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

November 22, 2016

Mr. R. Michael Glover
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 TO REVISE REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS APPLICABLE FOR 50 EFFECTIVE FULL POWER YEARS (CAC NO. MF7048)

Dear Mr. Glover:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 248 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). This amendment changes the HBRSEP2 Technical Specifications (TSs) in response to your application dated November 2, 2015, as supplemented by letters dated December 22, 2015; and March 31, May 9, and September 14, 2016. The amendment revises the reactor coolant system (RCS) pressure and temperature limits by replacing TS Section 3.4.3, "RCS Pressure and Temperature (P/T) Limits," Figures 3.4.3-1 and 3.4.3-2, with figures that are applicable up to 50 effective full power years.

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. 248 to DPR-23
2. Safety Evaluation

cc w/enclosures: Distribution via 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. 248
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 November 2, 2015, as supplemented by letters dated December 22, 2015; and March 31, May 9, and September 14, 2016, 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. 248 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 the date of its issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeanne A. Dion, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-23
and the Technical Specifications

Date of Issuance: November 22, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 248

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. 248 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 50 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 50 EFPY
Heatup Curves include +20°F and -80 psig Allowance for Instrumentation error.

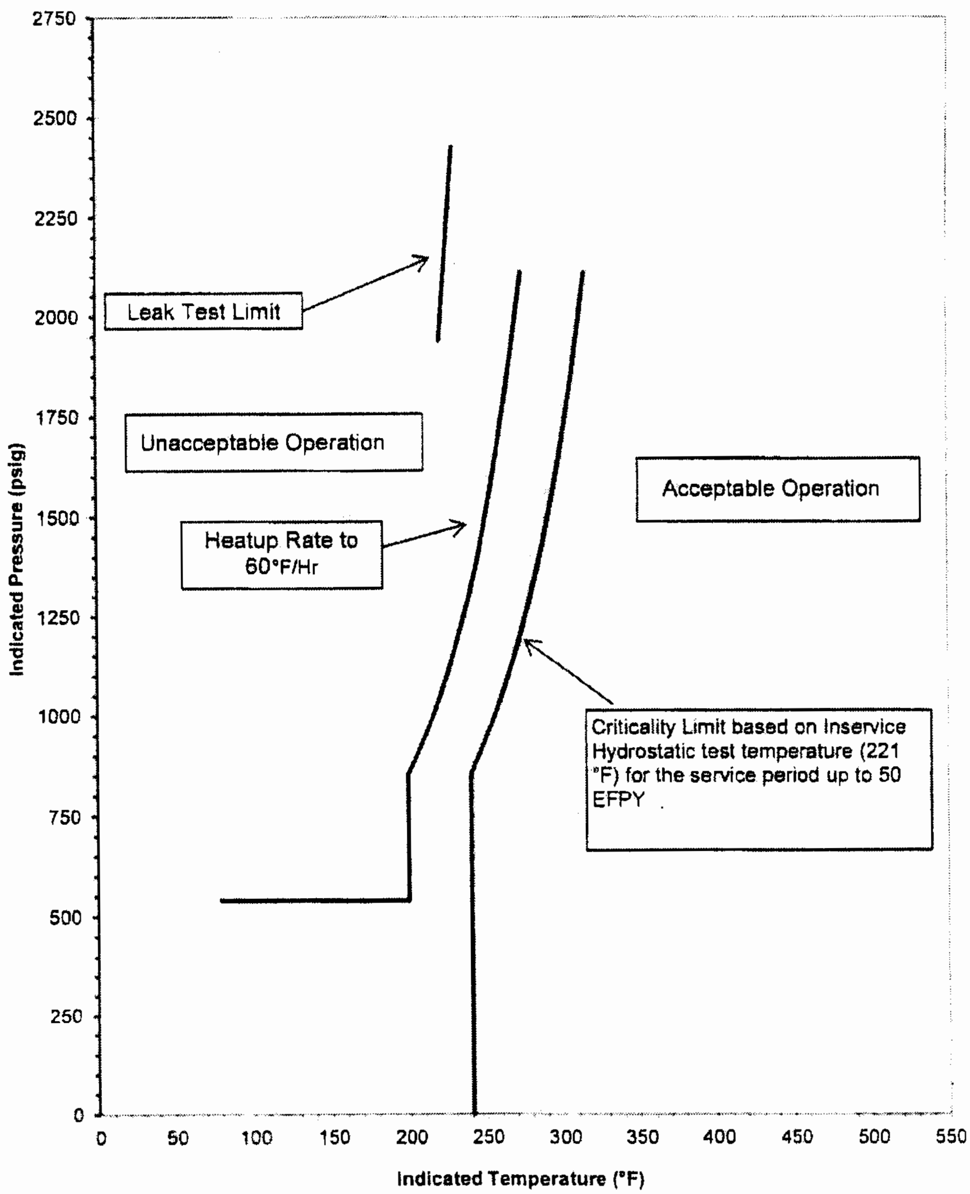

Figure 3.4.3-1
Reactor Coolant System Heatup Limits
Applicable Up to 50 EFPY

MATERIALS PROPERTIES BASE
Controlling Material. Upper Shell Plate W10201-1 & Girth Weld 10-273
Limiting ART Values at 50 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 50 EFPY.
Curves include +20°F and -80 PSIG Allowance for Instrumentation error.

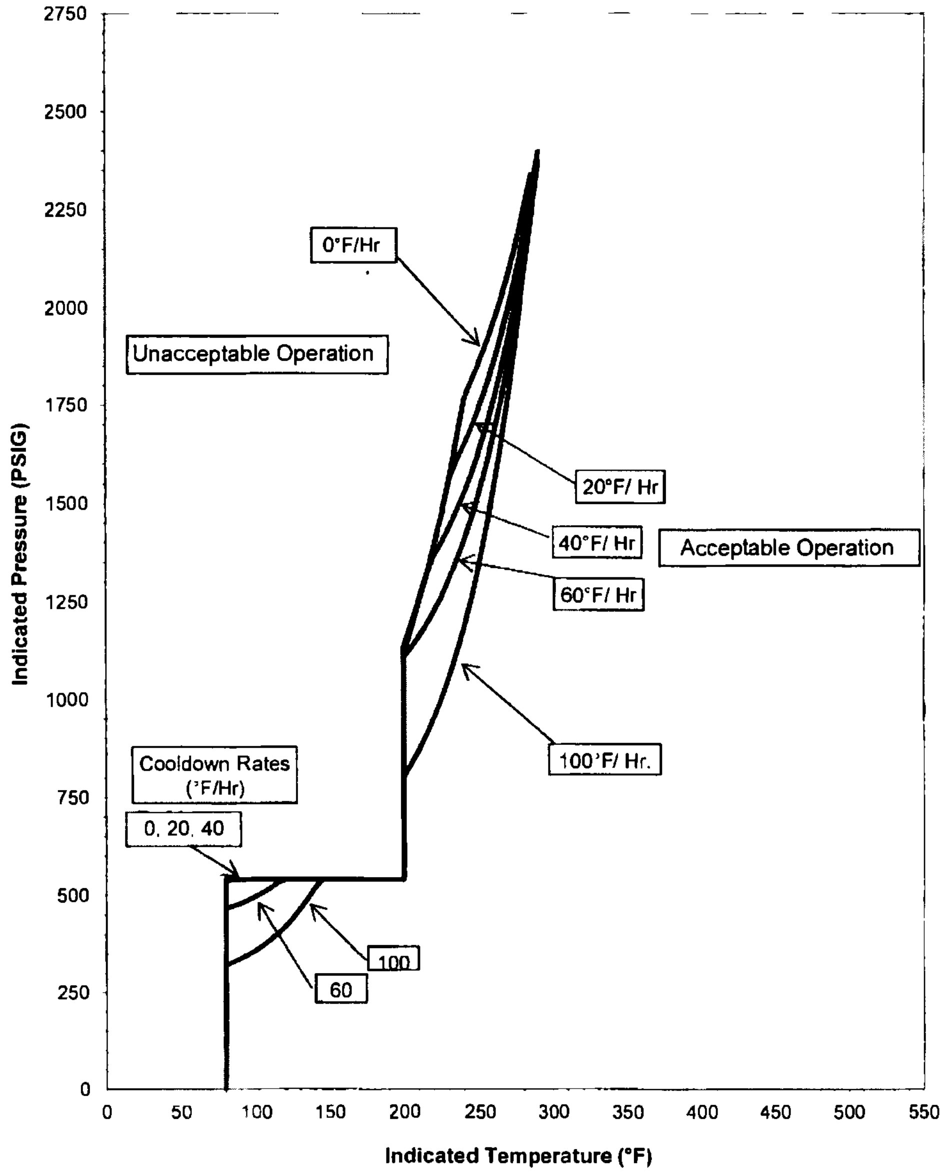

Figure 3.4.3-2
Reactor Coolant System Cooldown Limitations
Applicable Up to 50 EFPY

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 248 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 November 2, 2015 (Reference 1), as supplemented by letters dated December 22, 2015 (Reference 2); March 31, 2016 (Reference 3); May 9, 2016 (Reference 4); and September 14, 2016 (Reference 5), Duke Energy Progress, LLC, the licensee (previously operating as Duke Energy Progress, Inc.), submitted a license amendment request (LAR) for H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). The LAR proposed to revise Technical Specification (TS) Section 3.4.3, "RCS [Reactor Coolant System] Pressure and Temperature (P/T) [or P-T] Limits," Figures 3.4.3-1 and 3.4.3-2, with figures that are applicable up to 50 effective full power years (EFPYs).

The 50 EFPY P-T limits are based on the P-T limit curves developed in Westinghouse report, WCAP-15827, Revision 0, "H. B. Robinson Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," which was included as Attachment 4 to the LAR (Reference 1). The P-T limits were determined using the U.S. Nuclear Regulatory Commission (NRC)-approved methodology documented in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Reference 6). WCAP-15805, "Analysis of Capsule X From the Carolina Power and Light Company H.B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program" (Reference 7), documents dosimetry information from in-vessel surveillance capsules considered in the P-T limit calculations. WCAP-18100-NP, Revision 0, "Ex-Vessel Neutron Dosimetry Program for H. B. Robinson Unit 2 Cycles 16 through 29," documents ex-vessel dosimetry information used to confirm the P-T limit calculations. WCAP-18100-NP was included as an enclosure to the licensee's submittal dated May 9, 2015 (Reference 4).

As described in the licensee's December 22, 2015, letter (Reference 2), the NRC staff identified three acceptance review issues that needed supplemental information to support a detailed technical review. These issues are further discussed in Section 3.1.2 of this safety evaluation. The December 22, 2015, letter provided additional clarification and justification of approaches used in the LAR but did not change the scope of the LAR. The NRC staff subsequently determined that the LAR as supplemented was sufficient to support a technical review.

The supplements dated March 31, May 9, and September 14, 2016, 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 as published in the *Federal Register* on March 1, 2016 (81 FR 10678).[1]

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 evaluates 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 2010 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 2010 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 transition] 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 reactor vessel.

Appendix H to 10 CFR Part 50 establishes requirements for each facility related to its RPV material surveillance program.

---

1 The March 1, 2016, *Federal Register* notice (81 FR 10678) referenced the November 2, 2015, amendment request as supplemented by letter dated December 22, 2015.

Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" (Reference 8), contains guidance on methodologies the NRC considers acceptable for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation.

Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)" (Reference 9), requested that licensees submit the RPV data for their plants to the NRC for review. Supplement 1 to GL 92-01, Revision 1 (Reference 10), requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock" (Reference 11) describes acceptance criteria for determining the P-T limits for ferritic materials in the beltline of the RPV based on Appendix G to Section XI of the ASME Code methodology.

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001 (Reference 12), 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."

HBRSEP2 received its construction permit in 1967 and was licensed for operation in July 1970. On July 11, 1967, the Atomic Energy Commission published for public comment in the *Federal Register* (32 FR 10213), a revised and expanded set of 70 draft GDC (hereinafter referred to as the "draft GDC"). On February 20, 1971, the Atomic Energy Commission 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 (Reference 13), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the final GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the final GDC were formally adopted, was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission.

Based on a review of the HBRSEP2 Updated Final Safety Analysis Report (UFSAR), Section 3.1, "Conformance with General Design Criteria"; SRP Section 5.3.2; RG 1.190; and the licensee's application (Reference 1), the NRC staff identified the following draft GDC as being applicable to the proposed amendment:

In the HBRSEP2 UFSAR Section 3.1.2.9, "Reactor Coolant Pressure Boundary" (GDC 9), HBRSEP2 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 HBRSEP2 UFSAR Section 3.1.2.34, "RCPB Rapid Propagation Failure Prevention" (GDC 34), HBRSEP2 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 Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," issued October 14, 2014 (Reference 14), clarifies that P-T limit calculations for ferritic RPV materials other than those materials with the highest reference temperature may define P-T curves that are more limiting because the consideration of stress levels from structural discontinuities (such as RPV inlet and outlet nozzles) may produce a lower allowable pressure. RIS 2014-11 also clarifies that the beltline definition in 10 CFR Part 50, Appendix G, is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1 x 10^{17} neutrons per square centimeter (n/cm$^{2)}$ (E > 1 MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period.

3.0 TECHNICAL EVALUATION

Determination of P-T limit curves involves three basic steps: (1) calculation of neutron fluence projections for a particular EFPY value, (2) determination of the adjusted reference temperatures (ART) based on these fluence projections, and (3) determination of the P-T limit curves based on the updated ART values. Then, the effect of the updated P-T limit curves on the low temperature overpressure protection (LTOP) settings is evaluated.

3.1 Reactor Vessel Fluence

The NRC staff reviewed the LAR and LAR Attachment 4, WCAP-15827, Revision 0, (Reference 1). With respect to the neutron fluence calculational method, WCAP-15827 references WCAP-15805 (Reference 7). WCAP-15805 documents the results of calculated and measured dosimetry activity from all previous HBRSEP2 surveillance capsules in fulfillment of

the 10 CFR Part 50, Appendix H surveillance reporting requirement. In WCAP-15805, four capsule dosimetry analyses were performed specific to HBRSEP2 at the end of Cycles 1, 3, 8, and 20 (Capsules S, T, V, and X, respectively) to demonstrate the continued validity of the HBRSEP2 RPV fluence calculational methodology, WCAP-14040-NP-A, Revision 2 (Reference 6), which was previously found to adhere to RG 1.190 (Reference 12). This fluence calculational methodology was used to perform fluence projections through 50 EFPY for beltline RPV components calculated to exceed 1 x 10^{17} n/cm^2. The licensee also uses the least squares adjustment methodology of the FERRET Code (Reference 15) in WCAP- 15805 (dated March 2002) (Reference 7) to demonstrate compliance with RG 1.190. The use of this same least squares adjustment methodology was later approved by the NRC staff, as documented in WCAP-16083-NP-A, Revision 0, “Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,” dated May 2006 (Reference 16). The fluence projections are required input when using RG 1.99 (Reference 8) to satisfy applicable NRC regulations.

3.1.1 Methodologies Used

The NRC-approved generic Westinghouse RPV fluence methodology topical report WCAP-14040-NP-A, Revision 2, indicates that calculational variation less than 20 percent is expected when the WCAP-14040-NP-A fluence calculational methodology is used to determine pressurized-water reactor (PWR) RPV fluence calculations. The expected 1-sigma uncertainty in the Westinghouse method is an equally-weighted combination (i.e., Westinghouse uses square-root-sum-of-squares (SRSS)) of analytic uncertainty (11 percent[2]), benchmark comparison relative differences (~4 percent when combined), and “other factors” (5 percent) -- the estimated 1-sigma uncertainty is therefore 13 percent. When dosimeter activity measurement uncertainty is combined with the Westinghouse method uncertainty through SRSS, the maximum “expected” 1-sigma uncertainty for all benchmark sets is estimated to be approximately 14 percent, independent of surveillance capsule and dosimeter type.

WCAP-14040-NP-A, Section 2.2.2, “Determination of Best Estimate Pressure Vessel Exposure,” describes how fluence values determined using the RG 1.190-adherent method are used to generate P-T curves for a particular EFPY of plant operation. The beginning of Section 2.2.2 indicates that the best-estimate fast neutron exposure at the location of interest is determined by multiplying calculated fluence values by a plant-specific measured-to-calculated (M/C) bias factor derived from all available surveillance capsule and reactor cavity dosimetry data. WCAP-14040-NP-A states, in part: “In some cases the fluence at the EFPY of interest is obtained directly from the dosimetry analysis. However, if the fluence is not available from the dosimetry analysis, the peak vessel inner radius fluence at the EFPY of interest is calculated [based on Equation 2.2-2].” Since fluence is available from dosimetry analyses, this indicates that the best-estimate fluence values used for P-T curve generation should be divided by the overall average calculated-to-measured (C/M) ratio of 0.87 determined in WCAP-15805, Section 6.3.3, “Comparisons of Measurements and Calculations,” for the entire set of HBRSEP2 data. However, for consistency with RG 1.190, in LAR Section 1.2, “Basis for Proposed

[2] SRSS of the following uncertainty components: internals dimensions, vessel inner radius, water temperature, peripheral assembly source strength, axial power distribution, peripheral assembly burnup, and spatial distribution of the source.

Change," the licensee takes exception to the portion of the WCAP-14040-NP-A methodology in Section 2.2 that prescribes this bias correction.

Regarding the evaluation of dosimetry data, which is the subject of WCAP-15805, RG 1.190, Section 1.4.2, "Comparisons with Benchmark Measurements and Calculations," states, in part:

> Differences between measurements and calculations should be consistent with the combined uncertainty estimates for the measurements and calculations. (Note that the uncertainties in both the calculations and measurements will contribute to the observed measurement-to-calculation differences.) The calculated reaction rates (using the methods described in Regulatory Positions 1.1 through 1.3) typically agree with the measurements to within about 20% for in-vessel surveillance capsules and 30% for cavity dosimetry. Deviations greater than these values must be investigated and, when the cause of the deviation is determined to be an error in the calculation, the calculations must be modified.

RG 1.190 indicates that the combined uncertainty estimates for the measurements and calculations for the various benchmarks should be checked. If there are C/M values that fall outside of the 0.8-1.2 range, it should be verified that the 1-sigma standard deviation of the C/M differences (including any average C/M bias) are still within the combined uncertainty estimate and corrected as necessary; values of C/M outside of the expected range may indicate that the calculation is biased. If C/M values are outside of the 0.8-1.2 range, but the 1-sigma standard variation about the average is within the methodology uncertainty range, correction of calculated results should also be considered as implied by RG 1.190.[3] Equation 6 on page 20 of RG 1.190 shows how calculated values are to be adjusted if adjustment is found to be appropriate. Footnote 11, on the same page as Equation 6, provides further discussion on the adjustment of calculated values showing how the various uncertainty components can be weighted.

In the least squares adjustment methodology in WCAP-16083-NP-A, Revision 0, adjustments are made to measured reaction rates (i.e., through the neutron spectrum calculation and dosimetry and transport cross-section adjustments). In this process, changes within the uncertainties of these parameters can be made, and are consistently applied to all measurements in order to minimize the C/M bias consistent with the measured reaction rate data and its associated uncertainty. The corresponding NRC staff safety evaluation report mentions in its limitation that the "[least squares approach] is acceptable if the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry-measured reaction rates, and the dosimetry cross sections." Therefore, in the context of the least squares approach used in WCAP-16083-NP-A, the expectation is that the corresponding best-estimate-to-calculated (BE/C) values will be within 14 percent of the unadjusted M/C values on average.

---

[3] A calculational bias may still be present given very precise calculations and measurements.

3.1.2 Acceptance Review Issues

The NRC staff performed an initial review and confirmatory calculations to determine if the LAR provided sufficient information to perform a complete review. The NRC staff identified three issues that required additional clarification.

(1) The LAR initially appeared to be inconsistent with the WCAP-16083-NP-A methodology limitation described in the corresponding NRC staff safety evaluation. That is application of the WCAP-16083-NP-A least squares adjustment methodology in WCAP-15805 did not appear to meet the application that the adjustments to the calculated spectra are relatively small and well within the assigned uncertainties for the calculated spectra, measured sensor reaction rates, and dosimetry reaction cross-sections.

(2) There is a potential bias in the C/M dosimetry data that was not discussed in WCAP-15805. A bias in C/M dosimetry data may require bias correction as described in RG 1.190, Section 1.4.3, "Estimate of Fluence Calculational Bias and Uncertainty."

(3) The exclusion of all cobalt dosimeter data from the unadjusted C/M summary table provided in Table 6-11 of WCAP-15805 is not explained.

By letter dated December 22, 2015, the licensee supplemented its LAR dispositioning the NRC staff's concerns (Reference 2).

Regarding the first issue, the licensee showed that the least squares adjustment procedure produces an average calculated-to-best-estimate (C/BE) value of 0.89 with a standard deviation of approximately 8 percent, which shows that the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry-measured reaction rates, and the dosimetry cross sections as required by the WCAP-16083-NP-A limitation. Based on the explanations in the supplement dated December 22, 2015, the NRC was able to independently confirm the values provided by the licensee. Therefore, the licensee may apply the WCAP-16083-NP-A methodology in the HBRSEP2 fluence calculations and the first issue is resolved.

Regarding the second issue, RG 1.190 implies that 68 percent of the C/BE data should fall between 0.8 and 1.2 (i.e., consistent with a 1-sigma standard deviation about 1). It is seen that 67 percent of the raw data falls between 0.8 and 1.2. Based on this metric, there isn't strong evidence that the data is not consistent with the RG 1.190 fluence calculational method uncertainty allowance, however a bias is visually indicated when the C/BE data is plotted. Consequently, for HBRSEP2, analysis of the raw C/M data indicates that there may be a calculational bias[4], and RG 1.190 would allow for adjustment of the calculated fluence values under certain conditions since the fluence uncertainty analysis derived in WCAP-14040-NP-A is

---

[4] Especially considering the excellent agreement shown in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," dated May 2004, Section 2.2.2, "Validation of the Transport Calculations," (ADAMS Accession No. ML050120209) between M/C values for each of the various fast neutron sensors -- average M/C of 1.03 and standard deviation of 9.8 percent.

consistent with the observed raw 1-sigma C/M variation about the average.[5] American Society for Testing of Materials (ASTM) Standard E482-11e1, "Standard Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance" (ASTM E482), provides a bias existence indicator by taking the log-mean of the C/M ratios. ASTM E482, Section 3.2.2.7, notes that a non-zero log-mean of the C/M ratios indicates that a bias exists, and that possible sources of a bias are: (1) source normalization, (2) neutronic data, (3) transverse leakage corrections (if applicable), (4) geometric modeling, and (5) mathematical approximations.

To address the second issue, in the supplement dated December 22, 2015, the licensee demonstrated that the fluence calculational methodology meets RG 1.190 requirements using unadjusted activity measurement values instead of the WCAP-16083-NP-A-based adjusted activity measurement values. The response references Table 4-1, "Data Based Comparison for 104 In-Vessel Dosimetry Sets from 29 Reactors," from WCAP-16083-NP-A, Section 4.3, "Operating Power Reactor Comparisons." Regarding Table 4-1, the licensee noted that although some HBRSEP2 capsules deviate further from the mean than most capsules from other plants, this is expected and within the prescribed shape of the underlying distribution of Westinghouse's more extensive fluence calculational methodology validation database. However, the NRC staff noted that only data from Capsules V and T were discussed in the context of the observed distribution of BE/C values versus the expected distribution of BE/C values -- that is, data from Capsules S and X were not included in the discussion. The NRC staff also noted that the licensee's characterization of the observed distribution of the Capsule V and T data was inaccurate. Specifically, regarding where the extremes of the BE/C values lie in the observed distribution of the Capsule V and T data, the licensee states that the data falls into the +2-sigma category. However, the NRC staff noted several values that fall into the +3-sigma category.

Despite the apparent HBRSEP2-specific fluence method bias resulting from the evaluation of HBRSEP2 plant-specific dosimetry, application of the design basis fluence calculational method, described in WCAP-14040-NP-A, results in an associated bias and uncertainty that is consistent with the RG 1.190 uncertainty allowance of 20 percent when determining fluence values used to develop P-T curves for HBRSEP2. Furthermore, Westinghouse's discussion in WCAP-16083-NP-A, Section 4.3 indicates that inclusion of the HBRSEP2 surveillance data with Westinghouse's overall validation database, which is summarized in Table 4-1, would result in an associated bias and uncertainty that is generally consistent with the RG 1.190 uncertainty allowance of 20 percent when determining best-estimate fluence values, and bias correction is not necessarily warranted for HBRSEP2. Therefore, the NRC staff accepts that a bias correction does need to be included in the fluence calculation and the second issue is resolved.

[5] RG 1.190 cautions against applying bias corrections when the uncertainty in the bias is not substantially less than the bias itself. RG 1.190, Footnote 8 states: "The adequacy of the measurement data base for determining a bias in the calculations depends on the magnitude of the bias and is therefore problem dependent. For example, if the bias is small an accurate estimate of the bias will require either (1) a large number of reasonably accurate measurements or (2) a small number of very accurate measurements. Specifically, the uncertainty in the bias should be substantially less than the bias itself. For example, if the calculated fluence is to be increased by 10%, the uncertainty in this increase should be substantially less than 10%."

Regarding the third issue, the NRC staff noted that in WCAP-15805, while the 56 dosimetry activity measurements are identified, the unadjusted C/M summary table provided in Table 6-11 of WCAP-15805 excludes some 20 measurements based on the cobalt reaction product reaction rate from the reported benchmarking effort, thus reducing the total number of measurements for comparison to 36. WCAP-15805 does not explain why the cobalt data was excluded. This is important because the note to this table states that "the overall average C/M ratio for the set of sensor measurements is 0.87 with an associated standard deviation of 11.4%," and this conclusion is used as part of the basis for meeting RG 1.190 in WCAP-15805, Section 6.3.3 (i.e., in the case of direct comparison of measured and calculated sensor reaction rates).

To address the third issue, the licensee explained why the cobalt dosimeter data was removed from the fluence calculational methodology validation based on unadjusted measurements. Since the cobalt reaction cross-section of interest is primarily responsive to thermal and epithermal (i.e., not fast) neutrons, it does not provide meaningful validation for fast fluence calculations. The NRC staff finds this justification acceptable since 90 percent of the response range is still covered without consideration of the cobalt dosimeter data and the basis for removal allows for more relevant application-specific benchmarking.

3.1.3 Reactor Vessel Fluence Review

By e-mail dated March 30, 2016 (Reference 17), the NRC staff requested additional information regarding the accounting of operational history effects in the neutron fluence projections out to the period of extended operation. By letter dated March 31, 2016 (Reference 3), the licensee responded by stating that more recently, ex-vessel neutron dosimetry was evaluated for HBRSEP2 as documented in WCAP-18100-NP, Revision 0, "Ex-Vessel Neutron Dosimetry Program for H.B. Robinson Unit 2 Cycles 16 through 29." By letter dated May 9 2016 (Reference 4), the licensee submitted WCAP-18100-NP. The dosimetry documented in WCAP-18100-NP was pulled after Cycle 29 and corresponds to 33.18 EFPY. The results of the updated 50 EFPY peak neutron fluence projections in WCAP-18100-NP were calculated, using the WCAP-16083-NP-A fluence methodology as approved, to be approximately 5 percent lower than those previously projected based on the Capsule X analysis documented in WCAP-15805. Furthermore, the licensee explained that there are no core and/or operational design changes planned for HBRSEP2 for the period between 33.18 EFPY and 35 EFPY that would impact the fluence projections to 50 EFPY. Therefore, the licensee has addressed the NRC staff concerns about accounting for operational history during the period of extended operation.

3.1.4 Reactor Vessel Fluence Conclusion

Based on the LAR, the supplements, and the discussion above, the NRC staff finds that:

(1) Based on the demonstrated consistency with the RG 1.190 uncertainty allowance of 20 percent, the use of the fluence methods described in both WCAP-14040-NP-A, Revision 2, and WCAP-16083-NP-A, Revision 0, were applied appropriately for determining best-estimate fluence values for HBRSEP2.

(2) The licensee has performed appropriate plant-specific qualification activities necessary to verify the validity of the calculational fluence methods used for determining

best-estimate fluence values used in the development of the 50 EFPY HBRSEP2 P-T curves.

(3) The projections based on the Capsule X dosimetry evaluation in WCAP-15805 are acceptable because they bound the most recent ex-vessel dosimetry evaluation documented in WCAP-18100-NP, Revision 0 using the WCAP-16083-NP-A, Revision 0 calculational fluence method.

(4) The neutron fluence values used in the determination of the proposed 50 EFPY P-T curves are acceptable because WCAP-18100-NP, Revision 0, RPV component fluence projections account for the most recent operational history effects from 20.39 EFPY to 33.18 EFPY.

(5) The NRC staff has reasonable assurance that the scheduled date for the next capsule evaluation[6], and hence dosimetry evaluation, will be sufficient to allow for verification of the applicability of the currently proposed 50 EFPY fluence projections based on the demonstrated adequacy of past projections and the relatively low sensitivity of these projections to operational history effects for HBRSEP2. If the future calculated peak 50 EFPY fluence projections are found to increase relative to the WCAP-15805 projections using the design basis fluence method after the next scheduled dosimetry measurement evaluation as required by 10 CFR 50, Appendix H, the licensee must verify that the P-T curves remain applicable.

3.2 Vessel and Reactor Coolant Pressure Boundary Integrity

3.2.1 Licensee's Evaluation

Adjusted Reference Temperature Calculations

The licensee submitted ART values and P-T limit curves valid for up to 50 EFPY of plant operation in WCAP-15827 (Reference 1). The licensee identified the limiting material for the HBRSEP2 RPV as upper shell plate W10201-1, fabricated from plate heat A6623-1 and upper to intermediate shell plate circumferential weld 10-273, fabricated from weld heat W5214. The circumferential weld is limiting for only a portion of the cooldown curves. The licensee calculated the ART values for these limiting materials for both the one-quarter of the RPV wall thickness (1/4t) and three-quarters of the RPV wall thickness (3/4t) locations. The key parameters for the licensee's ART determination for the limiting materials are shown in the Table 1 below for HBRSEP2, which are from WCAP-15827 (Reference 1).

[6] Scheduled to occur at 38 EFPY as documented in the safety evaluation report regarding the HBRSEP2 reactor vessel surveillance capsule withdrawal schedule revision enclosed in a letter dated December 21, 2011 (ADAMS Accession No. ML11349A026).

Table 1. Licensee's ART Calculations for 50 EFPY Limiting RPV Materials for HBRSEP2

Applicable Curves	Limiting Material	Location	Initial RT_{NDT} (°F)	Fluence (n/cm^2)	Chemistry Factor (1) (°F)	ΔRT_{NDT} (°F)	Margin (2) (°F)	ART (°F)
Heatup and Part of Cooldown	Upper Shell Plate W10201-1	1/4t	69	1.43×10^{19}	62.9	69.2	34 $\sigma_I = 0$ $\sigma_\Delta = 17$	172
Heatup and Part of Cooldown	Upper Shell Plate W10201-1	3/4t	69	4.68×10^{18}	62.9	49.6	34 $\sigma_I = 0$ $\sigma_\Delta = 17$	153
Remaining Part of Cooldown	Circ. Weld 10-273	1/4t	-56	1.43×10^{19}	230.2	253.2	65.6 $\sigma_I = 17$ $\sigma_\Delta = 28$	263
Remaining Part of Cooldown	Circ. Weld 10-273	3/4t	-56	4.68×10^{18}	230.2	181.4	65.6 $\sigma_I = 17$ $\sigma_\Delta = 28$	191

Notes: (1) Determined from RG 1.99, Revision 2, Regulatory Position 1.1 (Reference 8).
(2) Margin = $2\sqrt{(\sigma_I^2 + \sigma_\Delta^2)}$. This margin term is based on the establishment of initial material property uncertainty (σ_I) and shift in material property uncertainty (σ_Δ) consistent with RG 1.99, Revision 2.

The licensee's ART calculations for the RPV inlet and outlet nozzles are shown in Table 2, which are also from WCAP-15827. The inlet and outlet nozzles are included in the 50 EFPY ART calculations because the fluence levels in the nozzles exceed 1×10^{17} n/cm^2 with energy greater than one million electron volts (E > 1 MeV) as specified in Appendix H to 10 CFR Part 50.

Table 2. Licensee's ART Calculations for 50 EFPY Inlet and Outlet Nozzles for HBRSEP2

Applicable Curves	Limiting Material	Location	Initial RT_{NDT} (°F)	Fluence (n/cm^2)	Chemistry Factor (1) (°F)	ΔRT_{NDT} (°F)	Margin (2) (°F)	ART (°F)
Cooldown	Inlet Nozzle	1/4t	60	2.24×10^{17}	20	3.7	34.2 $\sigma_I = 17$ $\sigma_\Delta = 1.85$	98
Cooldown	Inlet Nozzle	3/4t	60	7.35×10^{16}	20	1.8	34.0 $\sigma_I = 17$ $\sigma_\Delta = 0.89$	96
Cooldown	Outlet Nozzle	1/4t	60	1.45×10^{17}	113	15.8	37.5 $\sigma_I = 17$ $\sigma_\Delta = 7.92$	113
Cooldown	Outlet Nozzle	3/4t	60	4.73×10^{16}	113	7.2	34.8 $\sigma_I = 17$ $\sigma_\Delta = 3.64$	102

P-T Limit Curve Calculations - Vessel

WCAP-15827 documented detailed thermal and fracture mechanics evaluations to establish the proposed HBRSEP2 P-T limit curves for 50 EFPY. The licensee stated that the P-T limit curves

were determined based on the methodology documented in WCAP-14040-NP-A, which is based on the methodology in Appendix G to Section XI of the ASME Code. The numerical representation of the proposed P-T limit curves can be found in Appendix A "PT Curves Without Flange Requirement" to WCAP-15827, Table A9 "50 EFPY Heatup Curve Data Points Using 1996 App. G" and Table A10, "50 EFPY Cooldown Curve Data Points Using 1996 App. G." RPV temperatures at the inner wall, 1/4t, 3/4t, and the outer wall locations for various heatup and cooldown rates can be found in Appendix B, "Vessel Wall (1/4T, 3/4T and T), Temperatures" to WCAP-15827. Based on the temperature distribution across the RPV wall, the material K_{IC}, the thermal stresses, and, subsequently, the applied thermal stress intensity factors (K_{It}) at the tip of the postulated axial and circumferential flaws at the 1/4t and 3/4t locations were computed. Based on these applied K_{IC} and K_{It} values at the crack tips, the corresponding allowable applied pressure stress intensity factors (K_{Ip}) at the tip of the postulated flaw at the 1/4t and 3/4t locations were calculated. From the K_{Ip} values, the allowable pressures at various temperatures were calculated.

P-T Limit Curve Calculations - Nozzles

WCAP-15827 does not contain 50 EFPY P-T limit curves for the HBRSEP2 RPV inlet and outlet nozzles even though the projected fluence values are greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) as shown in Table 2 of this safety evaluation. The NRC issued RIS 2014-11 (Reference 14) as a reminder that Appendix G to 10 CFR 50 specifies fracture toughness requirements for all ferritic materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 (E > 1 MeV) of the RCS pressure boundary. In particular, RIS 2014-11 reminds licensees and applicants that ferritic RPV shell materials with the highest ART may not necessarily define the bounding P-T limit curves. The reason for this is that stress levels from structural discontinuities in other ferritic RPV components, such as inlet and outlet nozzles, may produce more bounding P-T limit curves. Therefore, by e-mail dated March 30, 2016 (Reference 17), the NRC staff requested the licensee in request for additional information (RAI)-3 to provide P-T limit calculations for the HBRSEP2 RPV inlet and outlet nozzles or otherwise demonstrate how the P-T limit curves developed for 50 EFPY in WCAP-15827 bound all ferritic pressure boundary components of the RPV.

By letter dated September 14, 2016 (Reference 5), the licensee responded by including in a non-proprietary supplemental analysis, 50 EFPY P-T limit curves for the HBRSEP2 RPV inlet and outlet nozzles. In this supplemental analysis, the licensee did not use the neutron fluence values from WCAP-15827 (reproduced in Table 2 of this safety evaluation). Instead, the licensee used the neutron fluence values in report WCAP-18100-NP, Revision 0, which reflects the most recent operational effects from 20.39 EFPY to 33.18 EFPY. These neutron fluence values were greater than 1 x 10^{17} n/cm^2 (E > 1 MeV). WCAP-18100-NP, Revision 0 is included in the licensee's response to RAI-2 in the submittal dated May 9, 2016 (Reference 4). In addition, the licensee did not use the initial RT_{NDT} values from WCAP-15827 (reproduced in Table 2 of this safety evaluation), and instead used updated initial RT_{NDT} values for the HBRSEP2 inlet and outlet nozzles, which are based on HBRSEP2 nozzle-specific data. The initial RT_{NDT} values in WCAP-15827, Table 1 "Summary of the Best Estimate Cu [Copper] and Ni [Nickel] Weight Percent and Initial RT_{NDT} Values for the H.B. Robinson Unit 2 Reactor Vessel Materials," for the inlet and outlet nozzles were generic values based on SRP Materials Engineering Branch (MTEB) 5-2 (Reference 18) because HBRSEP2 nozzle-specific data needed to calculate HBRSEP2 nozzle-specific initial RT_{NDT} did not exist at the time

WCAP-15827 was issued. SRP MTEB 5-2 is the former designation of Branch Technical Position (BTP) 5-3 in Chapter 5 "Reactor Coolant System and Connected Systems" of the SRP (Reference 19).

The licensee re-calculated values of ART for the HBRSEP2 inlet and outlet nozzles based on the updated fluence and initial RT_{NDT} values of the HBRSEP2 inlet and outlet nozzles discussed above. Finally, the licensee calculated the 50 EFPY P-T limit curves for the HBRSEP2 inlet and outlet nozzles based on the updated ART values and a postulated inside-surface nozzle corner flaw and compared them with the HBRSEP2 50 EFPY P-T limit curves for cooldown included in the licensee's LAR dated November 2, 2015 (Reference 1). The comparison showed that the 50 EFPY P-T limit curves for the HBRSEP2 inlet and outlet nozzles are bounded by the HBRSEP2 50 EFPY P-T limit curves for cooldown. Only the cooldown P-T limit curves need to be compared because the HBRSEP2 inlet and outlet nozzle P-T limit curves are based on fracture mechanics analysis of a postulated flaw at the blend radius on the inside surface of the nozzle, which receives the highest stresses during a cooldown.

Because updated neutron fluence values for the HBRSEP2 RPV and RPV welds were also reported in WCAP-18100-NP, the licensee performed confirmatory calculations to demonstrate that the HBRSEP2 50 EFPY P-T limit curves in the November 2, 2015 LAR remain bounding.

Low Temperature Overpressure Protection Settings

The licensee stated that no changes to the LTOP settings are required as a result of the revision of the P-T limit curves from 35 EFPY to 50 EFPY.

3.2.2 NRC Staff's Evaluation

The NRC staff's evaluation of the ART and P-T limit curve calculations, and the LTOP settings is provided in the following paragraphs. The NRC staff verified in its confirmatory calculations that the licensee has correctly implemented the following two exceptions to WCAP-14040-NP-A, Revision 2 (Reference 6) related to the scope of this review that is indicated in Section 1.2 of the LAR (Reference 1): (1) that K_{IC} is used instead of K_{Ia} and (2) that Appendix G to the 1996 Edition in lieu of the 1989 Edition of the ASME Code was used in generating the P-T limit curves.

Adjusted Reference Temperature Calculations

Using the methodology in RG 1.99, Revision 2 (Reference 8), the NRC staff independently calculated 50 EFPY ART values for several RPV materials, including all the materials shown in Table 1 of this SE, and confirmed that the limiting materials are the Upper Shell Plate W10201-1 and Circumferential Weld Seam 10-273. The NRC staff's confirmatory calculations included the intermediate terms used in the calculation of ART, such as the fluence values at the 1/4t and 3/4t locations and the margin term in Table 1 of this SE. The NRC staff finds the licensee's 50 EFPY ART calculations acceptable and valid for the P-T limit curve calculations.

P-T Limit Curve Calculations - Vessel

The NRC staff evaluated the licensee's proposed 50 EFPY P-T limit curves by independently calculating P-T limit curves based on the governing equation in Appendix G to Section XI of the ASME Code, but with K_{It} computed from a 1-dimensional thermal stress analysis across a vessel wall and the allowable pressure computed from a method similar to that used in Section 2.6, "Pressure-Temperature Curve Generation Methodology," of WCAP-14040-NP-A. The NRC staff's calculations included verifying that (1) the requirements in Table 1 "Pressure and Temperature Requirements for the Reactor Pressure Vessel" of Section IV.A of Appendix G to 10 CFR Part 50 and (2) the instrumentation errors specified in the licensee's submittal were correctly applied. The NRC staff also verified that for the lower temperature range of the cooldown P-T limit curve, the limiting material is the Upper Shell Plate W10201-1 and for the higher temperature range, the limiting material is the Circumferential Weld Seam 10-273 as reflected in Table 28 "50 EFPY Cooldown Curve Data Points Using 1996 App. G" of WCAP-15827. Based on these confirmatory calculations, the NRC staff finds the licensee's proposed 50 EFPY heatup and cooldown curves (including the leak test and criticality limits) reflected in the revised Figure 3.4.3-1, "Reactor Coolant System Heatup Limits Applicable Up to 50 EFPY," and revised Figure 3.4.3-2, "Reactor Coolant System Cooldown Limitations Applicable Up to 50 EFPY," respectively, of the HBRSEP2 TSs included in Attachments 2 and 3 of the LAR (Reference 1), acceptable.

P-T Limit Curve Calculations - Nozzles

The NRC staff evaluated and determined that the updated neutron fluence values reported in WCAP-18100-NP (Reference 4) are acceptable. Details of the NRC staff's evaluation and determination are in Section 3.1 of this safety evaluation.

In the licensee's supplemental analysis submitted in a letter dated September 14, 2016 (Reference 5), in response to RAI-3, the licensee determined initial RT_{NDT} values for the inlet nozzle based on Position 1.1(4) of SRP BTP 5-3 and for the outlet nozzle based on the methodology in Appendix B, Alternative Approach 2 of NRC-approved proprietary topical report BWRVIP-173-A (Reference 20). The licensee used Position 1.1(4) of SRP BTP 5-3 for the inlet nozzle because Charpy V-notch tests were performed at a single temperature and Alternative Approach 2 of BWRVIP-173-A for the outlet nozzle because drop-weight test data were not available.

The NRC staff evaluated whether topical report BWRVIP-173-A, which was prepared by the Boiling Water Reactors Vessel and Internals Project (BWRVIP) industry group, is applicable to PWRs. The objective of the report (page xvii) states:

> To perform statistical evaluations on data of SA508-2 forging materials used in reactor vessels and to determine an estimate for the chemistry properties to use in embrittlement prediction methods for determining the RT_{NDT} shift if limited or no data for a particular SA508-2 forging heat are available.

There is nothing in the objective that specifies that the topical report is applicable only to BWRs. Furthermore, the approach on page xvii of the report states that chemistry data of SA508-2 forgings in both BWRs and PWRs were included in the statistical analyses. The objective of

BWRVIP-173-A stated above is consistent with NRC staff's conclusions in its final safety evaluation of BWRVIP-173-A issued on July 16, 2010 (Reference 21), which states, in part:

> The NRC staff has reviewed the BWRVIP-173 report and found that the report is acceptable for providing an estimate of the properties to be used in current embrittlement prediction methods, if there is limited or no data available for SA508-2 RV nozzle forging materials.

The NRC staff confirmed in the HBRSEP2 UFSAR, Table 5.2.3-1, "Materials of Construction of the Reactor Coolant System Components," of Section 5.2, "Integrity of Reactor Coolant Pressure Boundary," that the HBRSEP2 RPV nozzle forgings are made of SA-502, Class 2 material. Therefore, the NRC staff determined that the use of BWRVIP-173-A, to determine the initial RT_{NDT} value for the HBRSEP2 outlet nozzle forging material, is acceptable.

The NRC staff performed confirmatory calculations of initial RT_{NDT} using Position 1.1(4) of SRP BTP 5-3 and Alternative Approach 2 of BWRVIP-173-A in conjunction with RG 1.99, Revision 2 (Reference 8), and determined that the licensee's initial RT_{NDT} values on Table 7 of the September 14, 2016, supplemental analysis for the HBRSEP2 inlet and outlet nozzle are reasonable, and therefore acceptable.

The licensee re-calculated the ART values for the HBRSEP2 inlet and outlet nozzles and determined the P-T limits curves for the nozzles by postulating a one quarter thickness (defined as the section thickness forty-five degrees from the nozzle corner) inside-surface flaw at the rounded corner (blend radius) of the nozzles. A salient element of the nozzle ART calculations is that if the delta RT_{NDT} (ΔRT_{NDT}) value is less than 25 degrees Fahrenheit (°F), embrittlement effects may be neglected, based on the results of the studies in NRC technical report TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," dated November 14, 2014 (Reference 22). The NRC staff performed the nozzle ART calculations and confirmed the licensee's values reported in Table 8 of the September 14, 2016, supplemental analysis. The fracture mechanics calculations for the postulated inside-surface nozzle corner flaw are based on methods in the NRC-sponsored report ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water reactor and Pressurized Water Reactor Pressure Vessel Nozzles – Revision 1" (Reference 23). The NRC staff performed the nozzle corner flaw fracture mechanics calculations to determine P-T limit curves for a 100 °F/hour cooldown and confirmed that, as shown in Figures 1 and 2 of the September 14, 2016, supplemental analysis, the licensee's 50 EFPY P-T limit curves for the HBRSEP2 inlet and outlet nozzles are bounded by the HBRSEP2 50 EFPY P-T limit curves for cooldown included in the licensee's November 2, 2015, LAR.

The licensee also qualitatively considered the stress levels in the vessel-to-nozzle weld as compared to those in the inside-surface nozzle corner and determined that the stresses at the nozzle corner would be bounding because of the discontinuity. Furthermore, the licensee considered the vessel-to-nozzle welds as part of the RPV P-T limits evaluation in WCAP-15827 and determined the vessel-to-nozzle welds were not the limiting location. The NRC staff agrees with this assessment and it is therefore acceptable.

Additionally, since updated neutron fluence values for the HBRSEP2 RPV and RPV welds were also reported in WCAP-18100-NP, the NRC staff confirmed that, considering the updated

neutron fluence values, the HBRSEP2 50 EFPY P-T limit curves included in the licensee's November 2, 2015, LAR are still bounding. Finally, the licensee addressed in its September 14, 2016, submittal (Reference 5), ferritic RCS pressure boundary components that are not part of the HBRSEP2 RPV. These components include the pressurizer, replacement vessel head, and replacement steam generators and are not expected to receive neutron fluence levels such that they need to be considered for P-T limit evaluation. The NRC staff reasonably infers from the licensee's considerations that there are no other ferritic materials of the HBRSEP2 RPV that needs to be considered for P-T limit evaluation. The NRC staff, therefore, finds the licensee's consideration of ferritic RCS pressure boundary components that are not part of the HBRSEP2 RPV acceptable.

Based on the above evaluation, the NRC staff determined that the licensee has adequately addressed all the concerns of the NRC staff in RAI-3 and has adequately demonstrated that the HBRSEP2 50 EFPY P-T limit curves included in the November 2, 2015, LAR bound all ferritic pressure boundary components of the RPV. Therefore, RAI-3 is resolved.

Low Temperature Overpressure Protection Settings

The LTOP system imposes a limit on the reactor coolant pressure at low temperatures so that the integrity of the RCPB is not compromised in low-temperature modes of operation. Appendix G to Section XI of the ASME Code requires that for plants with LTOP systems, the LTOP system shall be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a RPV metal temperature less than RT_{NDT} + 50 °F, whichever is greater. In addition, Appendix G to Section XI of the ASME Code limits the maximum pressure in the RPV to 100 percent of the pressure determined to satisfy Equation 1 of G-2215 of Appendix G to Section XI of the ASME Code. The NRC staff notes that Equation 1 of G-2215 is the basis for the P-T limit curves for heatup and cooldown conditions. Section 3.2.2, "Pressure Limits Selection," of WCAP-14040-NP-A, however, clarifies that since overpressure events are likely to occur during isothermal (steady-state) conditions, it is appropriate that the maximum pressure for the LTOP setting be based on steady-state conditions rather than on heatup and cooldown conditions. The licensee stated in its LAR that the LTOP limits remain unchanged as a result of the proposed 50 EFPY P-T limit curves. The licensee, however, did not include the LTOP temperature enable setting value and pressurizer power-operated relief valve (PORV) LTOP setting value in the LAR. Therefore by e-mail dated March 30, 2016 (Reference 17), the NRC staff requested in RAI-1 the LTOP temperature enable setting value and the PORV LTOP setting value. By letter dated March 31, 2016 (Reference 3), the licensee provided the LTOP temperature enable setting value and the PORV LTOP setting value, which are 350 °F and 400 pounds per square inch gauge, respectively. The licensee also provided a basis for why the LTOP limits remain unchanged. This basis included a comparison of the LTOP limits with the steady-state P-T values for 35 EFPY and 50 EFPY in Table 22, "35 EFPY Cooldown Curve Data Points Using 1996 App. G" and Table 28, "50 EFPY Cooldown Curve Data Points Using 1996 App. G", respectively, of WCAP-15827. The NRC staff calculated steady-state P-T curves and confirmed (1) the steady state P-T values in Tables 22 and 28 of WCAP-15827 and (2) the LTOP temperature enable setting and PORV LTOP setting values provided by the licensee meet the temperature and maximum RPV pressure requirements in Appendix G to Section XI of the ASME Code, considering the proposed 50 EFPY P-T limit curves. Therefore, RAI-1 is resolved. Based on the above, the NRC staff finds the licensee's proposed LTOP settings acceptable.

Evaluation of Applicable Technical Specifications

TS 3.4.3 addresses RCS P-T limits for all modes of operation. This includes the limiting condition of operation (LCO), the surveillance requirements (SRs), and the P-T limit curves in Figures 3.4.3-1 and 3.4.3-2. LCO 3.4.3 limits the pressure and temperature changes during RCS heatup and cooldown (i.e., to the right and below the P-T curves), to prevent the non-ductile RPV. The LAR extends the applicability of the P-T limits to 50 EFPY, which does impact the function of the LCO to limit RCS operation to within approved P-T limits. Based on the NRC staff finding above that Figures 3.4.3-1 and 3.4.3-2 are acceptable, the staff finds the use of the revised Figures 3.4.3-1 and 3.4.3-2 in LCO 3.4.3 meets 10 CFR 50.36(c)(2)(i) by providing for the requisite functional capability or performance level required for safe operation.

SR 3.4.3.1 verifies that that RCS operation is within the limits of Figures 3.4.3-1 and 3.4.3-2 when RCS pressure and temperature conditions are undergoing planned changes. The LAR extends the applicability of the P-T limits to 50 EFPY, which does not impact the function of SR 3.4.3.1 to verify that the RCS is operated within approved P-T limits. Therefore, the NRC staff finds the use of the revised Figures 3.4.3-1 and 3.4.3-2 in SR 3.4.3.1 meets 50.36(c)(3) by providing sufficient test, calibration, or inspection requirements to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

TS 3.4.12 “Low Temperature Overpressure Protection (LTOP) System” addresses the condition when the accumulator is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P-T limit curves provided in Figures 3.4.3-1 and 3.4.3-2. This includes the note in LCO 3.4.12 and the condition statement and required action in Conditions C and D of the LCO. LCO 3.4.12 provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. The notes permit accumulator surveillances to be performed when the accumulator pressure does not exceed the P-T limits. Action Statements C and D address and improperly isolated accumulator. The LAR extends the applicability of the P-T limits to 50 EFPY, which does not impact the conditions and actions in LCO 3.4.12. Therefore, the NRC staff finds the use of the revised Figures 3.4.3-1 and 3.4.3-2 in LCO 3.4.12, Action Statements C and D, meets 10 CFR 50.36(c)(2)(i) by providing for the requisite functional capability or performance level required for safe operation.

3.2.3 Vessel and Reactor Coolant Pressure Boundary Integrity Conclusion

Based on the licensee’s P-T limits curves conforming to an NRC-approved methodology, WCAP-14040-NP-A, and independent confirmatory NRC staff calculations, the NRC staff finds that the proposed changes to the HBRSEP2 TS in the LAR, as supplemented (i.e., the 50 EFPY P-T limit curves, comply with the acceptance criteria in Section 2.0 above. The NRC staff finds that the usage of Figures 3.4.3-1 and 3.4.3-2 in TS 3.4.3 and TS 3.4.12 meets 10 CFR 50.36(c)(2)(i) by providing for the requisite functional capability or performance level required for safe operation and meets 50.36(c)(3) by providing sufficient test, calibration, or inspection requirements to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Therefore, the staff finds the 50 EFPY P-T limit curves 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. 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 March 1, 2016 (81 FR 10678). 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. Glover, R. M., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Request for Technical Specification Change to Reactor Coolant System Pressure and Temperature Limits," November 2, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15307A069).

2. Glover, R. M., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Supplement to Request for Technical Specification Change to Reactor Coolant System Pressure and Temperature Limits," December 22, 2015 (ADAMS Accession No. ML15356A481).

3. Glover, R. M., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding License Amendment Request to Revise Reactor Coolant System Pressure and Temperature Limits," March 31, 2016 (ADAMS Accession No. ML16091A087).

4. Glover, R. M., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Supplement to Response to Request for Additional Information Regarding

License Amendment Request to Revise Reactor Coolant System Pressure and Temperature Limits," May 9, 2016 (ADAMS Accession No. ML16130A645).

5. Glover, R. M., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Regarding License Amendment Request to Revise Reactor Coolant System Pressure and Temperature Limits," September 14, 2016 (ADAMS Accession No. ML16258A343).

6. Westinghouse Electric Corporation, WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 15, 1996 (ADAMS Accession No. ML15324A233).

7. Westinghouse Electric Corporation, WCAP-15805, "Analysis of Capsule X from the Carolina Power and Light Company H.B. Robinson Unit 2 Reactor Vessel Radiation Surveillance Program," March 2002 (ADAMS Package Accession No. ML021190366).

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

9. U.S. Nuclear Regulatory Commission, Generic Letter 92-01, "Reactor Vessel Structural Integrity," Revision 1, March 6, 1992 (ADAMS Accession No. ML031070438).

10. U.S. Nuclear Regulatory Commission, Generic Letter 92-01, Supplement 1, "Reactor Vessel Structural Integrity," Revision 1, May 19, 1995 (ADAMS Accession No. ML031070449).

11. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," NUREG-0800 (SRP), Section 5.3.2, Revision 2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock," March 2007 (ADAMS Accession No. ML070380185)

12. 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).

13. U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum for SECY-92-223, "Resolution of Deviations Identified during the Systematic Evaluation Program," September 18, 1992 (ADAMS Accession No. ML003763736).

14. 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).

15. F. A. Schmittroth, *FERRET Data Analysis Code,* HEDL-TME 79-40, Hanford Engineering Development Laboratory, Richland, WA, September 1979.

16. Westinghouse Electric Company LLC, "WCAP-16083-NP-A, Revision 0, Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry," May 2006 (ADAMS Accession No. ML061600256).

17. Galvin, D. G., U.S. Nuclear Regulatory Commission, e-mail to Richard Glover, Duke Energy Progress, Inc., "H. B. Robinson Steam Electric Plant Unit No. 2 – Request for Additional Information - License Amendment Request to Change the Technical Specification Reactor Coolant System Pressure and Temperature Limits (TAC No. MF7048)," March 30, 2016 (ADAMS Accession No. ML16090A341).

18. U.S. Nuclear Regulatory Commission, NUREG-0800, SRP, Section 5.3.2, "Pressure-Temperature Limits," Revision 1, "Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," July 1981 (ADAMS Accession No. ML052340637).

19. U.S. Nuclear Regulatory Commission, NUREG-0800, SRP, Branch Technical Position 5-3, Revision 2, "Fracture Toughness Requirements," March 2007 (ADAMS Accession No. ML070850035).

20. Electric Power Research Institute, BWRVIP-173-A, "BWR Vessel and Internals Project: Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials," July 2011 (ADAMS Accession Nos. ML12083A268 (publicly available), ML12083A269 (proprietary information, not publicly available)).

21. Blount, T. B., U.S. Nuclear Regulatory Commission, letter to David M. Czufin, Chairman, BWR Vessel and Internals Project, "Final Safety Evaluation for Boiling Water Reactor Vessel and Internals Project, 'BWRVIP-173: BWR Vessel and Internals Project, Evaluation of Chemistry Data for BWR Nozzle Forging Materials' EPRI Technical Report 1014995 (TAC No. MD6255)," July 16, 2010 (ADAMS Accession Nos. ML101760447 (letter), ML101760460 (publicly available), ML101760454 (proprietary information, not publicly available)).

22. U.S. Nuclear Regulatory Commission, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," Technical Letter Report TLR-RES/DE/CIB-2013-01, November 14, 2014 (ADAMS Accession No. ML14318A177).

23. Oak Ridge National Laboratory, ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles – Revision 1," June 2012 (ADAMS Accession No. ML12181A162).

Principal Contributors: D. Dijamco
A. Patel

Date: November 22, 2016

November 22, 2016

Mr. R. Michael Glover
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 TO REVISE REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS APPLICABLE FOR 50 EFFECTIVE FULL POWER YEARS (CAC NO. MF7048)

Dear Mr. Glover:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 248 to Renewed Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP2). This amendment changes the HBRSEP2 Technical Specifications (TSs) in response to your application dated November 2, 2015, as supplemented by letters dated December 22, 2015; and March 31, May 9, and September 14, 2016. The amendment revises the reactor coolant system (RCS) pressure and temperature limits by replacing TS Section 3.4.3, "RCS Pressure and Temperature (P/T) Limits," Figures 3.4.3-1 and 3.4.3-2, with figures that are applicable up to 50 effective full power years.

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

Sincerely,

/RA/
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. 248 to DPR-23
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL2-2 R/F
RidsACRS_MailCTR Resource
RidsNrrDssStsb Resource
APatel, NRR
RidsNrrDorlLpl2-2 Resource
LRonewicz, NRR
RidsNrrPMRobinson Resource
RidsNrrDeEvib Resource
RGrover, NRR
RidsNrrLABClayton Resource
RidsRgn2MailCenter Resource
RidsNrrDssSnpb Resource
DDijamco, NRR

ADAMS Accession No.: ML16285A404 *via memo **via e-mail

OFFICE	NRR/DORL/LPL2-2/PM	NRR/DORL/LPL2-2/LA	NRR/DE/EVIB/BC*	NRR/DSS/SNPB/BC**
NAME	DGalvin	BClayton (PBlechman for)	DRudland	BLukes
DATE	10/27/2016	10/27/2016	10/12/2016	10/5/2016
OFFICE	NRR/DSS/STSB/BC**	OGC /NLO **	NRR/DORL/LPL2-2/BC(A)	NRR/DORL/LPL2-2/PM
NAME	AKlein	DRoth	JDion	DGalvin
DATE	10/13/2016	11/9/2016	11/22/2016	11/22/2016

OFFICIAL RECORD COPY