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

October 29, 2020

Mr. Daniel G. Stoddard
Senior Vice President and Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 286 AND 269 TO REVISE TECHNICAL SPECIFICATIONS TO ALLOW USAGE OF A FULL SPECTRUM LOSS-OF-COOLANT-ACCIDENT (LOCA) METHODOLOGY (EPID L-2019-LLA-0236)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 286 and 269 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station (North Anna), Unit Nos. 1 and 2, respectively. These amendments are in response to your application dated October 30, 2019, as supplemented by letter dated August 31, 2020.

The amendments revise North Anna Technical Specifications to include Westinghouse Topical Report WCAP-16996-P-A, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," in the list of methodologies approved for reference in the Core Operating Limits Report (COLR) in TS 5.6.5.b. The amendments also removes obsolete COLR references that are no longer used to support North Anna core reloads.

Enclosure 3 to this letter contains Proprietary information. When separated from Enclosure 3, this document is DECONTROLLED.

The NRC staff has determined that the related safety evaluation contains proprietary information pursuant to Title 10 of the Code of Federal Regulations Section 2.390, “Public inspections, exemptions, requests for withholding.” The proprietary information is indicated by text enclosed with double brackets. The proprietary version of the safety evaluation is provided as Enclosure 3. Accordingly, the NRC staff has also prepared a non-proprietary version of the safety evaluation, which is provided in Enclosure 4.

The Commission’s monthly *Federal Register* notice will include the notice of issuance.

Sincerely,

/RA/

G. Edward Miller, Project Manager
Special Projects and Process Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 286 to NPF-4
2. Amendment No. 269 to NPF-7
3. Safety Evaluation (Proprietary)
4. Safety Evaluation (Non-Proprietary)

Cc w/o Enclosure 3: Listserv

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 286
Renewed License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated October 30, 2019, as supplemented by letter dated August 31, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-4, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A, as revised through Amendment No. 286, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to Renewed Facility
Operating License No. NPF-4
and Technical Specifications

Date of Issuance: October 29, 2020

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 269
Renewed License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated October 30, 2019, as supplemented by letter dated August 31, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-7, as indicated in the attachment to this license amendment, and is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 269, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to Renewed Facility
Operating License No. NPF-7
and Technical Specifications

Date of Issuance: October 29, 2020

ATTACHMENT TO

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

LICENSE AMENDMENT NO. 286

RENEWED FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

AND LICENSE AMENDMENT NO. 269

RENEWED FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the 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
NPF-4, page 3	NPF-4, page 3
NPF-7, page 3	NPF-7, page 3

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

Remove	Insert
5.6-3	5.6-3
5.6-4	5.6-4

(2) Pursuant to the Act and 10 CFR Part 70, VEPCO to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report;

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

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

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

VEPCO is authorized to operate the North Anna Power Station, Unit No. 1, at reactor core power levels not in excess of 2940 megawatts (thermal).

(2) Technical Specifications

Technical Specifications contained in Appendix A, as revised through Amendment No. 286 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, VEPCO to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations as 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>

VEPCO is authorized to operate the facility at steady state reactor core power levels not in excess of 2940 megawatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 269 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Additional Conditions</u>

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the insurance of the condition or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission:

a. If VEPCO plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the North Anna Power Station, the

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. VEP-FRD-42-A, "Reload Nuclear Design Methodology."

2. Plant-specific adaptation of WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," as approved by NRC Safety Evaluation Report dated February 29, 2012.

3. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

4. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code."

5. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY—REFERENCE CORE REPORT."

6. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology."

7. VEP-NE-1-A, "VEPCO Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications."

8. WCAP-8745-P-A, "Design Bases for Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Function."

9. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

10. BAW-10168P-A, "RSG LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Volume II only (SBLOCA models).

11. DOM-NAF-2-A, "Reactor Core Thermal-Hydraulics Using the VIPRE-D Computer Code," including Appendix C, "Qualification of the Westinghouse WRB-2M CHF Correlation in the Dominion VIPRE-D Computer Code," and Appendix D, "Qualification of the ABB-NV and WLOP CHF Correlations in the Dominion VIPRE-D Computer Code."

12. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO" (Westinghouse Proprietary).

13. WCAP-16996-P-A, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes (FULL SPECTRUM LOCA Methodology)," (Westinghouse Proprietary).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 286 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-4

AMENDMENT NO. 269 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-338 AND 50-339

Proprietary information has been redacted from this document pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations*.

Redacted information is identified by blank space enclosed within [[double brackets]].

1.0 INTRODUCTION

By letter dated October 30, 2019 (Reference 1) as supplemented by letter dated August 31, 2020 (Reference 2), Virginia Electric and Power Company (Dominion Energy Virginia, the licensee) submitted a license amendment request (LAR) for North Anna Power Station (North Anna), Unit Nos. 1 and 2.

The amendments would revise Technical Specification (TS) 5.6.5 to add Westinghouse Topical Report 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) to the list of methodologies approved for Reference in the Core Operating Limits Report (COLR) for North Anna. The added Reference identifies the analytical methods used to determine core operating limits for the large break loss of coolant accident (LBLOCA) event described in the North Anna Updated Final Safety Analysis Report (UFSAR), Section 15.4.1, "Loss of Reactor Coolant From Ruptured Pipes or From Cracks in Large Pipes Including Double Ended Rupture That Actuates the Emergency Core Cooling System (Large Break Loss-of-Coolant Accident)." The amendments also propose to remove obsolete COLR References that are no longer used to support North Anna core reloads. The Augus 31, 2020, supplement provided additional information that clarified the application, but did not expand the scope of the application as originally noticed nor change the NRC staff proposed no significant hazards consideration determination as published in the Federal Register on December 31, 2019 (84 FR 72385).

Enclosure 4

2.0 REGULATORY EVALUATION

The NRC staff considered the following regulations and guidance during its review of the proposed changes.

Regulations

The regulation in 10 CFR 50.36, “Technical Specifications,” requires that TSs include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; (5) administrative controls; (6) decommissioning; (7) initial notification; and (8) written reports.

The regulations in 10 CFR 50.36(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 fregulations 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.

Guidance Documents

The NRC staff used the following documents to provide additional guidance on acceptable approaches to demonstrate that the above regulatory requirements are met.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," 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 (Reference 4).

- NRC Regulatory Guide (RG) 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," dated May 1989. (Reference 5)

- NRC Regulatory Guide 1.203, "Transient and Accident Analysis Methods," dated December 2005. (Reference 6)

- NRC Generic Letter 88-16, "Removal of Cycle Specific Parameter Limits from Technical Specifications," dated October 4, 1988. (Reference 7)

- NRC Information Notice 97-09, "Main Steam Safety Valve (MSSV) Setpoints and Performance Issues Associated with Long MSSV Inlet Piping."

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the licensee's LAR to determine whether the proposed changes would continue to meet the regulations and guidance provided in Section 3.0 of this safety evaluation. The NRC staff reviewed the licensee's proposed changes to verify that all limitations and conditions in applicable NRC-approved methods are met, the licensee appropriately applied the LOCA Evaluation Model (EM) to North Anna, and the acceptance criteria of 10 CFR 50.46(b)(1) through (4) are satisfied.

3.1 Description of FSLOCA Methodology

As described in WCAP-16996-P-A, Revision 1, the purpose of the Full Spectrum LOCA (FSLOCA™) EM is to build on the ASTRUM EM, described in WCAP-16009-P-A (Reference 3), by extending the applicability of the WCOBRA/TRAC Code to include the treatment of small break LOCA (SBLOCA) and intermediate break LOCA (IBLOCA) scenarios. The term "Full Spectrum" specifies that the new EM is intended to resolve the full spectrum of LOCA scenarios that result from a postulated break in the cold leg of a pressurized water reactor (PWR). The break sizes considered in the Westinghouse FSLOCA™ methodology include any break size in which break flow is beyond the capacity of the normal charging pumps, up to and including a double ended guillotine rupture with a break flow area equal to 2 times the pipe area.

The licensee is currently using the ASTRUM EM methodology described in WCAP-16009-P-A (Reference 8) to perform its LBLOCA licensing analyses. The licensee proposes to use the WCAP-16996-P-A (Reference 2) methodology in order to fulfill a prior commitment to the NRC to update its licensing basis to account for thermal conductivity degradation (TCD). FSLOCA™ EM is an analysis methodology for LOCAs that was reviewed and approved by the NRC, and further discussion of the methodology and its application to LOCAs can be found in WCAP-16996-P-A (Reference 3). The FSLOCA™ methodology divides the break spectrum into two parts: Region I (small break LOCA), and Region II (large break LOCA). The licensee proposes to use this methodology only for the Region II analyses.

3.2 Analysis

The analyses for North Anna, Units 1 and 2 analyses were performed by the licensee in accordance with the NRC-approved methodology in WCAP-16996-P-A. The analyses were performed assuming both loss of outside power (LOOP) and offsite power available (OPA). The FSLOCA™ EM, as approved by the NRC, is designed to perform analyses for both Regions I and II. Nonetheless, while the FSLOCA™ methodology divides the break spectrum into the two regions, independent analyses are performed to determine results within each region. Therefore, given that the FSLOCA™ analyses for Region I and Region II are separable, and do not influence each other, NRC staff finds it acceptable that the licensee is only performing analyses for Region II.

The major plant parameter and analysis assumptions used in the North Anna Units 1 and 2 FSLOCA EM are provided in Tables 1 through 4 of Attachment 4 to the LAR (Reference 1). Table 5 in the Attachment to the Supplement dated August 31, 2020 (Reference 2) provides results, Table 6 of Attachment 4 to the LAR provides a sequence of events and Table 7 summarizes the Region II LOOP and OPA uncertainty values used and the Decay Heat Uncertainty Multiplier analysis.

The NRC staff review concluded that the input assumptions such as Core Parameters, Reactor Coolant System parameters and Containment Parameters and the uncertainty values used in the analysis were reasonable and acceptable based on consistency with the North Anna plant configuration and current licensing basis.

The following table summarizes the peak cladding temperature (PCT), maximum local oxidization (MLO), and core wide oxidization (CWO) results from the North Anna analyses. The limiting PCT result is 1862 °F for the Region II analysis, however, an error correction from the gamma energy redistribution multiplier is estimated to increase the Region II analysis PCT by 31°F (see Section 4.3 of this safety evaluation, Limitation and Condition 2 evaluation for further discussion) with a total analysis PCT result of 1893 °F.

Results	Region II Value (LOOP)	Region II Value (OPA)
95/95 PCT	1862°F +31°F = 1893°F	1857°F + 31°F = 1888°F
95/95 MLO	6.43%	6.85%
95/95 CWO	0.79%	0.63%

The staff reviewed the analysis and the submittals provided by the licensee and determined that the North Anna Units 1 and 2 Region II analysis was performed by the licensee in accordance with the NRC-approved methodology, and accordingly, that the proposed change maintains sufficient safety margins and the relevant criteria of 10 CFR 50.46(b) are satisfied.

3.3 TS Evaluation

The proposed change adds a new methodology, WCAP-16996-P-A, "Realistic LOCA Evaluation Methodology Applied to the Full Spectrum of Break Sizes" (Reference 3). WCAP-16996-P-A would become the new method for LBLOCA analysis and, after transition, would replace WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)" (Reference 8) after transition in a subsequent reload cycle. The NRC staff considers the proposed TS change to be acceptable and notes that it would continue to provide administrative controls consistent with 10 CFR 50.36(c)(5).

The proposed amendment will also allow the Region II part of WCAP-16996-P-A methodology to be utilized to support future core reloads at North Anna. This proposed change assures the core operating limits have been calculated in accordance with NRC approved methodologies.

The current WCAP-16009-P-A LBLOCA analysis method would be retained in order to allow both units at North Anna to transition to the new method (Reference 3) between cycles, rather than having to update the method for both units at the same time. The NRC staff considers the proposed changes to revise the list of methodologies to reflect those necessary for future cycle specific operating limits in TS 6.2.C to be acceptable because the change would continue to provide administrative controls consistent with 10 CFR 50.36(c)(5).

Currently, TS 5.6.5b includes approved analytical methods that are no longer used to support North Anna reload cores. The proposed revision will remove the legacy analytical methods listed in TS 5.6.5b (Reference 1) (Section 2.4). The NRC staff considers the proposed deletion of an unused method in the TS to be acceptable because the proposed change is administrative in nature, superceded by NRC-approved methods, and would continue to meet 10 CFR 50.36(c)(5).

3.4 Limitations and Conditions

The safety evaluation for WCAP-16996-P-A, Revision 1 (Reference 9) contains 15 limitations and conditions that must be met in order for a licensee to implement the NRC-approved FSLOCA™ EM.

A summary of each limitation and condition and how it has been met, as stated by the licensee in its application dated October 30, 2019, and the associated NRC staff findings are provided below. The NRC confirmed the statements provided by the licensee in the LAR (Reference 1) by review of the analysis documentation associated with the review. The NRC staff also performed an audit of the review and the details of the audit are provided in the audit plan and the audit report summary (Reference 10) and (Reference 11), respectively.

Limitation and Condition 1 – Applicability with Regard to LOCA Transient Phases

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

The analysis for North Anna, Units 1 and 2 with the FSLOCA™ EM is only being used to demonstrate compliance with 10 CFR 50.46 (b)(1) through (b)(4). Given that the licensee is not using the FSLOCA™ EM to demonstrate compliance with 10 CFR 50.46(b)(5), the NRC staff finds that the licensee has met the requirements for Limitation and Condition 1.

Limitation and Condition 2 – Applicability with Regard to Type of PWR Plants

Applicability of FSLOCA™ EM is defined in terms of PWR-type plants so that analysis is approved for the Westinghouse-designed 3-loop and 4-loop PWRs [Pressurized Water Reactors] with cold-side injection only. Plant-specific licensing actions referencing FSLOCA™ analyses should include a statement summarizing the extent to which the FSLOCA™ methods and modeling were followed, and justification for any departures.

North Anna Units 1 and 2 are Westinghouse-designed 3-loop PWRs with cold-side injection, so they are eligible to use the FSLOCA™ EM. The analysis for North Anna Units 1 and 2 utilizes the NRC-approved FSLOCA™ methodology, with the following exceptions.

Since the safety evaluation was issued on the FSLOCA methodology (WCAP-16996-P-A, Revision 1), several changes and corrections have been made to the FSLOCA™ EM. In a letter to the NRC (Reference 12) Westinghouse reported the impact of errors in the emergency core cooling system (ECCS) evaluation models used by Westinghouse Electric Company. A description of the error corrections to the Westinghouse FSLOCA™ EM is provided to Reference 9. The NRC staff reviewed the errors and the resolution as documented in Reference 9 and identified concluded that only the following applies to Region II analyses of FSLOCA™ EM:

> Conservation of Non-Condensable Gas: Westinghouse identified that there existed an imbalance in the non-condensable gas mass which could occur in the WCOBRA/TRAC-TF2 code because it does not have the capability to implement the vapor property functions for temperatures below 32°F under certain conditions.

In the LAR (Reference 1), the licensee stated that this error was corrected in the loop components for the WCOBRA/TRAC-TF2 code. The resolution of this issue represents a Non-Discretionary Change in the Evaluation Model as described in Section 4.1.2 of WCAP-13451, "Westinghouse Methodology for Implementation of 10 CFR 50.46." The error had minimal impact on LOCA transient calculations, leading to an estimated peak cladding temperature impact of 0°F.

The NRC staff finds that changes to the conservation of non-condensable gas are appropriate and acceptable.

After completion of the analysis for North Anna Units 1 and 2, Westinghouse subsequently discovered three errors were in the WCOBRA/TRAC-TF2 code. The first error was regarding the calculation of radiation heat transfer to liquid. The second error was regarding vapor temperature resetting, where the vapor temperature could incorrectly be reset to the saturation temperature for heat transfer calculations. The licensee in the LAR (Reference 1) stated that these two errors were found to have a negligible impact on analysis results.

The third error impacted the gamma energy redistribution multiplier on the hot rod and hot assembly power used for the North Anna, Units 1 and 2, analyses. This error resulted in up to about a 5 percent deficiency in the hot rod and hot assembly rod linear heat rates on a run-specific basis, depending on the as-sampled value for the multiplier uncertainty.

The licensee stated in its LAR supplement letter (Reference 2) that the error impacting the gamma energy redistribution multiplier had only a limited impact on the power modeled for a single assembly in the core, and the error correction had a negligible impact on the system thermal-hydraulic response during the postulated LOCA for the Region II analysis. The PCT impact from the error correction was found by the licensee to be different for the transient phases (i.e., blowdown versus reflood) based on parametric PWR sensitivity studies. The correction of the error was estimated to increase the Region II analysis PCT by 31 °F, leading to a final PCT analysis result of 1893 °F for Region II. The NRC staff reviewed the Region II analysis and additional information that was provided by the licensee and confirmed that the analysis results, with inclusion of the error correction, continue to demonstrate compliance with the 10 CFR 50.46(b)(1) acceptance criteria. Therefore, the NRC staff finds that the licensee has met the requirements for Limitation and Condition Number 2.

Limitation and Condition 3 – Applicability for Containment Pressure Modeling

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.

The NRC staff reviewed the information provided in the Attachment 4 of the LAR (Reference 1) and confirmed that the containment pressure calculation for the North Anna, Units 1 and 2 analysis was performed consistent with the NRC-approved methodology. [The containment pressure is calculated for each LOCA transient in the analysis using the COCO code. The COCO containment code is integrated into the WC0BRA/TRAC-TF2 thermal-hydraulic code.] The licensee stated that appropriate design parameters and conditions were modeled, which can reduce the containment pressure. A minimum initial temperature associated with normal full-power operating conditions was modeled, and only containment coatings which are qualified and outside of the break zone-of-influence were credited.

The NRC staff finds that the licensee used NRC-approved methodology (i.e., the COCO or LOTIC2 models) for the Region II containment pressure calculation with the appropriated design parameters and conditions. Therefore, the NRC staff finds that the licensee has met the requirements for Limitation and Condition 3.

Limitation and Condition 4 – Decay Heat Modeling

The decay heat uncertainty multiplier **[[**
]]. The analysis simulations for the FSLOCA™ EM cannot be applied for transient time longer than 10,000 seconds following reactor trip unless the decay heat model is shown to be acceptable for the analyzed core conditions. The sampled values of the decay heat uncertainty multiplier as applied for the limited runs in Region I and Region II analysis results will be provided in the license amendment submittal in units of σ (sigma) and absolute units.

The licensee in the LAR (Reference 1) stated that the decay heat uncertainty multiplier **[[** **]]** for the North Anna, Units 1 and 2 Region II analysis. The analysis simulations were all executed for no longer than 10,000 seconds following reactor trip. The sampled values of the decay heat uncertainty multiplier for the cases which produced the Region II analysis results have been provided in units of σ and approximate absolute units in Table 7 of Attachment 4 to the LAR (Reference 1).

The NRC staff confirmed that that the licensee appropriately modeled decay heat per the limitation and condition 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 met the requirements for Limitation and Condition 4.

Limitation and Condition 5 – Fuel Burnup Limits

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

The NRC staff reviewed the analysis that was performed by the licensee and confirmed that maximum assembly and rod length-average burnup is less than or equal to **[[**
]] respectively, for North Anna, Units 1 and 2.

Based on the above, the NRC staff finds that the licensee has met the requirements for Limitation and Condition 5.

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

The fuel performance data for analyses with the FSLOCA™ EM should be based on the PAD5 code (Reference 14), 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.

PAD5 fuel performance data was utilized in the North Anna Units 1 and 2 analysis with the FSLOCA™ EM. The analyzed fuel pellet average temperatures bound the maximum values calculated in accordance with Section 7.5.1 of (Reference 14) , and the analyzed rod internal pressures were calculated in accordance with Section 7.5.2 of (Reference 14).

Given that the licensee used the latest NRC-approved fuel performance code (i.e., PAD5) and used appropriate conservative inputs, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 6.

Limitation and Condition 7 – Interfaciale Drag Uncertainty in Region I Analyses

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

]] given in Table 29.2.3-1 of (Reference 3).

The licensee proposed to use the FSLOCA™ methodology only for Region II analyses. Accordingly, this Limitation and Condition is not applicable, and the licensee did not need to perform a Region I uncertainty analysis in this application of the FSLOCA EM.

Limitation and Condition 8 – Biased Uncertainty Contributors in Region I Analyses

The **[[**

]]

The licensee proposed to use the FSLOCA™ methodology only for Region II analyses. Accordingly, this Limitation and Condition is not applicable, and the licensee did not need to perform a Region I uncertainty analysis in this application of the FSLOCA EM.

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

For PWR designs which are not Westinghouse 3-loop PWRs, a confirmatory analysis will be performed to assess the effect associated with the **[[**

]] for the plant design being analyzed..

North Anna, Units 1 and 2 are Westinghouse 3-loop PWRs, and the licensee proposed to use the FSLOCA™ methodology only for Region II analyses. Therefore, this Limitation and Condition does not apply to the licensee's LAR.

Limitation and Condition 10 – Boundary Between Region I and Region II Breaks

For PWR designs which are not Westinghouse 3-loop PWRs, a confirmatory evaluation will be performed to demonstrate that the applied break size boundary between Region I and Region II analyses serves the intended goal of **[[**

]].. Additionally, the minimum sampled break area for the analysis of Region II should be one square foot ($1ft^2$).

North Anna, Units 1 and 2 are Westinghouse 3-loop PWRs, and the licensee proposed to use the FSLOCA methodology only for Region II analyses. Therefore, the first part of this Limitation and Condition does not apply to the licensee's LAR.

Given that the minimum sampled break area for the North Anna Units 1 and 2 Region II analysis is $1ft^2$, the NRC staff finds that the licensee has met the applicable requirements of Limitation and Condition 10.

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

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

1) The **[[**

]] to be usef for the Region I and Region II uncertainty analyses will be declared and documented prior to performing the uncertainty analyses. The **[[** **]]** will not be adjusted as a result of the outcome.

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 10 CFR 50, 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.

This Limitation and Condition was met for the North Anna, Units 1 and 2 analyses as follows:

1) The **[[** **]]** the Region II analysis seeds, and the analysis inputs were declared and documented prior to performing the Region II uncertainty analyses. The **[[** **]]** and the Region II analyses seeds were not changed once they were declared and documented.

2) The analysis inputs were not changed once they were declared and documented.

3) The plant operating ranges which were sampled within the uncertainty analyses are provided for North Anna Units 1 and 2, in Table 1 of Attachment 4 of the LAR (Reference 1).

The NRC staff's review of the analysis and the information the licensee provided confirmed that the analysis inputs were not changed once they were declared and documented. Given that the licensee has declared and documented the appropriate inputs and did not change these values once declared and documented, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 11.

Limitation and Condition 12 – Steam Generator Heat Removal During SBLOCAs

in plant-specific applications, a check will be performed to confirm the effects associated with dynamic pressure losses from the steam generator secondary-side to the main steam safety valves areproperly considered and adequately accounted for in analysis with the FSLOCA^{TM} EM consistent with NRC Information Notice 97-09, "Main Steam Safety Valve (MSSV) Setpoints and Performance Issues Associated with Long MSSV Inlet Piping."

The licensee has stated in the LAR (Reference 1), that this Limitation and Condition applies to Region I (small break) transients. Region I calculations were not performed for North Anna Units 1 and 2; as such, the first stage main steam safety valve setpoint was used as a representative basis for the main steam safety valve setpoint calculation for the Region II analysis. Furthermore, Region II transients do not result in secondary side pressurization such that the Main Steam Safety Valve (MSSV) setpoint pressures would be reached.

Based on the above, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 12.

Limitation and Condition 13 – Upper Head Spray Nozzle Loss Coefficient

In plant-specific applications of the FSLOCA EM, 1) the **[[**

]] in the PWR model used to perform

the design-basis LOCA transient calculations, to capture the proper core two-phase level response should the core uncover. Additionally, the **[[** **]]** in such calculations.

The **[[** **]]** in the analyses for North Anna Units 1 and 2. The **[[** **]]** in the analyses.

Based on the above, the NRC staff finds that the licensee has met the requirements of Limitation and Condition 13.

Limitation and Condition 14 – Correlation for Oxidation

For analyses with 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 the 17 percent limit.

The licensee has stated in Attachment 4 (Page 7 of 33) of the LAR (Reference 1) that for the North Anna Units 1 and 2 analysis, the Baker-Just correlation was used to convert the LOCA transient time-at-temperature to an ECR. The resulting LOCA transient ECR was then summed with the pre-existing corrosion for comparison against the 10 CPR 50.46 local oxidation acceptance criterion of 17 percent.

The NRC staff finds that by using the Baker-Just correlation, converting to an ECR, and accounting for pre-existing corrosion, the licensee has met the requirements of Limitation and Condition 14.

Limitation and Condition 15 – LOOP versus OPA Treatment in Uncertainty Analyses for Region II

The Region II analysis will be executed twice; once assuming LOOP and once assuming OPA. The results from both analysis executions should be shown to be in compliance with the 10 CFR 50.46 acceptance criteria. The statistical analysis must adhere to the Limitation and Condition as specified in the NRC-approved methodology for the FSLOCA™ EM. For wach set, the calculated statistical results at the 95/95 probability confidence level should be demonstrated to comply with regulatory limits for PCT, MLO, and CWO. Specifically, the **[[** **]]**

> The Region II uncertainty analysis for North Anna Units 1 and 2 was performed by the licensee twice; once assuming a LOOP and once assuming OPA. The results from both analyses that were performed are in compliance with the current 10 CFR 50.46 acceptance criteria. The licensee stated in the LAR (Reference 1) that the statistical analysis adhered to the Limitations and Conditions as specified in the NRC-approved methodology for the FSLOCA™ EM. The results for this limiting condition analysis are in Table 7 of the LAR (Reference 1).

Given that the licensee has performed the Region II analysis for both LOOP and OPA and that the results from both are in compliance with the acceptance criteria in 10 CFR 50.46(b)(1) through (b)(4), the NRC staff finds that the licensee has met the requirements of Limitation and Condition 15.

3.5 Compliance with 10 CFR 50.46

The licensee presented the results for PCT, MLO, and CWO in Table 5 of the Attachment to the Supplement dated August 31, 2020 (Reference 2) for North Anna, Units 1 and 2.

To demonstrate compliance with 10 CFR 50.46(b)(1) through (b)(4), the following criteria must be met:

1. The calculated maximum fuel element cladding temperature must not exceed 2200° F.
2. The calculated total oxidation of the cladding must nowhere exceed 0.17 times the total cladding thickness before oxidation.
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam must 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. Calculated changes in core geometry must be such that the core remains amenable to cooling.

Each of the above four 10 CFR 50.46(b) criteria is discussed below.

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

Peak Cladding Temperature

The requirement of 10 CFR 50.46 (b)(1) states, "The calculated maximum fuel element cladding temperature shall not exceed 2200 °F." The licensee stated that the analysis for PCT corresponds to a bounding estimate of the 95th percentile PCT at the 95-percent confidence level and given that the resulting PCT is less than 2200 °F, the analyses with the FSLOCA EM confirm that 10 CFR 50.46 acceptance criterion (b)(1) is satisfied The licensee documented its results in Table 5 of Attachment 4 the LAR (Reference 1) for North Anna, Units 1 and 2. Given that the maximum calculated PCT is below the 2200 °F PCT limit, the NRC staff finds that the acceptance criterion of 10 CFR 50.46(b)(1) is met..

Maximum Cladding Oxidation

10 CFR 50.46(b)(2) states, in relevant part, that "[t]he calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

The licensee stated that the analysis for MLO corresponds to a bounding estimate of the 95th percentile MLO at the 95-percent confidence level. Since the resulting MLO is less than 17 percent after converting the time-at-temperature to an ECR using the Baker-Just correlation and adding the pre-transient corrosion, the analysis confirms that 10 CFR 50.46 acceptance criterion (b)(2) is satisfied. The licensee presented the results in Table 5 of the Attachment to the Supplement dated August 31, 2020 (Reference 2) for

North Anna, Units 1 and 2. Given that the resulting MLO is below the 17 percent limit, the NRC staff finds that the acceptance criterion of 10 CFR 50.46(b)(2) is met.

Maximum Hydrogen Generation

The requirement of 10 CFR 50.46(b)(3) states, "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."

The licensee stated that the analysis for CWO corresponds to a bounding estimate of the 95th percentile CWO at the 95-percent confidence level. The analysis confirms that the resulting CWO is less than 1 percent and that the 10 CFR 50.46 acceptance criterion (b)(3) is satisfied. The licensee presented the results in Table 5 of the Attachment to the LAR Supplement dated August 31, 2020. Given that the resulting CWO is below the 1 percent limit,, the NRC staff finds that the acceptance criterion of 10 CFR 50.46(b)(3) is met.

Coolable Geometry

The requirement of 10 CFR 50.46(b)(4) states, "Calculated changes in core geometry shall be such that the core remains amenable to cooling." The licensee stated that this criterion is met by demonstrating compliance with criteria 10 CFR 50.46(b)(1), (b)(2), and (b)(3), and by ensuring that fuel assembly grid deformation due to combined LOCA and seismic loads is specifically addressed.

Section 32.1 of the NRC-approved FSLOCA^{TM} EM documents that the effects of LOCA and seismic loads on the core geometry do not need to be considered unless fuel assembly grid deformation extends to inboard assemblies beyond the core periphery (i.e., deformation in a fuel assembly with no sides adjacent to the core baffle plates). The licensee stated that inboard grid deformation due to the combined LOCA and seismic loads was calculated to not occur for North Anna, Units 1 and 2.

Given that the criteria in 10 CFR 50.46(b)(1), (b)(2), and (b)(3) are met and the fuel assembly grid deformation due to the combined LOCA and seismic loads is specifically addressed, the NRC staff finds the acceptance criterion of 10 CFR 50.46(b)(4) is met.

4.0 TECHNICAL CONCLUSION

The licensee proposed to modify TS 5.6.5.b to replace the existing NRC-approved LOCA methodology (Reference 8) with the NRC-approved FSLOCA^{TM} EM (Reference 3). The NRC staff concludes that the proposed change is acceptable because the new methodology is an NRC-approved method. The NRC staff review has determined that the licensee appropriately applied the FSLOCA^{TM} EM to North Anna, Units 1 and 2; and finds that the resulting analysis meets 10 CFR 50.46(b)(1) through (4) requirements. In addition, NRC staff finds that the removal of selected obsolete COLR References that are no longer used to support North Anna reload cores.are acceptable. The NRC staff also finds the proposed changes continue to meet 10 CFR 50.36(c)(5) by providing provisions necessary to assure operation of the facility in a safe manner.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified an official from the Virginia Division of Radiological Health of the proposed issuance of the amendments on September 24, 2020. The Virginia official confirmed that the Commonwealth had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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 has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on December 31, 2019 (84 FR 72388). 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.

7.0 CONCLUSION

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

8.0 REFERENCES

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3 Westinghouse Electric Company (WEC), letter to U. S. Nuclear Regulatory Commission (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 Pkg No. ML17277A130).

4 U. S. Nuclear Regulatory Commission (NRC), "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).

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9 U. S. Nuclear Regulatory Commission (NRC) letter to J. Gresham, Westinghouse Electric Company (WEC), "Revised Final Safety Evaluation for Westinghouse Electric Company Topical Report WECAP-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 Pkg Accession No. ML17207A124).

10 U. S. Nuclear Regulatory Commission (NRC) Letter to D. G. Stoddard, , "North Anna Power Station, Units 1 and 2 and Surry Power Station Units 1 and 2 - Regulatory Audit Report Regarding License Amendment Request for Small Brak Loss-of-Coolant Accident Analysis Methodology," February 13, 2020 (ADAMS Accession No. ML20034F330).

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WCAP-16996-P-A, Revision 1 to the Core Operating Limits Report," April 17, 2020 (ADAMS Accession No. ML20099F873).

12 Westinghouse Elecric Company (WEC), Letter to U.S. Nuclear Regulatory Commission (NRC), "Information to Satisfy the FULL SPECTRUM LOCA (FSLOCA) Evaluation Methodology Plant Type Limitations and Conditions for 4-loop Westinghouse Pressurized Water Reactos (PWRs) (Proprietary / Non-Proprietary)," LTR-NRC-18-50, July 13, 2018 (ADAMS Accession No. ML18198A039).

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Principal Contributors: F. Forsaty, NRR

Date: October 29, 2020

SUBJECT: NORTH ANNA POWER STATION, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 286 AND 269 TO REVISE TECHNICAL SPECIFICATIONS TO ALLOW USAGE OF A FULL SPECTRUM LOSS-OF-COOLANT-ACCIDENT (LOCA) METHODOLOGY (EPID L-2019-LLA-0236) DATED OCTOBER 29, 2020

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