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1CAN081403

August 27, 2014

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

SUBJECT: License Amendment Request
Update the Reactor Coolant System Pressure and Temperature and the Low Temperature Overpressure Protection System Limits
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES

1 Entergy Letter to NRC, "Request for Exemption from Certain 10 CFR 50.61 and 10 CFR 50, Appendix G Requirements," dated March 20, 2014 (1CAN031403) (ML14083A640)

2. NRC Letter to Exelon Nuclear, "Three Mile Island Nuclear Station, Unit 1 – Exemption from Certain Requirements of 10 CFR Part 50, Appendix G and 10 CFR 50.61, For Initial $RT_{NDT}$ Values for Linde 80 Welds (TAC No. MF0425)," dated December 13, 2013 (ML13324A086)

3. NRC Letter to Exelon Nuclear, "Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment RE: Revision to the Pressure and Temperature Limit Curves and the Low Temperature Overpressure Protection Limits (TAC No. MF0424)," dated December 13, 2013 (ML13325A023)

4. NRC Letter to Duke Energy Carolinas, LLC, "Oconee Nuclear Station, Units 1, 2, and 3; Issuance of Amendments Regarding Pressure – Temperature Limits (TAC NOS. MF0763, MF0764, and MF0765)," dated February 27, 2014 (ML14041A093)

Dear Sir or Madam:

In accordance with the provisions of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.90, Entergy Operations, Inc. (Entergy) is submitting a request for an amendment to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TS) to revise the Reactor Coolant System Pressure and Temperature (P/T) Limits (TS 3.4.3); Pressurizer (TS 3.4.9); Pressurizer Safety Valves (TS 3.4.10); and Low Temperature Overpressure Protection (LTOP) System (TS 3.4.11). The current limits are applicable to 31 Effective Full Power Years (EFPYs). The proposed limits are applicable to the end of the current period of extended operation (54 EFPY).

A001
NRR

The P/T limits for the ANO-1 reactor pressure vessel were developed in accordance with 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, and NRC approved AREVA Topical Report BAW-10046A, Revision 2.

The projected fluence values at 54 EFPY are based on the NRC approved methodology presented in BAW-2241 P-A, Revision 2.

The initial reference temperature for nil-ductility transition ($RT_{NDT}$) values of the reactor vessel beltline welds (Linde 80 welds) were determined using methods provided in Topical Report BAW-2308, Revisions 1-A and Revision 2-A, rather than the methodology described within Topical Report BAW-10046A, Revision 2. The methodology in BAW-10046A, Revision 2, was used to evaluate the other beltline components (non-Linde 80 materials). Entergy requested an exemption from the requirements of 10 CFR 50.61 to allow use of the alternate initial $RT_{NDT}$ values provided in BAW-2308, Revisions 1-A and 2-A (Reference 1). The subsequent analyses assumed this exemption request was approved. The exemption request is currently being reviewed by the NRC.

Attachment 1 provides a description and assessment of the proposed TS changes. Attachment 2 provides markup pages of existing TS and TS Bases to show the proposed changes. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides a copy of AREVA Topical Report ANP-3300, "Arkansas Nuclear One (ANO) Unit 1 Pressure-Temperature Limits at 54 EFPY," June 2014. This report provides the technical basis for the proposed changes. The values associated with the Pressurized Thermal Shock assessment are provided in Attachment 5.

The current 31 EFPY P/T limits are estimated to be reached in April 2015. Entergy requests approval of the proposed license amendment by March 1, 2015, effective immediately with the amendment being implemented within 30 days of approval.

In accordance with 10 CFR 50.91(a)(1), "Notice for public comment," the analysis regarding the issue of no significant hazards consideration (NSHC) using the standards in 10 CFR 50.92 is being provided to the Commission in accordance with the distribution requirements in 10 CFR 50.4.

This amendment and the separate exemption request are similar to those approved for Three Mile Island Nuclear Station, Unit 1 (References 2 and 3) and Oconee Nuclear Station, Units 1, 2, and 3 (Reference 4). Three Mile Island and the three units at Oconee all have Babcock & Wilcox reactor vessels with Linde 80 welds similar to the ANO-1 reactor vessel.

In accordance with 10 CFR 50.91(b)(1), a copy of this application and the reasoned analysis about NSHC is being provided to the designated Arkansas state official.

If you have any questions or require additional information, please contact Stephenie Pyle at 479-858-4704.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on August 27, 2014.

Sincerely,

JGB/rwc

Attachments:

1. Description and Assessment of the Proposed Changes
2. Proposed Technical Specification and Bases Changes (mark-up)
3. Revised (clean) Technical Specification Pages
4. ANP-3300, "Arkansas Nuclear One (ANO) Unit 1 Pressure-Temperature Limits at 54 EFPY" June 2014
5. Pressurized Thermal Shock Assessment

cc: Mr. Marc L. Dapas
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**Attachment 1 to**

**1CAN081403**

**Description and Assessment of the Proposed Changes**

# DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGES

## 1.0 DESCRIPTION

In accordance with 10 CFR 50.90, Entergy Operations, Inc., (Entergy) requests an amendment to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1). The purpose of this License Amendment Request is to revise the pressure / temperature (P/T) limits in ANO-1 Technical Specification (TSs). The proposed amendment will revise the reactor coolant system 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, "Fracture Toughness Requirements."

The 54 EFPY time period will bound the operation of ANO-1 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.

Further, the proposed amendment also revises other ANO-1 TSs requirements to reflect the revised P/T limits of the reactor vessel. These changes rely on NRC approved methodologies for determining allowable P/T limits.

The proposed change includes the following TS revisions:

- TS Section 3.4.3 ("RCS Pressure and Temperature (P/T) Limits") is being revised to incorporate updated figures for P/T curves and reactor coolant pump restrictions. These figures have been recalculated to account for 54 EFPYs of plant operation.

- TS 3.4.9,"Pressurizer"; TS 3.4.10, "Pressurizer Safety Valves"; and TS 3.4.11, "Low Temperature Overpressure Protection (LTOP) System" are being revised to account for 54 EFPYs of operation. The changes include the electromatic relief valve lift setpoint being changed from 460 pounds per square inch – gauge (psig) to 563.8 psig. The enable temperature for this valve is being revised from 262 °F to 248 °F. These changes are as a result of the revised LTOP analyses and are consistent with the new P/T limits. Instrument uncertainty has not been included in these values.

TS Bases changes have been provided for information only.

## 2.0 TECHNICAL ANALYSIS

To address plant operation through the period of extended operation (54 EFPY), neutron fluence projections were updated; reactor vessel embrittlement analyses performed, and updated P/T and LTOP limits were developed. The P/T limits for the ANO-1 reactor vessel were developed in accordance with the requirements of 10 CFR 50, Appendix G, utilizing the analytical methods and flaw acceptance criteria of ASME Code Section XI, Appendix G and Topical Report BAW-10046A.

## Beltline Region Determination

Of particular interest in this analysis is the reactor vessel beltline, which is defined in 10 CFR 50, Appendix G, as the region of the reactor vessel 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." The beltline region experiences increased embrittlement over the operating period of the reactor vessel as a result of accumulated fast neutron radiation from the core.

10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," provides the requirements to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline resulting from the exposure to neutron irradiation and the thermal environment. Appendix H to 10 CFR 50 states that no material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods, that the peak neutron fluence at the end of operating period will not exceed 1E+17 neutrons/square centimeter (n/cm$^2$) with energy greater than one million electron volts (E > 1 MeV). Appendix G to 10 CFR states, "To demonstrate compliance with the fracture toughness requirements of Section IV of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part." Therefore, the fracture toughness requirements of 10 CFR 50, Appendix G, for the reactor vessel beltline are applicable to the reactor vessel materials with projected neutron fluence values greater than $1 \times 10^{17}$ n/cm$^2$ (E > 1 MeV) at the end of the operating period.

During operation, the physical region of the reactor vessel with fluence that exceeds this level can expand as a result of several factors, including power uprates, increased operating periods due to license renewal, and modified fuel design. The result is that changes in fracture toughness properties resulting from neutron embrittlement may occur in materials where the effects of radiation damage may not have been considered previously when developing the P/T limits for the vessel. In particular, this may be true for reactor vessel nozzle materials when the nozzles are positioned immediately above or below the active core height.

## Fluence Determination

Based on the considerations above, the fluence analysis performed for the latest cavity dosimetry exchange was expanded to determine if previously unevaluated regions of the reactor vessel had crossed the fluence threshold. The inside wetted surface neutron fluence values were determined following the method from BAW-2241 P-A, Revision 2. BAW-2241 P-A has been reviewed and accepted by the NRC, and is in compliance with NRC Regulatory Guide (RG) 1.190,"Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001. All the 1/4 vessel wall thickness (T) and 3/4T fluence values were generated using the inside wetted surface fluence values and the methodology from RG 1.99, Revision 2. The fluence values are provided in Tables 3-1 and 3-2 of the AREVA report in Attachment 4 of this submittal.

How ANO-1 meets the NRC requirements and conditions for the use of methodologies such as BAW-2241 is discussed in a later portion of this attachment.

P/T Limits Determination

The ability of the reactor pressure vessel 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 reactor pressure vessels is described in Appendix G to the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." This method utilizes fracture mechanics concepts and the reference temperature for nil-ductility transition, $RT_{NDT}$. The $RT_{NDT}$ is defined as the greater of the drop weight nil-ductility transition temperature (per American Society for Testing and Materials (ASTM) E208) or the temperature at which the material exhibits 50 foot-pounds 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 ($K_{1c}$). The $K_{1c}$ curve appears in Appendix G of ASME Code Section XI. When a given material is indexed to the $K_{1c}$ 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 histories.

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. 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 $K_{1c}$ 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 ($\Delta RT_{NDT}$), and a margin ($\sigma$) term. The $\Delta RT_{NDT}$ 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 RG 1.99, Revision 2, or from surveillance data. The FF is dependent upon the neutron fluence 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 RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial $RT_{NDT}$, the copper and nickel contents, the neutron fluence, and the calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

For the Linde 80 welds, alternate initial $RT_{NDT}$ values were used per the NRC-approved Topical Report BAW-2308, Revision 1-A and Revision 2-A. In order to utilize these alternate initial $RT_{NDT}$ values, an exemption request in accordance with 10 CFR 50.12 has been previously submitted (Reference 1). Using Linde 80 weld metal initial $RT_{NDT}$s from BAW-2308 requires a minimum CF of 167.0 and a margin term of 28.0 °F.

During the development of the new limits, AREVA informed Entergy that the generic $RT_{NDT}$ used in reactor vessel integrity calculations is non-conservative.

The generic initial $RT_{NDT}$ and its standard deviation are determined in BAW-10046, Revision 2A, for a population of SA-508 Class 2 forgings ordered subsequent to 1971. These values are used to determine the ART for other 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. The forgings in Babcock & Wilcox (B&W) designed reactor pressure vessels are generally ordered prior to 1971. When $RT_{NDT}$s from forgings manufactured prior to 1971 are included in the SA-508 Class 2 population, the newly calculated generic mean and standard deviation increase calculated ARTs.

AREVA now believes that the currently used dataset is not the most representative of the vessel forgings at the operating plants. AREVA further finds that the most representative datasets are those grouped by the manufacturer that performed the forging process. Using the more applicable dataset, the resultant ART can be higher, thus indicating that the current generic value may be less conservative (and potentially non-conservative).

AREVA has determined the appropriate new $RT_{NDT}$ and its uncertainty for ASTM SA-508 Class 2 forgings based upon these findings, and has confirmed that the ART values will not increase by greater than 4.5 °F.

A review of the ART calculations that support the development of the current P/T curves indicates that the limiting material is not affected at ANO-1. Only the non-limiting materials are affected. Therefore, the resultant P/T curves are unaffected by this finding.

The ANO-1 RV contains both axially and circumferentially oriented welds. Therefore, the P/T limits are based on the postulation of both axial and circumferential flaws in the most limiting axial and circumferential welds and the postulation of an axial flaw in the most limiting forging material of the reactor vessel.

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 uncorrected P/T limits were determined for 54 EFPY. Pressure correction factors were determined between the pressure sensor locations and the various regions of the reactor vessel.

Attachment 4 provides a summary of the technical basis leading to the development of the new P/T limits. Instrument uncertainty was not included in the limits listed in this attachment. These will be applied in the appropriate operating procedures.

Low Temperature Overpressurization Protection Limits

Low Temperature Overpressurization Protection (LTOP) limits were based on the ASME Code, Section XI, Article G-2215. This article requires that the 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. During a cooldown, the coolant temperature is always less than (or equal to) the 1/4T temperature; therefore, it is conservative to use the coolant temperature as the LTOP enable set-point. However, during a heatup, the 1/4T temperature is always less than the corresponding coolant temperature. To support the development of the LTOP system limits, the temperature differences between the reactor coolant in the downcomer region and the 1/4T wall locations are determined for the maximum heatup rate transient.

The current LTOP enabling temperature and electromatic relief valve (ERV) maximum lift setpoint are 262 °F and 460 psig, respectively. The proposed values are 248 °F and 563.8 psig, based on the criteria specified in Appendix G of the ASME Code, Section XI. These limits do not include instrument uncertainties. These will be applied in the operating procedures.

Pressurized Thermal Shock

A Pressurized Thermal Shock (PTS) assessment for the ANO-1 reactor vessel beltline materials with fluences greater than 1E+17 n/cm$^2$ was performed in accordance with 10 CFR 50.61. The PTS screening criterion is 270 °F for plates, forgings, and axial weld materials and 300 °F for circumferential weld materials.

The controlling material are the Lower Shell axial welds, WF-18, with a predicted $RT_{PTS}$ value of 191.1 °F. The remaining results are provided in Attachment 5 of this submittal.

Upper Shelf Energy and Equivalent Margins Analysis

Neutron fluence is part of the basis for Upper Shelf Energy (USE or $C_v$USE) and Equivalent Margins Analysis (EMA). The current analysis supported the license renewal update which demonstrated compliance with 10 CFR 50, Appendix G, IV.A.1. Reference 8 is the SE for the ANO-1 renewed license.

The current analysis remains bounding for the projected end of life fluence, except for the lower shell plate and upper shell plate axial (longitudinal) welds, WF-18. 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.

Comparing the quarter thickness fluence with BAW-2251A and the most recent ART yields

| Location | Material ID / Heat Number | 48 EFPY Fluence (n/cm$^2$) | 54 EFPY Fluence (n/cm$^2$) | Estimated 48 EFPY USE[1] |
|---|---|---|---|---|
| Nozzle belt forging (NBF) | AYN-131 / 528360 | 7.11E+18 | 8.48E+17[2] | 98 |
| Upper Shell (US) Plate | C5120-2 / C5120-2 | 8.06E+18 | 7.90E+18 | 65 |
| US Plate | C5114-2 / C5114-2 | 8.06E+18 | 7.90E+18 | 82 |
| Lower Shell (LS) Plate | C5120-1 / C5120-1 | 7.73E+18 | 7.78E+18 | 61 |
| LS Plate | C5114-1 / C5114-1 | 7.73E+18 | 7.78E+18 | 74 (71) |
| NB to US Circumferential (Circ.) Weld (100%) | WF-182-1 / 821T44 | 7.11E+18 | 7.14E+18 | 46 (55) |
| US Longitudinal Weld (Both 100%) | WF-18 / 8T1762 | 5.82E+18 | 6.32E+18 | 49 |
| US to LS Circ. Weld (100%) | WF-112 / 406L44 | 7.73E+18 | 7.60E+18 | 41 (44) |
| LS Longitudinal Weld (Both 100%) | WF-18 / 8T1762 | 5.71E+18 | 6.79E+18 | 49 |

1 RG 1.99 Revision 2 Position 1 (RG 1.99, Revision 2 Position 2)
2 Start at 8.4 inches portion of (lower) NBF

All reactor vessel locations not listed above have inside surface fluences below 1E+17 n/cm$^2$.

In Reference 8, the NRC made the following determination:

> Although not discussed by the applicant, Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have Charpy USE levels in the transverse direction for the base metal and along the weld for the weld material according to the ASME Code, of no less than 75 ft. lbs. (102 J) initially, and must maintain Charpy USE levels throughout the life of the vessel of no less than 50 ft. lbs. (68 J). However, Charpy USE levels below these criteria may be acceptable if it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that the 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.
>
> The 48 EFPY $C_v$ USE values determined for the ANO-1 reactor beltline materials are given in BAW-2251A, Table 4-4. The T/4 fluence values in this table were calculated in accordance with the ratio of the clad-to base metal interface fluence to T/4 fluence values (i.e., neutron fluence lead factors at T/4) determined in the last reactor vessel surveillance program report. Table 4-4 shows that the $C_v$USE is maintained above 50 ft-lbs for all base materials (plates and forgings), but weld materials nearly always fall below the 50 ft-lb limit at 48 EFPY. Appendix G of 10 CFR Part 50 provides for this situation by allowing lower values of $C_v$USE if it is demonstrated that the lower $C_v$USE will provide margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. An equivalent margins analysis was performed for 48 EFPY, and the results reported in Appendix A to BAW-2251A for service levels A, B, C, and D. For service levels A and B, the results demonstrate that there is sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated. Again, the results showed that there is a sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code.
>
> As mentioned earlier in this evaluation, the applicant submitted a response to an RAI for ANO-1 regarding Supplement 1 to GL-92-01, Revision 1. This response was BAW-2325, Revision 1. The "best estimate" chemistry composition (copper and nickel) was reported in BAW-2325, Revision 1. Best estimate chemistry compositions were also reported in BAW-2251A, and were summarized in Table A-1 of Appendix A to BAW-2251A for the various reactor vessel materials. The copper composition reported in BAW-2251A is equivalent to, or exceeds, the copper content reported in BAW-2325, Revision 1. In addition, the 48 EFPY fluence estimates were recalculated using the methodology described in Appendix B of BAW-2251A. It was shown that the fluence estimates listed in BAW-2251A remain conservative. Therefore the $C_v$USE values, given in Table 4-4 of BAW-2251A, remain conservative.
>
> The Appendix K analysis, from Section XI of the ASME Boiler and Pressure Vessel Code involves a quantitative assessment of the impact of low $C_v$USE on reactor vessel integrity. In Appendix K analysis, cracks are postulated at the inner reactor vessel wall. Since the neutron fluence decreases with depth into the vessel, the Appendix K analysis method assumes the fracture toughness at the crack tip will be greater than that at the inner wall of the vessel. The applicant's analysis was carried out using conservative stress assumptions for service levels A, B, C, and D for 48 EFPY. The analysis, given in Appendix B of BAW-2251A, shows that for service levels A and B, there is sufficient margin beyond that required by the acceptance criteria of Appendix K to Section XI of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated, and again the analytical results demonstrated that there is a sufficient margin beyond that required by

Appendix K to Section XI of the ASME Code. The applicant concludes that evaluations for all four service levels show the adequacy of safety against fracture for the ANO-1 vessel for 48 EFPY.

The staff found the B&WOG evaluation of the Charpy USE acceptable for all ANO-1 materials for the period of extended operation because the 48 EFPY analysis reported in Appendix B of BAW-2251A, and referenced in this application, meets the provisions of 10 CFR 54.21(c)(1)(ii) and applies to ANO-1.

## 3.0 REGULATORY ANALYSIS

### 3.1 No Significant Hazards Consideration Determination

Entergy Operations, Inc. (Entergy) proposes a change to the Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TSs) to revise the pressure / temperature (P/T) limits for the reactor coolant system.

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 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. Further, the proposed amendment revises the enable temperature and the lift setpoint for Low Temperature Overpressurization Protection (LTOP) requirements to reflect the revised P/T limits of the reactor vessel. 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 BAW-10046A, Revision 2, 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 incorporate methodologies that either have been approved or accepted for use by the NRC (provided that any conditions / limitations are satisfied). The P/T limits and LTOP limits will provide the same level of protection to 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 and associated LTOP 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."

### 3.2 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:

Appendix G to 10 CFR 50 requires that P/T limits be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society for Mechanical Engineering (ASME), Boiler and Pressure Vessel Code. Appendix G to 10 CFR 50 also provides minimum temperature requirements that must be considered in the development of the P/T limit curves. Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," advised licensees that the NRC

staff would use Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Material," Revision 2, to review P/T limits. RG 1.99, Revision 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation.

The GL 92-01, "Reactor Vessel Structural Integrity, "Revision 1, requested that licensees submit their reactor pressure vessel (RPV) materials property data for their plants to the NRC staff for review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RV integrity evaluations.

Standard Review Plan (STP), Branch Technical Position (BTP) 5-3, Revision 3, 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 Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor, $K_{1C}$, which is a function of the stress state and flaw configuration. ASME Code, Section XI, Appendix G, 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 the ASME Code, Section XI, Appendix G, has a depth that is equal to 1/4 of the RPV beltline thickness 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 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Appendix G to Section XI of the ASME Code requires that the adjusted reference temperature (ART or adjusted $RT_{NDT}$) be determined by evaluating material property changes due to neutron irradiation. The ART is defined as the sum of the initial $RT_{NDT}$, the mean value of the adjustment in reference temperature caused by irradiation ($\Delta RT_{NDT}$), and a margin ($\sigma$) term. The $\Delta RT_{NDT}$ 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 RG 1.99, Revision 2, or from surveillance data. The FF is dependent upon the neutron fluence 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 RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial $RT_{NDT}$, the copper and nickel contents, the neutron fluence and the calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

AREVA Topical Report BAW-2308, Revision 1-A and Revision 2-A provide NRC-approved alternate initial $RT_{NDT}$ and associated $\sigma_i$ values for various heats of Linde 80 beltline weld materials for RPV integrity evaluation applications.

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, $\Delta RT_{NDT}$ 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.

RG 1.190 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 RG 1.190, 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,

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 Reference 2 (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 Reference 3 (hereinafter referred to as "final GDC" or "GDC"). In accordance with Reference 4, 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.

### 3.4 Precedence

This amendment and the separate exemption request are similar to the ones approved for Three Mile Island Nuclear Station, Unit 1 (References 5 and 6) and Oconee Nuclear Station, Units 1, 2, and 3 (Reference 7). Three Mile Island and the three units at Oconee all have Babcock & Wilcox reactor vessels with Linde 80 welds similar to the ANO-1 reactor vessel.

### 3.5 Topical Report Conditions

The methodologies described in three separate topical reports were used in the development of this submittal. These topical reports are:

> BAW-10046A, Revision 6, “Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G”
>
> BAW-2241PA, Revision 2, “Fluence and Uncertainty Methodologies”
>
> BAW-2308, Revisions 1-A and 2-A, “Initial RTNDT of Linde 80 Weld Materials”

Each of these topical reports have been reviewed and approved by the NRC.

BAW-10046

The Safety Evaluation (SE) associated with BAW-10046 states that the NRC staff has determined that the methods are acceptable for application to the generation of P/T limit curves for pressurized water reactor (PWR) applications. The Staff found the report to be acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated SE.

A review of the SEs for all the revisions of this topical report did not identify any additional limitations in the use of this topical. It should be noted that the current 31 EFPY P/T limits were developed using this methodology.

BAW-2241

A review of the SEs associated with BAW-2241P-A, Revision 2 demonstrates that this revision is an extension of the PWR calculational methodology for application to boiling water reactors. This revision is not applicable to ANO-1.

The SE for Revision 1 of the topical report concluded that the proposed methodology is acceptable for referencing in licensing applications for determining the pressure vessel fluence of Westinghouse, Combustion Engineering and B&W designed reactors. In addition, there are three limitations imposed in the SE for Revision 1 of the topical. These limitations involved analysis of reactor designs not included in BAW-2241P-A database (e.g., partial length fluence assembly designs), changes in cross sections from those reviewed by the Staff, and any other changes in methodology. The design of the ANO-1 has been included in the BAW-2241P-A database. There are no changes in the cross sections or other changes to the methodology in the current application.

The NRC found that methodology presented in Revision 0 of BAW-2241 was acceptable for determining the pressure vessel fluence of B&W designed reactors and to be referenced in B&W designed reactor licensing actions. Three limitations were listed in the SE for this revision. These include that the methodology is applicable only to B&W designed reactors; changes in cross sections from those reviewed by the Staff, and provide the staff with a record of future modifications of the methodology. ANO-1 is a B&W designed reactor. As noted above, there are no changes in the cross sections from that previously reviewed and subsequent changes, if any, have been presented to the NRC and the Staff has reviewed those changes. See the discussions above related to Revisions 1 and 2 of the topical.

BAW-2308

The SE for BAW-2308, Revision 2-A provides an NRC-approved alternate initial $RT_{NDT}$ and associated $\sigma_i$ values for the Linde 80 weld material present in the beltline region of the reactor pressure vessels at Oconee Units 1, 2 and 3.

The following Conditions and Limitations are stated in the SE for BAW-2308, Revision 1-A

> Any license who wants to utilize the methodology of TR BAW-2308, Revision 1 as outlined in items (1) through (3) above, must request an exemption, per 10 CFR 50.12, from the requirements of Appendix G to 10 CFR Part 50 and 10 CFR 50.61 to do so.

Condition and Limitation (2) requires that a minimum chemistry factor of 167.0 °F be applied when the methodology of Regulatory Guide 1.99, Revision 2, and 10 CFR 50.61 is used to assess the shift in nil-ductility transition temperature due to irradiation.

Condition and Limitation (3) requires that a value of $\sigma_a$ = 28.0 °F be used to determine the margin term, as defined in Topical Report BAW-2308, Revision 1 and Regulatory Guide 1.99, Revision 2.

This exemption request was submitted to the Staff via Reference 1. The analyses performed to support the exemption request included the values listed in Condition and Limitations 2 and 3. As of the date of this submittal, the exemption request is being reviewed by the NRC Staff.

As demonstrated above, the limitations and conditions imposed on the three topical reports that were utilized in the development of the ANO-1 P/T limits have been satisfied and the reports are applicable to ANO-1.

## 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 Entergy Letter to NRC, "Request for Exemption from Certain 10 CFR 50.61 and 10 CFR 50, Appendix G Requirements," dated March 20, 2014 (1CAN031403) (ML14083A640)

2. Federal Register (32 FR 10213) on July 11, 1967

3. Federal Register (36 FR 3255) on February 20, 1971

4. NRC Staff Requirements Memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ML003763736)

5. NRC Letter to Exelon Nuclear, "Three Mile Island Nuclear Station, Unit 1 – Exemption from Certain Requirements of 10 CFR Part 50, Appendix G and 10 CFR 50.61, For Initial $RT_{NDT}$ Values for Linde 80 Welds (TAC No. MF0425)," dated December 13, 2013 (ML13324A086)

6. NRC Letter to Exelon Nuclear, "Three Mile Island Nuclear Station, Unit 1 – Issuance of Amendment RE: Revision to the Pressure and Temperature Limit Curves and the Low Temperature Overpressure Protection Limits (TAC No. MF0424)," dated December 13, 2013 (ML13325A023)

7. NRC Letter to Duke Energy Carolinas, LLC, "Oconee Nuclear Station, Units 1, 2, and 3; Issuance of Amendments Regarding Pressure – Temperature Limits (TAC NOS. MF0763, MF0764, and MF0765)," dated February 27, 2014 (ML14041A093)

8. NRC Cover Letter to Entergy, "Arkansas Nuclear One, Units 1, License Renewal Safety Evaluation Report," dated April 12, 2001 (ML011030091) (SER ML011020554)

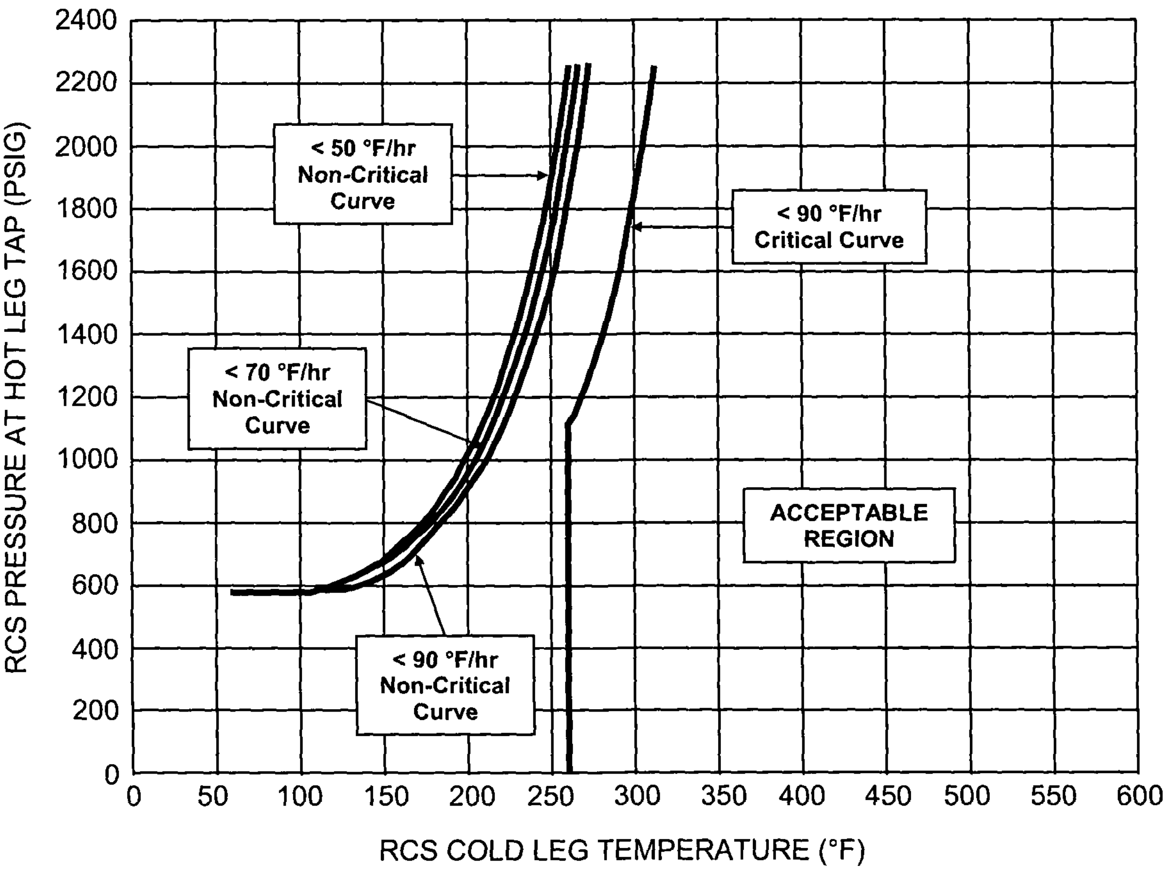
**Attachment 2 to**

**1CAN081403**

**Proposed Technical Specification and Bases Changes (mark-up)**

FIGURE 3.4.3-1

RCS Heatup Limitations to 54~~31~~ EFPY




Notes:

1. ~~These~~ Ccurves 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:

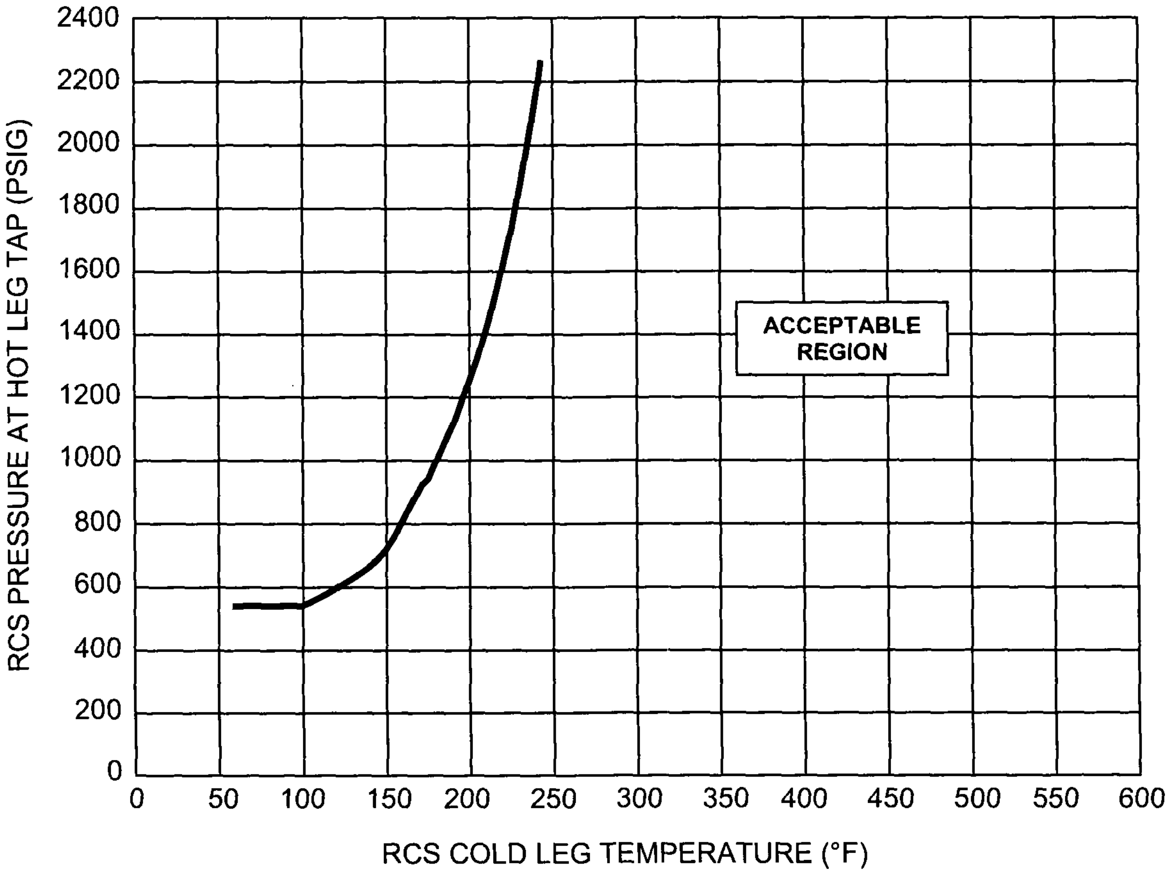
| RCS TEMP | RCP RESTRICTIONS |
|---|---|
| T ≥~~>~~ ~~30~~250 °F | None |
| ~~30~~250°F ≥~~>~~ T ≥ ~~225~~100 °F | ≤ 3 |
| ~~225 °F > T ≥ 84 °F~~ | ~~≤ 2~~ |
| T < 100~~84~~ °F | No RCPs operating |

4. Allowable Heatup Rates:

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

FIGURE 3.4.3-2

RCS Cooldown Limits to 54~~31~~ 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 ≥~~>~~ ~~30~~250 °F | None |
| ~~30~~250°F ~~≥~~> T ≥ ~~225~~100 °F | ≤ 3 |
| ~~225°F > T ≥ 84 °F~~ | ~~≤ 2~~ |
| T < 100~~84~~ °F | No RCPs operating |

4. Allowable Cooldown Rates:

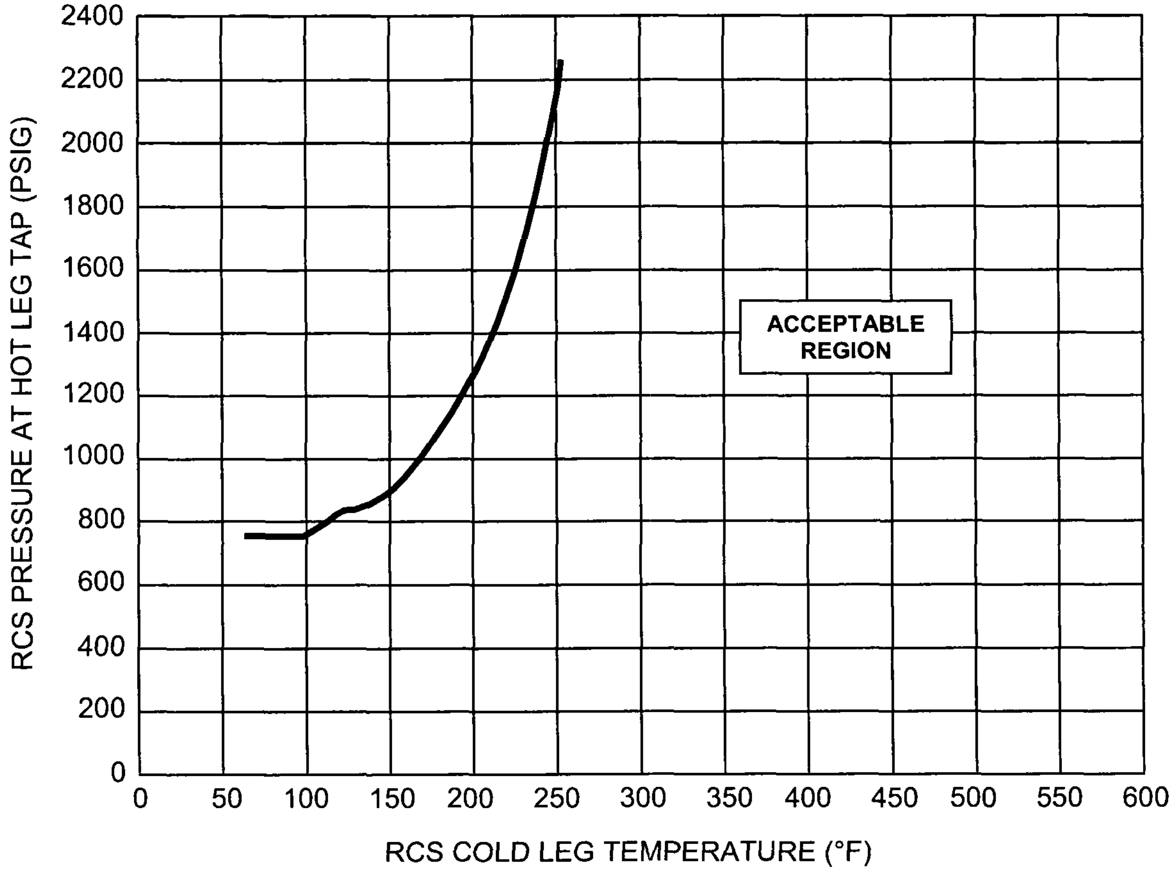
| RCS TEMP | C/D RATE | STEP CHANGE |
|---|---|---|
| T ≥ 280 °F | 100 °F/hr~~HR~~ | ≤ 50 °F in any 1/2 hr~~HR~~ |
| 280°F > T ≥ 150 °F | 50 °F/hr~~HR (Note 5)~~ | ≤ 25 °F in any 1/2 hr~~HR~~ |

| | | |
|---|---|---|
| T < 150 °F | 25 °F/hr~~HR~~ | ≤ 25 °F in any 1 hr~~HR~~ |

~~5. 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.~~

FIGURE 3.4.3-3

RCS Inservice Hydrostatic Test H/U & C/D Limits to 54~~31~~ 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 > ~~262~~248°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.<br>AND<br>B.2 Be in MODE 4 with RCS temperature ≤ ~~262~~248°F. | 6 hours<br><br>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.<br>AND<br>D.2 Be in MODE 4. | 6 hours<br><br>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 > ~~262~~248 °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 > ~~262~~248 °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 > ~~262~~248 °F.

----------

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > ~~262~~248 °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.<br><br>OR<br><br>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 > ~~262~~248 °F. | C.1 Be in MODE 4 with RCS temperature ≤ ~~262~~248 °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."

-------------------------------------------------------------------

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 ≤ ~~460~~563.8 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 ≤ ~~262~~248 °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 |

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, and unit transients. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 contain P/T limit curves for heatup, cooldown, inservice hydrostatic testing, and physics testing at RCS temperatures ≤ 525 °F, and the maximum rate of change of reactor coolant temperature. The methods and criteria employed to establish operating pressure and temperature limits are described in BAW-10046A (Ref. 1). These limit curves are applicable through ~~thirty-one~~fifty-four effective full power years (EFPY) of operation. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for the various operating reactor coolant pump combinations.

Each P/T curve defines an acceptable region for normal operation below and to the right of the limit curve. The curves are used to develop operational guidance for use during heatup or cooldown maneuvering.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel. The vessel is the component most subject to brittle failure due to the fast neutron embrittlement it experiences during power operation, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the reactor coolant pressure boundary (RCPB) materials. Reference 2 requires an adequate margin to brittle failure during normal operation, abnormalities, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 4). For the LInde 80 weld materials present in the ANO-1 reactor vessel beltline an alternative approach for determining the adjusted reference nil-ductility temperature as described in Topical Report BAW-2308, Revisions 1-A and 2-A (Ref. 12). The Master Curve methodology is accepted with exemption from the requirements of 10 CFR 50.61 (ref. 13) and 10 CFR 50, Appendix G (Ref.2)

BACKGROUND (continued)

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in ~~FTI Document 77-1258569-01~~CALC-14-E-0100-08 (Ref. 5). ~~The service period was reduced by one effective full power year from that assumed in Reference 5 to be conservative with respect to independent calculations performed by the NRC staff. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543 (Rev. 6).~~ The chemical composition of the limiting weld material is reported in the B&W report, BAW-~~2121P~~2317 (Rev. 7). The effect of neutron irradiation on the nil ductility reference temperature ($RT_{NDT}$) of the limiting weld material is reported in ~~FTI~~ Calculations ~~32-1245917-00 and 32-1257716-00~~CALC-14-E-0100-02 (Rev. 8) and CALC-14-E-0100-09 (Ref.14).

The actual shift in the $RT_{NDT}$ of the vessel beltline region material will be established periodically by removing and evaluating the irradiated reactor vessel material surveillance specimens, in accordance with Appendix H of 10 CFR 50 (Ref. 9). These specimens are installed near the inside wall ~~of this or a~~in other similar reactor vessels in the core region. The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

Prior to reaching ~~thirty-one~~fifty-foour effective full power years of operation, Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 must be updated for the next service period in accordance with 10 CFR 50, Appendix G. The service period must be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543 (Ref. 6) and Topical Report BAW-2308 (Ref. 12). The highest predicted adjusted reference temperature of all the beltline region materials is used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction is submitted for NRC staff review at least 90 days prior to the end of the service period.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The calculation to generate the inservice hydrostatic testing curve uses different safety factors (per Ref. 3) than the heatup and cooldown curves. The testing curve also extends to the RCS design pressure of 2500 psia.

LCO (continued)

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

The heatup and cooldown rates stated are intended as the maximum changes in temperature in one direction in the stated time periods. The actual temperature linear ramp rate may exceed the stated limits for a shorter time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the stated time period.

The acceptable P/T combinations are below and to the right of the limit curves which are applicable for the first ~~31~~fifty-four EFPY. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The magnitude of the departure from the allowable operating P/T regime or the magnitude of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

---

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or inservice hydrostatic testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

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ACTIONS (continued)

D.1 and D.2 (continued)

ASME Code, Section XI, Appendix E (Ref. 10), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition D is modified by a Note requiring Required Action D.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone, per Required Action D.1, is insufficient because higher than analyzed stresses may have occurred and may have affected RCPB integrity.

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SURVEILLANCE REQUIREMENTS

SR 3.4.3.1, SR 3.4.3.2, SR 3.4.3.3, and SR 3.4.3.4

Verification that operation is within the limits of the appropriate figure is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes.

This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or inservice hydrostatic testing may be discontinued when the definition given in the relevant unit procedure for ending the activity is satisfied.

The acceptable P/T combinations are below and to the right of the limit curves ~~which are applicable for the first 31 EFPYs~~. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation (as identified in Note 1 on each applicable Figure).

SR 3.4.3.1 is modified by a Note that requires this SR to be performed only during system heatup operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-1 which provides applicable heatup limitations, including reactor coolant pump (RCP) operating restrictions and allowable heatup rates. Figure 3.4.3-1 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated DHR system return temperature to the reactor vessel is the appropriate temperature indicator.

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REFERENCES

1. BAW-10046A, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G", Rev. 2, June 1986.

2. 10 CFR 50, Appendix G, Fracture Toughness Requirements.

3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

4. Regulatory Guide 1.99, Revision 2, May 1988.

5. ~~FTI Document 77-1258569-01~~CALC-14-E-0100-08, ANO-1 Corrected P-T Limits for 60 Years (54 EFPY).

6. BAW-1543, Master Integrated Reactor Vessel ~~Material~~ Surveillance Program (latest revision).

7. ~~BAW-2121P, Irradiation Induced Reduction in Charpy Upper Shelf Energy of Reactor Vessel Welds~~BAW-2313, Revision 6, B&W Fabricated Reactor Vessel Materials and Surveillance Data Information.

8. ~~FTI Calculations 32-1245917-00 and 32-1257716-00~~CALC-14-E-0100-02, ANO-1 ART (Adjusted Reference Temp) Values at 54 EFPY.

9. 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements.

10. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

11. 10 CFR 50.36, Technical specifications.

12. BAW-2308, Revisions 1-A and 2-A, Initial RTNDT of Linde 80 Weld Materials

13. 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

14. CALC-14-E-0100-09, ANO-1 Fluence Analysis Report, Cycles 21, 22, and 23 for RV Beltline.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level ≤ 320 inches ensures that a steam bubble exists prior to criticality. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires a minimum of 126 kW (nominal) of pressurizer heaters OPERABLE. To be considered OPERABLE, the required heaters must be powered from an ES bus. NUREG-0578 (Ref. 1) specifies that the minimum required pressurizer heaters are capable of being powered from redundant, emergency diesel generator backed sources. This provides assurance that sufficient heater capacity is available to provide RCS pressure control during a loss of off-site power. The amount needed to maintain pressure is dependent on the insulation losses, which can vary due to tightness of fit and condition.

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APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3 and, for pressurizer water level, for MODE 4 with RCS temperature > ~~262~~248 °F. The purpose is to prevent water solid RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations, such as reactor coolant pump startup. The temperature of ~~262~~248 °F has been designated as the cutoff for applicability because LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)," provides a requirement for pressurizer level at or below ~~262~~248 °F. The LCO does not apply to MODE 5 with loops filled because LCO 3.4.11 applies and provides adequate overpressure protection. This parameter value does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The LCO does not apply to MODES 5 and 6 with partial loop operation.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. The Applicability is modified by a Note stating that the OPERABILITY requirements on pressurizer heaters do not apply in MODE 4. For MODE 4, 5, or 6, the need to control pressure (by heaters) to ensure loop subcooling for heat transfer is significantly reduced when the Decay Heat Removal System is in service, and therefore the LCO is not applicable.

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ACTIONS

A.1

With pressurizer water level outside the limit, action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limit. The 1 hour Completion Time is considered to be a reasonable time for adjusting pressurizer level.

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. By shutting down the reactor and reducing reactor coolant temperature to at least MODE 3, the potential thermal energy of the reactor coolant mass for mass and energy releases is reduced.

Six hours is a reasonable time based upon operating experience to reach MODE 3 from full power in an orderly manner and without challenging unit systems. Further pressure and temperature reduction to MODE 4 with RCS temperature ≤ ~~262~~248 °F places the unit into a MODE where the LCO is not applicable. The 24 hour Completion Time to reach the non-applicable MODE is reasonable based upon operating experience.

C.1

If the required pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using non-ES bus powered heaters.

D.1 and D.2

If the Required Action and associated Completion Time are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3 within 6 hours and to MODE 4 within the following 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that pressurizer water level be maintained below the upper limit to provide a minimum space for a steam bubble. The value specified for pressurizer level does not contain an allowance for instrument error. Therefore, additional allowances for instrument uncertainties must be provided in the implementing procedures. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level.

SR 3.4.9.2

The SR requires sufficient pressurizer heaters which are connected to an ES bus verified to be capable of providing the required capacity. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

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REFERENCES

1. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.

2. 10 CFR 50.36 Technical specifications.

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

## B 3.4.10 Pressurizer Safety Valves

BASES

### BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection (Ref. 1). Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. One safety valve is required for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 2). The required lift pressure is 2500 psig + 1%, - 3%. The safety valves discharge steam from the pressurizer to a quench tank located in the reactor building. The discharge flow is indicated by acoustic flow monitoring devices, by an increase in temperature downstream of the safety valves, and by an increase in the quench tank temperature, pressure, and level.

The upper and lower as-left pressure limits are based on the ± 1% tolerance requirement for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

### APPLICABLE SAFETY ANALYSES

The overpressure protection analysis (Ref. 3) is based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). One pressurizer code safety valve is capable of preventing overpressurization in MODE 3 and in MODE 4 with RCS temperature > ~~262~~248 °F since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat (Ref. 1 and 4). These valves must accommodate pressurizer insurges that

APPLICABLE SAFETY ANALYSES (continued)

could occur during a startup, rod withdrawal, or ejected rod event. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at low power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

In MODES 1 and 2, pressurizer safety valves satisfy Criterion 3 of the 10 CFR 50.36 (Ref. 5). In MODE 3 and MODE 4 above the LTOP enable temperature, the pressurizer safety valves satisfy Criterion 4 of 10 CFR 50.36.

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LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The upper and lower as-left pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 2) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

The LCO is modified by four Notes. Note 1 states that in MODE 3 and MODE 4 with RCS temperature above ~~262~~248 °F, only one pressurizer safety valve is required to be OPERABLE. In this condition, one pressurizer safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than the sum of the available heat sources.

Note 2 allows entry into MODE 3, and into MODE 4 with RCS temperature > ~~262~~248°F, with the lift settings potentially outside the limits. This permits testing of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

Note 3 states that the LCO is not applicable in MODE 3, and in MODE 4 with RCS temperature > ~~262~~248°F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. During hydrostatic tests, the code safeties must be gagged to prevent them from relieving at the target test pressure. RCS pressure is carefully observed and compensatory measures are in place to provide assurance that the pressure is appropriately controlled during the performance of hydrostatic tests.

LCO (continued)

Note 4 states that the provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > ~~262~~248 °F. In the event no code safety valve is OPERABLE in this MODE, the Required Actions ensure that the RCS is placed in a condition in which the ERV is capable of relieving any potential LTOP pressure transient.

The parameter value (~~262~~248 °F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

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## APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP enable temperature, OPERABILITY of pressurizer safety valve(s) is required to ensure adequate relieving capacity is available to keep reactor coolant pressure below 110% of its design value during certain accidents.

The LCO is not applicable in MODE 4 with RCS temperature ≤ ~~262~~248 °F, in MODE 5, nor in MODE 6 when the reactor vessel head is on because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head removed.

The parameter value (~~262~~248 °F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

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## ACTIONS

A.1

With one pressurizer safety valve inoperable in MODES 1 and 2, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1

If the Required Action and associated Completion Time of Condition A are not met, or if both pressurizer safety valves are inoperable in MODES 1 and 2, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

ACTIONS (continued)

C.1

With the required pressurizer code safety valve inoperable, the RCS overpressure protection capability is significantly reduced and an overpressure event could challenge the integrity of the RCPB. Therefore, the unit must be placed in a condition in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 4 with RCS temperature at or below the LTOP enable temperature within 18 hours. The 18 hours allowed is reasonable, based on operating experience, to reach a low temperature within MODE 4 without challenging unit systems. With RCS temperature at or below ~~262~~248 °F, overpressure protection is provided by LTOP.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME OM Code (Ref. 6), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is + 1%, - 3% for OPERABILITY (Ref. 7); however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. SAR, Section 4.2.4.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1968.
3. SAR, Section 4.3.8.
4. SAR, Section 4.3.11.4.
5. 10 CFR 50.36.
6. ASME, Boiler and Pressure Vessel Code, Section XI.
7. ASME OM Code - 2001.

BACKGROUND (continued)

accommodate a coolant insurge and prevent a rapid pressure increase, allowing the operator time to stop the increase. The ERV, with reduced lift setting, or the RCS vent is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI deactivated, the ability to provide RCS coolant addition is restricted. To allow for coolant addition, the LCO does not require the makeup function to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the makeup function can provide flow through the makeup control valve.

ERV Requirements

As designed for the LTOP, the ERV is signaled to open if the RCS pressure reaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the ERV is signaled to open. Maintaining the lowered setpoint ensures the Reference 1 limits will be met in any event analyzed for LTOP.

RCS Vent Requirements

Once the RCS is depressurized, adequate pressure relief capability may be provided by a vent path to the reactor building atmosphere which is capable of relieving the flow of the limiting LTOP transient and maintaining pressure below P/T limits. The required vent capacity may be provided by one or more vent paths. Acceptable RCS vent paths include any of the following: removing a pressurizer safety valve, locking the ERV in the open position and disabling its block valve in the open position, or similarly establishing a vent by removing a steam generator (SG) primary manway, removing a SG primary hand hole cover, removing all control rod drive top closure assemblies (excluding reactor vessel level probe), or removing a pressurizer manway. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

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APPLICABLE SAFETY ANALYSES

Safety analyses (Refs. 4, 5, 6, and 7) demonstrate that the reactor vessel can be adequately protected against overpressurization transients during shutdown. The pressure and temperature limits are derived from fracture mechanics analyses. Transients are then evaluated to determine a required ERV setpoint and other unit conditions that will ensure that the P/T limits are not exceeded.

Fracture mechanics analyses ~~(using the safety margins of Reference 8)~~ established the temperature of LTOP Applicability at ~~262~~248 °F. Above this temperature, the pressurizer safety valves provide the reactor vessel overpressure protection. The actual temperature at which the allowable pressure falls below the pressurizer.

APPLICABLE SAFETY ANALYSES (continued)

safety valve setpoint increases as vessel material ductility decreases due to neutron embrittlement. P/T limits are periodically determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations. For the current limits, vessel materials are assumed to have a neutron irradiation accumulation equivalent to ~~31~~54 effective full power years (EFPYs) of operation. Each time the P/T limit curves are revised, the LTOP is re-evaluated to ensure that its functional requirements can still be met. The ERV setpoint is revised if necessary.

Transients that are capable of overpressurizing the RCS at low temperature result in either excessive mass input or excessive heat input. Such transients include: HPI actuation, CFT discharge, energization of the pressurizer heaters, failing the makeup control valve open, loss of decay heat removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and addition of nitrogen to the pressurizer. Without controls, HPI actuation and CFT discharge would be transients that result in exceeding P/T limits within the 10 minute period in which time no operator action can be assumed to take place. For the remaining events, operator action after that time precludes overpressurization.

This specification prevents exceeding the P/T limits by: 1) limiting the capability for rapid mass input to the RCS; and 2) ensuring that adequate vent capability exists to accommodate inadvertent mass or energy addition to the RCS. Pressurizer level is also limited to ensure that increasing pressure during a transient will be slow enough to preclude exceeding pressure limits within the 10 minutes assumed to be required for operator action to mitigate the transient. Mass input into the system is limited by disabling HPI (with specific exceptions) and by deactivating pressurized CFT discharge isolation valves in the closed position with their power breakers open (with specific exceptions). The analyses demonstrate that HPI transients involving one HPI pump can be accommodated by the ERV without exceeding the maximum allowable pressure.

The ERV setpoint is determined by modeling LTOP performance assuming the most limiting LTOP transient of a makeup control valve failing open. Pressure overshoot beyond the setpoint resulting from signal processing and valve stroke times is considered. The resulting ERV setpoint ensures the ~~r~~Reference 1 limits will not be exceeded.

Vent capability is required to ensure that the maximum allowable pressure is not exceeded in the event of full opening of the makeup control valve while one makeup pump is running. Acceptable vent paths have adequate capacity at a system pressure of 100 psig which is less than the maximum RCS pressure on the P/T limit curve in LCO 3.4.3.

APPLICABLE SAFETY ANALYSES (continued)

The ERV is an active component. Therefore, its failure represents the worst case single active failure of LTOP features. The other vent paths are passive and not subject to active failure.

The LTOP satisfies Criterion 2 of 10 CFR 50.36 (Ref. 9).

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LCO

The LCO requires an LTOP system OPERABLE with a limited coolant input capability and a pressure relief capability. To limit coolant input, the LCO requires the HPI deactivated, and the CFT discharge isolation valves closed and deactivated. For pressure relief, the LCO requires the pressurizer coolant level to be below a level which represents a water solid condition, and the ERV OPERABLE with a lowered lift setting or the RCS depressurized and a vent established.

HPI deactivation requires that the HPI system be incapable of causing a significant increase in RCS pressure (motor operated valves de-activated closed, HPI pump breakers racked down, or other configurations that prevent inadvertent HPI actuation). CFT isolation requires the CFT discharge valves to be closed and the circuit breakers for the motor operators to be opened.

The HPI deactivation and CFT isolation requirements are modified by five Notes. Note 1 indicates that the requirements are not applicable during ASME Section XI testing. This exception provides for required testing during these shutdown conditions rather than at power when the HPI and CFTs are required to be OPERABLE for the ECCS function. Note 2 indicates that the requirements are not applicable for the HPI deactivation during fill and vent of the RCS. The HPI pumps are used for this normal makeup function and must be available. Specific procedural controls are provided to prevent overpressurization during this activity. Note 3 indicates that the requirements are not applicable for the HPI deactivation during emergency RCS makeup. This exception is necessary to enhance the response capability to a loss of decay heat removal event without violating the TS (Ref. 10). Note 4 indicates that the requirements are not applicable for the HPI deactivation during valve maintenance. This exception allows maintenance to be performed during these shutdown conditions rather than at power when the HPI is required to be OPERABLE for the ECCS function. Note 5 states that CFT isolation is only required when CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This is acceptable since the CFT can not be the source of an overpressurization event when its pressure is less than the allowable RCS pressure.

The pressurizer is considered to represent a water solid condition when coolant level is > ~~105~~180 inches~~, when RCS pressure is > 100 psig, or > 150 inches, when RCS pressure is ≤ 100 psig~~. Although a vapor space still exists with pressurizer level above these values, from an analytical point of view, the unit is considered to be water solid. ~~These~~This parameter ~~values~~does not contain allowances for instrument error.

LCO (continued)

The pressurizer level requirements are modified by two Notes. Note 1 indicates that the requirements are not applicable during operation allowed by the Emergency Operating Procedures (EOPs). This exception provides for use of the "feed and bleed" process when necessary as determined by the EOPs. Note 2 indicates that the requirements are not applicable during RCS hydrotesting. Specific procedural controls are provided to prevent overpressurization during this activity.

OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path. For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be set at ≤ ~~460~~563.8 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the ERV and its control circuits. With the RCS depressurized, acceptable alternate vent paths include removing a pressurizer safety valve, locking the ERV in the open position and disabling its block valve in the open position, removing a SG primary manway, removing a SG primary hand hole cover, removing all control rod drive top closure assemblies (excluding reactor vessel level probe), or removing a pressurizer manway.

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APPLICABILITY

This LCO is applicable in MODE 4 with RCS temperature ≤ ~~262~~248 °F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The Applicability temperature of ~~262~~248 °F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above ~~262~~248 °F. With the vessel head off, overpressurization is not possible.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above ~~262~~248 °F.

The parameter value (~~262~~248 °F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

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ACTIONS

A.1, B.1, and B.2

With the pressurizer level not within its required limits, the time for operator action in a pressure increasing event is reduced. The postulated event most affected in the LTOP MODES is failure of the makeup control valve, which fills the pressurizer relatively rapidly. Restoration is required within 1 hour.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Verification of the pressurizer level at ≤ ~~105~~180 inches ~~when RCS pressure is > 100 psig or ≤ 150 inches when RCS pressure is ≤ 100 psig~~, by observing control room or other indications ensures that the unit is not in a water solid condition and that a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients (Ref. ~~3~~11). This parameter does not contain allowances for instrument error.

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when these evolutions are complete, as defined in unit procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.11.2 and SR 3.4.11.3

Verifications must be performed that the HPI is deactivated, and each pressurized CFT is isolated. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP. The Surveillances are required at 12 hour intervals.

The 12 hour intervals are shown by operating practice to be sufficient to assess coolant input capability and verify operation within the safety analysis.

SR 3.4.11.4

OPERABLE pressure relief capability must be provided to prevent overpressurization due to inadvertent full makeup system operation. Such a vent keeps the pressure from full makeup flow within the LCO limit. OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path.

For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be set at ≤ ~~460~~563.8 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the two valves and their control circuits. The parameter value of ~~460~~563.8 psig does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.4 (continued)

With the RCS depressurized, acceptable alternate vent paths include: a) removing a pressurizer safety valve; b) locking the ERV in the open position and disabling its block valve in the open position; c) removing a SG primary manway; c) removing a SG primary hand hole cover; d) removing all control rod drive top closure assemblies (excluding reactor vessel level probe); and e) removing a pressurizer manway.

For a vent path not locked open, the Frequency is every 12 hours. For a locked open vent path, the required Frequency is every 31 days.

The Frequency intervals are considered adequate based on operating practice to determine adequacy of pressure relief capability and verify operation within the safety analysis.

SR 3.4.11.5

The performance of a CHANNEL CALIBRATION is required every 18 months. The CHANNEL CALIBRATION for the LTOP ERV opening logic, including the ERV setpoint, ensures that the ERV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

The 18 month Frequency considers a typical refueling cycle and industry accepted practice.

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REFERENCES

1. 10 CFR 50, Appendix G, Fracture Toughness Requirements.
2. Generic Letter 88-11, Pressurizer Surge Line Thermal Stratification.
3. ANO-1 LTOP Safety Evaluation Report (1CNA058302) dated May 5, 1983.
4. Response to NRC Request for Additional Information (1CAN117608) dated November 15, 1976.
5. Response to NRC Request for Additional Information (1CAN127602) dated December 3, 1976.
6. Response to NRC Request for Additional Information (1CAN037716) dated March 24, 1977.

REFERENCES (continued)

7. ~~ANO-1 License Amendment Request (1CAN119608), dated November 26, 1988, and Operating License Amendment 188, (1CNA039703) dated March 14, 1997~~Deleted.

8. ~~ANO-1 Request for Exemption (1CAN119608), dated November 26, 1996, and Exemption from Requirements of 10 CFR 50.60, (1CNA039702) dated March 12, 1997~~Deleted.

9. 10 CFR 50.36, Technical specifications.

10. ANO-1 License Amendment Request (1CAN059008), dated May 22, 1990, and Operating License Amendment 138, (1CNA119002) dated November 1, 1990.

11. CALC-14-E-0100-13, ANO-1 Pressurizer Model for LTOP Design Bases Transient (54 EFPY).

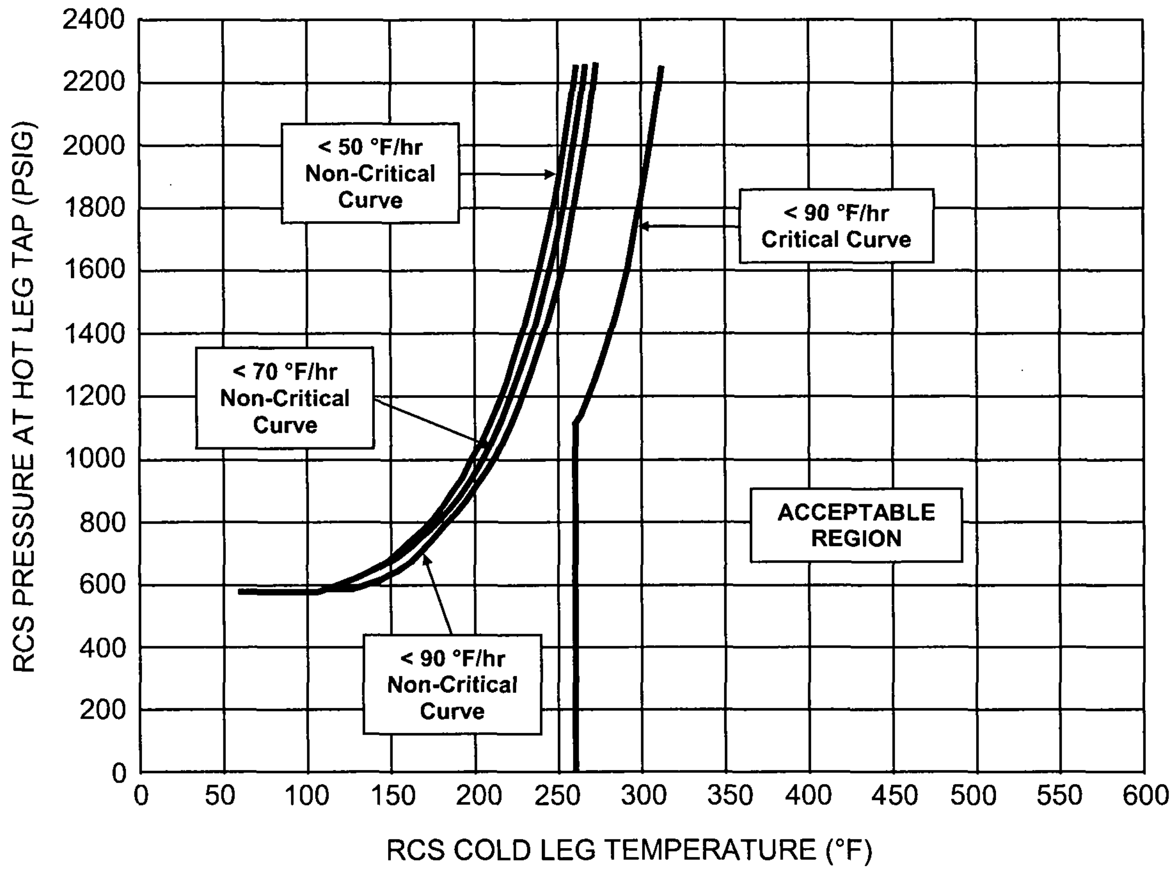
**Attachment 3 to**

**1CAN081403**

**Revised (clean) Technical Specification Pages**

FIGURE 3.4.3-1

RCS Heatup Limitations to 54 EFPY




Notes:

1. 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:

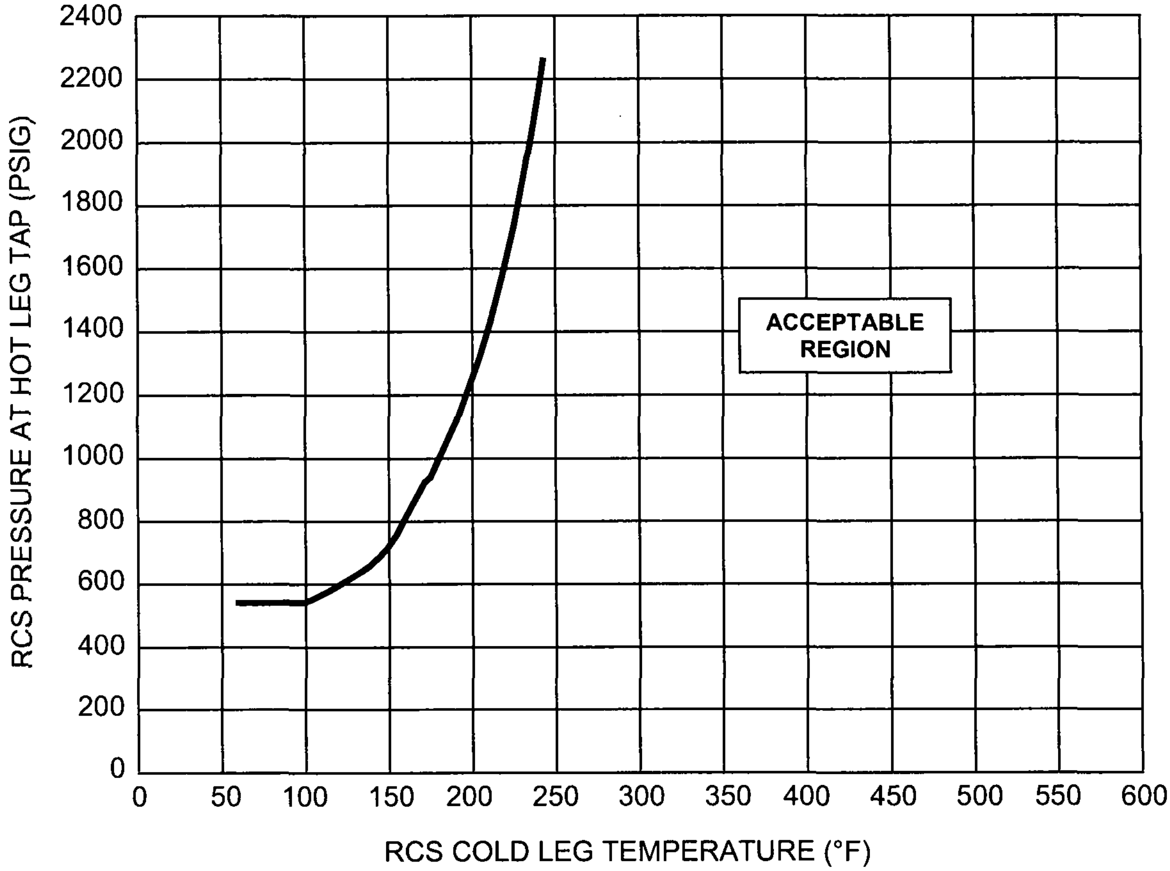
| RCS TEMP | RCP RESTRICTIONS |
|---|---|
| T ≥ 250 °F | None |
| 250°F > T ≥ 100 °F | ≤ 3 |
| T < 100 °F | No RCPs operating |

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:

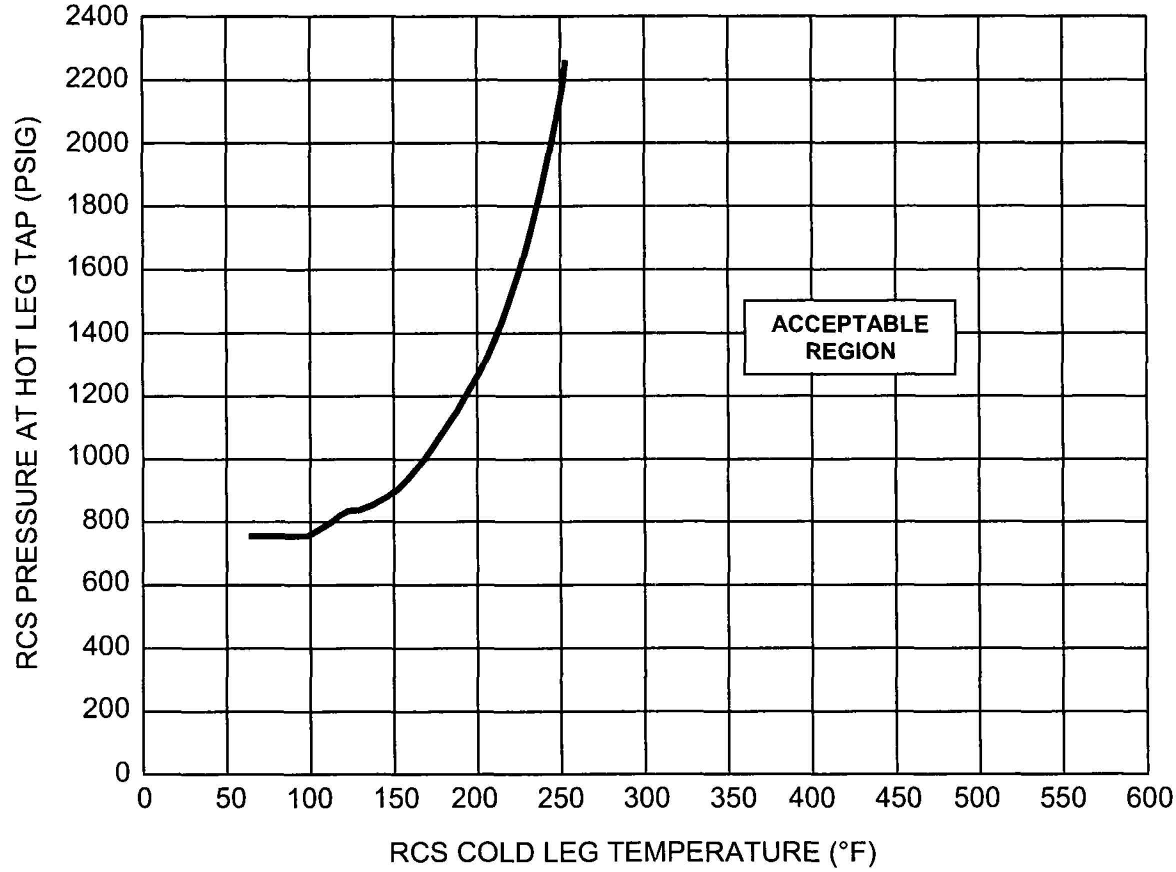
| RCS TEMP | RCP RESTRICTIONS |
|---|---|
| T ≥ 250 °F | None |
| 250 °F > T ≥ 100 °F | ≤ 3 |
| T < 100 °F | No RCPs operating |

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 > 248 °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.<br><u>AND</u><br>B.2 Be in MODE 4 with RCS temperature ≤ 248 °F. | 6 hours<br><br>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.<br><u>AND</u><br>D.2 Be in MODE 4. | 6 hours<br><br>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 > 248 °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 > 248 °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 > 248 °F.

----------

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > 248 °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.<br><br><u>OR</u><br><br>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 > 248 °F. | C.1 Be in MODE 4 with RCS temperature ≤ 248 °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."

-----

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 ≤ 563.8 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 ≤ 248 °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 |