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

August 4, 2020

Mr. Tom Simril
Site Vice President
Catawba Nuclear Station, Units 1 and 2
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 306 AND 302 TO REVISE TECHNICAL SPECIFICATION 3.4.3, "REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS" (EPID NO. L-2019-LLA-0141)

Dear Mr. Simril:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 306 to Renewed Facility Operating License No. NPF-35 and Amendment No. 302 to Renewed Facility Operating License No. NPF-52 for the Catawba Nuclear Station (Catawba), Units 1 and 2, respectively. The amendments are in response to your application dated July 2, 2019, as supplemented by letter dated February 13, 2020.

The amendments revise Technical Specification (TS) 3.4.3, "RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits," to reflect an update to the P/T limit curves in Figures 3.4.3-1 RCS Heatup Limitations (UNIT 1 ONLY) and 3.4.3-2 RCS Cooldown Limitations (UNIT 1 ONLY) for Catawba, Unit 1. The proposed change will also reflect that the revised Catawba, Unit 1 P/T limit curves in TS 3.4.3 are applicable until 42.7 effective full power years (EFPY). Although the TSs are common to both Catawba, Unit 1 and Unit 2, the proposed changes are only applicable to Catawba, Unit 1. The applicable TS 3.4.3-1 RCS Heatup Limitations (UNIT 2 ONLY) and TS 3.4.3-2 RCS Cooldown Limitations (UNIT 2 ONLY) for Unit 2 remain unchanged.

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

If you have any questions, please contact me at (301) 415-1438 or Karen.Cotton@nrc.gov.

Sincerely,

Karen R. Cotton Digitally signed by Karen R. Cotton
Date: 2020.08.04 10:54:58 -04'00'

Karen Cotton-Gross, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 306 to NPF-35
2. Amendment No. 302 to NPF-52
3. Safety Evaluation

cc: Listserv

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2– ISSUANCE OF AMENDMENT NOS. 306 AND 302 TO REVISE TECHNICAL SPECIFICATION 3.4.3, "RCS [REACTOR COOLANT SYSTEM] PRESSURE AND TEMPURATURE (P/T) LIMITS" (EPID NO. L-2019-LLA-0141) DATED AUGUST 4, 2020

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OFFICE	NRR/DORL/LPL2-1/PM	NRR/DORL/LPL2-1/LA	NRR/DNRL/NVIB/BC	NRR/DSS/SNSB/BC
NAME	KCotton-Gross	KGoldstein	HGonzalez*	SKrepel*
DATE	07/22/2020	07/24/2020 & 07/31/2020	07/24/2020	02/26/2020
OFFICE	NRR/DSS/SFNB/BC	NRR/DSS/STSB/BC	OGC / NLO	NRR/DORL/LPL2-1/BC
NAME	RLukes+	VCusumano*	MWoods+	MMarkley
DATE	6/26/2020	07/10/2020	7/30/2020	8/4/2020
OFFICE	NRR/DORL/LPL2-1/PM			
NAME	KCotton-Gross			
DATE	8/4/2020			

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DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 306
Renewed License No. NPF-35

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. NPF-35, filed by Duke Energy Carolinas, LLC (licensee), dated July 2, 2019, as supplemented by letter dated February 13, 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-35 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 306 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC 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 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley
Digitally signed by Michael T. Markley
Date: 2020.08.04 12:34:19 -04'00'

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

Attachment:
Changes to Renewed License No. NPF-35
and Technical Specifications

Date of Issuance: August 4, 2020

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

DUKE ENERGY CAROLINAS, LLC

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 302
Renewed License No. NPF-52

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

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. NPF-52, filed by Duke Energy Carolinas, LLC (the licensee), dated July 2, 2019, as supplemented by letter dated February 13, 2020, complies with the standards and requirements of the Atomic 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 302, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC 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 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley Digitally signed by Michael T. Markley
Date: 2020.08.04 12:34:56 -04'00'

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

Attachment:
Changes to Renewed License No. NPF-52
and Technical Specifications

Date of Issuance: August 4, 2020

ATTACHMENT

AMENDMENT NO. 306 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-35

CATAWBA NUCLEAR STATION, UNIT 1

DOCKET NO. 50-413

AMENDMENT NO. 302 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-52

CATAWBA NUCLEAR STATION, UNIT 2

DOCKET NO. 50-414

Operating Licenses

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-35, page 4	NPF-35, page 4
NPF-52, page 4	NPF-52, page 4

Appendix A, "Technical Specifications," of Renewed Facility Operating Licenses

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

Remove	Insert
3.4.3-3	3.4.3-3
3.4.3-5	3.4.3-5

(2) TECHNICAL SPECIFICATIONS

The Technical Specifications contained in Appendix A, as revised through Amendment No. 306 which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than December 6, 2024, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(4) Antitrust Conditions

Duke Energy Carolinas, LLC shall comply with the antitrust conditions delineated in Appendix C to this renewed operating license.

(5) Fire Protection Program

Duke Energy Carolinas, LLC shall implement and maintain in effect all provisions of the approved fire protection program that complies with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated September 25, 2013; as supplemented by letters dated January 13, 2015; January 28, 2015; February 27, 2015; March 30, 2015; April 28, 2015; July 15, 2015; August 14, 2015; September 3, 2015; December 11, 2015; January 7, 2016; March 23, 2016; June 15, 2016; August 2, 2016; September 7, 2016; and January 26, 2017, as approved in the SE dated February 8, 2017. 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.

(2) TECHNICAL SPECIFICATIONS

The Technical Specifications contained in Appendix A, as revised through Amendment No. 302, which are attached hereto, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Technical Specifications.

(3) Updated Final Safety Analysis Report

The Updated Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on December 16, 2002, describes certain future activities to be completed before the period of extended operation. Duke shall complete these activities no later than December 6, 2024, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement as revised on December 16, 2002, described above, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following issuance of this renewed operating license. Until that update is complete, Duke may make changes to the programs described in such supplement without prior Commission approval, provided that Duke evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section

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MATERIALS PROPERTY BASIS
Limiting Material: Lower Shell Forging 04,
Intermediate Shell Forging 05

Limiting ART at 42.7 EFPY: 1/4-T 47°F
3/4-T 34°F

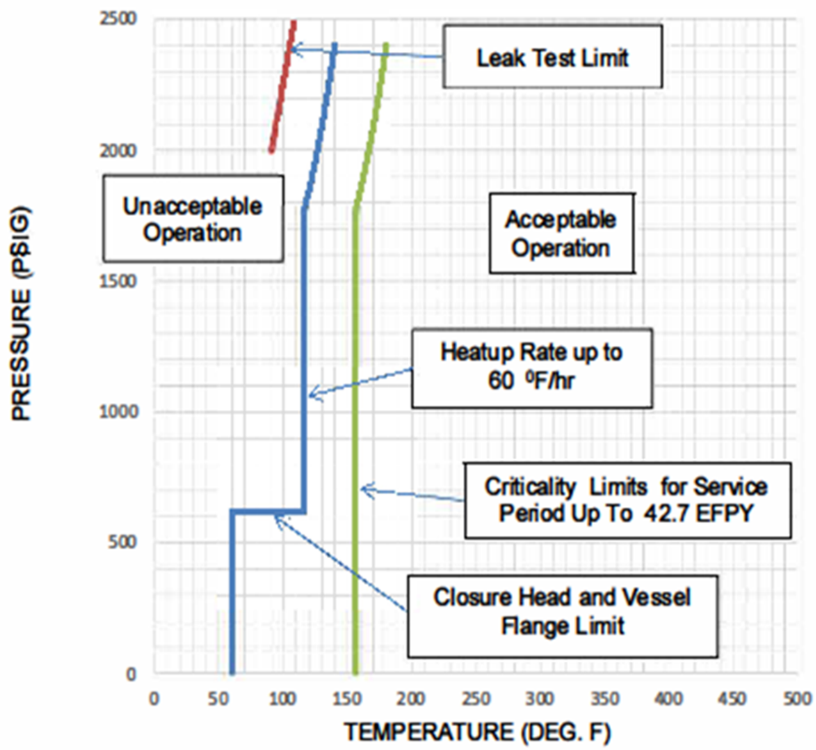

Figure 3.4.3-1
(UNIT 1 ONLY)
RCS Heatup Limitations (Without Margins for Instrument Errors)

MATERIALS PROPERTY BASIS
Limiting Material: Lower Shell Forging 04,
Intermediate Shell Forging 05

Limiting ART at 42.7 EFPY: 1/4-T 47°F
3/4-T 34°F

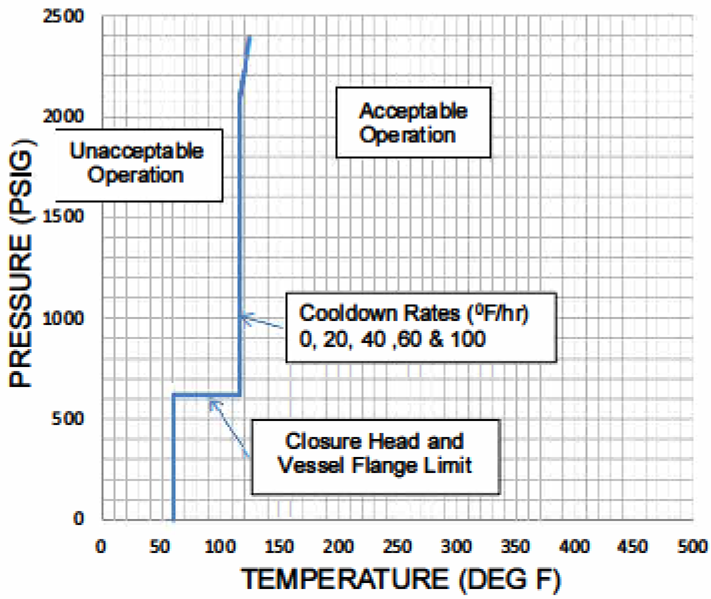

Figure 3.4.3-2
(UNIT 1 ONLY)
RCS Cooldown Limitations (Without Margins for Instrument Errors)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 306 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-35

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNIT 1

DOCKET NO. 50-413

AMENDMENT No. 302 TO RENEWED FACILITY OPERATING LICENSE No. NPF-52

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNIT 2

DOCKET NO. 50-414

1.0 INTRODUCTION

By letter to the U. S. Nuclear Regulatory Commission (NRC, Commission) dated July 2, 2019 (Reference 1), as supplemented by letter dated February 13, 2020 (Reference 2), Duke Energy Carolinas, LLC, (DEC or the licensee) requested changes to the Technical Specifications (TSs) for Renewed Facility Operating License No. NPF-35 for Catawba Nuclear Station, Unit 1 (henceforth Catawba Unit 1) and Renewed Facility Operating License No. NPF-52 for Catawba Nuclear Station, Unit 2. Although the TSs are common to both Catawba, Unit 1 and Unit 2, the proposed changes are only applicable to Catawba Unit 1. The applicable TS 3.4.3-1 RCS Heatup Limitations (UNIT 2 ONLY) and TS 3.4.3-2 RCS Cooldown Limitations (UNIT 2 ONLY) for Unit 2 remain unchanged. If approved, the proposed changes would permit the licensee to revise the pressure-temperature (P/T) limit curves for Catawba Unit 1 to be effective for licensed power operations up to and inclusive of 42.7 effective full power years (EPFY). The license amendment request (LAR) submittal included two Westinghouse Electric Company (WEC) non-proprietary WCAP reports as supporting information for the request: (1) WCAP-15448, Revision 1 (Reference 3), and (2) WCAP-17669-NP, Revision 0 (Reference 4).

The supplement dated February 13, 2020 provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination published in the *Federal Register* (FR) on November 5, 2019 (84 FR 59657).

2.0 REGULATORY EVALUATION

2.1 System Descriptions and Requirements

In Section 2.1, "System Design Description," of the LAR dated July 2, 2019, the licensee provided the following description:

> All components of the CNS [Catawba Nuclear Station] Reactor Coolant System (RCS) are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients and reactor trips. CNS is required to limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation.
>
> The CNS TS contain P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing and data for the maximum rate of change of reactor coolant temperature. Each P/T limit curve defines an acceptable region for normal operation. The typical use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.
>
> Operating limits are established that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB).
>
> CNS Updated Final Safety Analysis Report [UFSAR] Section 5.3.2 provides additional details regarding the methodology used to develop the P/T limit curves that are contained in the CNS TS.

2.2 Licensee's Proposed Changes

Catawba TS 3.4.3 addresses RCS Pressure and Temperature (P/T) Limits. Limiting condition for operation (LCO) 3.4.3 states, in part, that:

> RCS pressure and RCS temperature shall be limited during RCS heatup and cooldown, criticality, and inservice leak and hydrostatic testing in accordance with:
> a. A maximum heatup rate as specified in Figure 3.4.3-1;
> b. A maximum cooldown rate as specified in Figure 3.4.3-2;

Figures 3.4.3-1 and 3.4.3-2 are two unit-specific figures. In its LAR dated July 2, 2019, the licensee stated that the proposed changes are necessary because the existing Catawba, Unit 1 P/T limit curves in TS 3.4.3 are only applicable up to 30.7 EFPY, which Catawba, Unit 1 is expected to reach during operating cycle 26 (projected for early 2021), and that a new set of P/T limit curves with a longer term of applicability is required.

TS 3.4.3, Figure 3.4.3-1, "(UNIT 1 ONLY) RCS Heatup Limitations," proposed change is as follows:

- The existing RCS heatup limitations curve is superseded entirely by a new curve applicable up to 42.7 EFPY.

- The words “Upper Shell Forging 06, Intermediate Shell Forging 05, and Bottom Head Ring 03” for the “Limiting Material” are revised to state “Lower Shell Forging 04, Intermediate Shell Forging 05”.
- The words “Limiting ART [adjusted reference temperature] at 30.7 EFPY” are revised to state “Limiting ART at 42.7 EFPY”.
- The “1/4-T” value of “42°F” is revised to state “47°F”.
- The “3/4-T” value of “31°F” is revised to state “34°F”.

TS 3.4.3, Figure 3.4.3-2, “(UNIT 1 ONLY) RCS Cooldown Limitations,” proposed change is as follows:

- The existing RCS cooldown limitations curve is superseded entirely by a new curve applicable up to 42.7 EFPY.
- The words “Upper Shell Forging 06, Intermediate Shell Forging 05, and Bottom Head Ring 03” for the “Limiting Material” are revised to state “Lower Shell Forging 04, Intermediate Shell Forging 05”.
- The words “Limiting ART at 30.7 EFPY” are revised to state “Limiting ART at 42.7 EFPY”.
- The “1/4-T” value of “42°F” is revised to state “47°F”.
- The “3/4-T” value of “31°F” is revised to state “34°F”.

2.3 Regulatory Requirements and Guidance Applicable to P/T Limits

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36(c)(2), "Limiting conditions for operation," states that "[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclearreactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

The regulation in 10 CFR 50.36(c)(2)(ii)(B) states that a TS limiting conditions for a operation (LCO) must be established for:

> <u>Criterion 2</u>. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The regulation in 10 CFR 50.36(c)(2)(ii)(C) states that a TS LCO must be established for a:

> <u>Criterion 3</u>. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50 (Reference 5) requires, in part, that the P/T limits for an operating light water nuclear power reactor be at least as conservative as the limits obtained by following the methods of analysis and margins of safety of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” Division 1

(Reference 6). Table 1 in 10 CFR Part 50, Appendix G, establishes the specific minimum temperature requirements that must be incorporated into the calculations of P/T limit curves. The regulation in 10 CFR Part 50, Appendix G, Section IV.A.2.a (Reference 5) also states that the P/T limits and RCS minimum temperature requirement criteria for operating reactors are "defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical." For components located in the beltline of the Reactor Pressure Vessel (RPV), 10 CFR Part 50, Appendix G, Section IV.A requires the values of ART used in the development of the P/T limits to account for the effects of neutron irradiation, including incorporation of the results of the RPV materials surveillance program that is required and implemented in accordance with the requirements in 10 CFR Part 50, Appendix H, "Reactor Vessel Materials Surveillance Program Requirements." (Reference 7)

The methods in ASME Section XI, Appendix G define ASME International's methods for preforming calculation of site-specific P/T limit curves.

The regulation in 10 CFR Part 50, Appendix H, requires a licensed owner of a U.S. light water reactor to have surveillance monitoring program for ferritic components located in the beltline of the RPV, if the RPVs in the reactors are anticipated to operate with cumulative neutron fluence exposures in excess 1 x 10^{17} n/cm^2 (E > 1.0 MeV). Specifically, Section III.A in the 10 CFR Part 50, Appendix H states that "No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E 1 MeV)." Section III.B in 10 CFR Part 50, Appendix H, states that "Reactor vessels that do not meet the conditions in paragraph III.A of this appendix must have their beltline materials monitored by a surveillance program complying with ASTM E 185, as modified by this appendix."

The regulation in 10 CFR Part 50, Appendix H, also requires, in part, that for each capsule withdrawal, the test procedure for the testing of capsule specimens must meet the requirements specified in ASTM Standard Practice E185-82 (Reference 8) to the extent practicable for the configuration of the specimens in the capsules. Additionally, 10 CFR Part 50, Appendix H requires each capsule withdrawal and the test results for test specimens in the capsules to be the subject of a summary technical report that is required to be submitted within one year of the capsule withdrawal, unless an extension is granted by the Director of Nuclear Reactor Regulation.

2.4 Regulatory Guidelines and Staff-Approved Industry Guidelines That May Be Applied to P/T Limit or LTOP [low temperature overpressure protection] System Setpoint Calculations

NRC Regulatory Guide (RG) 1.99, Revision 2 (Reference 9) describes a method acceptable to the NRC staff for calculating ART values (and upper shelf energy values) of ferritic base metal or weld components that are located in the beltline of the RPV. For ART objectives, the methodology provides two positions and methods for calculating component-specific chemistry factors (CFs) that are part of the ART calculations:

(a) use of the component-specific copper and nickel alloying chemistries and the CF tables provided in the RG, or

(b) for RPV base metal and weld materials that are included in the RPV surveillance program, use of the applicable surveillance data for the materials.

This RG also provides a method that may be used to calculate attenuated neutron fluence values across the RPV wall thickness and a methodology for calculating the component-specific fluence factors that are used in the ART calculations.

NRC RG 1.190 (Reference 10) describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence with respect to the General Design Criteria (GDC) contained in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

The guidance in NUREG-0800, "Standard Review Plan," Section 5.2.2, Revision 3 "Overpressure Protection" (Reference 11) provides, in part, guidance for calculating PWR low temperature overpressure protection (LTOP) system pressure lift and system enable temperature (arming temperature) setpoints. Additional guidance for LTOP system setpoints is given in NUREG-0800, Branch Technical Position (BTP) 5-2 (Reference 12).

<u>Westinghouse Non-Proprietary Class 3 Report WCAP-14040-A, Revision 4</u>. This referenced Westinghouse report provides an NRC-approved methodology that may be used to generate P/T limit curves and LTOP system pressure lift and arming temperature setpoints for Westinghouse-design PWRs. The methodology WCAP-14040-A, Revision 4 (Reference 13) includes and applies the methodologies in several ASME Code Cases, including but not limited to those in:

- ASME Code Case N-640 (Reference 14), as referenced for use in Section 2.8.1 of WCAP-14040-A, Revision 4. The code case provides the basis for using a linear elastic fracture mechanics critical stress intensity factor (K_{IC}) in P/T limit curve calculations, as opposed to the use of a dynamic crack arrest stress intensity factor (K_{Ia}) for the curve calculations.

  ASME Code Case N-641 (Reference 15), as referenced in Section 3.3 of WCAP-14040-A, Revision 4. The methods of analysis in ASME Code N-641 provide, in part, alternate criteria that may be used for establishment of LTOP system arming temperature setpoints (Refer to the discussions in Section 3.2.2 of this SE for the applicant's application of Code Case N-641).

NRC Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (Reference 16) provides guidance that may be used to ensure that the scope of P/T limit evaluations includes all components in the RCPB that are made from ferritic steel materials, including assessment of RPV nozzle appurtenances that are made from ferritic steel materials and their ferritic nozzle-to-vessel pressure retaining welds.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 <u>Evaluation of the Proposed Changes to the P/T Limit Curves</u>

The licensee describes its basis for meeting the requirements specified in 10 CFR Part 50, Appendix G, for P/T limit calculations in Section 5.3.1.5 of the updated final safety analysis report (UFSAR) for Catawba, Unit1 and Unit 2 (Reference 17). The licensee describes its unit-specific RPV material surveillance programs for Catawba, Unit 1, and its bases for meeting the RPV surveillance program requirements of 10 CFR Part 50, Appendix H, in UFSAR Section 5.3.1.6 (Reference 18).

The NRC staff reviewed the proposed P/T limit curves for 42.7 EFPY in the referenced TS figures to ensure that the P/T limit curves meet the following regulatory objectives:

- the P/T limit curves given in LAR Figures 3.4.3-1 and 3.4.3-2 (as based on 51 EFPY fluences) are at least as conservative and incorporate the minimum safety margin requirements specified in Appendix G of the ASME Boiler and Pressure Vessel Code, Section XI;

- the ART values used in the development of the P/T limit curves have accounted for: (a) the increases in neutron irradiation (i.e., neutron fluence exposures) of the RPV beltline weld and base metal components needing assessment of ART, and (b) the results of the licensee's 10 CFR Part 50, Appendix H RPV materials surveillance program; and

- the proposed P/T limit curves have appropriately accounted for the minimum temperature requirements for P/T limit curves assessments in Section IV.A.2.b of the 10 CFR Part 50, Appendix G and that are specified in Table 1 of 10 CFR Part 50, Appendix G

The NRC staff also reviewed the licensee's basis for scaling back the P/T limit curves to 42.7 EFPY is valid and appropriately addresses the measurement uncertainty recapture LAR impacts evaluated in WCAP-17669-NP, Revision 0.

3.1.1 P/T Limit Curve Review – Neutron Fluence Assessment

Determination of P/T limit curves involves three basic steps: (1) calculation of neutron fluence projection for a particular EFPY value, (2) determination of ARTs based on these fluence projections, and (3) determination of the P/T limit curves based on the updated ART values. Then, the effect on the updated P/T limit curves on the LTOP settings is evaluated.

It is important to note that, in April 2016, DEC received NRC-approval for the implementation of a measurement uncertainty recapture (MUR) uprate at Catawba Unit 1 (hereafter MUR LAR) (Reference 19). As a result, two neutron fluence calculations are referenced in the licensee's current LAR submittal: a 1998 neutron fluence calculation at pre-MUR conditions that utilized the DORT 2D discrete ordinates radiation transport code (WCAP-15117-NP) (Reference 20), and a 2013 neutron fluence calculation at post-MUR conditions (WCAP-17669-NP, Revision 0; Reference 4) that utilized the RAPTOR-M3G 3D discrete ordinates radiation transport code (WCAP-16083-NP, Revision 1) (Reference 21). In 2001, the pre-MUR neutron fluence calculation results were used to generate a set of P/T limit curves (WCAP-15448-NP, Revision 1 (Reference 3)) with applicability, at the time, of 51 EFPY. The licensee's current submittal makes use of the post-MUR neutron fluence calculation results (WCAP-17669-NP, Revision 0 (Reference 4), to demonstrate the continued applicability of the pre-MUR P/T limit curves at a reduced EFPY applicability date. Because the licensee used post-MUR neutron fluence calculation results in assessing P/T curves based on pre-MUR neutron fluence calculation results, the NRC staff examined each set of results and compared them for consistency.

3.1.1.1 Post-Measurement Uncertainty Recapture Uprate Neutron Fluence Assessment

In its application dated July 2, 2019, the licensee indicated that the neutron fluence methodology utilized for Catawba, Unit 1, follows the guidance of RG 1.190. The NRC staff evaluated the neutron fluence methodology in accordance with that RG.

The guidance provided in RG 1.190 states that an acceptable neutron fluence calculation meets the following criteria:

- Performed using an acceptable methodology (Regulatory Positions 1.1-1.3);

- Contains methodology qualification and uncertainty estimates based on:
  - Analytic uncertainty analysis identifying possible sources of uncertainty (Regulatory Position 1.4.1)
  - Comparison with benchmark measurements and calculations from applicable test facilities (Regulatory Position 1.4.2) including:
    - A pressure vessel simulator benchmark
    - Fluence calculation benchmark (e.g., benchmark problems provided by the NRC in NUREG/CR-6115) (Reference 22)

- Estimation of fluence calculational bias and uncertainty (Regulatory Position 1.4.3); and
- Demonstrates plant-specific qualification by comparison to measured fluence values (Regulatory Position 1.4.2)

The NRC staff reviewed the LAR and LAR Attachment 3, WCAP-17669-NP (Reference 4), which presents the post-MUR [measurement uncertainty recapture] fluence calculational results. Concerning the calculational method itself, WCAP-17669-NP states that RAPTOR-M3G with the BUGLE-96 cross-section library (a 67 energy-group coupled neutron-gamma ray ENDF/B-VI-based dataset) was used to perform the neutron fluence calculations for Catawba, Unit 1, under MUR power uprate conditions. RAPTOR-M3G, as presented in WCAP-18124-NP-A, Revision 0 (Reference 23), is an NRC-approved methodology for neutron fluence calculations. However, WCAP-17669-NP does not reference the approved RAPTOR-M3G WEC report. Instead, WCAP-17669-NP references WCAP-16083-NP, Revision 1 (Reference 21), which provides comparisons of RAPTOR-M3G to the approved TORT code to demonstrate acceptability of the method.

While WCAP-16083-NP, Revision 1, is not approved for generic applicability, the NRC staff previously reviewed the report as part of the licensee's 2014 MUR LAR (Reference 24) and found the application of the methodology acceptable in that instance and that it satisfied the first criterion listed above from RG 1.190. The NRC staff's prior review of WCAP-16083-NP, Revision 1, also found that RAPTOR-M3G is supported by an analytic uncertainty analysis and methods qualification using the standard benchmark problems found in RG 1.190. Specifically, the licensee compared the calculations to benchmark measurements from the Poolside Critical Assembly simulator at the Oak Ridge National Laboratory (ORNL) and the H. B. Robinson Steam Electric Plant benchmark experiment. The licensee combined the results of these analyses to determine an overall calculational uncertainty and bias, with an estimated uncertainty of less than 20 percent and no systematic bias. Additionally, the prior NRC staff review of WCAP-16083-NP, Revision 1, found the analysis and overall uncertainty to be acceptable because it was in accordance with RG 1.190 and respectively satisfied the second and third criteria listed above from RG 1.190 as well.

Similarly, WEC report WCAP-17669-NP, Revision 0, which is included in the licensee's application dated July 2, 2019, was previously provided to and reviewed by the NRC staff as part of the licensee's MUR LAR. The NRC staff's prior review found WCAP-17669-NP contains acceptable plant-specific benchmarking for Catawba Unit 1. Specifically, the WCAP-17669-NP, Revision 0 fluence results were generated considering the appropriate number of cycles over which Catawba Unit 1 operated at pre- and post-MUR conditions, and the surveillance capsule fluence was calculated to within 20 percent of the measured fluence, which is in accordance with RG 1.190. This satisfied the fourth criterion listed above. Based on satisfying each of the above four criteria, the NRC staff found the RAPTOR-M3G neutron fluence calculation acceptable for the plant-specific MUR LAR approval.

The NRC staff previously reviewed WCAP-16083-NP, Revision 1, and found the fluence methodology acceptable for use in the licensee's MUR LAR. The Catawba, Unit 1, plant configuration, as presented in the licensee's MUR LAR, remains unchanged (that is, the core model and inputs used in the fluence methodology continue to accurately represent the plant). The NRC staff also verified that WCAP-16083-NP, Revision 1, continues to satisfy the first criterion listed above from RG 1.190. Additionally, as the analytical uncertainty analysis and the Catawba, Unit 1, plant-specific benchmarking, via measured-to-calculated surveillance capsule fluence comparisons, make use of the RAPTOR-M3G methodology as presented in WCAP-16083-NP, the NRC staff concludes that the uncertainty analysis and benchmarking remain acceptable. The methods qualification is acceptable because it uses standard benchmark problems found in RG 1.190. Thus, the remaining criteria listed above from RG 1.190 continue to be satisfied. Therefore, the NRC staff concludes that the RAPTOR-M3G neutron fluence calculations presented in WCAP-17669-NP is acceptable with respect to the current LAR.

3.1.1.2 Pre-Measurement Uncertainty Recapture Uprate Neutron Fluence Assessment

The pre-MUR Catawba Unit 1 neutron fluence calculation results presented in WCAP-15117-NP were generated using the DORT 2D discrete ordinates radiation transport code Version 2.8.14 and the BUGLE-93 cross-section library (a 67 energy-group coupled neutron-gamma ray ENDF/B-VI-based dataset). The NRC staff's previous review of WCAP-15117-NP for the Catawba license renewal (Reference 25) in 2003 found that the fluence computational methodology adhered to the guidance of RG 1.190 and that the results were acceptable. In that prior review, the NRC staff noted specifically that the assumed EFPY of operation to the end of the extended license for the Catawba Units is 54 EFPY and the projected neutron fluence values are conservative for reactor vessel fluence. As this prior review was not application-specific, and the neutron fluence calculation results were found to be conservative and adhered to the guidance of RG 1.190, the NRC staff concludes that the results are also acceptable for referencing in the licensee's current LAR application.

3.1.1.3 Comparison of Neutron Fluence Calculation Results

As the licensee is using the post-MUR neutron fluence calculation results to assess the continued applicability of P/T curves based on pre-MUR neutron fluence calculations, the NRC staff compared the two calculational results for consistency. The NRC staff performed a linear extrapolation of the neutron fluence calculational results from Table 2-2 of WCAP-17669-NP to 51 EFPY and compared the results to the 51 EFPY neutron fluence calculation results presented in Table 6-13 of WCAP-15117-NP. This comparison showed a 16.2 percent decrease in post-MUR neutron fluence at the reactor vessel clad/base metal interface. Given the 1.7 percent MUR power uprate of Catawba, Unit 1, which will have a proportionally linear increase in core neutron flux, a 16.2 percent decrease in post-MUR neutron fluence is unexpected.

The NRC staff noted a discrepancy between pre- and post-MUR fluence that was not addressed in the application. Therefore, the NRC staff requested additional information from the licensee. The licensee's response (Reference 2) indicated that low-leakage core loading pattern strategies were implement in Catawba, Unit 1, post-1998, and the fluence values reported in WCAP-15117-NP were calculated prior to the implementation of these low-leakage core loading patterns. This would result in a reduction of post-MUR calculated fluence values when compared to pre-MUR results. The implementation of low-leakage core loading patterns is a possible explanation for the observed reduction in projected end-of-life vessel fluence between pre-MUR and post-MUR results. The NRC staff noted that the fluence values reported in WCAP-17669-NP were developed using cycle-specific and plant-specific neutron transport analyses. Therefore, the fluence results of WCAP-17669-NP for cycles prior to 1998 will be representative of core loading patterns prior to the implementation of low-leakage strategies.

To assess the impact of the implementation of low-leakage core loading patterns, the NRC staff made additional comparisons of the fluence calculations at the reactor vessel clad/base metal interface from Table 2-3 of WCAP-17669-NP to those presented in Table 6-3 of WCAP-15117-NP, specifically focusing on how well the results agreed prior to 1998 and after. The results demonstrated very good agreement between the fluence calculations performed pre- and post-MUR for the cycles before 1998 (within ±4 percent) and a marked decrease in calculated fluence for all cycles after 1998 (between -12 percent to -25 percent). Given the very good agreement between fluence calculations prior to the implementation of low-leakage core loading pattern strategies and the stark contrast between the neutron fluence calculations after the loading pattern strategies were implemented, the NRC staff concludes that the licensee's response acceptably explains the noted decrease in calculated neutron fluence between pre- and post-MUR results at 51 EFPY.

3.1.1.4 Conclusion Regarding Neutron Fluence Assessment

Based on the above, the NRC staff has determined that the neutron fluence calculations provided by the licensee adequately address the four criteria of RG 1.190. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the use of the RAPTOR-M3G post-MUR neutron fluence calculation and the use of the WCAP-15117-NP (also reported in WCAP-15448-NP) pre-MUR neutron fluence calculation.

3.1.2 P/T Limit Curve Review – P/T Limit Conservatism Assessment

The staff noted that P/T limit curves for 42.7 EPFY proposed in TS Figure 3.4.3-1, "(UNIT 1 ONLY) RCS Heatup Limitations," and TS Figure 3.4.3-2, "(UNIT 1 ONLY) RCS Cooldown Limitations," of the LAR are based on NRC staff-approved methods of analysis in WCAP-14040-A, Revision 4, and the following P/T limit curve assessments for 51 EFPY in WCAP-15448, Revision 1: (a) for the pressure test limitation curve in Figure 3.4.3-1, the assessment in the WCAP report for pressure testing under steady state conditions, (b) for the heatup and core critical limitation curves in Figure 3.4.3-1, the assessments for heatups using a maximum heatup rate of 60 ºF/Hr, and (c) for the cooldown limitation curve in Figure 3.4.3-2, the assessment for reactor cooldowns using a maximum cooldown rate of 100 ºF/Hr.

The NRC staff verified the accuracy and validity of these P/T limit calculations through the performance of: (a) a review of the information included in the LAR and in WCAP-17669, Revision 1 to address regulatory matters raised in RIS 2014-11; (b) independent ¼-T ART and ¾-T ART value calculations for the beltline materials in the Catawba Unit 1 RPV, including the application of site-specific and sister-plant surveillance data for those RPV forging and weld materials that are included in the licensee's RPV surveillance program; and (c) independent P/T limit calculations for plant operations during heatups and cooldowns of the reactor, critical power operations, and operations during pressure testing of the RCPB.

P/T Limit Curve Review – RIS 2014-11 Review

The NRC staff verified that the licensee's basis for its evaluation of RPV nozzle impacts on the proposed P/T limit analyses was previously addressed and evaluated by the licensee in WCAP-17669, Revision 1, and previously approved in the staff's safety evaluation for the MUR LAR (Reference 19). Specifically, the staff verified that the neutron fluence assessment for the RPV was consistent with the licnesee's previous evaluation in the MUR LAR, in that it did not include any ferritic nozzles (e.g., ferritic RPV inlet and outlet nozzles or ferritic RPV instrumentation nozzles) whose neutron fluence exposures would exceed a neutron fluence exposure of 1×10^{17} n/cm^2 (E > 1.0 MeV) through 54 EFPY. The staff also verified that the stress assessments for the nozzles do not include any stress intensity factor considerations that, otherwise, might potentially cause the P/T limit assessments of the nozzles to be limiting relative to those calculated using the ¼-thickness (¼-T) and ¾-thickness (¾-T) ART values for the limiting ¼-T and ¾-T components in the beltline of the RPV. The NRC staff confirmed that the P/T limits analyses performed in the licensee's current LAR application have appropriately accounted for and follows guidance in NRC RIS No. 2014-11 (Reference 16). Based on the above, the NRC staff concludes that the licensee has demonstrated sufficiently that nozzles included in the RPV will not be limiting for the P/T limit calculations based on the assessed RPV nozzle stress levels and projected nozzle fluence exposures through 42.7 EFPY.

P/T Limit Curve Review – ART Value Assessment

The NRC staff verified that the licensee's ¼-T ART and ¾-T ART values used in the P/T limit calculations were established in accordance with the methods of analysis and criteria specified in WCAP-14040-A, Revision 4, which are based on the methods in RG 1.99, Revision 2. The NRC staff performed independent calculations of the ¼-T and ¾-T ART values to verify whether the following ¼-T and ¾-T ART values used by licensee are the appropriate limiting ART value inputs for the P/T limit calculations:

- A limiting ¼-T ART value of 47 ºF, as based on the calculation for RPV lower shell forging 04 (Material Heat 527708) using Regulatory Position C.1.1 in RG 1.99, Revision 2 and the applicable CF table for base metal components in the RG.

- A limiting ¾-T ART value of 34 ºF, as based on the calculation for RPV intermediate shell forging 05 (Material Heat 411343) using Regulatory Position C.2.1 in RG 1.99, Revision 2 and the applicable RPV surveillance program data for the forging component.

The NRC staff verified that the components referenced above are limiting components for the P/T limit calculations based on an evaluation of both the stress levels and ART values for the components. For these components, the staff calculated ¼-T and ¾-T ARTs for the components at 51 EPFY. The NRC staff's findings were equivalent to those calculated for the components by the licensee. Based on its independent calculations, the NRC staff concludes that the ART values referenced above are valid, limiting ¼-T and ¾-T ART values for the P/T limit calculations that were performed by the licensee for 51 EFPY and that the ART values are acceptable with respect to implementation of the proposed P/T limit curves.

P/T Curve Review - Stress Intensity Factor, Minimum Temperature Requirement, and P/T Limit Curve Conservatism Impacts.

The NRC staff confirmed that the licensee's proposed P/T limit curves included the following safety factors required by the methods in ASME Section XI, Appendix G: (a) for plant heatup and cooldown curves, an ASME-required safety factor of two (2) on applied primary membrane and primary bending stress intensity factors, and (b) an ASME-required safety factor of 1.5 on applied primary membrane and primary bending stress intensity factors for P/T limits that apply to plant operations during scheduled pressure testing of the RCPB.

The NRC staff determined that there were two potential gaps in relation to the P/T limit heatup curve for 51 EFPY in Attachment 1 of the LAR and information related to this P/T curve that WEC had included in WCAP-15448, Revision 1 (i.e., Attachment 2 of the LAR): (a) WEC did not include the 10 CFR Part 50, Appendix G, minimum temperature requirement in the P/T limit curve in WCAP 15448, Revision 1, for portions of the curve that cover pressures in excess of 20 percent of the RPV preservice hydrostatic test pressure (i.e., P > 621.25 psi for Catawba Unit 1),[1] and (b) the staff's independent calculation of its P/T limit heatup curve yielded slightly more conservative P/T limit points than those calculated by WEC for temperatures lower than 110 ºF. However, the NRC staff noted that the DEC's LAR appropriately addressed and resolved these two potential gaps through DEC's inclusion of the appropriate lowest service temperature and minimum temperature requirements in the P/T limit heatup curve that was provided in Figure 3.4.3-1 of LAR Attachment 1.

Although the NRC staff is not endorsing or approving any of the P/T limit heatup curves that were provided by WEC in WCAP-15448, Revision 1, the NRC staff confirmed that DEC's plant-specific heatup curve in the LAR appropriately includes a lowest service temperature/minimum temperature requirement of 60 ºF for RCS operating pressures less than or equal to 621.25 psi, and a minimum temperature requirement of 116 ºF for RCS operating pressures greater than to 621.25 psi. Because the NRC staff confirmed that the P/T limit heatup curve provided in LAR Attachment 1 includes the appropriate minimum temperature requirements required by Table 1 in 10 CFR Part 50, Appendix G, the staff concludes that the P/T limit heatup curve in Figure 3.4.3-1 of LAR Attachment 1 is acceptable for implementation by the licensee.

The NRC staff confirmed that the licensee's proposed P/T limit curves provided in Figure 3.4.3-1 of LAR Attachment 1 for pressure test conditions and for operational changes with the reactor in the critical condition and in Figure 3.4.3-2 of LAR Attachment 1 for reactor cooldown conditions were conservative relative to those calculated by the staff in accordance with ASME Section XI, Appendix G. The NRC staff also confirmed that the P/T limit curves included the appropriate minimum temperature/lowest service temperature requirements in Table 1 of Appendix G of 10 CFR Part 50. Additionally, for the P/T limit curve that applies to plant operations when the reactor is critical , the NRC staff noted that the licensee's curve conservatively sets a minimum temperature requirement (T_{min}) of 156 ºF (i.e., T_{min} = limiting $RT_{NDT\text{-}Closure\ Flange}$ of -4 ºF + 160 ºF)

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[1] For the P/T limit evaluations that apply to Catawba Unit 1, the preservice hydrostatic test pressure is set at 125% of the rated normal operating pressure of 2485 psi (i.e., $P_{Hydrostatic\ Test}$ = 1.25 * 2485 psi = 3106.25 psi). Thus 20% of the pre-service hydrostatic test pressure equals 20% of 3106.25 psi or 621.25 psi. This is consistent with the P/T limit heatup curve proposed and provided in the Figure No. 3.4.3-1 of the LAR.

throughout the RCS operating pressure range assessed in the P/T limit analysis. The NRC staff considers this to be conservative for the core critical curve because for RCS operating pressures less than or equal to 20 percent of the preservice hydrostatic test pressure (i.e., for P ≤ 621.25 psi), the licensee could have set the minimum temperature requirement to the minimum temperature for performing the system pressure test (i.e., to a T_{min} of 108 ºF), as allowed for in Table 1 of 10 CFR Part 50, Appendix G.

Based on the above, the NRC staff concludes that the P/T limit points provided in Figures 3.4.3-1 and 3.4.3-2 of LAR Attachment 1, are acceptable because: (a) DEC corrected any potential non-compliances or issues with the curves for 51 EPFY provided in WCAP-15448, Revision 1 in the P/T limit curves that were provided in Figures 3.4.3-1 and 3.4.3-2 of LAR Attachment 1, (b) the P/T limit curves provided in Figures 3.4.3-1 and 3.4.3-2 of the LAR appropriately incorporate the minimum temperature criteria required by Table 1 in 10 CFR Part 50, Appendix G, and the minimum safety margins on membrane stresses required by ASME Section XI Appendix G, and (c) the staff verified that the proposed P/T limit curves are at least as conservative than those that would be obtained if the methods of analysis in ASME Section XI, Appendix G were used for generation of the P/T limit curves, as required by 10 CFR Part 50, Appendix G, Section IV.A.2.b. The NRC staff's evaluation of the licensee's basis for scaling the proposed P/T limit curves in LAR Figures 3.4.3-1 and 3.4.3-2 to 42.7 EFPY from 51 EFPY is addressed in the next subsection.

Scaling back of the P/T limit curves to 42.7 EPFY.

The neutron fluence values and ART values that the licensee used to develop the P/T limit curves for 51 EFPY in WCAP-15448, Revision 1, did not account for the increase in the neutron fluence exposure to the RPV beltline components that would result from the licensee's implementation of MUR power uprate for the facility, which was approved in Amendment No. 281 for Catawba Unit 1 (Reference 19). Therefore, the licensee used the difference in the reported fluences between those reported in WCAP-15448, Revision 1, and those reported in WCAP-17669-NP, Revision 0, as the basis for scaling back the effective date of the P/T limit curves for 51 EFPY to 42.7 EPFY.

In performing the scaling, the licensee first compared the pre- and -post MUR limiting ART values and materials as determined from the respective fluence calculations. The pre-MUR fluence results reported in WCAP-15448, Revision 1 are for 51 EFPY, and the post-MUR fluence results reported in WCAP-17669-NP, Revision 0 are for 54 EFPY. The licensee stated that the ¼-T limiting ART value at 51 EFPY is 47 °F, which is less than the ¼-T limiting ART value at 54 EFPY of 51 °F. Because the pre-MUR limiting ¼-T ART value used in the development of the 51 EFPY P/T limit curves is slightly lower than the post-MUR limiting ¼-T ART value, the 51 EFPY P/T limit curves do not bound the post-MUR results. As a result, the licensee identified that the limiting material for the ¼-T location changed from the upper shell forging 06 to the lower shell forging 04 (which is associated with the 51 EFPY limiting ¼-T ART value) and calculated the fluence this material would need under post-MUR conditions to yield the limiting ART of 47 °F. This fluence value was then used to determine a revised EFPY applicability date for the P/T limit curves by linearly interpolating on the post-MUR fluence projections. To summarize, the licensee used the post-MUR fluence results to determine the EFPY at which the ¼-T limiting material would achieve an ART of 47 °F. The results showed the effective date of the P/T limit curves must be decreased from 51 EFPY to 42.7 EFPY.

Because fluence is neutron exposure accumulated over time, using post-MUR fluence results to determine an updated time at which a pre-MUR limiting ART value will occur under post-MUR conditions will account sufficiently for the increase in neutron exposure from the power uprate, provided both sets of fluence calculations are found acceptable (as discussed in Section 3.1.1). In assessing the acceptability of the licensee's scaling results, the NRC staff performed a series of confirmatory calculations. These calculations yielded results consistent with that of the licensee, supporting the revised applicability limit of 42.7 EFPY for the P/T curves as discussed in the LAR. The NRC staff, therefore, concludes that the licensee's scaling approach is acceptable.

3.2 Low Temperature Overpressure Protection (LTOP) Analysis Review

The licensee is proposing to replace the current P/T Limits in TS 3.4.3, applicable to 30.7 EFPY, with new P/T limits applicable to 42.7 EFPY. As part of this LAR, the licensee also provided a discussion of the LTOP analysis to support the proposed P/T limits.

The NRC staff reviewed the LTOP analysis and determined that the analysis was acceptable for supporting the proposed new P/T limits in TS 3.4.3, as described below.

3.2.1. LTOP Evaluation

Acceptance Criterion 4 on page 5.2.2-7 in Section 5.2.2 of NUREG-0800, Revision 3 (Reference 11) specifies that the LTOP system be designed in accordance with the guidance of Branch Technical Position 5-2 (Reference 12), which specifies that the LTOP system be capable of relieving pressure during all anticipated overpressure events at a rate sufficient to satisfy the TS limits while operating at low temperatures.

3.2.1.1 LTOP Analysis

Section 5.2.2.2 of the UFSAR for Catawba, Unit 1 and Unit 2, discusses the LTOP system, which consists of two power-operated relief valves (PORVs) with reduced lifted settings, two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve for pressure relief. Two RCS relief valves were required for redundancy to satisfy the single failure criterion. One RCS relief valve would have adequate relief capacity to keep from overpressure for the required coolant input capability. Events that were capable of overpresurizing the RCS were categorized as either mass addition or energy input events. Typical mass addition events included inadvertent safety injection of one high-head pump and one intermediate-head pump, and charging/ letdown flow mismatch. Energy input events included inadvertent actuation of pressurizer heaters, loss of RHR cooling, and reactor coolant pump startup with temperature asymmetry within the RCS or between the RCS and steam generator.

On page 5 of its LAR dated July 2, 2019, the licensee discussed its LTOP analysis and indicated that the maximum calculated pressure for Catawba Unit 1 was 609.5 psig [pounds per square inch] (with inclusion of location adjustment and instrumentation uncertainty). The calculation assumed the current LTOP PORV setpoint of 400 psig and RHR suction relief valve setpoint of 463 psig. The licensee showed that the calculated new limiting pressure for the heatup and cooldown curve at 42.7 EFPY was 1,089 psig for the reactor vessel beltline region and remained 621 psig for the closure head and vessel flange region. Since the maximum calculated pressure of the LTOP analysis of 609.5 psig was below the new limiting pressure for the heatup and cooldown curve at 42.7 EFPY, the licensee stated that the current PORV low-pressure setpoint (400 psig) and RHR suction relief valve setpoint (463 psig) are valid for plant operation to 42.7 EFPY for Catawba Unit 1.

During the review, the NRC staff found that the relief valve setpoints assumed in the LTOP analysis were in the lower range that is allowed by TS LCO 3.4.12. For example, the assumed PORV lift setpoint of 400 psig was lower than the maximum TS allowable value of 425 psig and the assumed RHR suction relief valve setpoint of 463 psig was lower than the maximum setpoint of 509 psig in TS 3.4.12.

The assumed lift setpoints (which are in the lower range of than the TS values) would cause the opening of the relief valves to occur at a lower pressure, and, therefore, result in calculations that did not reflect the worst alloweable theoretical plant conditions (i.e., the calculation was nonconservative).

The NRC staff requested the licensee to justify the use of the lower RCS relief valves setpoints in the LTOP analysis to support plant operation to 42.7 EFPY. In its response (Reference 2), the licensee clarified that the 400 psig value for the LTOP PORV setpoint and 463 psig value for the RHR relief valve setpoint were the actual calibration setpoints for these devices. The allowable TS setpoints represent the actual calibration setpoinsts with the drift tolerance. For the design basis calculation, the actual calibration setpoints with inclusion of the maximum allowable drift tolerance were used to obtain the maximum RCS pressure, which is conservative.

An LTOP analysis for determination of the maximum RCS pressure during plant operations at low temperature conditions was performed by the licensee for two limiting events. They were: (1) a mass addition event assuming injection from two high-pressure injection pumps; and (2) a heat addition event assuming start of a reactor coolant pump at a 50 ^{0}F temperature differential between a steam generator and the RCS. For each event analyzed, the analysis assumed that a single PORV or single RHR relief valve was open for pressure relief. The licensee's anlaysis assumed the following setpoints:

> For a PORV, it was 460 psig. This value represented 400 psig for the actual calibration setpoint and 60 psig for the uncertainty. For an RHR relief valve, it was 509.3 psig. This value represented 463 psig for the actual calibration setpoint and 46.3 psig for 10 percent setpoint tolerance.

The NRC staff found that the assumed setpoint of 460 psig bounded the TS upper limit of 425 psig for a PORV setpint, and, therefore, determined that it was conservative and acceptable. The NRC staff also found that the assumed setpoint of 509.3 psig bounded the TS upper limit of 509 psig for an RHR relief valve setpoint, and therefore, determined that it was conservative and acceptable. Also, the licensee's analysis accounted for valve accumulation at the transient flow rate, RCS differential pressure for different reactor coolant pump combinations and elevation correction between the relief device pressure input and the reactor vessel head flange.

Since the LTOP analysis for each event was based on conservative valve setpoints, accounted for plant conditions, and resulted in a peak pressure less than the limiting reactor vessel closure head flange limit, the NRC staff determined that the LTOP analysis was acceptable to support plant operation to 42.7 EFPY for Catawba Unit 1.

The licensee indicated on page 5 of the LAR that the calculated new limiting pressure for the heatup and cooldown curves shown in in Figure 3.4.3-1 and Figure 3.4.3-2 at 42.7 EFPY in the

LAR was 1,089 psig for the reactor vessel beltline region and remained 621 psig for the closure head and vessel flange region. Based on the above, the NRC staff questioned whether both calculated pressure limits included measurement uncertainty or not.

The NRC staff requested the licensee to clarify whether the calculated pressure limits of 1,089 psig and 621 psig were consistent with the pressure limits shown in Figure 3.4.3-1 and Figure 3.4.3-2, which seemed to indicate that the pressure limits did not include margins for instrument errors. In its response (Reference 2), the licensee confirmed that the values of 1,089 psig and 621 psig and the pressure limts in Figures 3.4.3-1 and 3.4.3-2 for 42.7 EFPY were consistent with that in Figure 3.4.3-1 and Figure 3.4.3-2, in that none of the limits included measurement uncertainty.

Based on the above, the NRC staff determined that the pressure limits of 1,089 psig and 621 psig were consistent with the pressure limits shown in proposed Figures 3.4.3-1 and 3.4.3-2 and that none of them reflected measurement uncertainty. The NRC staff also found that the pressure limits of 1,089 psig and 621 psig and the applicable pressure limits in the proposed Figures 3.4.3-1 and 3.4.3-2 bounded the maximum calculated pressure of the LTOP analysis of 609.5 psig, and, therefore, concluded that the above cited pressure limits were acceptable for the LTOP protection.

Based on the above, the NRC staff concludes that: (1) the LTOP analysis adequately supported the limiting pressure for the heatup and cooldown curve at 42.7 EFPY; and (2) the limiting pressure for the heatup and cooldown curve at 42.7 EFPY would meet 10 CFR 50.36(c)(2), which requires, in part, that TSs include items in the category of LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Therefore, the NRC staff determined that the limiting pressure for the heatup and cooldown curve was acceptable for plant operation to 42.7 EFPY for Catawba Unit 1.

3.2.2 LTOP Analysis – LTOP System Arming Temperature Setpoint Assessment

In its LAR, the licensee indicated that the minimum 42.7 EFPY LTOP system enable temperature (arming temperature) is 87 ºF, and that this value is less than the minimum required LTOP system enable temperature of 210 ºF (i.e., 200 ºF with a 10 ºF measurement uncertainty allowance) currently established in TS Section 3.4.12, "Lower Temperature Overpressure Protection (LTOP) System." The licensee stated that the current LTOP system enable temperature of 210 ºF remains acceptable for plant operations of Catawba Unit 1 through 42.7 EFPY.

The NRC staff verified that the operational requirements for activating (enabling) the reactor unit's LTOP system are given in TS Section 3.4.12. This TS section requires the LTOP system to be activated whenever the unit is in one of the following TS Applicability requirements for the system:

- Operating Mode 4 [Hot Standby], when any RCS cold leg temperature is ≤ 210 ºF
- Operating Mode 5 [Cold Standby]
- Operating Mode 6 [Refueling], when the reactor vessel head is on (i.e., when the RPV closure head is bolted and tensioned)

The licensee's basis for establishing the operational requirements in TS Section 3.4.12 are provided in UFSAR Section 5.2.2.2. The UFSAR indicates that the requirements for activating the LTOP system are given in ASME Code Case N-641. The NRC staff noted that use of alternate methods in ASME Code Case N-641 would require the licensee to activate the LTOP system at a minimum RCS operating temperature of 200 ºF, at a temperature equivalent to the ART for the RPV component containing the limiting inside surface axial oriented flaw plus 40 ºF (i.e., ART + 40 ºF), or at a temperature equivalent to the ART for the RPV component containing the limiting inside surface circumferentially oriented flaw minus 85 ºF (i.e., ART – 85 ºF), whichever process yields the greater system enable temperature.

For this assessment, the NRC staff confirmed that the licensee's limiting LTOP system enable temperature value of 210 ºF includes a 10 ºF uncertainty allowance on the RCS thermocouple-instrumentation monitoring capability. Based on the above, the NRC staff concludes that the licensee's LTOP system enable temperature setpoint of 210 ºF is acceptable and will remain valid for plant operations through 42.7 EFPY because: (1) use of ASME Code Case N-641 is referenced as an acceptable methodology in WCAP-14040-A, Revision 4, and (2) the staff has confirmed that the LTOP system arming temperature setpoint value is limited using the 200 ºF setpoint criterion stated in ASME Code Case N-641 and includes a 10 ºF margin on the setpoint value. Therefore, the NRC staff concludes that the licensee's assessment basis does not have any impact on the validity of the existing enable temperature applicability requirements specified in TS Section 3.4.12.

3.3 Technical Specification Review

The NRC staff has reviewed the information provided in the licensee's submittals dated July 2, 2019, and February 13, 2020. Based on the NRC staff's evaluation of the proposed changes to TS Figures 3.4.3-1 and 3.4.3-2, as discussed in Section 3.1 of this SE, the NRC staff has determined that the proposed P/T limit curves provided in Figures 3.4.3-1 and 3.4.3-2 of the licensee's LAR Attachment 1 are acceptable for incorporation into the TS and implementation by DEC because these figures represent at least the lowest functional capability or performance levels of equipment required for safe operation of the facility. The staff concludes that the P/T limit curves in these TS figures are acceptable for implementation through 42.7 EFPY and collectively represent the new TS LCO operating restrictions on pressure and temperature for safe operations of the RCPB through 42.7 EPFY, as are required to be established in a site-specific TS LCO, in accordance with the requirement in 10 CFR 50.36(c)(2)(ii)(B). Therefore, the NRC staff concludes that the proposed changes continue to meet the requirements of 10 CFR 50.36.

4.0 <u>STATE CONSULTATION</u>

In accordance with the Commission's regulations, NRC staff notified the South Carolina State official of the proposed issuance of the amendments on July 7, 2020 (Reference 26). The State of South Carolina official had no comments (Reference 27).

5.0 <u>ENVIRONMENTAL CONSIDERATION</u>

The amendments change a requirement with respect to the installation or use of facility components 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 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 November 5, 2019 (84 FR 59657). 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Principal Contributors: J. Medoff, NRR
S. Sun NRR
K. Heller, NRR
R. Grover, NRR
K. Cotton, NRR

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