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

April 24, 2015

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: UPDATE THE REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS AND THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM LIMITS (TAC NO. MF5292)

Dear Sir or Madam:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 254 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 21, 2014, as supplemented by letters dated February 6, March 10, March 25, and April 7, 2015.

The amendment revises TS 3.4.3, "RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits," TS 3.4.9, "Pressurizer," TS 3.4.10, "Pressurizer Safety Valves," and TS 3.4.11, "Low Temperature Overpressure Protection (LTOP) System," to update the RCS P/T limits to 54 effective full power years (EFPY). The current P/T limits are applicable up to 31 EFPY.

In its analysis for the updated P/T limits, ANO-1 utilized an alternate methodology to determine the initial nil-ductility reference transition temperature (RT_{NDT}) to that prescribed in Appendix G to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 and 10 CFR 50.61. By letter dated March 24, 2014, the licensee requested an exemption to certain portions of 10 CFR 50.61 and Appendix G to 10 CFR Part 50. The exemption was approved by the NRC staff on March 16, 2015 (80 FR 15634; March 24, 2015).

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

Sincerely,

Andrea E. George, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 254 to DPR-51
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 254
Renewed License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee), dated November 21, 2014, as supplemented by letters dated February 6, March 10, March 25, and April 7, 2015, 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 license 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. DPR-51 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 254, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

3. The license 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

Michael T. Markley

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

Attachment:
Changes to the Renewed Facility Operating License No. DPR-51 and Technical Specifications

Date of Issuance: April 24, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 254

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Renewed Facility Operating License No. DPR-51 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
3.4.3-4	3.4.3-4
3.4.3-5	3.4.3-5
3.4.3-6	3.4.3-6
3.4.9-1	3.4.9-1
3.4.10-1	3.4.10-1
3.4.10-2	3.4.10-2
3.4.11-1	3.4.11-1

(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 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;

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

(1) Maximum Power Level

EOI is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 254, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

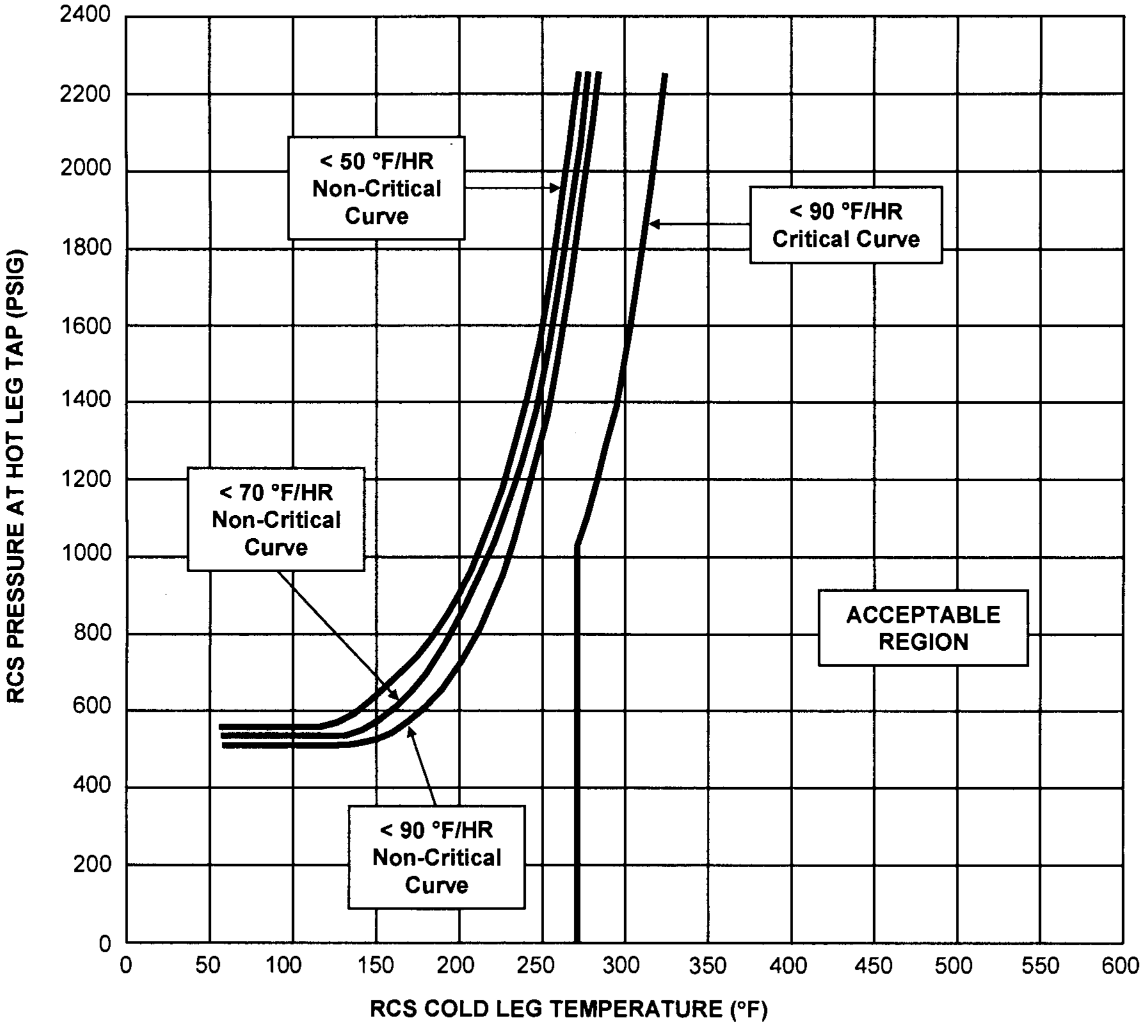
(3) Safety Analysis Report

The licensee's SAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 14, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than May 20, 2014.

(4) Physical Protection

EOI shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Arkansas Nuclear One Physical Security Plan, Training and Qualifications Plan, and Safeguards Contingency Plan," as submitted on May 4, 2006.

FIGURE 3.4.3-1
RCS Heatup Limitations to 54 EFPY



Notes:

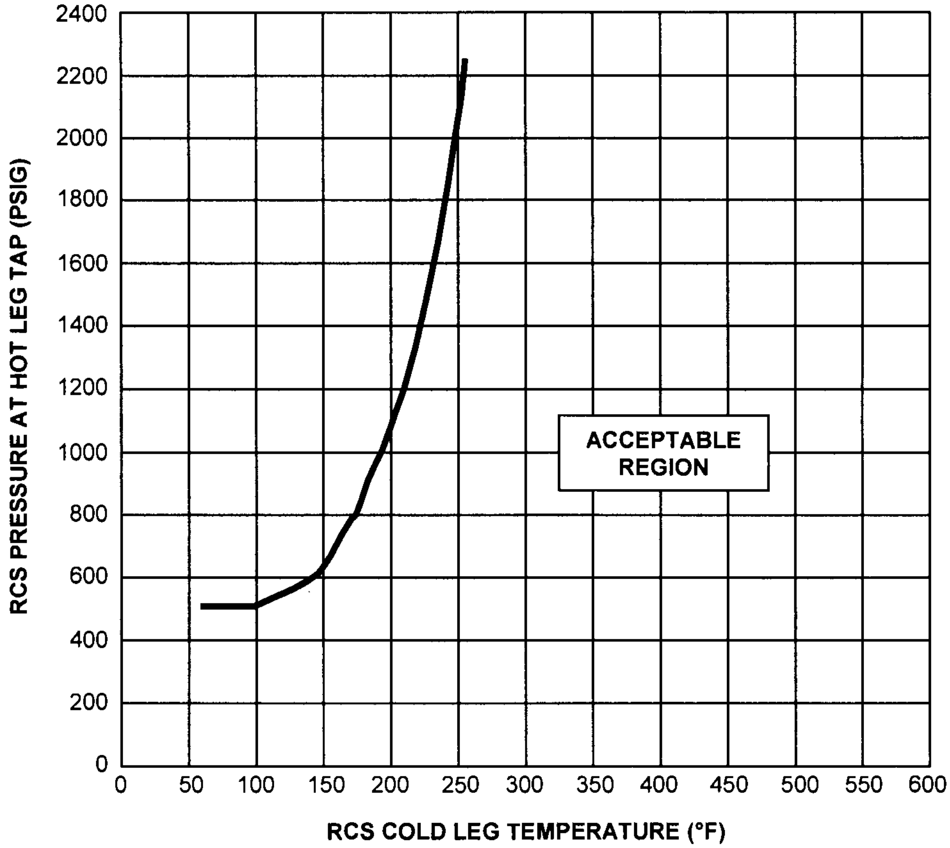
1. These curves are not adjusted for instrument error and shall not be used for operation.
2. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 300 °F	None
300 °F ≥ T ≥ 225 °F	≤ 3
225 °F > T ≥ 84 °F	≤ 2
T < 84 °F	No RCPs operating

4. Allowable Heatup Rates:

RCS TEMP	H/U RATE
60 °F < T ≤ 84 °F	≤ 15 °F/HR
T > 84 °F	As allowed by applicable curve

FIGURE 3.4.3-2
RCS Cooldown Limits to 54 EFPY



Notes:

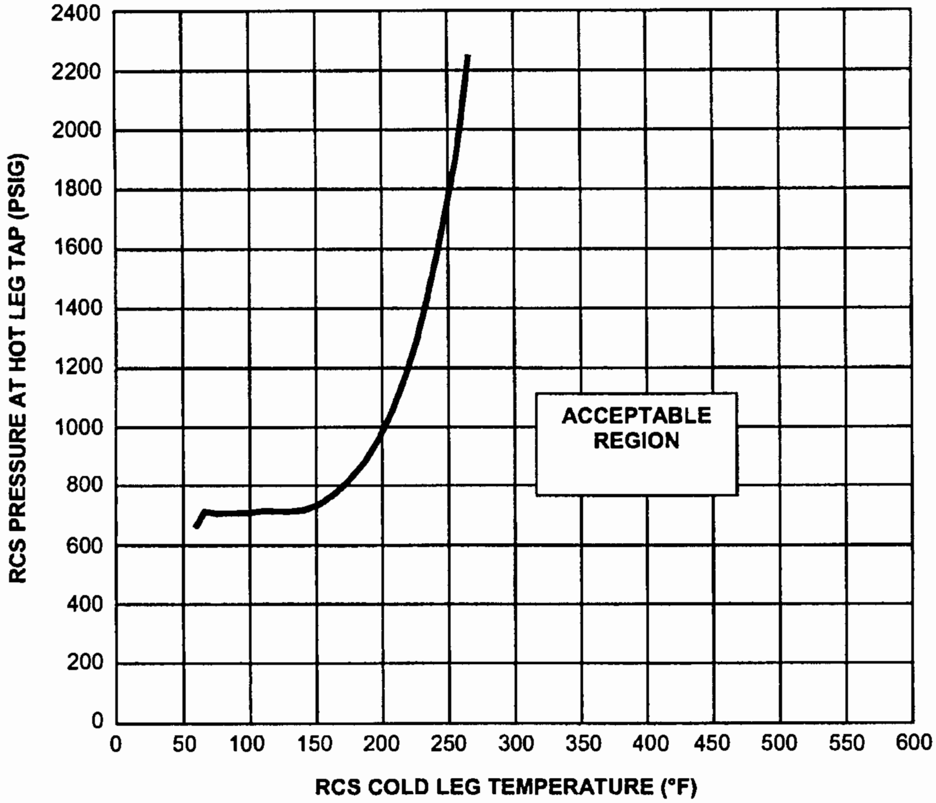
1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25 °F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 255 °F	None
150 °F ≤ T ≤ 255 °F	≤ 2
T < 150 °F	No RCPs operating

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280 °F	100 °F/HR	≤ 50 °F in any 1/2 HR
280 °F > T ≥ 150 °F	50 °F/HR	≤ 25 °F in any 1/2 HR
T < 150 °F	25 °F/HR	≤ 25 °F in any 1 HR

FIGURE 3.4.3-3
RCS Inservice Hydrostatic Test H/U & C/D Limits to 54 EFPY



Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. All Notes on Figure 3.4.3-1 are applicable for heatups. This curve is based on a heatup rate of < 90 °F/HR.
3. All Notes on Figure 3.4.3-2 are applicable for cooldowns.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 320 inches; and

b. A minimum of 126 kW of Engineered Safeguards (ES) bus powered pressurizer heaters OPERABLE.

---NOTE---
OPERABILITY requirements on pressurizer heaters do not apply in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > 259 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limits.	A.1 Restore level to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with RCS temperature ≤ 259 °F.	6 hours 24 hours
C. Capacity of ES bus powered pressurizer heaters less than limit.	C.1 Restore pressurizer heater capacity.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE.

-----NOTES-----

1. Only one pressurizer safety valve is required to be OPERABLE in MODE 3, and in MODE 4 with RCS temperature > 259 °F.
2. The lift settings are not required to be within limits for entry into MODE 3 or the applicable portions of MODE 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.
3. Not applicable in MODE 3, and in MODE 4 with RCS temperature > 259 °F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.
4. The provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > 259 °F.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > 259 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable in MODES 1 or 2.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two pressurizer safety valves inoperable in MODES 1 or 2.	B.1 Be in MODE 3.	6 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required pressurizer safety valve inoperable in MODE 3 or MODE 4 with RCS temperature > 259 °F.	C.1 Be in MODE 4 with RCS temperature ≤ 259 °F.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each required pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.11 An LTOP System shall be OPERABLE with high pressure injection (HPI) deactivated and the core flood tanks (CFTs) isolated and:

-----NOTES-----

1. HPI deactivation and CFT isolation not applicable during ASME Section XI testing.
2. HPI deactivation not applicable during fill and vent of the RCS.
3. HPI deactivation not applicable during emergency RCS makeup.
4. HPI deactivation not applicable during valve maintenance.
5. CFT isolation is only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature curves provided in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

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a. Pressurizer level such that the unit is not in a water solid condition and an OPERABLE electromatic relief valve (ERV) with a setpoint of ≤ 508 psig; or

   -----NOTES-----
   1. Pressurizer level not applicable as allowed by Emergency Operating Procedures.
   2. Pressurizer level not applicable during system hydrotest.

   ---

b. The RCS depressurized and the RCS open.

APPLICABILITY: MODE 4 with RCS temperature ≤ 259 °F,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer level not within required limits.	A.1 Restore pressurizer level to within required limits.	1 hour

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 254 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By application dated November 21, 2014, as supplemented by letters dated February 6, March 10, March 25, and April 7, 2015 (Agencywide Documents Access and Management System (ADAMS Accession Nos. ML14330A246, ML15041A065, ML15071A054, ML15086A019, and ML15097A517, respectively), Entergy Operations, Inc. (Entergy, the licensee), requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit No. 1 (ANO-1). Portions of the letters dated February 6, March 10, and March 25, 2015, contain sensitive unclassified non-safeguards information (proprietary) and, accordingly, have been withheld from public disclosure.

The supplemental letters dated February 6, March 10, March 25, and April 7, 2015, 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) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 6, 2015 (80 FR 524).

The license amendment request (LAR) proposed changes, which would replace the current reactor pressure vessel (RPV) pressure and temperature (P/T) limits in TS 3.4.3, "RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits," applicable to 31 effective full power years (EFPY), with new P/T limits applicable to 54 EFPY for ANO-1. In addition, the LAR proposed to change the applicable temperatures and pressures for the Pressurizer (TS 3.4.9, "Pressurizer"), the Pressurizer Safety Valves (TS 3.4.10, "Pressurizer Safety Valves") and the Low Temperature Overpressure Protection (LTOP) System (TS 3.4.11, "Low Temperature Overpressure Protection (LTOP) System"). The licensee revised the P/T limits based on NRC-approved AREVA topical report (TR) BAW-10046A, Revision 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of [Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50)], Appendix G," June 1986 (ADAMS Legacy Accession No. 8607230131).

In its analysis for the updated P/T limits, ANO-1 utilized an alternate methodology, described in NRC-approved AREVA TR BAW-2308, Revisions 1-A and 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials" (NRC Safety Evaluations (SEs) dated August 4, 2005, and March 24, 2008, available at ADAMS Accession Nos. ML052070408 and ML080770349, respectively), to determine the initial nil-ductility reference transition temperature (RT_{NDT}) to that prescribed in Appendix G, "Fracture Toughness Requirements," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 and 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." By letter dated March 20, 2014 (ADAMS Accession No. ML14083A640), the licensee requested an exemption to certain portions of 10 CFR 50.61, and 10 CFR Part 50, Appendix G. The exemption was approved by the NRC staff in a letter dated March 16, 2015 (ADAMS Accession No. ML15056A367), and published in the *Federal Register* on March 24, 2015 (80 FR 15634).

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in 10 CFR 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls.

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. The NRC staff evaluates the acceptability of a facility's proposed P/T limits based on the following NRC regulations: Appendix G to 10 CFR Part 50; Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50; and 10 CFR 50.61.

Section IV.A.1 of Appendix G to 10 CFR Part 50 requires that RPV beltline materials must have upper-shelf energy (USE) of no less than 75 foot-pounds (ft-lb) initially and 50 ft-lb throughout the life of the vessel, "unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code."

Section IV.A.1 of Appendix G to 10 CFR Part 50 requires that facility P/T limit curves for the RPV be at least as conservative as those obtained by applying the linear elastic fracture mechanics methodology of Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Appendix H to 10 CFR Part 50 establishes methodologies for determining the increase in transition temperature and the decrease in USE resulting from neutron radiation.

The proposed 54 EFPY P/T limits for ANO-1 have been established based on the 2001 Edition through the 2002 Addenda of the ASME Code which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G. The regulations in 10 CFR Part 50, Appendix G, specify the fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB 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 regulations in 10 CFR Part 50, Appendix G, also impose minimum temperature requirements based on the limiting properties of the materials in the RPV closure flange region that are highly stressed by bolt preload when system pressure is at or above 20 percent of the preservice hydrostatic test pressure.

The regulations in 10 CFR 50.61 establish regulatory requirements designed to protect against fracture resulting from an event or transient in pressurized-water reactors (PWRs) causing severe overcooling concurrent with or followed by significant pressure in the reactor vessel (pressurized thermal shock or PTS). The regulations in 10 CFR 50.61 provide screening criteria for PTS, and stipulate that the plant may not continue to operate without justification if the calculated PTS reference temperature (RT_{PTS}) is above the screening criteria, which is established to be 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials.

The NRC staff used the following guidance in its review of this LAR:

- Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988 (ADAMS Accession No. ML003740284);

- Generic Letter (GL) 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)," dated March 6, 1992 (ADAMS Accession No. ML031070438);

- GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," dated May 19, 1995 (ADAMS Accession No. ML031070449); and

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock" (ADAMS Accession No. ML070380185).

GL 92-01, Revision 1, requested that licensees submit the RPV data for their plants to the NRC staff for review, and GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their requirements related to facility RPV material surveillance programs. RG 1.99, Revision 2, contains RPV integrity evaluations. SRP Section 5.3.2 provides an acceptable method for determining the P/T limit curves for ferritic materials in the beltline of the RPV based on the ASME Code, Section XI, Appendix G methodology.

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001 (ADAMS Accession No. ML010890301), 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 Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. In consideration of the guidance set forth in RG 1.190, GDC 14, "Reactor coolant pressure boundary," GDC 30, "Quality of reactor coolant pressure boundary," and GDC 31, "Fracture prevention of reactor coolant pressure boundary," are applicable to this LAR. GDC 14 requires the design, fabrication, erection, and testing of the

RCPB so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30 requires, among other things that components comprising the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31 pertains to the design of the RCPB, stating:

> 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.

Regarding the applicability of the GDC to ANO-1, the licensee stated the following in its LAR dated November 21, 2014:

> The construction permit for ANO-1 was issued by the Atomic Energy Commission (AEC) on December 6, 1968, and an operating license was issued on May 21, 1974. The ANO-1 operating license was issued based on compliance with the proposed GDC published by the AEC in [*Federal Register* notice dated July 11, 1967 (32 FR 10213)] (hereinafter referred to as "draft GDC"). The AEC published the final rule that added Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," in [*Federal Register* notice dated February 20, 1971 (36 FR 3255)] (hereinafter referred to as "final GDC" or "GDC"). In accordance with [NRC Staff Requirements Memo dated September 18, 1992[1]], the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971, which includes ANO-1.
>
> ANO-1 Safety Analysis Report (SAR) section 1.4.10 incorporates the current GDC 14. SAR Section 1.4.26 discusses GDC 30, and GDC 31 is discussed in SAR Section 1.4.27.

Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," issued October 2014 (ADAMS Accession No. ML14149A165), 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 1×10^{17} n/cm^2 (E > 1 MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period.

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[1] U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum, "SECY-92-223 – Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736).

Additionally, the LTOP system controls RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the pressure and temperature limits required by 10 CFR Part 50, Appendix G.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

TS 3.4.3, RCS Pressure and Temperature (P/T) Limits, Figure 3.4.3-1

The current title of Figure 3.4.3-1, "RCS Heatup Limitations to 31 EFPY" would be changed to "RCS Heatup Limitations to 54 EFPY."

The changes to the RCS P/T heatup limitations curves in Figure 3.4.3-1 reflect an updated analysis for protection against brittle fracture during plant operation from 31 EFPY to 54 EFPY.

TS 3.4.3, RCS Pressure and Temperature (P/T) Limits, Figure 3.4.3-2

The current title of Figure 3.4.3-2, "RCS Cooldown Limits to 31 EFPY" would be changed to "RCS Cooldown Limits to 54 EFPY."

The changes to the RCS P/T cooldown limit curve in Figure 3.4.3-2 reflect an updated analysis for protection against brittle fracture during plant operation from 31 EFPY to 54 EFPY.

Current Figure 3.4.3-2, Note 3 states:

3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 255 °F	None
150 °F ≤ T ≤ 255 °F	≤ 2 (See Note 5)
T < 150 °F	No RCPs operating

Revised Figure 3.4.3-2, Note 3 would state:

3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 255 °F	None
150 °F ≤ T ≤ 255 °F	≤ 2
T < 150 °F	No RCPs operating

Current Figure 3.4.3-2, Note 4 states:

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280 °F	100 °F/HR	≤ 50 °F in any 1/2 HR
280 °F > T ≥ 150 °F	50 °F/HR (Note 5)	≤ 25 °F in any 1/2 HR
T < 150 °F	25 °F/HR	≤ 25 °F in any 1 HR

Revised Figure 3.4.3-2, Note 4 would state:

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280 °F	100 °F/HR	≤ 50 °F in any 1/2 HR
280 °F > T ≥ 150 °F	50 °F/HR	≤ 25 °F in any 1/2 HR
T < 150 °F	25 °F/HR	≤ 25 °F in any 1 HR

Current Figure 3.4.3-2, Note 5, which states "If RCPs are operated < 200 °F, then the RCS cooldown rate from 150 °F ≤ T ≤ 180 °F is reduced to 30 °F in 15 hours," would be deleted.

TS 3.4.3, RCS Pressure and Temperature (P/T) Limits, Figure 3.4.3-3

The current title of Figure 3.4.3-3, "RCS Inservice Hydrostatic Test H/U & C/D Limits to 31 EFPY" would be changed to "RCS Inservice Hydrostatic Test H/U & C/D Limits to 54 EFPY."

The changes to the RCS inservice hydrostatic test heatup and cooldown limits curve in Figure 3.4.3-3 reflect an updated analysis for protection against brittle fracture during plant operation from 31 EFPY to 54 EFPY.

TS 3.4.9, Pressurizer

The Applicability for current LCO 3.4.9 states:

MODES 1, 2, and 3,
MODE 4 with RCS temperature > 262 °F.

The revised Applicability for LCO 3.4.9 would state:

MODES 1, 2, and 3,
MODE 4 with RCS temperature > 259 °F.

Current Required Action B.2 states:

Be in MODE 4 with RCS temperature ≤ 262 °F.

Revised Required Action B.2 would state:

Be in MODE 4 with RCS temperature ≤ 259 °F.

TS 3.4.10, Pressurizer Safety Valves

Current LCO 3.4.10, Note 1 states:

Only one pressurizer safety valve is required to be OPERABLE in MODE 3, and in MODE 4 with RCS temperature > 262 °F.

Revised LCO 3.4.10, Note 1 would state:

Only one pressurizer safety valve is required to be OPERABLE in MODE 3, and in MODE 4 with RCS temperature > 259 °F.

Current LCO 3.4.10, Note 3 states:

Not applicable in MODE 3, and in MODE 4 with RCS temperature > 262 °F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Revised LCO 3.4.10, Note 3 would state:

Not applicable in MODE 3, and in MODE 4 with RCS temperature > 259 °F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Current LCO 3.4.10, Note 4 states:

The provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > 262 °F.

Revised LCO 3.4.10, Note 4 would state:

The provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > 259 °F.

Current LCO 3.4.10 Condition C states:

Required pressurizer safety valve inoperable in MODE 3 or MODE 4 with RCS temperature > 262 °F.

Revised LCO 3.4.10 Condition C would state:

Required pressurizer safety valve inoperable in MODE 3 or MODE 4 with RCS temperature > 259 °F.

Current LCO 3.4.10 Required Action C.1 states:

> Be in MODE 4 with RCS temperature ≤ 262 °F.

Revised LCO 3.4.10 Condition C would state:

> Be in MODE 4 with RCS temperature ≤ 259 °F.

TS 3.4.11, Low Temperature Overpressure Protection (LTOP) System

Current LCO 3.4.11.a states, in part:

> Pressurizer level such that the unit is not in a water solid condition and an OPERABLE electromatic relief valve (ERV) with a setpoint of < 460 psig; or

Revised LCO 3.4.11.a would state, in part:

> Pressurizer level such that the unit is not in a water solid condition and an OPERABLE electromatic relief valve (ERV) with a setpoint of < 508.0 psig; or

The Applicability for current LCO 3.4.11 states:

> MODE 4 with RCS temperature ≤ 262 °F,
> MODE 5,
> MODE 6 when the reactor vessel head is on

Applicability for revised LCO 3.4.11 would state:

> MODE 4 with RCS temperature ≤ 259 °F,
> MODE 5,
> MODE 6 when the reactor vessel head is on

3.2 Licensee's Evaluation of the ANO-1 P/T Limits

Attachment 1 to the licensee's LAR submittal dated November 21, 2014, discusses the P/T limits, LTOP limits, and PTS assessment, with which the licensee proposed to bound the operation of ANO-1 to 54 EFPY (i.e., the end of the current period of extended operation). The LAR also discussed the USE and EMA analyses for which the licensee stated that the calculations "remain bounding for close to 54 EFPY." Attachments 2 and 3 to the November 21, 2014, submittal provided markup and clean versions of the TSs that are to be updated as a result of the LAR. Attachment 4 to the LAR contained ANP-3300, Revision 1, "Pressure-Temperature Limits at 54 EFPY," November 2014 (ADAMS Accession No. ML14330A250), which provides a detailed technical basis for the P/T limits. Attachment 5 to the LAR provided the RT_{PTS} values for all the RPV beltline materials (ADAMS Accession No. ML14330A250).

With respect to P/T limits determination, Attachment 1 to the LAR provided a high-level summary of the technical and regulatory bases, since a more detailed evaluation of P/T limits was provided in Attachment 4. Attachment 1 notes that alternate initial RT_{NDT} values were used for Linde 80 welds per NRC-approved TR BAW-2308, Revision 1-A and Revision 2-A. Attachment 1 also notes that generic initial RT_{NDT} and its standard deviation are determined in TR BAW-10046A, Revision 2 for SA-508 Class 2 forgings when sufficient material test data is not available to determine a heat-specific initial RT_{NDT} in accordance with RG 1.99, Revision 2.

Technical Specifications

Attachments 2 and 3 to the LAR provided markup and clean versions of the TSs that are to be updated as a result of this LAR. TS 3.4.3 is updated to extend the applicability of the heatup and cooldown limit curves from 31 EFPY to 54 EFPY. In the LAR, TS 3.4.3 was also modified to reflect reactor coolant pump (RCP) operating restrictions for both heatup and cooldown; however, by letter dated April 7, 2015, the licensee provided revised TS 3.4.3 pages to eliminate the modification to the RCP operating restrictions. Thus, the RCP operating restrictions that are applicable to the currently approved heatup and cooldown limit curves (up to 31 EFPY) are extended to 54 EFPY. With respect to cooldown, TS 3.4.3 is modified to eliminate the requirement for a cooldown rate of 30 °F in 15 hours when RCPs are operated between 180 °F and 150 °F. TSs 3.4.9 and 3.4.10 are updated to change the LTOP enable temperature from 262 °F to 259 °F. TS 3.4.11, as revised by the licensee's April 7, 2015, supplement, is updated to change the LTOP enable temperature from 262 °F to 259 °F and to change the electromatic relief valve (ERV) lift setpoint from 460 pounds per square inch gauge (psig) to 508 psig.

Technical Basis and Evaluation of P/T Limits

ANP-3300, Revision 1, provided the technical basis for the proposed P/T limits for ANO-1 based on the AREVA TR BAW-10046A, Revision 2 methodology, supplemented by the alternative initial RT_{NDT} of BAW-2308, Revisions 1-A and 2-A. The alternative approach, which used fracture toughness data to determine the initial RT_{NDT}, is not consistent with 10 CFR Part 50, Appendix G, and 10 CFR 50.61, which require a method based on Charpy V-notch (C_v) and drop weight data. Therefore, Entergy submitted a request for exemption from the 10 CFR Part 50, Appendix G, and 10 CFR 50.61 requirements dated March 20, 2014. This exemption request was approved by the NRC staff by letter dated March 16, 2015. The technical details of the ANO-1 P/T limits analysis are found in Sections 3 through 7 of ANP-3300, Revision 1.

ANP-3300, Revision 1, contains input parameters such as the initial RT_{NDT}, chemical composition, and adjusted reference temperature (ART) values used in the P/T limits calculations. These material parameters are documented in Section 3 of ANP-3300, Revision 1. In a supplemental letter dated February 6, 2015, the licensee provided input parameters for the outlet nozzle forgings which were not included in ANP-3300, Revision 1. Detailed information regarding the generation of the P/T limits for ANO-1 indicated that the proposed P/T limits consist of the allowable pressures for the controlling RPV beltline, inlet and outlet nozzles, and closure head, with their highest ART values reproduced in Table 1 below. The Upper Shell Plate 1 is the limiting material (i.e., the material in the RPV with the highest ART), but Table 1 includes ART values for the weld and nozzle forging material with the highest ART values.

Table 1: ART Values For the Limiting Material and Other Components

Component	ART (°F)	
	1/4T	3/4T
Upper Shell Plate 1, C-5120-2 (limiting material)	179.3	145.5
Outlet Nozzle Forging, 124W479VA1 (limiting nozzle)	59.4	N/A
Lower Shell 1 to Lower Shell 2 Longitudinal Weld, WF-18 (per BAW-2308, limiting weld)	166.8	121.6

Section 4 of ANP-3300, Revision 1, presented design basis information for the P/T limits. Included in this section was a detailed discussion of heatup and cooldown transients. The licensee analyzed three sets of normal heatup transients:

60 °F – 84 °F: 15 °F/hr, then 50 °F/hr above 84 °F

60 °F – 84 °F: 15 °F/hr, then 70 °F/hr above 84 °F

60 °F – 84 °F: 15 °F/hr, then 90 °F/hr above 84 °F

The critical and non-critical curves for the heatup transients are shown in TS Figure 3.4.3-1.

The licensee also analyzed a step cooldown transient and a ramp cooldown transient. The cooldown curve for these transients is shown in TS Figure 3.4.3-2. TS Figure 3.4.3-2 limits the maximum cooldown rate to 100 °F per hour at temperatures greater than or equal to 280 °F, 50 °F per hour at temperatures below 280 °F down to 150 °F, and 25 °F per hour at temperatures below 150 °F . The licensee analyzed both ramp and step transients to determine the most limiting P/T curve. The TS also limit the maximum step temperature changes as follows: at temperatures greater than or equal to 280 °F, 50 °F in any one-half hour, at temperatures less than 280 °F down to 150 °F, 25 °F step in any one-half hour, and at temperatures below 150 °F, a maximum step change of 25 °F in any 1 hour.

The details of the step cooldown transient and ramp cooldown transient can be found in Section 4.6.2 of Attachment 4 of the LAR submittal dated November 21, 2014, and are consistent with the transients allowed by TS 3.4.3. Both the step cooldown transient and the ramp cooldown transient were analyzed by the licensee with the last RCP tripping at three different temperatures (255 °F, 200 °F, and 150 °F). Therefore, a total of six cooldown transient variations were analyzed by the licensee. The evaluation of all the cooldown transient variations also included initiation of the decay heat removal system (DHRS) which occurs at a reactor coolant temperature of 270 °F. DHRS initiation was modeled as a step change from 270 °F to 249 °F, with a hold at 249 °F for 1 minute, followed by a step temperature increase to 263 °F. For the cooldown transients where the last RCP trips at 150 °F, the licensee has eliminated a requirement for a special 15-hour cooldown between 180 °F and 150 °F since the cooldown ramp rates and step changes are defined in the TSs as being maintained over the entire temperature range of the normal cooldown.

To generate the in-service leak and hydrostatic (ISLH) test P/T limits, the licensee evaluated the same set of heatup and cooldown transients as were evaluated for normal operation, with the only difference from normal application being the application of a safety factor of 1.5 rather than 2.0. The ISLH test P/T limit is determined from the most limiting points of all these transients.

Sections 5 and 6 of ANP-3300, Revision 1, provided a technical basis for the P/T limits. For the thermal analyses, ANP-3300, Revision 1, adopted a one-dimensional axisymmetric thermal model and used a finite difference technique to develop the temperature distribution. Resulting thermal stresses were then obtained through numerical integration based on the temperature distribution, and the thermal stress intensity factors (K_{It}) at 1/4T and 3/4T locations were developed using the formulas in Appendix G to Section XI of the ASME Code. Separately, the membrane stress intensity factor due to unit pressure (K_{Im}) was obtained using the formulas in Appendix G to Section XI of the ASME Code for beltline materials. In the final step, the allowable pressure is determined by subtracting K_{It} from K_{Ic} and dividing by the safety factor and K_{Im}. The licensee used a safety factor of 2 for normal operation and 1.5 for the ISLH testing.

Section 4.4 of ANP-3300, Revision 1, stated that the P/T limits for the reactor vessel head-to-flange closure region for normal operation and ISLH operation were derived for the ANO-1 reactor vessel closure head based on the K_{Ic} fracture toughness curve, and that the P/T limits derived for the reactor vessel head-to-flange satisfy the minimum temperature requirements specified in Table 1 of Appendix G to 10 CFR Part 50. More detail on the methodology for the closure head is provided in TR BAW-10046, Revision 2.

For the outlet nozzles, the licensee stated that it used the K_{Im} formula for nozzles from Welding Research Council (WRC) Bulletin 175, "PVRC [Pressure Vessel Research Committee of the WRC] Recommendations on Toughness Requirements for Ferritic Materials." Based on these different sets of P/T limits, the bounding P/T limits were determined. The resulting P/T limits were further modified to consider minimum boltup temperature and the closure flange limits.

Table 6-1 of ANP-3300, Revision 1, provided limiting location correction factors. These factors correct for the difference in actual pressure at the measurement location, at the hot-leg piping pressure taps, to the locations of interest in the RPV (beltline, outlet nozzle, closure head, and core flood nozzle). These correction factors are thus subtracted from the P/T limits determined as described above.

Section 7 of ANP-3300, Revision 1, provided a summary of results for the ANO-1 P/T limits at 54 EFPY, which included P/T limits for normal heatup (HU), normal cooldown (CD), and ISLH HU/CD. The P/T limits were provided in both numeric and graphic format. Criticality limits were determined from the ISLH HU/CD composite data. Table 7-1 provided the pressure and temperature values used to generate the curves on Figure 7-1 for normal (noncritical) heatup. Table 7-2 provided the values regarding minimum temperature for criticality and the pressure limits for heatup while critical, both of which are factors in the reactor criticality limit curve of Figure 7-1. Table 7-3 provided the pressure and temperature values used to generate the curves on Figure 7-2 for normal cooldown. Table 7-4 provided the pressure and temperature values used to generate the ISLH composite P/T limit curves on Figure 7-3. The limiting location correction factors have already been subtracted from the pressures in the figures and tables. These figures are consistent with, and provide the technical basis for, the proposed TS Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 provided in the LAR, as supplemented.

LTOP

The LTOP system was also discussed in the LAR, as supplemented. The LTOP enable temperature would drop from 262 °F to 259 °F, and the ERV lift setpoint (as revised by the licensee's April 7, 2015, letter in response to an NRC staff request for additional information (RAI) dated April 2, 2015; ADAMS Accession No. ML15093A006) would increase from 460 psig to 508 psig, based on the data extending the P/T limits from 31 EFPY to 54 EFPY.

PTS

Attachment 5 to the LAR dated November 21, 2014, provided the RT_{PTS} values for all the beltline materials. In Attachment 1 of its LAR, the licensee stated that the PTS assessment was performed in accordance with 10 CFR 50.61. Also as stated in Attachment 1 of the LAR, and as shown by the data in Attachment 5, the controlling material for PTS would be the Upper Shell Plate 1 (heat C5120-2) with a predicted RT_{PTS} value of 197.4 °F at 54 EFPY. This is below the regulatory screening criteria in 10 CFR 50.61(b)(2) for PTS, which is 270 °F for plates.

USE and EMA

In Attachment 1 to the licensee's LAR submittal, the licensee stated, in part, that

> The current analysis remains bounding for the projected end of life fluence, except for the Upper Shell Plate 1 Material. The USE and EMA calculations also remain bounding for close to 54 EFPY as the fluence calculated per BAW-2241P-A methodology following Cycles 21, 22, and 23 is lower, or only marginally higher, than the conservative fluence used in BAW-2251A. The copper content has also decreased.

3.3 NRC Staff Evaluation – Reactor Vessel Fluence

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

- A fluence calculation performed using an acceptable methodology
- Analytic uncertainty analysis identifying possible sources of uncertainty
- Benchmark comparison to approved results of a test facility
- Plant-specific qualification by comparison to measured fluence values

For input to its P/T Limits, Entergy performed fluence calculations in accordance with AREVA NP, Inc. Licensing TR BAW-2241NP-A, Revision 2, "Fluence and Uncertainty Methodologies," April 2006 (ADAMS Accession No. ML073310660), which was approved by the NRC staff in an SE dated April 26, 2006 (ADAMS Accession No. ML061220721).

After reviewing the calculations, the NRC staff concludes that the neutron calculations are performed in a manner consistent with the guidance set forth in RG 1.190. A solution to the

Boltzmann transport equation is approximated using the two-dimensional (2D) discrete ordinates code (DOT). The licensee uses a cross-section library based on the ENDF/B-VI nuclear data. Numeric approximations include a P3 Legrende expansion for anisotropic scattering and the modeling uses S8 order of angular quadrature. These cross-section data and approximations are in accordance with the modeling guidance contained in in RG 1.190. Since the licensee used NRC-approved RG 1.190 adherent methods to determine the vessel fluence for ANO-1, the NRC staff concludes that the fluence calculations are acceptable.

Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Three-dimensional (3D) flux solutions are constructed using a synthesis of azimuthal, axial, and radial flux. Source distributions include a cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions, which are used to develop spatial- and energy-dependent core source distributions that are averaged over each fuel cycle. This method accounts for source energy spectral effects by using an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history for each fuel assembly. The neutron source and transport calculations, as described above, were performed in accordance with the guidance set forth in RG 1.190. Based on the consistency with the guidance contained in RG 1.190, the NRC staff concludes that the source and transport calculations used in support of the development of the ANO-1 P/T curves are acceptable.

The NRC-approved methods are supported by an analytic uncertainty analysis, and the estimated uncertainty is less than 20 percent, which is in accordance with RG 1.190 and, therefore, the NRC staff concludes that this is acceptable. Details of the analytic uncertainty analysis are provided in BAW-2241NP-A, Revision 2.

BAW-2241NP-A, Revision 2, describes the methods qualification using the standard benchmark problems found in RG 1.190. The calculations were compared with the benchmark measurements from the Poolside Critical Assembly (PCA) simulator at the Oak Ridge National Laboratory (ORNL) and vessel fluence benchmark problems provided in NUREG/CR-6115, "PWR and BWR [Boiling-Water Reactor] Pressure Vessel Fluence Calculation Benchmark Problems and Solutions," provided by Brookhaven National Laboratory (BNL), dated September 2001 (ADAMS Accession No. ML012900043). The NRC staff concludes that these test facilities are specifically referenced in RG 1.190, and are, therefore, acceptable.

BAW-2241NP-A, Revision 2, contains acceptable plant-specific benchmarking for ANO-1 as it contains a database of PWR dosimetry benchmarking. The ANO-1 unit-specific geometry, a Babcock and Wilcox (B&W) RPV, is well represented within the database. B&W-specific benchmarking documented in BAW-2241NP-A, Revision 2, indicates that surveillance capsule fluence can be calculated within 20 percent of measured values, which is in accordance with RG 1.190. The NRC staff concludes, therefore, that these uncertainties are acceptable.

Also, in its RAI dated January 21, 2015 (ADAMS Accession No. ML15021A428), the NRC staff requested that the licensee demonstrate the bootstrap, spatial modeling, and synthesis techniques for the transport calculations are adequate to produce reliable fluence estimates in the lower nozzle belt forging (SRXB-RAI-1(a)). In its February 6, 2015, RAI response, the licensee stated that the bootstrap technique is no longer being used, and that a more explicit representation of the spatial modeling is being used. The spatial and synthesis modeling uses a

2D RZ model expansion to include the upper and lower internal structures, with the method of determining 3D flux results in internal structures and other areas not varying from the beltline method. The NRC staff notes that elimination of the bootstrap technique provides for a more explicit representation of the problem geometry. Additionally, the spatial and synthesis modeling remain consistent with the NRC-approved methods documented in BAW-2241NP-A, Revision 2. Based on these considerations, the NRC staff concludes that the licensee's response to SRXB-RAI-1(a) is acceptable.

The NRC staff also requested in its RAI dated January 21, 2015, that the licensee demonstrate that the angular quadrature chosen for the transport solution is adequate (SRXB-RAI-1(b)). In its RAI response dated February 6, 2015, the licensee stated that the multi-variable and parameter sensitivity evaluations indicated that the BAW-2241NP-A, Revision 2, fluence method is unbiased, with the reactor cavity dosimetry results providing for a demonstration of the adequacy of the S8 angular quadrature. The licensee also provided information indicating that S8 angular quadrature was also compared to higher-order models and showed consistent results. The consistency between the higher order angular quadrature modeling and the S8 modeling in the nozzle demonstrates the adequacy of the S8 results. Therefore, the NRC staff concludes that the licensee demonstrated that the S8 angular quadrature is adequate for the nozzle region based on accuracy to a higher order angular quadrature. Based on this consideration, the NRC staff concludes that the licensee's response to SRXB RAI-1(b) is acceptable.

ANP-3300, Revision 1, included as Attachment 4 to the LAR, included fluence estimates for the lower nozzle belt forging. Figure 2-1 of Attachment 4 provides a drawing of this forging as located immediately below the outlet nozzle forging. Although the figure did not indicate the location of the reactor core, it appeared that the top of active fuel may be below the lower nozzle belt forging.

ANP-3300, Revision 1, indicates that the fluence was calculated in accordance with BAW-2241NP-A, Revision 2, and that this method complies with RG 1.190. The NRC staff noted that the guidance in RG 1.190 applies primarily to the region of the reactor vessel that directly surrounds the effective height of the active core. Furthermore, the NRC staff determined that the qualification of BAW-2241NP-A, Revision 2, is not well established for determining fluence at or near RPV nozzle locations.

In its RAI dated January 21, 2015, the NRC staff requested that the licensee provide a qualified estimate of the accuracy and uncertainty of the fluence method for the nozzle locations, and to demonstrate that the uncertainty in the fluence estimate is within the 20 percent margin term in the reference temperature calculations (SRXB-RAI-1(c)). In its February 6, 2015, RAI response, the licensee stated that the qualified estimate of accuracy and uncertainty of the fluence methods for the nozzle locations was not completed for BAW-2241NP-A, Revision 2; however, the qualification data were reviewed to establish that there were no axial dependences in the qualification database. In other words, the licensee indicated that the BAW-2241NP-A, Revision 2, methods have been shown to perform as well at upper core locations as at the core mid-plane. The licensee also demonstrated through EMA that nozzle-to-shell attachment welds are not limiting material with regard to neutron embrittlement damage. The NRC staff concludes that the licensee demonstrated that there is not a strong axial height dependency, and that the EMA demonstrated the nozzle-to-shell attachment welds are not limiting. Based on this consideration, the NRC staff concludes that the licensee's response to SRXB RAI-1(c) is acceptable.

Based on the above, the NRC staff concludes that the licensee provided adequate information for justification of the neutron fluence calculation outside the beltline at the lower nozzle belt forging. The NRC staff further concludes that the use of the fluence calculations in support of the development of the revised P/T limit curves for use up to 54 EFPY is acceptable.

3.4 NRC Staff Evaluation – Vessel and Reactor Coolant Pressure Boundary Integrity

3.4.1 P/T Limits

The NRC staff reviewed the licensee's methodology for determining the P/T limits for the beltline described in ANP-3300, Revision 1, and determined that the methodology is consistent with NRC staff approved TR BAW-10046A, Revision 2, and that described in the ASME Code, Section XI, Appendix G. The NRC staff notes that BAW-10046A, Revision 2, uses the dynamic crack arrest fracture toughness (K_{Ia}) curve while the methodology of ANP-3300, Revision 1, uses the static crack initiation fracture toughness (K_{Ic}) curve, consistent with the current version of Appendix G to Section XI of the ASME Code.

The licensee based its ISLH test P/T limits on the composite curve determined from the most limiting P/T points of all the same heatup and cooldown transients evaluated for normal operation, applying a safety factor of 1.5 consistent with the ASME Code. Since the ASME Code, Section XI, Appendix G allows P/T limits for hydrostatic testing to be based on steady-state or isothermal conditions, the NRC staff concludes that the licensee's methodology is conservative compared to the ASME Code since it is based on non-steady state transients.

With respect to the methodology used by the licensee to develop the outlet nozzle P/T limits, the ASME Section XI Code of record for ANO-1 (2001 Edition through 2002 Addenda) does not provide any guidance for determining the fracture toughness requirements for nozzles. However, Paragraph G-2223 (b) of the ASME Code, 2001 Edition through 2002 Addenda, does state that Welding Research Council Bulletin (WRCB) 175, "PVRC Recommendation on Toughness Requirements for Ferritic Materials," dated August 1972, provides an approximate method for analyzing the inside corner of a nozzle and cylindrical shell for elastic stresses due to internal pressure stress. This is consistent with the methodology used by the licensee. The ASME Code, Section XI, 2013 Edition, Appendix G, Paragraph G-2223, incorporated a method for determining the stress intensity factor due to internal pressure loading for nozzles (K_{Ip}). Two different methods are permitted, one of which is identical to the WRCB 175 method, while the other method is identical to the sharp corner solution from ORNL Technical Letter Report 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 ASME Code, Section XI, 2013 Edition does not contain any guidance for determining K_{It}.

Material Adjusted Reference Temperature

To verify the licensee-calculated ARTs for the beltline materials, the NRC staff performed ART calculations using the materials information provided in the LAR dated November 21, 2014. As an additional verification, the NRC staff also examined the RPV materials information in the license renewal application for ANO-1, as well as TR BAW-2251-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel," dated June 1996 (ADAMS Accession No. ML003670224). The NRC staff found that, for Upper Shell Plates 1 and 2 and for Lower Shell Plates 1 and 2, the Initial RT_{NDT} and margin data found in the LAR is different for the information for the same materials found in the license renewal application and TR BAW-2251-A. In its RAI dated January 21, 2015, the NRC staff requested that the licensee to describe how the initial RT_{NDT} and margin values were determined for selected materials and to describe why it was determined that the method of determining initial RT_{NDT} and margin values should be changed for this LAR (EVIB-RAI-4).

In its RAI response dated February 6, 2015, the licensee clarified that the initial RT_{NDT} for Upper Shell Plates 1 and 2 and for Lower Shell Plates 1 and 2 was determined from measured Charpy values, but that the orientation of the Charpy specimens could not be conclusively determined. Subsequently, the licensee located data from Charpy specimens oriented in the transverse (weak) direction that would support recalculated initial RT_{NDT} values for Upper Shell Plate 2 and Lower Shell Plate 2. This data is documented in BAW-1440, "Analysis of Capsule ANI-E from Arkansas Power & Light Company Arkansas Nuclear One – Unit 1," dated April 1977 (not publicly available), and supports the initial RT_{NDT} values of 10 °F for Upper Shell Plate 2 and 30 °F for Lower Shell Plate 2 that are documented in the LAR. The NRC staff concludes that the licensee's use of data from Charpy specimens oriented in the transverse direction for Lower Shell Plate 2 and Upper Shell Plate 2 meets the requirements of the ASME Code which requires that Charpy specimens for all plate material shall be oriented in a direction normal to the principal rolling direction.

Also, in its EVIB-RAI-4 response, the licensee stated that it could not locate data from Charpy specimens oriented in the transverse direction that would support recalculated initial RT_{NDT} values for Upper Shell Plate 1 and Lower Shell Plate 1. Therefore, the licensee used generic values consistent with the guidance in RG 1.99, Revision 2, to recalculate the initial RT_{NDT} values for Upper Shell Plate 1 and Lower Shell Plate 1. The margin term for Upper Shell Plate 1 and Lower Shell Plate 1 also had to be recalculated in accordance with the guidance in RG 1.99, Revision 2, since surveillance data was not used. The NRC staff notes that RG 1.99, Revision 2, recommends that if generic mean values are used for initial RT_{NDT}, the σ_i value in the margin term should be equal to the standard deviation of the data set used to establish the mean RT_{NDT}. The staff notes that the generic initial RT_{NDT} values and margin values used by the licensee for Upper Shell Plate 1 and Lower Shell Plate 1 are consistent with those accepted by the NRC staff for use by several other B&W plants. The licensee's use of the generic values results in a higher initial RT_{NDT} value and margin term resulting in a net increase to the ART value of about 40 °F, not including the additional increase due to the fluence increase from 31 EFPY to 54 EFPY.

Based on the above, the NRC staff concludes that the licensee's response to EVIB-RAI-4 is acceptable, and that the method of calculating initial RT_{NDT}, margin, and ART for the beltline materials as described in the LAR is acceptable because it meets the requirements of the ASME Code, or is consistent with the guidance RG 1.99, Revision 2, for those materials that use generic initial values.

The NRC staff verified that the initial RT_{NDT} values for Linde 80 welds used by the licensee were determined in accordance with the NRC staff SEs for BAW-2308, Revisions 1-A and 2-A, as permitted by the exemption granted on March 16, 2015. Although the Linde 80 welds are limiting materials for the current P/T limits which are valid through 31 EFPY, as a result of the new data used from BAW-2308, the Linde 80 welds are no longer limiting for the proposed P/T limits which would be valid through 54 EFPY.

RIS 2014-11 and Nozzle Materials

As clarified in RIS 2014-11, 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. Additionally, since TR BAW-10046A, Revision 2, which the licensee used to develop its P/T limits for nozzles, does not provide guidance for evaluating the effects of neutron fluence on the nozzle RT_{NDT}, the NRC staff requested in its RAI dated January 21, 2015 (EVIB-RAI-1), that the licensee describe how neutron fluence was considered in the evaluation of the nozzles, and to provide RT_{NDT} and fluence values for the limiting nozzle.

In its response to EVIB-RAI-1 dated February 6, 2015, the licensee described how fluence was calculated for both the inlet and outlet nozzle forgings. The outlet nozzle forgings receive a higher fluence than the inlet nozzle forgings and have an end-of-life fluence projected to be greater than 1×10^{17} n/cm^2. Thus, the outlet nozzle forgings are defined to be part of the beltline, in accordance with RIS 2014-11 and Appendix G to 10 CFR Part 50. Although P/T limits analysis is performed at 1/4 T and 3/4 T wall-thickness locations, the licensee used peak wetted-surface fluence values to perform its analysis of the nozzles. Since peak wetted-surface fluence would be greater than fluence at the 1/4 T and 3/4 T wall-thickness locations, the NRC staff notes that this is a conservative method of evaluating P/T limits.

The licensee's response to EVIB-RAI-1 also discussed how RT_{NDT} and ART values were determined for the limiting outlet nozzle forgings. The licensee's original initial RT_{NDT} values were determined using Charpy specimens which were all oriented in the strong (L-T or longitudinally oriented) direction and were all tested at a single temperature of 10 °F. This method of testing was allowed by the ASME Code prior to 1971. SRP Branch Technical Position (BTP) 5-3, "Fracture Toughness Requirements" (ADAMS Accession No. ML070850035), paragraph 1.1(4) states that "if limited Charpy V-notch tests were performed at a single temperature to confirm that at least 41 J (30 ft-lbs) was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 61 J (45 ft-lbs) was obtained if the specimens were longitudinally oriented." In its RAI response, the licensee stated that a minimum of 90 ft-lbs was obtained. Thus, BTP 5-3, paragraph 1.1(4) would support an initial RT_{NDT} of 10 °F for the ANO-1 outlet nozzle forgings using the data referenced by the licensee.

However, BTP 5-3, paragraph 1.1(4) has been shown to be potentially nonconservative (see NRC presentations from a February 19, 2015, public meeting at ADAMS Accession Nos. ML15061A065 and ML15061A075). Because the BTP 5-3, paragraph 1.1(4) is potentially nonconservative, the licensee chose not to use the value of 10 °F for the initial RT_{NDT} but rather performed an evaluation to determine an appropriate bounding estimate of the initial RT_{NDT} for the ANO-1 outlet nozzle forgings. Using data from ASME SA-508 Class 2 forgings located in B&W-designed reactor vessels for which the initial RT_{NDT} was determined by both the pre-1971 method (Charpy specimen oriented in the strong direction) and in accordance with the ASME Code, Section III, NB-2331 (Charpy specimen oriented in the weak direction), the licensee constructed a graph of initial RT_{NDT} (current ASME method) versus Charpy energy measured in the L-T direction (pre-1971 method). The licensee then graphed an upper bound line of the data set, and took the intersection of this line with the minimum obtained impact energy of 90 ft-lbs as the adjusted initial RT_{NDT} for the ANO-1 outlet nozzle forgings. Based on this graph, the licensee obtained an initial RT_{NDT} value of 40 °F, which is 30 °F higher than the initial RT_{NDT} value that would be determined in accordance with BTP 5-3, paragraph 1.1(4). Using the initial RT_{NDT} value of 40 °F, the licensee also determined 1/4T ART using the guidance of RG 1.99, Revision 2, but using the expected 54 EFPY inner diameter neutron fluence as the fluence at the 1/4T location. The limiting 1/4T ART at 54 EFPY for the outlet nozzle forgings was determined by the licensee to be 59.4 °F, which is bounded by the RT_{NDT} value of 60 °F used in the nozzle region to support the ANO-1 P/T limits analysis.

The NRC staff concludes that the licensee's evaluation in its response to EVIB-RAI-1 is acceptable because: (1) the licensee used a representative data base for ASME SA-508, Class 2 forgings, and (2) the upper bound line for initial RT_{NDT} versus strong direction Charpy energy at 10 °F bounds all data with an adequate margin. The NRC staff notes that further use of this position/method without generic NRC guidance must be justified on a case-by-case basis. The NRC staff also notes the following conservatisms that support the licensee's calculations of 1/4T ART for nozzles for ANO-1. First, the data used to calculate initial RT_{NDT} had a minimum Charpy impact energy of 90 ft-lbs which is twice that allowed by BTP 5-3, paragraph 1.1(4) for establishing initial RT_{NDT} based on longitudinal specimens tested at a single value. Second, the initial RT_{NDT} was increased by 30 °F. Third, wetted-surface fluence rather than attenuated fluence at the 1/4T location was used at the 1/4T location. These factors result in an estimate of 1/4T ART which is conservative for the outlet nozzle. To further demonstrate the conservatism of the licensee's method for calculating initial RT_{NDT}, by using the plot of the upper bound line of RT_{NDT} versus Charpy energy in the strong direction, the NRC staff notes that in order to use an initial RT_{NDT} value of 10 °F, the licensee would need to have impact data at that temperature (10 °F) with a Charpy energy of nearly 120 ft-lbs. This is a margin of nearly 75 ft-lbs above the BTP 5-3, paragraph 1.1(4) allowance.

Therefore, the NRC staff concludes that the licensee determined the ART of the outlet nozzle forgings in an acceptable manner because it used an appropriate database and methodology to predict a conservative initial RT_{NDT}, and also conservatively predicted the effect of neutron fluence on the nozzle ART. Therefore, EVIB-RAI-1 is resolved.

The licensee stated in paragraph 4.2.b of the ANP-3300, Revision 1 (Attachment 4 to its LAR dated November 21, 2014), that a deep corner flaw is postulated on the inside surface of the reactor vessel outlet nozzles and that the outlet nozzle bounds the inlet nozzle and the core flood nozzle. In an RAI dated March 4, 2015 (ADAMS Accession No. ML15065A084), the NRC staff requested that the licensee demonstrate that the outlet nozzles bound the inlet nozzles and the core flood nozzles (EVIB-RAI-8). In its RAI response dated March 10, 2015, the licensee provided two factors which make the outlet nozzle bounding. The outlet nozzle is significantly larger than the inlet nozzle and core flood nozzle and as the radius of the nozzle increases, the magnitude of the stress intensity factor increases for a constant assumed flaw. Secondly, the bottom of the outlet nozzle is closer to the active core and receives more fluence than the bottom of the inlet nozzle. Based on an evaluation of the information provided by the licensee in response to EVIB-RAI-8, the NRC staff concludes that the licensee's response to EVIB-RAI-8 is acceptable. Therefore, EVIB-RAI-8 is resolved.

In summary, the NRC staff concludes that the licensee's evaluation of nozzles meets the requirements of 10 CFR 50, Appendix G, the ASME Code, and the guidance in RIS 2014-11, and is, therefore, acceptable.

Confirmatory Calculations of P/T Limits

The NRC staff performed independent P/T limit calculations for ANO-1, in order to verify the P/T limits presented in Section 7 of ANP-3300, Revision 1 (Attachment 4 to the LAR), for ANO-1, considering all relevant information in ANP-3300, Revision 1 and the February 6, 2015, RAI responses. The staff's calculations addressed both the minimum temperature requirements of 10 CFR 50, Appendix G, which are based on the limiting properties of the RPV materials in the closure flange region that are highly stressed by bolt preload, and the P/T limits when the coolant temperatures exceed the minimum temperature requirements, which are referred to as the "ASME Code, Appendix G Limits," in 10 CFR 50, Appendix G.

While performing verification of the compliance of the licensee's P/T limits with the minimum temperature requirements of 10 CFR 50, Appendix G, Section IV.A.2, Table 1, the NRC staff required additional data to confirm that the limiting material in the closure flange region that is highly stressed by the bolt preload would not be controlling for the proposed P/T limits, and that operations using the proposed P/T limits would not result in a violation of the 10 CFR Part 50, Appendix G, Section IV.A.2, Table 1 requirements for ISLH, normal operation heatup and cooldown with the core not critical, and normal operation heatup and cooldown with the core critical. In its RAI dated February 6, 2015, the NRC staff requested that the licensee provide the RT_{NDT} for the limiting material in the closure flange region and to describe how the heatup and cooldown curves in the LAR comply with 10 CFR Part 50, Appendix G, Section IV.A.2 (EVIB-RAI-3). In its RAI response dated February 6, 2015, the licensee provided the requested information. Using the information provided by the licensee, the staff calculated the minimum temperature requirements for ISLH, normal operation heatup and cooldown with the core not critical, and normal operation heatup and cooldown with the core critical. The NRC staff verified that the licensee complies with all the minimum temperature requirements of 10 CFR Part 50, Appendix G, Section IV.A.2, Table 1. The NRC staff also verified that the closure flange region is not controlling for any portion of the P/T curves for ISLH, normal operation heatup and cooldown with the core not critical, and normal operation heatup and cooldown with the core critical. With respect to criticality, the NRC staff verified that the minimum temperature as stated in the LAR is controlled by the minimum ISLH test temperature.

ASME Code, Section XI, Appendix G, Subparagraph G-2214.3 specifies two different methods for determining the applied stress intensity factor due to a radial thermal gradient (K_{It}). The first method determines the maximum K_{It} as a function of the cooldown rate and reactor vessel wall thickness. The through-wall temperature difference and metal temperature at any depth (such as the 1/4T or 3/4T locations) may then be determined from the maximum K_{It} using Figures G-2214-1 and G-2214-2 of Appendix G. The second method uses a polynomial fit to a stress distribution. The temperature and stress distributions are determined using a finite element analysis or similar technique, and vary as a function of time through the transient. In Section 5.2 of ANP-3300, Revision 1, the licensee stated that the second method for determining K_{It} was used in the development of the proposed ANO-1 54 EFPY P/T curves.

To support its confirmatory calculations, in its RAI dated March 4, 2015, the NRC staff requested that the licensee provide the K_{It} values and the metal temperatures at the 1/4T and 3/4T locations for the transients evaluated by the licensee (EVIB-RAI-6). In the same RAI, the NRC staff requested similar information for the outlet nozzle for the 1/4T location (EVIB-RAI-6). Section 5.4 of ANP-3300, Revision 1, describes the calculation of the unit pressure stress intensity factor for the outlet nozzle, which is a function of several geometrical parameters of the nozzle. In EVIB-RAI-7, the NRC staff requested the licensee to provide the values of the nozzle geometrical parameters for use in their confirmatory calculations.

Table 6-1 of ANP-3300, Revision 1, provides the limiting location pressure correction factors for ANO-1. These factors were used by the licensee to correct for the difference in pressure between the pressure instrument location (the pressure tap on the RCS hot leg) and the evaluated RPV locations (e.g., beltline, outlet nozzle, reactor vessel closure head, or core flood nozzle). In EVIB-RAI-9 dated March 4, 2015), the NRC staff requested clarification with respect to whether these correction factors were applied by subtracting them from the licensee's allowable pressures in Tables 7-1 through 7-4 of ANP-3300, Revision 1, and whether the correction factors were included in the values reported in the tables. In EVIB-RAI-9, the NRC staff also requested that the licensee confirm whether the beltline was the limiting location at all temperatures and pressures for all the transients evaluated and, therefore, whether the correction factor for the beltline was subtracted for all calculated allowable pressures in the resulting P/T curves.

In its March 10, 2015, response to EVIB-RAI-9, the licensee confirmed that the limiting pressure correction factors were included in the table values and that the beltline was limiting. The licensee also indicated that the P/T curves, in some cases, had been conservatively adjusted to avoid steps or negative slopes in allowable pressure caused by the evaluation of step changes in temperature. In its April 7, 2015, RAI response, the licensee provided a revision to ANP-3300, Revision 1, providing revised location pressure correction factors specific to heatup or cooldown, revising Table 6-1 and adding Table 6-2.

By letter dated March 10, 2015, the licensee provided the requested information in response to EVIB-RAI-5, EVIB-RAI-6, and EVIB-RAI-7. Using this information and the information provided in the LAR, the NRC staff performed confirmatory calculations of the licensee's proposed P/T curves. The staff used the methodology of the ASME Code, Section XI, Appendix G, in conjunction with the K_{It} and metal temperature values provided in the RAI responses, to independently calculate the ANO-1 P/T curves. The NRC staff compared its calculated curves to the licensee's curves defined in Tables 7-1, 7-2, 7-3, and 7-4 of ANP-3300, Revision 1, as modified to account for the correction factors in Table 6-1 and 6-2 of the licensee's April 7, 2015, RAI response. The NRC staff's calculated beltline curves are in very good agreement with the licensee's curves, with the licensee's curves being equally or more conservative than those developed by the NRC staff.

As previously noted in Section 3.2 of this SE, the licensee evaluated step transients in addition to ramp transients to determine the P/T limits for normal cooldown, since the TSs place limits on the maximum allowable step change during cooldown. The handling of thermal stresses during stepped transients was previously addressed during the review of the Oconee Nuclear Station, Units 1, 2, and 3 P/T limits, which used the same methodology as ANO-1 for development of the P/T limits. The response to RAI-6 related to the Oconee P/T limits, in Attachment 1 of Duke Energy's September 10, 2013, letter to the NRC (ADAMS Accession No. ML13259A120), and stated, in part, that "...[s]ince it is not expected that the operator can follow a stepped transient nor can a step change in temperature actually be imposed on the reactor coolant, stepped transients are treated as a series of hold points, such that the allowable pressures for a stepped transient are calculated at the end of the step or hold period." The NRC staff found this method acceptable in its SE supporting the issuance of the license amendment for the Oconee, Units 1, 2 and 3 P/T limits dated February 27, 2014 (ADAMS Accession No. ML14041A093). Therefore, for ANO-1, the NRC staff verified that the allowable pressures at the end of each evaluated hold period were greater than or equal to the licensee's allowable pressures.

As an independent check on the licensee's determination of the K_{It} values and metal temperatures, the NRC staff compared the licensee's values to values determined using the methods of ORNL/NRC/LTR-03/03, "Tabulation of Thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating Pressure-Temperature Curves per ASME Section XI – Appendix G," dated March 2003, and ORNL/TM-2012/567, "Fracture Analysis of Vessels – Oak Ridge FAVOR, v12.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations," dated November 2012 (ADAMS Accession Nos. ML100840745 and ML13008A015, respectively). Based on the staff's comparisons, the K_{It} and metal temperature values used by the licensee were found to be in reasonable agreement with those determined using the methods described in the above ORNL/NRC research reports, and are, therefore, acceptable.

To independently verify the licensee's nozzle methodology, the NRC staff generated P/T curves for the ANO-1 outlet nozzle (most limiting, as previously discussed) using both methods allowed by Paragraph G-2223 of the ASME Code, Section XI, Appendix G, 2013 Edition, using the licensee's K_{It} and metal temperature values. The results of these calculations showed that the method of G-2223(c) (identical to the WRCB 175 method used by the licensee) results in more conservative P/T limits for the nozzle than the method of G-2223(d) (identical to the ORNL/TM-2010/246 sharp corner method). Since the ASME Code does not provide any guidance on how to determine K_{It} for nozzles, the NRC staff compared the licensee's K_{It} values for the outlet nozzle cooldown to K_{It} values generated using the method of ORNL/TM-2010/246, which showed that the licensee's peak K_{It} values are conservative. In addition, the NRC staff's independent calculation of the nozzle P/T limits shows that nozzle P/T curves generated using both methods are not limiting (i.e., are bounded by the beltline curves).

Based on its confirmatory calculations, the NRC staff concludes that the proposed ANO-1 54 EFPY P/T limits for normal heatup, normal cooldown, and ISLH testing meet the requirements of 10 CFR 50, Appendix G because: (1) the limits are equally or more conservative than limits obtained by following the methods of analysis and margins of safety in Appendix G of Section XI of the ASME Code, and (2) the limits comply with the minimum temperature requirements of 10 CFR Part 50, Appendix G, Section IV.A.2, Table 1.

3.4.2 LTOP Enable Temperature

Appendix G to Section XI of the ASME Code requires that for plants with LTOP systems, the system shall be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50 °F, whichever is greater. In its LAR, the licensee proposed to change the LTOP enable temperature from 262 °F to 259 °F. The current LTOP enable temperature, as discussed in a November 26, 1996, submittal from the licensee (ADAMS Legacy Accession No. 9612120073), was based on the ASME Code requirements in place at the time. For the 31 EFPY limits, Weld W-112 (the upper to lower shell circumferential weld) was the limiting material with a RT_{NDT} of 212 °F at 31 EFPY. In the LAR dated November 21, 2014, the licensee indicated that Weld W-112 was no longer the limiting material since the alternative initial RT_{NDT} of BAW-2308, Revisions 1-A and 2-A was used. The licensee stated in its LAR that the Upper Shell Plate 1 was now the limiting material with a RT_{NDT} of 179.3 °F at 54 EFPY. Therefore, the ASME Code would require the LTOP enable temperature to be at coolant temperature corresponding to a reactor vessel metal temperature less than 229.3 °F (ART plus 50 °F). If isothermal conditions are postulated, the coolant temperature would correspond to the reactor vessel metal temperature. However, for transients postulated where the system is not at thermal equilibrium, the coolant temperature during these transients will differ from the reactor vessel metal temperature. The coolant temperature will be lower than the metal temperature during a cooldown and higher than the metal temperature during heatup. Based on the data provided by the licensee in its March 10, 2015, response to EVIB-RAI-5 for the 90 °F/hr heatup transient, the NRC staff has confirmed that an LTOP enable temperature of 259 °F is appropriate considering the temperature difference between the reactor coolant and 1/4T location, when assuming a reactor vessel metal temperature of 229.3 °F. The NRC staff concludes that the licensee's LTOP enable temperature is acceptable because it meets the requirements of the ASME Code.

3.4.3 PTS

The regulations in 10 CFR 50.61 define a PTS event as "an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel." To verify the licensee's calculated RT_{PTS} for the beltline materials provided in Attachment 5 to the LAR, the NRC staff performed confirmatory calculations in accordance with the requirements of 10 CFR 50.61. The staff's confirmatory calculation results were consistent with the licensee's calculations, which stated that the controlling material for PTS would be the Upper Shell Plate 1 (heat C5120-2) with a predicted RT_{PTS} value of 197.4 °F at 54 EFPY. This value is below the 10 CFR 50.61(b)(2) regulatory screening criteria for PTS, which is 270 °F for plates. Based on information provided by the licensee and its confirmatory calculations, the NRC staff concludes that the licensee's PTS assessment is in accordance with the requirements of 10 CFR 50.61, and is, therefore, acceptable.

3.4.4 USE and EMA

In Attachment 1 to the LAR dated November 21, 2014, the licensee discussed the USE and EMA analyses and provided a table with updated fluence values for 54 EFPY for several beltline materials. In Section 2.0 of Attachment 1 of the LAR, under the subsection titled "Upper Shelf Energy and Equivalent Margins Analysis," the licensee stated that "the current analysis remains bounding for the projected end of life fluence, except for the Upper Shell Plate 1 Material." However, the NRC staff had concerns regarding this statement since the projected 54 EFPY fluence for the Upper Shell Plate 1 material appeared to be bounded by the 48 EFPY fluence previously projected in BAW-2251A, while the 54 EFPY fluence of other materials, such as the Longitudinal Welds, appeared to be greater than previously projected. Accordingly, the NRC staff requested that the licensee provide the following information (EVIB-RAI-2) in its RAI dated January 21, 2015:

- how the analysis for USE and EMA would not remain bounding for the Upper Shell Plate 1 material,
- how the analysis for USE and EMA would remain bounding for both Upper and Lower Shell Longitudinal Welds, and
- provide an updated analysis for USE and EMA if the current analysis does not remain bounding for any material.

In its response to EVIB-RAI-2 dated February 6, 2015, the licensee stated that the following statement from the LAR should be removed:

> The current analysis remains bounding for the projected end of life fluence, except for the Upper Shell Plate 1 Material. The USE and EMA calculations also remain bounding for close to 54 EFPY as the fluence calculated per BAW-2241P-A methodology following Cycles 21, 22, and 23 is lower, or only marginally higher, than the conservative fluence used in BAW-2251A. The copper content has also decreased.

The licensee stated that the above statement should be replaced with the following:

> The current USE and EMA analyses (BAW-2251A, Reference (6), Note: Appendix B of the Report contains BAW-2275 that addresses the EMA analyses) remain valid through 48 EFPY. For the EMA analysis, comparing the current projected 48 EFPY wetted surface fluence values of the limiting welds of ANO-1 with the EMA calculations reported in BAW-2275A, it can be shown that the EMA analyses for ANO-1 remains valid through 48 EFPY.

Also, in its response to EVIB-RAI-2, the licensee stated that updates to the current USE and EMA calculations will be necessary in the evaluation period prior to the projected fluence exceeding the fluence on which the current USE and EMA calculations were based.

The NRC concludes that the licensee's response to EVIB-RAI-2 is acceptable, in that it clarifies that the current USE and EMA analyses, which are valid through 48 EFPY, cannot be extended to 54 EFPY based on the currently approved evaluation. The NRC staff notes that the licensee is required by the regulations at 10 CFR Part 50, Appendix G, Paragraph IV.A.1.c to submit the analysis of USE and EMA "as specified in § 50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of section IV.A.1 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate."

3.5 NRC Staff Evaluation – Low Temperature Overpressure Protection and Electromatic Relief Valve Setpoint

In its LAR dated November 21, 2014, the licensee stated that the LTOP enable temperature for 54 EFPY is 259 °F, which is 3 °F lower than the current 31 EFPY LTOP enable temperature of 262 °F. Also in its LAR, the licensee proposed an LTOP system pressure limit of 553.8 psig, as compared to the previously approved 31 EFPY pressure limit of 460 psig. The ERV setpoint is applicable whenever the RCS temperature is below the LTOP enable temperature. The ERV is designed to open via signal if the RCS pressure reaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors the RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setpoint, the ERV is signaled to open. Maintaining a lowered setpoint ensures that the P/T limits will not be violated for any analyzed LTOP event. In its review of the LAR, the NRC staff determined that the minimum allowable pressure for ANO-1 is 513 psig for the 90 degree/hour heatup transient (the most limiting heatup transient, see Table 7.1 of ANP-3300, Revision 1, which is Attachment 4 to the LAR) and 508 psig for the cooldown transients (the most limiting transient, see Table 7.3 of ANP-3300, Revision 1). The ASME Code, Section XI, Appendix G, G-2215, Equation 1 states that "LTOP systems shall limit the maximum pressure in the vessel to 100% of the pressure determined to satisfy [Equation 1]." The NRC staff notes that Equation 1 of Paragraph G-2215 is the basis for the allowable pressure equations for heatup and cooldown conditions. In an RAI dated March 20, 2015 (ADAMS Accession No. ML15097A005), the NRC staff requested that the licensee provide an explanation for how the proposed LTOP setpoint meets the ASME Code, Section XI, Appendix G requirements since the proposed setpoint was greater than the minimum allowable pressures for the heatup and cooldown transients, as described above.

In its RAI response dated March 25, 2015, the licensee stated that the 54 EFPY ERV LTOP setpoint was determined using the methodology in ASME Code, Section XI, Appendix G, 2001 Edition through 2002 Addenda. The licensee stated that its 54 EFPY methodology was consistent with the 31 EFPY methodology, namely through the use of isothermal P/T curves and application of pressure correction factors, with the exception that the proposed 54 EFPY limits did not utilize ASME Code Case N-514, "Low Temperature Overpressure Protection, Section XI, Division 1." The licensee also stated that its LTOP setpoint is based on 100 percent of the fracture mechanics ASME Section XI, Appendix G, P/T limits for isothermal conditions limits, rather than the transient heatup/cooldown P/T limits.

Since the ASME Code, Section XI, does not specify that only the isothermal P/T limits need to be considered in determining the LTOP setpoint, in an RAI dated April 2, 2015, the NRC staff requested that the licensee explain how the proposed LTOP setpoint limits the maximum pressure in the vessel to 100 percent of the pressure determined to satisfy Equation 1 since the LTOP pressure setpoint, as shown in Figure 1 of the licensee's March 25, 2015, RAI response, is higher that the cooldown limits calculated in accordance with Equation 1 of G-2215 (RAI 4).

In its April 7, 2015, RAI response, the licensee stated that the new proposed ERV LTOP setpoint would be reduced from 553.8 psig to 508 psig. The licensee further stated that this value, after adjustment for measurement and opening uncertainty, is to be used for the ERV setpoint whenever the RCS temperature is below the LTOP enable temperature. Additionally, in its RAI response, the licensee stated that the 508 psig ERV setpoint will not result in operational restrictions at low temperatures relative to net positive suction head requirements that would preclude heatup and cooldown. The NRC staff concludes that the new proposed setpoint of 508 psig is acceptable because once the ERV setpoint is adjusted for measurement and opening uncertainty, it will prevent exceeding 100 percent of the ASME Code, Section XI, Appendix G limits.

Additionally, in its RAI dated April 2, 2015, the NRC staff requested that the licensee describe a limiting mass and energy analysis for the new ERV setpoint at a level of detail consistent with SRP Section 5.2.2, "Overpressure Protection" (ADAMS Accession No. ML070540076) (RAI 1).

In its April 7, 2015, RAI response, the licensee responded that the NRC staff evaluated the limiting mass and energy analysis for LTOP and single failure in the "Safety Evaluation Resolving Multi-Plant Action item B-04, Reactor Vessel Overpressure Protection," dated May 5, 1983 (ADAMS Legacy Accession No. 8305180743). The licensee further stated that the most limiting credible mass input transient is a makeup control valve failing fully open, and that the steam or nitrogen volume in the pressurizer would allow for at least 10 minutes after the operator is alerted of the transient by the makeup line high-flow alarm before the pressure would reach the TS limit. The licensee stated that the most limiting heat addition transient was determined to be a loss of DHRS capability. The analysis demonstrated that if no operator action were taken, then the RCS pressure would increase to the ERV setpoint in approximately 15.7 minutes. With the failure of the pressurizer ERV, the 15.7 minutes would allow sufficient time for the operator to take corrective action to mitigate the transient. The mass and energy analysis is bounding with the ERV setpoint at 508 psig, in that it provides sufficient time for the operator to take corrective action to mitigate the transient.

The licensee identified to the NRC staff how the ERV setpoint meets the ASME Code, Section XI requirements. Additionally, the licensee described the limiting mass and energy analysis for the new ERV setpoint at a level of detail consistent with SRP Section 5.2.2. Based on the considerations discussed above, the NRC staff concludes that the licensee's proposed ERV setpoint of 508 psig will prevent RCS pressure from exceeding the P/T limits generated in accordance with ASME Code, Section XI, Appendix G, at and below the LTOP enable temperature. Therefore, the NRC staff concludes that the proposed LTOP setpoint is acceptable.

3.6 Summary and Conclusions

Based on the NRC staff's review of the information provided in the licensee's LAR, as supplemented, the NRC staff concludes that the proposed ANO-1 RPV P/T limits, valid for 54 EFPY, meet the requirements of Appendix G to 10 CFR Part 50 (taking into account the associated exemption granted on March 16, 2015). The NRC staff's conclusion is based on independent NRC staff evaluations and verification that the proposed P/T limits were developed appropriately. Therefore, the NRC staff concludes that the licensee's proposed TS revisions to reflect the use of these P/T limits are acceptable.

The NRC staff concludes the licensee's PTS analysis demonstrated that all RPV materials will not exceed the screening criteria of 10 CFR 50.61 through 54 EFPY.

The NRC staff concludes that the licensee has demonstrated that its USE and EMA analyses for the RPV materials will remain acceptable until 48 EFPY. The NRC staff further emphasizes, as discussed in Section 3.4.4 of this SE, that 10 CFR Part 50, Appendix G, Paragraph IV.A.1.c requires that the analysis of USE and EMA "must be submitted, as specified in § 50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of section IV.A.1 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate."

The NRC staff concludes that the licensee's proposed LTOP setpoint will prevent RCS pressure from exceeding the P/T limits generated in accordance with the ASME Code, Section XI, Appendix G, at and below the LTOP enable temperature, and is, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State 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 January 6, 2015 (80 FR 524). 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: April 24, 2015

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

Sincerely,

/RA/

Andrea E. George, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Amendment No. 254 to DPR-51
2. Safety Evaluation

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