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

November 27, 2018

ANO Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
N-TSB-58
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 - ISSUANCE OF AMENDMENT RE: UPDATING THE REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS (EPID L-2017-LLA-0396)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 311 to Renewed Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 20, 2017, as supplemented by letters dated August 1, 2018, and October 10, 2018.

The amendment revises the ANO-2 TSs to replace the current pressure-temperature limits for heatup, cooldown, and the inservice leak hydrostatic tests for the reactor coolant system presented in TS 3.4.9, "Pressure/Temperature Limits," which expire at 32 Effective Full Power Years (EFPY), with limitations that extend out to 54 EFPY.

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

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 311 to NPF-6
2. Safety Evaluation

cc: Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 311
Renewed License No. NPF-6

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated November 20, 2018, as supplemented by letters dated August 1, 2018, and October 10, 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, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 311, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications

3. This amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. NPF-6 and
Technical Specifications

Date of Issuance: November 27, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 311

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

Replace the following pages of the Renewed Facility Operating License No. NPF-6 and 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.

Operating License

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
3/4 4-22	3/4 4-22
3/4 4-24	3/4 4-24
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26

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

(5) EOI, 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 calibration or associated with radioactive apparatus or components; and

(6) EOI, 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 the operation of the facility.

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

(1) <u>Maximum Power Level</u>

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 3026 megawatts thermal. Prior to attaining this power level EOI shall comply with the conditions in Paragraph 2.C.(3).

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 311, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

Exemptive 2nd paragraph of 2.C.2 deleted per Amendment 20, 3/3/81.

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

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

2.C.(3)(a) Deleted per Amendment 24, 6/19/81.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2A, 3.4-2B and 3.4-2C during heatup/criticality, cooldown, and inservice leak and hydrostatic testing operations with:

a. A maximum heatup of 50 °F, 60 °F, 70 °F or 80 °F in any one hour period in accordance with Figure 3.4-2A.

b. A maximum cooldown rate of 100 °F per hour (constant) or 50 °F in any half hour period (step) for RCS cold leg temperatures between 60 °F and 560 °F.

c. A maximum temperature change of ≤ 10 °F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

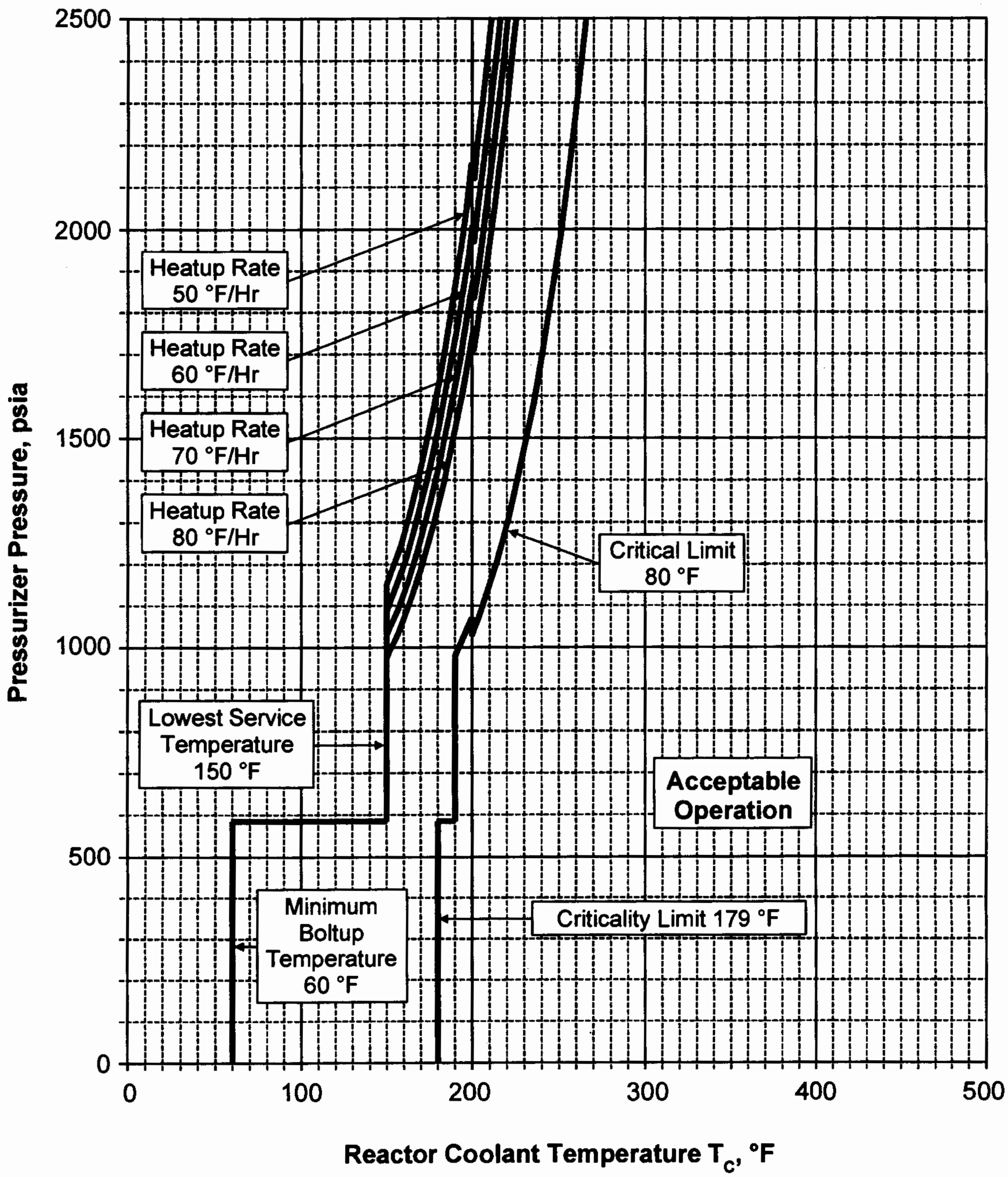
APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the acceptable region of the applicable curve within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tc and pressure to less than 200 °F and less than 500 psia, respectively, within the following 30 hours.

Figure 3.4-2A

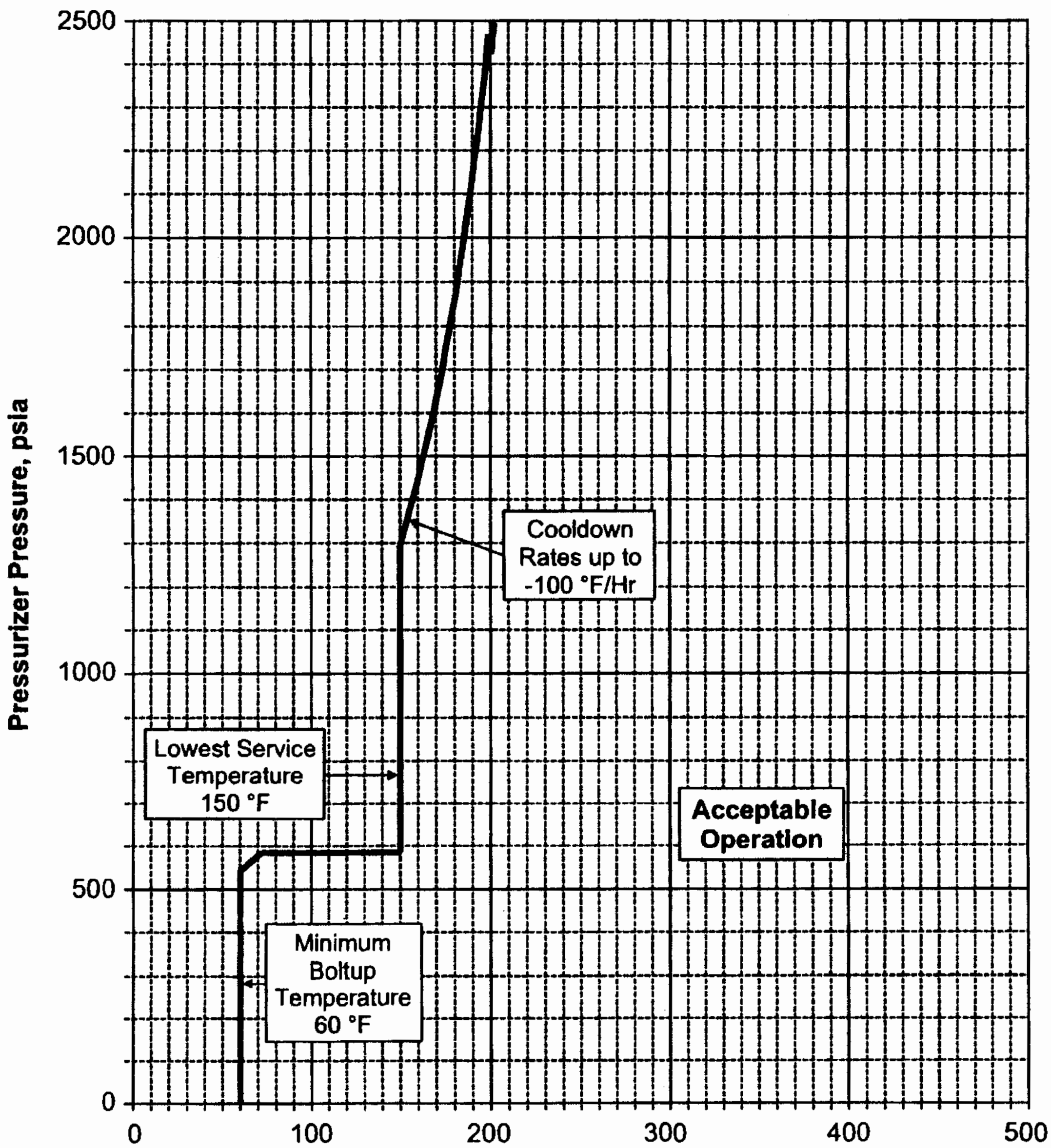
HEATUP CURVE – 54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



(Curves do not include margins for instrument uncertainties)

Figure 3.4-2B

COOLDOWN CURVE – 54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS

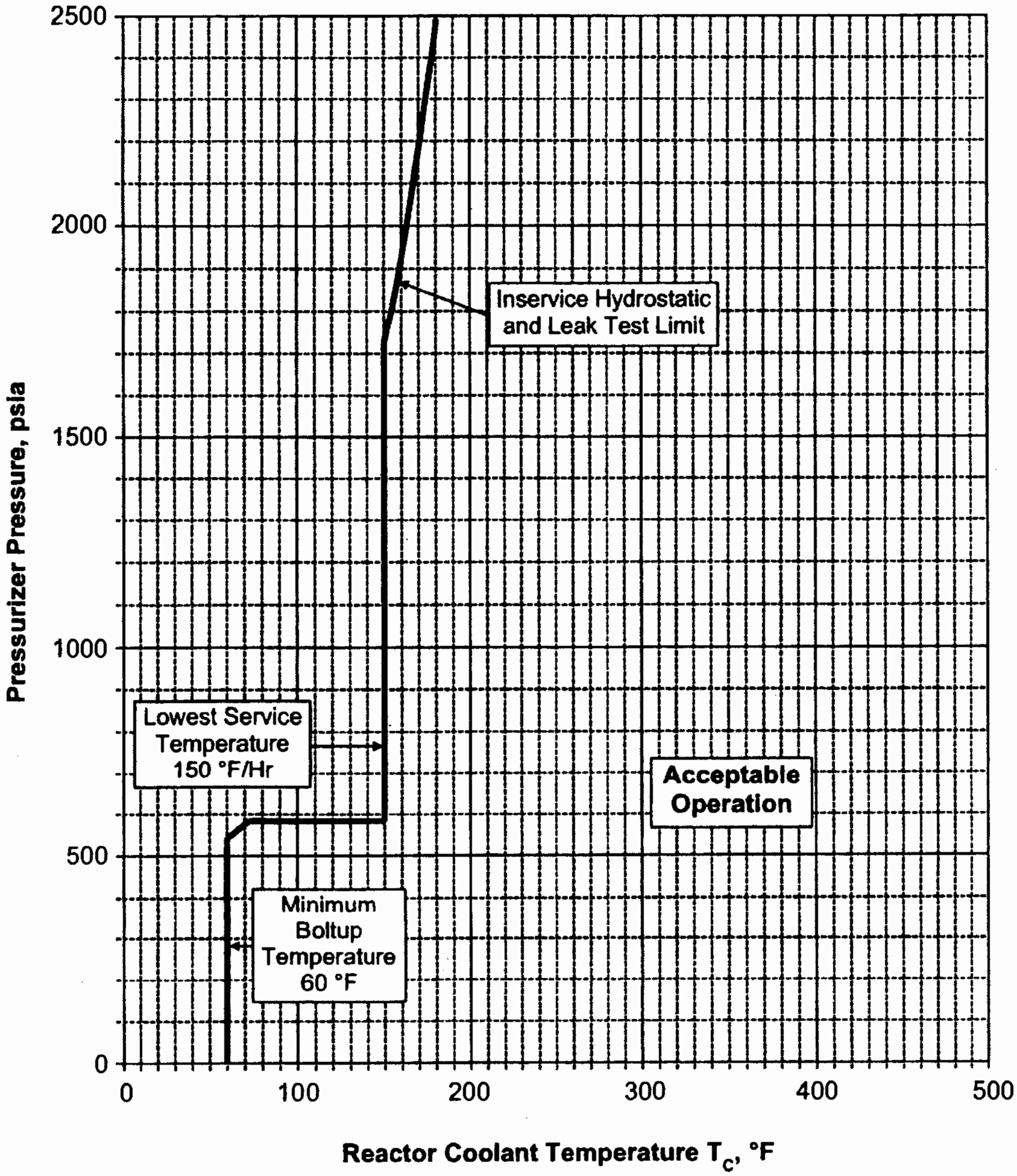


Reactor Coolant Temperature T_c, °F

(Curves do not include margins for instrument uncertainties)

Figure 3.4-2C

INSERVICE HYDROSTATIC TEST CURVE – 54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



(Curves do not include margins for instrument uncertainties)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 311 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By application dated November 20, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17326A379), as supplemented by letters dated August 1 and October 10, 2018 (ADAMS Accession Nos. ML18215A198 and ML18283A599, respectively), Entergy Operations, Inc. (the licensee) requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 2 (ANO-2).

The proposed changes would revise the ANO-2 TSs to replace the current pressure-temperature (P-T) limits for heatup, cooldown, and the inservice leak hydrostatic tests for the reactor coolant system (RCS) presented in TS 3.4.9, "Pressure/Temperature Limits," which expire at 32 Effective Full Power Years (EFPY), with limitations that extend out to 54 EFPY.

The supplemental letters dated August 1 and October 10, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 27, 2018 (83 FR 8514).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications," identifies the requirements for the contents of TSs. Paragraph 50.36(c)(2) of 10 CFR requires, in part, establishing limiting conditions for operations (LCOs) to assure safe operation of a nuclear reactor.

Section 50.60 of 10 CFR, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,” states:

> (a) Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.
>
> (b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12.

The regulation at 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” states, in part:

> This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The provisions of 10 CFR Part 50, Appendix G, require that the P-T limits for an operating light-water nuclear power reactor be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) were used to generate the P-T limits. Appendix G also requires that applicable surveillance data from reactor pressure vessel (RPV) material surveillance programs be incorporated into the calculations of plant-specific P-T limits, and that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials.

Table 1 of 10 CFR Part 50, Appendix G, provides the NRC staff's criteria for meeting the P-T limit requirements of the ASME Code, Section XI, Appendix G, as well as the minimum temperature requirements for the RPV during normal heatup, cooldown, and pressure test operations. In addition, the NRC staff regulatory guidance related to P-T limit curves is contained in Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” dated May 1988 (ADAMS Accession No. ML003740284); and NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition,” Section 5.3.2, Revision 2 “Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock,” dated March 2007 (ADAMS Accession No. ML070380185).

The P-T limit curve calculations are based, in part, on the nil ductility reference temperature (RT_{NDT}) for the material, as specified in the ASME Code, Section XI, Appendix G. Appendix G of the ASME Code requires that RT_{NDT} values for materials in the RPV beltline region be adjusted to account for the effects of neutron irradiation.

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," states, in part:

> The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Section IV of Appendix G to Part 50.
>
> ASTM [American Society for Testing and Materials] E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels"; and ASTM E 185-79, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels"; and ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register.

ANO-2 was designed and constructed to meet the intent of the Atomic Energy Commission's (AEC's) general design criteria (GDC), as originally proposed in July 1967, and thus, the design and construction were initiated and proceeded to a significant extent based upon the criteria proposed in 1967. The ANO-2 Safety Analysis Report (SAR) Amendment 26 (ADAMS Accession No. ML16132A517), Section 3.1, "Conformance with AEC General Design Criteria," describes the manner in which the ANO-2 GDC meet the intent of the corresponding GDC published as Appendix A to 10 CFR Part 50 in 1971. The regulations in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," include the following GDC applicable to fracture prevention of the reactor coolant pressure boundary, as described in this license amendment request (LAR):

- GDC 31, "Fracture prevention of reactor coolant pressure boundary," which states:

  The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

In addition, the NRC staff determined that the following GDC of 10 CFR Part 50, Appendix A, are also applicable to this LAR:

- GDC 15, "Reactor coolant system design," which states:

  The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- GDC 30, "Quality of reactor coolant pressure boundary," which states:

  Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

- GDC 32, "Inspection of reactor coolant pressure boundary," which states:

  Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor vessel.

Requirements related to fracture toughness of ferritic RPV materials are provided in 10 CFR Part 50, Appendix G. Paragraph A of Section IV, "Fracture Toughness Requirements," of 10 CFR Part 50, Appendix G, states that limits intended to ensure adequate protection of vessel ductility "must account for the effects of neutron radiation." Licensees typically account for such effects by calculating the reactor vessel neutron fluence expected at the end of the applicability period for the P-T limit curves. This estimated fluence is then converted to a fluence factor that is included in the formulation of the P-T limit curves.

2.2 Applicable Regulatory Guidance

The following regulatory guidance is applicable to this LAR:

Revision 2 of RG 1.99 describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. RG 1.99 contains methodologies for calculating the adjusted reference temperature (ART) due to neutron irradiation. The ART is defined as the sum of the initial (unirradiated) RT_{NDT}, the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term. The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99 or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth assumed for the P-T limit calculations. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value, and whether the CF was determined using the tables in RG 1.99 or from surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the neutron fluence, and the calculational procedures. Revision 2 of RG 1.99 describes a methodology that may be used in calculating the margin term.

In March 2001, the NRC staff issued RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (ADAMS Accession No. ML010890301). RG 1.190 provides an acceptable methodology to calculate fluence for use in ART and P-T limit curve analyses.

On October 14, 2014, the NRC issued Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (ADAMS Accession No. ML14149A165), which clarified that the beltline definition in 10 CFR Part 50, Appendix G, is applicable to all RPV ferritic materials with projected neutron fluence values greater than 1×10^{17} neutrons/centimeter-squared (n/cm^2) with energy greater than 1 million electron volts (E > 1 MeV), and this neutron fluence threshold remains applicable for the licensed operating period.

Branch Technical Position (BTP) 5-2, Revision 3, "Overpressurization Protection of Pressurized-Water Reactors while Operating at Low Temperatures," of NUREG-0800, provides guidance to the NRC staff in reviewing overpressurization protection of pressurized water reactors (PWRs) while operating at low temperatures (ADAMS Accession No. ML070850008). Paragraph B.1 of BTP 5-2 specifies that the low temperature overpressure protection (LTOP) system be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to satisfy the TS limits while operating at low temperatures.

2.3 Acceptable Fluence Calculations

RG 1.190 describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence with respect to the GDC contained in Appendix A to 10 CFR Part 50. In consideration of the guidance set forth in RG 1.190, GDC 14, 30, and 31 are applicable.

- GDC 14, "Reactor coolant pressure boundary," requires the design, fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

- GDC 30, "Quality of reactor coolant pressure boundary," requires, in part, that components comprising the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical.

- GDC 31, "Fracture prevention of reactor coolant pressure boundary," pertains to the design of the reactor coolant pressure boundary, and states:

  The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

The guidance provided in RG 1.190 indicates that the following elements comprise an acceptable fluence calculation:

1. Determination of the geometrical and material input data,
2. Determination of core neutron source,
3. Propagation of the neutron fluence from core to vessel and into the cavity, and
4. Qualification of the calculational procedure.

The NRC's review of the fluence calculation was performed to establish that elements 1 through 4, above, of the calculational method adhere to the regulatory positions set forth in RG 1.190.

2.4 Proposed TS Changes

The licensee proposed to make revisions to the ANO-2 TS 3.4.9, "Pressure/Temperature Limits," as follows:

Proposed Revision to TS 3.4.9.1.b

- Current TS 3.4.9.1.b states:

> A maximum cooldown rate of 100°F [degrees Fahrenheit] per hour (constant) or 50°F in any half hour period (step) for RCS cold leg temperatures between 50°F and 560°F.

- Revised TS 3.4.9.1.b would state:

> A maximum cooldown rate of 100 °F per hour (constant) or 50 °F in any half hour period (step) for RCS cold leg temperatures between 60 °F and 560 °F.

Proposed Revision to TS Figures

- Replace current Figure 3.4-2A, "HEATUP CURVE – 32 EFPY," with a new figure applicable to 54 EFPY.
- Replace current Figure 3.4-2B, "COOLDOWN CURVE – 32 EFPY," with a new figure applicable to 54 EFPY.
- Replace current Figure 3.4-2C, "INSERVICE HYDROSTATIC TEST CURVE – 32 EFPY," with a new figure applicable to 54 EFPY.

Based on the new P-T limits, the licensee evaluated the enable temperature and other related LTOP limits presented in TS 3.4.12 and determined that this TS did not need to be revised.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The technical basis for ANO-2's revised P-T limits is provided in Attachment 3 to the LAR dated November 20, 2017: Non-proprietary Westinghouse Report No. WCAP-18169-NP, Revision 0, "Arkansas Nuclear One Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," December 2016. Westinghouse Report No. WCAP-18169-NP states that the proposed 54 EFPY heatup and cooldown limit curves were generated using ART values for the most limiting RPV beltline shell material, plus an additional margin, based on the NRC-approved generic P-T limits methodology documented in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown

Limit Curves," May 2004 (ADAMS Accession No. ML050120209). Westinghouse Report No. WCAP-18169-NP also states that the neutron transport evaluation methodologies for determining RPV beltline fluence followed the guidance of RG 1.190 and are consistent with the NRC-approved methodology described in WCAP-14040-A.

Westinghouse Report No. WCAP-18169-NP includes a detailed description of the methods employed for generating the P-T limit curves, which are based on the methodology detailed in ASME Code Section XI, Appendix G. The key parameters necessary for generating P-T limit curves are the RPV material fracture toughness, K_{IC}, and the applied stress intensity factors due to pressure and thermal stresses, K_{IM} and K_{IT}, respectively. For all RPV beltline materials, K_{IC} was established based on the ART for the material, consistent with the 1998 Edition through 2000 Addenda of the ASME Code, Section XI, Appendix G. Westinghouse Report No. WCAP-18169-NP describes how the ART values for the RPV beltline materials were determined by calculating the effects of projected neutron embrittlement through 54 EFPY using the procedures in RG 1.99, Revision 2. The K_{IM} and K_{IT} values were calculated using the formulations specified in WCAP-14040-A, which are the same as those specified in ASME Code, Section XI, Appendix G, paragraph G-2214.

Westinghouse Report No. WCAP-18169-NP also addresses the impact of complex geometries of RPV components (i.e., RPV nozzles, penetrations, other structural discontinuities) outside of the RPV beltline shell region that experience higher local stresses than the RPV shell. The P-T limit curves were developed for the inlet and outlet nozzle inside corner regions since the geometric discontinuities results in high stresses due to internal pressure and the cooldown transient. The licensee stated that P-T limit curves generated based on the limiting cylindrical beltline material bound the P-T limit curves for the vessel inlet and outlet nozzles.

The LAR also addresses the reevaluation of the LTOP limits. The LTOP limits, which were based on ASME Code, Section XI, Article G-2215, were reanalyzed to address a higher backpressure in the system and a higher high pressure safety injection flowrate. The licensee stated that the current LTOP relief valve settings were shown to be acceptable when compared to the limiting LTOP transient in the 54 EFPY P-T limits. The enable temperature was also reevaluated and determined to be acceptable as is. Based on these evaluations, the licensee stated that ANO-2 TS 3.4.12 did not need to be revised based on the change to the 54 EFPY P-T limits.

The LAR also addresses pressurized thermal shock (PTS). The licensee stated that a PTS assessment was performed for the reactor vessel beltline materials with fluence greater than $1x10^{17}$ n/cm^2. The licensee identified the Lower Shell Plate 8010-1 as the controlling material with a predicted RT_{PTS} value of 122.4 °F. The PTS screening criterion is 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials.

The LAR also addresses upper-shelf energy (USE). The licensee stated that all of the beltline and extended beltline materials in its RPV are projected to remain above the USE screening criteria of 50 foot-pounds (ft-lb) through 54 EFPY. The licensee stated that the limiting USE value for the ANO-2 RPV at 54 EFPY is 60.2 ft-lb, which is for the Upper Shell Plate C-8008-2.

3.2 NRC Staff Evaluation

The NRC staff reviewed the licensee's LAR submittals, including WCAP-18169-NP, to determine whether the proposed 54 EFPY P-T limit curves are in compliance with the requirements of 10 CFR Part 50, Appendix G. The NRC staff verified that the proposed

54 EFPY P-T limits were developed by taking into account all portions of the RPV, including the inlet and outlet nozzles. The NRC staff noted that based on the evaluation of all regions of the RPV, the bounding P-T limits for 54 EFPY are controlled by the RPV beltline shell region and the minimum boltup temperature, which was revised in accordance with WCAP-14040-A. The NRC staff also reviewed the licensee's evaluations of the LTOP limits, PTS, and USE. The details of the NRC staff's evaluation are discussed below.

3.2.1 Evaluation of the Neutron Fluence Values Used for Determining the RPV Beltline Region ARTs

The licensee provided a detailed description of the neutron fluence calculations used to determine the PT limits in Section 2, "Calculated Neutron Fluence," of WCAP-18169-NP (Attachment 3 to the LAR dated November 20, 2017). According to the methodology described in WCAP-18169-NP, the licensee performed its fluence evaluation using the methods described in WCAP-14040-A, Revision 4. Fluence methods are described in Chapter 2, "Pressure-Temperature Limit Curves," of WCAP-14040-A. The methods described in WCAP-14040-A were reviewed and approved for use by the NRC staff, based on their adherence to the guidance in RG 1.190.

The NRC staff reviewed the information contained in Section 2 of WCAP-18169-NP. Based on its review, the NRC staff determined that the plant-specific fluence calculations were performed in a manner consistent with the NRC-approved methodology contained in WCAP-14040-A, Revision 4. This includes the level of detail represented in the geometric modeling, the use of fuel-cycle specific neutronic data for past operating cycles, and the use of discreet ordinates transport methods and a flux synthesis technique to capture three-dimensional aspects of the transport problem. Although benchmarking is addressed generically in WCAP-14040-A, the licensee also provided comparisons to ANO-2-specific capsule dosimetry to confirm that the transport calculations agree with measured data within 20 percent, as recommended by RG 1.190.

Since the plant-specific calculation was performed in a manner consistent with an NRC-approved methodology that adheres to RG 1.190, the NRC staff determined that the plant-specific calculations are also consistent with the guidance contained in RG 1.190. Based on these considerations, the NRC staff determined that the fluence calculations are acceptable.

3.2.2 Evaluation of the ART Values and P-T Limit Curves for RPV Beltline Shell Region

The NRC staff reviewed the licensee's proposed 54 EFPY P-T limits to determine if they were calculated based on an evaluation of the RPV beltline shell region, accounting for projected neutron embrittlement through 54 EFPY, as documented in WCAP-18169-NP. The licensee projected neutron embrittlement through 54 EFPY by calculating the ARTs for the RPV beltline materials using the procedures of RG 1.99, Revision 2. The licensee's ART calculations for the RPV beltline shell region at the one quarter thickness (1/4T) and three quarter thickness (3/4T) locations are provided in Tables 7-2 and 7-3 of WCAP-18169-NP, respectively, including all of the input parameters necessary for calculating the ART values.

The NRC staff verified that the initial RT_{NDT}, Copper content, Nickel content, and CF values used for calculating the beltline material ARTs are consistent with those identified in Table 4.2-2 of the ANO-2 license renewal application (LRA) (ADAMS Accession No. ML032890483), and approved by the NRC staff in the ANO-2 LRA Safety Evaluation Report, NUREG-1828, "Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 2," June 2005

(ADAMS Accession No. ML051730233), with one exception. The staff noted that for the Intermediate to Lower Shell Girth Weld 9-203 (Heat No. 83650), the LRA stated the initial RT_{NDT} was -10 °F. In the LAR for the proposed 54 EFPY P-T limits, the applicant stated that the initial RT_{NDT} is -40 °F. The LAR states that this new initial RT_{NDT} is based on drop-weight data and Charpy V-notch test data. The staff noted that, even though the LAR used a less conservative value for this material, there is sufficient margin from the ART of the limiting material.

With regard to the calculation of the RT_{NDT} values, Section 7, "Calculation of Adjusted Reference Temperature," of WCAP-18169-NP cited NRC Technical Letter Report TLR-RES/DE/CIB-2013-01, "Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels," dated November 14, 2014 (ADAMS Accession No. ML14318A177) as a basis for not considering the shift due to irradiation for RPV materials for which the predicted shift in the reference temperature (ΔRT_{NDT}) is less than 25 °F.

Discounting the shift in RT_{NDT}, due to irradiation if the predicted shift is less than 25 °F, is inconsistent with RIS 2014-11. Therefore, the NRC staff requested in a request for additional information (RAI), in a letter dated May 10, 2018 (ADAMS Accession No. ML18129A425), that the licensee revise its P-T limit evaluation to include RT_{NDT} values for all RPV beltline and extended beltline materials calculated in accordance with 10 CFR Part 50, Appendix G.

In its response to the RAI, by letter dated August 1, 2018 (ADAMS Accession No. ML18215A177), the licensee revised Tables 7-2 and 7-3. The updated tables provide the ART values that are calculated without the use of TLR-RES/DE/CID-2013-01; therefore, the calculated ΔRT_{NDT} values less than 25 °F are not set to zero. The NRC staff reviewed the updated tables and confirmed that the updated values did not affect the limiting ART values that were used to develop the 54 EFPY P-T limit curves. The staff's concern in the RAI is resolved.

A summary of the limiting ART values is provided in Table 7-4 of WCAP-18169-NP. These tables indicate that the limiting beltline shell material is the Lower Shell Plate C-8010-1 (Heat No. C8161-2) based on Regulatory Position 1.1 of RG 1.99. The limiting ART values are 122 °F at the 1/4T location and 109 °F at the 3/4T location at 54 EFPY. The NRC staff independently verified that the 54 EFPY ART values for the limiting beltline shell material were calculated correctly using the procedures in Regulatory Position 1.1 of RG 1.99 and that Lower Shell Plate C-8010-1 is limiting at the 1/4T and 3/4T locations.

The NRC staff verified that the licensee correctly applied RPV material surveillance data in accordance with Regulatory Position 2.1 of RG 1.99 to determine that the RPV surveillance materials, Intermediate Shell Plate C-8009-3 (Heat No. C8182-2), Intermediate to Lower Shell Girth Weld 9-203 (Heat No. 83650), and Upper Shell Plate C-8008-1 (Heat No. C8182-1) are not limiting. The licensee stated that surveillance results from the Intermediate Shell Plate C-8009-3 also apply to the Upper Shell Plate C-8008-1, because the two plates were made from the same heat of material (Heat No. C8182). The licensee determined that this material was also not limiting. The NRC staff verified that the licensee's analysis incorporated data from the latest surveillance capsule pulled from the ANO-2 RPV (Capsule 284°), as documented in WCAP-18166-NP, Revision 0, "Analysis of Capsule 284 from the Entergy Operations, Inc. Arkansas Nuclear One, Unit 2 Reactor Vessel Radiation Surveillance Program," dated September 2016 (ADAMS Accession No. ML16293A584). Therefore, the NRC staff determined that the licensee's consideration of the RPV surveillance data is acceptable.

Since the NRC staff verified that the licensee correctly determined the ART values for the RPV beltline materials in accordance with RG 1.99, based on valid input parameters, the staff determined that the licensee's ART analysis of the RPV beltline region is acceptable.

The NRC staff verified that the licensee's proposed 54 EFPY P-T limits were calculated in accordance with WCAP-14040-A, Revision 4, and are based on an evaluation of the limiting RPV beltline shell material, including the 54 EFPY ART inputs documented above. However, the licensee stated that P-T limits were developed without margins for instrumentation error. The NRC staff noted that TS Bases 3/4.4.9 states that instrument uncertainty is added in station procedures. For the limiting beltline shell material, the NRC staff performed a set of confirmatory calculations to verify that the licensee's 54 EFPY P-T limits are consistent with WCAP-14040-A, Revision 4 and the ASME Code, Section XI, Appendix G. Using the licensee's thermal stress intensity factor (K_{IT}) values, the NRC staff was able to reproduce the licensee's P-T limits. Therefore, the NRC staff determined that the licensee's proposed 54 EFPY P-T limits for the limiting beltline shell material are acceptable.

3.2.3 Evaluation of the Licensee's Analysis of the RPV Inlet and Outlet Nozzles

The licensee provided ART calculations for the inlet and outlet nozzle materials based on 54 EFPY RPV nozzle fluence values from WCAP-18169-NP. The licensee determined that the 54 EFPY neutron fluence values are 7.96 x 10^{16} n/cm^2 (E > 1.0 MeV) at the lowest extent of the inlet nozzles and 9.80 x 10^{16} n/cm^2 (E > 1.0 MeV) at the lowest extent of the outlet nozzles. The licensee noted that these 54 EFPY neutron fluence values are conservative and bounding relative to the consideration of these nozzles for the 54 EFPY P-T limit curves. The licensee also provided the summary of the fracture-toughness related parameters in Table B-1 in WCAP-18169-NP. For initial RT_{NDT} values, the licensee stated that these values were determined using BTP 5-3, "Fracture Toughness Requirements," of NUREG-0800, Positions 1.1(3) (a) and 1.1(3) (b) (ADAMS Accession No. ML070850035), and that the most limiting value was chosen for each nozzle material. The licensee stated that, per RIS 2014-11, since the projected fluence values are less than 1 x 10^{17} n/cm^2, embrittlement of the nozzle material does not need to be considered. Therefore, the licensee stated that the initial RT_{NDT} values will be the ART values for the nozzle materials. The NRC staff confirmed that the neutron fluence values used for the inlet and outlet nozzles are consistent with those listed in WCAP-18169-NP. Furthermore, these neutron fluence values are conservatively based on the nozzle-to-shell weld location and were chosen at an elevation lower than the actual elevation of the postulated flaw, which is at the inside corner of the nozzle. Therefore, the NRC staff determined that the neutron fluence values are acceptable and that the licensee's ART values for the inlet and outlet nozzles are acceptable.

The licensee also stated that the nickel, manganese, and phosphorus weight percent of the inlet and out nozzles were obtained using the average of material-specific analyses. The copper content of the outlet nozzles were also determined using the average of available material-specific analyses. The licensee stated that, if the copper content of the inlet nozzle materials is needed in future evaluations, it will use the best-estimate copper weight percent value available from Section 4 of the NRC-approved BWRVIP (proprietary) report, BWRVIP-173-A, "Evaluation of Chemistry Data for BWR Vessel Nozzle Forging Materials." The NRC staff's review of the Westinghouse Report No. WCAP-18169-NP was limited to the evaluation of the inlet and outlet nozzle based on 54 EFPY. The NRC staff did not review the acceptability of using the best-estimate copper weight percent. The licensee's use of the best-estimate copper weight percent in embrittlement evaluations will be reviewed by the NRC staff, if applicable and as needed, in any future licensing action requests.

The licensee generated P-T limit curves for the inlet and outlet nozzles using the 54 EFPY ART values, based on a 100 °F per hour cooldown rate and a postulated inside surface 1/4T nozzle corner flaw. The licensee stated that the stress intensity factor correlations used for the nozzle corners were calculated based on the methodology provided in the Oak Ridge National Laboratory (ORNL) study, ORNL/TM-2010/246, "Stress and Fracture Mechanics Analyses of Boiling Water Reactor and Pressurized Water Reactor Pressure Vessel Nozzles - Revision 1," dated June 2012 (ADAMS Accession No. ML12181A162).

The specific nozzle total stress intensity factor (K_I) formulation described in the ORNL/TM-2010/246 report is based on a linear elastic fracture mechanics model that is generally considered to be applicable to postulated corner flaws in rounded corner nozzle forgings, irrespective of plant design. Based on the review of these analyses, the NRC staff finds that the licensee's methods for calculating the K_{IP} and K_{IT} values for the nozzles are acceptable.

The licensee stated that the resulting 54 EFPY nozzle P-T limit curves are less limiting than the proposed 54 EFPY P-T limits for cooldown conditions developed for the beltline. The licensee indicated that this demonstrates that the nozzle P-T limits are less restrictive, and the limiting RPV beltline shell material is controlling. The NRC staff performed confirmatory calculations using methods from the ORNL/TM-2010/246 report and confirmed that the calculations for the inlet and outlet nozzles are consistent with the ASME Code, Section XI, Appendix G, and 10 CFR Part 50, Appendix G, and that the resulting P-T limits are less restrictive than those calculated for the limiting beltline shell material. Therefore, the staff finds the licensee's assessment of the inlet and outlet nozzles to be acceptable.

3.2.4 Evaluation of the RPV Minimum Boltup Temperature

Table 1 of 10 CFR Part 50, Appendix G, includes minimum temperature requirements for the RPV, which must be incorporated into plant's P-T limit curves. For normal conditions, these minimum temperature requirements are established based on Footnote 2 to Table 1, which refers to the highest RT_{NDT} value "of the material in the RPV closure flange region that is highly stressed by the [closure head] bolt preload." Per the methodology, in WCAP-14040-A, the minimum boltup temperature should be 60 °F or the limiting unirradiated RT_{NDT} of the closure flange region, whichever is higher.

In the current P-T limits for 32 EFPY, the minimum boltup temperature is 50 °F. In its evaluation for 54 EFPY, the licensee stated that, since the limiting unirradiated RT_{NDT} of the closure flange region is below 60 °F, the minimum boltup temperature was revised to 60 °F. The NRC staff verified that the licensee's minimum boltup temperature of 60 °F, as established in the proposed TS P-T limit curves, is in compliance with the minimum temperature requirements of WCAP-14040-A and Table 1 of 10 CFR Part 50, Appendix G.

3.2.5 Evaluation of the Proposed Revision to Low Temperature Overpressure Protection (LTOP) Limits

As a result of the piping configuration modifications made during the ANO-2 replacement steam generator (SG) project, certain piping configurations were changed and the high pressure safety injection (HPSI) flow rate increased due to modifications made to the HPSI pump impellers. As part of this LAR, the licensee performed a reanalysis of the limiting LTOP events and included the effects of the modified piping configurations and higher HPSI flow rate to support the new P-T limits in TS 3.4.9.

Section 5.2.2, Revision 3, "Overpressure Protection," of NUREG-0800, dated March 2007 (ADAMS Accession No. ML070540076), specifies that the LTOP system be designed in accordance with the guidance of BTP 5-2, which specifies that the LTOP system be capable of relieving pressure during all anticipated over pressurization events at a rate sufficient to satisfy the TS limits while operating at low temperatures.

3.2.5.1 LTOP Analysis of Record (AOR)

Section 5.2.2.4 of the ANO-2 SAR, discusses the LTOP system, which is provided with redundant LTOP relief valves 2PSV-4732 and 2PSV-4742, and two LTOP isolation valves. The relief valve setpoint of less than or equal to 430 pounds per square inch gauge (psig) specified in TS LCO 3.4.12 was determined based on the results of the analysis of two events: (1) the most limiting energy addition event, a single idle reactor coolant pump (RCP) start with a secondary-to-primary temperature differential of 100 °F; and (2) the most limiting mass addition event, simultaneous injection to the RCS from one HPSI and three charging pumps resulting from an inadvertent safety injection actuation signal. The LTOP transient analysis showed that the maximum pressures are 539 pounds per square inch absolute (psia) and 522.2 psia for the limiting energy addition event and the limiting mass addition event, respectively. In the response to the NRC staff's RAI SRXB-1, in the letter dated August 1, 2018, the licensee indicated that the maximum pressures in the SAR discussed above were determined by two sets of analyses. The first set of analyses were performed with the OVERP code, which was expressively developed to treat water solid systems. The analyses assumed that: (1) a maximum of two HPSI pumps were aligned during LTOP operation; (2) the RCS was water solid at the time of the limiting transient; and (3) a nominal LTOP relief valve backpressure for liquid discharge of 100 psig was assumed to determine valve capacity. The peak pressures of 539 psia for the limiting energy addition event and 522.2 psia for the limiting mass addition event calculated by this set of the analyses were incorporated in the SAR.

The second set of the analyses was performed with the CENTS code, which is capable of modeling the steam volume and liquid volume in the pressurizer. The analyses assumed: (1) a maximum of one HPSI pump is aligned during LTOP operation; and (2) a maximum nominal pressurizer water volume of 910 cubic feet (ft^3) is maintained prior to starting of the first RCP during LTOP operation. The above LTOP conditions assumed in the second set of the analyses are consistent with the operating restrictions in the plant TSs. TS LCO 3.4.12, requires the LTOP system to be operable with a maximum of one HPSI pump capable of injecting into the RCS. The footnote to the APPLICABILITY section of TS LCO 3.4.12 further requires that when starting the first RCP, the maximum pressurizer water volume is 910 ft^3. The CENTS code included an input of the LTOP relief valve flow as a function of the backpressure, which was determined with a RELAP5 analysis of the piping network downstream of the LTOP relieve valve. The results of the analyses showed that the peak pressures in SAR Section 5.2.2.4 remained the limiting pressures.

3.2.5.2 LTOP Transient Reanalysis

Page 5 of the enclosure to the LAR dated November 20, 2017, stated that the limiting LTOP events were reanalyzed to include the effects of the modified piping configurations and a higher HPSI flowrate. The reanalysis showed that the current LTOP relief valve setting of less than or equal to 430 psig specified in TS LCO 3.4.12 remained valid to meet the 54 EFPY P-T limits. In the response to NRC's RAI SRXB-1 in the letter dated August 1, 2018, the licensee provided, in Figure 1, the backpressure values as a function of the relieving flow for the AOR and reanalysis.

The backpressures were calculated based on RELAP models of the piping configuration assuming pressurizer condition consistent with the TS 3.4.12 requirements with a saturated liquid volume of 910 ft^3 and a saturated steam volume of 323 ft^3 for two cases of initial pressurizer saturated liquid corresponding to saturation temperatures of 417.4 °F at 300 psia, and 444.6 °F at 400 psia. The results showed that the system backpressures versus flow rates for liquid discharge in the LTOP transient AOR and reanalysis were not significantly different, even though the piping configurations were modified as a result of the piping configuration modifications made during the replacement SG project. The licensee also provided, in Figure 2 of the RAI response dated August 1, 2018, the RCS pressure versus the flow rate for one HPSI pump for the AOR and reanalysis. The figure showed that one HPSI flow was higher for the LTOP transient reanalysis due to modifications made to the HPSI pump impellers.

LTOP Transient Reanalysis for Mass Addition Event

In the RAI response dated October 10, 2018, the licensee discussed the LTOP transient reanalysis for a mass addition event. The mass addition reanalysis assumed that sometime after the relief valve opens, an equilibrium between the mass input and valve discharge would occur. The equilibrium pressure was determined at the intersection of the relief valve capacity curve with the mass input curve. For conservatism to assure a higher calculated peak pressure, the licensee assumed that when the determined equilibrium pressure was less than the valve maximum opening pressure, the peak transient pressure was assumed to be equal to the maximum valve opening pressure, and when the determined equilibrium pressure was greater than the valve maximum opening pressure, the equilibrium pressure became the peak transient pressure. The maximum valve opening pressure was identified as 487.7 psia, which was equal to 110 percent of the relief valve setpoint of 430 psig. The RAI response dated August 1, 2018, clarified that the mass input used in the analysis of the mass addition event was the total amount of the charging flow rate and HPSI flow rate with allowances added for fluid expansion due to heat sources including decay heat, heat from two RCPs, and pressurizer heaters at their maximum heat rate. This method used in the reanalysis was the same as the LTOP mass addition AOR. The relief valve inlet piping pressure drop was calculated based on the flow rate corresponding to the equilibrium pressure. This inlet piping pressure drop was added to the peak transient pressure to obtain the peak transient pressure at the pressurizer location. The peak pressure at the pressurizer location for the limiting mass addition event was determined to be 498 psia.

The NRC staff found that the method used for the mass addition analysis was the same as the mass addition AOR, and that the assumptions used in the analysis were conservative, resulting in a highest peak transient pressure; therefore, the staff concludes that the reanalysis is acceptable.

LTOP Transient Reanalysis for Energy Addition Event

The RAI response in the letter dated August 1, 2018, discussed the LTOP energy addition analysis. The LTOP reanalysis was performed using the CENTS code to evaluate the system transient for the most limiting energy addition event, which was previously identified by the licensee as an event initiated from a single idle RCP start with a maximum nominal water inventory of 910 ft^3 in the pressurizer and with a secondary-to-primary temperature differential of 100 °F pressure. The input to the CENTS code included the LTOP relief valve flow rates calculated by using RELAP5 with inclusion of the effects of a calculated inlet piping pressure drop. The results in the RAI response dated October 10, 2018, showed that the peak calculated pressure from the CENTS calculation for the energy addition event was 466 psia, which was lower than 487.7 psia (110 percent of the relief valve setpoint of 430 psig). For conservatism to assure a higher calculated peak pressure, the peak transient pressure at the pressurizer location was based on 110 percent of the relief valve setpoint plus a maximum inlet piping pressure, which was based on the maximum water flow rate through the relief valve. This approach is consistent with the LTOP energy addition AOR. The peak pressure at the pressurizer location thus determined for the limiting energy addition event was 497.5 psia.

Since the method used for the energy addition analysis was the same as the energy addition AOR, the analysis was performed for the most limiting energy addition event, and that the assumptions used in the analysis were conservative, resulting in a highest peak transient pressure, the NRC staff concludes that the reanalysis was acceptable.

Current Peak LTOP Transient Pressure Compliance with the P-T Limits

A comparison of the results of the LTOP transient reanalysis discussed in the above Subsections 3.2.1.1 and 3.2.1.2 of this safety evaluation (SE) showed that the peak pressure of 498 psia from the reanalysis of the limiting mass addition event was the maximum peak transient pressure at the pressurizer for the LTOP conditions. A pressure drop in the pressurizer surge line of 2.2 pounds per square inch was added to the LTOP transient peak pressure of 498 psia prior to addressing the compliance of the calculated peak transient pressure with the P-T limits. As shown in TS Figures 3.4-2A and 3.4-2B, the limiting P-T points were 588 psia at 60 °F for the heatup curve and 543 psia at 60 °F for the cooldown curve, respectively.

Since the LTOP transient peak pressure of 500.2 (498 plus 2.2) psia was within both the limiting P-T values of 588 psia for the heatup curve and 543 psia for the cooldown curve, the NRC staff determined that the proposed P-T limits were adequately supported by the LTOP transient reanalysis in meeting GDC 15 as it is related to the requirements of the reactor coolant pressure boundary, and BTP 5-2 as it is related to the guidance of over pressurization protection of PWRs while operating at low temperatures. Therefore, the NRC staff concludes that the P-T limits in TS Figures 3.4-2A and 3.4-2B are acceptable.

Basis for Not Including 100 °F Assumption in LTOP TSs

Page 8 of the RAI response in the letter dated August 1, 2018, indicates that for the reanalysis of the most limiting mass and energy addition events, a maximum nominal pressurizer level of 910 ft^3 was assumed as an initial pressurizer water level during LTOP operation. The NRC staff noted that the operating limit of the pressurizer water volume of 910 ft^3 assumed in the LTOP analyses was included in a footnote to the Applicability of ANO-2 TS LCO 3.4.12.

Also, page 11 of the RAI response states, in part, that the limiting energy addition event assumed that the SGs were filled with water at the initial temperature of 100 °F above the primary system T_{cold}. However, it was not clear whether this limitation was identified in the ANO-2 TSs.

In an RAI, the NRC staff requested the licensee to identify the location in the ANO-2 TSs where the operating limit for the SG water temperature difference of 100 °F referenced above was specified. If this limitation was not currently defined in the TSs, the licensee was requested to address compliance with the requirements of 10 CFR 50.36(c)(2)(ii)(B), Criterion 2, which requires inclusion of an LCO in TSs for plant process variables, design features, or operating restrictions that are used as an initial condition of a design transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. In its response to the RAI in the letter dated October 10, 2018, the licensee indicated that the 100 °F temperature difference was an assumed value used in the analysis to bound the result of the energy addition event and this value was unlikely to occur in LTOP applicability modes. During shutdown modes when LTOP requirements were applicable, the water inventory in each SG was supplied from the condensate storage tank (CST). During winter operations, the temperature of water in the CST is controlled by the auxiliary steam system or electric heaters. The heating systems are operated to control CST temperature between approximately 80 °F and 90 °F. During summer operations, the CST water temperature was maintained by the ambient temperature. TS 3.4.9 specifies the RCS P-T limit. Its associated revised (clean) copies of TS Figures 3.4-2A, 3.4-2B, and 3.4-2C showed that the minimum boltup temperature of the reactor vessel head is 60 °F. Since the TS LTOP requirements were applicable when the reactor vessel head was installed, the feedwater source for the SGs during LTOP conditions was unlikely to increase above this minimum allowable RCS temperature by greater than 100 °F during the hottest summer months, which recorded the highest local temperature of less than 115 °F. Based on the above discussion, the NRC staff concludes that the subject 100 °F temperature difference would be unlikely to occur in LTOP applicable modes, and the 100 °F LTOP assumption does not meet the intent of 10 CFR 50.36(c)(2)(ii)(B), Criterion 2, for inclusion in TSs. The intent of Criterion 2 for TS inclusion is applicable to the limits of the variables, designed features, or operating restrictions that are used as initial conditions in the safety analyses and can be exceeded without controls and verifications. Therefore, the NRC staff concludes that the licensee's proposal to not add the 100 °F temperature difference to ANO2 LTOP TS was acceptable.

Reevaluation of the LTOP Enable Temperature

In its response to an RAI in the letter dated August 1, 2018, the licensee stated that the LTOP enable temperature in TS LCO 3.4.12 was reevaluated per ASME Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure System Requirements, Section XI, Division 1," based on the limiting 1/4T ART. ASME Code Case N-641 presents alternative procedures for calculating P-T relationships and LTOP system effective temperatures (T_e), and allowable pressures. ASME Code Case N-641 provides temperature and pressure conditions that LTOP systems may follow to provide protection against failure during reactor startup and shutdown operation due to LTOP events. The conditions state that the LTOP systems shall be effective below the higher temperature determined in accordance with: (1) a coolant temperature of 200 °F; and (2) a coolant temperature corresponding to a reactor vessel metal temperature, for all vessel beltline materials. The code case provides two sets of equations to calculate the coolant temperature corresponding to a reactor vessel metal temperature that is based on the enable temperature, either of which the licensee can use.

In its response, the licensee provided its calculations, including its parameter inputs, for both equations to calculate the enable temperature and the coolant temperature corresponding to the reactor vessel metal temperature. In both cases, the licensee calculated the coolant temperature to be below 200 °F. Therefore, to identify the minimum enable temperature, the licensee took the highest temperature determined in accordance with the conditions of the code case, which was determined to be 200 °F. The licensee then added an instrument uncertainty of 20 °F and concluded that the LTOP system shall be effective below the temperature of 220 °F for 54 EFPY.

To verify the licensee calculated LTOP enable temperatures provided in the response, the NRC staff performed confirmatory calculations in accordance with ASME Code Case N-641. The staff's confirmatory calculations were consistent with the licensee's response which determined the highest minimum enable temperature to be 200 °F. Therefore, the NRC staff determined that the licensee's LTOP enable temperature for 54 EFPY is acceptable and the staff's concerns in the RAI are resolved.

3.2.5.3 Conclusion of NRC Staff Technical Evaluation of Proposed Revision to LTOP Limit

Based on the above discussion, the NRC staff concludes that: (1) the revised P-T limits in TS 3.4.9 were adequately supported by the LTOP reanalysis, (2) the revised TS 3.4.9 in combination with current TS 3.4.12 would reasonably assure that the reanalysis remained valid in meeting the requirements GDC 15 as it relates to the requirements of the reactor coolant pressure boundary, and BTP 5-2 as it is related to the guidance of overpressure protection of PWRs while operating at low temperatures; and (3) the proposed revised P-T limits in TS 3.4.9 will continue to meet the requirements of 10 CFR 50.36(c)(2). Therefore, the NRC staff determined that the proposed P-T limits in TS 3.4.9 are acceptable for conditions applicable to 54 EFPY, an exposure that corresponds to roughly 60 calendar years of operation.

3.2.6 Evaluation of the Proposed Revision to Pressurized Thermal Shock

To verify the licensee calculated RT_{PTS} for the limiting beltline materials provided in the LAR submittal dated November 20, 2017, the NRC staff performed confirmatory calculations in accordance with the requirements of 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." The staff's confirmatory calculations were consistent with the licensee's submittal, which stated that the controlling material for PTS would be the Lower Shell Plate C-8010-1 with a predicted RT_{PTS} value of 122.4 °F at 54 EFPY. This is below the regulatory screening criteria for PTS, which is 270 °F for plates. The staff concludes that the licensee's PTS assessment is in accordance with the requirements of 10 CFR 50.61.

3.2.7 Evaluation of Upper Shelf Energy

Regarding USE, in its LAR dated November 20, 2017, the licensee stated, in part:

> The limiting USE value for the ANO-2 RPV at 54 EFPY is 60.2 ft-lb. This value corresponds to the Upper Shell Plate C-8008-2. Section 5 of Attachment 3 presents the complete results of the USE evaluation. As can be seen in Section 5, all of the beltline and extended beltline materials in the ANO-2 RPV are projected to remain above the USE screening criteria value of 50 ft-lb through 54 EFPY.

Section 5 of WCAP-18169-NP provides the licensee's evaluation of CFs and does not contain the evaluation of USE. Therefore, the NRC staff requested in an RAI, by letter dated May 10, 2018, that the licensee provide the evaluation and results for USE.

In its response to the RAI by letter dated August 1, 2018, the licensee provided a summary and results of its USE evaluation. The licensee stated that the projected 54 EFPY USE values were calculated using the methodology in RG 1.99, Revision 2, as well as the plant surveillance data from the latest ANO-2 surveillance capsule analysis report. The licensee provided the results of its USE evaluation in Table 1 of the response.

The NRC staff reviewed the licensee's USE evaluation in accordance with the requirements of 10 CFR Part 50, Appendix G. The staff confirmed that the limiting USE value, which corresponds to the Upper Shell Plate C-8008-2, as well as the USE values of the remaining beltline and extended beltline materials are projected to be above 50 ft-lb. The results, which were calculated using a staff approved methodology, meets the regulatory screening criteria for USE. Therefore, the staff concludes that the licensee's USE evaluation is in accordance with the requirements of 10 CFR Part 50, Appendix G.

3.3 NRC Staff Technical Evaluation Conclusion

Based on its evaluation in Section 3.2 of this SE, the NRC staff determined the following:

(1) The licensee's proposed 54 EFPY TS P-T limit curves in TS Figures 3.4-2A, 3.4-2B, and 3.4-2C meet the criteria of the ASME Code, Section XI, Appendix G, and are in compliance with the fracture toughness requirements of 10 CFR 50.60 and 10 CFR Part 50, Appendix G through 54 EFPY.

(2) The licensee's proposed change to update the minimum boltup temperature of 60 °F, as established in the proposed TS P-T limit curves, is in compliance with the minimum temperature requirements of WCAP-14040-A and Table 1 of 10 CFR Part 50, Appendix G.

(3) Concerning the LTOP reanalysis: (1) the revised P-T limits in TS 3.4.9 are adequately supported by the LTOP reanalysis, (2) the revised TS 3.4.9 in combination with current TS 3.4.12 provide reasonable assurance that the reanalysis remains valid in meeting GDC 15 as it relates to the requirements of the reactor coolant pressure boundary, and BTP RSB 5-2 as it relates to the guidance of overpressure protection of PWRs while operating at low temperatures; and (3) the proposed revised P-T limits in TS 3.4.9 will continue to meet 10 CFR 50.36(c)(2).

(4) The licensee's proposed revision to its PTS assessment is in accordance with the requirements of 10 CFR 50.61. The licensee's USE evaluation is in accordance with the requirements of 10 CFR Part 50, Appendix G.

(5) The NRC staff review established that the fluence calculations supporting the requested update to the PT limit curves contained in ANO-2 TS 3.4.9 are acceptable, because they adhere to the guidance contained in RG 1.190.

Therefore, the NRC staff concludes that Entergy's proposed TS revisions for the P-T limit curves and LTOP protection limits are acceptable for incorporation into the ANO-2 TSs for 54 EFPY, and that there is reasonable assurance that the applicable regulatory requirements will continue to be met.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment on November 7, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 February 27, 2018 (83 FR 8514). 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.

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Date: November 27, 2018

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 - ISSUANCE OF AMENDMENT RE: UPDATING THE REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS (EPID L-2017-LLA-0396) DATED NOVEMBER 27, 2018

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