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10 CFR 50.90

2CAN111702

November 20, 2017

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: License Amendment Request
Updating the Reactor Coolant System Pressure-Temperature Limits
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

REFERENCES:

1. Entergy submittal to NRC, "Reactor Vessel Surveillance Capsule Technical Report," dated October 17, 2016 (ML16293A583)

2. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004 (ML050120209)

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests an amendment to the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TS) to revise the Reactor Coolant System Pressure / Temperature Limits (TS 3.4.9). The proposed changes presented in this license amendment request (LAR) will replace the current reactor pressure vessel pressure-temperature (P-T) limits, applicable to 32 Effective Full Power Years (EFPY), with new P-T limits applicable to 54 EFPY (approximately 60 calendar years). In addition, the minimum boltup temperature is being revised in accordance with Reference 2. This impacts the range of temperatures for the cooldown curves provided in TS 3.4.9.

Other P-T related TSs (i.e., TS 3.4.12, "Low Temperature Overpressure Protection (LTOP) System) were evaluated as part of this LAR and it was determined that no additional changes were required.

Reference 1 submitted the results from the evaluation of the last reactor vessel material surveillance capsule removed from ANO-2. These results were used in the development of the analyses performed to support this LAR.

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The proposed P-T limits for the reactor pressure vessel were developed in accordance with Reference 2 and the requirements of 10 CFR 50, Appendix G, using the analytical methods and flaw acceptance criteria of American Society of Mechanical Engineers (ASME) Code Section XI, Appendix G.

The projected fluence values at 54 EFPY are consistent with the NRC approved methodology presented in Reference 2.

The enclosure provides a detailed description and analysis of the proposed changes. Attachment 1 of the enclosure provides the annotated TS pages showing the proposed changes. Attachment 2 of the enclosure provides the clean copy of the TS pages showing the proposed TS changes and Attachment 3 of the enclosure is the Westinghouse developed report WCAP-18169-NP, Revision 0, "Arkansas Nuclear One Unit 2 Heatup and Cooldown Limit Curves for Normal Operation". Attachment 4 provides the proposed changes to the TS bases for information.

The requested changes involve no significant hazards consideration and do not contain any new regulatory commitments.

It is conservatively estimated that ANO-2 will reach 32 EFPY by March 3, 2019. Based on this, Entergy requests approval of the proposed amendment by February 1, 2019. Once approved, the amendment shall be implemented within 30 days.

In accordance with 10 CFR 50.91, Entergy is notifying the State of Arkansas of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

If there are any questions or if additional information is needed, please contact Stephenie Pyle, Manager, Regulatory Assurance at (479) 858-4764.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on November 20, 2017.

Sincerely,

Richard L. Anderson

RLA/rwc

Enclosure: Evaluation of the Proposed Change

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Enclosure to

2CAN111702

Evaluation of the Proposed Change

EVALUATION OF THE PROPOSED CHANGE

1.0 DESCRIPTION

The proposed amendment would modify the Technical Specifications (TS) associated with Arkansas Nuclear One, Unit 2 (ANO-2) Renewed Operating License NPF-6 to replace the current pressure-temperature (P-T) limits for heatup, cooldown, and the inservice leak hydrostatic tests for the Reactor Coolant System (RCS) presented in TS 3.4.9 that expire at 32 Effective Full Power Years (EFPY) with limitations that extend out to 54 EFPY.

Based on the operating history of ANO-2 and the predicted operation of the unit, ANO-2 will reach 32 EFPY by March 3, 2019. This is when the current limitations will expire. The 54 EFPY time period will bound the operation of ANO-2 until the end of the current period of extended operation (i.e., 60 calendar years). Entergy has assumed that the unit would operate with an average capacity factor of 90% over this time period.

The proposed amendment would revise TS 3.4.9 to incorporate updated figures for the P-T limits. These figures have been recalculated to account for 54 EFPY of plant operation. The minimum boltup temperature for the ANO-2 vessel has been revised as well. Instrument uncertainty has not been included in these limits. The requested change involves no significant hazards consideration as outlined in Section 3.3 of this enclosure.

Based on the new limits, the enable temperature and other related Low Temperature Overpressure Protection System (LTOP) limits presented in TS 3.4.12 were evaluated and it was determined that this TS did not need to be revised. In addition, this license amendment request (LAR) does not include changes to the heatup or cooldown rates, reactor coolant pump configurations, or the design bases LTOP transients.

Attachment 1 of this enclosure contains the affected TS pages annotated with the proposed changes. A clean copy of the TS pages is provided in Attachment 2. Attachment 3 contains WCAP-18169-NP, "Arkansas Nuclear One Unit 2 Heatup and Cooldown Limit Curves for Normal Operation". This Westinghouse developed report provides the methodology and technical results of the generation of the heatup and cooldown P-T limit curves for the ANO-2 reactor vessel. Attachment 4 provides a copy of the marked up TS bases for information.

2.0 TECHNICAL EVALUATION

The NRC has established requirements in 10 CFR 50, Appendix G (Reference 1) in order to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. Reference 1 requires that the P-T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Reference 2) were used to generate the P-T limits. Also, Reference 1 requires that applicable surveillance data from reactor pressure vessel (RPV) material surveillance programs be incorporated into the calculations of plant-specific P-T limits, and the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials.

Appendix H to 10 CFR 50 (Reference 3) establishes the requirements related to the reactor vessel material surveillance programs. The reactor vessel material surveillance capsule technical report was submitted to the NRC via Reference 4. The capsule was withdrawn at approximately 29.24 EFPY. The results from the testing of this capsule were used in the development of the proposed limits.

The reference temperature for nil-ductility (RT_{NDT}) of material increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting reference temperature at any time period, this change in (RT_{NDT}) must be added to the unirradiated reference temperature. The extent of the shift in the reference temperature is enhanced by certain chemical elements (such as copper and nickel) present in the vessel steels. Regulatory Guide (RG) 1.99, Revision 2 (Reference 5), contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron irradiation. These methodologies were used in the development of the proposed 54 EFPY limitations.

Reference 6 was developed to define a methodology for RPV P-T limit curve development and, consistent with the guidance provided in NRC Generic Letter (GL) 96-03 (Reference 7) for the development of plant-specific Pressure-Temperature Limit Reports (PTLRs). While this LAR for ANO-2 is not to relocate the P-T and LTOP system to a PTLR, the methodology used to develop the proposed limits are the same as described in Reference 5. The methodology outlined in Reference 5 is not dependent upon the use of a PTLR. Attachment 3 is considered to be a plant-specific application report of Reference 6.

Four specific topics are addressed in the context of the development of the methodology presented in Reference 6: (1) the calculation of neutron fluences for the RPV and the RPV surveillance capsules; (2) the evaluation of the RPV material properties due to changes caused by neutron radiation; (3) the development of appropriate P-T limit curves based on these pressure vessel material properties and the establishment of the LTOP system setpoints and controls to protect the pressure vessel from brittle fracture; and, (4) the development of a pressure vessel material surveillance program to monitor changes in pressure vessel material properties due to radiation.

The proposed 54 EFPY P-T limitations were developed in accordance with References 1 and 6, using the analytical methods and flaw acceptance criteria of Reference 2. Specifically, the K_{IC} methodology of the 1998 through the 2000 Addenda Edition of the ASME Code, Section XI, Appendix G, was used.

The minimum boltup temperature has been revised from the previous 50 °F to 60 °F in accordance with Reference 6.

To address plant operation through the period of extended operation (54 EFPY), neutron (E > 1 MeV) fluence projections were updated, reactor vessel embrittlement analyses were performed, and updated P-T limits were developed in accordance with the guidance provided above.

Beltline Region Determination

Of particular interest in this analysis is the reactor vessel beltline, which is defined in Reference 1, as:

> The region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

As described in NRC Regulatory Issue Summary (RIS) 2014-11 (Reference 8), any reactor vessel material predicted to experience a neutron fluence exposure greater that 1.0×10^{17} n/cm^2 (E > 1 MeV) at the end of the licensed operating period should be considered in the development of P-T limits. This means the traditional beltline of the reactor vessel may have expanded.

The methodology and evaluations used to determine the initial RT_{NDT} values for the beltline and the extended beltline base metal materials were reviewed and updated, as appropriate, in support of this LAR. Section 3 of Attachment 3 provides the specific methodologies, evaluations and the values of the initial RT_{NDT} for each of the beltline and extended beltline materials. Also the initial RT_{NDT} values for the reactor vessel flange, reactor vessel closure head, the replacement reactor vessel closure head, and the balance of the RCS are provided in Section 3 of Attachment 3.

The boltup temperature is the minimum allowable temperature at which the vessel closure head bolts can be preloaded and is determined by the highest RT_{NDT} in the closure flange region. This requirement is established in Reference 1. In accordance with the Reference 6 methodology, the minimum boltup temperature should be 60 °F or the limiting unirradiated RT_{NDT} of the closure flange region, whichever is higher. A review of Section 3 of Attachment 3 demonstrates the limiting unirradiated RT_{NDT} of this region is less than 60 °F; therefore the minimum boltup temperature is 60 °F. This value does not include instrument uncertainties.

Fluence Determination

The fluence and the fluency benchmarking methodologies as outlined in Reference 5 were found to adhere to the guidance of RG 1.190 (Reference 9) as well as to the appropriate American Society of Testing and Materials (ASTM) standards and is acceptable for use.

An ANO-2 specific discrete ordinates transport (DORT) analysis was performed to determine the neutron (E > 1 MeV) radiation environment within the RPV. In the analysis, radiation exposure parameters were established on a plant- and fuel-cycle-specific basis. An evaluation of the dosimetry sensor sets from the 284° and 97° surveillance capsules is provided in Reference 4. The dosimetry analysis shows that the ± 20% acceptance criterion specified in Reference 9 is met. The validated calculations form the basis for providing projections of the neutron exposure of the reactor pressure vessel for operating periods extending to 54 EFPY. Details of the fluence evaluation are provided in Attachment 3.

In Reference 10 the NRC staff stated that there are several NRC-approved methods for a licensee may reference in developing a PTLR and would be acceptable for implementation of a PTLR. The staff has confirmed previously that Reference 6 addresses fluence in a manner that adheres to the guidance contained in Reference 9 (for Revision 4).

Pressure-Temperature Limits

The ability of the RPV to resist fracture is the primary factor in ensuring the safety of the primary system in light water-cooled reactors. A method for guarding against brittle fracture in RPVs is described in Reference 2. This method utilizes fracture mechanics concepts and the RT_{NDT}. The RT_{NDT} is defined as the greater of the drop weight nil-ductility transition temperature (per ASTM E208) or the temperature at which the material exhibits 50 foot-pounds (ft-lb) absorbed energy and 35 mils lateral expansion minus 60 °F. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IC}). The K_{IC} curve appears in Reference 2. When a given material is indexed to the K_{IC} curve and applied thermal stress intensity factors and unit pressure stress intensity factors determined, then the allowable pressures can be obtained for this material as a function of temperature. Operating P-T limits can then be determined for a given heatup or cooldown temperature - time history. The RT_{NDT} of the reactor vessel materials must be adjusted to account for the effects of irradiation. Neutron embrittlement and the resultant changes in mechanical properties of a given pressure vessel steel are monitored by a surveillance program. See ANO-2 Safety Analysis Report (SAR) Section 5.2.4.4 for a description of the reactor vessel material surveillance program for ANO-2. The increase in the Charpy V-notch temperature is added to the unirradiated RT_{NDT} to adjust it for neutron embrittlement. This adjusted RT_{NDT} (ART) is used to index the material to the K_{IC} curve, which in turn, is used to set new operating limits. These new limits take into account the effects of irradiation on the vessel materials.

The ART is defined as the sum of the initial RT_{NDT}, the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (σ) term. The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor (FF). The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in Reference 5 or from surveillance data. The FF is dependent upon the neutron fluence (E > 1 MeV) at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in Reference 5 or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel content, the neutron fluence, and the calculational procedures. Reference 5 describes the methodology to be used in calculating the margin term.

Appendix D of Attachment 3 contains a credibility evaluation for weld Heat # 10137 considering all applicable sister plant surveillance program data.

The ANO-2 RPV contains both axially and circumferentially oriented welds. Therefore, the P-T limits are based on the postulation of axial flaws and circumferential flaws, as appropriate. An axial flaw (which is more limiting than a circumferential flaw) is postulated for axial weld and plate materials. A circumferential flaw is postulated for circumferential weld materials as allowed by Code Case N-588 and Code Case N-641. Since the limiting material in the ANO-2 reactor vessel is a plate material, an axial flaw was postulated in this material.

Attachment 3 provides a summary of the technical basis leading to the development of the new P-T limits. The data presented in Attachment 3 was used to create the TS figures presented in Attachment 2. The limits are generated for normal operation heatup, normal operation cooldown, inservice leak and hydrostatic (ISLH) test conditions, and reactor core operations. These limits are expressed in the form of curves of allowable pressure versus temperature. The P-T limits were determined for 54 EFPY. Pressure Corrections factors were developed to adjust the reactor vessel (RV) beltline limits to the Pressurizer where the operators monitor RCS pressure. Instrument uncertainty was not included in the limits listed in this attachment. The instrument uncertainties will be applied in the appropriate operating procedures.

Appendix B of Attachment 3 contains a P-T limit evaluation of the reactor vessel inlet and outlet nozzles based on a 1/4T flaw postulated at the inside surface of the corner of the nozzle. The P-T limit curves generated based on the limiting cylindrical beltline material bound the P-T limit curves for the vessel inlet and outlet nozzles.

Low Temperature Overpressurization Protection Limits

LTOP limits were based on the ASME Code, Section XI, Article G-2215., than credited in the analysis of record. This article requires that the updated LTOP system ensures that the maximum pressure from the limiting P-T curve is not exceeded when the 1/4T temperature is less than the ART+ 50 °F.

The limiting LTOP events were reanalyzed to address a higher backpressure in the system and a higher High Pressure Safety Injection flowrate. The current LTOP relief valve setting was shown to be acceptable by comparison of the limiting LTOP transients to the 54 EFPY P-T limits. The enable temperature was reevaluated as well and was determined to be acceptable as is. Based on these evaluations, ANO-2 TS 3.4.12 is not being revised due to the change to the P-T limits.

Pressurized Thermal Shock

In 1985, the NRC issued 10 CFR 50.61 (Reference 11) that established the screening criteria for pressurized water reactor vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of the license, termed RT_{PTS}. The PTS screening criterion is 270 °F for plates, forgings, and axial weld materials and 300 °F for circumferential weld materials.

A PTS assessment for the ANO-2 reactor vessel beltline materials with fluences greater than 1 E+17 n/cm^2 was performed. The controlling material is the Lower Shell Plate C-8010-1, with a predicted RT_{PTS} value of 122.4 °F. The remaining results are provided in Section 4 of Attachment 3 of this submittal.

Upper-Shelf Energy

The requirements related to the Upper-Shelf Energy (USE) are included in Reference 1. Reference 1 requires an analysis at least three years prior to the time that the USE of any pressure vessel material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

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

3.0 REGULATORY EVALUATION

3.1 Applicable Regulatory Requirements/Criteria

The NRC has established requirements in Title 10 of the Code of Federal Regulations (10 CFR) Part 50, to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates the P/T limits based on the following regulations and guidance.

Reference 1 requires that P-T limits be at least as conservative as those obtained by applying the methodology of Reference 2. Reference 1 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves. GL 88-11 (Reference 12) advised licensees that the NRC staff would use Reference 5 to review P-T limits.

GL 92-01 (Reference 13) requested that licensees submit RPV materials property data for the respective plant to the NRC staff for review. GL 92-01, Revision 1, Supplement 1 (Reference 14), requested that licensees provide and assess data from other licensees that could affect the related RV integrity evaluations.

The Standard Review Plan, Branch Technical Position 5-3, Revision 2, of NUREG-0800, provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RV based on the linear elastic fracture mechanics methodology of Reference 2. The basic parameter of this methodology is the stress intensity factor, K_{IC}, which is a function of the stress state and flaw configuration. Reference 2 requires a safety factor of 2.0 on stress intensities resulting from pressure during normal and transient operating conditions, and a safety factor of 1.5 on these stress intensities for hydrostatic testing curves.

The flaw postulated in Reference 2 has a depth that is equal to 1/4T and a length equal to 1.5 times the RV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limits are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Reference 2 requires that the ART be determined by evaluating material property changes due to neutron irradiation.

Section 50.60 of 10 CFR imposes fracture toughness and material surveillance program requirements, which are set forth in 10 CFR 50, Appendices G and H. In the "Definitions" section of Appendix G, Paragraph G.II.D(ii) states, "For the reactor vessel beltline materials, ΔRT_{NDT} must account for the effects of neutron radiation." In the "Fracture Toughness Requirements" section, Paragraph G.IV.A states in part, " ... the values of RT_{NDT} and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part." The effects of neutron radiation are determined, in part, by estimating the neutron fluence on the reactor vessel.

Reference 11 established the screening criteria for pressurized water reactor vessel embrittlement, as measured by the maximum reference nil-ductility transition temperature in the limiting beltline component at the end of the license.

The methodologies for determining the increase in transition temperature and the decrease in USE resulting from neutron radiation can be found in Reference 5.

Reference 9 describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence with respect to the General Design Criteria (GDC) contained in Appendix A of 10 CFR 50.

In consideration of the guidance set forth in Reference 9, GDC 14, 30, and 31 are applicable. GDC 14 requires the design, fabrication, erection, and testing of the reactor coolant pressure boundary 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 reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical.

GDC 31 pertains to the design of the reactor coolant pressure boundary, 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 non-brittle 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.

Additionally GDC 15 and 32 are applicable to this request. GDC 15 requires the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel. (GDC 32)

In the original SER for ANO-2 (Reference 15), the NRC stated the following with regards to the GDCs:

> Arkansas Nuclear One – Unit 2 was designed and is being constructed on the proposed AEC General Design Criteria which were published July 11, 1967. Design and construction were thus initiated and proceeded to a significant extent based upon the criteria proposed in 1967. Since July 15, 1971, when the Atomic Energy Commission published the General Design Criteria of Appendix A of 10 CFR 50 the applicant has attempted to comply with the newer criteria to the extent practical. Recognizing work already accomplished and design commitments made, the applicant discusses in Section 3.1 of the Final Safety Analysis

Report the design of ANO-2 with respect to the criteria of July 15, 1971. As a result, our technical review assessed the plant against the General Design Criteria now in effect and we have concluded that the plant design conforms to the intent of these newer criteria.

Criteria 14 and 15 are discussed in Section 3.1.2 of the ANO-2 SAR. Criteria 30, 31, and 32 are discussed in SAR Section 3.1.4.

The NRC found that Reference 6 was acceptable for referencing as a PTLR methodology, subject to the following conditions:

1. Licensee who wish to use WCAP-14040, Revision 3, as their PTLR methodology must provide additional information to address the methodology requirements discussed in provision 2 in the table of Attachment 1 to GL 96-03 related to the reactor pressure vessel material surveillance program.

   This LAR for ANO-2 is not to relocate the P-T limits to a PTLR; however, the methodology described in Reference 6 was used in the development of the proposed limit curves. To be complete, the requested information is provided below.

   Provision 2 in the table of Attachment 1 to Reference 7 provides the minimum required information to be included in the PTLR is the surveillance capsule withdrawal schedule, or reference by title and number the documents in which the schedule is located. Reference the surveillance capsule reports by title and number if ARTs are calculated using surveillance data.

   The ANO-2 reactor vessel material surveillance program is described in the ANO-2 SAR, Section 5.2.4.4. The results of the capsule reports are described in Section 5.2.4 except for the latest surveillance capsule report. That was transmitted to the NRC via Reference 4. The current surveillance capsule withdrawal schedule is provided in ANO-2 SAR Table 5.2-12. Reference 16 transmitted a revision to the schedule to the NRC for review and approval. That request is still under review.

2. Contrary to the information in WCAP-14040, Revision 3, licensee use of the provisions of ASME Code Cases N-588, N-640, or N-641 in conjunction with the basic methodology in WCAP-14040, Revision 3, does not require an exemption since the provisions of these Code Cases are contained in the edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a. When published, the approved revision (Revision 4) of WCAP-14040 should be modified to reflect this NRC staff conclusion.

   Reference 6 is the NRC-approved version of WCAP-14040, Revision 3. The requirement of an exemption does not exist in Reference 6. No further actions are required.

3. As stated in WCAP-14040, Revision 3, until Appendix G to 10 CFR Part 50 is revised to modify / eliminate the existing reactor pressure vessel flange minimum temperature requirements or an exemption request to modify / eliminate these requirements is approved by the NRC for a specific facility, the stated minimum temperature must be incorporated into a facility's pressure-temperature curves.

   The pressure vessel flange minimum temperature requirements are discussed in Attachment 3. These requirements were incorporated into the proposed P-T limits.

The proposed changes to the ANO-2 TS do not affect compliance with the above regulations or guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met.

3.2 Precedent

The proposed amendment would allow use of Reference 6 for the development of pressure and temperature limit curves. The NRC previously approved the WCAP-14040-NP-A, Revision 4, for the development of pressure and temperature limit curves at the Vogtle Electric Generating Plant (Reference 17), Comanche Peak (Reference 18), and Wolf Creek (Reference 10).

3.3 No Significant Hazards Consideration Analysis

Entergy Operations, Inc. (Entergy) proposes a change to the Arkansas Nuclear One, Unit 2 (ANO-2) Technical Specifications (TSs) that would revise the pressure-temperature limits for the Reactor Coolant System (except the Pressurizer).

Entergy has evaluated the proposed changes to the TS using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

Basis for no significant hazards consideration determination: As required by 10 CFR 50.91(a), Entergy analysis of the issue of no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change will revise the pressure-temperature (P-T) limits for heatup, cooldown, and inservice leak hydrostatic test limitations for the Reactor Coolant System (RCS) to a maximum of 54 Effective Full Power Years (EFPY) in accordance with 10 CFR 50, Appendix G. This is the end of the period of extended operation for the renewed ANO-2 operating License. The P-T limits were developed in accordance with the requirements of 10 CFR 50, Appendix G, utilizing the analytical methods and flaw acceptance criteria of Topical Report WCAP-14040, Revision 4, and American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G. These methods and criteria are the previously NRC approved standards for the preparation of P-T limits. Updating the P-T limits for additional EFPYs maintains the level of assurance that reactor coolant pressure boundary integrity will be maintained, as specified in 10 CFR 50, Appendix G.

   The proposed changes do not adversely affect accident initiators or precursors, and do not alter the design assumptions, conditions, or configuration of the plant or the manner in which the plant is operated and maintained. The ability of structures, systems, and components to perform their intended safety functions is not altered or prevented by the proposed changes, and the assumptions used in determining the radiological consequences of previously evaluated accidents are not affected.

   Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed changes implement methodologies that have been approved by the NRC (provided that any conditions / limitations are satisfied). The P-T limits will ensure the protection consistent with assuring the integrity of the reactor coolant pressure boundary as was previously evaluated. Reactor coolant pressure boundary integrity will continue to be maintained in accordance with 10 CFR 50, Appendix G, and the assumed accident performance of plant structures, systems and components will not be affected. These changes do not involve any physical alteration of the plant (i.e., no new or different type of equipment will be installed), and installed equipment is not being operated in a new or different manner. Thus, no new failure modes are introduced.

   Therefore, this change does not create the possibility of a new or different kind of accident from an accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed changes do not affect the function of the reactor coolant pressure boundary or its response during plant transients. By calculating the P-T limits using NRC-approved methodology, adequate margins of safety relating to reactor coolant pressure boundary integrity are maintained. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. These changes will ensure that protective actions are initiated and the operability requirements for equipment assumed to operate for accident mitigation are not affected.

   Therefore, this change does not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above, Entergy concludes that the requested change involves no significant hazards consideration, as set forth in 10 CFR 50.92(c), "Issuance of Amendment."

3.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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.

4.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

5.0 REFERENCES

1. 10 CFR 50, Appendix G, "Fracture Toughness Requirements"

2. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Non-mandatory Appendix G

3. 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

4. Entergy submittal to NRC, "Reactor Vessel Surveillance Capsule Technical Report," dated October 17, 2016 (2CAN101602) (ML16293A583)

5. U. S. NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988

6. Westinghouse Report WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004 (ML050120209)

7. U. S. NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996

8. U. S. NRC Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," dated October 14, 2014 (ML14149A165)

9. U. S. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001

10. NRC letter to Wolf Creek Nuclear Operating Corporation, dated January 27, 2009, Wolf Creek Generating Station – Issuance of Amendment Re: Application to Revise Technical Specification 5.5.6, Reactor Coolant System Pressure and Temperature Limits Report (TAC No. MD9217) (ML083430224)

11. 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events"

12. Generic Letter 88-11, “NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations”

13. Generic Letter 92-01, “Reactor Vessel Structural Integrity,” Revision 1, March 6, 1992

14. Generic Letter 92-01, "Reactor Vessel Structural Integrity,” Revision 1, Supplement 1, May 19, 1995

15. NUREG-0308, Safety Evaluation Report related to the operation of Arkansas Nuclear One, Unit 2, November 1977, (ML102850078)

16. Entergy submittal to NRC, “Proposed Revision to Reactor Vessel Surveillance Capsule Withdrawal Schedule,” dated September 14, 2017 (2CAN091702) (ML17257A121)

17. NRC letter to Southern Nuclear Operating Company, Inc., dated March 28, 2005, Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Exemption and Amendments Re: Request to Revise Technical Specifications and Pressure Temperature Limits Report and Relocate the Cold Overpressure Protection System (COPS) Arming Temperature (TAC Nos. MC2225, MC2226, MC2227, MC2228, MC3090, and MC3091) (ML050690216)

18. NRC letter to TXU Power, dated February 22, 2007, Comanche Peak Steam Electric Station, Units 1 and 2 – Issuance of Amendment Re: Revise Technical Specification 5.6.6 on Reactor Coolant System Pressure and Temperature Limits Report (TAC Nos. MC9500 and MC9501) (ML070320823)

<u>ATTACHMENTS</u>

1. Proposed Technical Specification Changes (mark-up)

2. Revised (clean) Technical Specification Pages

3. WCAP-18169-NP, “Arkansas Nuclear One Unit 2 Heatup and Cooldown Limit Curves for Normal Operation,” Revision 0, December 2016

4. Proposed Technical Specification Bases Changes (mark-up)

Enclosure Attachment 1 to

2CAN111702

Proposed Technical Specification Changes (mark-up)

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

b. A maximum cooldown rate of 100 °F per hour (constant) or 50 °F in any half hour period (step) for RCS cold leg temperatures between 6~~5~~0 °F and 560 °F.

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

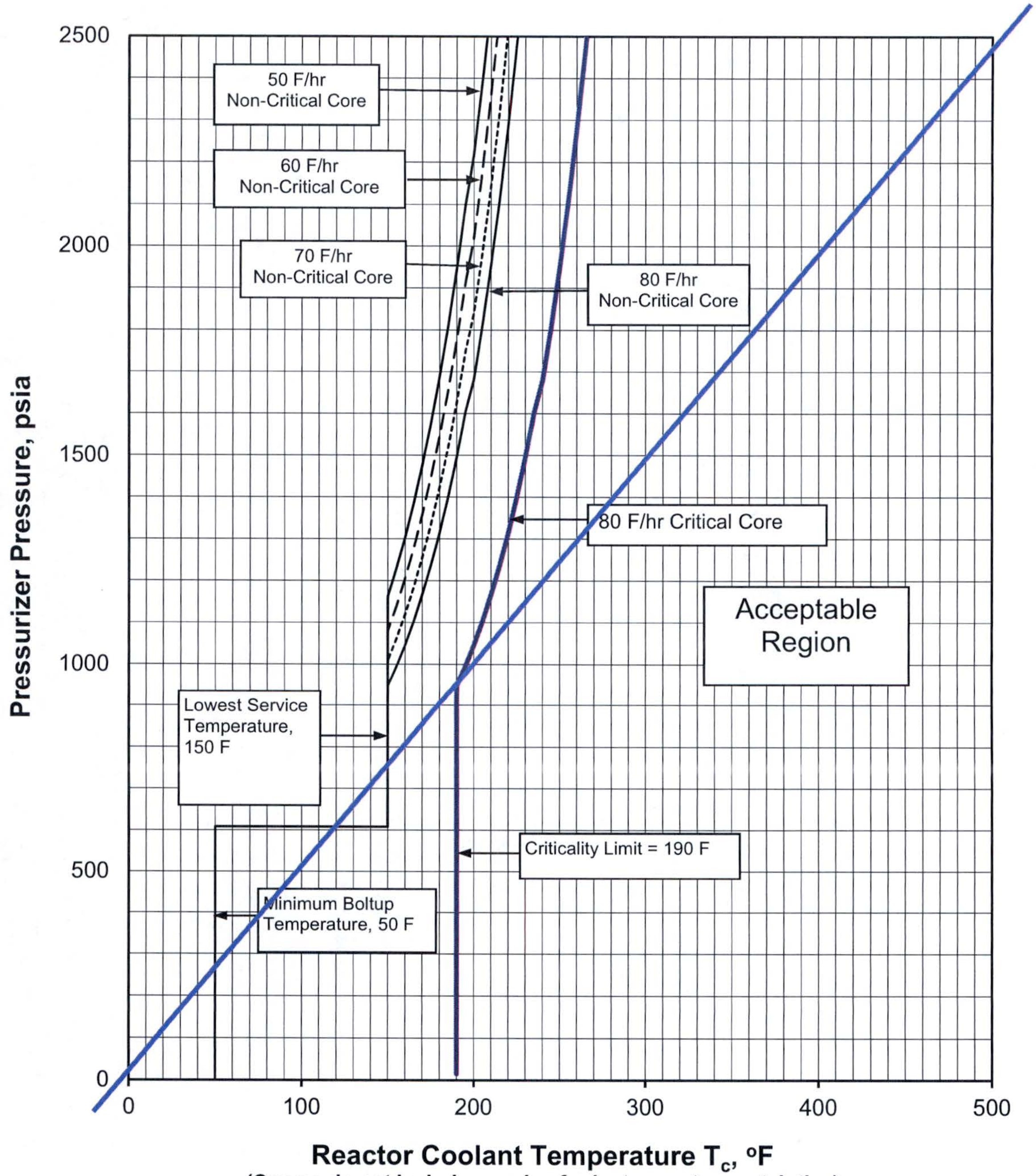
APPLICABILITY: At all times.

ACTION:

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

Figure 3.4-2A

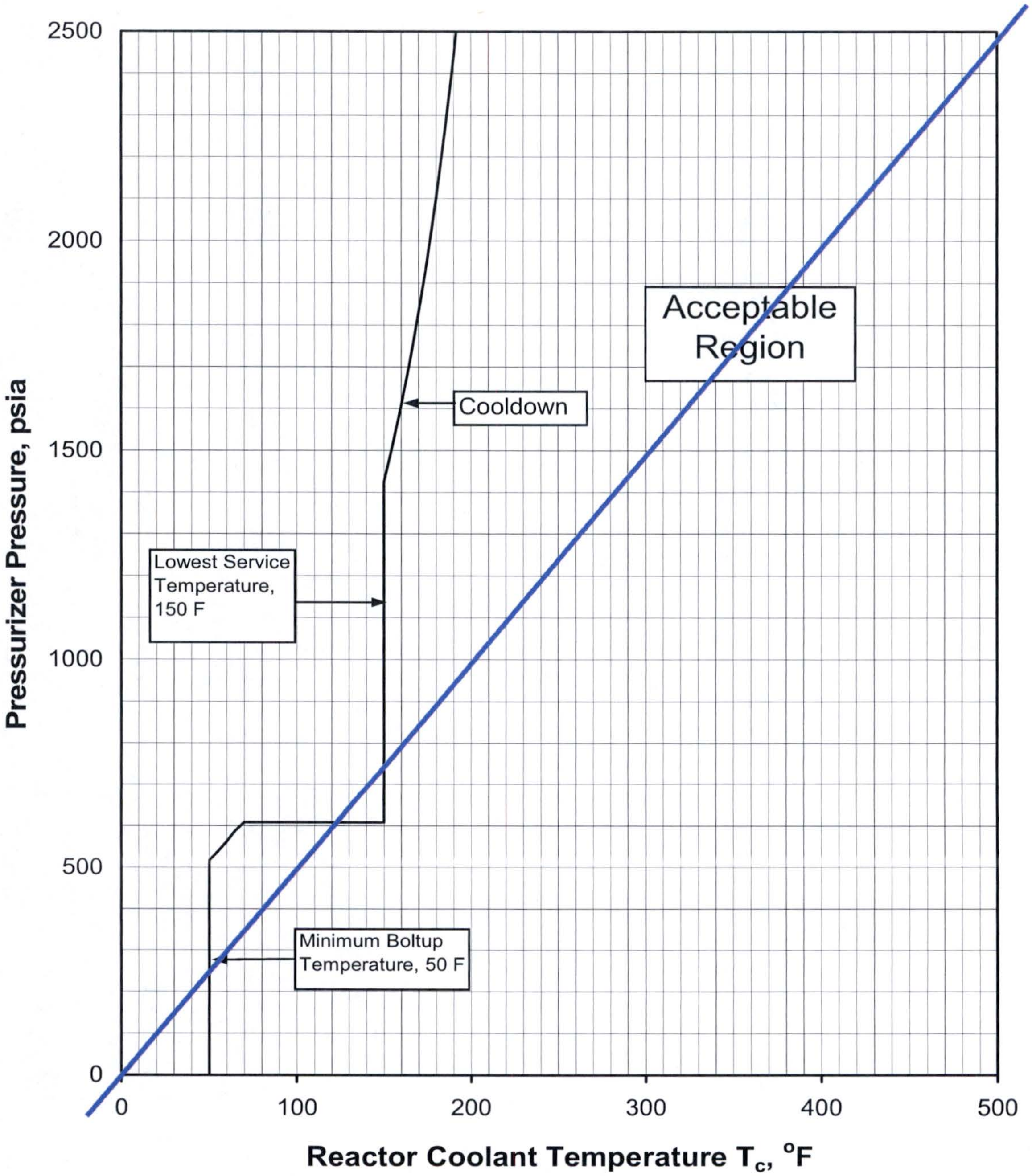
HEATUP CURVE – ~~32~~54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



(Curves do not include margins for instrument uncertainties)

Figure 3.4-2B

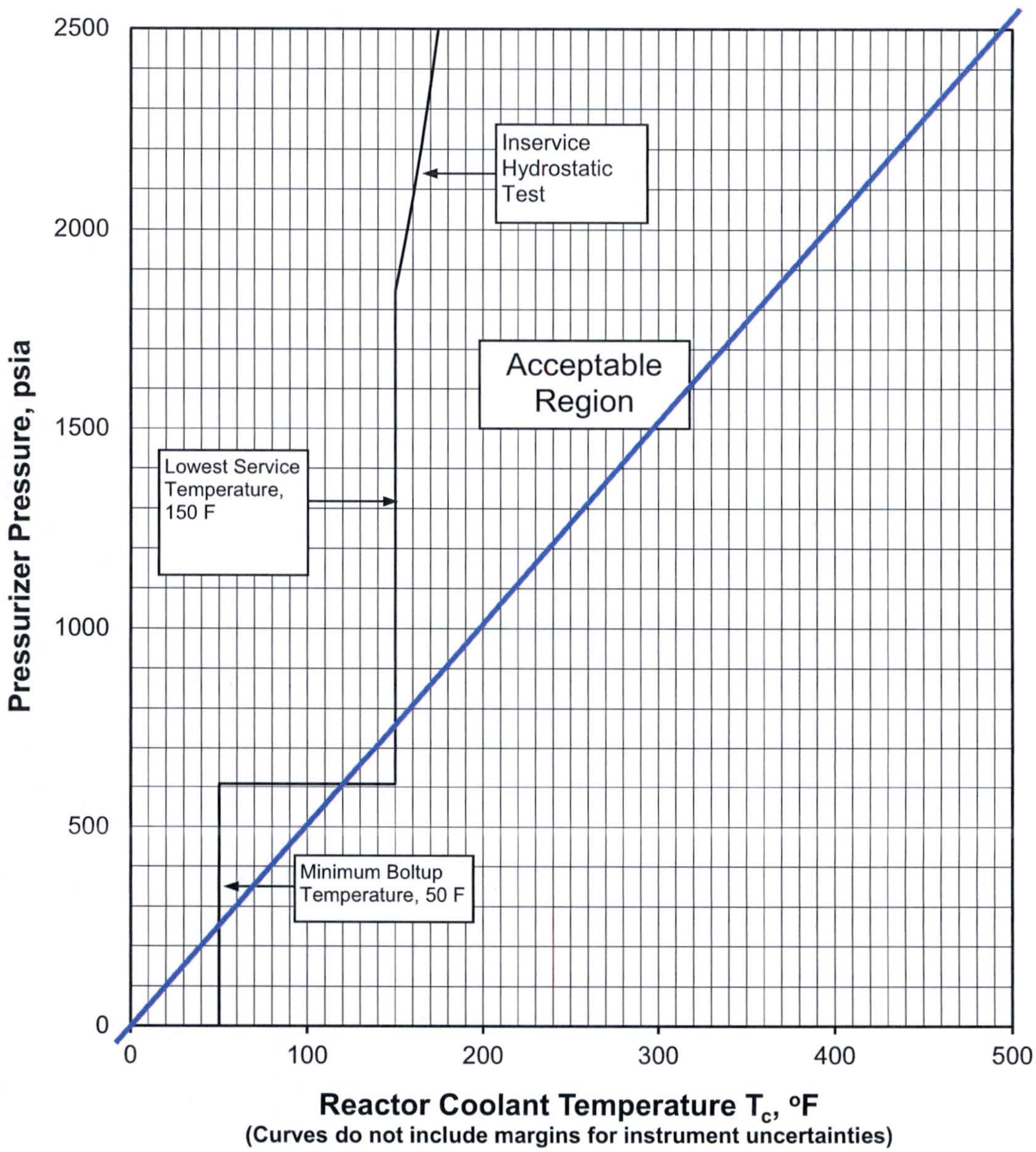
COOLDOWN CURVE – ~~32~~54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



(Curves do not include margins for instrument uncertainties)

Figure 3.4-2C

INSERVICE HYDROSTATIC TEST CURVE~~S~~ – ~~32~~54 EFPY
REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS



Reactor Coolant Temperature T_c, °F
(Curves do not include margins for instrument uncertainties)

Enclosure Attachment 2 to

2CAN111702

Revised (clean) Technical Specification Pages

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

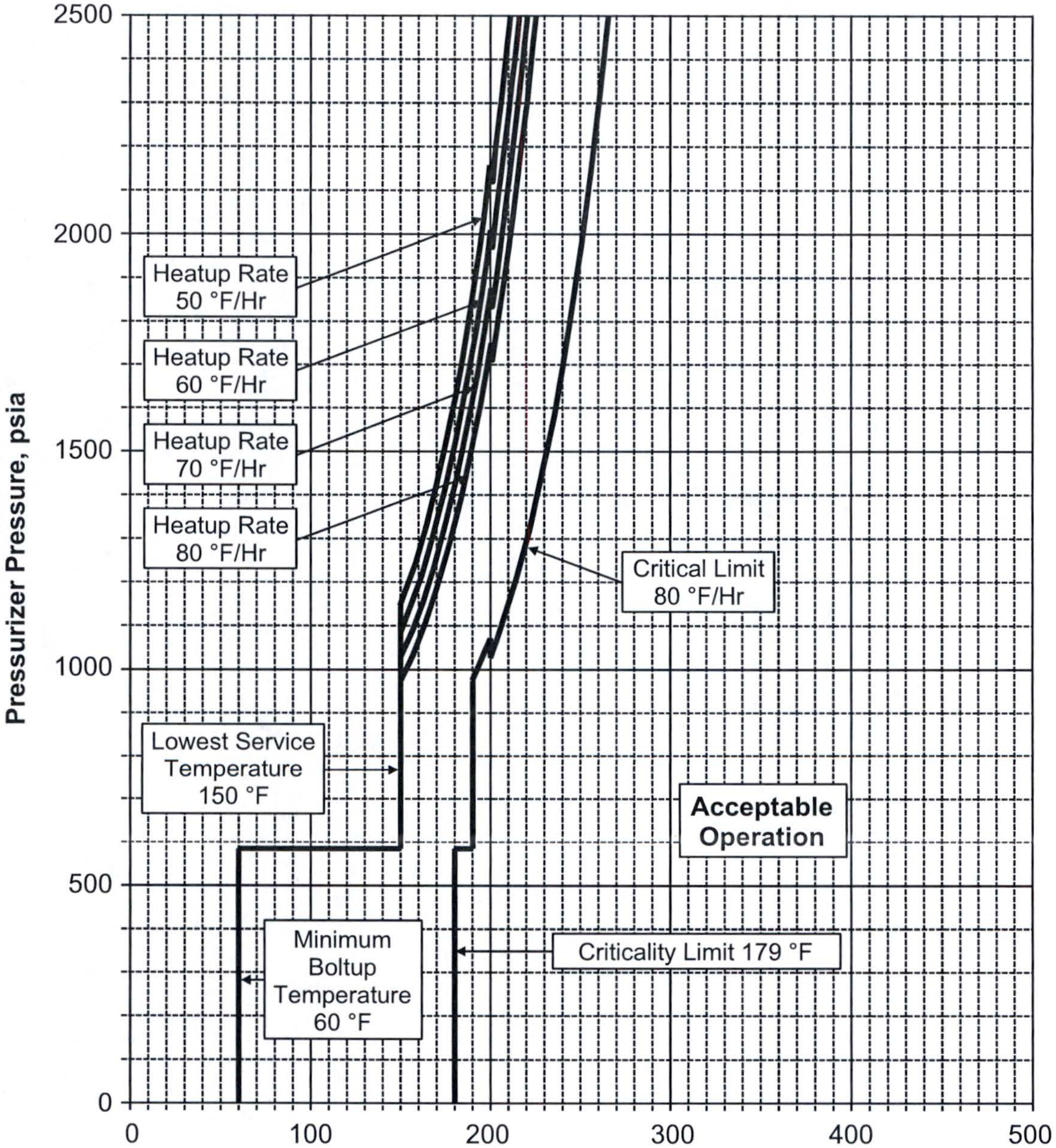
APPLICABILITY: At all times.

ACTION:

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

Figure 3.4-2A

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

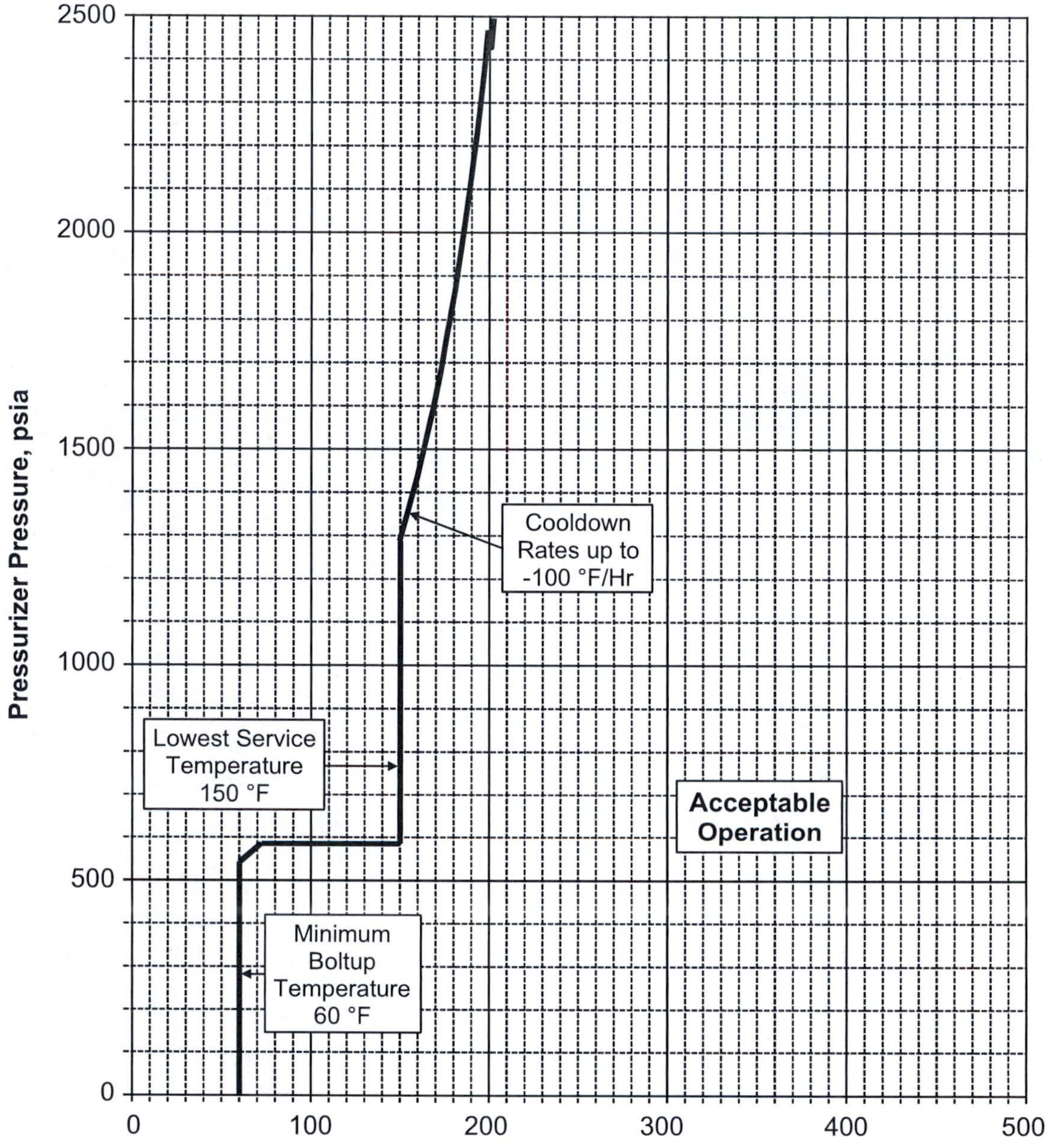


Reactor Coolant Temperature T_c, °F

(Curves do not include margins for instrument uncertainties)

Figure 3.4-2B

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

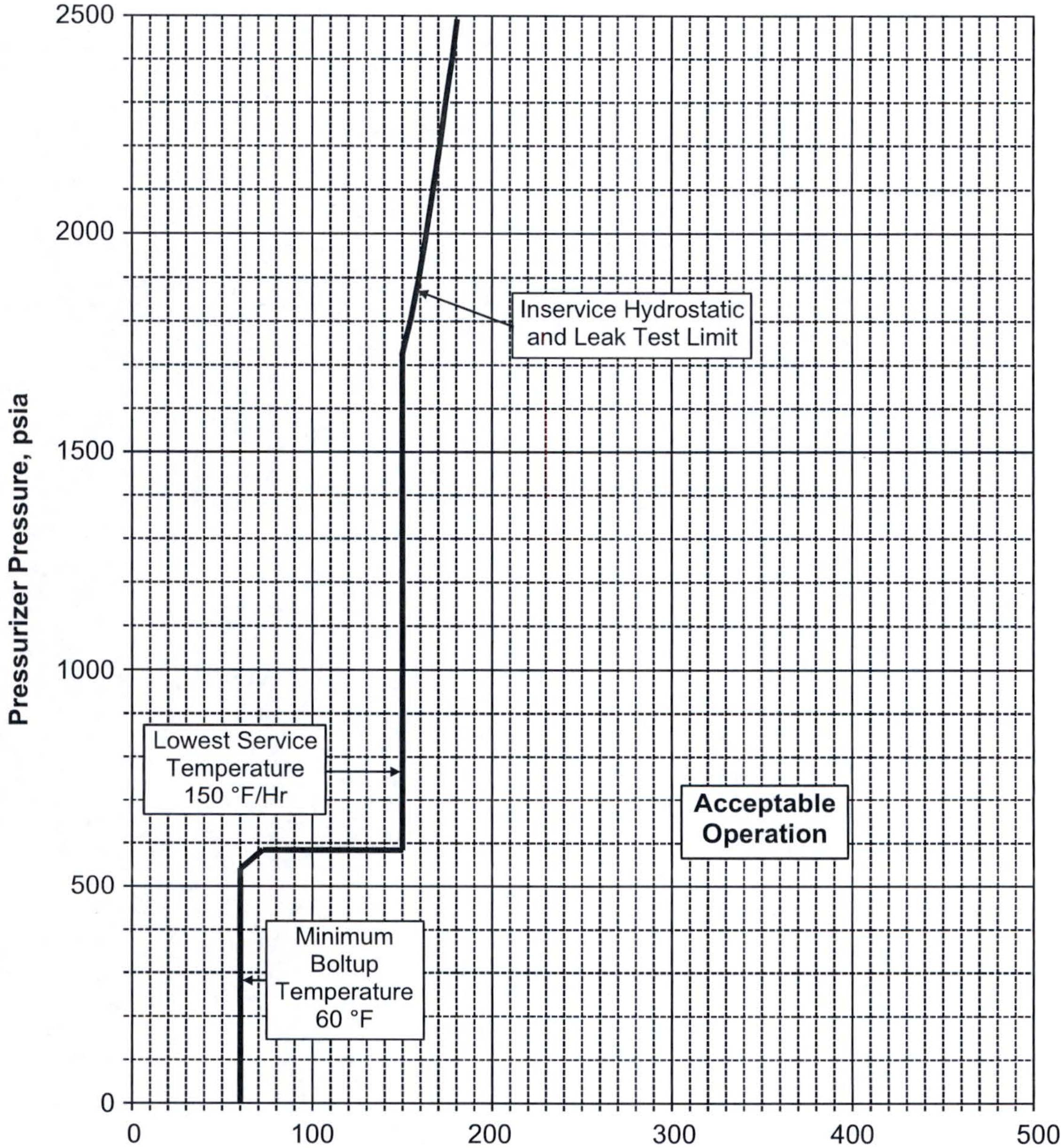


Reactor Coolant Temperature T_c, °F

(Curves do not include margins for instrument uncertainties)

Figure 3.4-2C

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



Reactor Coolant Temperature T_c, °F

(Curves do not include margins for instrument uncertainties)

Enclosure Attachment 4 to

2CAN111702

Proposed Technical Specification Bases Changes (mark-up)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. The heatup/criticality, cooldown, and hydrostatic test limit curves for 54~~32~~ EFPY shown on Figures 3.4-2A, 3.4-2B, and 3.4-2C include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for the location ~~and for possible errors in~~of the pressure ~~and temperature~~ sensing instruments. It should be noted that the location adjustment considered the operation of three RCPs from a RCS temperature equal to or above the LTOP enable temperature and a maximum of two RCPs operating while below the LTOP enable temperature. Instrument uncertainty in these curves is not included, but is added in station procedures.

The shift in the limiting material fracture toughness, as represented by RT_{NDT}, is calculated using Regulatory Guide 1.99, Revision 2. For 54~~32~~ EFPY, at the 1/4t position, the adjusted reference temperature (ART) value is ~~112.7~~116.8 °F. At the 3/4t position the ART value is ~~98.8~~103.4 °F. These values are conservatively based on a reactor vessel clad / base metal interface~~inner surface~~ fluence of ~~3.79~~4.98 x 10^{19} n~~vt~~/cm^2. The fluence at the 1/4t point is ~~2.29~~3.10 x 10^{19} n~~vt~~/cm^2 and the fluence of the 3/4t point is ~~8.92~~1.21 x $10^{1\cancel{8}9}$ n~~vt~~/cm^2. These ART values were conservatively increased to 122 °F (1/4t) and 109 °F (3/4t), and these increased ART values are used with procedures developed in the ASME Boiler and Pressure Vessel Code, Section ~~III~~XI, Appendix G and Code Case N-641 to calculate heatup and cooldown limits in accordance with the requirements of 10 CFR Part 50, Appendix G.

To develop composite pressure/temperature limits for the heatup transient, the isothermal, 1/4t heatup, and 3/4t heatup pressure/temperature limits are compared for a given thermal rate. Then the most restrictive pressure/temperature limits are combined over the complete temperature interval resulting in a composite limit curve for the reactor vessel beltline for the heatup event.

To develop composite pressure/temperature limit for the cooldown event, the isothermal pressure/temperature limits must be calculated. The isothermal pressure/ temperature limit is then compared to the pressure/temperature limit associated with both the constant cooldown rate and the corresponding step change rate. A step change is a change in temperature that occurs instantaneously. Examples of step changes are when the last RCP is secured during a cooldown or if a SDBCS valve fails open. A ramp change is a continuous change in temperature over time. A rapid cool down initiated after an event such as a SGTR or normal heating of the RCS with RCPs are examples of ramp changes. Any step change less than the specified TS limit will require a reduction in, or potentially cessation of the cooldown to ensure the applicable one hour maximum cooldown rate limit is not exceeded. The more restrictive allowable pressure/temperature limit is chosen resulting in a composite limit curve for the reactor vessel beltline.

Both 10 CFR Part 50, Appendix G and ASME Code Section ~~III~~XI, Appendix G, require the development of pressure/temperature limits which are applicable to inservice hydrostatic tests. The minimum temperature for the inservice hydrostatic test pressure can be determined by entering the curve at the test pressure and locating the corresponding temperature. This curve is shown for 54~~32~~ EFPY on Figure 3.4-2C.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (continued)

Similarly, 10 CFR Part 50 specifies that core critical limits be established based on material considerations. This limit is shown on the heatup curve, Figure 3.4-2A. Note that this limit does not consider the core reactivity safety analyses that actually control the temperature at which the core can be brought critical.

The Lowest Service Temperature is the minimum allowable temperature at pressures above 20% of the pre-operational system hydrostatic test pressure (~~624~~637 psia). This temperature is defined as equal to the most limiting RT_{NDT} for the balance of the Reactor Coolant System component (conservatively estimated as 50 °F) plus 100 °F, per Article NB 2332 of Section III of the ASME Boiler and Pressure Vessel Code.

The horizontal line between the minimum boltup temperature and the Lowest Service Temperature is defined by the ASME Boiler and Pressure Vessel Code as 20% of the pre-operational hydrostatic test pressure.

The minimum boltup temperature is the minimum allowable temperature at pressures below 20% of the pre-operational system hydrostatic test pressure. The minimum is defined as the initial RT_{NDT} for the material of the higher stressed region of the reactor vessel plus any effects for irradiation per Article G-2222 of Section III of the ASME Boiler and Pressure Vessel Code. The initial reference temperature of the reactor vessel and closure head flanges was determined using the certified material test reports and Branch Technical Position ~~MTEB 5-2~~5-3. The maximum initial RT_{NDT} associated with the stressed region of the vessel flange is 30 °F. The minimum boltup temperature is set to the minimum value of 60 °F per WCAP-14040-A, Revision 4~~of 30 °F plus a 20 °F conservatism is~~ 6~~50 °F~~.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided SAR Table 5.2-12 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

The reactor coolant vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one vent path from the reactor vessel head and the reactor coolant system high point ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are in addition to the valve testing required by the INSERVICE TESTING PROGRAM.

For inservice testing periods up to and including 2 years, Code Case OMN-20 provides an allowance to extend the inservice testing periods by up to 25%. For inservice testing periods of greater than 2 years, OMN-20 allows an extension of up to 6 months.