected.

- GDC 42, because the ECCS and containment heat removal system capabilities for core cooling and containment heat removal are not impaired due to LOCAs.

- GDC 44, because the results of the revised FSLOCA analysis shown in Table 2 above confirm that the 10 CFR 50.46(b) parameters are within the specified limits and, therefore, the ECCS will limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the RCS pressure boundary, including a double-ended rupture of the largest pipe.

- GDC 49, since the containment design and its heat removal systems are unaffected, therefore, the containment can still accommodate its design leakage rate, pressure and temperature resulting from the largest M&E release following a LOCA.

- GDC 52, because the containment heat removal system is unaffected under accident conditions and will perform its safety function to prevent the containment design pressure from being exceeded, assuming a single active failure.

3.5 Limitations and Conditions

NRC SER for WCAP-16996-P-A (Reference 3), Table 22 provides a list of limitations and conditions required to be satisfied in order for the licensee to implement the NRC-approved FSLOCA EM. The licensee summarized each limitation and condition in the LAR, as given below. The NRC staff reviewed the licensee's summaries against the actual limitations and conditions documented in Table 22 of the NRC's SER above (Reference 18) and finds the summaries to be acceptable because the licensee appropriately described the specific requirement of each limitation and condition in its summary. The NRC staff evaluation of how each limitation and condition is satisfied for the Turkey Point LBLOCA analysis is provided below.

3.5.1 Limitation and Condition 1: FSLOCA EM Applicability with Regard to LOCA Transient Phases

Summary:

The FSLOCA EM is not approved to demonstrate compliance with 10 CFR 50.46 acceptance criterion (b)(5) related to the long-term cooling.

Evaluation

The licensee did not use the FSLOCA EM to show compliance with 10 CFR 50.46(b)(5) because it is not an NRC-approved methodology to analyze LBLOCA long-term cooling. Therefore, the NRC staff finds this limitation and condition is satisfied.

3.5.2 Limitation and Condition 2: FSLOCA EM Applicability with Regard to Type of PWR Plants

Summary:

The FSLOCA EM is approved for the analysis of Westinghouse-designed 3-loop and 4-loop PWRs with cold-side injection. Analyses should be executed consistent with the approved method, or any deviations from the approved method should be described and justified.

Evaluation

Turkey Point is a Westinghouse-designed 3-loop PWR with cold-side injection. The licensee used the NRC-approved FSLOCA EM which included changes and error corrections reported to the NRC, pursuant to 10 CFR 50.46, as described and justified in letters LTR-NRC-18-30 (Reference 12) and LTR-19-6 (Reference 13).

The NRC staff therefore finds this limitation and condition is satisfied.

3.5.3 Limitation and Condition 3: FS LOCA EM Applicability for Containment Pressure Modeling

Summary:

For Region II, the containment pressure calculation will be executed in a manner consistent with the approved methodology (i.e., the COCO or LOTIC2 model will be based on appropriate plant-specific design parameters and conditions, and engineered safety features which can reduce pressure are modeled). This includes utilizing a plant-specific initial containment temperature, and only taking credit for containment coatings which are qualified and outside of the break zone-of-influence.

Evaluation

The NRC staff reviewed the information provided in the Attachment 1 of the LAR (Reference 1) and finds that the LOCA containment pressure response analysis is performed using NRC-accepted Westinghouse COCO methodology (References 10 and 11). For this analysis, the licensee integrated the COCO code into the WCOBRA/TRAC-TF2 thermal-hydraulic code. The licensee stated that appropriate input parameters, including the effects of all the installed pressure reducing systems, i.e., assuming all trains of containment spray system and fan cooler in operation, were used for a calculation of a conservatively low containment back pressure as a boundary condition at the break. A plant-specific minimum initial temperature associated with normal full-power operating conditions was modeled, and conservatively, no coatings were credited on any of the containment structures.

The NRC staff finds this limitation and condition is satisfied because (a) the licensee used an acceptable plant-specific initial temperature for the containment pressure response, (b) the licensee used NRC-approved methodology for the LBLOCA (Region II) containment pressure calculation using inputs that minimized the containment back pressure as a boundary condition to the break for a conservative PCT calculation, and (c) conservatively, the licensee did not credit any coatings on any of the containment structures.

3.5.4 Limitation and Condition 4: Decay Heat Modeling in FSLOCA EM Applications

Summary:

The decay heat uncertainty multiplier will be **[[**
]] The analysis simulations for the FSLOCA EM will not be executed for **[[** **]]** following reactor trip unless the decay heat model is appropriately justified. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region I and Region II analysis results will be provided in the analysis submittal in units of sigma and absolute units.

Evaluation

The NRC staff reviewed Table 11 in Attachment 1 of the LAR (Reference 1) and confirmed that for both Region I and II analysis the licensee used sampled values of the decay heat uncertainty multiplier (DECY_HT) which are absolute in units of σ (sigma) and are **[[**

]] The analysis simulations were all executed for [[
]] following reactor trip.

The NRC staff finds that that the licensee appropriately modeled decay heat and correctly reported the resulting sampled values in units of σ and absolute units for the limiting cases. Therefore, the NRC staff finds that the licensee has satisfied the requirements for this limitation and condition.

3.5.5 Limitation and Condition 5: Fuel Burnup Limits in FSLOCA EM Applications

Summary:

The maximum assembly and rod length-average burnup is limited to **[[**
]] respectively.

Evaluation

In response to this limitation and condition, the licensee confirmed in the LAR (Reference 1), Attachment 1, that the maximum analyzed assembly and rod length-average burnups for Turkey Point FSLOCA analysis were less than or equal to **[[**
]] which are the limits contained in the FSLOCA EM.

Based on the above, the NRC staff finds that the licensee has satisfied the requirement for this limitation and condition.

3.5.6 Limitation and Condition 6: WCOBRA/TRAC-TF2 Interface with PAD 5.0

Summary:

The fuel performance data for analyses with the FSLOCA EM should be based on the PAD5 code (at present), which includes the effect of thermal conductivity degradation. The nominal fuel pellet average temperatures and rod internal pressures should be the maximum values, and the generation of all the PAD5 fuel performance data should adhere to the NRC-approved PAD5 methodology.

Evaluation

The NRC staff reviewed Attachment 1 of LAR (Reference 1) and confirmed that the licensee used PAD5 (Reference 8) in the analysis along with the FSLOCA EM. PAD5 is the latest version of the fuel performance code which explicitly models TCD and is benchmarked to high burnup data in WCAP-17642-P-A (Reference 8). The licensee stated that the FSLOCA EM considers the effects of fuel pellet TCD and other burnup-related effects by initializing to fuel rod performance data input generated by the PAD5 code. In the analysis, the fuel pellet average temperatures conservatively bounded the maximum values calculated in accordance with Section 7.5.1, “Maximum Fuel Temperatures,” of WCAP-17642-P-A (Reference 8). The analyzed rod internal pressures were calculated in accordance with Section 7.5.2, “Rod Internal Pressure,” of WCAP-17642-P-A (Reference 8).

The NRC staff finds that the Turkey Point analysis for FSLOCA satisfies the requirements of this limitation and condition because the licensee used PAD5 which is the latest version of

NRC-approved fuel performance code and explicitly includes the effect of TCD using conservative inputs.

3.5.7 Limitation and Condition 7: Interfacial Drag Uncertainty in FSLOCA EM Region I Analyses

Summary:

The YDRAG uncertainty parameter should be **[[**
]]

Evaluation

In response to this limitation and condition, the licensee confirmed in the LAR (Reference 1), Attachment 1, that the interfacial drag multiplier YDRAG was **[[**

]] established in the FSLOCA EM as described in WCAP-16996-P-A (Reference 3), Section 29.1.5. The lower value of the YDRAG reduces the two-phase mixture producing a lesser swell and therefore results in the calculation of a conservative rod heat-up. The NRC staff finds this limitation and condition is satisfied.

3.5.8 Limitation and Condition 8: Biased Uncertainty Contributors in Region I Analyses

Summary:

The **[[**

]]

Evaluation

The licensee stated that the **[**

] for the Turkey Point Region I analysis. The NRC staff finds that the limitation and condition is satisfied because in the Region I analysis, the biasing of (a) **[[** **]]** is consistent with the FSLOCA EM, Section 29.1.6, and (b) **[[** **]]** is consistent with the FSLOCA EM, Section 29.1.7.

3.5.9 Limitation and Condition 9: Effect of Bias in Applications for Region I

Summary:

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to confirm that the **[[**
]] for the plant design being analyzed. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Evaluation

Turkey Point is a 3-loop Westinghouse PWR. Therefore, this limitation and condition is not applicable because it only applies to PWRs that are not Westinghouse 3-loop plants.

3.5.10 Limitation and Condition 10: Boundary Between Region I and Region II Breaks

Summary:

For PWR designs which are not Westinghouse 3-loop PWRs, a sensitivity study will be executed to: 1) demonstrate that no unexplained behavior occurs in the predicted safety criteria across the region boundary, and 2) ensure that the **[[**

]] must cover the equivalent 2 to 4-inch break range using RCS-volume scaling relative to the demonstration plant. This sensitivity study should be executed once, and then referenced in all applications to that particular plant class.

Additionally, the minimum sampled break area for the analysis of Region II should be 1 ft^2.

Evaluation

The portion of the requirement of this limitation and condition that is applicable to PWRs that are not Westinghouse 3-loop plants does not apply to Turkey Point because it is a Westinghouse 3-loop plant.

Consistent with FSLOCA EM (Reference 3), the licensee confirmed that the Region II analysis was performed for a minimum sampled break area of 1 ft^2. The NRC staff finds that this limitation and condition and condition is satisfied.

3.5.11 Limitation and Condition 11: **[[** **]]** in FSLOCA Uncertainty Analyses for Region II and Documentation of Reanalysis Results for Region I and Region II

Summary:

There are various aspects of this Limitation and Condition, which are summarized below:

1. The **[[** **]]** the Region I and Region II analysis seeds, and the analysis inputs will be declared and documented prior to performing the Region I and Region II uncertainty analyses. The **[[** **]]** and the Region I and Region II analysis seeds will not be changed throughout the remainder of the analysis once they have been declared and documented.

2. If the analysis inputs are changed after they have been declared and documented, for the intended purpose of demonstrating compliance with the applicable acceptance criteria, then the changes and associated rationale for the changes will be provided in the analysis submittal. Additionally, the preliminary values for PCT, MLO, and CWO which caused the input changes will be provided. These preliminary values are not subject to Appendix B verification, and archival of the supporting information for these preliminary values is not required.

3. Plant operating ranges which are sampled within the uncertainty analysis will be provided in the analysis submittal for both regions.

Evaluation

In response to this limitation and condition, the licensee confirmed in the LAR, Attachment 1, that in the Turkey Point models for analysis with the FSLOCA EM the **[[** **]]** the Region I and Region II analysis seeds, and the analysis inputs were declared and documented prior to performing the Region I and Region II uncertainty analyses and were not changed after declared and documented. The licensee provided the Turkey Point plant operating ranges which were sampled within the uncertainty analysis in LAR, Attachment 1, Table 1. Based on this information, the NRC staff finds this limitation and condition is satisfied.

3.5.12 Limitation and Condition 12: Steam Generator Heat Removal During SBLOCAs

Summary:

The plant-specific dynamic pressure loss from the steam generator secondary-side to the main steam safety valves (MSSVs) must be adequately accounted for in analysis with the FSLOCA EM.

Evaluation

In response to this limitation and condition, the licensee confirmed in the LAR, Attachment 1, that the Turkey Point model used to perform the FSLOCA analysis includes a bounding plant-specific dynamic pressure loss from the SG secondary side to the MSSVs. The NRC staff finds this limitation and condition is satisfied.

3.5.13 Limitation and Condition 13: Upper Head Spray Nozzle Loss Coefficient

Summary:

In plant-specific models for analysis with the FSLOCA EM, specific modeling considerations for the upper head spray nozzles should be followed as required by the NRC-approved methodology.

In plant-specific models for analysis with the FSLOCA EM: 1) the **[[**

] and 2) the **[[** **]]**

Evaluation

In response to this limitation and condition, the licensee confirmed in the LAR, Attachment 1, that in the Turkey Point models for analysis with the FSLOCA EM, the **[**

]] and the **[[** **]]** The NRC staff finds this limitation and condition is satisfied.

3.5.14 Limitation and Condition 14: Correlation for Oxidation

Summary:

For analyses with the FSLOCA EM to demonstrate compliance against the current 10 CFR 50.46 oxidation criterion, the transient time-at-temperature will be converted to an equivalent cladding reacted (ECR) using either the Baker-Just or the Cathcart-Pawel correlation. In either case, the pre-transient corrosion will be summed with the LOCA transient oxidation. If the Cathcart-Pawel correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 13 percent limit. If the Baker-Just correlation is used to calculate the LOCA transient ECR, then the result shall be compared to a 17 percent limit.

Evaluation

In response to this limitation and condition, the licensee stated in the LAR, Attachment 1, that in the Turkey Point models for analysis with the FSLOCA EM, the Baker-Just correlation was used to convert the LOCA transient time-at-temperature into an ECR. The licensee met the 10 CFR 50.46(b)(2) MLO criterion of 17% after summing up the pre-existing corrosion with the resulting LOCA transient ECR. Therefore, the NRC staff finds this limitation and condition is satisfied.

3.5.15 Limitation and Condition 15: LOOP versus OPA Treatment in FSLOCA EM Uncertainty Analyses for Region II

Summary:

The Region II analysis will be executed twice: once assuming loss of offsite power (LOOP) and once assuming offsite power available (OPA). The results from both analysis executions should be shown to be in compliance with the 10 CFR 50.46 acceptance criteria.

The **[[ ]]**

Evaluation

The NRC staff review of LAR, Attachment 1 confirmed that the Region II analysis for Turkey Point was performed for LOOP and OPA cases. The licensee performed the statistical analysis for both cases in accordance with the FSLOCA EM. The results from both analyses are in compliance with the 10 CFR 50.46 criteria as evaluated in Section 3.3 above.

The licensee used a **[[**
]]

Based on the above evaluation, the NRC staff finds this limitation and condition is satisfied.

3.6 Technical Evaluation Summary

Based on the foregoing evaluation of the proposed change, the NRC staff conclusions are as follows:

- The 10 CFR 50.36(c)(5) requirement is satisfied because the licensee added the approved FSLOCA EM to the TS 6.9.1.7 COLR reference list as a provision for administrative controls.

- The deletion of the obsolete COLR references in TS Section 6.9.1.7 is acceptable because they will no longer be used.

- The guidance in NRC GL 1988-16 (Reference 4) continues to be implemented because the proposed change specifies an NRC-approved methodology for the determination of core operating limits.

- The FSLOCA EM is NRC-approved and satisfies the requirements of 10 CFR 50.46(a)(1)(i) and 10 CFR Part 50, Appendix K, Part II documentation requirements.

- The licensee appropriately applied the FSLOCA EM for FSLOCA analysis after making changes and correcting errors reported in 10 CFR 50.46 letters LTR-NRC-18-30 (Reference 12) and LTR-NRC-19-6 (Reference 13).

- The FSLOCA analysis results satisfy the requirements of 10 CFR 50.46(b)(1) through (b)(4).

- The FSLOCA EM is not used to confirm compliance with 10 CFR 50.46(b)(5) because it is not NRC-approved for LOCA long-term core cooling analysis.

- The current compliance with 10 CFR 50.46(b)(5) is not affected by the FSLOCA EM implementation, so the current licensing basis remains as previously approved by the NRC.

- Conformance with the applicable draft 1967 GDCs listed in Section 2.3 above remains unaffected.

- NRC IN 2011-21 (Reference 6) on TCD for fuel pellets is appropriately considered while confirming compliance with 10 CFR 50.46(b)(1).

- NRC IN 1998-29 (Reference 15) regarding predicted increase in fuel rod cladding oxidation is appropriately considered while confirming compliance with 10 CFR 50.46(b)(2) on the MLO.

- All limitations and conditions listed in the SER for WCAP-16996-P-A for the FSLOCA analysis are satisfied.

Based on the above conclusions, the NRC staff finds the proposed changes in TS 6.9.1.7 are acceptable, and the requirements in 10 CFR 50.46(b) continue to be met.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Florida State official was notified of the proposed issuance of the amendment on February 8, 2022. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has

determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission previously issued a proposed finding that the amendment involves no significant hazards consideration published in the *Federal Register* on September 7, 2021 (86 FR 50188), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Florida Power and Light letter to NRC, "License Amendment Request 273, Update Listing of Approved LOCA Methodologies to Adopt FULL SPECTRUMTM LOCA Methodology," April 15, 2021 (ADAMS Accession No. ML21105A848).

2. Florida Power and Light letter to NRC, "Response to Request for Additional Information Regarding License Amendment Request 273, Update Listing of Approved LOCA Methodologies to Adopt FULL SPECTRUM™ LOCA Methodology," November 19, 2021 (ADAMS Accession Nos. Proprietary ML21323A154 and Non-Proprietary ML21323A153).

3. Westinghouse letter to NRC, "Submittal of WCAP-16996-P-A/WCAP-16996-NP-A, Volumes I, II, III and Appendices, Revision 1, 'Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Metodology),'" October 2, 2017, (ADAMS Accession Nos. Proprietary ML17277A143, ML17277A144, ML17277A149, and ML17277A150, and Non-Proprietary ML17277A131, ML17277A132, ML17277A133, ML17277A134, and ML17277A135).

4. NRC Generic Letter (GL) 1988-16, "Removal of Cycle Specific Parameter Limits from Technical Specifications," October 4, 1988 (ADAMS Accession No. ML031130447).

5. Westinghouse letter and reports, to U.S. Nuclear Regulatory Commission (NRC), "Transmittal of Proprietary [and Non Proprietary] Information" WCAP 16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)," March 11, 2005 (ADAMS Accession No. ML050910157).

6. NRC Information Notice (IN) 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," December 13, 2011 (ADAMS Accession No. ML113430785).

7. Florida Power and Light letter to NRC, "Schedule for Re-Analysis of Turkey Point Licensing Basis Analyses Affected by PAD5 Implementation," March 27, 2018 (ADAMS Accession No. ML18086A154).

8. Westinghouse letter to NRC, "LTR-NRC-17-75 'Submittal of WCAP-17642-P-A / WCAP-17642-NP-A, Revision 1 'Westinghouse Performance Analysis and Design Model (PAD5),' (Proprietary / Non-Proprietary)," November 27, 2017 (ADAMS Package Accession No. ML17335A334).

9. Virginia Electric and Power Company letter to NRC, "Virginia Electric and Power Company North Anna Power Station Units 1 and 2 Proposed License Amendment Request Addition of Analytical Methodology to the Core Operating Limits Report for a Full Spectrum Loss of Coolant Accident (FSLOCA) Gamma Energy Redistribution Information," August 31, 2020 (ADAMS Accession No. ML20244A336).

10. Westinghouse, "Westinghouse Emergency Core Cooling System Evaluation Model – Summary," WCAP-8339, June 1974 (ADAMS Accession No. ML092430562).

11. Westinghouse, WCAP-8327, "Containment Pressure Analysis Code (COCO)," July 1974 (ADAMS Accession No. ML092460709, *Proprietary Version, Non-Publicly Available*).

12. Westinghouse letter to NRC, LTR-NRC-18-30, "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2017," July 18, 2018 (ADAMS Accession No. ML19288A174).

13. Westinghouse letter to NRC, LTR-NRC-19-6, "U.S. Nuclear Regulatory Commission 10 CFR 50.46 Annual Notification and Reporting for 2018," February 7, 2019 (ADAMS Accession Package No. ML19042A378).

14. NUREG-0800, Standard Review Plan, Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," March 2007 (ADAMS Accession No. ML070550016).

15. NRC IN 1998-29, "Predicted Increase in Fuel Rod Cladding Oxidation," August 3, 1998.

16. NRC Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," May 1989 (ADAMS Accession No. 003739584).

17. NRC Regulatory Guide 1.203, "Transient and Accident Analysis Methods," December 2005 (ADAMS Accession No. ML053500170).

18. NRC "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" (TAC NO. ME5244)," October 19, 2016 (ADAMS Accession No. ML16142A001).

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Date: May 24, 2022

SUBJECT: TURKEY POINT NUCLEAR GENERATING UNIT NOS. 3 AND 4 - ISSUANCE OF AMENDMENT NOS. 296 AND 289 REGARDING IMPLEMENTATION OF FULL SPECTRUM LOSS-OF-COOLANT ACCIDENT (FSLOCA) METHODOLOGY (EPID L-2021-LLA-0070) DATED MAY 24, 2022

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