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

August 16, 2018

Mr. Ernest J. Kapopoulos, Jr.
Site Vice President
H. B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA01
Hartsville, SC 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENT NO. 260 REGARDING REQUEST TO REVISE TECHNICAL SPECIFICATION REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS TO REFLECT 24-MONTH FUEL CYCLES (EPID L-2017-LLA-0033)

Dear Mr. Kapopoulos:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 260 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated February 7, 2018.

The amendment revises TS Section 3.4.3 "RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits," to reduce the applicability terms from 50 effective full-power years to 46.3 effective full-power years in Figures 3.4.3-1 and 3.4.3-2, as a result of the removal of part-length fuel assemblies and the migration to 24-month fuel cycles.

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

Sincerely,

Dennis J Galvin

Dennis J. Galvin, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosures:
1. Amendment No. 260 to DPR-23
2. Safety Evaluation

cc: Listserv

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 260
Renewed License No. DPR-23

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee) (previously Duke Energy Progress, Inc.), dated February 7, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment. Paragraph 3.B. of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

    B. Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 260 are hereby incorporated in the 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 120 days from issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License
and the Technical Specifications

Date of Issuance: August 16, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 260

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace page 3 of Renewed Facility Operating License No. DPR-23 with the attached 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 Pages	Insert Pages
3.4-7	3.4-7
3.4-8	3.4-8

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

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

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; 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:

A. Maximum Power Level

The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2339 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 260 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 176.

MATERIALS PROPERTIES BASE
Controlling Material: Upper Shell Plate W10201-1 & Girth Weld 10-273
Limiting ART Values at 46.3 EFPY: 1/4T, 172°F & 263°F
3/4T, 153°F & 191°F

Curves applicable for heatup rates up to 60°F/Hr for service period up to 46.3 EFPY
Heatup Curves include +20°F and -80 psig Allowance for Instrumentation error.

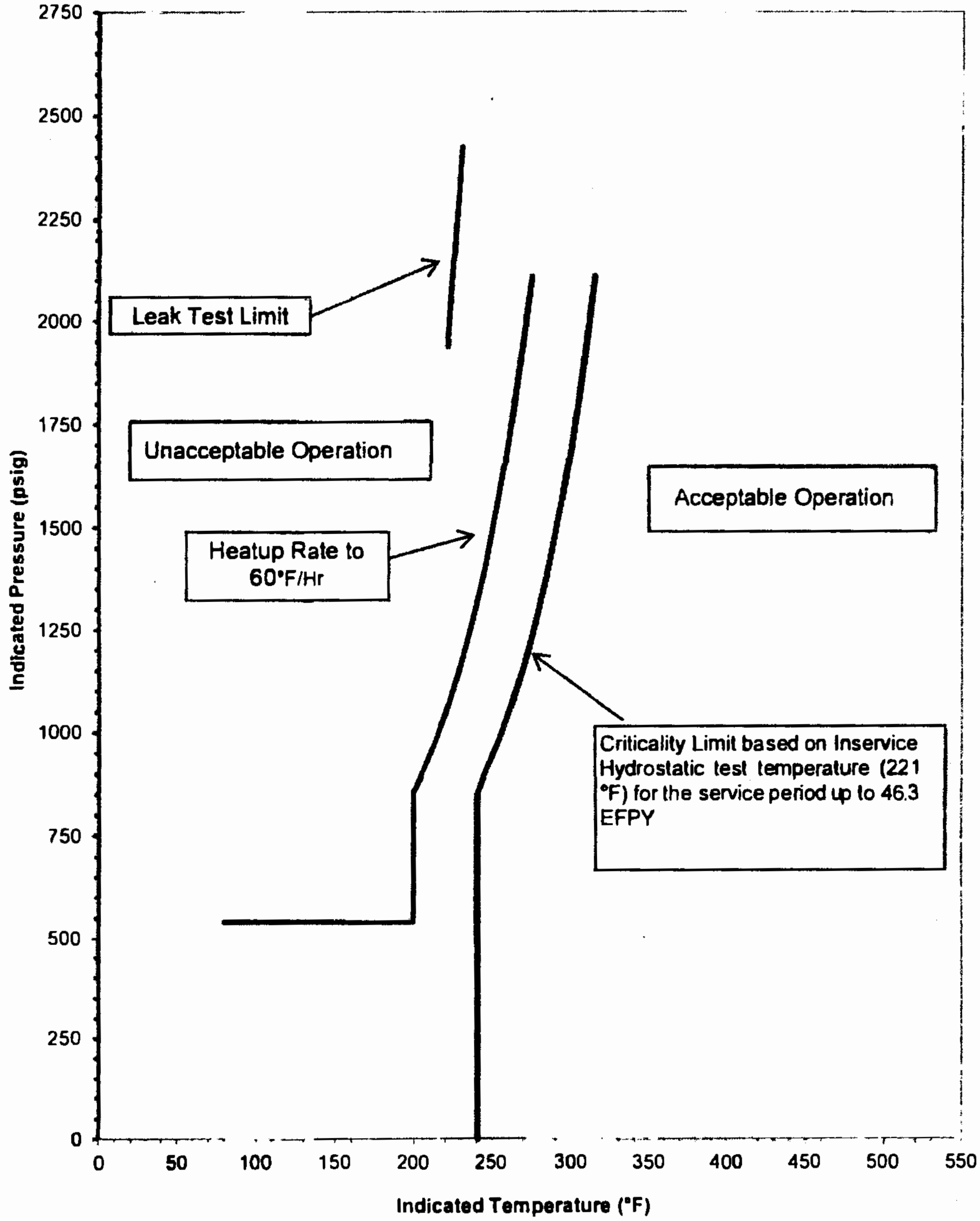

Figure 3.4.3-1
Reactor Coolant System Heatup Limits
Applicable Up to 46.3 EFPY

MATERIALS PROPERTIES BASE
Controlling Material. Upper Shell Plate W10201-1 & Girth Weld 10-273
Limiting ART Values at 46.3 EFPY: 1/4T. 172°F & 263°F
3/4T, 153°F & 191°F

Curves applicable for cooldown rates up to 100°F/Hr for the service period up to 46.3 EFPY.
Curves include +20 F and -80 PSIG Allowance for Instrumentation error.

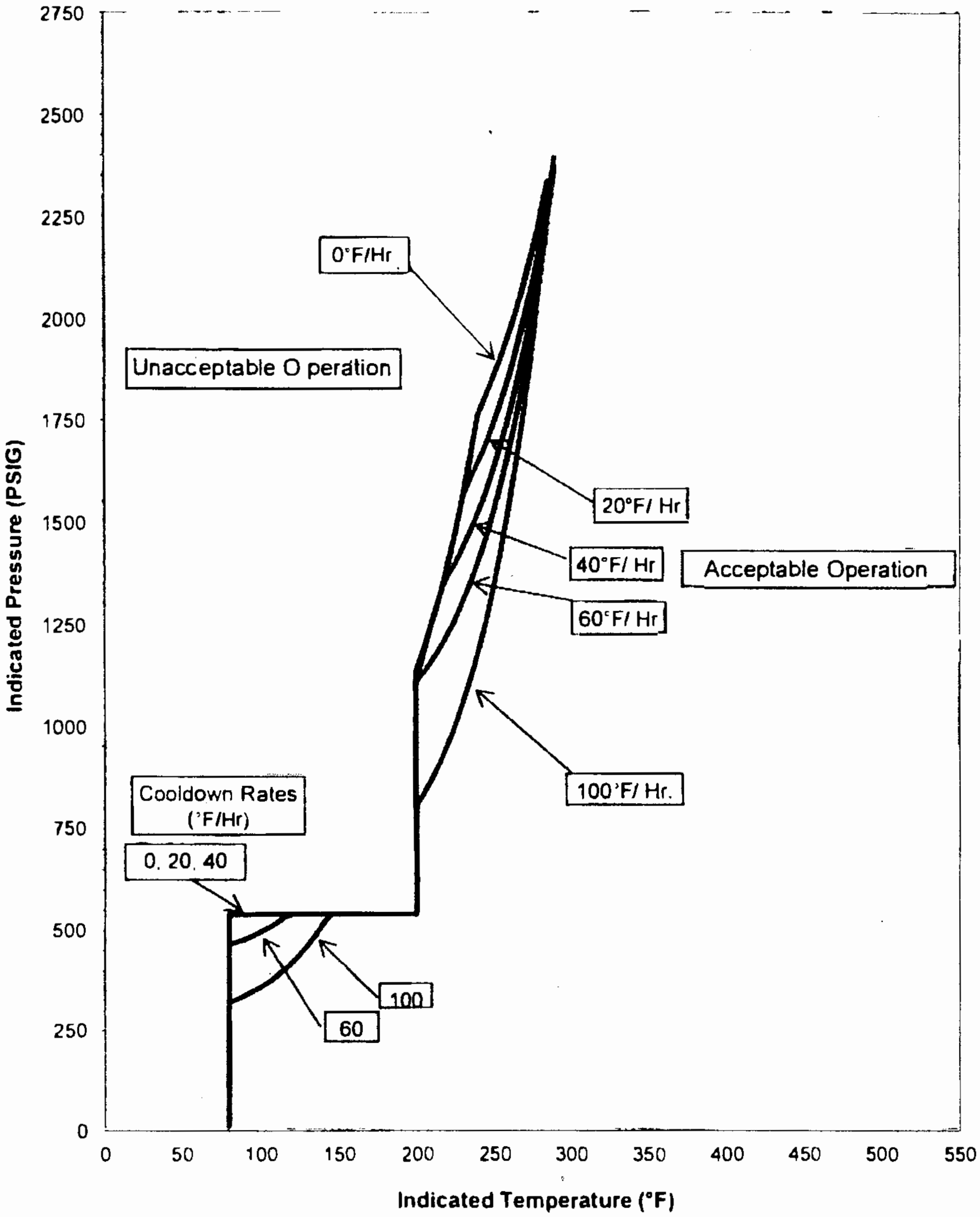

Figure 3.4.3-2
Reactor Coolant System Cooldown Limitations
Applicable Up to 46.3 EFPY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 260 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DUKE ENERGY PROGRESS, LLC

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By application dated February 7, 2018 (Reference 1), Duke Energy Progress, LLC (Duke Energy, the licensee), submitted to the United States Nuclear Regulatory Commission (NRC), a license amendment request (LAR) to revise Technical Specification (TS) 3.4.3 "RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits" of the H. B. Robinson Steam Electric Plant, Unit 2 (HBRSEP). Specifically, the licensee proposed to change the applicability of the existing HBRSEP P/T limits from 50 effective full-power years (EFPY) to 46.3 EFPY, as a result of removal of part-length shield assemblies (PLSAs) and migration of the facility to a 24-month fuel cycle. Attachment 2 to the LAR, Westinghouse Commercial Atomic Power (WCAP) report WCAP-18215-NP "H. B. Robinson Unit 2 End-of-License Extension Reactor Vessel Integrity Evaluations and Feasibility Study," Revision 0, provides the technical basis for the proposed changes.

2.0 REGULATORY EVALUATION

The NRC established requirements in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluated the acceptability of a facility's proposed P/T limits based on the following NRC regulations and guidance.

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

Section 50.60 of 10 CFR, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," imposes fracture toughness and material surveillance program requirements, which are set forth in 10 CFR Part 50, Appendices G,

"Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Program Requirements."

Appendix G to 10 CFR Part 50 requires that the P/T limits for the facility's reactor pressure vessel (RPV) be at least as conservative as those obtained by following the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The 2013 edition is the most recent version of Appendix G to Section XI of the ASME Code that has been endorsed by the NRC in 10 CFR 50.55a, "Codes and standards." The 2013 edition of Appendix G to Section XI of the ASME Code incorporates ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," and ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P/T Limit Curves."

Appendix G to 10 CFR Part 50, paragraph IV.A, states, in part: "For the reactor vessel beltline materials, including welds, plates and forgings, the values of RT_{NDT} [Reference Temperature Nil Ductility] and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part." The effects of neutron radiation are determined, in part, by estimating the neutron fluence on the RPV.

Appendix H to 10 CFR Part 50 establishes requirements for each facility related to monitoring of effects of neutron radiation on RPV material through a surveillance capsule program.

HBRSEP received its construction permit in 1967 and was licensed for operation in July 1970. The plants' design approval for the construction phase was based on the proposed general design criteria (GDC) published by the Atomic Energy Commission (AEC) for public comment in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as the "draft GDC"). On February 20, 1971, the AEC published in the *Federal Register* (36 FR 3255) a final rule that added Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants" (hereinafter referred to as the "final GDC"). Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the NRC Staff Requirements Memorandum for SECY-92-223, "Resolution of Deviations Identified during the Systematic Evaluation Program," dated September 18, 1992, the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971.

Based on a review of the HBRSEP Updated Final Safety Analysis Report (UFSAR) (Reference 2), Section 3.1, "Conformance with General Design Criteria," the NRC staff identified the following draft GDC as being applicable to the proposed amendment:

In the HBRSEP UFSAR Section 3.1.2.9, "Reactor Coolant Pressure Boundary" (GDC 9), HBRSEP states that:

> The reactor coolant pressure boundary (RCPB) shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

In the HBRSEP UFSAR Section 3.1.2.34, "RCPB Rapid Propagation Failure Prevention" (GDC 34), HBRSEP states that:

> The RCPB shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given:

a) To the provisions for control over service temperature and irradiation effects which may require operational restrictions

b) To the design and construction of the reactor pressure vessel (RPV) in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range, and for absorption of energy by plastic deformation

c) To the design and construction of RCPB piping and equipment in accordance with applicable codes.

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001 (Reference 3), describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence.

RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" (Reference 4) contains guidance on methodologies the NRC staff considers acceptable for determining the shift in RPV material transition temperature due to neutron radiation.

Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (Reference 5) clarifies that P/T limit calculations for ferritic RPV materials other than those materials with the highest reference temperature may result in more limiting P/T curves because of higher stresses due to structural discontinuities, such as those in RPV inlet and outlet nozzles.

3.0 TECHNICAL EVALUATION

3.1 Scope of Review

The NRC staff noted that WCAP-18215-NP includes information not pertinent to the proposed change to P/T limit applicability. The scope of this safety evaluation (SE) is to review only the information pertinent to the proposed change to P/T limit applicability.

3.2 Licensee's Evaluation

The licensee proposes to change the applicability of the HBRSEP P/T limits from 50 EFPY to 46.3 EFPY as a result of removal of PLSAs and migration of the facility to a 24-month fuel cycle from an 18-month fuel cycle. The licensee presented its technical basis for this change in WCAP-18215-NP (included as Attachment 2 of the LAR). Removal of PLSAs and migration to a 24-month fuel cycle increase neutron fluence, and therefore, the degree of embrittlement of the RPV materials used in determining P/T limits.

In WCAP-18215-NP, the licensee presented two scenarios to evaluate the applicability change to the existing HBRSEP 50 EFPY P/T limits. Scenario No. 1, included for comparison only, is defined to be the case with the current fuel design (i.e., with the PLSAs) and the existing 18-month fuel cycle, with fluence values calculated in WCAP-18100-NP, "Ex-Vessel Neutron Dosimetry Program for H. B. Robinson Unit 2 Cycles 16 through 29" (enclosure to Reference 6). Scenario No. 2 is the case without the PLSAs and migration to a 24-month fuel cycle.

In Section 2, "Calculated Neutron Fluence" of WCAP-18215-NP, the licensee calculated neutron fluence values for Scenario No. 2 and provided the values at various locations in the RPV beltline and inlet and outlet nozzles in Tables 2-2 through 2-8 of WCAP-18215-NP.

In Section 6, "Heatup and Cooldown Pressure-Temperature Limit Curves," of WCAP-18215-NP, the licensee evaluated the applicability of the requested change to the existing 50 EFPY HBRSEP P/T limits. In this section, for Scenarios No. 1 and No. 2, the licensee determined RPV material embrittlement by calculating the adjusted reference temperature (ART) for the RPV beltline materials, including the inlet and outlet nozzles. These RPV beltline materials are the same materials evaluated in WCAP-15827 (Attachment 4 to Reference 7) and in a 2016 Westinghouse evaluation (attachment to Reference 8), both of which were technical bases for the existing 50 EFPY P/T limits.

The licensee summarized the 50 EFPY ART calculations for Scenario No. 1 in Tables 6.1-1 and 6.1-2 of WCAP-18215-NP at the quarter thickness (1/4T) and three-quarter thickness (3/4T) locations, respectively; and for Scenario No. 2 in Tables 6.1-3 and 6.1-4 of WCAP-18215-NP at the 1/4T and 3/4T locations, respectively. For the material with the highest ART, the Upper Shell to Intermediate Shell Circumferential Weld 10-273 (heat No. W5214), the licensee calculated the chemistry factor (CF) value with no surveillance data considered (Position 1.1 of RG 1.99, Revision 2) and with surveillance data considered (Position 2.1 of RG 1.99, Revision 2). In calculating the CF using Position 2.1 of RG 1.99, Revision 2, the licensee integrated surveillance capsule data for weld heat No. W5214 from HBRSEP's sister plants, Indian Point, Units 2 and 3, and Palisades. As precedent, the licensee referenced the evaluation of surveillance data for weld heat No. W5214 performed by Structural Integrity Associates (Reference 9), which Entergy applied to its pressurized thermal shock (PTS) evaluation of Palisades in 2010. The NRC staff approved this 2010 Palisades PTS evaluation in 2011 (Reference 10). In the 2010 Palisades PTS evaluation, the CF value was calculated using Position 2.1 of RG 1.99, Revision 2, which also integrated surveillance capsule data from sister plants, which for Palisades are HBRSEP and Indian Point, Units 2 and 3.

Finally, the licensee summarized its evaluation of the applicability change of the existing 50 EFPY P/T limits in Table 6.1-5 of WCAP-18215-NP. This table shows only the limiting materials weld, Upper Shell to Intermediate Shell Circumferential Weld 10-273 (heat No. W5214), and limiting plate, Upper Shell Plate W10201-1. Under Scenario No. 2, the licensee determined that the 50 EFPY ART value for the Upper Shell to Intermediate Shell Circumferential Weld 10-273 (heat No. W5214) does not exceed the ART for the existing HBRSEP 50 EFPY P/T limit. For the Upper Shell Plate W10201-1 under Scenario No. 2, the licensee determined that the 50 EFPY ART value exceeds the ART for the existing HBRSEP 50 EFPY P/T limit; and that at 46.3 EFPY, the ART value, and therefore the P/T limit curves, is equivalent to the existing HBRSEP 50 EFPY P/T limits. Therefore, the licensee concluded that the existing HBRSEP 50 EFPY P/T limits are valid through 46.3 EFPY.

3.3 NRC Staff Evaluation

The NRC staff reviewed the licensee's technical basis in WCAP-18215-NP for the proposed change of P/T limit applicability and provides its findings below for each element of the technical basis.

3.3.1 Fluence Calculations

The NRC staff reviewed the fluence calculations for Scenario No. 2 in Section 2 of WCAP-18215-NP and found the fluence calculations acceptable for the following reasons:

- A neutron fluence calculational methodology consistent with that approved by the NRC staff in WCAP-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Reference 11) was used to perform the analysis. The NRC staff notes that removal of the PLSAs for post-Cycle 31 fluence calculations (Scenario No. 2) results in a less complex fluence analysis due to removal of the axial asymmetry introduced by differing material regions within the PLSAs. This allows for simplified neutron source modeling, in that only one set of neutron flux synthesis calculations is needed instead of two to determine RPV fluence. The combination of fluence using two sets of synthesis calculations is discussed in detail in WCAP-18100-NP.

- No new beltline materials are identified with the updated fluence analysis based on Scenario No. 2; therefore no additional fluence calculational methodology uncertainty analysis considerations are warranted.

- The limiting beltline components remain the same with the updated fluence analysis based on Scenario No. 2.

3.3.2 Confirmation of Materials with Limiting Adjusted Reference Temperature

The NRC staff confirmed that the RPV materials with limiting ART in Tables 6.1-1 through 6.1-4 of WCAP-18215-NP are the same RPV materials with limiting ART determined for the 50 EFPY P/T limits in WCAP-15827. The RPV materials with limiting ART are the Upper Shell to Intermediate Shell Circumferential Weld 10-273 (heat No. W5214) and Upper Shell Plate W10201-1.

3.3.3 Surveillance Data Set for Chemistry Factor Calculation

Calculating ART consists of computing a margin term (M) and a shift in nil-ductility reference temperature (ΔRT_{NDT}) and adding them to the initial nil-ductility reference temperature. RG 1.99, Revision 2 provides guidance in calculating M and ΔRT_{NDT}. Calculating ΔRT_{NDT} requires calculating the CF either by Position 1.1 or Position 2.1 of RG 1.99, Revision 2.

For the material with the highest ART, the Upper Shell to Intermediate Shell Circumferential Weld 10-273 (heat No. W5214), the licensee calculated the CF both by Position 1.1 and Position 2.1 of RG 1.99, Revision 2. The NRC staff approved the licensee's calculation of CF using Position 1.1 of RG 1.99, Revision 2 in its SE of the HBRSEP 50 EFPY P/T limits (Reference 12). The licensee's calculation of CF for the limiting material using Position 2.1 of RG 1.99, Revision 2, relied on the method of CF calculation in the 2010 Palisades PTS evaluation, which used surveillance data of sister plants for weld heat No. W5214. The licensee referenced the 2016 Westinghouse evaluation (attachment to Reference 8) or details of its CF calculation. This 2016 Westinghouse evaluation also used surveillance data of sister plants to calculate CF. The NRC staff noted that the surveillance capsule data set for weld heat No. W5214 in the 2016 Westinghouse evaluation and in the 2010 Palisades PTS evaluation are identical, and that in both evaluations, the CF value calculated from capsule data of all sister plants were adjusted to the subject plant's capsule data (i.e., adjusted to HBRSEP's capsule data in the 2016 Westinghouse evaluation and to Palisades' capsule data in the 2010 Palisades PTS evaluation). The adjustment of surveillance capsule data is consistent with guidance from

the 1998 RPV integrity workshop (Reference 13), which recommended that surveillance capsule ΔRT_{NDT} be adjusted for the subject plant's CF (calculated using Position 1.1 of RG 1.99, Revision 2) and mean operating temperature. The use of this adjustment approach was found acceptable in the SE for the 2010 Palisades PTS evaluation (Reference 10). The NRC staff finds the licensee's integration of sister plant surveillance capsule data to calculate CF acceptable and evaluates the scatter in the data in the next section.

3.3.4 Scatter in Surveillance Capsule ΔRT_{NDT}

RG 1.99, Revision 2 specifies 5 criteria in evaluating the credibility of surveillance capsule data. WCAP-15827, Section 2, summarizes the credibility evaluation for the HBRSEP surveillance capsules for weld heat No. W5214. WCAP-15827, Section 2, states in part:

> Surveillance weld metal (Heat W5214) from Robinson only was determined to be not credible. Note that there exists surveillance data of the same heat from Indian Point Units 2 and 3. If this data were integrated with Robinson, the surveillance weld data would then be credible.

The NRC staff noted that the licensee integrated surveillance capsule data not only from Indian Point, Units 2 and 3, but also from Palisades, and that RG 1.99, Revision 2 states that the standard deviation (or margin term) on ΔRT_{NDT} (σ_Δ) may be cut in half when surveillance capsule data is used in determining ART. However, instead of deeming the integrated surveillance data fully credible and applying half of σ_Δ, the licensee deemed the integrated surveillance data not fully credible and applied the full σ_Δ of 28 °F in the 2016 Westinghouse evaluation, consistent with the approach in the 2010 Palisades PTS evaluation. The NRC staff further noted that in the 2010 Palisades PTS evaluation, a detailed evaluation of the scatter in the surveillance capsule ΔRT_{NDT} was performed. Table 7, "Scatter in Fit to all Surveillance Capsule Results Containing Weld Heat No. W5214," of the 2010 Palisades PTS evaluation showed the calculated scatter of ΔRT_{NDT} of the surveillance capsules (in the column labeled "Adjusted – Predicted") and noted that the scatter of ΔRT_{NDT} of four surveillance capsules exceeded σ_Δ of 28 °F, but that the scatter of all surveillance capsules was below $2\sigma_\Delta$ of 56 °F. The NRC staff found this scatter in ΔRT_{NDT} acceptable in its SE of the 2010 Palisades PTS evaluation (Reference 10), citing the following statement in Criterion 3 of RG 1.99, Revision 2 for its acceptability: "Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values." The NRC staff interpreted "twice those values" as $2\sigma_\Delta$.

In reviewing the HBRSEP application, the NRC staff calculated the scatter in the surveillance capsule ΔRT_{NDT} in the 2016 Westinghouse evaluation with adjustments for HBRSEP's CF (using Position 1.1 of RG 1.99, Revision 2) and mean operating temperature, consistent with the methodology in the 2010 Palisades PTS evaluation. The NRC staff determined that the scatter of two surveillance capsules exceeded σ_Δ of 28 °F, but the scatter of all surveillance capsules was below $2\sigma_\Delta$ of 56 °F. Since the scatter is similar to 2010 Palisades PTS evaluation and below two standard deviations, the NRC staff finds the scatter in the surveillance capsule ΔRT_{NDT} in the 2016 Westinghouse evaluation acceptable.

3.3.5 Chemistry Factor and Adjusted Reference Temperature Calculations and Change in Pressure and Temperature Limit Applicability

The NRC staff verified the licensee's calculation of CF in the 2016 Westinghouse evaluation for weld heat No. W5214, using Position 2.1 of RG 1.99, Revision 2, and the licensee's calculations

of ART in Tables 6.1-1 through 6.1-4 of WCAP-18215-NP. The NRC staff finds the calculations acceptable.

To verify the licensee's conclusion that the existing HBRSEP 50 EFPY P/T limits remain valid through 46.3 EFPY, the NRC staff determined the trend in ART as a function of EFPY for Scenario No. 2 for the limiting materials in Table 6.1-5 of WCAP-18215-NP. The NRC staff used the fluence values at various EFPYs provided in Section 2 of WCAP-18215-NP, and based on these fluence values calculated ART. The NRC staff determined that ART varies linearly with EFPY and from this linear relation, verified the licensee's conclusion that at 46.3 EFPY for Scenario No. 2 for Upper Shell Plate W10201-1, the ART value, and therefore the P/T limit curves, is equivalent to the existing HBRSEP 50 EFPY P/T limits. The NRC staff also verified that because of the lower CF for weld heat No. W5214 calculated using Position 2.1 of RG 1.99, the ART value for this weld heat does not exceed the ART value used in the existing 50 EFPY P/T limits. The NRC staff, therefore, finds the licensee's conclusion that the existing HBRSEP 50 EFPY P/T limits remain valid through 46.3 EFPY acceptable.

3.3.6 Pressure and Temperature Limits for Reactor Pressure Vessel Inlet and Outlet Nozzles

The NRC staff determined in its SE of the HBRSEP 50 EFPY P/T limits that the HBRSEP inlet and outlet nozzles were bounded by the RPV P/T limits. With the new fluence values for Scenario No. 2 for the current LAR, the NRC staff verified that the P/T limits for the HBRSEP inlet and outlet nozzles are still bounded by the RPV P/T limits.

3.4 Technical Evaluation Conclusion

The NRC staff reviewed the licensee's LAR to change the applicability of the HBRSEP P/T limits from 50 EFPY to 46.3 EFPY. Based on the evaluation in Section 3.3 of this SE, the NRC staff concludes that the licensee's proposed change meets the fracture toughness requirements of Appendix G to 10 CFR Part 50. The NRC staff also concludes that the licensee's proposed change complies with the acceptance criteria in Section 2.0 above. Therefore, the NRC staff finds the incorporation of the licensee's proposed change to the HBRSEP TS 3.4.3 acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment on July 18, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on April 10, 2018 (83 FR 15415). Accordingly, the amendment meets 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 amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Kapopoulos, E. J., Duke Energy Progress, LLC, letter to the U.S. Nuclear Regulatory Commission, "License Amendment Request Proposing to Revise Technical Specification 3.4.3, 'RCS Pressure and Temperature (P/T) Limits'," February 7, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18038B289).

2. Kapopoulos, E. J., Duke Energy Progress, LLC, letter to the U.S. Nuclear Regulatory Commission, "Submittal of Updated Final Safety Analysis Report, Revision No. 27," September 25, 2017 (ADAMS Accession No. ML17298A847).

3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001 (ADAMS Accession No. ML010890301).

4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988 (ADAMS Accession No. ML003740284).

5. U.S. Nuclear Regulatory Commission, Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," October 14, 2014 (ADAMS Accession No. ML14149A165).

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Date: August 16, 2018

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENT NO. 260 REGARDING REQUEST TO REVISE TECHNICAL SPECIFICATION REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS TO REFLECT 24-MONTH FUEL CYCLES (EPID L-2017-LLA-0033) DATED August 16, 2018

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